CHAPTER 12

Radiation Protection and Environmental Safety

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***Summary**:*

Crucial to operation of a CANDU nuclear plant is protection of workers, the public, and the environment. This chapter discusses:

1. *Basic fundamentals of radiation physics as they pertain to radiation interactions that have the potential to cause biological harm in living systems.*
2. *Concepts of regulatory guidance which governs the "as low as reasonably achievable" radiation dose paradigm.*
3. *Radiation detection and monitoring techniques used in CANDU plant environs.*
4. *External and internal radiation hazards, including discussions on shielding and personal protective equipment.*
5. *Radiation management plans, worker dose monitoring, and control and waste management issues.*
6. *Radiation releases to the environment, derived release limits, and environmental protection.*

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1 Introduction

In Chapter 3, the fundamentals of nuclear physics, radioactive decay processes, and radiation interactions with matter were discussed. This chapter expands upon the concepts in Chapter 3 and explores the fundamentals of radiation protection and environmental safety, which are of vital importance to the safe operation of CANDU stations. In a holistic sense, the overarching field that deals with radiation protection and environmental safety is the area of health physics. Health physics is the branch of science that deals with protection of workers, the public, and the environment from potential detrimental effects from exposure to ionizing and non-ionizing radiation. The three primary pillars of health physics in the context of CANDU operations are presented in Figure 1. Radiation shielding is presented as a pillar because it is an important aspect of radiation protection that is often a field in and of itself.

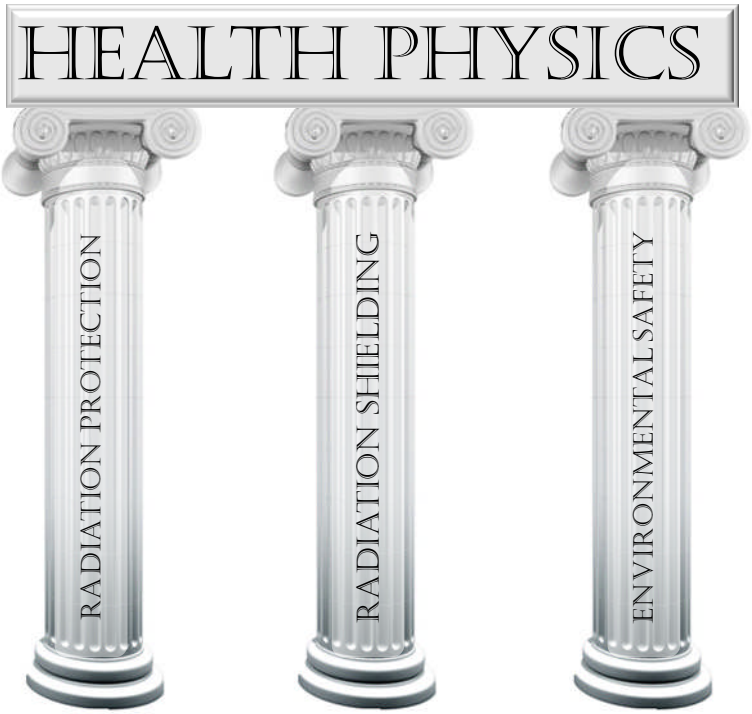

Figure 1 Three pillars of health physics

CANDU stations use experts in health physics, radiation protection, shielding, and environmental safety to ensure that reactor operations comply with Canadian regulations in a safe and controlled manner.

1.1 Overview

Experts in health physics generally require formal training in radiation protection, environmental radioactivity, and shielding design. In this chapter, the concepts of biological effects, radiation protection, health physics, shielding, and environmental safety are discussed with an emphasis on CANDU nuclear plant operations. This chapter represents the principal concepts related to the three pillars (Figure 1) which provide the well-rounded understanding of health physics that is required for a career in CANDU nuclear operations. Specific concepts discussed are: (1) basic fundamentals of radiation biology and the basics for understanding biological harm in living systems; (2) concepts of regulatory guidance which govern the “as low as rea-

sonably achievable" radiation dose paradigm; (3) radiation detection and monitoring techniques that may be used in CANDU plant environs; (4) external and internal radiation hazards, including discussions on shielding and personal protective equipment; (5) radiation management plans, worker dose monitoring, and control and waste management issues; and (6) radiation releases to the environment, derived release limits (DRL), and personnel and environmental protection.

1.2 Learning Outcomes

The goal of this chapter is for the student to understand:

- The biological basis for health impact from radiation exposure
- National and international regulatory guidance principles involving radiation exposure
- Overview of radiation-protection instrumentation used in CANDU plants
- The equations governing external exposure from point and extended sources
- Concepts of radiation shielding of external sources
- The equations governing internal exposure to radiation
- Concepts of radiation protection and management programs for CANDU plants
- Environmental radioactivity from CANDU plants
- Protection of the environment.

2 Biological Effects of Radiation

The principles of radiation biology are germane to background radiation-safety guidance for workers and the public with respect to CANDU operations. Personnel operating in and around CANDU plants, fuel-processing facilities, mining activities, and used nuclear fuel storage must have a fundamental understanding of how radiation may interact with the human body and how protective practices can be used to reduce any potential risk from exposure. In short, radiation biology is the backbone of all radiation protection and health physics.

Before discussing the mechanisms of how radiation affects living organisms, which form the basis of radiation protection, the basics of radiation interactions with matter must be understood. These concepts were discussed in Chapter 3 (Nuclear Processes and Neutron Physics). This section focuses on the fundamentals of how different types of radiation deposit energy in tissue through interactions and the principal target which drives radiation-protection regulation: deoxyribonucleic acid, or DNA. More information on the historical aspects of radiation biology may be found in [Preston2005].

2.1 Basic Radiation Interactions with Tissue

As discussed in Chapter 3, radiation interacts with materials in a variety of ways. Radiation can, for example, scatter, be absorbed, or create secondary particles as it passes through matter. The exact way in which radiation interacts with matter is dictated by the type of radiation and the nature of the matter being traversed. The endpoint(s) of the interaction(s) are dictated by the material being traversed. For example, neutrons impinging upon polymers can have the net effect of creating displaced atoms in the lattice (displacement damage), which can result in embrittlement of the material. The same neutrons impinging upon human tissue can cause changes to the DNA molecules, inducing chromosome aberrations which can lead to cancer. It is worth-while to note here, however, that radiation can also be used to strengthen polymers

and to treat cancer, so that the mode of application of radiation to material becomes important in determining the system endpoint. The fundamentals of radiation interactions with tissue begin with the concept of energy deposition.

Energy that is deposited in biological material can lead to two principal effects on an individual atom: excitation or ionization. Excitation raises an orbital-shell electron to a higher energy state without ejecting it from the atom; radiation can be emitted during this process. Ionization, on the other hand, is the ejection of an electron from an orbital shell of an atom; other kinds of radiation can also be emitted during this process. It is generally the ionization process that causes damage to DNA through direct or indirect action.

If DNA is the primary critical target for radiation-protection applications, then consideration must be given to DNA damage from direct and indirect action of the radiation to which tissue is exposed. DNA is a macromolecule which is the main constituent of chromosomes and the material that contains and communicates genetic information about all life. The DNA molecule is formed as a double helix and is coded with nucleotide base pairs along a twisted backbone of alternating sugar and phosphate groups, as depicted in Figure 2.

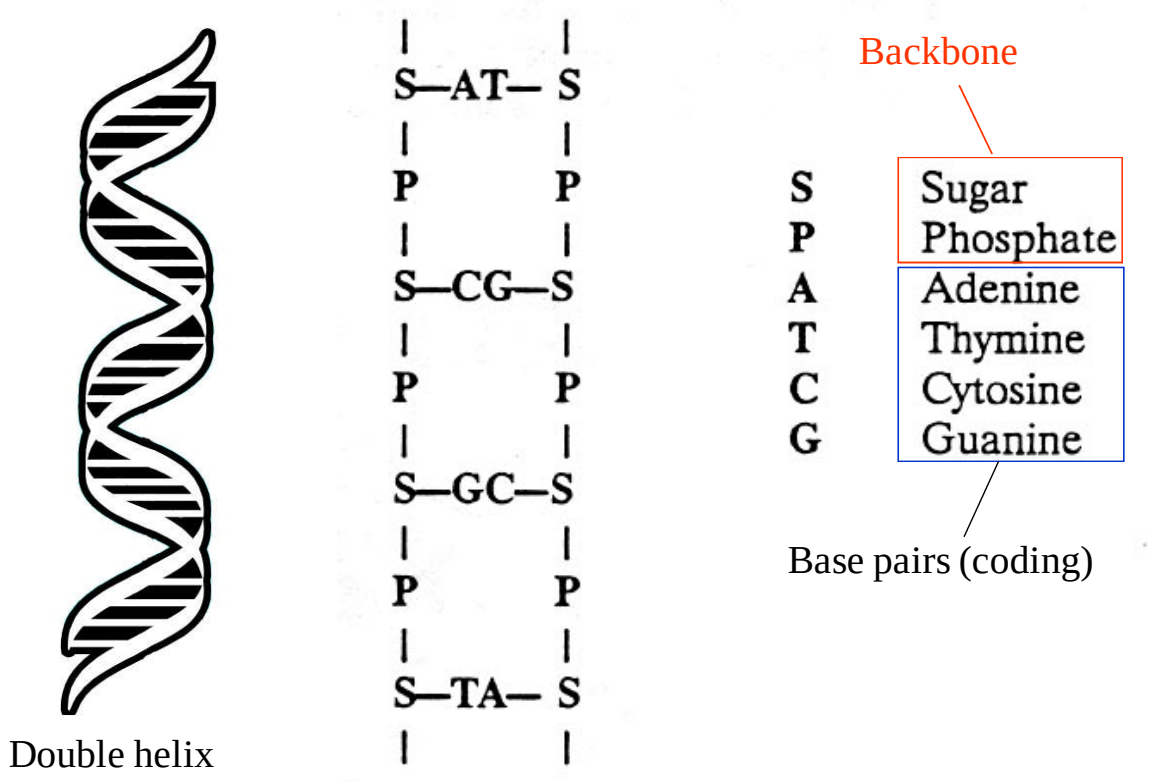

Figure 2 DNA double helix: target for radiation damage

Direct damage to DNA can occur when ionizing radiation interacts directly with the DNA molecule. For example, direct energy deposition on a backbone strand can have the effect of breaking the strand at that location. Direct damage is more probable for high linear energy transfer (LET) radiation, such as α-particles and neutrons. Indirect damage is a more complex process involving radiation interactions with water molecules. On average, the human body is composed of approximately 60% water [Cember2009], which is contained mostly within the body's

cells. Therefore, most direct interactions of all types of radiation on the human body are with water.

The process of indirect DNA damage proceeds as follows: radiation ionizes a water molecule, as described by Eq. (1):

$$H_2O \rightarrow H_2O^+ + e^- . \quad (1)$$

The products of water dissociation, H_2O^+ and e^-, interact further with water molecules. The electron interaction with water yields a hydrogen free radical, $H^\bullet$, and a hydroxide ion, OH^-, as shown in Eq. (2):

$$H_2O + e^- \rightarrow H_2O^- \rightarrow H^\bullet + OH^- . \quad (2)$$

The positive ion undergoes immediate dissociation as described in Eq. (3):

$$H_2O^+ \rightarrow H^+ + OH^\bullet , \quad (3)$$

which yields a positive hydrogen ion, H^+, and a free hydroxyl radical, $OH^\bullet$, the neutral form of the hydroxide ion, OH^-. The free hydrogen radicals tend to form gaseous hydrogen (H_2) which is relatively harmless in the human body. The hydroxyl radical, however, is extremely reactive and can damage DNA. These hydroxyl radicals are, however, short-lived and therefore need to be created in the vicinity of target DNA to do damage. An alternate damage mechanism is for the short-lived hydroxyl radicals to combine to form hydrogen peroxide as indicated in Eq. (4):

$$OH^\bullet + OH^\bullet \rightarrow H_2O_2 . \quad (4)$$

Hydrogen peroxide is a strong oxidizing agent which is fairly stable in the human body and can therefore migrate to distances far from the creation point and damage DNA.

A number of events on varying time scales must occur before deleterious effects from exposure are observed. The complex chain of events from radiation exposure to a biological endpoint is summarized in Figure 3.

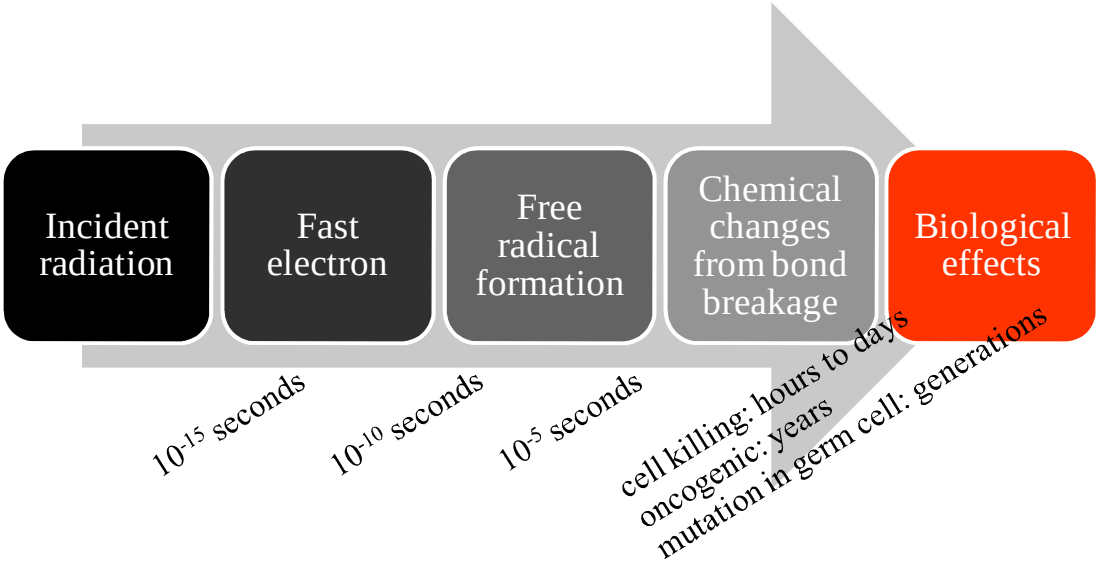

Figure 3 Process chain for radiation interaction with tissue

The time scales extend from femtoseconds for initial ionization, to tens of nanoseconds related to ion-radical lifetimes, to tens of microseconds for free-radical lifetimes, and finally to years post-exposure for onset of cancer.

The dominant modes of interactions for ionizing radiation that are important to CANDU reactors are contrasted in Table 1. The direct and indirect interaction mechanisms of the creation of charged particles by neutral particles are depicted in Figure 4. Dose assessment will be considered in Section 5 for external irradiation and Section 6 for internal irradiation.

Table 1 Radiation-induced DNA damage mechanisms

Radiation	Tissue interaction	Dominant damage mechanism
X- and γ rays	Indirect ionization	Production of fast-moving electrons creating free radicals which induce strand breaks
Neutrons		Production of recoil protons, alpha particles, and heavier atoms, creating free radicals which induce strand breaks
Beta particles (electrons)	Direct ionization	Direct action with DNA, inducing strand breaks. Directly ionizing particles can also generate free radicals.
Alpha particles (helium nuclei)		

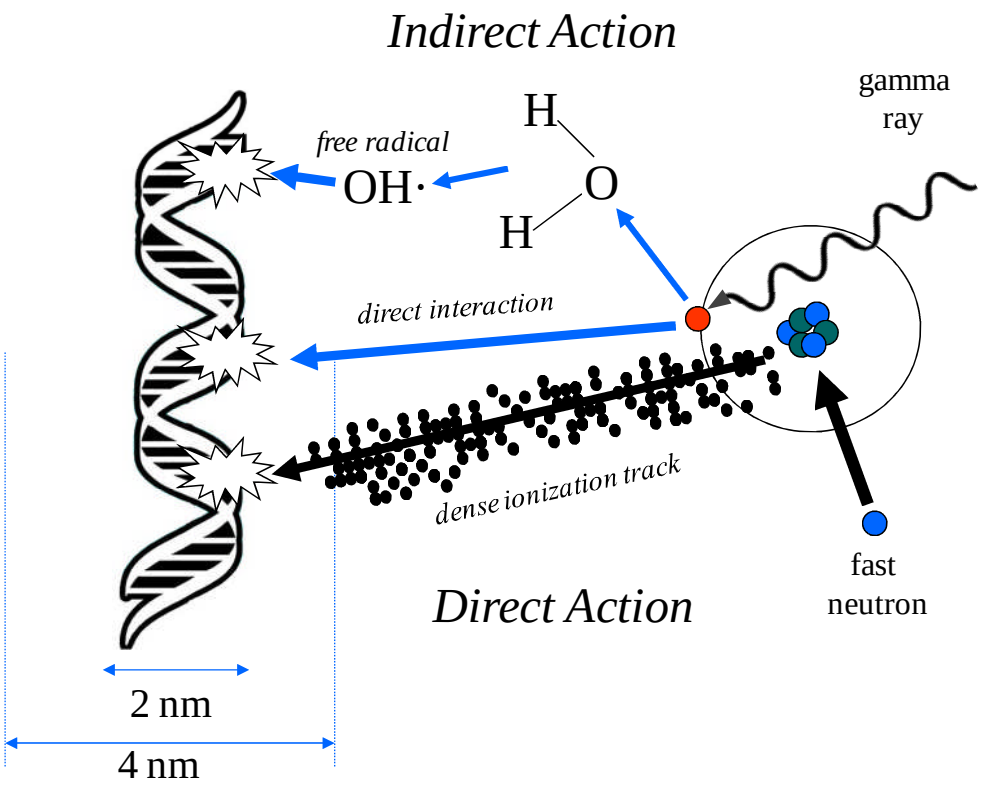

Figure 4 Direct and indirect radiation interaction mechanisms with DNA

2.2 Biological Radiosensitivity

The effect of radiation on living systems varies with the magnitude of exposure, duration of exposure, and region of the body exposed. Responses are characterized as deterministic (threshold) and stochastic (mutagenic, carcinogenic, or teratogenic). The law describing cellular radiosensitivity is attributed to Bergonie and Tribondeau [Bergonie1906] and states that the radiosensitivity of a cell is directly proportional to its reproductive activity and inversely proportional to its degree of differentiation. Cells most active in reproduction and cells not fully mature will be most harmed by radiation exposure. Conversely, the more mature and specialized the cell is, the less sensitive it is to radiation. One of the main reasons for the cause-effect relationship between radiation exposure and rapidly dividing cells is that the effect is essentially seen more rapidly when cells divide more rapidly.

2.2.1 Cell cycle

Cells are made up of various components. The cell membrane is the first line of defence for the cell and regulates what enters or exits the cell. The endoplasmic reticulum provides a transport mechanism from the cell membrane to the cell nucleus. Ribosomes in the cell are producers of proteins, and Golgi apparati store, package, help transmit, and customize proteins. Lysomes break down unwanted material in the cells, and mitochondria generate energy for the cell. For radiation-exposure risk, the most important component is the nucleus, specifically the chromosomes, which are depicted in Figure 5. Chromosomes are made up of DNA segments that code for proteins. Humans have 23 pairs of chromosomes. The DNA, an extremely small molecule in the chromosomes, carries the genetic code for all proteins. There are approximately $3x10^9$ base pairs, but only ~5% are active genes (the other 95% are the “glue” that holds the active genes together). Radiation can do harm when it hits an active gene.

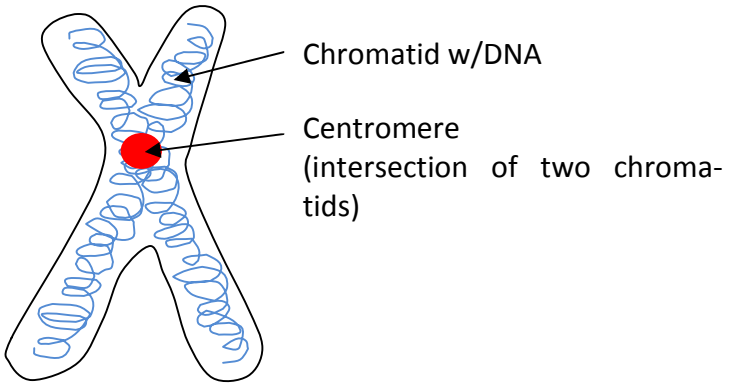

Figure 5 Chromosome

The cell cycle is depicted in Figure 6. The cell progresses from mitosis (cell division) to gap 1 (inactivity) for each divided cell which goes on to its own cycle, then to DNA synthesis, and then to gap 2 (inactivity) before mitosis.

The cell has maximum radiation sensitivity during the mitotic phase of the cell cycle. The cell cycle involves a number of stages before replication, as depicted in Figure 7.

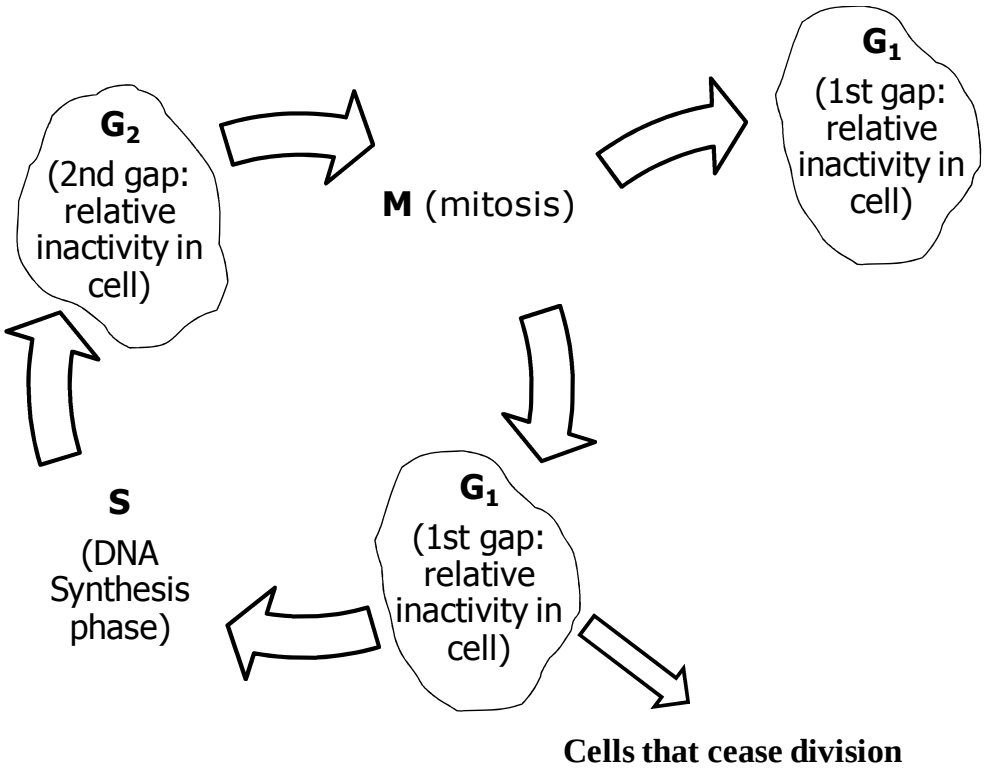

Figure 6 The cell cycle

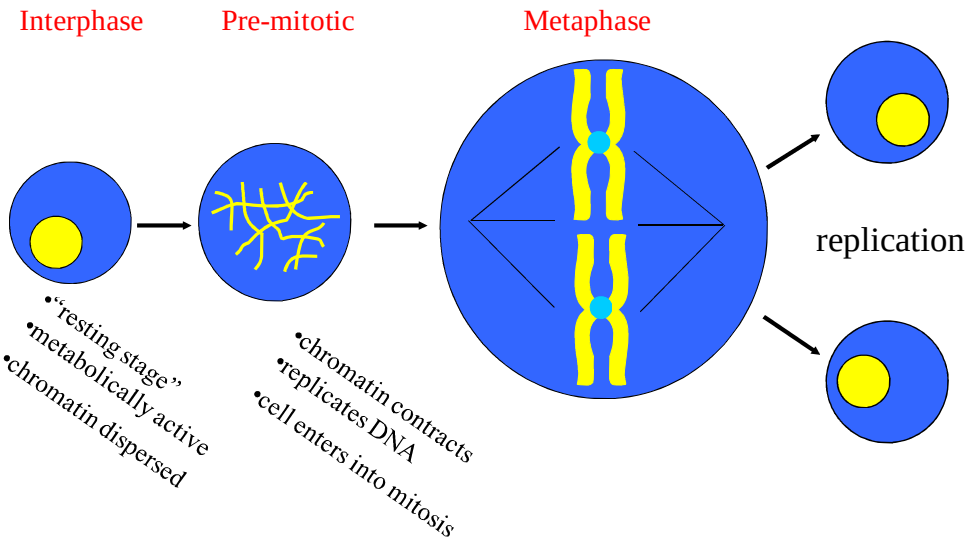

Figure 7 Cell mitosis

2.2.2 DNA damage

As previously stated, the primary target for cell killing is DNA. The primary lesion is the DNA strand break, and in the case of permanent damage, the DNA double-strand break must be considered. A DNA single-strand break is depicted in Figure 8. Radiation or free radicals interact with one of the sugar/phosphate backbone strands. This type of damage is of little risk because the DNA will either repair itself or, at worst, die upon replication.

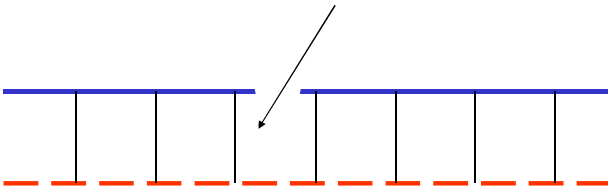

Figure 8 DNA single-strand break

A DNA double-strand break, on the other hand, is depicted in Figure 9. This type of damage can result either from a single, highly ionizing particle interacting with the DNA (such as an alpha particle) or from two events occurring so close in time that they occur, for all intents and purposes, at the same time. In this case, it is also possible for the DNA to repair itself if the strands reattach, but also for the DNA to misrepair itself, either with its own twisted DNA or with another DNA fragment. In this case, genetic coding can be produced that does not die upon replication, yet is not native to normal cellular function. This endpoint can be a precursor to cancer. This type of damage can be observed by examining chromosomes. Radiation damage to DNA can produce aberrations such as rings (a chromosome wrapped upon itself), fragments (small bits of "orphan" chromosomes), or dicentrics (two broken chromosomes which form together to make up a larger macromolecule).

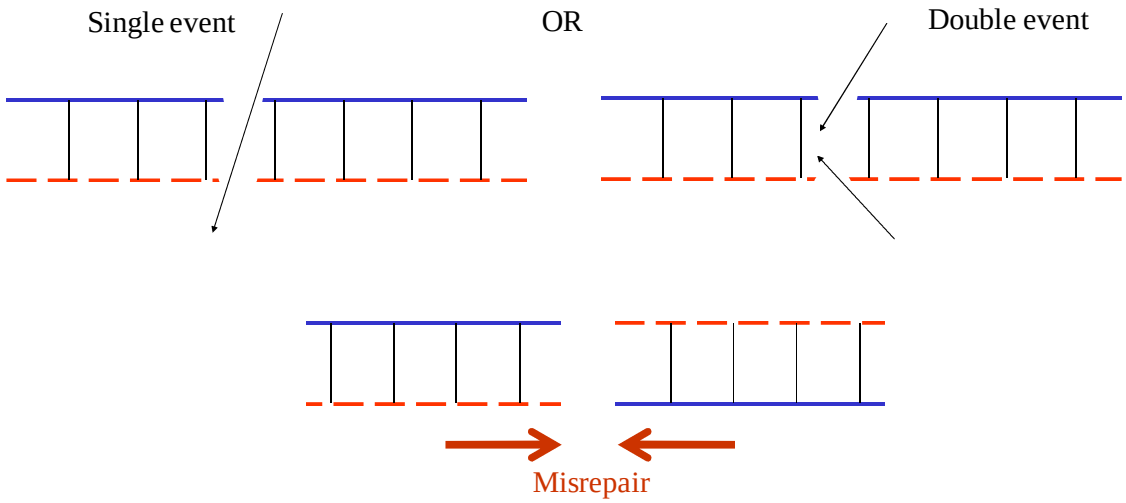

Figure 9 DNA double-strand break

A complete depiction of how radiation damages cells can be seen in Figure 10. During the interphase, radiation-induced strand breaks are generated. This leads to chromosome misrepair and abnormally repaired chromosomes during DNA replication, yielding dicentrics, fragments, (and rings). Note that there is a background level of approximately one dicentric per 1000 lymphocytes in human blood, and therefore excess dicentrics are indicative of radiation damage. During metaphase, the cells divide and can undergo either apoptosis or mutation.

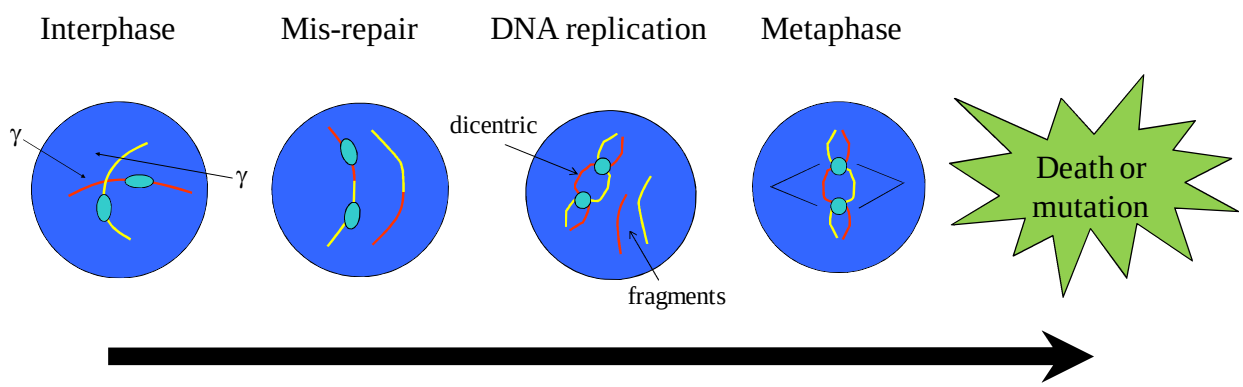

Figure 10 Radiation damage to cells

The possible endpoints of damaged DNA are graphically depicted in Figure 11. Upon DNA damage from ionizing radiation exposure, the DNA may (1) repair itself without coding error (common with single-strand breaks), (2) perform cell "suicide" (apoptosis), or (3) mutate into a viable, yet undesirable, DNA molecule that may have oncogenic (tumour-producing) properties upon cell division.

Various factors can impact the extent of radiation damage at the cellular level, including:

- Dose – generally speaking, as the dose increases, the surviving fraction of cells decreases. In other words, increasing the dose increases the damage.
- Dose rate – as the dose rate increases, the surviving fraction decreases.
- Radiation quality and relative biological effectiveness – for low linear energy transfer (LET) radiation, it requires more than one track to cause a dicentric; for high-LET radiation, one track can cause a dicentric. As a general rule, high-LET radiation is more damaging to cells than low-LET radiation.
- Oxygenation – aerated cells are more radiosensitive. Oxygen reacts with free radicals to produce peroxide, which is highly toxic.
- Radioprotectants – if present, radioprotectors such as aminothiols can scavenge free radicals before DNA damage occurs.

The biological effects of exposure are explored in Section 2.3.

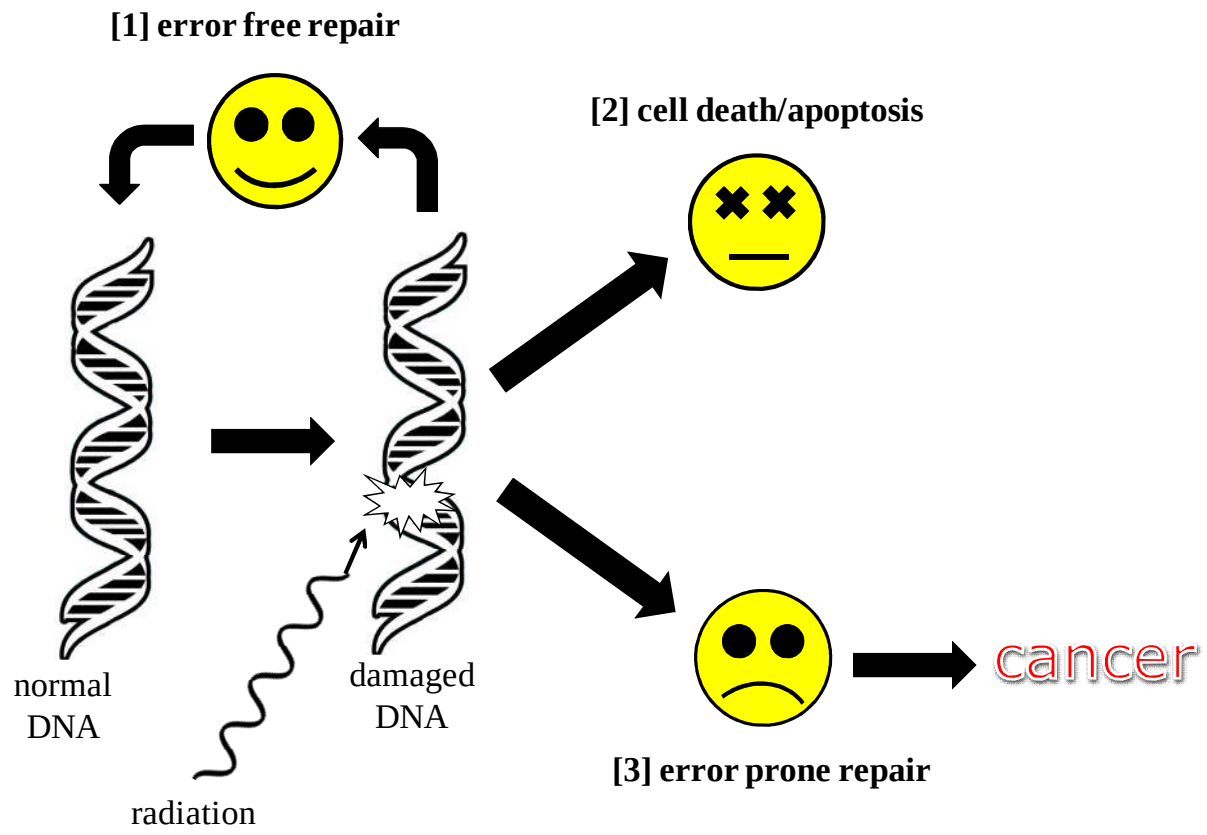

Figure 11 DNA damage endpoints

2.3 Biological Effects of Exposure

In the previous section, factors affecting the extent of radiation damage to cells were explored. A variety of factors that influence the biological effects of exposure must be taken into account, such as:

- Total absorbed energy (dose)
- Rate of dose delivery
    - Acute (seconds-minutes)
    - Chronic (days-years)
    - Type of radiation
- Source of exposure
    - External
    - Internal
- Age at exposure
- Time since exposure
- Area of exposure
    - Localized (cells; organs)
    - Extremities (hands; feet)
    - Whole body
    - Superficial (skin only)
    - Deep tissue.

As presented above, exposure can be categorized in many ways. Two very important concepts related to risk are the rate of dose delivery for exposure and the endpoint.

The rate of dose delivery for exposure is conveniently divided into two principal categories: (a) acute and (b) chronic. Acute exposure is normally considered as a single, large, and short-term whole-body dose, and the effects are observed in a short time frame post-exposure. The effects are categorized by four sequential stages: (i) Initial (prodromal) which lasts approximately 48 hours, (ii) latent, which lasts 48 hours to 3 weeks, (iii) manifest illness, which lasts 6–8 weeks, and (iv) recovery, which can last weeks to months (if death does not occur). Chronic exposure, on the other hand, is typically defined as a lower, protracted dose over a long period of time. The two principal categories describing the endpoint effects of radiation exposure in humans are (a) deterministic effects and (b) stochastic effects. Dose definitions (Gy, Sv, etc.) are provided in Section 3.1.

2.3.1 Deterministic effects

Deterministic effects are those that manifest at some threshold dose, with increasing severity with increasing dose. Some estimated deterministic effects as a function of absorbed dose from X-ray, gamma-ray, and electron exposure of target organs is provided in Table 2 [NCRP2001].

Table 2 Deterministic effects [NCRP2001]

Effect	Target organ	Absorbed dose (Gy)
Sterility (temporary)	Testes	0.15
Nausea	Whole body	0.35
RBC depression	Bone marrow	0.5
Skin reddening (reversible)	Skin	2
Sterility (permanent)	Ovaries Testes	2.5–6 3.5
Vomiting	Whole body	3
Hair loss (temporary)	Skin	3–5
Erythema	Skin	5–6
Death ($LD_{50/30}$)†	Whole body	4

†$LD_{50/30}$ is the Lethal Dose where 50% of an exposed population dies within 30 days. $LD_{50/60}$ is a similar definition at 60 days post-exposure.

There are three general categories of acute radiation sickness, as presented below.

1. Hematopoietic
   - Dose range 3–8 Gy
   - Radiation damages precursors to red and white blood cells and platelets
   - Prodromal phase may occur immediately
   - Symptoms include septicemia
   - Mixed survival
   - Examples include Chernobyl personnel (203 exhibited symptoms; 13 died)

2. Gastrointestinal
   - Dose > 10 Gy
   - Symptoms include abdominal pain/fever, diarrhea, and dehydration
   - Death in 3–10 days (no record of human survivors above 10 Gy)
   - Examples include Chernobyl firefighters
3. Cerebrovascular
   - Dose > 100 Gy
   - Death in minutes to hours
   - Examples include criticality accidents.

Radiation exposure to high doses is well known to induce changes in blood-count levels. There are numerous components of blood and a variety of ways to subdivide blood characteristics. The four factors that are often examined when high radiation exposure is expected are lymphocytes, neutrophils, thrombocytes, and hemoglobin, described below:

- Lymphocytes are white blood cells that govern the body's immune response (directly fighting disease and infection)
- Neutrophils are a subset of white blood cells that fight infection
- Thrombocytes (platelets) are active in blood clotting
- Hemoglobin is the component of red blood cells that carries oxygen.

An example of a hypothetical complete blood count with differential (CBC w/ diff) after exposure to a large dose (insult > 2 Gy whole body) of radiation is provided in Figure 12. It may be seen that immediately after exposure, there is a sharp decrease in lymphocytes, followed by a spike in neutrophils. The neutrophil spike is the body's response to what it believes may be an infection. Three distinct stages of illness are evident through the exposure: (1) latent period from exposure to approximately 20 days, (2) manifest illness, where the exposed person is extremely sick, from day 20 to 35, followed by (3) recovery, assuming the exposed person has survived the blood counts dropping to almost zero. Note that Figure 12 represents a very high whole-body exposure and that the actual blood counts and time frames can vary greatly from person to person. Note also that models have been developed to determine the severity of radiation exposure retrospectively using sequential lymphocyte counts post-exposure [Goans1997].

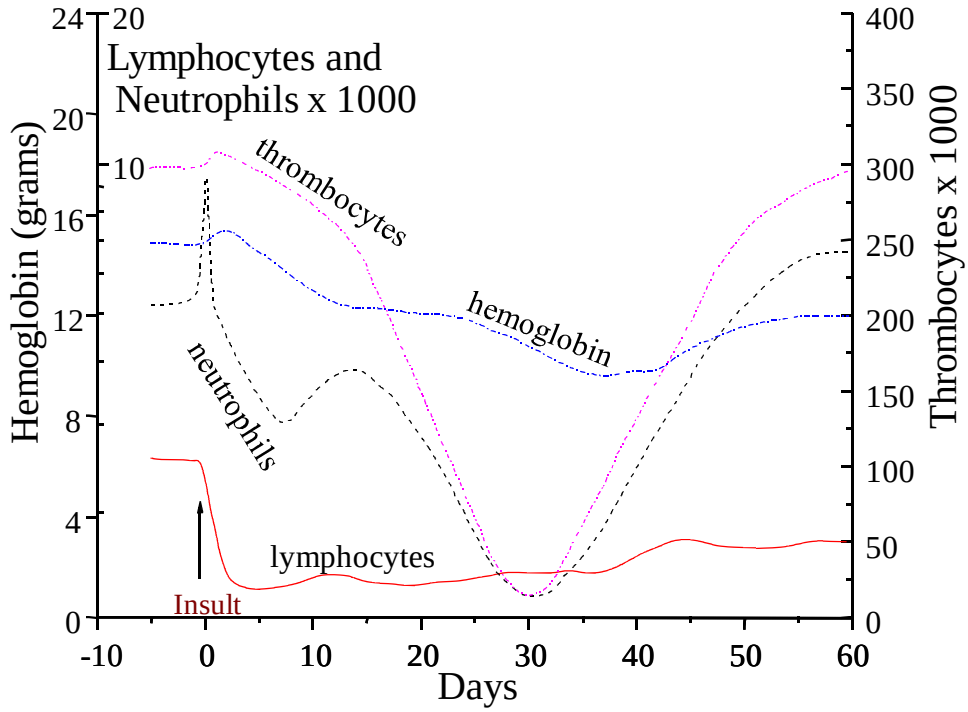

Figure 12 Differential blood count after severe radiation insult (adapted from [Gusev2001] and [Cember2009])

The universal principles of radiation protection dictate that deterministic effects must be avoided in occupational exposure. However, this is not the case with medical exposure because a deterministic-level dose may be required for the medical procedure. A typical example is radiation therapy for treatment of cancer, where deterministic-level doses are routinely applied to treat the cancer.

2.3.2 Stochastic effects

Stochastic effects are those effects which are assumed not to have a threshold dose and for which the severity of the effect is independent of absorbed dose. Stochastic effects may manifest in delayed or late somatic effects. The usual endpoint stochastic effect considered for radiation exposure is cancer induction (for example, leukemia, bone cancer, or lung cancer), which has been inferred at absorbed-dose levels on the order of hundreds of mGy. In addition, genetic and hereditary effects have been observed using animal studies, although these effects have never been observed in human populations exposed to ionizing radiation. Stochastic effects can occur from both acute and chronic exposures.

The often-quoted value for fatal cancer risk as a function of whole-body radiation exposure is 5% per Sv (dose definitions are found in Section 3.1). In other words, if a population of 100,000 were exposed to a 1-Sv whole-body dose, then 5000 excess fatal cancers would be expected (note that there is approximately a 40% baseline fatal cancer incidence across the population). A tabulation of the current ICRP [ICRP2007] guidance on stochastic risk coefficients is provided in Table 3. ICRP recommends continued use of the 5% per Sv value.

Table 3 Stochastic risk coefficients (adapted from ICRP (2007))

Exposed population	Cancer (%/Sv)	Heritable (%/Sv)	Total (%/Sv)
All	5.5	0.2	5.7
Adult only	4.1	0.1	4.2

A study performed by [Brenner2003] indicated that, for doses within a range of 50–100 mSv (protracted exposure) and 10–50 mSv (acute exposure), there was direct epidemiological evidence from human populations to indicate that exposure to ionizing radiation increased the risk of certain cancers. The primary argument to support linear non-threshold response at low dose is not from scientific observation, but from mechanistic arguments and assumptions. Based upon the scientifically observed evidence, there is no conclusive proof of any deleterious endpoint effect below 10 mSv and only weak evidence of an effect below about 50 mSv. In any case, the risk factors below approximately 100 mSv are very low for all exposure scenarios.

Note that effects from radiation exposure cannot be definitively attributed to an individual. That is to say, a deterministic effect that may be observed in an individual at one dose level may not be observed in another individual at the same dose level. This occurs because individuals have different sensitivities to ionizing radiation exposure. Likewise, cancer risk from radiation exposure cannot be extrapolated to an individual for a given exposure. Radiosensitivity is organ-specific because not all organs have the same vulnerability to radiation. Table 4 depicts the relative sensitivity variation among some critical organs.

Table 4 Relative human-organ vulnerabilities to ionizing radiation

	LOW MEDIUM HIGH VERY HIGH
Lymphocytes	
Intestinal epithelium	
Spermatogonia	
Urinary bladder epithelium	
Gastric mucosa	
Epidermal epithelium	
Optic lens	
Growing bone	
Pulmonary epithelium	
Renal epithelium	
Thyroid	
Mature hematopoietic cells	
Mature bone	
Muscle	
Ganglion	

Although cellular and animal studies provide indicators of risk, the primary source of data for radiation-exposure risk modeling is epidemiological studies [UNSCEAR2008]. The primary source of data used for risk models is the Japanese atomic-bomb survivors. Other exposed groups that have contributed data to risk modeling include radiotherapy cancer patients (cervical, endometrial, childhood, breast, Hodgkin's lymphoma, etc.), radiotherapy patients with non-malignant conditions (spondylitis, thymus, tonsils, ringworm, etc.), diagnostic radiology patients (tuberculosis fluoroscopy, pelvimetry, scoliosis, etc.), workers with occupational

exposure (radium dial painters, miners, radiologists, nuclear workers, etc.), and people experiencing environmental exposure (nuclear weapons fallout, Chernobyl, Techa River, etc.).

Radiation is an incredibly weak carcinogen, and as therefore estimation of risk from exposure at low doses requires the use of risk models, which are described in the following section.

2.4 Radiation Risk Models

In the simplest terms, risk may be defined by Eq. (5):

$$\text{Risk} = \left(\text{Probability of occurrence}\right) \times \left(\text{Consequence}\right) \tag{5}$$

Therefore, it is possible to have a low-probability event with high consequence with a low risk factor, or a high-probability event with low consequence with a low risk factor. However, if neither probability of occurrence nor consequence is non-trivial, then it is possible to have a high risk factor. In terms of radiation risk, the BEIR Committee [BEIRVII2006] defines risk as:

> "the chance of injury, loss, or detriment; a measure of the deleterious effects that may be expected as a result of action or inaction".

To understand risk from radiation exposure, a fundamental understanding of what the risk detriment parameter is (for example, cancer) and the background incidence of this detriment is required. Incidence of effect, or endpoint, can be estimated using *in vitro* or *in vivo* studies of humans, animals, or plants (of organisms, components of organisms, or both, down to the DNA level). Risk to populations from exposure can be estimated using the principles of epidemiology, which is the study of the causal factors of the frequency of disease in humans.

Risk from radiation exposure is typically quantified as some probability of endpoint effect (incidence) per unit dose exposure. The relative risk (RR) is a common measure of risk and is defined by Eq. (6):

$$\text{RR} = \frac{\text{Incidence in exposed population}}{\text{Incidence in unexposed population}}. \tag{6}$$

An important metric for risk estimate is based on the increased likelihood of an endpoint (cancer) as a result of exposure to a carcinogen (radiation) and is termed the excess relative risk (ERR), defined by Eq. (7):

$$\text{ERR} = \text{RR} - 1 = \frac{\text{Incidence in exposed population}}{\text{Incidence in unexposed population}} - 1. \tag{7}$$

The excess absolute risk (EAR), sometimes called the attributable risk, is a measure of the discrepancy in incidence rates between exposed and non-exposed populations and is given by Eq. (8):

$$\text{EAR} = \text{Incidence in exposed population} - \text{Incidence in unexposed population}. \tag{8}$$

For example, if the cancer incidence in an exposed population is 0.5 Gy^{-1} and in an unexposed population is 0.4, the RR is 1.25 Gy^{-1}, and the ERR is 0.25 Gy^{-1}. The EAR requires that the incidence in the unexposed population be resolved to a percentage by multiplying by the exposure (Gy). For a 100-mGy exposure, RR = 0.125, ERR = 0.025, and EAR is negative. When RR >1, then

exposure increases the risk, and if RR <1, then exposure does not increase risk. In this hypothetical case, the exposure does not increase cancer risk.

It is often desirable to estimate whether a given exposure is the cause of an observed endpoint (for example, cancer). The metric used for this is the probability of causation (PC), which is given by Eq. (9):

$$\mathrm{PC}=\frac{\mathrm{ERR}}{1+\mathrm{ERR}}. \tag{9}$$

In the example above, the PC is calculated as 0.024.

Understanding the risks of radiation exposure means understanding the cause-effect relationships between endpoints and exposure. Some cancers are more strongly linked to radiation exposure than others. Table 5 presents a variety of cancers in the following categories: (i) strong association with radiation exposure, (ii) moderate association with radiation exposure, although strongly influenced by other risk factors, and (ii) little or no association with radiation exposure.

Table 5 Association of cancers with radiation exposure

Strong	Moderate	Weak to none
Leukemia (excluding chronic lymphocytic)	Childhood leukemia	Bone
Thyroid	Bladder	Brain
	Breast	Esophageal
	Colorectal	Kidney
	Lung	Multiple myeloma
	Stomach	Non-Hodgkin's lymphoma
		Ovarian

Stochastic risk, such as the incidence of cancer, can be plotted as a function of dose to generate a dose-response curve that represents the incidence of an endpoint as a function of increasing dose. There are numerous sources of data for generating risk curves; some major cohorts where data have been obtained are presented in Table 6.

Table 6 Epidemiological categories for radiation risk studies

Population Category	Comments
Japanese atomic bomb survivors	Numerous endpoints studied
Environmental exposures	Events: Chernobyl NPP4 accident Techa River cohort Fukushima disaster Nuclear-weapons fallout Natural background radiation
Occupational exposures	Cohorts: Radium dial painters Miners Radiologists Technologists Nuclear-energy workers Veterans
Radiotherapy cancer treatments	Endpoint cancers: Cervical Endometrial Childhood Breast Hodgkin's lymphoma
Radiation therapies (non-cancer)	Conditions treated: Spondylitis Mastitis Thymus Infertility Tonsils Menstrual disorders Otitis Media Ulcers Ringworm Hemangioma
Diagnostic radiology	Conditions diagnosed: Tuberculosis Scoliosis Pelvimetry All other diagnostic procedures
Radionuclide internalizations	Isotopes of: Th, I, P, Ra, U, Pu

Generally speaking, much of our knowledge about dose response has been obtained from relatively high dose / high dose rate exposure, whereas most non-medical exposure is low dose / low dose rate exposure. Figure 13 depicts a hypothetical, yet realistic, dose response curve (incidence versus dose, where incidence could be relative risk of cancer, for example). The solid

circles represent points for which there are statistically significant data with reasonable confidence levels (for example, data from Japanese atomic bomb survivors). The open circles represent non-statistically significant data with low confidence. “High dose” would generally refer to data obtained above a 1-Gy whole-body dose; “low dose” would generally refer to data below approximately 0.1 Gy. Some risk estimates (for example, leukemia) are generated using a linear quadratic curve [BEIRVII2006] through the data and through the point (0,0), which represents “zero risk at zero dose”. Most risk estimates, however, are generated using a linear curve through the data and through point (0,0). This is termed the linear no-threshold (LNT) hypothesis of radiation risk and is very contentious in the radiation-protection community [Bond1996] [Scott2008] [Siegel2012].

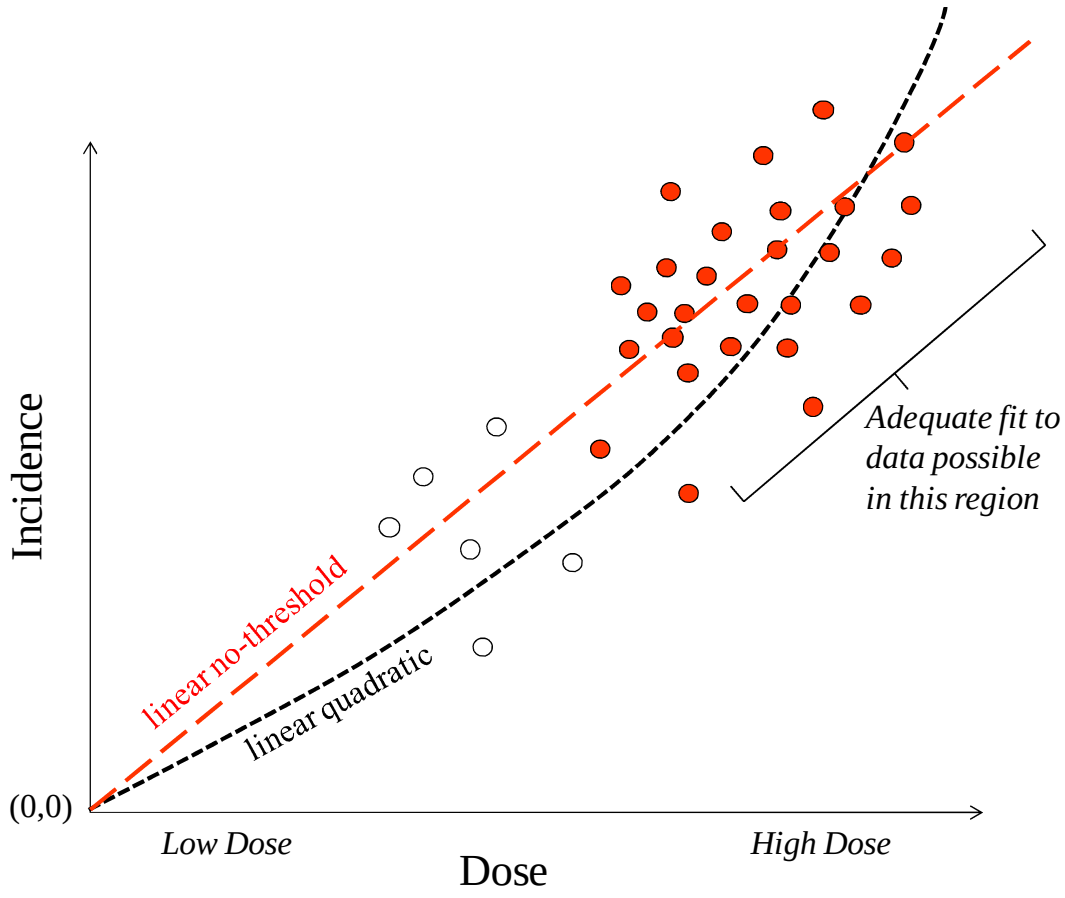

Figure 13 Dose-response curve showing two common models

The primary reason why the linear no-threshold hypothesis of radiation risk is contentious is that, as discussed in Section 2.3, there is no conclusive evidence below approximately 100 mGy to demonstrate deleterious effects of exposure to ionizing radiation. That being said, regulators have adopted the linear no-threshold hypothesis for radiation risk because it is considered conservative (that is, to overestimate the risk from ionizing-radiation exposure).

The various models for dose response [Cember2009] can be represented by a generalized expression given by Eq. (10):

$$f(D) = \left(\alpha_0 + \alpha_1 D + \alpha_2 D^2\right) e^{\left(-\beta_1 D - \beta_2 D^2\right)}, \tag{10}$$

where the dose-response function is given by *f(D), D* is the dose, α_0 is the natural incidence of the effect (for example, spontaneous cancer incidence), α_1 and α_2 are linear and quadratic fitting parameters, and β_1 and β_2 are parameters used in representation of cell killing or mortality only at higher doses (that is, the killing function dominates at high dose).

In terms of relative risk (RR), Eq. (10) becomes Eq. (11):

$$RR = \frac{f(D)}{\alpha_0} = \frac{\left(\alpha_0 + \alpha_1 D + \alpha_2 D^2\right) e^{\left(-\beta_1 D - \beta_2 D^2\right)}}{\alpha_0}. \qquad (11)$$

A graphical representation of the various functions for Eq. (10) is provided in Figure 14. The values for the fitting parameters have been selected arbitrarily, and a background incidence (of a given effect) was arbitrarily selected as 10^{-3}. At low dose, the linear, linear quadratic, and full expressions are basically identical. At high dose, the quadratic function matches the linear quadratic function, and clearly cell killing/mortality attenuates the linear quadratic curve.

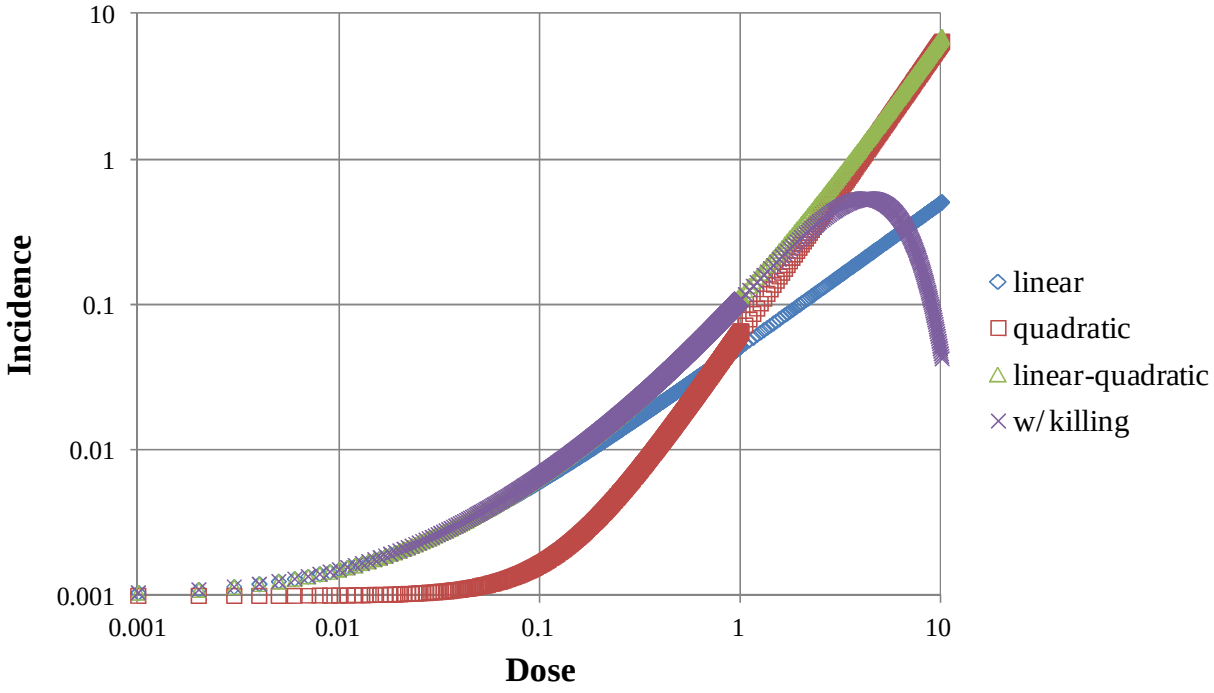

Figure 14 Generalized dose-response curves

Alternative theories of dose response include models that incorporate threshold responses (that is, risk only increases after a threshold dose) and hormetic response (see Figure 15).

Hormesis may be defined as "a process in which exposure to a low dose of a given insult that is damaging at higher doses induces an adaptive beneficial effect on the cell or organism" [Mattson2008]. There has been an abundance of literature published in support of both threshold and hormetic responses (see, for example, the journal Dose-Response [DoseResponse2013]). The relative risk-response curve may be adjusted to compensate for low-dose hormetic effects using Eq. (12):

$$RR = \frac{\left(\alpha_0 + \alpha_1 D + \alpha_2 D^2\right) e^{\left(-\beta_1 D - \beta_2 D^2\right)}}{\alpha_0} \left[1 - B(D) * PROFAC\right], \quad (12)$$

where *PROFAC* is the protection factor, which is the population average probability of cancer prevention given activated natural protection and *B(D)* is the benefit function, which is the probability of activated natural protection [Scott2012], which increases at low doses far more than *f(D)*. At low doses, $f(D)/\alpha_0$ is essentially unity, and at low doses, the relative risk may be approximated by Eq. (13):

$$RR \approx \left[1 - B(D) * PROFAC\right]. \quad (13)$$

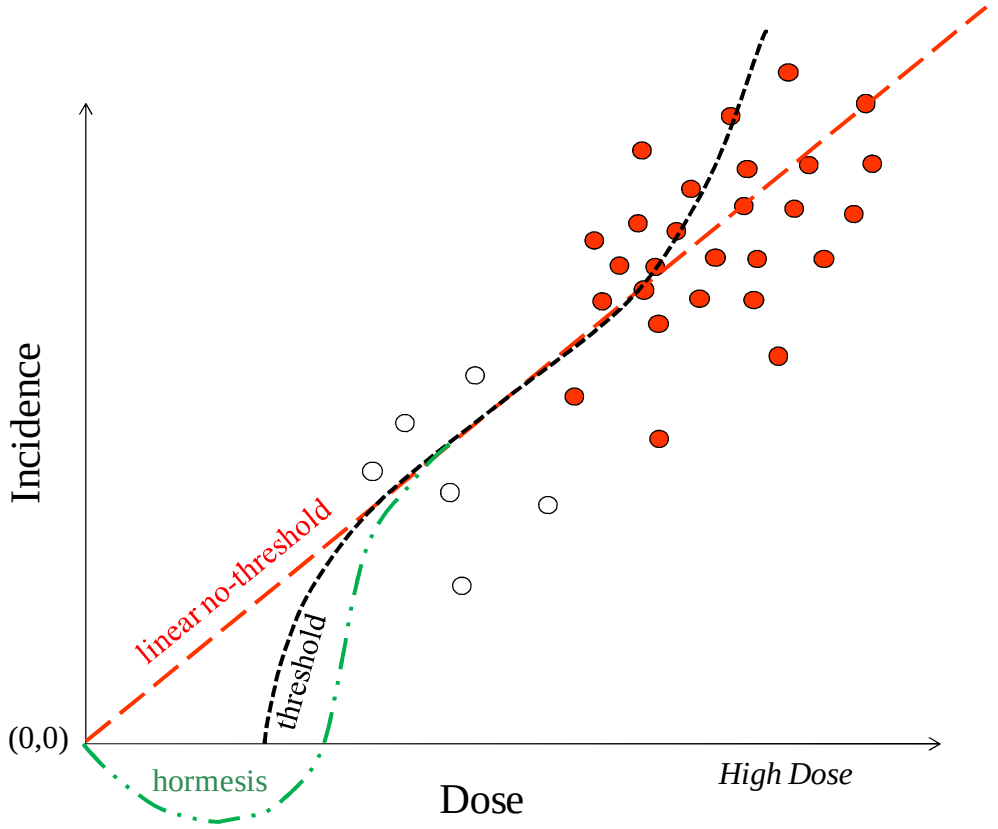

Figure 15 Alternative dose-response curves

2.4.1 Radiation risk controversy

Almost immediately after the discovery of X-rays by Wilhelm Roentgen in late 1895, the emanations were being put to use for therapeutic and diagnostic medical applications. The following year, reports of skin damage due to X-ray exposure were published. Early radiation-protection guidance focused on eliminating deterministic effects such as skin erythema. By the beginning of the 20th century, evidence was appearing that radiation might be responsible for tumours in radiologists, and supporting evidence of cancer induction due to radiation was obtained by studying pitchblende miners and radium dial painters. No formal standards for radiation protection existed until 1913, and at that point, they were established at thresholds for limiting skin erythema and subsequently expanded for internalized radionuclides in terms of maximum permissible body burden. Our current system of understanding of effects of ionizing radiation exposure has been developed over the past 100 years by (a) performing cellular-level studies,

(b) performing animal studies, and (c) performing epidemiological studies on exposed populations [BEIRVII2006].

Scientific debate on low-level radiation effects in the radiation-protection community involves the model used to determine risk from exposure. There is no unanimous agreement on the model of stochastic risk as low doses. There is scientific support for the linear-non-threshold model, threshold dose models, and also hermetic models. Although our current system of radiation-protection guidance is based upon the linear non-threshold model of dose-response, the concept of threshold dose is apparent as early as the writings of Paracelsus (1493-1541), in his famous statement "*What is there that is not poison? All things are poison and nothing is without poison. Solely the dose determines that a thing is not a poison*" [Deichmann1986]. In abbreviated terms, Paracelsus' statement becomes "The dose makes the poison". This suggests that, upon exposure to an anthropogenic chemical, there is a threshold dose above which something becomes poisonous to the body. The concept of threshold dose, established in the 16th century, was abandoned for stochastic-effect radiation-protection purposes in the 20th century. As early as the 1955 Recommendations of the International Commission on Radiological Protection, statements were promulgated that "*no radiation level higher than natural background can be regarded as absolutely 'safe'* " [Clarke2005], despite there being no conclusive proof that low-dose radiation causes any harm, stochastic or non-stochastic. It is this belief, manifest in the so-called linear non-threshold (LNT) hypothesis of radiation risk described above, that has divided public opinion and worker understanding of the effects of ionizing radiation. Various misapplications and misinterpretations of radiation risk using the linear non-threshold hypothesis [Bond1996] [Siegel2012] have made general understanding of the risk of exposure to low-dose radiation confusing and controversial.

2.5 Effects on Pregnancy

In much of the literature, the terms zygote, embryo, fetus, and conceptus are used. Accepted medical definitions are provided as follows [Mosby1990]:

Zygote [Gk *zygon+sporos* seed]: the developing ovum from the time it is fertilized until it is implanted in the uterus.

Embryo [Gk *en* in, *bryein* to grow]: 1. Any organism in the earliest stages of development; 2. in humans, the stage of prenatal development between the time of implantation of the fertilized ovum about 2 weeks after conception until the end of the seventh or eighth week. The period is characterized by rapid growth, differentiation of the major organ systems, and development of the main external features.

Fetus [L, fruitful]: the unborn offspring of any viviparous animal after it has attained the particular form of the species, more specifically, the human being *in utero* after the embryonic period and the beginning of the development of the structural features, usually from the eighth week after fertilization until birth.

Conceptus [L *concipere* to take over]: the product of conception; the fertilized ovum and its enclosing membranes at all stages of intrauterine development, from the time of implantation to birth.

2.5.1 Teratogenesis

The conceptus is a blanket definition which encompasses the zygote, embryo, and fetus stages. Teratogenic insult may be defined as any exposure that is capable of harming the conceptus (for example, inducing birth defects). Radiation is extremely harmful to the conceptus, and to children in general, for two simple reasons: (1) a majority of the cells are in active growth stages in young organisms, and as was demonstrated by the law of Bergonie and Tribondeau, rapidly dividing cells are more radiosensitive; and (2) exposure in the young allows for more time for observable effects to manifest before death.

The human conceptus has an approximate nine-month gestation period, with different radiation sensitivities over this time frame. During the first two weeks of gestation (the zygote stage), the egg is fertilized and implanted in the uterine wall. During this phase, the cells divide at a rapid rate and are highly sensitive to radiation. During this stage, cell killing is the most likely outcome from large doses of radiation. During the following approximately three to seven weeks of gestation (the embryonic stage), differentiated organ development begins, and the embryo is susceptible to cell killing or congenital malformations at moderate to high doses of radiation exposure. From approximately week 8 until birth (the fetal stage), there is rapid growth which makes the fetus susceptible to birth anomalies such as mental retardation. After approximately 17 weeks gestation, all the brain cells have formed and do not further divide, and therefore the most sensitive period for teratogenic effects such as mental retardation is believed to be between 8 and 17 weeks gestation for moderate- to high-dose radiation exposure.

The manifestation of stochastic effects (such as cancer-cell induction) is more probable for *in-utero* exposure. This is recognized by the International Commission on Radiological Protection by suggesting that the risk is a few times that of the population as a whole. Using the value of 5.5%/Sv from Table 3, the cancer-risk coefficient for *in-utero* exposure is estimated at approximately 17%/Sv. Note also that most regulatory agencies have adopted a "balance of pregnancy" dose limit on workers, which is designed to limit *in-utero* exposure. Operationally, when a worker declares a pregnancy to her employer, the dose to the conceptus must be limited to a pre-determined value (in the 2007 ICRP recommendations [ICRP2007], this is 1 mSv).

2.5.2 Perception of risk

Of interest in pregnancy and radiation exposure is the perception of risk by the mother. There is very little scientific evidence that points to increased risk of harm to the conceptus at low doses. To attempt to answer the question, "*How do pregnant women perceive the risk of low-level radiation exposure?*" [Bentur1991] examined the perception of teratogenic risk for exposure to ionizing radiation during pregnancy, specifically considering women who were scheduled to receive diagnostic radiation exams during pregnancy. The control group for the study was women exposed to non-teratogenic drugs and chemicals during their pregnancy. In this study, [Bentur1991] claimed that, *"The two probable factors in creating this misperception of risk are pregnancy-induced anxiety and misinformation. Anxiety is enhanced by the known effects of nuclear disasters and due to continuous flow of information in the medical literature about the long-term effects of therapeutic irradiation on fertility and carcinogenicity".* In the study, women were counselled about the low risk factors for diagnostic radiology procedures. Before counselling, the pregnant women exposed to radiation assigned themselves a higher risk than the non-exposed population, even though their doses from diagnostic procedures would

be extremely low. After counselling, the perception of teratogenic risk did not differ between the groups. This result is very important because it suggests that education and consultation with pregnant women who may be exposed to ionizing radiation can reduce stress and anxiety from the exposure and possibly reduce the risk of the mother selecting therapeutic abortion due to unfounded perceptions of the exposure.

2.6 Summary

From a toxicological standpoint, more is known about the effects of radiation on living organisms than any other insult, including chemicals. As far as carcinogenic toxins go, ionizing radiation is one of the weakest carcinogens on the planet. Effects have been investigated using studies of DNA/molecules/cells, studies on animals, and human epidemiological studies. The time frames for observable effects extend from subseconds (physical) to seconds (chemical) to many years (biological). The endpoints for effects may be (i) radiation enters the body, but misses important targets (a highly likely "endpoint"); (ii) radiation does not cause any damage to the targets; (iii) radiation damages the target, but the target repairs itself; (iv) damaged cells may die; and (v) damage cells may change (mutate). High doses of radiation may cause prompt deterministic effects, such as skin burns. Both high and moderate doses of radiation may increase the risk of stochastic effects such as cancer. Low-dose radiation exposure had not been observed to cause deleterious effects in the populations studied, although the risk of exposure has been extrapolated from high-dose observable effects to the low-dose regime using the linear no-threshold (LNT) dose-response model.

What is known about ionizing radiation exposure can be summarized as:

- Radiation is a very weak carcinogen
- Probability of cancer is a function of dose (increased risk with increased dose; severity of cancer is not a function of dose)
- There is weak-to-no evidence of cancer effects at doses below approximately 100 mSv.

What is NOT known about ionizing radiation exposure:

- Whether there are any negative effects of exposure below approximately 100 mSv
- Whether there are any beneficial effects of exposure below approximately 100 mSv
- Whether there are any effects at low dose other than cancer and leukemia
- Whether there are any inherited effects at any exposure.

Some practical considerations for understanding ionizing-radiation exposure:

- "Normal" cancer incidence is very high (30%–40% over the population)
    - Cause-effect relationships cannot be proven on an individual basis; relative risk can only be established for large exposed groups
- Stochastic effects have a long latent period
    - Leukemia: 2–7 years post-exposure
    - Other cancers: 10–50 years post-exposure
- Exposure must occur to tissue for radiation-induced cancer to form
    - Cancer will not initiate in an organ of the body unless that organ has received a dose
- Certain cancers are not associated with low to moderate doses of ionizing radiation
    - Hodgkin's disease

- Non-Hodgkin's disease
- Chronic lymphocytic leukemia
- Cutaneous malignant melanoma
- Uterine cancer
- Prostate cancer

- Risk from radiation exposure to child and conceptus are enhanced
  - Rapidly dividing cells are radiosensitive
  - Longer lifespan post-exposure enables latent cancers to manifest.

The risks of ionizing radiation exposure have been quantified using a variety of dose-response models. A commonly used model is the linear no-threshold (LNT) model, which is generally considered to be conservative and as such is used extensively by regulators. There are numerous other models, including hormesis models, which are actively debated in the scientific literature and about which scientists have shown evidence for and against. Dose-protection guidance, the LNT approximation, and how the "as low as reasonably achievable, social and economic factors taken into account" (ALARA) mantra is used are discussed in Section 3.

3 Dosimetry, Dose Limitations, and Guidance

The underlying principle of any process that uses ionizing radiation is that the benefit from potential exposure outweighs the potential risk from exposure. Radiation protection is guided by regulations that eliminate deterministic effects and limit stochastic effects from exposure. The relationship between scientific discovery, international synthesis of data, and adoption of national regulations is illustrated in Figure 16.

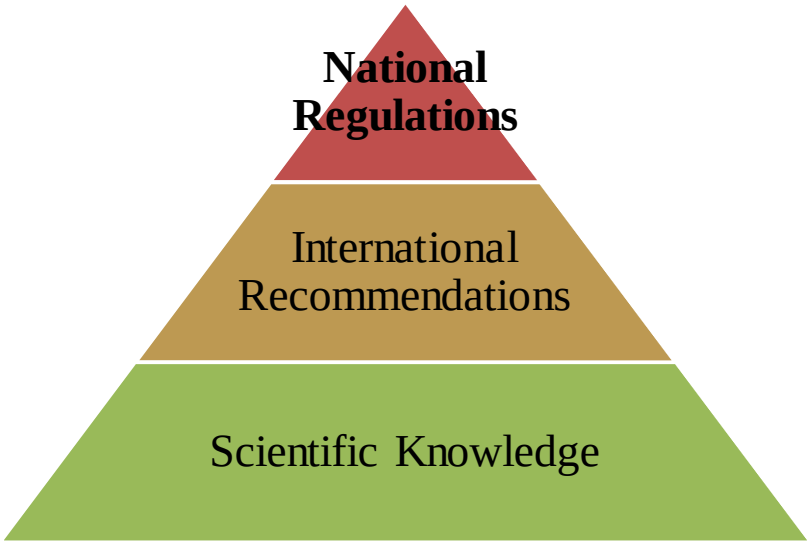

Figure 16 Radiation protection regulation pyramid

Scientific knowledge and discovery is the broad base from which we derive fundamental radiation-protection quantities. This knowledge is disseminated through national reports, industrial reports, peer-review publications, conferences, and many other scientific venues. International bodies periodically synthesize the available data from the scientific literature and produce recommendations on various aspects of radiation protection. International recommendations form the basis of national standards, but are not themselves regulations. National regulations are adopted by government bodies country by country, and although these bodies

tend to follow international recommendations, they are not obligated to do so. In Canada, there are various governmental agencies involved in establishing radiation-protection guidelines, with different areas of responsibility depending upon the source of radiation. For example:

- Nuclear reactor operations – Canadian Nuclear Safety Commission (CNSC)
- Radioactive sources and waste – Canadian Nuclear Safety Commission (CNSC)
- Uranium mining and milling operations – Canadian Nuclear Safety Commission (CNSC)
- High-energy particle accelerators – Canadian Nuclear Safety Commission (CNSC)
- X-ray regulations – each province regulates independently
    - X-ray use – Health Canada (HC) guidelines
- Naturally Occurring Radioactive Material (NORM) - each province regulates independently
    - NORM mitigation – Health Canada (HC) guidelines.

Other agencies that have input into radiation protection regulations include Environment Canada (EC) and Transport Canada (TC). A number of national agencies in Canada are involved in safety and security of radioactive material, including (i) all the above-named agencies, (ii) Public Safety Canada, (iii) Canadian Security Intelligence Service (CSIS), (iv) Department of National Defence, and (v) Canada Border Services Agency (CBSA).

This section discusses fundamental definitions related to ionizing-radiation dose, background radiation, and national/international guidance regarding dose and dose limitations. Background information on the historical development of the radiation protection-system can be found in [Inkret1995] [Lindell1996] [Jones2005] [Clarke2005] [Walker2000].

3.1 Dose Definitions

The concepts of absorbed dose, equivalent dose (radiation weighting factors), and effective dose (tissue weighting factors) are discussed here in relation to risk concepts. In addition, radon guidance will be introduced.

3.1.1 Exposure

Before discussing dose, it is worthwhile to discuss, in a historical sense, exposure. In the early days of X-rays, a common dosimeter was a piece of dental film attached to a paper clip. The daily allowable exposure was an exposure that was just enough so that some "fogging" of the film could be observed on processing. This was known as a "paper-clip unit" of radiation exposure and amounted to an early dosimeter. For larger doses such as might be used in therapeutic medicine, a "skin erythema unit" was used, which was an exposure that would just cause visible reddening of the skin. Neither unit was what could be considered biologically meaningful, although they did provide a measure of protection.

Exposure was defined for X- (and gamma) radiation in terms of air ionization, and the original unit of air ionization established in 1928 was the Roentgen (*R*). The current definition is given as:

$$1R \equiv 2.58 \times 10^{-4}\, C\, kg^{-1}\, of\ air\ .$$

Absorbed dose was subsequently defined as the energy absorbed per unit mass from any kind of ionizing radiation in any target. As such, the absorbed dose is a physical quantity. To calcu-

late the absorbed dose in air from a given exposure, the ionization potential of air (which is 33.7 eV/ion pair or 33.7 J/C) can be used. Using this value, the absorbed dose in air from an exposure of 1 Roentgen can be calculated as:

$$1R \equiv \left(2.58\times10^{-4}\,\frac{C}{kg} \right)\left(33.7\,\frac{J}{C} \right) = 8.7\times10^{-3}\,\frac{J}{kg}\,.$$

A similar calculation of the soft-tissue ionization potential yields $9.5\text{x}10^{-3}$ J kg^{-1}.

Because energy deposition per unit mass was deemed to be close to actual biological damage, a new unit was developed, called the Rad (standing for Radiation Absorbed Dose). The Rad was defined as an energy deposition of 100 erg per gram of material, where 10^7 erg = 1 Joule. The Roentgen could therefore be defined as 87 erg/g in air (or 95 erg/g in soft tissue), and the Rad was defined as 100 erg/g. For regulatory purposes, it was therefore assumed that if one could measure the ionization of air in Roentgen (R), this would be approximately equivalent to the absorbed dose in tissue (Rad_{tissue}). To convert exposure to absorbed dose for any medium, Eq. (14) can be used:

$$D\left(Rad\right) = \frac{87}{100} \times \frac{\dfrac{\mu_m}{\rho_m}}{\dfrac{\mu_a}{\rho_a}} \times E(Roentgen)\ , \qquad (14)$$

where

D = absorbed dose in Rad
E = exposure in Roentgen
μ_m = energy-absorption coefficient for tissue *(or any medium)*
μ_a = energy-absorption coefficient for air
ρ_m = tissue density (or density of any medium)
ρ_a = air density.

Although the historical unit of Rad is still used in some instances, SI units are more common internationally and are promulgated through all major international guidance bodies such as IAEA, ICRP, and UNSCEAR.

3.1.2 Absorbed, equivalent, and effective dose

Three fundamental quantities are required when considering radiation protection:

- Absorbed dose (*D*) is energy absorbed per unit mass. 1 gray (Gy) = 1 joule per kilogram (J/kg)
- Equivalent dose (H_T, units of Sievert, Sv) takes into account the relative biological effectiveness of different radiation types
- Effective dose (*E*, units of Sievert, Sv) takes into account the potential for detrimental effects to the various organs and tissues.

It is important to realize that absorbed dose is a physical quantity, whereas equivalent and effective dose are derived quantities used for radiological-protection purposes. In addition, these derived quantities use reference "individuals" and assigned factors, roughly approximate risk on a population basis, and are not useful for accurate estimation of risk on an individual basis.

The absorbed dose can be expressed as in Eq. (15):

$$D_{T,r} = \frac{E_T}{m_T}, \tag{15}$$

where

$D_{T,r}$ is the absorbed dose to target *T* from radiation *r* (J/kg, or Gy)
E_T is the energy deposited in target *T* (Joule)
m_T is the mass of target *T* (kg).

The equivalent dose is calculated by summing over all different radiations interacting in the target material using Eq. (16):

$$H_T = \sum w_r D_{T,r}, \tag{16}$$

where

H_T is the equivalent dose to target *T* (Sv)
w_r is the radiation weighting factor for the radiation type.

Radiation weighting factors are related to the relative biological effectiveness (RBE) of different kinds of radiation, which is an indication of the relative amount of radiation damage done to tissue for a given absorbed dose. The recommended radiation weighting factors from ICRP103 are provided in Table 7.

Table 7 Radiation weighting factors [ICRP2007]

Radiation	Radiation weighting factor, w_r
Photons (X- and gamma)	1
Electrons (incl. beta)	1
Protons	2
Alpha particles and fission fragments	20
Neutrons (continuous function of energy) N.B.: in ICRP60 [ICRP1990], the centre expression represented all w_r)	$2.5+18.2e^{\frac{-(\ln E)^2}{6}} \quad E<1MeV$ $5.0+17.0e^{\frac{-(\ln(2E))^2}{6}} \quad 1MeV \le E \le 50\,MeV$ $2.5+3.25e^{\frac{-(\ln(0.04E))^2}{6}} \quad E>50\,MeV$

The effective dose, *E*, is calculated using the equivalent dose and tissue weighting factors using Eq. (17):

$$E = \sum w_T H_T \ , \tag{17}$$

where w_T is the tissue weighting factor for target *T*.

Tissue weighting factors have been established as indicators of the relative sensitivities of different tissues to radiation exposure. The ICRP103 tissue weighting factors, along with ICRP60 and ICRP26, are given in Table 8. Blank entries mean that the organ was not uniquely identified in the guide and was included in “remainder”. The sum of tissue weighting factors over all tissues is unity.

Table 8 Tissue weighting factors [ICRP1977,1990,2007]

Organ	Tissue weighting factor, w_T		
	ICRP26 (1977)	ICRP60 (1990)	ICRP103 (2007)
Gonads	0.25	0.20	0.08
Red Bone Marrow	0.12	0.12	0.12
Lung	0.12	0.12	0.12
Breast	0.15	0.05	0.12
Thyroid	0.03	0.05	0.04
Bone surface	0.03	0.01	0.01
Remainder	0.30	0.05	0.12
Colon		0.12	0.12
Stomach		0.12	0.12
Bladder		0.05	0.04
Liver		0.05	0.04
Esophagus		0.05	0.04
Skin		0.01	0.01
Salivary glands			0.01
Brain			0.01
Sum	1.00	1.00	1.00

Unfortunately, ICRP adopted the same base unit (Sv) for both equivalent dose and effective dose, which has the potential to lead to confusion in dose representation.

The equivalencies between classical and SI dose units (and activity, which was discussed in Chapter 3) are provided in Table 9. Dose rate units are the same base units as a function of time (for example, Gy/h).

Table 9 Dose units

Unit	Classical unit	SI unit	Unit conversion
Activity	Curie (Ci)	Becquerel (Bq)	1 Ci = 3.7×10^{10} Bq
Absorbed dose	Radiation absorbed dose (Rad)	Gray (Gy)	100 Rad = 1 Gy
Equivalent dose	Roentgen equivalent man (Rem)	Sievert (Sv)	100 Rem = 1 Sv
Effective dose	Roentgen equivalent man (Rem)	Sievert (Sv)	100 Rem = 1 Sv

In Canada, it is not uncommon to find references to classical units used as operational quantities in nuclear power plants. However, SI units are the approved units of radiation dosage according to the CNSC regulations.

As was discussed in Section 2.3.2 and demonstrated in Table 3, after exposure to low-dose-rate radiation, the ICRP [ICRP2007] suggests detriment-adjusted nominal risk coefficients for cancer and heritable effects in terms of risk per Sv (% per Sv).

The relationship between the various dose quantities and endpoint estimates is depicted in Figure 17. The relative uncertainty in physical and derived quantities is shown by increasing arrow thicknesses, with the uncertainty compounding with each step to calculate the endpoint risk. This demonstrates that, although absorbed-dose estimates may be made with relatively good confidence, the ability to estimate population risk from low dose / dose rate exposure is highly uncertain.

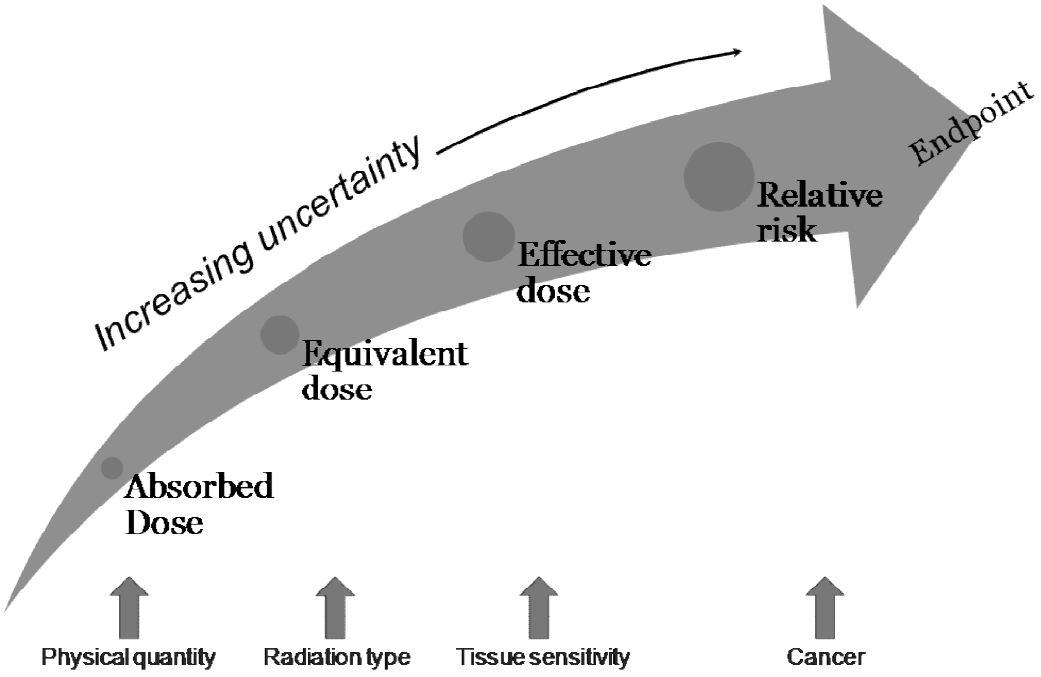

Figure 17 Relationship between dose and endpoint

3.1.3 Committed dose

The committed effective dose (CED) is the effective dose that an individual will eventually receive after having taken a radionuclide into the body (inhalation, ingestion, injection, or skin/wound absorption). Committed usually means to 50 years, although 70 years is used when a childhood intake has occurred.

3.1.4 Collective dose

The collective dose is the sum of individual doses over a population, with units of person-Sv. If the collective dose is divided by the number of persons in the exposed population, then an average individual dose can be estimated. Although this quantity has been used in the literature, ICRP 103 does not recommend the use of collective dose.

3.1.5 Derived quantities

Derived quantities are values or results that are made up, or derived, from other values. Often, derived quantities are made up of other fundamental quantities. Four primary derived quantities are discussed in this section: (i) dose-conversion factors (DCF), (ii) annual limits on intake (ALI), (iii) derived air concentration (DAC), and (iv) radon progeny (WL; WLM).

3.1.5.1 Dose-conversion factors (DCF)

Dose-conversion factors (DCF) are tabulated derived values that enable conversion from one quantity to another. For example, dose-conversion factors are tabulated for inhalation and ingestion for workers in ICRP68 [ICRP1994] and the public in ICRP72 [ICRP1996a]. A dose-conversion factor for an internalized radionuclide typically has units of Sv/Bq committed effective dose (CED), and therefore if the intake (in Bq) is known, the estimated CED in Sv can be immediately calculated. In the case of internalized radionuclides, values are available for inhalation and ingestion, for differing particle sizes, and for the different chemical forms of many radionuclides. Examples of tabulated dose conversion factors for ^{14}C are shown in Table 10.

Table 10 Sample inhalation dose coefficients [ORNL2013]

Inhalation dose coefficients (Sv / Bq) from ICRP 68				
Chemical form=	CO_2	CO	CH_4	Vapour
Effective (ICRP 60)	6.5E-12	8E-13	2.9E-12	5.8E-10

These conversion factors are derived using compartmental biokinetic models for internal dosimetry. Other dose-conversion factors can include external DCFs such as those found in ICRP74 [ICRP1996b] and discussed in Section 5.

3.1.5.2 Annual limit on intake (ALI)

The annual limit on intake (ALI) is the activity (Bq) of a radionuclide which, if taken internally, would result in a dose equal to the annual limit. As such, the ALI is normally defined as the dose limit divided by the effective dose per unit intake. Both equivalent and effective doses are considered; the more restrictive limit is used. Note that simply multiplying the number of ALIs taken in by 20 mSv to calculate committed effective dose may not be a correct estimate because some ALIs are based on effective dose and some on equivalent dose. The ALI is calculated using Eq. (18):

$$ALI\left(Bq\right)=\frac{D_{\text{limit}}\left(Sv\right)}{\sum w_T H_T\left(\frac{Sv}{Bq}\right)}=\frac{0.02\ Sv}{DCF\ \frac{Sv}{Bq}}. \tag{18}$$

3.1.5.3 Derived air concentration (DAC)

The derived air concentration (DAC) is defined as the air concentration (Bq/m^3) that would lead to the inhalation of 1 ALI by a reference person over 2000 hours (50 weeks x 40 h/week)

assuming that the reference person inhales 1.2 m^3/h (light work). The DAC is calculated using Eq. (19):

$$DAC\left(\frac{Bq}{m^3}\right)=\frac{ALI\left(Bq\right)}{2000\,hr\cdot 1.2\,\frac{m^3}{hr}}=\frac{ALI\left(Bq\right)}{2400\,m^3}\ . \tag{19}$$

DACs enable rapid estimation of committed effective dose rate because the DAC represents 1 ALI (in the case of Canadian regulations, 20 mSv) in 2000 h, or stated in another manner, 1 DAC = 10 µSv/h. The number of DAC-h corresponds to the CED.

3.1.5.4 Radon-derived quantities (WL; WLM)

Radon gas (^{222}Rn) is considered harmful primarily because it has numerous short-lived daughters such as polonium, bismuth, and lead which are solids and readily adsorbed on particles of fine dust than can be inhaled. When air containing radon is breathed in, it can be readily expelled without inducing any damage whatsoever. However, the radon daughters, both from gas decay and dust, can be trapped in lung tissue, where they can induce local damage to lung tissue through alpha-particle interactions. The combined alpha energies of the daughters per unit volume of air are the most important parameters to be considered when estimating damage to lung tissue. There is a special unit used to define this quantity, called the working level (WL).

The working level (WL) is defined as any combination of the short-lived decay products of radon (^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po) in one dm^3 (1 litre) that will result in the ultimate emission of $1.3\text{x}10^5$ MeV of alpha energy (or $2.08\text{x}10^{-5}$ J/m^3). An atmosphere containing 3.6 Bq/dm^3 (~100 pCi/L) of radon in equilibrium with its daughters is 1 WL. Note that the short-lived daughters are not necessarily in equilibrium with the radon parent, and for this reason, conversion of radon concentration to daughter concentrations uses a factor of about 0.7 for uranium mines (0.4 for homes). For example, finding 3.6 Bq/dm^3 in a mine is approximately equivalent to 0.7 WL. It is common practice to use an equilibrium factor of 0.5, and therefore a WL would be equal to 100 pCi/L (3.7 Bq/L).

A working-level month (WLM) is defined as the exposure to the equivalent radon daughter concentration of 1 WL for a time period of 170 hours. For regulatory purposes, an abundance of data was obtained from uranium miners, which showed that miners exposed to 100 or more WLM had an increased risk of developing lung cancer. A regulatory limit of 4 WLM per year was therefore set for uranium workers. This was set to correspond to the fact that a 25-year work duration at a level of 4 WLM would lead to a cumulative exposure of 100 WLM. Dose-conversion factors for radon are presented in ICRP50 [ICRP1987]. A worker dose-conversion factor of 5 mSv/WLM is often used for radiation-protection purposes.

3.2 Background Radiation Exposure

The Earth is radioactive and has been since its formation. The Earth contains radioisotopes that originated from planetary formation (primordial radioisotopes) and also from extra-terrestrial origins (cosmogenic radioisotopes produced continuously in the upper atmosphere through cosmic-ray interactions which deposit on Earth through atmospheric dispersion, deposition,

and precipitation). Some important primordial and cosmogenic radioisotopes are listed in Table 11.

Table 11 Primordial and cosmogenic radioisotopes

Primordial			Cosmogenic	
^{40}K	^{142}Ce	^{174}Hf	^{10}Be	^{22}Na
^{50}V	^{144}Nd	^{180}Ta	^{26}Al	^{35}S
^{87}Rb	^{147}Sm	^{187}Re	^{36}Cl	^{7}Be
^{113}Cd	^{148}Sm	^{190}Pt	^{14}C	^{33}P
^{115}In	^{152}Gd	^{204}Pb	^{32}Si	^{32}P
^{123}Te	^{156}Dy	$^{238, 235}$U	^{39}Ar	^{28}Mg
^{138}La	^{176}Lu	^{232}Th	^{3}H	^{24}Na

In addition to the two natural source categories, radioisotopes are introduced into the environment from anthropogenic sources such as nuclear medicine, certain industrial processes, routine nuclear power operations, historical nuclear weapons testing, and nuclear reactor accidents. Theoretically, all radioisotopes on the nuclide chart can be introduced into the environment through anthropogenic processes. In practice, only a few anthropogenic radioisotopes are persistent in the environment, for example, ^{137}Cs. All the above-listed sources contribute to the ubiquitous background that constantly surrounds us [Waller2013].

The most recent analysis of U.S. population dose due to natural and anthropogenic radionuclides was conducted by the NCRP and published in Report No. 160 [NCRP2009]. The report indicated that roughly half the yearly exposure to a member of the U.S. population is from natural background radiation, primarily radon (originating from uranium in the ground) and thoron (originating from thorium in the ground) gases, whereas the other half is from anthropogenic (human-made) sources, of which the majority is due to medical procedures (see Figure 18). In Canada, it can be expected that the ratios will be similar to the U.S. values.

Note that in the U.S. report [NCRP2009], the average estimated yearly dose corresponding to Figure 18 is approximately 6.2 mSv. The data indicated a large increase in average yearly dose, which was exclusively due to medical procedures. Before the release of NCRP Report No. 160, the average yearly dose was estimated to be approximately 3.6 mSv from all sources [NCRP1987]. Therefore, the dose increase of 3.6 mSv to 6.2 mSv (almost 75% greater) due to medical procedures is significant and worthy of increased surveillance.

Because Canada is a country in which advanced medical procedures such as computer tomography (CT) are widely used, it is likely that our yearly average dose is proportionally similar to that shown in Figure 18.

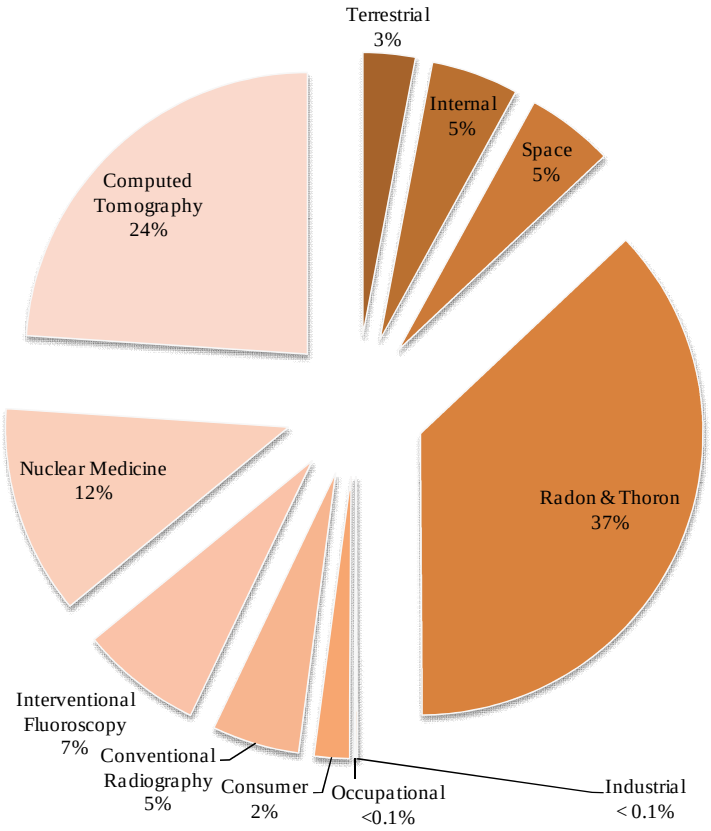

Figure 18 Origin of average yearly radiation exposure (adapted from [NCRP2009])

However, since the use of radiation for medical procedures is more strictly controlled in Canada, it is also likely that the average yearly dose is proportionally smaller that U.S. values, with a correspondingly higher percentage assigned to radon and thoron compared to, for example, CT.

A study done by [Grasty2004] for natural background radiation in Canada indicated that the average annual radon and thoron component of the dose is 926 µSv, compared to the [NCRP2009] U.S. estimate of 2294 µSv. In fact, taking into account all sources of natural (not anthropogenic) radiation, it was found [Grasty2004] that the average yearly effective dose from natural sources was 1769 µSv, which is significantly smaller than the 2422 µSv worldwide average reported by [UNSCEAR2000]. Note, however, that the natural background can vary greatly from location to location in Canada (and, of course, world-wide).

3.3 International Guidance

The history of radiation protection effectively begins in 1895 with the discovery of X-rays by Wilhelm Roentgen (for which he received the Nobel Prize in physics in 1901). Radiation-induced dermatitis was first observed by Emil Grubbé in the U.S. and Henry Drury in the U.K. in 1896. By the end of 1896, within a year of the discovery of X-rays, the first radiation-protection advice was proposed by Wolfram Fuchs in the United States and can be paraphrased as:

- Coat skin with Vaseline™ and leave extra on the most exposed area
- Do not stand within 12 inches of the X-ray tube
- Make exposure as short as possible.

As crude as the above recommendations are, they embody the underlying protective mantra of radiation shielding that we still use today: shielding, distance, and time. As a result, in 1902, a first crude limit corresponding to approximately 100 mGy per day (or 30000 mGy per year) was proposed, corresponding to the lowest radiation exposure that could be detected by fogging on a photographic plate [Inkret1995]. However, despite an acknowledgment of potential detrimental effects, between the time of discovery of X-rays and the mid-1920s, numerous injuries due to radiation exposure were reported (especially among early radiologists). Following increasing radiation-safety concerns by radiologists and the public, the first International Congress of Radiology was held in 1925 in London, England. Until that point, quantification of dose was not consistent and therefore was considered a priority. As a result, the International Commission on Radiation Units and Measurements (ICRU) was formed. In 1928, the second Congress was held in Stockholm, Sweden, which led to the formation of the International X-Ray and Radium Protection Committee, which was the predecessor of the International Commission on Radiological Protection (ICRP). Throughout the years, the trend in radiation protection has involved a lowering of the acceptable radiation dose for workers, and by the 1960s, of limitations on acceptable dose for members of the public. The trend in dose standards is presented in Figure 19.

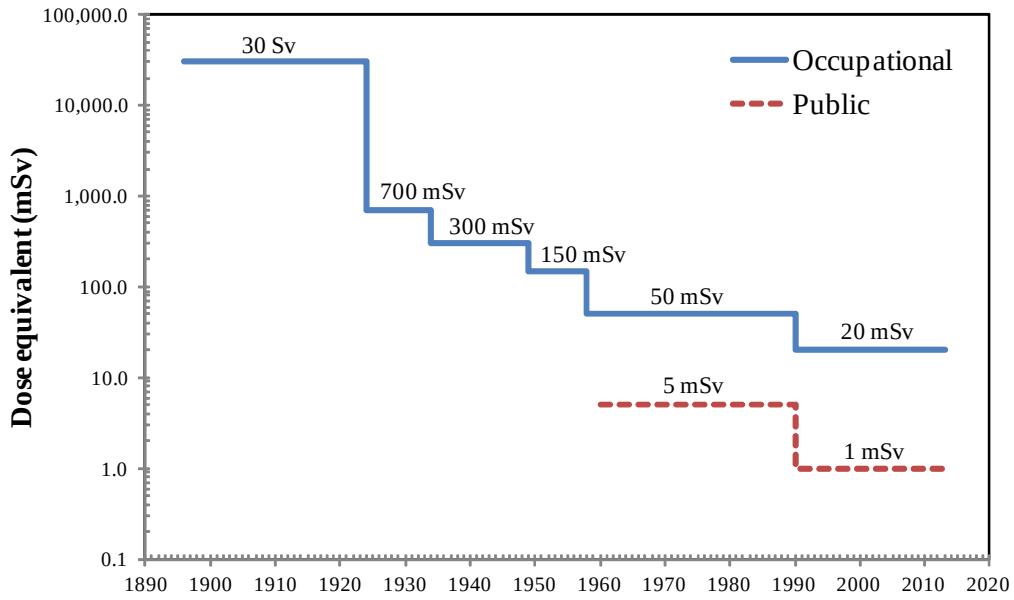

Figure 19 Chronology of radiation-protection guidance (adapted from [Inkret1995])

The change in philosophy for radiation-protection standards has shifted since 1896 as more knowledge about the detrimental effects became known and as more opportunities for population studies of effects (epidemiological studies) became available. A prime example is the paradigm shift from protecting only for deterministic effects (Figure 20, left) to acknowledgement, quantification, and protection standards to minimize risk from stochastic effects (Figure 20, right).

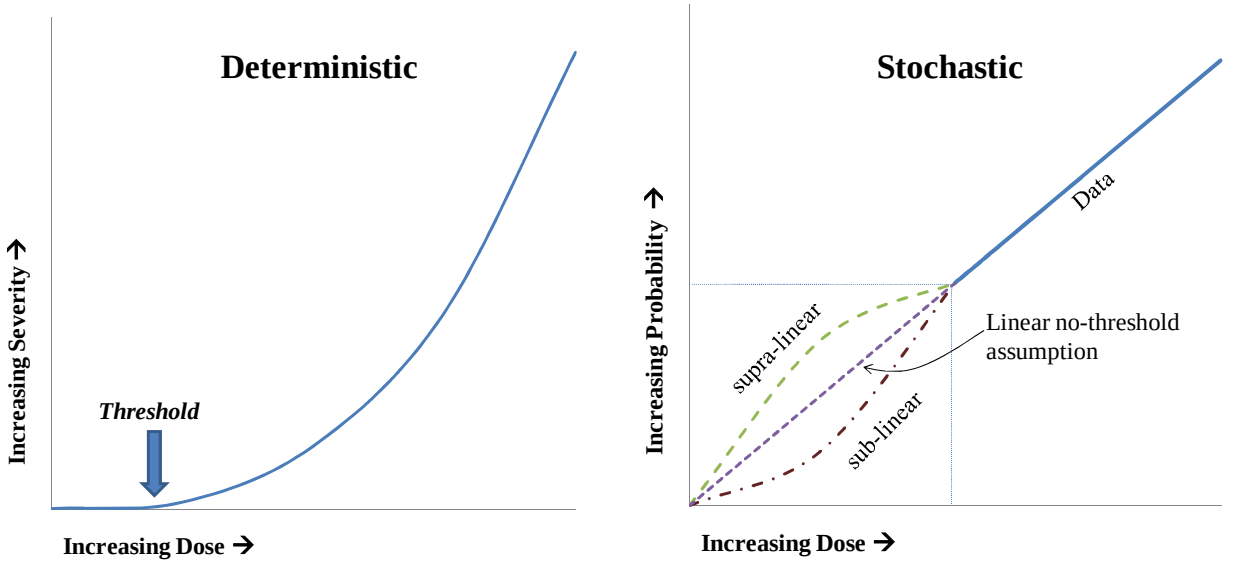

Figure 20 Paradigm shift from deterministic to stochastic effect protection

The paradigm shift in basic radiation-protection philosophy lead to numerous fundamental changes in the way radiation protection was being administered on an international scale. Although the original uses of ionizing radiation were predominantly in medicine, increased non-medical uses became more common throughout the years and were reflected in adaptations of international guidance. Practical guidance shifted to quantified dose limitation and optimization. Although it had always been assumed that if humans were adequately protected from the deleterious effects of ionizing radiation exposure, then the environment was also protected, a change in philosophy moved towards the need for demonstration of adequate environmental protection. Although the concept of the greatest good for the greatest number dominated radiation-protection practice for many years, current philosophy also considers adequate individual protection as well as that of the group. The historical progression of radiation-protection principles is depicted in Figure 21.

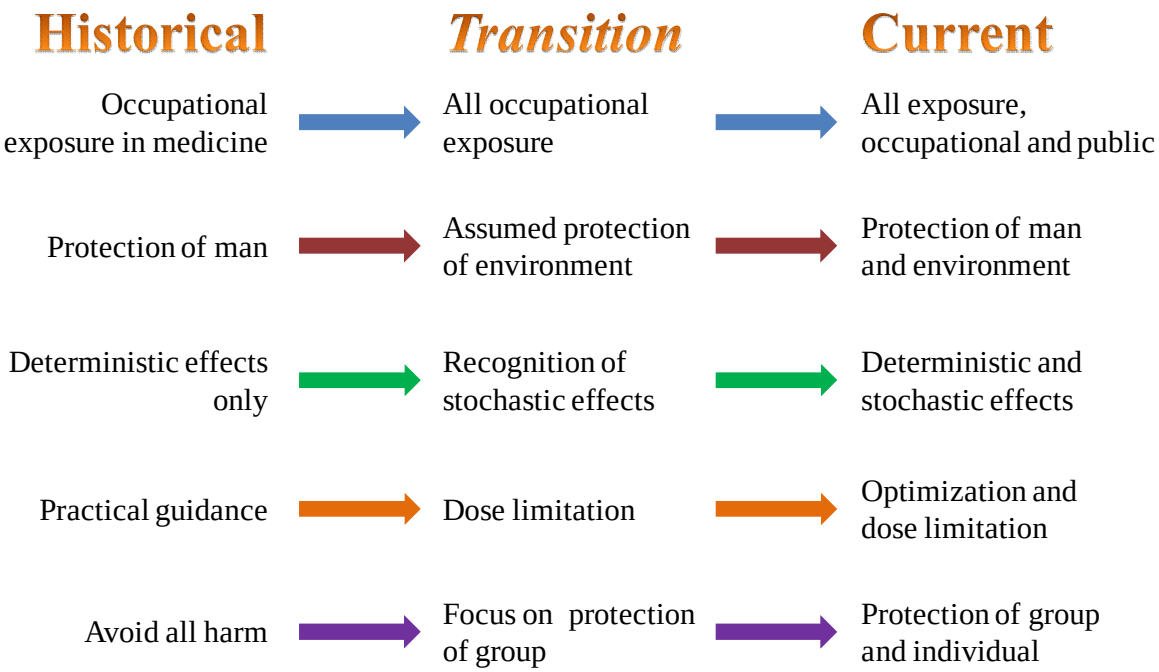

Figure 21 Progression of radiation protection principles (adapted from [Clement2009])

A number of international organizations are pivotal in setting radiological protection standards and providing guidance. The essential organizations that provide input to national regulatory standards are depicted in Figure 22. These organizations are discussed in the following sections.

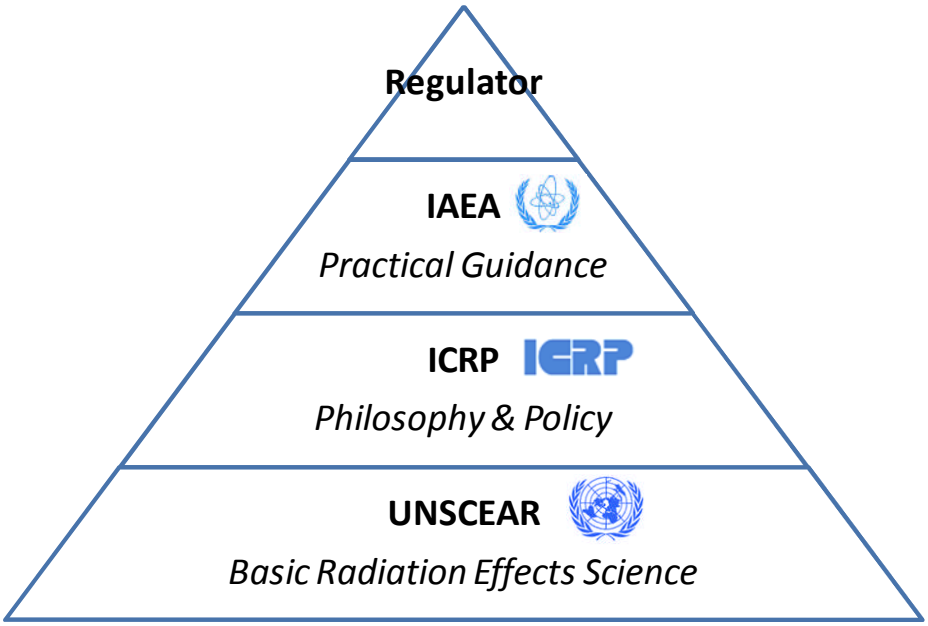

Figure 22 Relationship of international bodies for radiation protection regulations

3.3.1 UNSCEAR

UNSCEAR is the United Nations Scientific Committee on the Effects of Atomic Radiation, which was established by the United Nations in 1955 to assess and report levels and effects of exposure to ionizing radiation and provides the scientific basis for evaluating radiation risk and for

establishing protective measures. UNSCEAR provides the scientific guidance that is consulted when developing ICRP recommendations. UNSCEAR publications can be found online at:

http://www.unscear.org/unscear/en/publications.html .

3.3.2 ICRP

ICRP is the International Commission on Radiological Protection, established in 1928 to advance for the public benefit the science of radiological protection, in particular by providing recommendations and guidance on all aspects of protection against ionizing radiation. The main objective of the Commission's recommendations is to provide an appropriate standard of protection for humans without unduly limiting the beneficial practices giving rise to radiation exposure. The ICRP produces recommendations on radiological protection that are adopted worldwide based on science and value judgements. The ICRP guidance documents form the basis for IAEA regulatory recommendations. ICRP publications can be found online at: http://www.icrp.org/publications.asp.

3.3.3 ICRP 103

The current recommendations of the ICRP [ICRP2007] form the basis for the radiation-protection system. The primary aim of the recommendations is to provide an appropriate level of protection for people and the environment without unduly limiting the desirable human activities that may be associated with radiation exposure. The three primary goals of radiation protection are:

- Justification – more benefit than detriment from exposure
- Optimization – exposure has been optimized to be as low as reasonably achievable
- Dose limitation – limit the dose (except for medical exposures).

The three goals form the core of the radiation-protection system and apply to three different situations:

- Existing situations – natural events, past practices, past events; the situation already exists when a decision on control is required, including natural background and residues from past practices.
- Planned situations – practices, medical exposures; the situation involves planned operation of sources, including decommissioning and disposal of radioactive waste.
- Emergency situations – preparedness, response; the situation is unexpected and requires urgent action to mitigate consequences.

In ICRP103, three exposure types are also considered:

- Occupational exposure - all exposure incurred by workers in the course of their work, with the exception of (i) excluded exposures and exposures from exempt activities involving radiation or exempt sources, (ii) medical exposure, and (iii) normal local natural background radiation.
- Public exposure - exposure incurred by members of the public from radiation sources, excluding any occupational or medical exposure and the normal local natural background radiation.
- Medical exposure - exposure incurred by patients as part of their own medical or dental diagnosis or treatment; by persons, other than those occupationally exposed, knowingly,

while voluntarily helping in the support and comfort of patients; and by volunteers in a programme of biomedical research involving their exposure.

Dose limits corresponding to planned public and occupational exposure are provided in Table 12. Note that the numerical values are essentially the same as reported in ICRP60 [ICRP1990], from which the current CNSC regulations are derived. Note that the dose-limit guidance refers to exposures above background radiation (as discussed in Section 3.2) and as such may be considered as an additional dose over background. It should also be noted that, operationally, action levels are often used that are a fraction of the dose limit. Exceeding an action level would initiate an investigation into the reason for the dose and may initiate a review of procedures to make doses as low as reasonably achievable, social and economic factors being taken into account (ALARA). The ALARA principle is also found in CNSC document G-129, Rev1 [CNSC2004].

Table 12 ICRP103 dose limits for planned exposure scenarios [ICRP2007]

	Occupational	Public
Effective Dose (Whole Body)	20 mSv/a averaged over 5 years (50 mSv/a max)	1 mSv/a
Equivalent Dose (Lens of the Eye)	150 mSv/a	15 mSv/a
Equivalent Dose (Skin)	500 mSv/a	50 mSv/a
Equivalent Dose (Hands and Feet)	500 mSv/a	n/a

ICRP103 also considers the target for exposure in defining a “representative person”. A representative person is an individual receiving a dose that is representative of the more highly exposed individuals in the population. This term is the equivalent of, and replaces the notion of, the “average member of the critical group” described in prior ICRP documents (for example, [ICRP1990]).

Finally, ICRP103 considers explicitly the need for protection of the environment by delineating the requirement for scientific evidence to demonstrate that protection is adequately afforded and the need for improved protection as required. Further guidance is provided by ICRP108 [ICRP2008].

3.3.4 IAEA

The IAEA is the International Atomic Energy Agency, established within the United Nations framework in 1956 as the world’s “Atoms for Peace” organization to promote safe, secure, and peaceful nuclear technologies. The Agency works with its member states and multiple partners world-wide to promote safe, secure, and peaceful nuclear technologies. Three main areas of work underpin the IAEA’s mission: Safety and Security; Science and Technology; and Safeguards and Verification. The IAEA has a wide range of programmes, including development of safety stan-

dards in regulatory language. IAEA Safety Series documents are often used as the principal references for national regulatory policy. The most fundamental IAEA radiation protection standard was the "International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources" [IAEA1996], which has been superseded by an interim General Safety Requirements Part 3 report [IAEA2011]. IAEA publications can be found online at: http://www-pub.iaea.org/books/.

3.4 Canadian Guidance

The regulatory body for nuclear power activities in Canada is the Canadian Nuclear Safety Commission (CNSC). The regulatory system is designed to protect people and the environment from licensed sources of anthropogenic radiation resulting from the use of nuclear energy and materials. This is accomplished through a licensing process that requires the licensee to prove that their operations are safe. The basis of the regulatory system is the principle that no technology is fail-proof and that therefore licensees must incorporate multiple layers of protection (defence in depth) whenever radioactive materials are used. The CNSC also licenses the import, export, and transportation of nuclear materials and other prescribed substances, equipment, technology, and dual-use items. CNSC staff evaluates the performance of nuclear power plants and radioactive material users, as well as participating in international activities for non-proliferation of nuclear weapons.

The CNSC operates and enforces regulations under the Nuclear Safety and Control Act (NSCA, 1997). As the Canadian federal regulator, the CNSC executes licensing decisions made by the Commission or its designates continually monitors licensees to ensure that they comply with safety requirements that protect workers, the public, and the environment, and also upholds Canada's international commitments to the peaceful use of nuclear energy. Regulatory requirements are codified in the NSCA, its associated regulations, licences, and directives provided by the CNSC. A number of regulations are referred to under the act and are listed below:

- Canadian Nuclear Safety Commission By-Laws (SOR/2000-212)
- Canadian Nuclear Safety Commission Cost Recovery Fees Regulations (SOR/2003-212)
- Canadian Nuclear Safety Commission Rules of Procedure (SOR/2000-211)
- Class I Nuclear Facilities Regulations (SOR/2000-204)
- Class II Nuclear Facilities and Prescribed Equipment Regulations (SOR/2000-205)
- Directive to the Canadian Nuclear Safety Commission Regarding the Health of Canadians (SOR/2007-282)
- General Nuclear Safety and Control Regulations (SOR/2000-202)
- Nuclear Non-Proliferation Import and Export Control Regulations (SOR/2000-210)
- Nuclear Security Regulations (SOR/2000-209)
- Nuclear Substances and Radiation Devices Regulations (SOR/2000-207)
- Packaging and Transport of Nuclear Substances Regulations (SOR/2000-208)
- Radiation Protection Regulations (SOR/2000-203)
- Uranium Mines and Mills Regulations (SOR/2000-206)

Of significant interest are the Radiation Protection Regulations [NSCA2000], which set limits on the amount of radiation that the public and nuclear energy workers (NEWs) may receive. Canadian regulations are consistent with the most recent recommendations of the International Commission on Radiological Protection (currently adhering to guidance in ICRP60

[ICRP1990]). In Canada, standards and practices to protect people from radiation exposure are also developed by the Federal-Provincial-Territorial Radiation Protection Committee (FPTRPC), which provides a national forum on radiation-protection issues.

The CNSC offers instruction, assistance, and information on these requirements in the form of regulatory documents, such as policies, standards, guides, and notices. Licensee compliance is verified through inspections and reports.

In Canada, the current (2013) radiation-protection regulations are based on ICRP60 [ICRP1990]. However, the numerical values of the dose limits have not been changed in ICRP103 [ICRP2007], and therefore the basic values on which Canadian regulatory guidance is based will continue for many years into the future. The operational values used by the CNSC are provided in Table 12 with the inclusion of a balance-of-pregnancy dose limit for workers of 4 mSv. The effective dose is calculated using Eq. (20) and compared to the dose limit for regulatory purposes [NSCA2000]:

$$E_{total} = E + \left(5\frac{mSv}{WLM} \cdot RnP \right) + \left(20\,mSv \sum \frac{I}{ALI} \right), \tag{20}$$

where

E_{total} is the total effective dose from all sources (mSv)
E is the effective dose from external sources + committed effective dose measured directly or from excreta (mSv)
RnP is the exposure to radon progeny in working-level months (WLM)
I is the intake of radionuclides not already accounted for (Bq)
ALI is the annual limit on intake (Bq to give 20 mSv in 50 years).

3.5 Summary

Radiation-protection standards have evolved over the years to provide increased levels of protection for workers and the public. Dose limits are established on the basis of scientific knowledge which forms policy, best practices and guidance, and then national regulations.

The basic dose quantities used are absorbed dose, equivalent dose, and effective dose. Although nuclear power plant operations often use obsolete units (such as Ci, Rad, and Rem) in their plants, most of the nuclear industry uses SI units. Dose quantities for regulatory purposes, as promulgated through the Nuclear Safety and Control Act, use SI units. Dose limits derived from International Commission on Radiological Protection guidance are established in Canada, and derived quantities are used in operations.

Background radiation is made up of natural and anthropogenic components. Dose limits are established to provide protection above the ubiquitous background level, which has a wide annual dose range world-wide. In Canada, the background dose is estimated to be approximately 3.6 mSv per year. Evidence from U.S. estimates indicates that increased use of advanced medical imaging procedures such as computed tomography is resulting in a larger estimated background dose; however, no such study has indicated that this is the case in Canada.

As research continues into large epidemiological studies and low-dose rate radiation effects, dose-limit guidance may change in the future.

4 Radiation Instrumentation

Detection of ionizing radiation is vitally important to all aspects of the nuclear energy industry. The ability to identify sources of radiation, quantities of radiation, and specific radioisotopes enables the administration of comprehensive radiation protection, environmental monitoring, and security programs.

Before discussing the mechanisms of how radiation interacts with detection materials, the basics of radiation interactions with matter must be understood. These concepts were discussed in Chapter 3 (Nuclear Processes and Neutron Physics). This section considers radiation instrumentation that may be pertinent to CANDU operations. The basics of detection, as well as gas-filled, scintillation, and semiconductor detectors, are discussed. Dosimetry detectors, especially as they relate to CANDU plant operations, are explored.

4.1 Basics of Detection

Ionizing radiation is tasteless, odourless, and colourless and cannot be detected by the human senses. Therefore, devices to aid in detection and identification of ionizing radiation are required for safety purposes. Anything that responds to ionizing-radiation interactions can be considered a radiation detector. In fact, the mechanisms of radiation interactions with tissue and the human body, as discussed in Section 2, are fairly closely related to radiation detectors. The primary differences are in the observable endpoints, where radiation detectors are designed to provide an indication of radiation level through chemical/electric/electronic indications, whereas the endpoints in radiation biology tend to be deterministic or stochastic health effects. That being said, the human body itself is a radiation detector of sorts. The first "radiation dosimeters" used were based on skin reddening (or erythema). The variability in individual human response to radiation (and of course the fact that skin reddening occurs only at unsafe moderate to high radiation doses) makes this type of radiation detector impractical.

Radiation detectors work on one of two basic principles: (i) ionization and (ii) excitation.

Ionization

In an ionization-based detector, electrons bound to the atoms or molecules of the material through which charged particles such as alpha and beta particles pass are released. The separated electrons and ions (Figure 23) can be collected at two electrodes by imposing a potential difference across the detector space; their presence is then measured as pulses or a current. An example of generation of electron-ion pairs in a gas-filled counter (for example, a Geiger-Müller tube) is depicted in Figure 24.

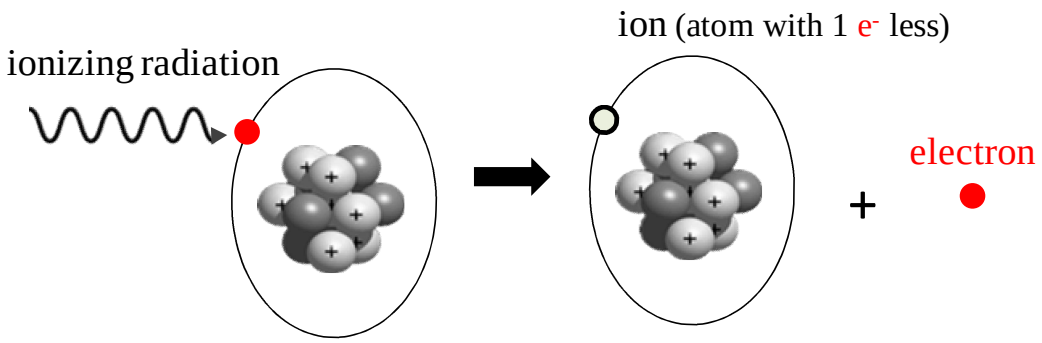

Figure 23 Ionization process

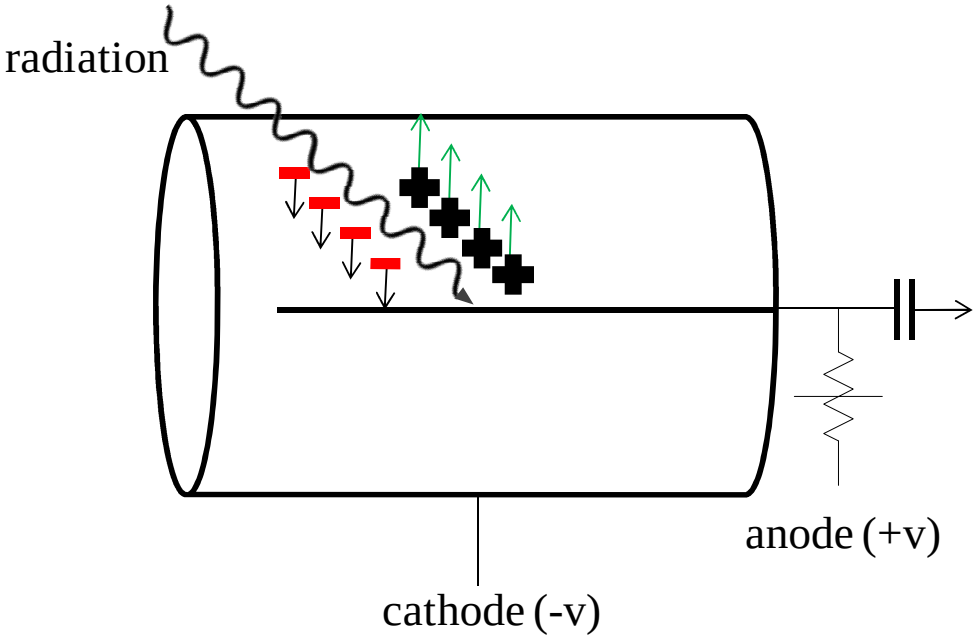

Figure 24 Ionization in a gas-filled tube

Excitation

In an excitation-based detector, part of the radiation energy is transferred to bound electrons and raises them to an excited state in the atom or molecule. When the excited species returns to its ground-state energy level, the excited atom or molecule may emit electromagnetic energy in the ultraviolet to visible region (Figure 25). This light can be detected by a photomultiplier tube (PMT), which generates secondary electrons using a photocathode and multiplies them to generate a detectable signal. This process is depicted in the scintillation-type counter system shown in Figure 26.

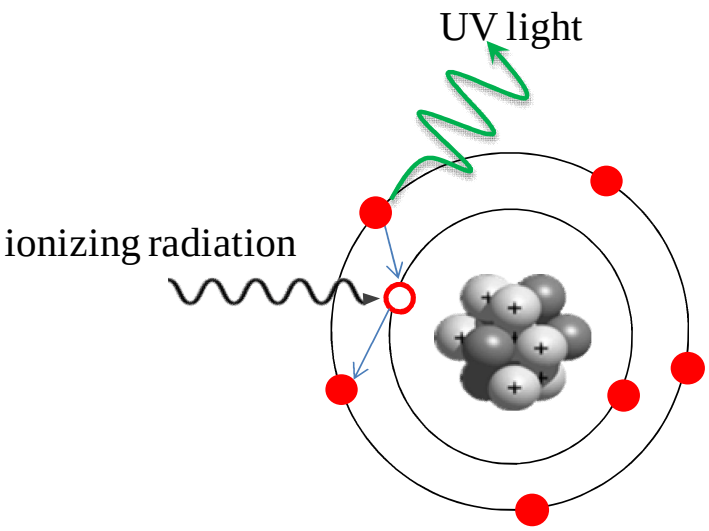

Figure 25 Excitation process

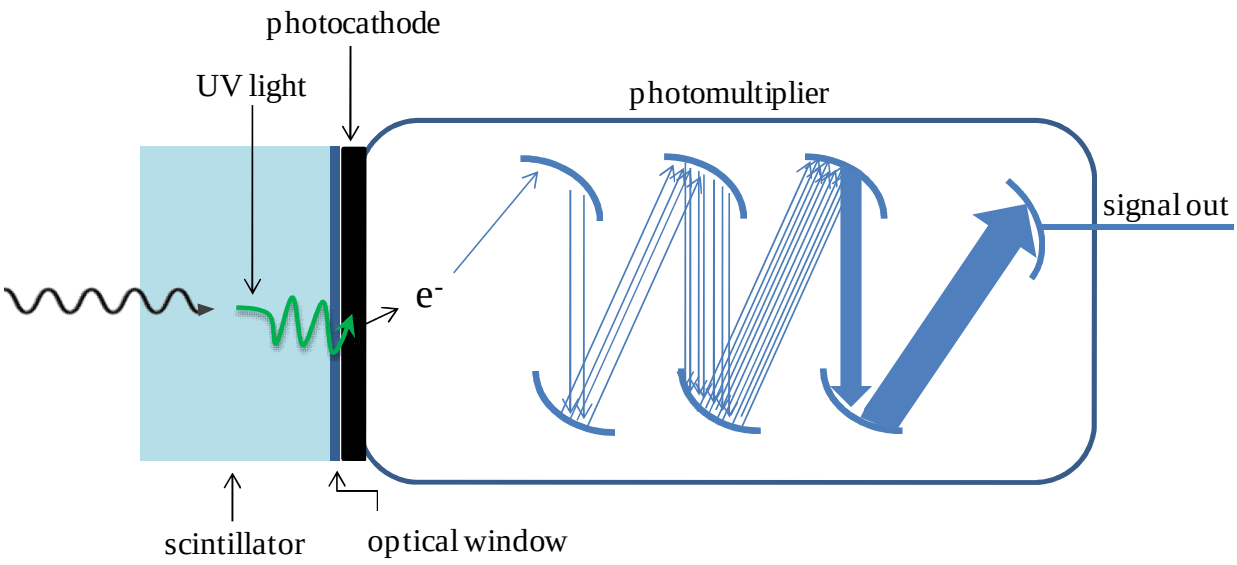

Figure 26 Excitation in a scintillator-type detector

Advances in radiation-detection instrumentation took place in the 1920s and have continued ever since. A concise discussion of the history of radiation-detection instrumentation is provided by [Frame2005]. Important operational characteristics for all detector types include resolving time and efficiency and are discussed below. An excellent reference on particle counting is ICRU Report 52 [ICRU1994].

4.1.1 Resolving time

When ionizing radiation interacts with a detector, a signal in the detector material (whether it be a gas, scintillator, or solid-state device) is produced that is related to the radiation event that generated it. During the time it takes the system to process a pulse that is generated in the detector, the detector may be unavailable to process additional events. For example, during the time it takes for positive ions in a GM tube to reach the cathode, the tube is insensitive to any radiation. During this time, if a second ionizing ray interacts with the detector, it will not be observed because the tube cannot distinguish that there is another electron avalanche present; the system observes only one large electron avalanche until it has been reset after detection. In essence, the counter cannot produce pulses for more than one event because the counter is "occupied" with the first event. This phenomenon is sometimes called coincidence, and as a result, the observed counts are always lower than the true counts. The "resolving time" is often referred to as "dead time", because in essence the detector is "dead" and cannot detect any other radiation in this time window. Note, however, that the resolving time is actually the sum of the dead time and recovery time. In terms of definitions, dead time is the time required in a detector before another unique pulse can form; recovery time is the time required after the dead time until the pulse size is large enough to pass a discriminator; and resolving time is the minimum time from the detection of one interaction until the next detection can occur. True resolving times span a range from a few microseconds to 1000 microseconds, depending on the detector. The loss of particles is important, especially when high count rates are involved and the losses accumulate into large numbers.

For a counting rate, R, from a radioactive source, the presence of coincidence will mean that the rate actually measured, r, will be less than the expected value (r < R). If the detector has a dead time of T, then the true count rate is given by Eq. (21):

$$R = r + rRT \,. \tag{21}$$

The counting rate can therefore be corrected for dead time using Eq. (22):

$$R = \frac{r}{1 - rT} \,. \tag{22}$$

The value of dead time, T, can be determined using a two-source method by measuring the activity from two known sources r_1 and r_2. In theory, the measurement, r_3, should be the simple sum of the two sources as given by Eq. (23):

$$r_1 + r_2 = r_3 + b \,, \tag{23}$$

where b is the background counting rate. If each of these counting rates is corrected for dead time, then Eq. (23) becomes Eq. (24):

$$\frac{r_1}{1 - r_1 T} + \frac{r_2}{1 - r_2 T} = \frac{r_3}{1 - r_3 T} + b \,. \tag{24}$$

Because the background count rate may be considered negligible for this measurement, Eq. (24) can be expressed in the form of a quadratic equation:

$$r_1 r_2 r_3 T^2 - 2 r_1 r_2 T + r_1 + r_2 - r_3 = 0 \,. \tag{25}$$

T should be on the order of microseconds, thereby making T^2 negligible, enabling the simple solution for the dead time, *T*, given by Eq. (26):

$$T = \frac{r_1 + r_2 - r_3}{2 r_1 r_2} \,. \tag{26}$$

The above expressions are useful in the limit of low interaction rates and are considered part of what is known as the non-paralyzable model. If the dead time approaches 30% or more, the above model fails, and the paralyzable model must be used (which is not developed here; the reader is referred to [Knoll2010]. The significance of non-paralyzable versus paralyzable, from a practical perspective, relates to ability to count radiation in extreme fields. An ideal detector will have a linear response of count rate with interaction rate. A non-paralyzable detector will have an increasing count rate with interaction rate (but not necessarily linear at high interaction rate), but will tend to flatten out at high interaction rate. A paralyzable detector is one which has an increasing response at lower interaction rate, a peak response, and then a decreasing response with higher interaction rate. Essentially, in a paralyzable system, an interaction that occurs in the detector during the dead time will extend the dead time. The danger of a paralyzable detector is that if the count rate goes down as the interaction rate goes up, a person entering a progressively higher radiation field will think that he is moving into an area of lower radiation. For example, a GM counter in a high radiation field will tend to become paralyzed and may severely under-predict the radiation hazard.

4.1.2 Energy response and efficiency

All radiation detectors respond to energy deposition differently, based upon the type of radiation interacting with the detector and the energy of the radiation. For example, in a thin-window GM tube, beta particles interact best with the filling gas, followed by alpha particles and then gamma rays. This is due to the range of the respective particles in the gas in relation to the anode and cathode of the GM detector.

No radiation detector counts all the particles which are emitted from a source for various reasons. For example, resolving time affects the registered counts. In addition, many of the particles do not strike the tube at all because they are emitted uniformly in all directions from the source. The combination of effects that prevent the detector from counting all particles is known as the efficiency. Efficiency of a radiation counting system is achieved by comparing the measured count rate in the system to the disintegration rate (activity) of the source when the activity is given in units of disintegrations per second (dps; 1 dps = 1 Bq), where the conversion factor is 1 Ci = 3.7×10^{10} Bq.

The formula for determining the absolute efficiency is straightforward and is given as Eq. (27):

$$\%\,Efficiency = \varepsilon = \frac{C_{measured}}{A} \times 100\%, \tag{27}$$

where $C_{measured}$ is the measured count rate and A is the calculated source activity. Note that the time units of the numerator and denominator must be the same.

In fact, the absolute efficiency is a product of two separate phenomena, known as the geometric efficiency and the intrinsic (or quantum) efficiency. The relationship for absolute efficiency, ε, is given by Eq. (28):

$$\begin{aligned} \varepsilon &= \frac{\text{Number detected}}{\text{Number emitted}} \times 100\% \\ &= \frac{\text{Number reaching detector}}{\text{Number emitted}} \times \frac{\text{Number detected}}{\text{Number reaching detector}} \times 100\%. \end{aligned} \tag{28}$$

$$\begin{array}{cc} \Uparrow & \Uparrow \\ Geometric\ efficiency & Intrinsic\ efficiency \end{array}$$

The intrinsic efficiency is determined by the energy of the particles, atomic number of the detector, density of the detector, thickness of the detector, and other factors. A variety of geometrical efficiency examples are depicted in Figure 27. In the left figure, the source is a certain distance from the detector, and the geometrical efficiency will be low and based upon the angle subtended by the source to the detector edge. In the centre figure, the geometrical efficiency approaches optimal for a planar detector (often called a "2π geometry" because the solid angle of a cone intersecting with the detector is approximately $\Omega = 2\pi(1-\cos(\pi/2)) = 2\pi$). In the right figure, the geometrical efficiency approaches unity because the source is surrounded by detector material. This configuration is called a well detector (often called a "4π geometry" because the solid angle of a cone intersecting with the detector is approximately $\Omega = 2\pi(1-\cos(\pi)) = 4\pi$). In practice, detector efficiencies are specified as a function of radioisotope (or energy) for a given geometry.

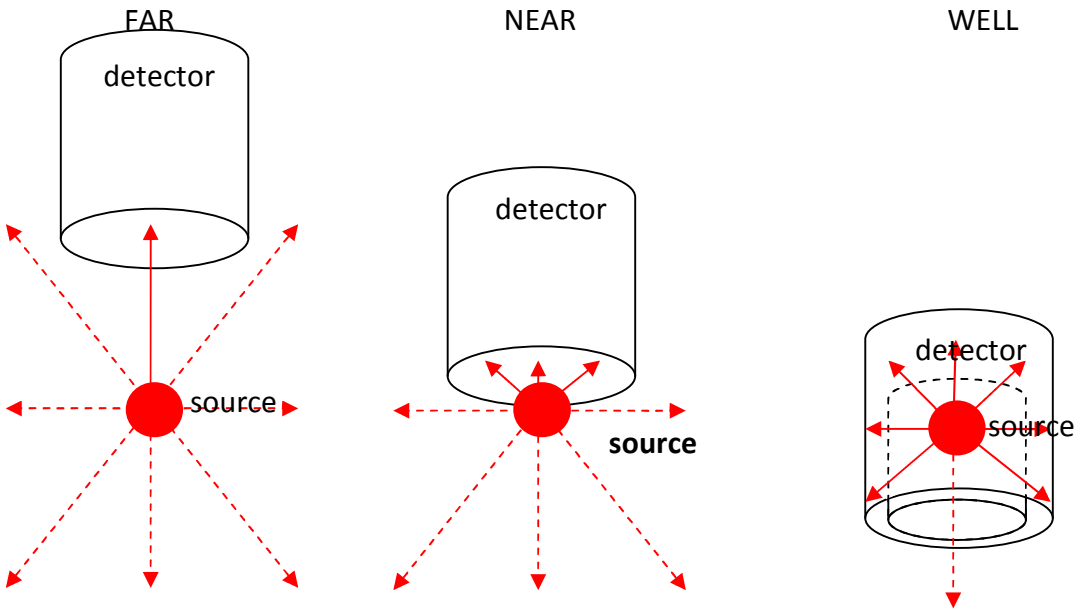

Figure 27 Geometric efficiency examples (far planar, near planar, and well detectors)

For detectors that perform spectroscopy (measurement of incident radiation spectra), both energy and efficiency calibration are generally required. As an example of how this calibration is performed, consider a simple scintillation (sodium iodide) gamma spectroscopy system. When a gamma ray (emitted during a change in an atom's nucleus) interacts with a sodium iodide crystal, NaI(Tl), the gamma ray will frequently give all its energy to an atomic electron through the photoelectric effect (PE). This electron travels a short, erratic path in the crystal, converting its energy into photons of light by colliding with many atoms in the crystal. The more energy the gamma ray has, the more photons of light will be created. A photomultiplier tube (PMT) converts each photon interaction into a small electrical current, and because the photons arrive at the PMT at about the same time, the individual currents combine to produce a larger current pulse. This pulse is converted into a voltage pulse with size proportional to the gamma-ray energy. The voltage pulse is amplified and measured by an analog-to-digital conversion (ADC) process. The result of this measurement is an integer between 1 and 1024 for a 10-bit ADC. One is the measured value for a voltage pulse less than a hundredth of a volt, and 1024 is the measured value for a pulse larger than approximately 8 volts (or the largest voltage pulse in the ADC). Pulses between 0V and 8V are proportionately assigned an integer measured value between 1 and 1024. This measure is called the channel number. The analog-to-digital conversion process is performed, and the computer records the measurements as the number of gamma rays observed for each integer measurement or channel number.

The spectrum is a visual display of the number of gamma rays as a function of the channel number. In the case of known sources (location of the energy peak(s) is well known), then a correlation of channel number with gamma-ray energy can be generated. The result is a graph of the channel number (memory location) as a function of gamma-ray energy (photopeak). The slope of this graph (energy/channel) is the energy calibration (a fitted equation may also be

used if the relationship is not linear over all energy values). An example of energy calibration is shown in Figure 28.

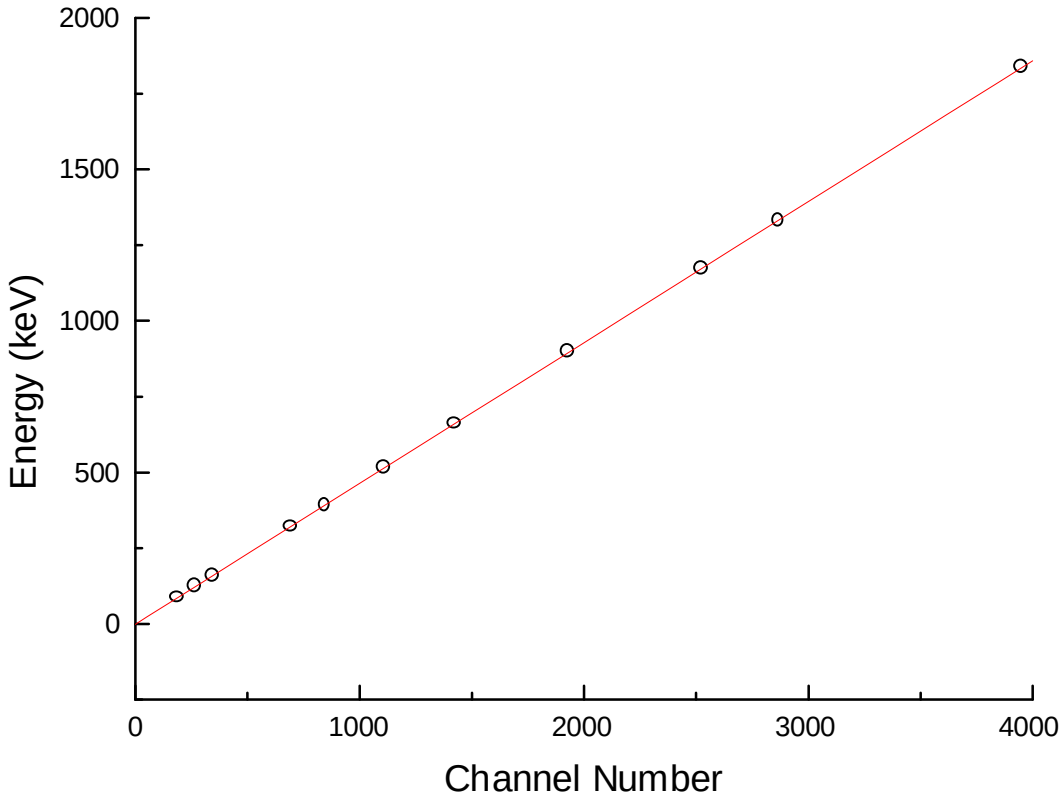

Figure 28 Example energy calibration graph

If the geometry is fixed and the activity of the calibration source is known, the peak area (integral counts) can be correlated with activity to determine the efficiency for each photopeak, as determined by Eq. (27). The efficiency can be plotted as a function of photopeak energy to obtain an efficiency curve, as depicted in Figure 29. This curve enables determination of the activity of an unknown source in a spectroscopic system.

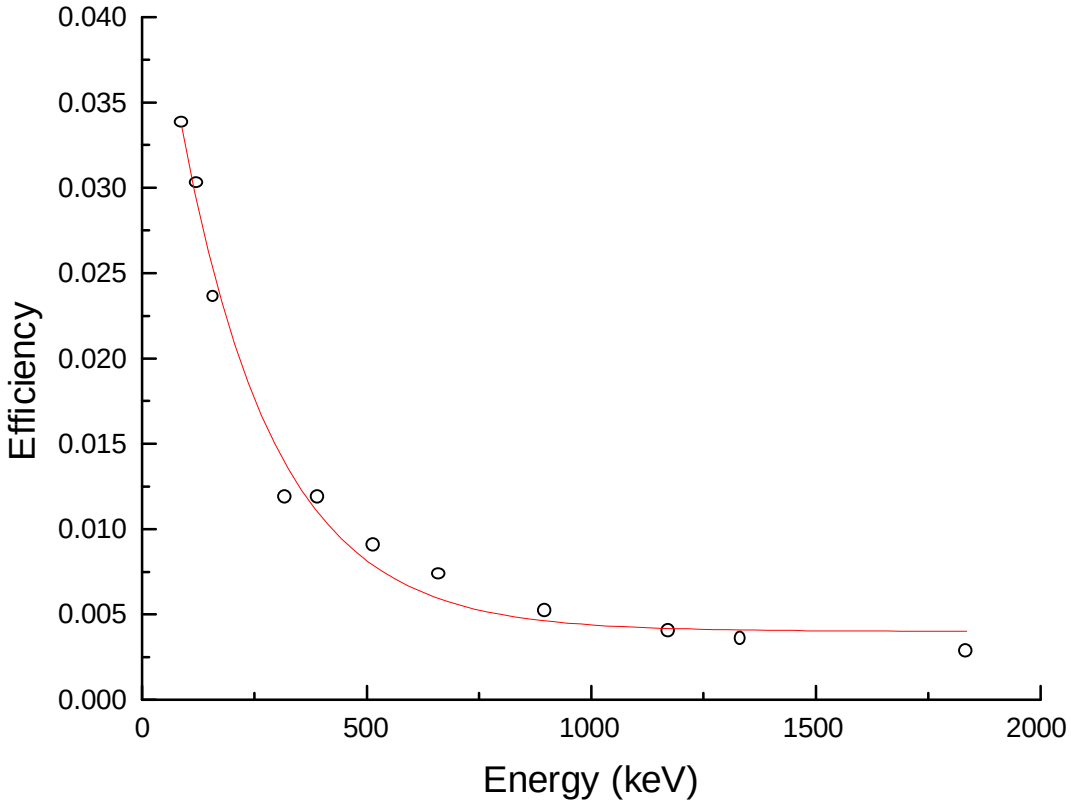

Figure 29 Example efficiency calibration graph

Energy calibration is essential for isotope *identification*, and efficiency is essential for *activity* determination.

4.1.3 Counting statistics and decision levels

Statistics play a very important role in particle counting because atomic concentrations are on the order of Avogadro's number. Therefore, it is impossible to deal with atoms individually, and statistics are used to assist in prediction of behaviour. Statistics uses two main quantities in predicting behaviour, the *mean* and the *standard deviation*.

The mean ($\bar{x}$) is computed as in Eq. (29):

$$\bar{x} = \sum_{i=1}^{N} \frac{x_i}{N}, \tag{29}$$

where the number of measurements (x_i) ranges from 1 to N.

The standard deviation of the data is computed as in Eq. (30):

$$\sigma = \sqrt{\frac{\sum_{i=1}^{N} \left(\bar{x} - x_i\right)^2}{N-1}}. \tag{30}$$

The standard deviation is a measure of the variation of the data without regard to position above or below the mean (that is, an absolute value using the square). Because most measurements tend to follow a Gaussian distribution about the mean (i.e., they are symmetrical),

the mean and standard deviation are widely used measures to predict behaviour. However, particle counting is not symmetrical about a mean. Radioactive decay is a random process that occurs very infrequently for low-activity samples and very frequently for high-activity samples. Three statistical distributions useful in particle counting will be discussed below.

4.1.3.1 Binomial distribution

The binomial distribution is used when there are two possible outcomes of an event. The probability of an outcome is a constant independently of the number of trials, and the selection of either outcome does not affect the outcome of subsequent trials. Assume an experiment where there are only two outcomes: (A) and (B). In a binomial distribution, *p* is defined as the probability of obtaining one of the outcomes (A), and *q* is the probability that the other (B) outcome occurs. Occasionally, statistical references use *1-p* in place of *q* (which is how *q* is calculated). The binomial distribution gives the probability, *P(n)*, that *n* out of *N* objects are A, is given by Eq. (31):

$$P(n) = \frac{N!}{(N-n)!n!} p^n q^{N-n} = \frac{N!}{(N-n)!n!} p^n (1-p)^{N-n} . \quad (31)$$

For this distribution, the mean, *m*, is given by Eq. (32):

$$m = pN , \quad (32)$$

and the standard deviation is calculated using Eq. (33):

$$\sigma = \sqrt{mq} = \sqrt{Npq} = \sqrt{Np(1-p)} . \quad (33)$$

To study radioactive decay, a distribution is required that represents a large number of counts and a small probability of success (large *N* and small *p*). The Poisson distribution is appropriate for this purpose.

4.1.3.2 Poisson distribution

The Poisson distribution is a special case of the binomial distribution given:

1. large number of counts, and
2. small probability of success.

The Poisson distribution, described by Eq. (34), gives the probability, *P(n)*, that *n* out of *N* objects are outcome (A):

$$P(n) = \frac{m^n}{n!} e^{-m} , \quad (34)$$

where *m* is the mean of the distribution (*m = Np*). A primary feature of the Poisson distribution is the ease of calculating the standard deviation using Eq. (35):

$$\sigma = \sqrt{m} . \quad (35)$$

For individual measurements, the standard deviation is the square root of the number of counts, (*N)*, as given by Eq. (36):

$$\sigma = \sqrt{N} \,. \tag{36}$$

For large values of *m* (*m* > 20), a more appropriate distribution, called the Gaussian distribution, is used.

4.1.3.3 Gaussian distribution

The Gaussian (normal, or bell) distribution is used when the sample size is large. An important feature of this distribution is that it uses continuous variables, unlike the Poisson and binomial distributions which use discrete variables. The Gaussian distribution is described by Eq. (37):

$$P(n) = \frac{1}{\sigma\sqrt{2\pi}} e^{-\frac{(x-m)^2}{2\sigma^2}} \,. \tag{37}$$

For a large number of measurements, the data should follow a Gaussian distribution and yield the mean and standard deviation presented by Eqs. (29) and (30).

4.1.3.4 Decision level

An important circumstance involving counting statistics is when low levels of radiation are present (which is often the case, for example, in environmental measurements). When a detector measurement is made at low counts, it must be determined whether the measurement is indicative of anthropogenic radiation or due to fluctuations in the natural background which is also being measured by the radiation-detection system.

The background can be estimated using two approaches. The first is a Poisson approach using Eqs. (29) and (30) for the mean and standard deviation of the measurements and making a judgement as to what multiple of the background measurement indicates a reading that is not due to background radiation. For example, it may be decided that the background mean + 3 standard deviations of the mean is appropriate.

The second, more rigorous approach is to use detection limit statistics. A full discussion is beyond the scope of this chapter, and the user is referred to [Currie1968] and [Chambless1992] for broad discussions of detection limit statistics.

The decision criterion (sometimes called the detection limit, L_D) is a value above some critical level (L_C) in which there is a quantifiable confidence that, if exceeded, you have actually measured radioactivity. The primary indicator is the critical level (L_C), which is defined as the go/no-go value above which a measurement is considered to represent radiation above a blank (or background) sample. The critical level (L_C) is defined by Eq. (38):

$$L_C = \sqrt{2} k \sigma_b \,, \tag{38}$$

where k is the Gaussian parameter related to area under the curve (sometimes termed "z") and σ_b is the standard deviation of the background radiation. Consider a normal distribution of background counts, a 95% confidence level, and the fact that for a normal distribution, the 95% confidence level occurs at k = 1.645; in this case, the critical level is defined as in Eq. (39):

$$L_C = \sqrt{2} \cdot 1.645 \cdot \sigma_b = 2.33 \sigma_b \,. \tag{39}$$

Therefore, the procedure to determine the critical level (the decision criterion) is:

1. Obtain numerous background measurements (for example, 20), spatially and temporally separated, but NOT near contamination or radioactive sources.
2. Determine the mean and standard deviation of the measurements
3. Multiply the standard deviation by 2.33 and add it to the mean to obtain the decision criterion.

For example, if the background-count rate mean is 30 cpm and the standard deviation is 5 cpm, the decision criterion is 30 cpm+12 cpm = 42 cpm. If a measurement above 42 cpm is obtained, it can be attributed to non-background radiation.

4.1.4 Particle counting versus spectroscopy

Generally speaking, gas-filled detectors are used for counting applications. In these, ionizing radiation interacts within the detection material to create a pulse that is counted by an electronic counting system. Scintillation and semiconductor detectors can also be used strictly as particle counters as well. In pulse mode, the signal from each interaction is processed individually. In current mode, the electrical signals from individual interactions are averaged together, forming a net current signal. In current mode, all information regarding individual interactions is lost. If the amount of electrical charge collected from each interaction is proportional to the energy deposited by that interaction, then the net current is proportional to the dose rate in the detector material. This mode is used for detectors subjected to very high interaction rates. The output from a particle-counting system will typically be counts per second (cps) or counts per minute (cpm). It is also possible through calibration to convert the pulses counted to dose rate or to integrated dose (Sv/h; Sv). A simple depiction of a particle-counting system is shown in Figure 30. Radiation interacts in the detector and creates pulses. If the pulses are above a discrimination threshold, they are passed to an integrator or counter circuit, and then if being used with a digital display, through an analog-to-digital converter and digital display. In addition, the count rate may be converted to dose rate using a calibrated radiation source and mapping count rate against known dose rate. The important aspect of a particle-counting system is that single values of count (rate) and/or dose (rate) are presented to the user as a function of time. The user does not generally have any direct information about the radioisotope or the activity being measured. To perform radioisotope identification and quantification, spectroscopy is normally used.

Spectroscopy is strictly defined as the identification of a radioisotope using the energy of emission of the decay particles. Spectrometry, on the other hand, is the quantification (i.e., activity determination) of a radioisotope using the intensity of the decay particle signal. For all practical purposes, spectroscopy is used to denote identification and quantification.

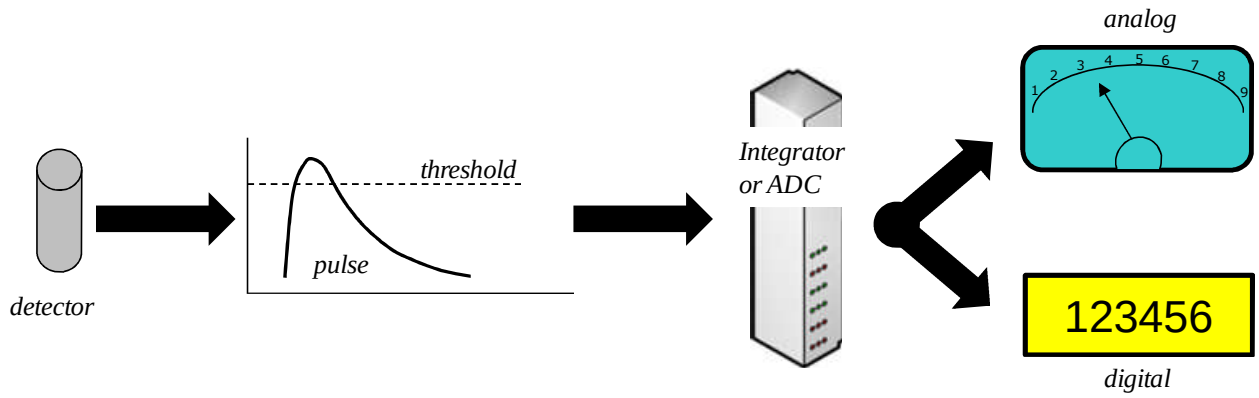

Figure 30 Simple particle-counting system

A simple depiction of a spectroscopy system is provided in Figure 31. In the system, radiation is emitted from the source and interacts with a material that responds to radiation with pulse heights that vary with, and are proportional to, the incident radiation energy. The various pulse heights, which are represented by a range of voltages, are processed in a multi-channel analyzer (MCA). The purpose of the multi-channel analyzer is to process the pulse heights into bins that are determined by memory allocation units (typically called channels). For example, if the pulses are amplified to have a maximum voltage of 10 volts and there are 1024 memory locations, then each memory location is 9.8 mV wide. The first channel is 0–9.8 mV, the second is 9.8–19.6 mV, and so on. It is normal to use a threshold discriminator to eliminate noise that would otherwise be counted as real events, and event logging can be controlled by a lower-level discriminator (LLD) and an upper-level discriminator (ULD). Events that fall within the window will be processed by the multi-channel analyzer. Each pulse is counted within its respective bin (analog-to-digital conversion) and as more counts accumulate, a spectrum is produced which is displayed with counts on the y-axis and channel number on the x-axis. The spectrum produced is characteristic of the radioisotope, and therefore identification is possible.

It can be readily seen that spectroscopy systems can also be used for particle counting. It is not unusual for portable spectroscopy systems to use a scintillation detector for both dose rate and spectral ID functions.

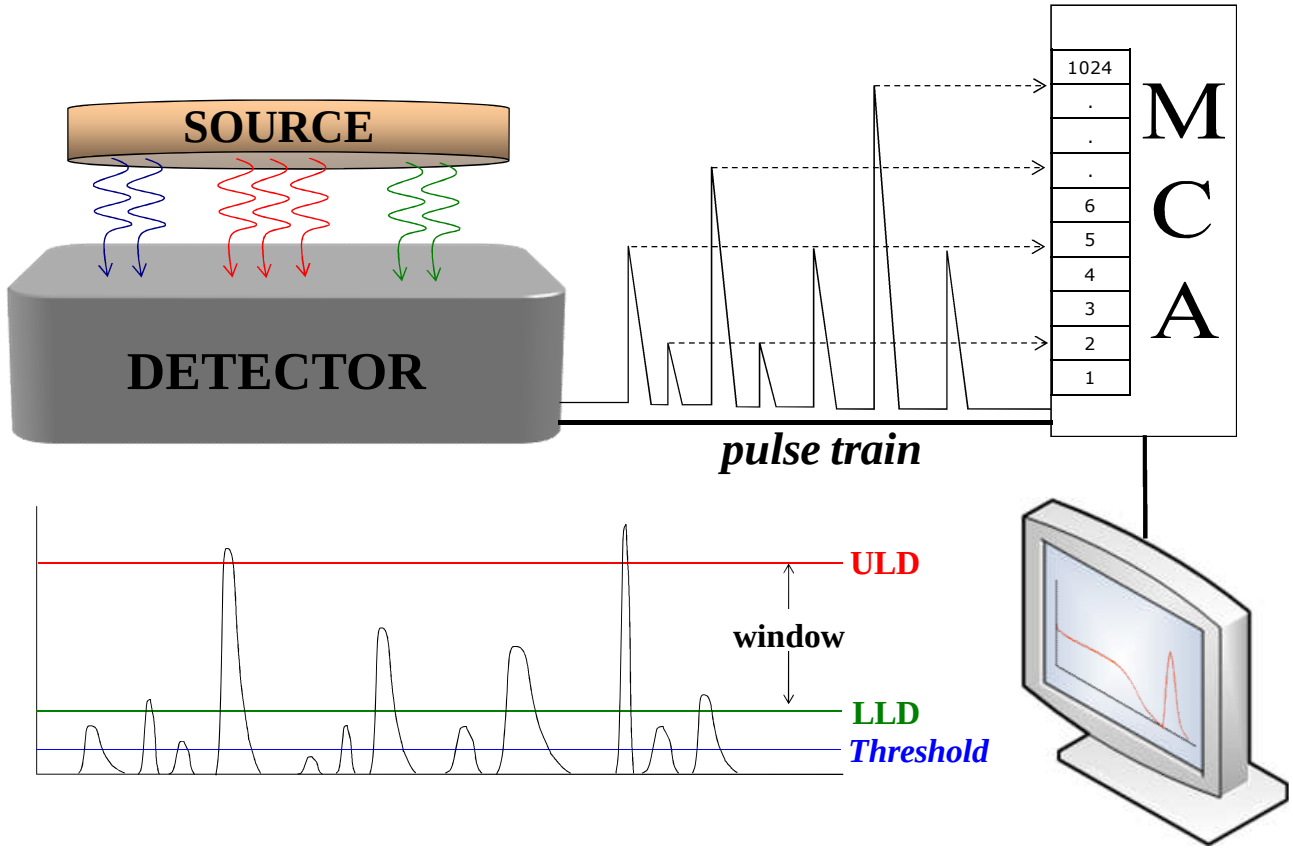

Figure 31 Simple spectroscopy process

Spectrometer systems can be summarized by the following steps:

- When a gamma ray interacts with a detector, a pulse is produced. For gamma spectroscopy, the pulses are produced when gamma rays interacting in the detector create electrons through the photoelectric effect, Compton scattering, or pair production.
- The size of the pulse reflects the energy deposited in the detector.
- An analog-to-digital converter (ADC) sorts the pulses into bins according to size.
- The results of the ADC analysis are displayed. The display, a plot of number of pulses (counts) versus pulse size (channel number), is referred to as a spectrum.
- If a large number of pulses have a similar size, they are sorted into adjacent channels and appear on the spectrum as a peak. A peak represents a number of pulses of similar size.

In an ideal spectrometer system, a peak would be produced that corresponds to the photopeak energy(ies) of the radioisotopes being measured, and the peak would be very narrow (approaching a delta function). In reality, a number of processes can occur within (a) the detector and (b) the shielding that is often present around the system. These processes generally produce undesirable effects in the spectrum, and therefore attempts are made to minimize the impact of these effects. The important aspect of that energy deposited in the detector (which is what the spectrometer measures and displays) is simply the energy in minus the energy out ($E_{dep} = E_{in} - E_{out}$).

4.1.4.1 Detector Interactions

The primary interactions of importance in the detector are as follows:

- **Photopeak.** A photopeak is generated in the spectrum when gamma rays deposit all their original energy in the detector through the photoelectric (PE) interaction. This is the primary signature in a gamma spectroscopy system and is used both to identify and to quantify the radiation.
- **X-Ray Escape Peak.** An X-ray escape peak is generated in the spectrum when gamma rays deposit all their original energy in the detector through the photoelectric effect (PE), except for the energy of an iodine or germanium X-ray that leaves the detector.
- **Compton Continuum.** The Compton continuum is produced when gamma rays interact in the detector through Compton scattering (CS) and then leave the detector.
- **Double Escape Peak.** The double escape peak is produced when gamma rays interact in the detector through pair production (PP) and two 511-keV annihilation photons escape the detector.
- **Single Escape Peak.** The single escape peak is produced when gamma rays interact in the detector through pair production (PP) and one 511-keV annihilation photon escapes the detector.

4.1.4.2 Shield interactions

If there is significant shielding around a spectroscopy system (for example, environmental radioactivity gamma spectroscopy systems often use a scintillator or semiconductor detection system in a lead shield, which is sometimes called a "castle"), then interaction occurring in the shield can produced scattered or secondary radiation in the detector that will be logged in the spectrum.

- **Lead X-Rays.** Lead X-rays are generated in the spectrum when gamma rays interact with a lead shield through the photoelectric effect (PE) and the resulting X-rays produced in the shield interact with the detector. The lead X-rays are around an energy level of 75 keV (73, 75, and 85 keV). If the shield were made of steel, iron X-rays might be seen, but these would be of such low energy that they would probably go undetected.
- **Backscatter Peak.** A backscatter peak is generated in the spectrum when gamma rays interact with the shield through Compton scattering (CS) and the 180^{o}-scattered gamma rays interact with the detector.
- **Annihilation Peak.** An annihilation peak is generated in the spectrum when gamma rays interact with the shield through pair production (PP) and the resulting annihilation photons at 511 keV interact with the detector.
- **Summation Peak.** A summation peak occurs when two gamma rays deposit all their energy in the detector at the same time. The energy of this peak is the sum of the two photopeak energies.
- **Bremsstrahlung.** Bremsstrahlung ("braking" radiation) is observed in the spectrum when high-energy beta particles are produced and slow down in the sample.

4.2 Gas-Based Detectors

The various operating regions for gas-filled detectors are shown in Figure 32.

At low applied voltage, below the ion-saturation region, there is a region called the recombination region. Many electrons and ions produced in the gas recombine because the voltage

applied between cathode and anode is not large enough to collect all the electrons. This region is not useful for counting radiation.

The next region is the ion-saturation region. The potential difference is sufficient to collect all freed electrons. A detector working in the saturation region is called an ionization chamber, and its output is proportional to the deposited radiation energy. Internal or thin-window ionization chambers are used as alpha-particle and fission-fragment detectors.

The next important region is the proportional region. The applied voltage is large enough that the electrons freed by the initial radiation are accelerated, so that they in turn ionize additional atoms or molecules (secondary ionization) to free more electrons. This electron multiplication generates an avalanche toward the anode for each primary electron that was freed. The applied voltage domain is called the proportional region because each avalanche is characterized by the same electron multiplication at a given applied voltage. The output signal is directly proportional to the deposited energy, although each pulse is many times larger than in the ionization region. The limited proportional region is at slightly higher applied voltage, and the proportionality of the output signal to the deposited energy at a given applied voltage no longer applies. Amplification of the greater deposited energy reaches its limit while that for the lesser deposited energy continues to increase. This region is usually avoided as a detection region.

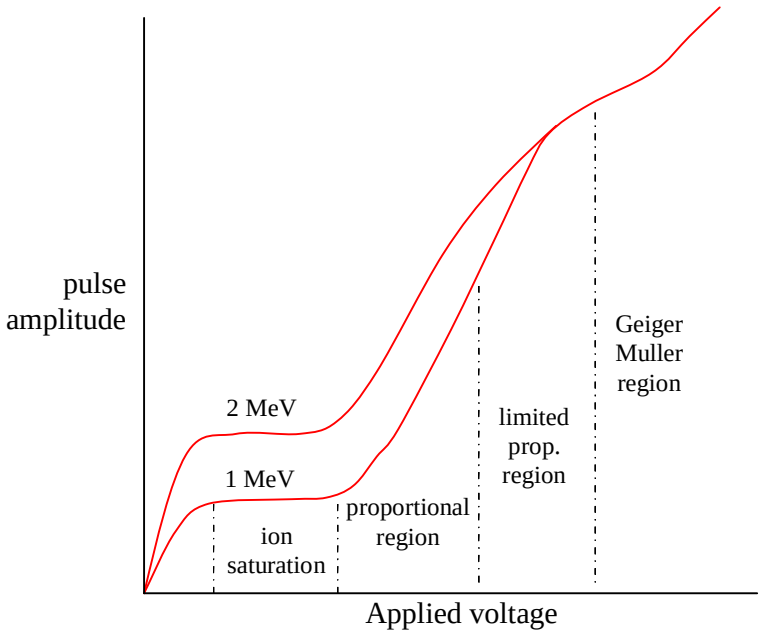

Figure 32 Operating regions of gas-filled counters

The next usable region is the Geiger-Müller region, which is characterized by an applied voltage high enough that any deposited energy produces sufficient secondary electrons to discharge the entire counting gas. A linear amplifier is no longer needed because there is sufficient electron production to generate a usable signal. In this region, it is not possible to distinguish between small and large depositions of energy. At higher applied voltages, electrical discharge occurs between the electrodes. This voltage region, called the continuous-discharge region, has been used for some purposes, but generally is avoided because the discharge can disable the detector.

Details about the two most common types of radiation detectors used in nuclear operations, namely the proportional counter and the Geiger-Müller (GM) counter, are discussed in the following sections.

4.2.1 Proportional counters

The proportional counter is a type of gas-filled detector introduced in the late 1940s. These counters are similar to GM counters in that they rely on gas multiplication to generate a signal. The primary difference between a GM and a proportional counter is that a GM counter produces a pulse when radiation interacts with it, irrespective of incident particle type or energy. A proportional counter generates different-size pulses that are dependent on the incident particle energy. It is, therefore, possible to distinguish between alpha and beta particles using a proportional counter, or between two different energies. Although proportional counters do not have the resolution required to perform spectroscopy, they are extremely useful for discriminating between alpha and beta particles.

The primary use of proportional counters in the nuclear industry is therefore in alpha-beta counters (for example, swipe counters), where it is desirable to discriminate between alpha and beta particles.

4.2.2 Geiger-Müller counters

Geiger-Müller (GM) counters were invented by H. Geiger and E.W. Müller in 1928 and are used to detect radioactive particles (α and β) and rays (γ and x). A GM tube usually consists of an airtight metal cylinder closed at both ends and filled with a gas that is easily ionized (usually neon, argon, or a halogen). One end consists of a "window" made of a thin material, mica, which allows alpha particles to enter (these particles can be shielded easily). A wire, which runs lengthwise down the centre of the tube, is positively charged with a relatively high voltage and acts as an anode. The tube itself acts as the cathode. The anode and cathode are connected to an electric circuit that maintains the high voltage between them.

When the radiation enters the GM tube, it will ionize some of the atoms in the gas. Because of the large electric field created between the anode and the cathode, the resulting positive ions and negative electrons accelerate toward the cathode and the anode. Electrons move or drift through the gas at a speed of about 10^4 m/s, which is about 10^4 times faster than the positive ions move. The electrons are collected a few microseconds after they are created, while the positive ions take a few milliseconds to travel to the cathode. As the electrons travel toward the anode, they ionize other atoms, producing a cascade of electrons called a gas multiplication or avalanche. The multiplication factor is typically 10^6 to 10^8. The resulting discharge current causes the voltage between the anode and cathode to drop. The counter (electric circuit) detects this voltage drop and recognizes it as a signal of a particle's presence. There are additional discharges triggered by UV photons liberated in the ionization process that start avalanches away from the original ionization site. These short-lived discharges are called GM discharges and do not affect performance. Once the avalanche of electrons is started, it must be stopped (quenched) because the positive ions may still have enough energy to start a new cascade. One early method was external quenching, which was done electronically by quickly ramping down the voltage in the GM tube after a particle was detected. This meant that any more electrons or positive ions created were not accelerated towards the anode or cathode. The electrons and ions would recombine and no more signals would be produced. The modern

method is called internal quenching. A small concentration of a polyatomic gas (organic or halogen) is added to the gas in the GM tube. The quenching gas is selected to have a lower ionization potential (~10 eV) than the fill gas (26.4 eV). When the positive ions collide with the molecules of the quenching gas, they are slowed or absorbed by giving their energy to the quenching molecule. They break down the gas molecules in the process (dissociation) instead of ionizing the molecule. Any quenching molecule that may be accelerated to the cathode dissociates upon impact, producing no signal. If organic molecules are used, GM tubes must be replaced periodically because they permanently break down over time (about one billion counts). GM tubes can also use a halogen molecule, which naturally recombines after breaking apart.

Different Geiger-Müller (GM) tubes have varying operating characteristics due to differences in their fabrication. Consequently, each GM counter has a different high voltage that must be applied to obtain optimal performance from the instrument. If a source of ionizing radiation is positioned beneath a tube and the voltage of the GM tube is ramped up (slowly increased by small intervals) from zero, the tube does not start counting right away. The tube must reach the starting voltage where the electron "avalanche" can begin to produce a signal. As the voltage is increased beyond that point, the counting rate increases quickly before it stabilizes. Where the stabilization begins is a region commonly referred to as the knee, or threshold value. Past the knee, further increases in voltage produce only small increases in the count rate. This region is the plateau we are seeking. Determining the optimal operating voltage starts with identifying the plateau. The end of the plateau is found when increasing the voltage produces a second large rise in count rate. This last region is called the discharge region. To preserve the life of the tube, the operating voltage is selected near the middle, but towards the lower half of the plateau (closer to the knee). If a GM tube operates near the discharge region and there is a change in its performance, the tube will operate in a "continuous-discharge" mode, which can damage it. Figure 33 shows a typical plateau shape for a GM tube. Above approximately 1100 volts, the tube enters the continuous-discharge region.

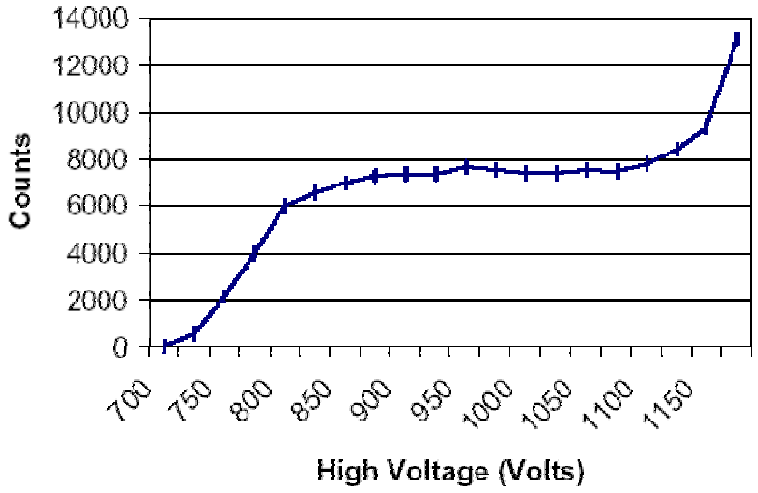

Figure 33 GM counter sample plateau

For hand-held radiation detection instrumentation commonly used in nuclear power operations, GM counters are the most common type of detectors; they can be found in radiation survey instruments, contamination detectors (for example, pancake detectors), and in personal dosimeters.

4.3 Scintillation-Based Detectors

Scintillation detectors are the most commonly used detectors for field applications because they do not require bulky cooling strategies. There are two stages in a scintillation detector unit: the scintillator and the photomultiplier. The scintillation detection process is outlined as follows:

1. Ionizing radiation deposits energy in the scintillator.
2. A portion of the absorbed energy causes electrons in the scintillator to move to a higher energy level.
3. A portion of these electrons immediately fall back down to a lower energy level. As they do, they emit a photon of light. Collectively, the photons of light form a flash, or scintillation.
4. Scintillations are converted into an electronic pulse by a photomultiplier tube (PMT). The brighter the scintillation, the larger is the pulse.

One particle of radiation (alpha, beta, gamma ray, X-ray, or neutron) interacting in the scintillator results in one scintillation (flash) of light and therefore one pulse. The greater the energy transferred to the scintillator, the greater is the number of excited electrons and the greater is the number of photons of light emitted in the scintillation. In other words, the greater the deposition of energy, the brighter is the flash of light and the larger is the pulse.

4.3.1 Scintillators

Scintillators are categorized as fluors as opposed to phosphors. Fluorescence has a lifetime in the excited state from 10^{-7} to 10^{-10} s, whereas phosphorescence has a lifetime in the excited state of 10^{-3} s or longer. It is desirable to have a material with a short excited-state lifetime to achieve a shorter resolving time. Scintillation materials can be classified as solid, liquid, or gas, as detailed below.

- **Gas scintillators:** The radiation energy excites the electrons of the gas molecules, and light photons are emitted when the electrons de-excite. Because these photons are typically in the UV range, a PMT must be chosen that is UV-sensitive. Gas scintillators are extremely fast, but their light output is poor (low efficiency). The most extensively investigated gas scintillators use the noble gases xenon and helium.
- **Liquid scintillators:** Liquid scintillators consist of an organic scintillator dissolved in an appropriate solvent. Normally, the sample to be counted is also dissolved in the scintillator. The advantage of this arrangement is a high counting efficiency for beta particles, even for betas of the lowest energies (e.g., ^{3}H). In some cases, the liquid scintillator is used to measure external radiation sources (e.g., cosmic rays). In such cases, the liquid scintillator is packaged and treated as if it were a solid crystal. One advantage of liquid scintillators is the fact that extremely large detectors can be constructed. This is the basis of the commonly found liquid scintillation counter (LSC).
- **Solid scintillators:** This is the most common type of material used for gamma spectroscopy.

Aside from their physical form, scintillators may also be classified chemically as organic or inorganic:

- **Inorganic:** gas scintillators (e.g., He, Xe), inorganic crystals (e.g., NaI, ZnS, CsI, LiI, BGO), glass scintillators.

- **Organic**: liquid scintillators (e.g., PPO, Bis-MSB, POPOP), organic crystals (e.g., anthracene, stilbene), plastic scintillators.

Two extremely common solid scintillation materials used in the nuclear industry are ZnS and NaI, which are discussed below.

Zinc Sulfide

Zinc sulfide is often doped (activated) with silver in the form of ZnS(Ag). It is typically used in alpha detectors and for heavy-particle detection. In the past, ZnS was combined with ^{226}Ra to form radium paint, which was used to paint clock and aircraft dials so that they would "glow in the dark". ZnS is fairly opaque and has high scintillation efficiency (~40%) with a long decay time.

Sodium Iodide

Sodium iodide is an alkali halide. Its use as a scintillator dates back to the late 1940s, and it is often activated using thallium in the form NaI(Tl). Thallium is added in trace amounts, at a level of approximately one atom in one thousand. NaI is used primarily for photon (X- or gamma) radiation detection. NaI has a high light yield with a scintillation efficiency of approximately 12%. In other words, approximately 12% of the gamma-ray energy deposited in the crystal will be emitted as light. The average energy of a photon emitted by NaI activated with thallium is 3 eV (which has a wavelength of approximately 415 nm, a wavelength readily detected by most photomultiplier tubes). For a hypothetical 1-MeV gamma ray that deposits all its energy in the NaI scintillator, one would expect 120 keV (12% of 1 MeV) to be emitted as light. With an average photon energy of 3 eV, this would be equivalent to 40,000 photons of light. NaI crystals can be grown to relatively large sizes for application in airborne/carborne spectroscopy systems or portal monitors.

Scintillators produce photons of light in the UV to near-visible region and require photomultiplier tubes (PMT) as part of the detection process (Figure 26).

4.3.2 Photomultiplier tubes (PMT)

A photomultiplier tube is used to detect very weak light signals. It is a photoemissive device in which the absorption of a photon results in the emission of an electron. It works by amplifying electrons generated by a photocathode impinged upon by a UV or near-visible light source. Photomultipliers acquire light through a glass or quartz window that covers the photosensitive surface (photocathode), which then releases electrons that are multiplied by electrodes known as metal channel dynodes. At the end of the dynode chain is an anode or collection electrode. The current flowing from the anode to ground is directly proportional to the photoelectron flux generated by the photocathode.

Electrons emitted by the photocathode are accelerated toward the dynode chain, which may contain in excess of 14 elements. Focussing electrodes are usually present to ensure that photoelectrons emitted near the edges of the photocathode will be likely to land on the first dynode. Upon impacting the first dynode, a photoelectron will invoke the release of additional electrons that are accelerated toward the next dynode. Note that photomultipliers produce a signal even in the absence of light due to dark current arising from thermal emissions of electrons from the photocathode, leakage current between dynodes, or background high-energy radiation.

4.4 Semiconductor-Based Detectors

Electrons in a solid can occupy some energy levels, but not others. These levels are not single discrete energies, but ranges of energies called bands. The highest energy band occupied by electrons is referred to as the valence band. The conduction band is of even higher energy than the valence band, but it is normally nearly void of electrons. The range of energies between the valence and conduction bands (forbidden to electrons) is called the band gap (forbidden band).

When the valence band is full, the electrons in the band are essentially fixed in place, i.e., tied to a particular site in the solid. Such a material will not conduct electricity unless electrons in the valence band can be given the energy to reach the conduction band where they would be free to move. If the valence band is not completely full, the electrons in it are free to move, and the solid will conduct electricity.

Solids can be divided into the following three categories:

- **Conductor**. A material in which the valence band is partly, but not completely, full.
- **Insulator**. A material in which the valence is completely full and for which the band gap is greater than 3–5 electron volts.
- **Semiconductor**. A material for which the valence band is completely full and for which the band gap is less than 3–5 electron volts. Semiconductor materials will not normally conduct electricity. However, the band gap is small enough that it is possible to lift enough electrons up into the conduction band for electrical conduction to occur.

When alpha or beta particles enter a semiconductor detector, they create positive and negative ion pairs (electron-hole pairs). Under the influence of an applied electric field, the positive members of the ion pairs (called holes) and the negative members of the ion pairs (electrons) move to the cathode and anode respectively. The result is an electronic pulse.

In the case of gamma detection, the gamma rays must first interact with the solid to produce secondary electrons (through the photoelectric effect, pair production, or Compton scattering). The secondary electrons then move through the detector and create electron-hole pairs which are collected at the electrodes. A semiconductor detector is often considered to be the solid equivalent of an ionization chamber.

Germanium is one of the most common semiconductor detector materials. A very common semiconductor detector used for environmental investigation is the high-purity germanium detector (HPGe). This detector needs to be cooled to near-liquid-nitrogen temperatures (~77°K) to operate. This type of spectrometer is normally found in a health physics laboratory and is used when high-resolution gamma spectroscopy is required. In general, for higher efficiency, scintillation detectors are used because they tend to be much larger-volume detectors, whereas semiconductor detectors are used when higher resolution is required. A typical commercial HPGe spectroscopy system is shown in Figure 34. At the bottom of Figure 34, the details of the opened lead castle can be seen, with the cylindrical HPGe detector element in the castle. The castle is designed to hold cylindrical geometries. For environmental sample analysis, it is very common to use a Marinelli beaker, which is a one-litre geometric design which covers the HPGe detector element for maximum efficiency.

Figure 34 Commercial HPGe system (top) and detail of lead castle (bottom)

4.5 Portable Instrumentation

All nuclear facilities make use of various types of portable instrumentation. Portable or hand-held instruments can be fabricated based on any detector technology, although some are more suited than others to the mission. Broad categories of portable instrumentation include: (a) general-purpose survey meter, (b) area contamination meter, (c) neutron meter, and (d) portable spectrometer.

4.5.1 General-purpose survey meter

A general-purpose survey meter (GPSM) is almost always a gamma detection system using scintillator, GM, or proportional detectors. This type of instrument is often called a gamma survey meter and is used to detect and quantify ambient gamma-radiation fields. The instruments have wide response to both incident photon energy and direction and provide readings

in counts per minute, counts per second, or both. If the instruments are calibrated, readings are in dose rate (μ-, mSv, or Gy per hour) and if they have an integration function, in total dose.

4.5.2 Area contamination meter

An area contamination meter (ACM) is an instrument used to detect and quantify alpha or beta/gamma surface contamination. It must be capable of detecting small quantities of radioactive material, but is not capable of determining whether contamination is fixed to a surface or loose. An ACM must be held very close to the surface to ensure accurate readings; however, there are no suitable portable instruments for detecting small quantities of tritium (^{3}H). A common type of area contamination detector is the pancake probe (see Figure 35), which is a relatively small-area thin-window GM tube that is very useful for frisking personnel for contamination. ACMs are usually based on a GM or other gas-filled tube. A common alpha detector uses a ZnS screen to convert alpha-particle interactions to UV photons, which are then detected by a photomultiplier tube. In this case, no gas is required.

Figure 35 Portable contamination meter with pancake probe

4.5.3 Neutron meter

Portable neutron meters are commonly proportional counters which use neutron detection reactions depending on the filling gas and generate charged particles. Two common reactions (and hence, detection-tube types) are BF_3- and ^{3}He-based respectively according to the reactions in Eqs. (40) and (41):

$$^{10}B + n \rightarrow \begin{matrix} ^{7}Li + \alpha + 2.79\,MeV \\ ^{7*}Li + \alpha + 2.31 MeV \rightarrow {}^{7}Li + \gamma + 0.478\,MeV \end{matrix} \quad (40)$$

$$^{3}He + n \rightarrow {}^{3}H + p + 765 keV\,. \quad (41)$$

These reactions are most likely to occur at low neutron energies, and therefore the counters are wrapped in hydrogenous (neutron-moderating) material. Detector-response functions can be altered using various techniques, such as perforated cadmium sleeves, so that a flat response over a wide energy range of incident neutrons can be achieved. There are also a variety of physical configurations, such as cylindrical, spherical, or even rectangular. Portable neutron detectors have historically been called “neutron REM-meters”, “long counters”, “Bonner spheres” (see Figure 36), “REM balls”, or by other names, depending on their origin and other physical characteristics.

Figure 36 Neutron Bonner sphere

4.5.4 Portable gamma spectrometer

The possibility of portable gamma spectroscopy became a reality as electronics, processors, microprocessors, and computers in general became smaller. Portable gamma spectrometers typically use scintillation-type detectors with small photomultiplier tubes. Some "portable" spectroscopy systems in the past have used semiconductor detectors; however, these are actually quite bulky and heavy, and their true portability is doubtful. Although portable spectroscopy generally cannot give the same performance as a laboratory system, these instruments can perform simplified radioisotope identification and typically offer count, count rate, dose, and dose rate functions.

4.6 Specialized Detectors

A number of specialized detector systems are commonly used in nuclear facilities to process personnel for potential internal or external contamination. These detection systems include portal monitors, whole-body counters, hand/cuff/foot counters, and thyroid counters.

4.6.1 Portal monitors

Portal monitors are used to monitor people in a passive way as they move between areas. They may be used as entrance and exit points from secure areas in nuclear plants or between clean and dirty operational areas. A portal monitor typically is constructed as a tubular frame which houses scintillation detectors (Figure 37). Portal monitors primarily respond to incident gamma radiation that may arise from radioisotope contamination on personnel or equipment. If a person is working in a potentially contaminated (dirty) area and some contamination is transferred to clothing, skin, or equipment, when that person passes through the portal monitor, the

gamma emissions from the contamination should be detected by the scintillators and trigger an alarm on the electronic panel. This will elicit a response from health physics personnel to clean the contamination and generate an incident report.

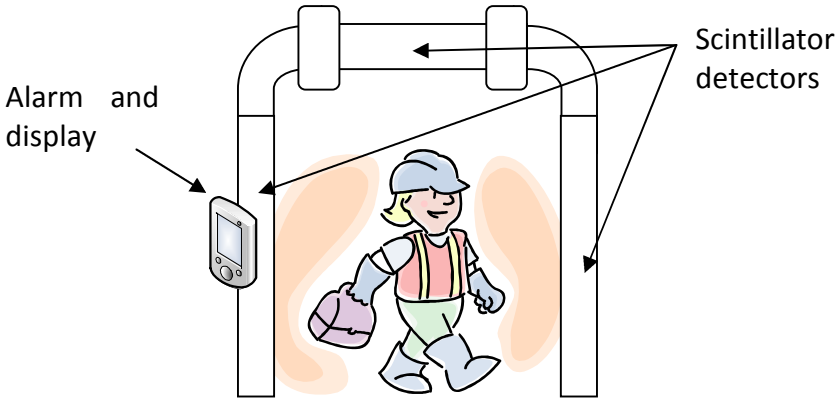

Figure 37 Portal monitor

Portal monitors are passive in that personnel only need to walk through the frame. Although portal monitors cannot readily distinguish between external and internal contamination, some are capable of determining radioisotopes and estimating activity (although this would tend to be a rough estimate). A commercial portal monitor is shown in Figure 38.

Figure 38 Commercial portal monitor

4.6.2 Whole-body counters

Similar to portal monitors (and sometimes synonymous) in that they are often used at entry and exit points, whole-body counters also detect radiation, although their primary mission is to detect and quantify internal contamination (presumably after a person has been externally decontaminated) through gamma emissions. Whole-body counters can use either scintillation or semiconductor spectroscopy detectors. Primary differences from portal monitors are that whole-body counters tend to have extensive shielding to reduce background radiation, they require a dwell time to obtain the measurement, and they may be designed for either standing or prone operation. A commercial whole-body counter is shown in Figure 39. Thin-window detectors can be seen along the length of the body, the head, and the feet.

Figure 39 Commercial whole-body counter

4.6.3 Hand/cuff/foot counters

For radiation workers exiting contaminated areas, it is vital to determine that the extremities, which have the highest probability of becoming contaminated, are adequately monitored. For this purpose, there are specialized hand-, cuff-, and foot-contamination monitoring systems. Generally, the person must stand on a detector and insert or place the hands or cuffs onto or into a detection screen. The detectors may be scintillator or gas-flow proportional counters. These types of counters are effective at detecting both alpha and beta radiation, which have much shorter ranges than penetrating gamma radiation.

4.6.4 Thyroid and lung counters

Thyroid counters, as the name suggests, are specialized detectors used specifically for detecting (principally iodine) radioisotopes in the thyroid (which may be present from fission-product releases that migrate into the coolant system from failed fuel elements). The thyroid counter is generally a scintillation-type detector aimed directly at the neck and thyroid area of the person being monitored. A lung counter is used specifically to detect and quantify radioisotopes that may have been inhaled. The detectors are arranged around the lung area, and due to the geometrical constraints and the longer count times required (especially for actinides), the person being monitored is generally sitting in a reclined seat, with the detectors near the lung region. Detectors are usually scintillation- or semiconductor-based.

4.7 Dosimetry Techniques

This section will consider thermo-luminescent dosimeters (TLD, as dosimeter of record), electronic personal dosimeters (EPD), and liquid scintillation counting (LSC) for bioassay dose estimation.

4.7.1 Thermo-luminescent dosimetry (TLD)

Thermo-luminescence (TL) is a thermally stimulated phenomenon; that is, material absorbs energy from ionizing radiation or light. This energy is stored, and part of it is released in the form of light when the material is heated. This phenomenon was first formally reported by Robert Boyle in his address to the Royal Society of London on October 28, 1663, when he described, "*observation of a glimmering light when he heated a diamond in the dark*" [Uchrin1988]. In the 1950s, research suggested the possibility of applications of thermo-luminescent properties, including radiation dosimetry, which led to development of more sophisticated TL phosphors and commercially available TL readers, making thermo-luminescent dosimetry (TLD) for ionizing radiation widely possible in the 1960s.

The main advantages of thermo-luminescent dosimeters (TLD) are their applicability over a wide dose range (10^{-6}–10^{5} Gy), their usability for various types of radiation including mixed fields, and their small physical size, which enables them to be readily worn as personal dosimeters. One of the fundamental disadvantages of TLDs is that the act of reading the dosimeter effectively "zeros" the dosimeter (through a process called annealing), which means that there is no permanent record of the dose.

The thermo-luminescence mechanism is highly complex and based on solid-state physics. TL materials are insulators with a crystalline structure. In a perfect crystal lattice, the atomic electronic levels are broadened into a series of continuous bands (the conduction band) separated by a several electron volt-wide "forbidden" energy region from the highest filled band, called the valence band. Usually, the conduction band is empty, and the valence band is filled. Crystals always contain imperfections: thermal or intrinsic defects, extrinsic defects, or substitutional impurities and radiation-induced defects [Uchrin1988]. The presence of lattice imperfections and impurities is essential for thermo-luminescent processes. The energy levels presented by them are situated in the forbidden region and act as traps or recombination centres for electrons and holes created in the crystal due to an excitation from interaction with ionizing radiation.

Electrons and holes will remain in traps provided that they do not acquire sufficient energy to escape. The number of trapped states is directly proportional to the number of ionizing-radiation interactions with the material and hence the total dose. When the temperature of the material is raised, the trapped charge carriers are given sufficient energy to escape from their traps to the conduction band, where they recombine at a luminescence centre, and the excess energy is radiated as visible or UV light, which is recorded by a photomultiplier tube (see Section 4.3.2). The TLD reader unit provides a correspondence between the signal generated from the measured light and the radiation exposure of the TL material.

Reading a TLD chip is relatively simple. The TL material is heated from the ambient temperature up to 300°C–400°C, and the emitted light is collected and measured quantitatively. The TLD reader consists of four components: (i) heating unit, (ii) light collection and detection system (PMT), (iii) signal measuring system, and (iv) data recording system. The emission of light from heated TLD material is often called a "glow", and the resulting spectrum from heating the material to release the traps is called a "glow curve". A typical glow curve obtained from a Harshaw model 3500 TLD reader for LiF TLD material irradiated to 500 µSv total dose from a 1 Ci ^{137}Cs irradiator is depicted in Figure 40. The portion of the curve corresponding to the delivered dose is denoted by the region of interest marked with vertical lines.

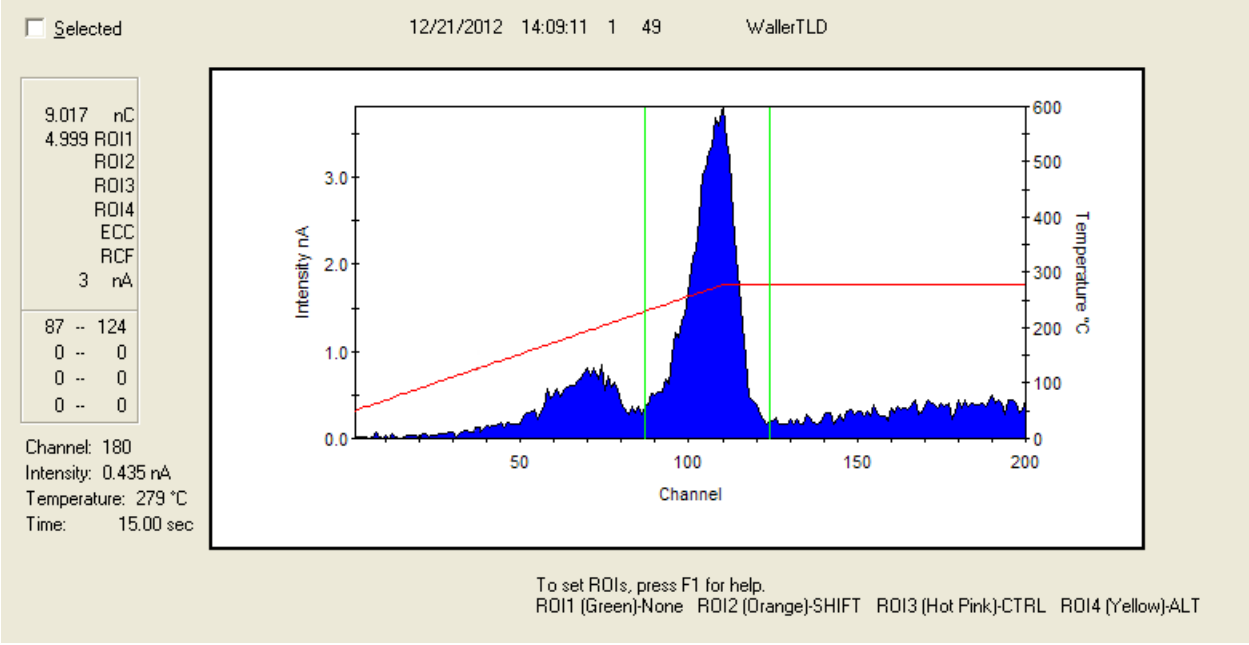

Figure 40 TLD glow curve from irradiated LiF chip

A variety of physical and chemical forms of thermo-luminescent dosimeters are used for different radiations and dose ranges. For example, physical forms of TLDs can include chips, disks, powders, and rods; a subset of materials used as dosimeters include LiF, $Li_2B_4O_7$, CaF_2, $CaSO_4$, and Al_2O_3. TLDs are characterized by their dose-response properties, energy dependence, sensitivity, and fading characteristics (fading is unintentional loss of stored signal due to thermal or optical release of traps). The observed glow curve from reading TLDs may be affected by chemo-luminescence (luminescence due to chemical reactions from material impurities) and tribo-luminescence (luminescence caused by mechanical effects in material preparation). These effects can be greatly reduced by inert gas (typically N_2) flushing. A commercial TLD reader and LiF chips are shown in Figure 41.

Figure 41 Commercial TLD reader (top) and tray of LiF chips (bottom)

The TLD is the dosimeter of record for nuclear energy workers (NEW) in Canada (National Dose Registry, Radiation Protection Bureau, Health Canada), as it is in most countries world-wide. All nuclear energy workers in Canada wear TLD dosimeters if they are working around sources of radiation exposure and as directed by the health physics department.

4.7.2 Electronic personal dosimeter (EPD)

An electronic personal dosimeter (EPD) is a detector calibrated to provide dose rate and dose readings in real time. The readings may be present on a display directly on the dosimeter, stored for later retrieval, or transmitted to a base station. The primary advantage of an electronic personal dosimeter is that it can be set with alarm points to warn the user (through audible and visual alarms) in real time if a dose or dose rate is being exceeded. This makes it a very useful tool for ALARA adherence. Key features of EPDs include: small size, light weight, fast response, continuous update and display, and timer/stay time functions. EPDs are generally

made from small GM tube or semiconductor detectors, and although the most common are sensitive to X- and gamma radiation, some have neutron-detection capabilities.

Although EPDs do not provide a legal dose of record, they are often worn in conjunction with TLDs by personnel working in nuclear-facility radiation environments, especially in non-homogeneous fields or in areas where exposure may be in excess of action or regulatory limits.

4.7.3 Liquid scintillation counting (LSC)

Liquid scintillation counting (LSC) is a radioanalytical technique developed in the 1950s and defined by incorporation of a radioisotope analyte into a scintillating liquid. It is a very sensitive and widely used technique for detection and quantification of radioactivity and is applicable to most forms of nuclear decay emissions (alpha and beta particles, electron capture, and gamma-ray-emitting radionuclides).

Consider the case of a beta-emitting radioisotope in a liquid scintillation material. The beta particle dissipates energy by collisions in the liquid, and the energy is absorbed by the medium in three forms: heat, ionization, and excitation of the molecules in the solution. Excitation of the solution molecules is the mechanism of the liquid scintillation technique. Facilitation of efficient transfer of energy between beta particles and the solution is accomplished using a solvent material and a scintillation solute solution (together called a “cocktail”). The scintillation solute is a fluor, and excited solvent molecules can transfer energy to one another and also to the solute. An excited solvent molecule creates an excited state in the solute, and as the excited orbital electrons of the solute molecule return to the ground state, a photon of UV or near-visible light is generated, which can be detected by a photomultiplier tube. The intensity of light from the scintillation process is proportional to the initial energy of the beta particle.

The detection process for beta decay is summarized in Figure 42.

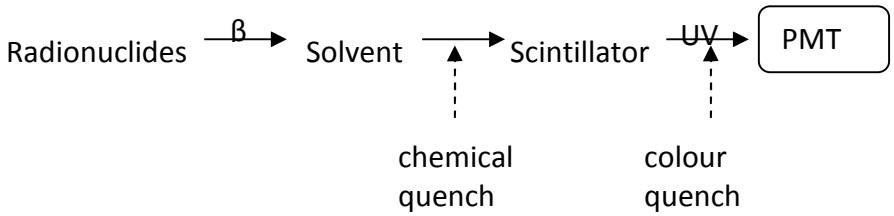

Figure 42 LSC detection process

The counting efficiency of the solvent-scintillator system is affected by many different factors that may reduce detection efficiency. Two dominant factors are: (i) chemical quenching (sometimes called impurity quenching), which causes energy losses in the transfer from solvent to solute, and (ii) colour quenching, which is the attenuation of UV light photons in the solution. Both quench factors can be compensated for through calibration. Other effects that may degrade LSC performance include: (a) thermionic effects, which are noise pulses that are extraneous to the true signal, yet resemble electronic pulses resulting from nuclear decay events; (b) photoluminescence, which results in activation of the cocktail or vial by ultraviolet light, which can occur by exposure to sunlight (LSCs rely on light-tight counting chambers); and (c) static electricity, which is a very common source of counting interference due to buildup and subsequent discharge of static electricity on LSC vials.

The electronic process of light detection and pulse generation in an LSC analyzer is depicted in Figure 43. In most scintillation counters, two photomultiplier tubes collect the total light produced within the scintillation vial that either (a) falls directly onto the two photocathodes or (b) is reflected onto each photocathode by a reflector centrally mounted between the two PMTs. Radioactive decay events produce approximately 10 photons per keV of energy. Beta decay yielding a multiplicity of photons will stimulate both PMTs at the same instant in time, and the signal from each PMT is fed into a summing circuit which produces an output only if the two signals occur simultaneously, which is called coincidence. Because electrical noise from the PMTs is produced randomly over time, it occurs at a sufficiently low rate to be excluded by the coincidence circuit (below the equivalent of 1 keV). The sample in the counting chamber and the PMTs are surrounded by lead, typically about 5 cm in all directions, which generally reduces the background radiation to low levels.

The output from the analog-to-digital converter is processed with a spectrum analyzer calibrated in keV, and regions between 0 and 2000 keV are used for sample analysis. For example, the region of interest for ^{3}H is 0–18.6 keV (^{3}H maximum beta energy = 18.6 keV) and for ^{14}C is 0–156 keV (^{14}C maximum beta energy = 156 keV).

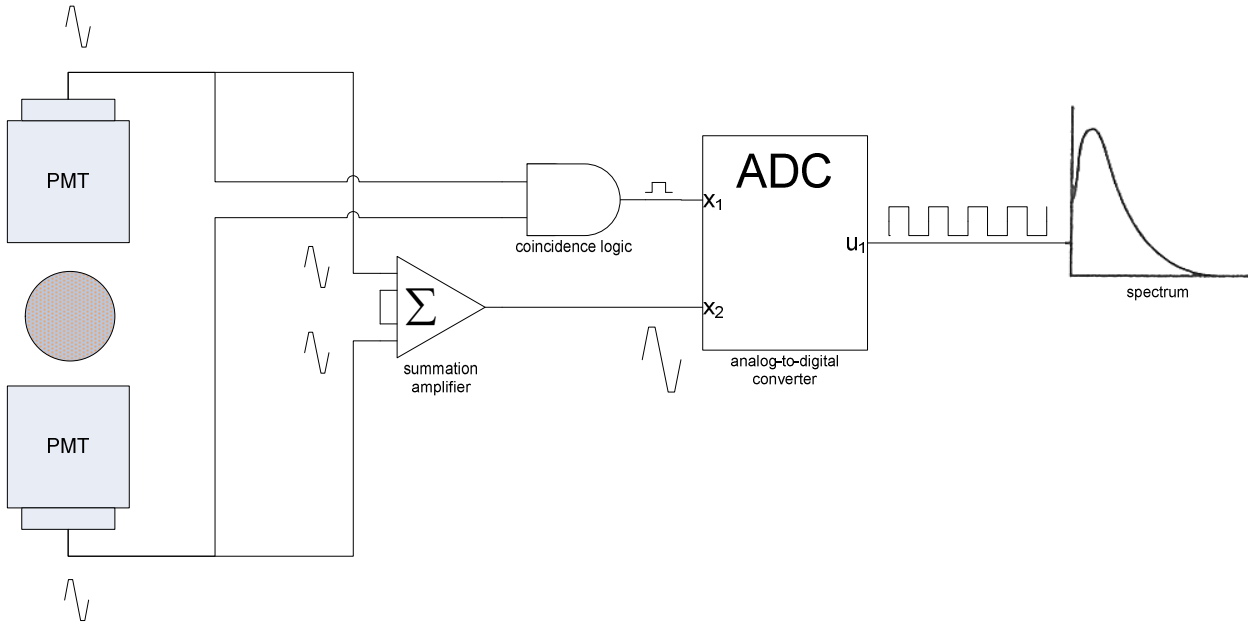

Figure 43 LSC electronics

Liquid scintillation counting is used in the nuclear industry for swipe analysis, environmental sample analysis, and urine (bioassay) analysis. LSC is vitally important to detection of one important radionuclide in CANDU nuclear power stations: tritium (^{3}H). Tritium is extremely hard to detect with field-portable instrumentation because the average energy of the beta particles emitted is very low (5.6 keV). In nuclear plants, LSC is routinely used for determining tritium intake through urine analysis. When tritium contamination is suspected on surfaces, quartz fibre or paper swipes can be used to wipe down a surface and then be analyzed in an LSC. Likewise, when performing environmental sample analysis for tritium, especially on water samples, LSC is the analytical technique of choice for tritium. A commercial LSC (for urine bioassay) is shown in Figure 44.

Figure 44 Commercial LSC counter (top) and detail of trays with vials (bottom)

4.8 Summary

Ionizing radiation cannot be detected by the human senses, and therefore instrumentation is required. A variety of radiation detection instruments is used in nuclear facilities to support routine operations such as dosimetry, radiation protection, and environmental monitoring. Instrumentation is selected by health physics personnel based on its applicability to the mission design. Radiation fields to be measured in nuclear facilities may be in the form of external fields, water effluent, airborne gases, or particulates. Instrumentation is of critical importance for the safe operation of nuclear facilities such as nuclear reactors and therefore is an integral part of the radiation-safety management plan.

5 External Radiation Hazards and Shielding

External radiation hazards from beta, gamma, and neutron sources are discussed in this section, with consideration of the ALARA principle. External radiation fields are always present within nuclear reactors, and therefore consideration of expected doses and strategies to minimize dose are required. Some historical perspective on external dosimetry and on shielding can be found in [Poston2005] and [Shultis2005] respectively.

5.1 External Sources and Dosimetry

The concepts of exposure and dose were explored in Section 3.1. The following subsections discuss the interactions of the primary radiations of concern to CANDU health physics—specifically, light charged particles (such as beta particles and electrons), X- and gamma rays, and neutrons. Note that under normal operating conditions, the primary radiation hazards to personnel expected in a CANDU plant are from gamma, neutron, and beta radiation. Heavy charged particles such as alpha particles are not an external hazard in CANDU plants because they occur only as radioactive decay emissions and because their energy (typically on the order of 4–5 MeV) is not enough to penetrate the outer layer of skin.

Beta radiation exposure is explored as surface contamination, hot particles on the skin, and a submersion dose. Gamma-ray equations for point sources (specific gamma constant), line sources, and volume sources are developed, and fast and thermal neutron-dose calculations are introduced.

5.1.1 Beta radiation

The ways in which an external beta radiation dose can be delivered to human targets are summarized in Figure 45. Generally speaking, it is possible to be submerged in a plume of beta-emitting radioactive material and thereby be exposed in an isotropic field of beta particles (with most radionuclides, there will be a corresponding gamma field that will require analysis). It is possible to be exposed through a plane source of beta particles (for example, a spill of beta-emitting radionuclides) where the target tissue is at some distance from the source, and it is also possible to encounter a contact source of beta-emitting material (for example, by touching a spill or a "hot" particle) that can generate a significant dose. Beta radiation can be significantly attenuated by clothing, boots, gloves, and goggles. Detailed descriptions of charged particle ranges and stopping powers can be found in Chapter 3.

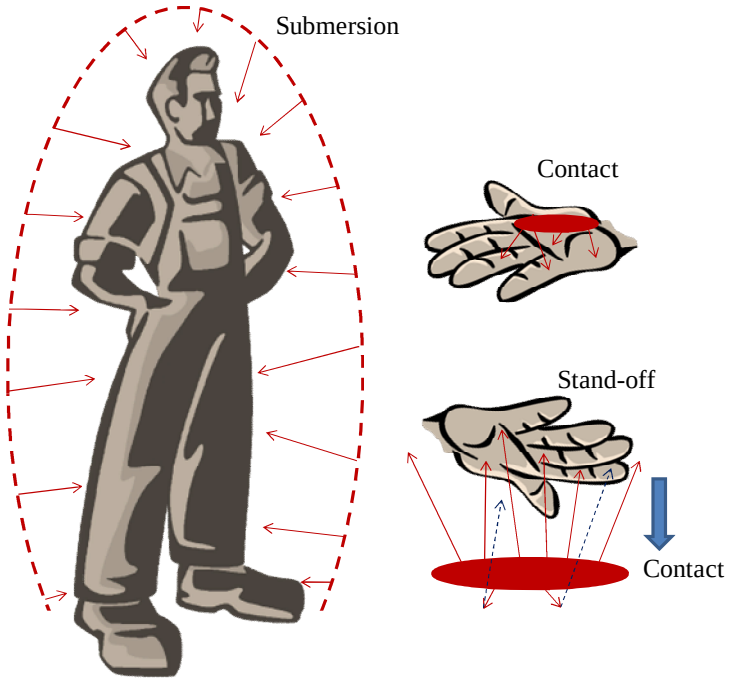

Figure 45 Beta dose delivery geometries

The intensity of beta radiation for depths less than the range of the beta particles can be described by Eq. (42):

$$\phi = \phi_o e^{-\mu_\beta t} \quad , \tag{42}$$

where

φ is the beta intensity at depth t (particle flux, energy flux, etc.)
φ_o is the initial intensity (same units as φ)
μ_β is the beta-ray absorption coefficient (cm^2/g)
t is the depth(density thickness) in material (g/cm^2).

The beta attenuation coefficients for tissue and air are given by Eqs. (43) and (44) respectively:

$$\mu_{\beta,\,tissue} = 18.6\left(E_{\max} - 0.036\right)^{-1.37} \quad \frac{cm^2}{g} \tag{43}$$

$$\mu_{\beta,\,air} = 16\left(E_{\max} - 0.036\right)^{-1.4} \quad \frac{cm^2}{g} \quad , \tag{44}$$

where E_{max} is the maximum beta-ray energy in MeV.

The dead layer of skin provides some beta shielding and is typically represented by 0.007 g/cm^2 density thickness. After the beta particles pass through the dead layer of skin, all beta energy is deposited in living tissue (betas travel no more than approximately 1 cm in tissue), and the energy deposited is essentially equal to the dose.

Beta doses from common industrial scenarios are presented below.

5.1.1.1 Beta dose from a source on a surface

The dose calculated is from a surface contamination scenario with and without a stand-off distance of the tissue from the source. A contaminated surface may be represented by Figure 46. From a dose perspective, this may be considered as a "2π" geometry; half the betas travel up towards the receptor, and half travel down into the surface where the contamination is located.

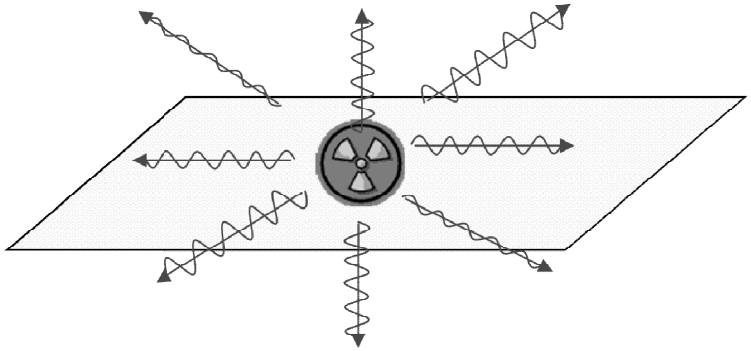

Figure 46 Beta surface contamination

The energy flux at the surface can be represented by Eq. (45):

$$\varphi = 0.5 C_a \overline{E}\,, \tag{45}$$

where

φ is the energy flux at the surface (MeV cm^{-2} s^{-1})
$\bar{E}$ is the average energy per disintegration (MeV)
C_a is the areal activity (Bq cm^{-2}).

The beta surface dose rate, considering the uncollided particles, may be determined using the beta attenuation coefficient as in Eq. (46):

$$\dot{D}_\beta = \phi\ \mu_\beta = 0.5 C_a \bar{E} \mu_\beta \left(\frac{MeV}{g-s} \right). \tag{46}$$

Rewriting Eq. (46) in standard SI dose units yields Eq. (47):

$$\dot{D}_\beta = 0.5 C_a \bar{E} \mu_\beta \left(\frac{MeV}{g-s} \right) = 2.88 \times 10^{-7} C_a \bar{E} \mu_\beta \left(\frac{Gy}{h} \right). \tag{47}$$

Of interest also is the contact dose to the skin (the scenario in which the tissue comes into contact with a surface that has beta-emitting contamination). The dose rate to tissue on contact is represented by Eq. (48):

$$\dot{D}_\beta = \frac{0.5 \times f_\beta \times C_a \frac{Bq}{cm^2} \times \bar{E} \frac{MeV}{transform} \times 1.6 \times 10^{-13} \frac{J}{MeV} \times 3600 \frac{s}{h} \times \mu_\beta \times e^{-\mu_\beta \times 0.007 \frac{g}{cm^2}}}{0.001\ J\ /\ g/Gy}, \tag{48}$$

and the dose rate for tissue on contact with a contaminated surface is given by Eq. (49):

$$\dot{D}_{contact} = \left(3.6 \times 10^{-4}\right) \cdot C_a \left(\frac{Bq}{cm^2} \right) \cdot \bar{E} \left(\frac{MeV}{transform} \right) \cdot \mu_\beta \cdot e^{-0.007 \times \mu_{\beta_{tissue}}} \left(\frac{mGy}{h} \right), \tag{49}$$

which assumes that approximately 25% of the beta particles going down into the contaminated surface are backscattered [Cember2009] towards the dose point (i.e., f_β=1.25) and the exponential represents attenuation through 0.007 g/cm^2 of tissue (the dead tissue-layer density thickness).

For tissue at some distance *d* away from the contaminated surface, air attenuation must also be taken into account, as in Eq. (50). Subsequent layers of attenuating material (such as gloves on hands) may be taken into account using similar exponentials:

$$\dot{D}_{at\ d} = \left(3.6 \times 10^{-4}\right) \cdot C_a \left(\frac{Bq}{cm^2} \right) \cdot \bar{E} \left(\frac{MeV}{transform} \right) \cdot \mu_\beta \cdot e^{-d \cdot \mu_{\beta_{air}}} \cdot e^{-0.007 \times \mu_{\beta_{tissue}}} \left(\frac{mGy}{h} \right). \tag{50}$$

5.1.1.2 Beta dose rate from contamination on skin (or hot particles)

For the case where the beta contaminant is in direct contact with external tissue with no backing material, the beta dose rate is expressed by Eq. (51). Note that no backscatter (f_b) factor is required in this scenario because there is air "behind" the source:

$$\dot{D}_\beta = \frac{0.5 \times C_a \frac{Bq}{cm^2} \times \bar{E} \frac{MeV}{transform} \times 1.6 \times 10^{-13} \frac{J}{MeV} \times 3600 \frac{s}{h} \times \mu_\beta \times e^{-\mu_\beta \times 0.007 \frac{g}{cm^2}}}{0.001\ J/g/Gy} = \left(2.88 \times 10^{-4}\right) \cdot C_a \cdot \bar{E} \cdot \mu_\beta \cdot e^{-\mu_\beta \times 0.007} \left(\frac{mGy}{h}\right) \quad (51)$$

If the beta emitter is not in contact with the external tissue, other attenuating material can be added using exponential attenuation factors with appropriate attenuation coefficients. Some materials that can attenuate betas for skin doses in an industrial setting are listed in Table 13. If the material is hydrocarbon-based, the tissue beta attenuation coefficient may be an appropriate approximation in the exponential.

Table 13 Standard thickness of various beta attenuators (from [Cember2009])

Material	Thickness (mm)	Density (g/cm^3)
Lab Coat (Plastic)	0.1	0.036
Cotton Glove Liner	0.3	0.3
Surgeon's Glove	0.5	0.9
Outer Glove (thick)	0.45	1.1
Ribbed Outer Glove	0.55	0.9
Plastic Bootie	0.2	0.6
Rubber Shoe Cover	1.2	1.0

5.1.1.3 Beta dose from submersion in a plume

In an "infinite" cloud, the rate of energy emission is equal to the rate of energy absorption, and the air dose is given by Eq. (52) for dry air:

$$\dot{D}_{\inf} = \frac{k \cdot C \cdot \overline{E}}{\rho_{air}} = 4.45 \times 10^{-7}\ C\ \overline{E} \quad \left(\frac{mGy}{h}\right), \quad (52)$$

where

k is a constant to yield units in mGy/h (5.76×10^{-7})
C is the air concentration of the radionuclide (Bq m^{-3})
$\bar{E}$ is the average beta energy (MeV)
ρ_{air} is the air density (1.293 kg/m^3).

More interesting is the dose to tissue for a person submerged in an infinite cloud of beta-emitting radionuclide. In this case, the expression given by Eq. (52) is modified by a factor of 1.1 for the approximation that tissue absorbs 10% more energy than air on a per-kg basis and by a factor of 0.5 representing the fact that in the cloud, half the beta particles move inward (towards tissue) and half move outward (away from tissue). Finally, a factor of $e^{-\mu * 0.007}$ compensates for passage through the dead skin layer. Equation (52) therefore becomes Eq. (53):

$$\dot{D}_\beta = \frac{0.5\times 1.1\times C\,\frac{Bq}{m^3}\times 1\frac{tps}{Bq}\times \bar{E}\,\frac{MeV}{transform}\times 1.6\times 10^{-13}\,\frac{J}{MeV}\times 3600\,\frac{s}{h}\times e^{-\mu\cdot 0.007}}{1.293\,\frac{kg}{m^3}\times 1\frac{J/kg}{Gy}} \quad . \qquad (53)$$

$$= 2.45\times 10^{-10}\cdot C\cdot \bar{E}\cdot e^{-\mu\cdot 0.007}\ \frac{Gy}{h}$$

Note that Eq. (53) can be modified for attenuation though clothing by multiplying by an exponential factor $e^{-\mu * t}$, where μ represents the beta-attenuation coefficient for the material and t is the thickness of the material.

5.1.2 Gamma radiation

Gamma rays are neutrally charged quanta of energy (photons). The principal interaction mechanisms for photons, namely the photoelectric effect, Compton scattering, and pair production were introduced in Chapter 3. In addition, photon cross sections and the Klein-Nishina formula for scattering, as well as interaction cross sections, were introduced in Chapter 3.

Some ways in which external neutral particles such as gamma radiation deliver dose are depicted in Figure 47. The exposure can originate from point, area, volume, or distributed source regions.

5.1.2.1 Point source - specific gamma constant (Γ)

A point source may be considered as small with respect to the target (vanishingly small compared to the target). Consider the energy absorbed per unit mass of air at a specified distance from a point source, as given by Eq. (54):

$$\frac{f_i\ (\gamma/t)\times E_i\,(MeV/\gamma)\times 1.6\times 10^{-6}\,(erg/MeV)\times 3.7\times 10^{10}\,(dps/Ci)\times 3600\ (s/h)\times \mu\ (m^{-1})}{4\times\pi\times(1m)^2\times\rho\ (kg/m^3)\times 87.7\ (erg/g/R)}, \qquad (54)$$

where

f_i is the fraction of transformations that yield a photon with energy E_i
μ is the linear energy absorption coefficient for dry air at density 1.293 kg/m^3 (m^{-1}).

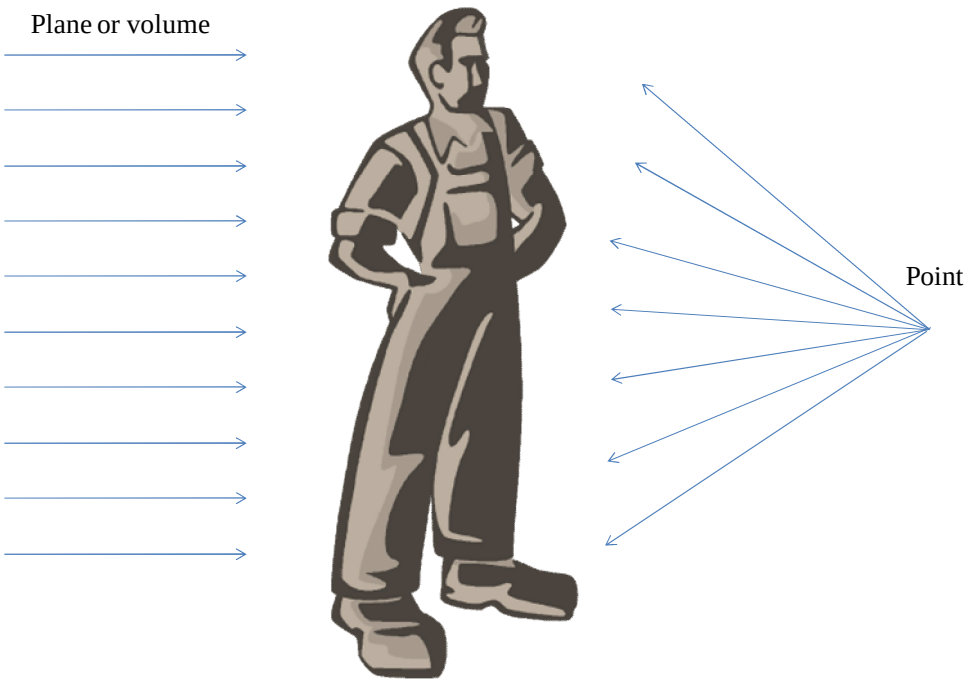

Figure 47 Neutral-particle radiation dose-delivery geometries

By expressing the linear energy-absorption coefficient for air as a constant (which is a good approximation over a large energy range of interest for most radionuclides, as shown in Figure 48), the terms in Eq. (54) can be reduced to a single conversion constant of 0.5, as shown in Eq. (55):

$$\frac{1.6\times10^{-6}\,\frac{\text{erg}}{\text{MeV}}\times3.7\times10^{10}\,\frac{\text{Bq}}{\text{Ci}}\times0.0035\ \text{m}^{-1}\times3600\,\frac{\text{s}}{\text{h}}}{\left[4\times\pi\times(1\ \text{m})^2\right]\times1293\,\frac{\text{g}}{\text{m}^3}\times87.7\,\frac{\text{erg}/\text{g}}{\text{R}}}\simeq0.5\,\frac{R\cdot m^2}{Ci\cdot h\cdot MeV}, \tag{55}$$

which yields the specific gamma constant (Γ) in conventional units as Eq. (56) or in SI units as Eq. (57). The importance of the specific gamma constant in conventional units is that, from an operational perspective, it is much easier to memorize compared to the SI unit form:

$$\Gamma=0.5\sum_i f_i\times E_i\ \ \frac{R\cdot m^2}{Ci\cdot h} \tag{56}$$

$$\Gamma=1.24\times10^{-7}\sum_i f_i\times E_i\ \ \frac{Sv\cdot m^2}{MBq\cdot h}\ . \tag{57}$$

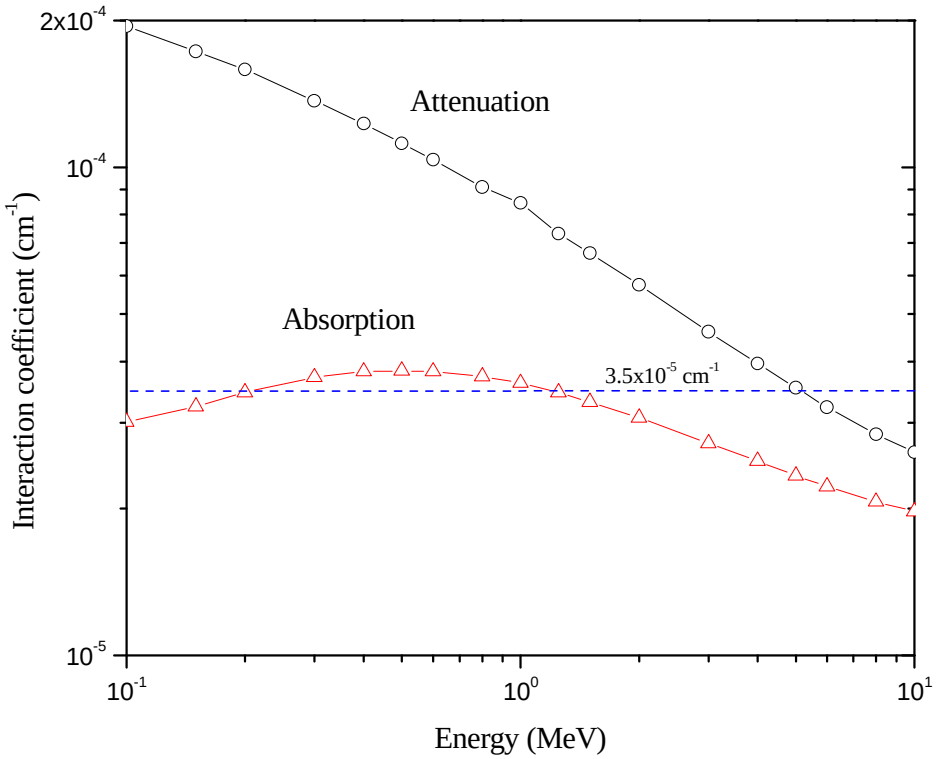

Figure 48 Linear interaction coefficients for dry air

Use of the specific gamma constant (Γ) to determine dose equivalent rate from a point source is straightforward, as shown in Eq. (58):

$$\dot{H}_p = \frac{\Gamma \cdot A}{d^2} \cdot w_r \left(\frac{R \text{ or } Sv}{h} \right), \tag{58}$$

where

$\dot{H}_p$ is the dose equivalent rate (Rem or Sv per hour, depending upon which form of Γ is used) at point p (distance d away from the source)
A is the source activity (Ci or MBq, depending upon which form of Γ is used)
d is the distance from source to receptor (m)
w_r is the radiation weighting factor (=1 for gamma rays).

5.1.2.2 Line source

A line source configuration is one that is infinitely thin but of finite length, such as a pipe carrying contaminated radioactive liquid. The geometry of such a system is described by Figure 49. The receptor is at point *p* at distance *h* from the line, at an arbitrary distance along the line.

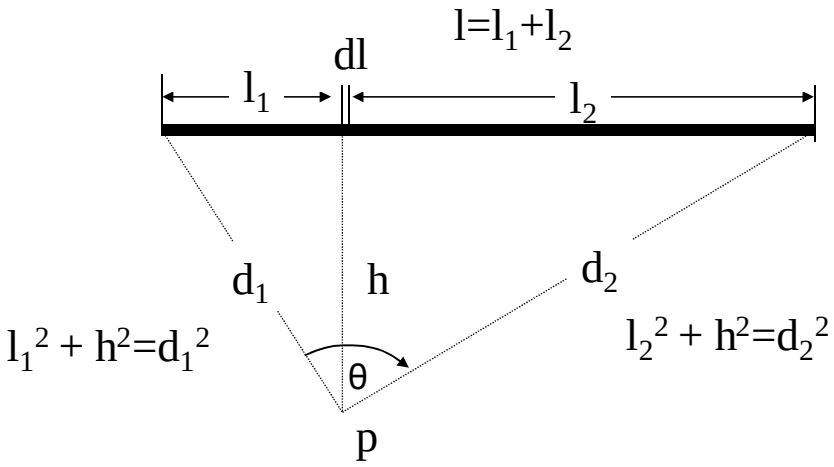

Figure 49 Line-source geometry

For a linear concentration of radioactivity C_i (Ci/m or MBq/m), the dose rate at point *p* can be expressed as a function of the point source-specific gamma constant as defined in Section 5.1.2.1, using Eq. (59):

$$\dot{H}_p = \frac{\Gamma C_l}{h}\left(\tan^{-1}\frac{l_1}{h} + \tan^{-1}\frac{l_2}{h}\right)$$

$$or \qquad , \qquad (59)$$

$$\dot{H}_p = \frac{\Gamma C_l \theta}{h}$$

where

$\dot{H}_p$ is the dose rate (R/h or Sv/h, consistent with Γ)

l_1, l_2, and h are defined in Figure 49 (m)

θ is the opening angle (radians)

Γ is the specific gamma constant for a point source (conventional or SI units)

C_l is the linear activity (Ci m^{-1} or MBq m^{-1}, consistent with Γ).

5.1.2.3 Planar source

A planar source in this case is represented as a disk source which is infinitely thin with area πr^2. This type of geometry could arise, for example, from a spill of radioactive material and is represented by Figure 50.

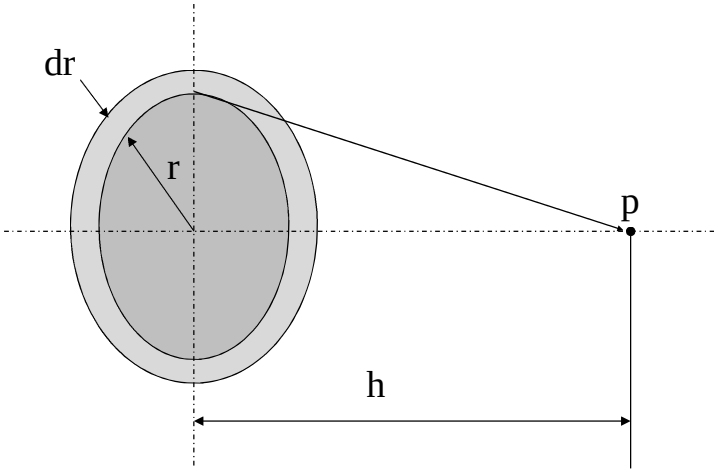

Figure 50 Planar source geometry

For an areal concentration of radioactivity C_i (Ci m^{-2} or MBq m^{-2}), the dose rate at point *p* can be expressed as a function of the point source-specific gamma constant as defined in Section 5.1.2.1, using Eq. (60):

$$\dot{H}_p = \pi \, \Gamma C_a \ln\left(\frac{r^2 + h^2}{h^2}\right), \tag{60}$$

where

$\dot{H}_p$ is the dose rate (R/h or Sv/h, consistent with Γ)

r and *h* are defined in Figure 50 (m)

Γ is the specific gamma constant for a point source (conventional or SI units)

C_a is the areal activity (Ci m^{-2} or MBq m^{-2}, consistent with Γ).

5.1.2.4 Volume source

A volume source is one that has finite dimensions, such as a drum containing radioactive material. The geometry of a cylindrical system used as an example is described by Figure 51. The receptor is at point *p* on the central axis at distance *h* from the surface of the volume.

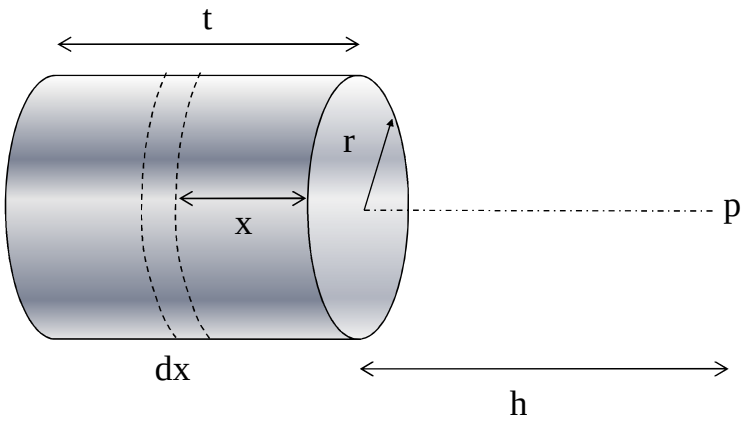

Figure 51 Volume-source geometry

For a uniform volume concentration of radioactivity C_v (Ci m^{-3} or MBq m^{-3}), the dose rate at point *p* can be expressed as a function of the point source-specific gamma constant as defined in Section 5.1.2.1, using Eq. (61):

$$\dot{H}_p = \pi\,\Gamma\frac{C_v}{\mu}\left(1-e^{-\mu\cdot t}\right)\ln\left(\frac{r^2+h^2}{h^2}\right), \tag{61}$$

where

$\dot{H}_p$ is the dose rate (R/h or Sv/h, consistent with Γ)

r, *t,* and *h* are defined in Figure 51 (m)

Γ is the specific gamma constant for a point source (conventional or SI units)

C_v is the volume activity (Ci m^{-3} or MBq m^{-3}, consistent with Γ)

μ is the linear absorption coefficient of the volume source material (cm^{-1}).

It is possible to extend this volume treatment to other simple geometries in a similar manner.

5.1.3 Neutrons

Neutrons can be classified in a variety of ways. The most common way to classify neutrons is by kinetic energy. Thermal neutrons and neutrons in thermal equilibrium with their environment are distributed according to a Maxwell-Boltzmann distribution. Their most probable energy is 0.025 eV at 20°C. At higher energies from 0.01–0.1 MeV, neutrons may be classified as slow, intermediate, or resonance. Fast neutrons have energies from a few MeV up to approximately 20 MeV, and relativistic neutrons have still higher energies. Neutrons are uncharged (neutral) like photons and can travel appreciable distances without interacting. Neutrons are attenuated exponentially under good geometry and do not interact appreciably with electron fields. Neutrons collide with atomic nuclei in elastic and inelastic collisions. With elastic collisions, the total energy is conserved, and the energy lost by the neutron is equal to the total energy of recoil of the nucleus. With inelastic collisions, the nucleus absorbs some of the energy and is left in an excited state, which upon de-excitation can yield neutrons, gammas, and light ions. Fast neutrons undergo a series of primarily elastic scattering reactions, being slowed down in a process called moderation. As neutron energy decreases, scattering continues, but the probability of capture by another nucleus increases. If the neutron reaches thermal energies, it will randomly move around until absorbed by a nucleus.

5.1.3.1 Neutron reactions

A number of reactions important to neutron interactions are listed in Table 14. Some of these reactions are important from a dose-to-tissue perspective, whereas some are important for considering neutron detection.

Table 14 Important neutron reactions

Reaction	Typical use
$^{1}H(n,\gamma)^{2}H$	Dose to tissue
$^{3}He(n,p)^{3}H$	Neutron proportional counters
$^{6}Li(n,T)^{4}He$	Thermal neutron detection
$^{10}B(n,\alpha)^{7}Li$	
$^{14}N(n,p)^{14}C$	Dose to tissue
$^{23}Na(n,\gamma)^{24}Na$	Blood dosimetry
$^{40}Ar(n,\gamma)^{41}Ar$	Internal dosimetry
$^{32}S(n,p)^{32}P$	Neutron spectroscopy
$^{113}Cd(n,\gamma)^{114}Cd$	
$^{115}In(n,\gamma)^{116m}In$	
$^{197}Au(n,\gamma)^{198}Au$	
$^{235}U(n,f)$	Thermal neutron detection

Besides the interactions presented in Table 14, neutron activation of materials can also be a significant source of gamma dose in nuclear plants. Neutron activation is the production of a radioactive isotope by absorption of a neutron and is governed by Eq. (62):

$$\frac{dN}{dt} = \varphi \sigma n - \lambda N$$

$$or$$

$$\lambda N = \varphi \sigma n \left(1 - e^{-\lambda t}\right) \equiv A \quad , \tag{62}$$

where

A is the activity of the radioisotope produced through neutron activation (Bq)
ϕ is the neutron flux (n cm^{-2} s^{-1})
σ is the activation cross section (cm^2)
λ is the decay constant of the produced isotope (s^{-1})
N is the number of produced atoms
n is the number of target atoms.

Although the radioisotopes produced are generally a gamma radiation hazard, activation is an important reaction with respect to this source of gamma (or beta) dose.

5.1.3.2 Neutron dosimetry

The absorbed dose from neutrons, from a biological dose perspective, involves primarily neutron interaction with tissue elements H, C, O, and N and the dose resulting from these interactions. Neutron interactions produce recoil protons and heavy charged particles with short ranges that deposit their energy locally in tissue. Dosimetry estimates may be made empirically be examining separately (i) fast neutrons and (ii) thermal neutrons.

I. Fast neutrons

The dose rate from fast neutrons in a material made up of a number of elements can be estimated using Eq. (63):

$$\dot{D}_n(E) = \frac{\varphi(E) \cdot E \cdot \sum_i N_i \sigma_i f_i}{1 \frac{J}{kg} \Big/ Gy} \quad \left(\frac{Gy}{s} \right), \tag{63}$$

where

$\phi(E)$ is the fast neutron flux (n $cm^{-2}s^{-1}$)

E is the neutron energy (Joules; 1.6E-19 Joules per eV)

N_i is the number of atoms per kg for the i^{th} material element

σ_i is the scattering cross section (cm^2) for the i^{th} material element

f_i is the fractional energy transferred from neutrons for the i^{th} material element, given by $f_i = \frac{2M_i}{(M_i+1)^2}$, where M_i is the atomic mass of the i^{th} material element. f_i represents the average fraction of neutron energy transferred in an elastic collision assuming isotropic scattering.

The fast neutron dose can be calculated as a function of different energy groups (corresponding to the neutron energy spectrum) to determine the total fast neutron dose.

II. Thermal neutrons

The most important specific thermal neutron reactions for tissue dosimetry (hydrogen and carbon) are described below.

Hydrogen capture - $^1H(n,\gamma)^2H$

This is an example of a radiative capture process. The neutron is absorbed, followed by immediate emission of a gamma photon, ${}_0^1n + {}_1^1H \rightarrow {}_1^2H + {}_0^0\gamma$, with photon energy 2.22 MeV. This reaction is of significance when tissue is exposed to thermal neutrons. This reaction yields a uniformly distributed gamma-emitting radioisotope, and therefore initially one must solve for the specific activity (A_s) of this source using Eq. (64)

$$A_s = \varphi N_H \sigma_H \quad \left(\frac{Bq}{kg} \right), \tag{64}$$

where

ϕ is the thermal neutron flux (n $cm^{-2}s^{-1}$)

N_H is the number of hydrogen atoms per kg of tissue (5.98×10^{25} atoms/kg)
σ_H is the absorption cross section for hydrogen (0.33×10^{-24} cm^2).

Knowing the specific activity, it is possible to solve for the internal dose using approaches presented in Section 6.

Nitrogen capture - $^{14}N(n,p)^{14}C$

This reaction has a large thermal cross section of 1.78 barns and an energy release of 0.626 MeV in tissue. This is a significant contributor to dose when tissue is irradiated by neutrons because the proton and the nucleus recoil have limited range and the energy is deposited locally. The dose rate due to the energy released by charged particles for this reaction can be determined by Eq. (65):

$$\dot{D}_{np} = \frac{\varphi \cdot N_N \cdot \sigma_N \cdot Q \cdot (1.6 \times 10^{-13} \frac{J}{MeV})}{1 \frac{J}{kg} \Big/ Gy} \quad \left(\frac{Gy}{s} \right), \tag{65}$$

where

ϕ is the thermal neutron flux (n cm^{-2}s^{-1})
N_N is the number of nitrogen atoms per kg of tissue (1.49×10^{24} atoms/kg)
σ_N is the absorption cross section for nitrogen (1.75×10^{-24} cm^2)
Q is the energy release in tissue (0.626 MeV).

5.2 External Hazards

A variety of potential external radiation hazards are associated with nuclear technology. When a reactor is under power, a variety of areas inside containment can produce significant neutron and gamma radiation fields. Generally speaking, these areas are well characterized, and administrative measures are put in place to minimize any hazard from these fields. When the reactor is on power, there are hazards from fission neutrons, fission prompt gamma rays, and neutron-capture gamma rays, as well as fission-product gamma rays, activation gamma rays, and photoneutrons. When the nuclear chain reaction is halted, the direct neutron component of the hazard rapidly reduces to an insignificant proportion. The hazards that remain are therefore fission-product gamma rays, activation gamma rays, and to a lesser extent, photoneutrons. In an ideal situation, the fission products produced in the fuel would remain in the fuel cladding. However, fuel failure does occur and contaminates water systems, which can lead to deposition of fission products in piping and associated systems. In addition, activation products can migrate away from their original locations due to corrosion processes.

The most common operational hazards from external radiation in a nuclear plant are due to:

(a) Fission-product activity in the core (present regardless of the power condition of the core)
(b) Activation products in the core and surrounding material (including water)
(c) Fission products that migrate and deposit at various locations in the reactor
(d) Activation products that migrate and deposit around various reactor systems

(e) Contamination, in liquid or aerosol form, due to fission and activation products that become inadvertently uncontained (which also may pose a significant external beta hazard and internal alpha/beta hazard if inhaled).

The possible sources of the radiation that may contribute to external radiation dose while a CANDU reactor is operating are depicted in Figure 52. The simplified drawing depicts the biological shield (this shield also encompasses the thermal shield and reflector region), the core (where the fission reaction takes place), and the primary heat transport (PHT) system that circulates the hot pressurized water directly to the steam generators. The reactor operates as a "closed primary circuit", and therefore radioisotopes generated as a result of fission, activation, or other nuclear processes may be circulating in the primary heat transport (PHT) system. The primary circuit is physically separated from the secondary steam-generation circuit at the steam generators. In theory, there should be no sources that migrate from the primary to the secondary side in a CANDU, and therefore no radioactivity at the turbines. Referring to Figure 52, the potential sources may be described as follows (after [Goldstein1962]):

- Prompt fission neutrons – emitted within first few μs after fission
- Delayed fission neutrons – emitted from excited nuclei up to a few minutes after fission
- Activation neutrons – emitted from nuclear reaction products
- Photoneutrons – produced through threshold (γ,n) reactions
- Prompt-fission gammas – emitted in coincidence with fission (< μs)
- Short-period fission gammas – emitted by fission products within a few minutes (~10 min) after fission
- Long-period fission gammas - emitted by fission products after ~10 minutes
- Capture gammas – emitted in (n,γ) reactions
- Inelastic-scatter gammas – emitted from excited nuclei after neutron inelastic scattering
- Reaction-product gammas – emitted from products of charged-particle reactions induced by neutrons
- Activation-product gammas – emitted from radioactive products of nuclear reactions
- Annihilation photons – emitted from positron annihilation due to certain activation or fission products
- Bremsstrahlung photons – emitted from decay electrons (beta particles) slowing down in material.

Note that not all the listed sources are of equal importance, and often their importance is dictated by their physical location in the reactor with respect to receptors (workers). In addition, the importance of sources is affected by the state of the reactor. On shutdown, some of the sources disappear, and the relative importance of the remaining sources changes. There is residual fission power in a CANDU core after shutdown, and therefore there are still prompt fission gammas and neutrons, as well as capture gammas produced in the core. However, they are produced at levels several orders of magnitude lower than during operation. Possible external radiation sources ~10 minutes after shutdown are depicted in Figure 53.

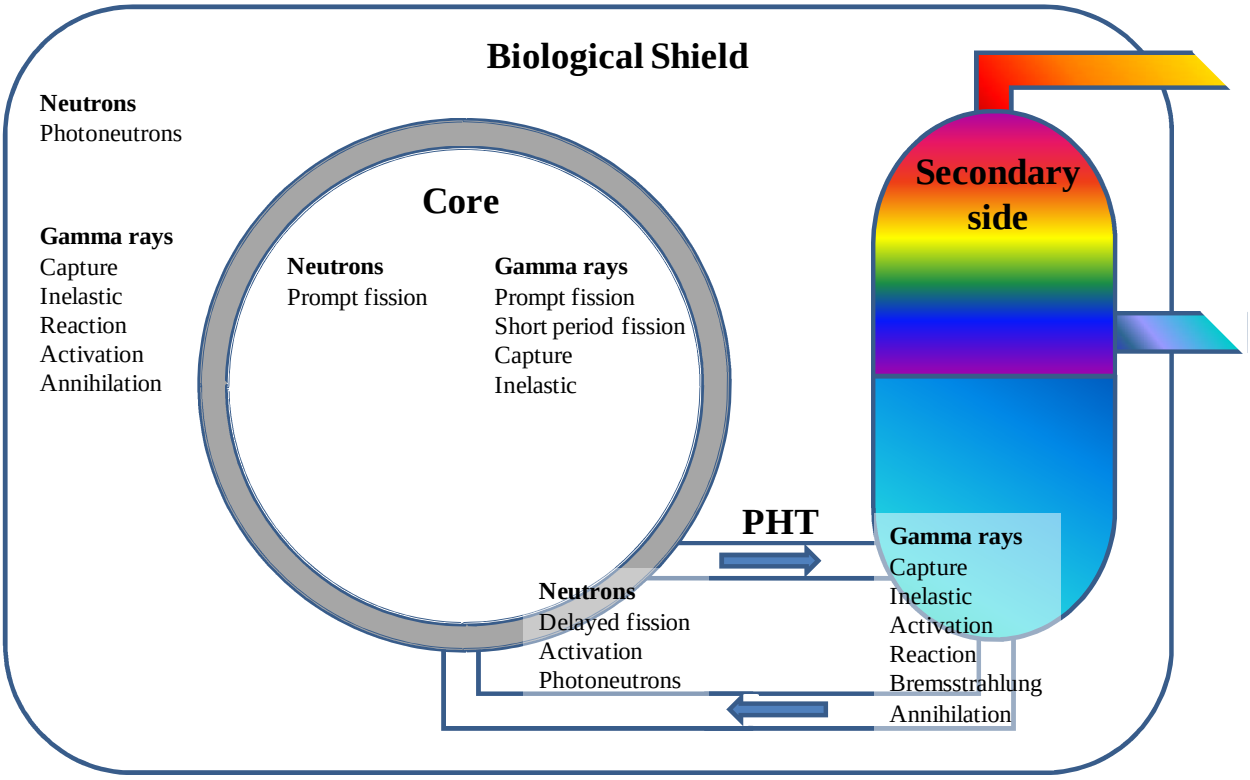

Figure 52 Potential external exposure sources during reactor operation

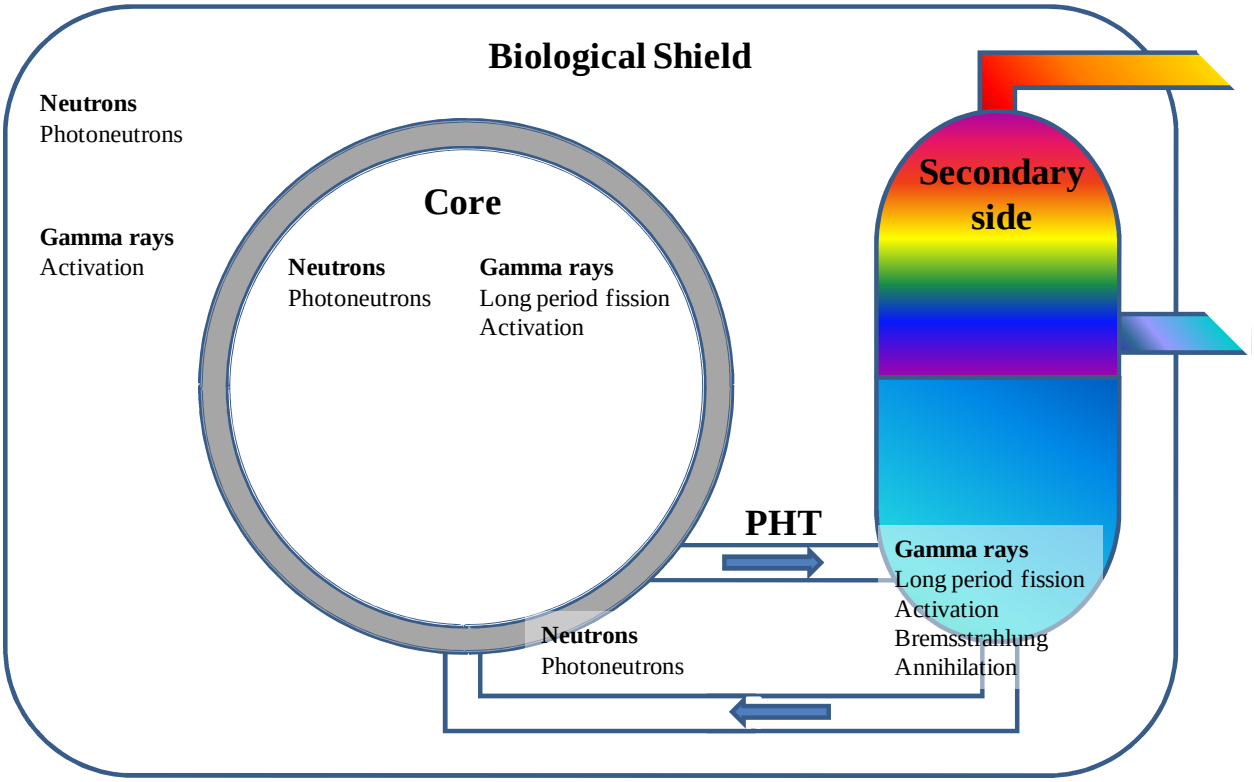

Figure 53 Potential external exposure sources after shutdown (> 10 min)

Some examples of external hazards in a nuclear reactor environment include (but are not limited to) the following:

- Mixed neutron/gamma fields from normal operation (under power or shut down) at various locations around the reactor
- Contaminated coolant in a pipe (circulating or uncirculating)
- Contaminated resin columns
- Fission or activation products that have lodged or attached somewhere unexpectedly in a piping system
- Leakage from a contaminated water system
- Outage activities that may involve opening piping or pump systems, welding and grinding activities on potentially contaminated systems, etc.

Protection against these external hazards includes proper work planning (discussed in Section 7) along with use of the ALARA principle and shielding (discussed in Section 5.3).

5.3 External Protection (Radiation Shielding)

The essential principle of external radiation protection is embedded within the so-called ALARA (as low as reasonably achievable) principle. The ALARA principle was formalized by the ICRP in their 1977 recommendations [ICRP1977] and may be considered as the fundamental philosophy of radiation protection that suggests that no practice dealing with radioactive material should be adopted unless there is a net benefit arising from it. If radiation is to be used, all efforts must be made to ensure that the dose is the lowest absolutely required for the given task. The complete text of the ALARA principle clarifies the context of applicability and has been paraphrased as [ICRP1977] making every reasonable effort to maintain exposures to radiation as far below the dose limits **as is practical**, consistent with the purpose for which the activity is undertaken, taking into account (i) the state of technology, (ii) the economics of improvements in relation to the state of technology, (iii) the economics of improvements in relation to benefits to public health and safety, (iv) other societal and socioeconomic considerations, and (v) relationships to use of nuclear energy and radioactive materials in the public interest.

The typical radiation hazards for CANDU reactor operations are depicted in Figure 54. In applied external radiation protection for CANDU reactors, it is normal to consider gamma, neutron, and beta radiation. Alpha radiation, although a great concern for inhalation dose (considered in Section 6), is of no concern as an external dose because the radiation cannot penetrate the external dead layer of tissue and certainly not any clothing that a person may be wearing. Beta radiation (discussed in Section 5.1.1) can be of concern externally if a worker is not protected and is in contact with a source, submerged in a plume, or gets a “hot” particle on unprotected skin. However, in this type of scenario, there are usually corresponding gamma emitters that require shielding, and therefore the beta particles are generally well shielded as a result. For other applications like nuclear medicine, low atomic number (low-Z) materials can provide effective shielding from beta radiation (while simultaneously yielding less bremsstrahlung radiation compared to high-Z shields). It is common to use Plexiglas™ shielding for beta-emitting radioisotopes. Therefore, for normal shielding, the focus is on gamma and neutron radiation.

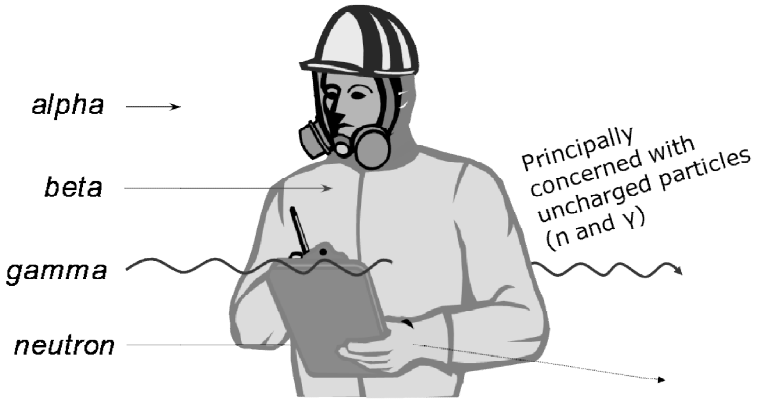

Figure 54 External hazards

The reduction of doses to ALARA for external radiation is primarily accomplished using three principal concepts of radiation protection: shielding, distance, and time. Simply stated, shielding is placing something between the source and receptor that blocks (or attenuates) radiation, distance is keeping space between the source and the receptor, and time is spending as little time around the source as required to reduce exposure. In addition, it is common to use the effect of radioactive decay for short-lived radioisotopes as part of the shielding strategy (for example, delaying entry into an area after shutdown to allow time for radioactive decay).

Time is an obvious dose parameter because dose is accumulated over time and is delivered as a function of time from an external source (dose rate). Therefore, less time = less dose. The role of distance is readily understood by considering the inverse-square law (which is also important in shielding discussions) depicted in Figure 55.

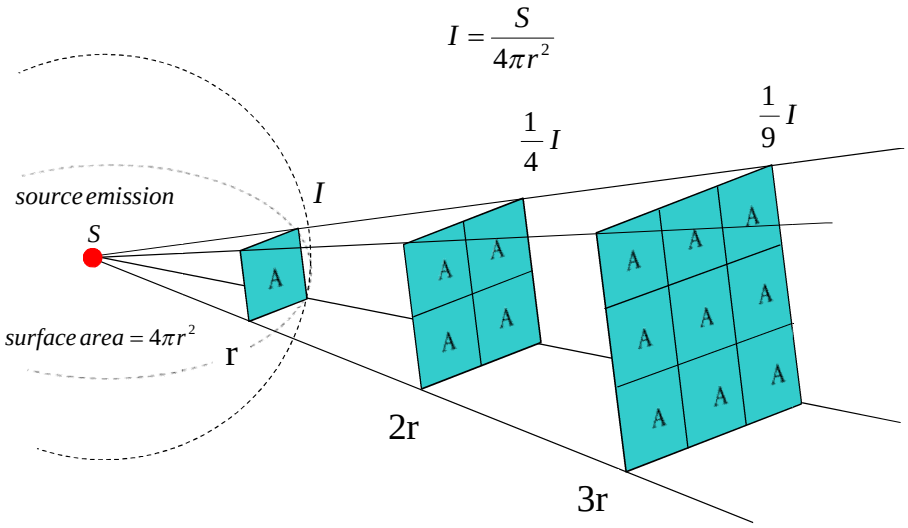

Figure 55 Inverse-square law

The inverse-square law can be readily understood by ray-tracing from a point source outwards and considering the intensity per unit area at the intersection with an arbitrary sphere of radius *r*. For every increase in radius, the intensity decreases by the square of the increased radius because the intensity is propagating outwards across the expanding sphere. For example, twice the radius leads to one-fourth the intensity, three times the radius, one-ninth the intensity, four times the radius, one-sixteenth the intensity, and so on. Because dose is directly proportional to

radiation intensity, this demonstrates that dramatic decreases in dose can be achieved with moderate changes in distance from the source, particularly close to the source.

As discussed earlier, shielding is the act of placing a material between the source of radiation and the receptor, which has the effect of reducing the amount of radiation impinging on the receptor. In radiation protection, shielding is often called biological shielding because its primary purpose is to reduce the biological dose to humans. As such, shielding materials are critically important to the effectiveness of shields. The following sections consider shielding for gamma and neutron radiation.

5.3.1 Gamma shielding

In Section 5.1.2, the principles of gamma dosimetry were discussed. The relationship to determine intensity (fluence, flux, or dose) of radiation after passage through a shield is given by Eq. (66):

$$I = I_0 e^{-\mu x}, \tag{66}$$

where

I is the intensity of the radiation after passing through the shield (fluence, flux, or dose units)
I_o is the intensity of the radiation before passing through the shield (same units as I)
μ is the linear attenuation coefficient for that material (cm^{-1})
x is the thickness of the shield (cm).

Taking into account both the distance from the source and receptor and the influence of the shielding material, the intensity at a point *r* units away from a source shielded by a thickness *x* of material is given by Eq. (67):

$$I = \frac{I_0 e^{-\mu x}}{4\pi r^2}. \tag{67}$$

The physical conditions that describe this type of attenuation are that (i) the radiation is a narrow beam interacting with the shield and (ii) the shield is thin with respect to the mean free path (1/μ) of the gamma radiation in the shield. This scenario is depicted in Figure 56. In this case, the radiation, depicted by the dashed lines, scatters away from the receptor.

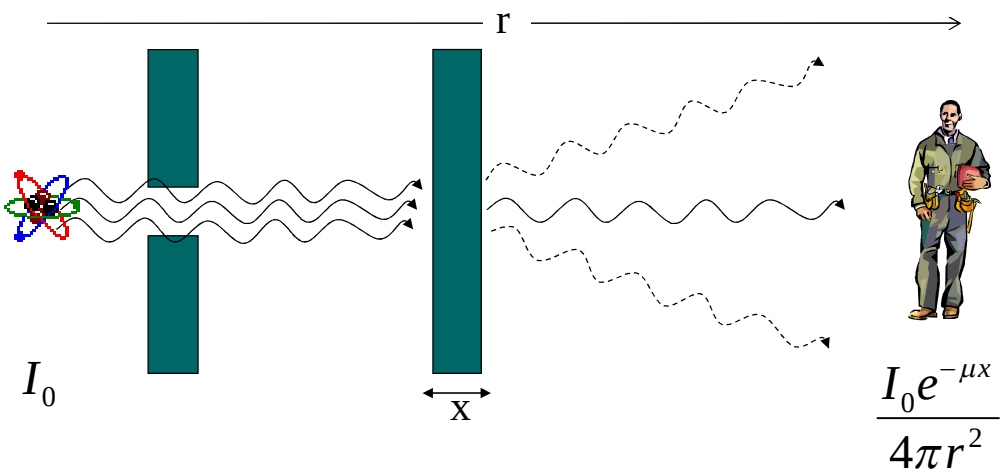

Figure 56 Narrow-beam or thin-shield attenuation

5.3.1.1 Buildup

Although approximation of Eq. (66). may be valid in some cases, there are a number of situations in which the radiation source may be considered a broad beam with respect to the shield or the shield may be thick with respect to the mean free path of radiation in the shield. This case is depicted in Figure 57. It has been observed that in this scenario, some of the radiation that would have scattered away from the receptor actually scatters towards the receptor and contributes additional intensity to the receptor location that is not predicted by Eq. (66). In this case, a factor is used to modify the standard attenuation equation to account for the scattering. This factor is known as the buildup factor (B) and is dimensionless. The intensity equation therefore becomes Eq. (68):

$$I = \frac{I_0 B e^{-\mu x}}{4\pi r^2} \quad . \tag{68}$$

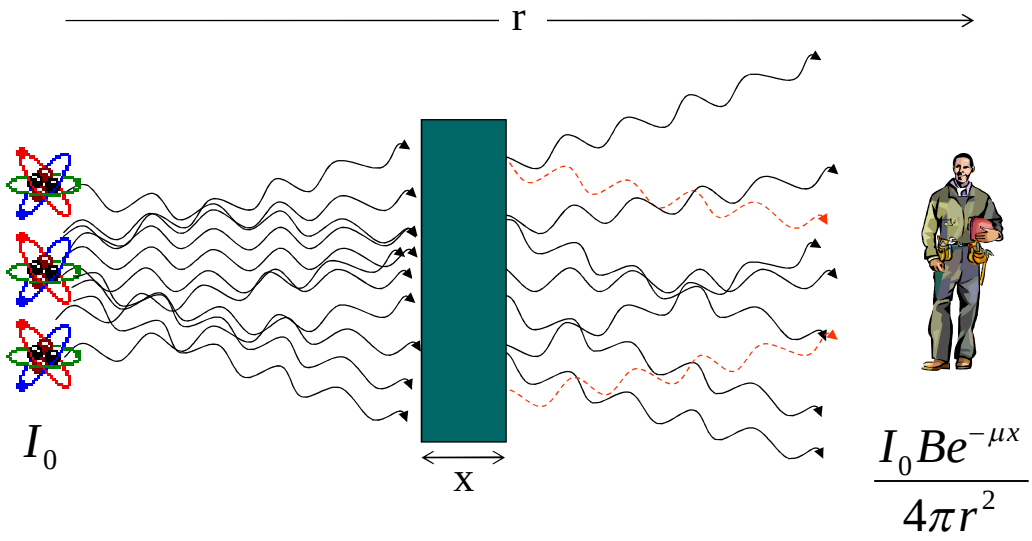

Figure 57 Broad-beam or thick-shield attenuation

The buildup factor is both a function of energy and the shield material (μx, denoted as the optical length, relaxation length, or number of mean free paths) and is given as Eq. (69):

$$B(E,\mu x) \equiv \frac{I_0(E,\mu x)+I_s(E,\mu x)}{I_0(E,\mu x)} = 1+\frac{I_s(E,\mu x)}{I_0(E,\mu x)}, \quad (69)$$

where

B is the buildup factor (dimensionless)
I_0 is the non-collided intensity (fluence, dose, or dose rate) at the receptor
I_s is the scattered (collided) radiation contribution at the receptor.

The buildup factor is the ratio of the radiation intensity, including both primary and scattered radiation, at any point in a beam to the intensity of primary radiation only at that point. The buildup factor accounts for scattering into an area due to poor (broad-beam) geometries and is a function of shield material and shield thickness (*x*). Examining Eq. (69) leads to the conclusion that B is always ≥ 1. Values for B have been experimentally determined; however, the bulk of data available are from calculations, and many evaluations are presented in tables or in a graphical format.

Two common ways of using the buildup factor for solving a shielding problem are: (i) knowing the source and shield, calculate the intensity, and (ii) knowing the desired intensity at the receptor point, determine the amount of shielding required.

Calculating intensity

The steps in calculating intensity (fluence, dose, or dose rate) are given as follows:

1. Look up B for:
   - Material of interest (e.g., lead, iron, etc.)
   - Energy of photons (e.g., 1 MeV)
   - Relaxation length of shield (μx) – Relaxation length is the thickness of shield material that will attenuate a narrow beam to 1/e of its original value.
2. Solve equation: $\mathbf{I=I_0*B*e^{-ux}}$ for fluence, dose, or dose rate.

Calculating required shield thickness

The calculation of the thickness of a shield required for a given desired fluence, dose, or dose rate is difficult because the equation must be solved simultaneously for two variables, *B* and *x*, which can be solved for only by iteration. The steps in calculating the required shielding thickness are the following:

1. Define the target exposure I (fluence, dose, dose rate)
2. Solve the narrow-beam equation $\mathbf{I=I_0*e^{-ux}}$ to obtain an initial shielding thickness *x*. This thickness will be too small.
3. Add one half-value of thickness to *x* ($\mu = (\ln 2) / x_{1/2} \rightarrow x_{1/2} = (\ln 2) / \mu$)
   - (*new* x) = ($x + x_{1/2}$)
4. Look up B for material, energy, (*new* x)* μ.
5. Solve $\mathbf{I=I_0*B*e^{-ux}}$ with new values
   - Check: Is the calculated value of I close to the target I?
6. If the calculated I is still too large, then add another $x_{1/2}$ and recalculate (i.e., go to step 3).

Note that the shield may be placed anywhere between the source and the receptor; however, when the shield is placed closer to the source, it provides the greatest solid-angle protection to the receptor. Tabulated buildup factors can be found in a number of sources including, for example, through the American Nuclear Society [ANS1991] and in work by Shultis and Faw [Shultis2000].

A number of numerical techniques can be used to perform gamma-radiation transport calculations suitable for use in shielding estimates, for example, point-kernel techniques, discrete-ordinate techniques, and Monte Carlo simulation. The need for advanced gamma-shielding approaches arises from requirements to simulate complex geometries, multilayer shields, complex materials, complex source terms, and differing cross-section evaluations.

5.3.1.2 Materials

In general, there are three categories of shield materials: (1) natural materials, (2) construction materials, and (3) special materials. Natural materials include air (insofar as it can be a scattering medium), water, and soil, and often shielding design involves making judicious use of these natural materials. Construction materials are a very advantageous design material from an optimization standpoint because they can serve two purposes, as structural and as shield materials. Construction materials may include concrete, steel, wood, plasterboard, and glass. Special materials for gamma shielding are generally high-atomic-number materials such as steel, iron, lead, tungsten, and uranium. The usefulness of a shield for any given application is related to the mass-attenuation coefficient (directly related to the interaction cross section) and the mass density of the material as a function of the photon energy spectrum. A plot of the interaction cross sections of photons in lead is provided in Figure 58. It can be seen that the photoelectric effect is the dominant interaction at energies below approximately 0.2 MeV, Compton scattering is competing or dominant between approximately 0.2 and 4 MeV, and pair production dominates above 4 MeV (further discussion of photon interactions is provided in Chapter 3).

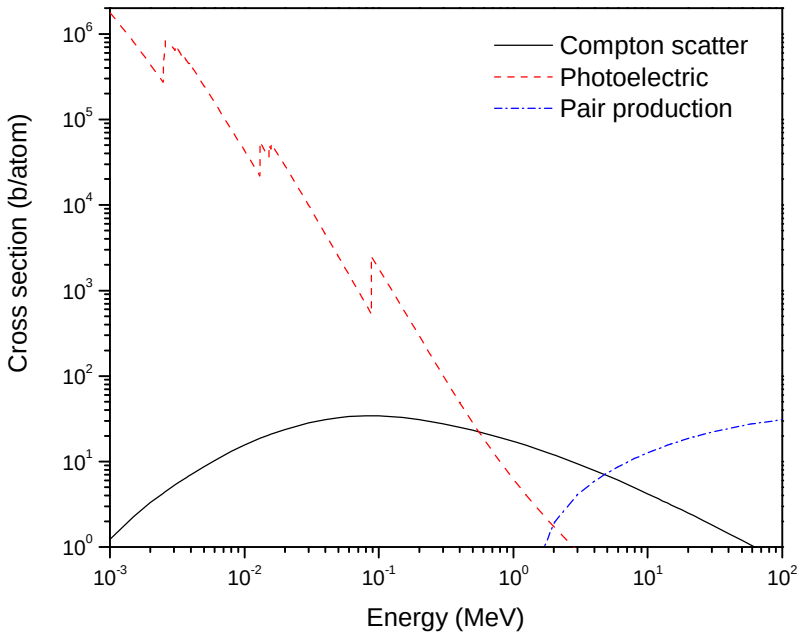

Figure 58 Photon interaction cross sections for lead

Photon total-interaction coefficients for uranium, lead, aluminum, and water are shown in Figure 59. It can be seen that for high-Z materials such as uranium and lead, the interaction mechanisms are very similar. For lower-Z materials such as aluminum and water, the interaction coefficients are significantly lower at most energies. Note that the total interaction coefficients at moderate energies (0.5–3 MeV) are similar for all four materials and that therefore their attenuating power comes primarily from their different mass densities because $\left(\frac{\mu}{\rho}\right)\rho \cdot x$ is the argument of the exponential function when μ is the linear attenuation coefficient.

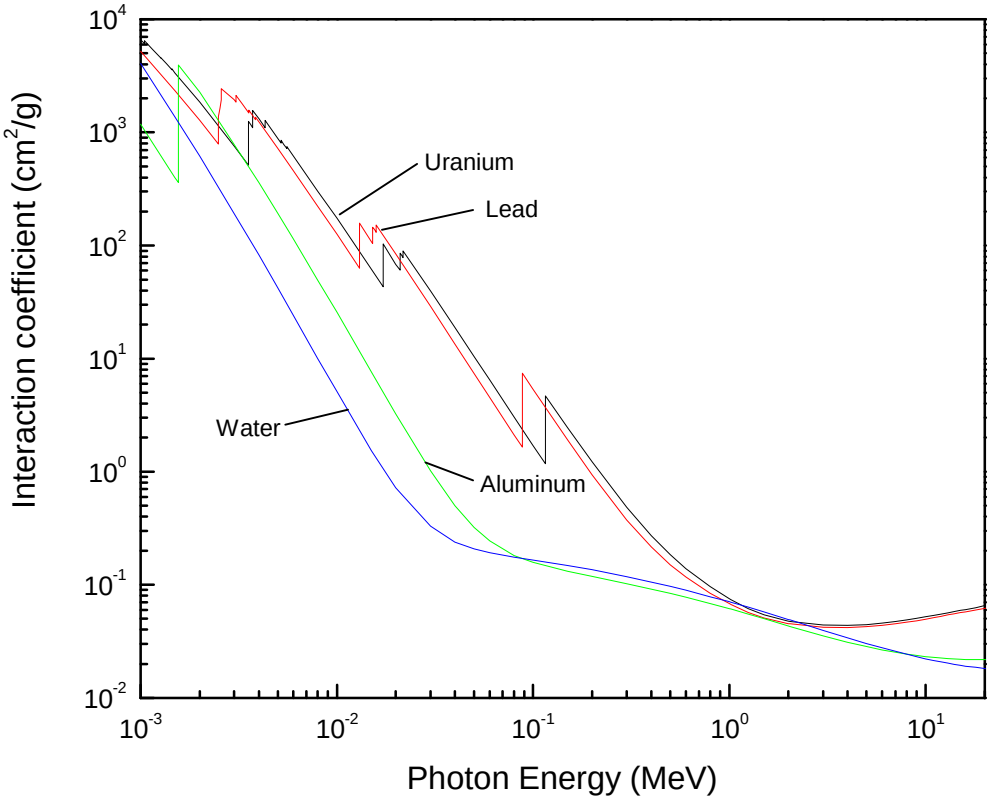

Figure 59 Photon interaction coefficients

5.3.2 Neutron shielding

Compared to the simplified gamma-shielding techniques, fast-neutron shielding is somewhat more complicated. These complications arise from a number of considerations:

- Primary neutrons (usually fast) have a high scattering probability
- Inelastic scattered neutrons produce gamma photons
- Thermal neutrons are produced after slowing down
- Capture gamma photons from (n,*γ*) reactions arise mainly from thermal neutrons
- Secondary neutrons from (n,2n), fission, etc., can be generated
- Neutron production is usually accompanied by primary gamma photons
- Neutron absorption can activate shield material, which in turn emits delayed gamma photons.

Taking all this into account, a simplified approach such as using buildup factors (similar to the approach with photons) is not generally practical for neutron problems because [Shultis2000]:

- Neutrons scatter much more than photons, thereby making B very large
- There is a large buildup of low-energy neutrons because the absorption cross section (σ_a) is small in the slowing-down energy region
- B depends strongly on the
    - Isotopic (material) composition of the shield
    - Physical geometry of the shield
    - Incident neutron-energy spectrum.

Due to the difficulties outlined above, the buildup-factor approach will not be developed for neutrons. However, in addition to the above, because neutrons have radiation-weighting factors that are energy-dependent, the difference between dose conversion factors (response functions) for air kerma, ambient dose, and dose equivalent can be significant, as shown in Figure 60. This can lead to significant differences in dose estimation.

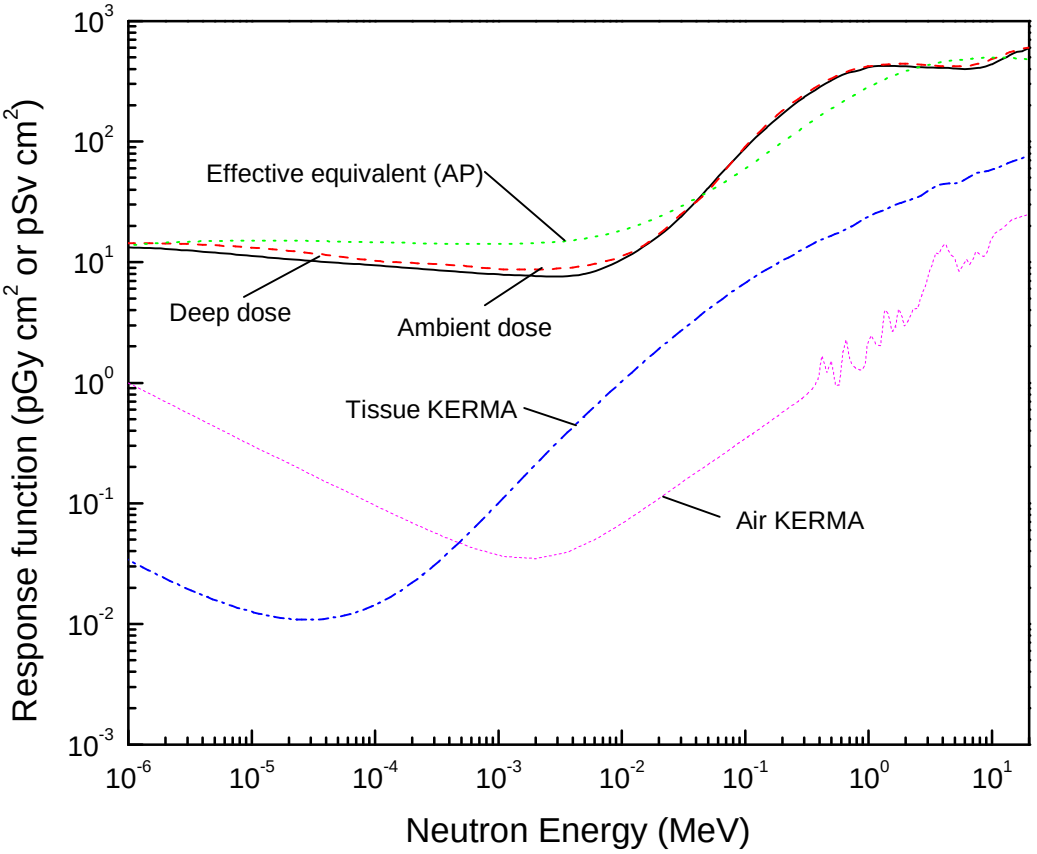

Figure 60 Response functions for neutrons

General considerations with respect to neutron shielding include:

- For biological shields, fast neutrons are of most concern because thermal neutrons are readily absorbed
- Q (w_R) varies with neutron energy and is greatest for 0.1 < E < 2 MeV
- Materials with large σ_a are used to absorb thermal neutrons (e.g., Cd, In, B, Li). However, one must consider the resulting capture gamma radiation

- To stop fast neutrons, they can be thermalized by
    - moderation (low-atomic-number materials) and
    - inelastic scattering (e.g., by iron)

  and then absorbed as thermal neutrons.

Most neutron-shielding analyzes require complex transport calculations. However, a few simplified techniques are available. For example, point kernels can be used for fission neutrons in hydrogenous and non-hydrogenous media, diffusion and removal theories for intermediate energy neutrons, and point kernel techniques for capture gamma photons. In addition, tabulated data for neutron shielding (for example, concrete shields) are often available.

5.3.2.1 Removal techniques

The removal cross section can be used to estimate the flux of fission neutrons in a hydrogenous material (e.g., water) of distance *r* from source to receptor that is shielded by a non-hydrogenous material of thickness *t*. In this case, the flux is given by Eq. (70) [Lamarsh2001]:

$$\begin{aligned} &\phi(r) = S \cdot G(r) \cdot e^{-\Sigma_r t} \\ &where \\ &G(r) = \frac{0.12 \cdot e^{-\Sigma_{r_{water}} r}}{4\pi r^2} \end{aligned}, \tag{70}$$

where

φ is the fission (> 1 MeV) neutron flux at r (n cm^{-2} s^{-1})
S is the source emission rate (n s^{-1})
Σ_r is the macroscopic removal cross section for the shield material (cm^{-1})
Σ_{rwater} is the macroscopic removal cross section for water (0.103 cm^{-1})
t is the thickness of the shield (cm)
r is the distance of the water from source to receptor (cm).

The macroscopic removal cross section can be expressed as a function of the microscopic cross section, as in Eq. (71):

$$\begin{aligned} &\Sigma_r = N\sigma_r \\ &or, \ for\ mixtures \\ &\Sigma_r = \sum_i N_i \sigma_{ri} \end{aligned}. \tag{71}$$

Macroscopic and microscopic cross sections for a variety of elements and mixtures are provided in Table 15 (adapted from [Shultis2005]).

Table 15 Removal cross sections for neutrons

Material	Σ_r (cm^{-1})	σ_r (b)
Aluminum		1.31
Hydrogen		1.00
Deuterium		0.92
Beryllium	0.132	1.07
Boron		1.07
Carbon	0.065	0.81
Oxygen		0.92
Sodium	0.032	1.26
Iron	0.168	1.98
Zirconium	0.101	2.36
Lead	0.118	3.53
Tungsten		3.36
Uranium	0.174	3.60
Zirconium		2.36
Water	0.103	
Paraffin		80.50
Heavy water	0.092	2.76
Concrete (6% water)	0.089	

An approximation for the removal cross section (where $\Sigma_r = \mu_r$) in terms of μ_r/ρ is given by Eq. (72):

$$\frac{\mu_r}{\rho} \simeq 0.206 A^{-1.3} Z^{-0.294} \left(\frac{cm^2}{g} \right). \tag{72}$$

Functionally, the removal technique can be used to estimate the flux at the surface of a shield, and then this flux can be used to estimate the flux at a position outside the shield. Knowing the

fast neutron flux at a receptor point outside the shield, the dose rate can be estimated using Eq. (63). Note that this is a simplified approximation, and a more complex analysis (such as the Monte Carlo technique) is required to determine the intensity accurately.

5.4 Summary

External radioisotope hazards associated with CANDU reactors include hazards directly associated with the nuclear chain reaction (neutron and gamma fields) or indirectly associated as is the case with fission products from fuel (whether intact or defective) and activation products (whether fixed or mobile). Dosimetry can be accomplished from first principles using source activity and geometry or by using dose-conversion factors and fluence measurements.

External hazards in CANDU reactors involve gamma, neutron, and beta fields, and protection of the worker from external hazards is accomplished by adhering to the principles of "as low as reasonably achievable" dose and the concepts of maximum shielding (of source), maximum distance (from source), and minimum time (around source) for any given work function.

6 Internal Radiation Hazards

Internal radiation hazards from alpha, beta, and gamma sources are discussed in this section. Pathways for internal radiation exposure are periodically present within nuclear reactors, and therefore consideration of expected doses and strategies to minimize dose are required. Some historical perspective on internal dosimetry can be found in [Potter2005].

6.1 Internal Pathways and Dosimetry

When a material containing radioisotopes is inhaled, ingested, or absorbed through the skin or a wound, this will lead to an internal dose, as depicted in Figure 61. For an ingestion pathway, typically (1) contamination (for example, a radioactive spill) occurs. If the contamination is touched (whether protected or unprotected), the radioisotope(s) can be either (2a) transferred to the hands or (2b) transferred through a wound if present and directly to the blood. If the hands are contaminated, the radioisotope(s) can be (3) transferred to food or the face directly, from where they can be (4a) transported to the gastrointestinal (GI) tract. Alternately, for an inhalation pathway, if a person is in proximity to airborne radioisotopes, the person may (4b) take in the radiation directly from the plume into the respiratory tract. In both cases, radioisotope(s) will be transported and (5) taken up into the blood and GI tract. From there, they can be transported to target organ(s) and deposit decay energy, or in other words, a dose.

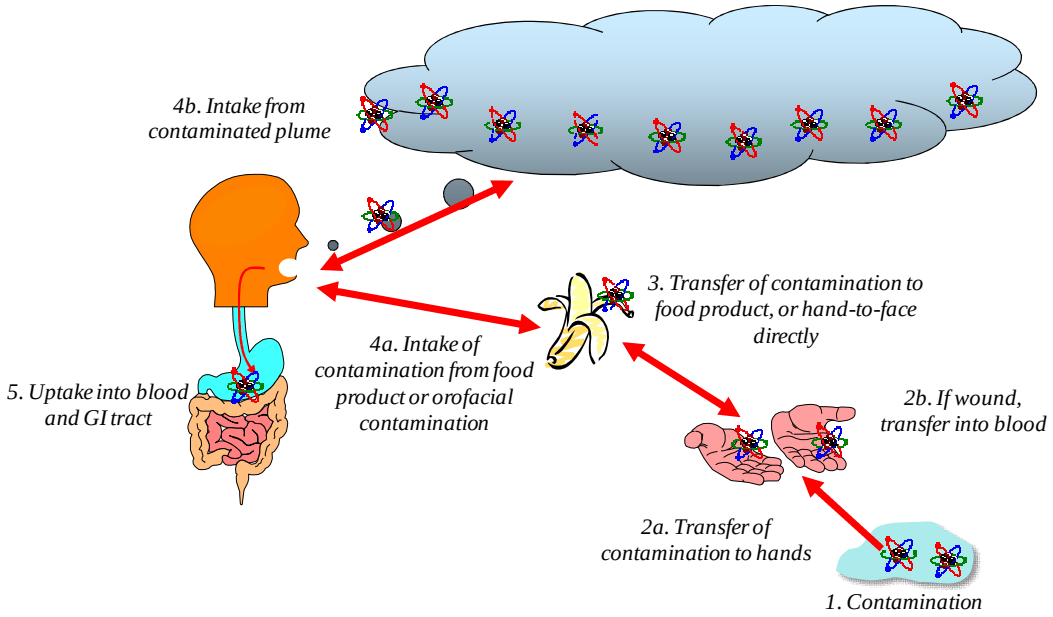

Figure 61 Internal contamination pathways

As was discussed in Section 3.4, the total dose assigned to an individual is the sum of all external and internal exposures. In an industrial environment, the dominant pathway for internal exposure is through inhalation of airborne radioisotopes.

Neutrons are generally not considered when discussing internal dose because the possibility of internalizing a neutron emitter (for example, a spontaneous fission-neutron source) is extremely remote. However, in contrast to external dosimetry, alpha-emitting radioisotopes can be a major contributor to the committed effective dose that a person might receive.

A number of factors affect internal dose, as outlined in Table 16. Note that not only does the type and quantity of radioactive material contribute to dose, but also other factors such as the chemical form of the material and its biological function in the human body.

Table 16 Parameters affecting internal dose

Physical Characteristics	Biological Characteristics
Radionuclide	Metabolic behaviour
Physical half-life	Biological half-life
Chemical & physical form	Chemical toxicity
Emitted radiation	Tissue sensitivity
Intake route	Age of individual
Duration of intake	Individual health
Total intake	Personal habits

The internal dose mechanism can be described as shown in Figure 62. Internal dose is facilitated by basic mechanisms for intake into the body (inhalation, ingestion, or absorption), followed by uptake into extra-cellular fluids for circulation through the body, and finally deposition of radioisotopes into target organs and tissues where decay energy is deposited.

The dominant intake regimes related to transport through the body to excretion are outlined in Figure 63. This is a simple representation of what is called a compartmental model. Dosimetry can be accomplished by solving mathematically for the transport, deposition, and excretion of radioisotopes through “compartments” in the body. Most internal-dosimetry computer codes perform this type of calculation (for example, see [IMBA2010]). Compartmental modelling of internal dose is beyond the scope of this chapter.

Inhalation into the respiratory system can be subdivided into three regions: (i) nasal-pharyngeal (inhalable), (ii) tracheal-bronchial (thoracic), and (iii) pulmonary (respirable, or deep lung). Particle deposition takes place in the three regions with an efficiency based on particle diameter. Generally speaking, particles from submicron to ten microns are considered respirable, whereas larger particles deposit in the upper airway regions. Although upper-airway deposition can lead to dose, the respirable fraction is of primary significance for internal dosimetry. Inhalation leads to deposition in one of the three regions of the respiratory tract, and if deposited in the deep lung, particles can be absorbed into the blood for transport to target organ(s). Inhalation can also lead to a fraction being removed mechanically (by coughing, etc.) from the respiratory tract into the GI tract, where it may be treated as an ingestion and similarly absorbed into the blood for distribution to other organs before excretion. Excretion may occur through urine, feces, sputum, sweat, tears, etc.; however, the dominant modes of biological excretion are through urine or feces. Before radioisotopes are excreted, they will deposit energy in the target tissue(s). The deposition of this energy into a mass of tissue is the definition of absorbed dose.

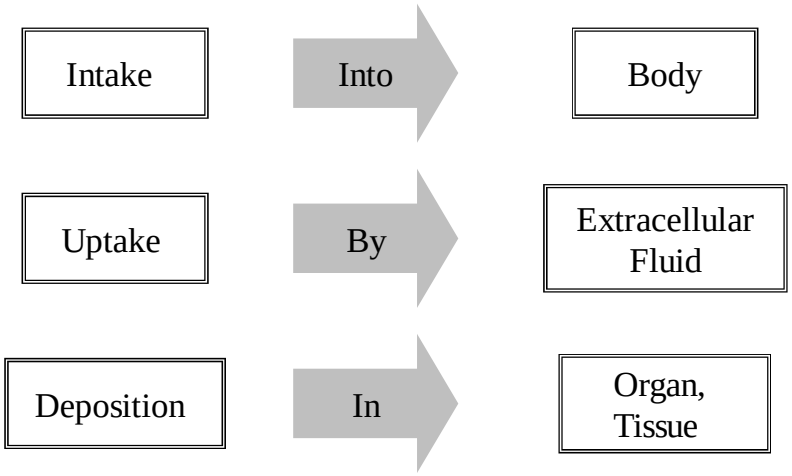

Figure 62 Basic mechanisms for internal dose

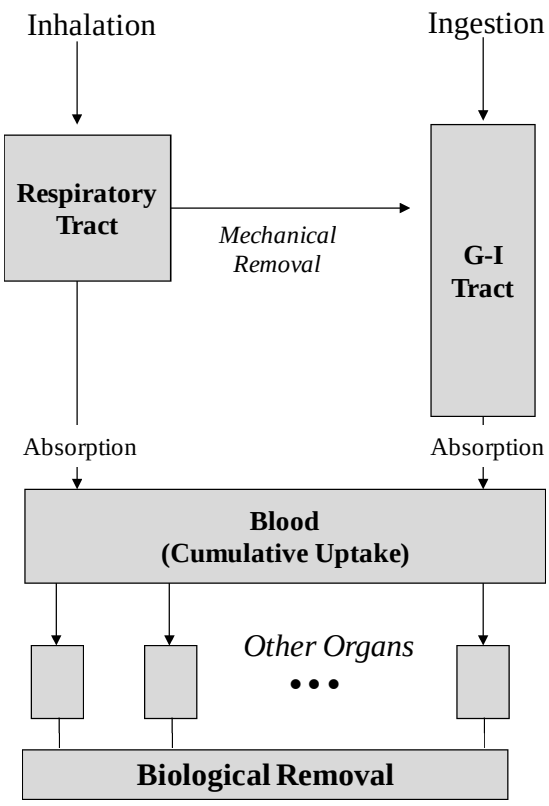

Figure 63 Intake to excretion pathways

The computation of internal dose requires three (3) critical pieces of information:

1. Rate of dose deposition (dose rate in tissue)
2. Elimination of radioactivity from tissue (decay and removal)
3. Total dose "committed" to tissue (committed dose).

In addition, simplified approximations can be made for short-range (high linear energy transfer) particles such as alpha and beta radiation, compared to most penetrating (low linear energy transfer) X- and gamma radiation. These approaches will be developed in Sections 6.1.1 and 6.1.2.

6.1.1 Alpha and beta internal emitters

The determination of absorbed dose from internal emitters follows directly from the definition of dose (J/kg, or Gray). Assumptions of an infinitely large medium and a uniform distribution of radioisotope in tissue are used. "Infinitely large" means that the target-tissue dimensions exceed the range of the radiation. For particles with short ranges (alpha and most beta radiation), the energy absorbed by the surroundings equals the energy emitted by the radioisotope. These assumptions are captured by the concept of specific effective energy (SEE) given by Eq. (73):

$$SEE\left(\alpha \text{ or } \beta\right) = \frac{\overline{E}\left(\alpha \text{ or } \beta\right)}{m} \quad \left(\frac{MeV}{transform \cdot kg}\right), \tag{73}$$

where

$\overline{E}$ is the average energy of the emitted particle per disintegration (MeV per transform)

m is the mass of the target organ or tissue (kg).

The instantaneous dose rate at time=0 from intake of a short-range (charged particle) emitter of activity C_t (Bq) can therefore be calculated from the SEE using Eq. (74):

$$\dot{D}_{\alpha/\beta} = \frac{\left(C_t\ \text{Bq}\right)\left(1\frac{\text{dps}}{\text{Bq}}\right)\text{SEE}\left(\bar{E}\frac{\text{MeV}}{\text{dis}\times\text{kg}}\right)\left(1.6\times10^{-13}\frac{\text{J}}{\text{MeV}}\right)\left(86400\frac{\text{s}}{\text{d}}\right)}{\left(1\frac{\text{J/kg}}{\text{Gy}}\right)}\ \left(\frac{\text{Gy}}{\text{d}}\right), \quad (74)$$

$$= 1.382\times10^{-8}\times C_t\times\text{SEE}\ \left(\frac{\text{Gy}}{\text{d}}\right)$$

where

C_t is the radioisotope activity located in the target tissue (Bq)
SEE is the specific effective activity of the radioisotope contained with mass m_t (MeV/dis-kg).

Radioactivity is removed by the body through two mechanisms: (i) physical decay of the radionuclide, and (ii) biological elimination of the radionuclide. To take into account both processes, the concept of effective half-life is introduced. The effective half-life describes the residence time of a radionuclide in a tissue or an organism and considers both radiological (physical) decay (T_R) and biological elimination (T_B). If biological elimination follows first-order kinetics, then it can be described by a loss-rate constant, λ_B, which is related to half-life by the relation $\lambda_B = (\ln 2/ T_B)$. The biological rate constant can be summed with the radiological rate constant, λ_R, to yield the effective elimination constant, $\lambda_E = \lambda_B + \lambda_R$, which is related to half-life by $\lambda_E = (\ln 2/ T_E)$.

Biological (T_B) and radiological (T_R) half-lives together eliminate a radionuclide from the body faster than either one alone. Combined, they become the effective half-life given by Eq. (75):

$$T_E = \frac{T_B \times T_R}{T_B + T_R}\ . \quad (75)$$

Effective half-lives are generally dependent upon the chemical form of the internal emitter. Note that, although radiological half-lives are well-known quantities, biological half-lives are specific to individuals, and therefore tabulated biological half-lives are approximations for some reference person that may or may not represent any particular individual. With that caveat, some representative examples of radiological, biological, and effective half-lives are given in Table 17.

Table 17 Effective half-life

Radioisotope	half-life		
	radiological	biological	effective
H-3 (tritium)	12 y	12 days	12 days
C-14	5560 years	10 days	10 days
P-32	14 days	257 days	14 days
S-35	87 days	90 days	44 days
Co-60	5 years	10 days	9.5 days
Sr90	28 years	50 years	18 years
I-131	8 days	138 days	7.6 days
Po-210	138 days	60 days	42 days
Ra-226	1620 years	45 years	44 years

Using Eq. (74) as a basis, the estimated dose rate at some time t after intake of a radioisotope is given by Eq. (76):

$$\dot{D}_t = \dot{D}_{\alpha/\beta} e^{-\lambda_e t} = \dot{D}_{\alpha/\beta} e^{-\frac{0.693}{T_E} \times t} . \tag{76}$$

Finally, the dose may be determined by integrating over time as in Eq. (77):

$$\begin{aligned} D &= \dot{D} \int_0^t e^{-\lambda_E t} dt \\ &= \frac{\dot{D}_{\alpha/\beta}}{\lambda_E} \left(1 - e^{-\lambda_E t}\right) , \end{aligned} \tag{77}$$

where D is the dose (Gy), $\dot{D}_{\alpha/\beta}$ is the instantaneous dose rate at time=0 (Gy/d), and λ_E is in units of d^{-1} (any time base will work as long as the dose rate and the effective elimination constant have the same base unit). For long times (on the order of seven half-lives or more), the dose approximation becomes Eq. (78):

$$D = \frac{\dot{D}_{\alpha/\beta}}{\lambda_E} \text{ (Gy)} . \tag{78}$$

For regulatory purposes, t = 50 years is normally used for a committed dose, and the absorbed dose (or dose rate) can be converted to equivalent and effective quantities using the appropriate weighting factors.

6.1.2 Gamma internal emitters

In the previous section, expressions were developed that are appropriate for short-range particles such as alpha and beta particles. Long-range neutral particles such as gamma rays require a different approach because they do not deposit all their energy locally. In other

words, the absorbed dose cannot be calculated by assuming the tissue or organ to be infinitely large because gammas are highly penetrating (unlike alphas or betas). Therefore, only a fraction of the energy carried by photons originating in the tissue containing the radioisotope will be absorbed within that tissue. Therefore, an alternative approach to that used for charged particles is required. For this scenario, the following parameters must be considered in the calculation of absorbed dose:

- Energy per decay
- Total activity
- Mass of tissue or organ (target)
- Fraction of energy emitted which is absorbed in the target
- absorbed fraction, φ
    - φ = (Energy absorbed by target)/(energy emitted by source)
    - φ = 1 for both alpha and beta particles (therefore, this approach will also work for these particles).

The concept of absorbed fraction is illustrated in Figure 64. The absorbed fraction can be calculated as $\varphi = \frac{\Delta E}{E_{in}}$.

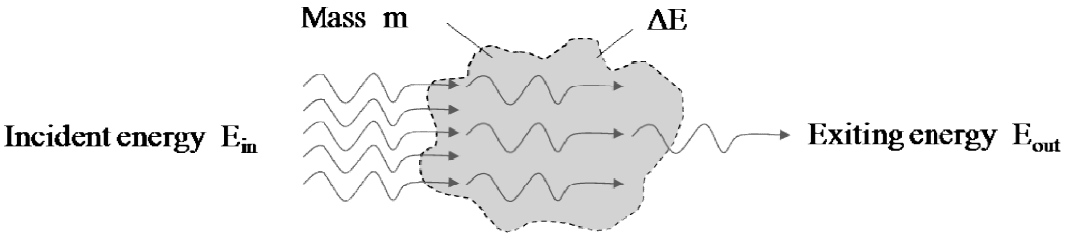

Figure 64 Absorbed fraction in mass

The specific absorbed fraction Φ is defined as the absorbed fraction φ divided by the mass of the target tissue, as given by Eq. (79):

$$\Phi = \frac{\varphi}{m} \quad \left(kg^{-1}\right). \tag{79}$$

Specific absorbed fractions are computed using mathematical phantom calculations (Monte Carlo simulations) and are tabulated as a function of target organ and photon energy for sources in various tissues [MIRD1978]. The generic equation for calculating absorbed dose rate using the absorbed fraction concept is given by Eq. (80):

$$\dot{D} = \frac{k \cdot A \cdot \sum_i n_i E_i \varphi_i}{m}, \tag{80}$$

where

D is the absorbed dose rate (Rad/s or Gy/s)
A is the activity in the source organ (Ci or Bq)
n_i is the yield of particles at energy E emitted per disintegration (also denoted as f_i or Y_i)
E is the average energy per disintegration (MeV per transform)
φ_i is the absorbed fraction

m is the mass of target (kg)

k is a proportionality constant to produce the required dose-rate units. For example, if the desired dose-rate units are Gy/s, the proportionality constant is k = (1.6E-13 J/MeV)(1 dps/Bq)/(1 J/kg/Gy) = $1.6\text{x}10^{-13}$.

The total absorbed dose can be determined using a cumulative activity $\tilde{A}$ which is the integral of the activity as a function of time. The time-dependent activity $A_s(t)$ and the integrated activity $\tilde{A}$ in the source are given, assuming first-order kinetics, by Eq. (81):

$$A_s(t) = A_s(0)e^{-\lambda_E t}$$
$$\tilde{A} = \int_0^\infty A_s(t)\,dt = A_s(0)\int_0^\infty e^{-\lambda_E t}dt = \frac{A_s(0)}{\lambda_E}, \tag{81}$$

where $A_s(0)$ is the activity in the source organ at time = 0. The units of $\tilde{A}$ are activity-time (for example, Ci-h or Bq-s, etc.). The total dose is then given by:

$$\mathrm{D} = \frac{\mathrm{k}\cdot\tilde{\mathrm{A}}\cdot\sum_{\mathrm{i}}\mathrm{n_i E_i}\varphi_{\mathrm{i}}}{\mathrm{m}}. \tag{82}$$

To calculate total dose in the body, consideration must be given to the source (S) and the target (T). The absorbed fractions are usually denoted as $\varphi(T \leftarrow S)$, and if there are multiple sources or targets, this approach can be extended to sum all contributions. For example, for multiple sources (S=1,2,…) contributing to a single target (T=1) of interest, the total-dose equation becomes:

$$D_1 = \frac{k\tilde{A}_1\sum_i n_i E_i \phi_i(1\leftarrow 1)}{m_1} + \frac{k\tilde{A}_2\sum_i n_i E_i \phi_i(1\leftarrow 2)}{m_1} + \cdots. \tag{83}$$

An alternative method of computation, given by [MIRD1978] for the calculation of internal dose, is given by the expressions in Eq. (84):

$$\dot{D}_i = \frac{A_s}{m}\phi_i\Delta_i$$
$$D = \frac{\tilde{A}}{m}\sum\phi_i\Delta_i \quad . \tag{84}$$
$$where$$
$$\Delta_i = \left(1.6\times10^{-13}\right)\cdot n_i\cdot\bar{E}_i$$

Note that consistent base units must be used in the evaluations to achieve the dose or dose-rate units required. Also, as previously discussed, for alpha or beta particles, the absorbed fractions will be equal to unity (φ=1).

6.2 Internal Hazards

A variety of internal hazards are associated with nuclear power plants and nuclear facilities in general, and the dominant pathway to committed dose is through inhalation. The primary hazards with respect to CANDU nuclear power stations under normal operating conditions are tritium, carbon-14, radio-iodines, and short- or long-lived particulates, which are discussed below.

6.2.1 Tritium

Tritium is an isotope of hydrogen with one proton and two neutrons. Tritium may be denoted as ^{3}H, H-3, or T for short. A tritium nucleus decays by emitting a single beta particle with a half-life of 12.3 years. The maximum beta energy is 18 keV (average beta energy ~ 5.6 keV). Tritium can contribute a large portion of the committed dose in a CANDU reactor and has been responsible for 30% to 40% of the radiation dose received by nuclear station staff in the past [Burnham1992]. Tritium, when taken into the body, distributes itself homogeneously throughout the entire body.

Because a beta particle requires at least 70 keV to be able to penetrate the dead surface layer of the skin, tritium is not a significant external hazard; however, it can be a significant internal hazard in large quantities. Tritium is produced through neutron absorption in deuterium ($^{1}n + {}^{2}H \rightarrow {}^{3}H + \gamma$), which is abundant in the moderator and the primary heat-transport (PHT) system of a CANDU reactor core. When neutrons irradiate heavy water (D_2O), some of the deuterium atoms (D) in the D_2O absorb neutrons to become tritium atoms (T) and generate TDO (tritiated heavy water). T_2O will also be produced, but in insignificant amounts.

In the case of heavy water that spends many years in the reactor core (for example, moderator water), after a few years, the tritium content of the D_2O can be on the order of several TBq ^{3}H per kg D_2O. For heavy water that spends most of its time outside the reactor core, for example, the primary heat-transport system water, after a few years, the tritium content of the D_2O can be on the order of a few hundred GBq ^{3}H per kg D_2O. A moderator leak can produce significant committed dose rates if unprotected. For example, for 2 TBq/kg of moderator water that leaks into the vault, an upper estimate of the committed dose rate is approximately 3 Sv/h, or 50 mSv/m [Burnham1992]. Workers around the moderator system must therefore be aware of the enormous tritium risk that can exist there if D_2O leaks are present, and therefore evaluation of the concentration of tritium in air is required before entry. People working with moderator water or performing maintenance on moderator D_2O systems must wear water-resistant (plastic) suits and breathe supplied air.

Because the PHT system operates at high temperature and pressure (300°C / 10 MPa), leaks are inevitable in a CANDU reactor. Each fuelling machine also spills about 1 L of D_2O during each channel visit, the result being that the atmosphere of the fuelling-machine vaults and the boiler room can be expected to have levels of tritiated water vapour up to 50 μSv/h under normal conditions [Burnham1992]. Design improvements to the fuelling-machine (FM) snout assembly to catch D_2O that escapes during re-fuelling operations have been made to reduce tritium-in-air levels in the FM vaults and boiler room [Aydogdu2013]. Any spill or leak of water in any of the CANDU reactor water-handling systems should be assumed to be radioactive, and workers must be aware that high localized levels of exposure may be possible.

Due to the relatively long half-life of tritium, it should be noted that the risk is present whether or not the reactor is on power and remains for a long time after shutdown.

6.2.2 Radio-iodines

Several radioactive isotopes of iodine (I-131, -132, -133, -134, -135) are produced in the fuel as fission products. These are volatile and can easily escape from defective fuel. All radioiodines decay by beta emission with associated gamma emission. Radioiodine is routinely expected to be circulating in the PHT system, and therefore any leakage from the PHT system gives rise to an airborne radioiodine hazard. Radioiodines, when taken into the body, migrate rapidly to, and are taken up by, the thyroid.

Radio-iodines can exist as elemental iodine (I_2 molecules), as organic iodine (frequently as methyl iodide (CH_3I) from ion-exchange resins in the PHT purification circuits), and as volatile hypoiodous acid (HOI). If a leak of PHT steam occurs, the radioiodines enter the air and are picked up by dust particles to exist in particulate form [Burnham1992].

Both tritium and radio-iodines are produced when the reactor operates, and production of both stops when the reactor shuts down. However the radio-iodines, with half-lives on the order of hours to days, decay away over a period of weeks, whereas the tritium does not. Therefore, the radio-iodines are a transient hazard, whereas the tritium is a persistent hazard in a CANDU reactor. There are gaseous fission-product and delayed-neutron monitoring systems in CANDU reactors to detect and locate failed fuel promptly. The FM can then remove the failed fuel from the channel to reduce radio-iodine levels in the coolant. The heavy-water purification system also removes radio-iodines from the coolant. Note that the fuel-failure rate in CANDU reactors is low (<< 0.1%), and therefore, out of approximately 5000 bundles discharged from the core annually, only a few bundles have a single fuel-element defect [Aydogdu2013].

6.2.3 Particulates

Radioactive particulates can be present in CANDU reactors as (i) fission products, (ii) activation products, or (iii) other radioactive volatiles attached to dust particles. Radiation hazard from particulates arises when airborne particulates are inhaled or when “hot” particles come into close proximity to or contact with tissue. There are two general categories of particulates: short-lived and long-lived.

Short-lived particulates, principally fission-product noble gas progeny ^{88}Rb and ^{138}Cs, are not a significant internal hazard due to their short half-lives (18 and 32 min respectively [Burnham1992]).

Long-lived particulates such as ^{141,144}Ce, ^{140}Ba, ^{140}La, ^{134,137}Cs, ^{131}I, ^{103,106}Ru, ^{95}Zr/^{95}Nb, ^{90}Sr, ^{65}Zn, ^{60}Co, ^{59}Fe, ^{54}Mn, ^{51}Cr, and ^{14}C are all beta emitters, and most also emit gamma radiation. These particulates may be found at various locations in a CANDU reactor, essentially anywhere that a leak can occur or maintenance is being done.

6.2.4 Carbon-14

Carbon-14 decays through beta emission with a maximum energy of 156 keV (average ~50 keV) and has a 5730-year half-life. Carbon-14 is generated in CANDU reactors by three processes:

$$^{1}n + {}^{17}O \rightarrow {}^{14}C + \alpha \quad \text{moderator and PHT water}$$
$$^{1}n + {}^{14}N \rightarrow {}^{14}C + p \quad \text{air}$$
$$^{1}n + {}^{13}C \rightarrow {}^{14}C + \gamma$$

About 1985, during a pressure-tube change outage at the Pickering NGS, ^{14}C was found to be widely distributed within the plant. Because of the large volume of moderator (and therefore relatively large quantity of ^{17}O), generation of ^{14}C in CANDU reactors is about an order of magnitude greater than in other pressurized-water reactors. ^{14}C was also generated in large quantities from ^{14}N in the N_2-filled annulus gas spaces of the pre-1985 Pickering A reactors and is also formed in CANDU reactors whenever there is air in-leakage into the core (for example, from the flux detectors). Carbon-14 was primarily a problem in the Pickering A Unit 1 & 2 CANDU station due to the annulus-filling gas and the lack of awareness of the extent of the condition. ^{14}C particulates were released when pressure tubes were replaced at Pickering A in the 1980s and is still present as residuals in the annulus space and in waste from the pressure-tube refurbishment. Although ^{14}C is always present in CANDU reactors, it is less of a hazard in other CANDU reactors than in Pickering A. Since the annulus gas was changed from N_2 to CO_2 in the Pickering A reactors, ^{14}C formation in the annulus gas has been reduced significantly. The production of ^{14}C in the calandria vault air applies to the Pickering A reactors because the vault atmosphere is air. The calandria is inside a water-filled tank in Bruce A, B, and Darlington reactors and inside a water-filled concrete vault in CANDU 6 and Pickering B reactors [Aydogdu2013]. Note, however, that due to its long half-life, ^{14}C can be a persistent problem for operation, shutdown, and subsequent decommissioning and disposal activities.

6.3 Internal Protection

The two primary techniques to prevent internal radiation from entering the body are (1) to block portals of entry into the body (control of the worker), and (2) to interrupt the transmission of radioactivity from the source to the worker (control of the source). Both techniques are standard industrial hygiene practice and are discussed below.

6.3.1 Control of the source

In an industrial setting, the airborne hazard is the most significant. Therefore, it is important to understand and characterize aerosols. Airborne particles are often classified by size and manner of production. Fumes are airborne solid particles formed by vapour condensation with a diameter range of 0.001–1.0 μm. Mists are suspended liquid droplets formed by condensation and can be of any diameter (generally in a range of 0.01–1.0 μm). Dusts are solid particles formed by mechanical action, with a diameter range of 0.1–30.0 μm. Smoke is a product of combustion with a diameter range from 0.1 to 1.0 μm. Inertials are large particles with diameters > 50 μm. In CANDU nuclear plants, particulates are made up of fission products and activation products. Gases are made up of noble fission gases which escape from fuel defects, such as ^{133}Xe, ^{135}Xe, and ^{85}Kr, as well as activation products such as ^{41}Ar. However, these noble gases are more of an external than an internal hazard. Vapours can be present in the form of radio-iodines and tritium and are readily absorbed by the human body.

The primary way to control radioactive emissions from the source is by ventilation. There are two ways to provide ventilation: (1) general exhaust ventilation (GEV), which is the removal of

contaminants by movement of the entire air mass into, around, and out of the workplace [ANSI2007], and (2) local exhaust ventilation (LEV) by systems that remove air at the point where the hazard is generated [ANSI2006,2003] [CSA2004]. The selection of a given ventilation solution depends upon multiple factors. General guidance principles are the following: GEV for non-toxic contaminants, multiple sources, widely distributed sources, and non-contaminated dilution air; LEV for moderately to highly toxic contaminants, one or a few sources, and when there is risk of direct worker exposure. Nuclear facilities use combinations of GEV and LEV with highly monitored and filtered air discharge systems.

6.3.2 Control of the worker

The main method for protecting the worker is the proper use of personal protective equipment (PPE), and for internal dose mitigation, the main protective equipment is respiratory protection appropriately used, although other protective clothing such as suits, gloves, and eye protection will help prevent skin absorption and contamination transfer.

In areas where a worker may be in contact with (or submerged in) radioactive material, plastic or water-resistant protective outer garments are suitable protection from beta emitters (alphas are also blocked). Gamma and neutron radiation will not be attenuated greatly by these garments, hence the need for dose control and shielding (ALARA) for these hazards. A worker performing a radiological survey around contaminated components is shown in Figure 65. The worker is shown wearing a Tyvek^{TM} polyethylene suite with hood, double latex gloves, safety glasses, and a half-face respirator (P100 toxic particulate filter).

Respirators fall into three broad categories: (i) escape, (ii) filtering, and (iii) supplied air. Escape respirators are for emergency purposes and are designed for one use or for one year from placement. Filtering and supplied-air respirators are fit-for-purpose assigned and depend on the nature of the contaminant, the nature of the work, and the level of protection required.

The expression for respirator performance is defined by the ratio of the contaminant concentration outside the mask (C_{out}) to the concentration inside the mask (C_{in}), which is called the protection factor (PF) and is given by Eq. (85):

$$\mathrm{PF} = \frac{C_{out}}{C_{in}} \,. \tag{85}$$

Protection factors are estimated by the manufacturer; however, an accurate protection factor can be determined only after performing a fit test on the respirator [ANSI2001]. Guidance for proper selection, use, and care of respirators is provided by the Canadian Standards Association [CSA2012].

Figure 65 Worker in PPE

Respirators can also be categorized as (i) air-purifying (for particulates only) and (ii) atmosphere-supplying (for particulates, gases, and vapours), which can be further subdivided into (ii-a) air-line respirators and (ii-b) self-contained breathing apparatus (SCBA). Combination-type respirators also exist which incorporate both air purification and atmosphere supply. Respirators are also classified by modes of regulator operation as follows: negative pressure (NP), positive pressure (PP), continuous flow (CF), demand (D), pressure demand (PD), and recirculating pressure demand (RP). The protection factors afforded can range from approximately 5–1000 for air-purifying respirators and 5–10,000+ for atmosphere-supplying respirators, with the largest protection factors being for self-contained breathing apparatus. Although manufacturers state nominal protection factors for respirators, the actual PF for any usage is determined through an individual fit test and check. For internal dosimetry calculations, manufacturer-listed protection factors (assuming that the respirator has been fitted and used properly by the worker) are often sufficient. Air-purifying respirator cartridges are colour-coded by contaminant type and designated by usage type and efficiency, as shown in Table 18.

Table 18 Air-purifying cartridge designations

Contaminant type	Use	Efficiency (%)
White – acid gases	N – Solid and water-based, particulates only (Not oil resistant)	99.97 (denoted as 100)
Black – organic vapours	R – Any particulate; one shift only for oily particulates (oil Resistant)	99
Green – ammonia gas	P – Any particulate (oil Proof)	95
Yellow – acid gases and organic vapours		
Purple – toxic particulates		

For example, the worker depicted in Figure 65 is wearing P100 toxic-particulate cartridges on a half-face air-purifying respirator. The P100 is the most common filter type for work in low-level-contaminated particulate environments. In many environments found in CANDU plants, the contamination is in the form of contaminated water, aerosols, or vapour, and therefore atmosphere-supplying respirators would be required. Air-line supply may be preferable when work will take longer than an SCBA tank full of air will allow, or when complete body coverage is required (such as might be the case in a tritiated water-vapour environment). SCBA may be preferable when increased mobility of the worker is required or when a higher inhalation protection factor is needed. As previously stated, the choice of respirator solution must fit the objectives of the task.

In addition to personal protective gear, another strategy for protection from radio-iodine is to use a thyroid blocker in the form of stable iodine (potassium iodide (KI)). The mechanism of KI is to saturate the thyroid with non-radioactive iodine so that when radioiodine is inhaled, it will be rejected by the thyroid and excreted, primarily through urine. The use of KI is not recommended for routine operations and is therefore only a strategy for emergency response. Charcoal-based cartridges for air-purifying respirators provide radioiodine protection in the absence of atmosphere-supplying respirators.

6.4 Summary

A variety of internal radioisotope hazards are associated with CANDU reactors. Dosimetry methods are well established, either using dose conversion factors on activity estimates (determined from bioassay measurements) or from first principles and compartmental models.

Internal hazards in CANDU reactors involve alpha, beta, and gamma emitters and include ^{3}H (the principal internal hazard), ^{14}C, radio-iodines, and particulates that can be present from leaks of fission products (defective fuel) and activation products (directly or through corrosion). Protection of the worker from internal hazards is accomplished by ventilation systems with filter trains, heavy-water vapour-recovery systems, atmospheric separation and access control features, control of sources of contamination, and judicious selection of personal protective

equipment. Respiratory protection can afford a worker a significant protection factor when used appropriately.

7 Radiation Safety Management

Management of radiation safety programs and actions is essential to the overall effectiveness of CANDU plant operations. Management includes reporting doses and keeping records of dose history for workers, generation of work plans for radiation areas, assignment of personal protective equipment, contamination control principles, waste management, and emergency preparedness. General principles of radiation-protection management can be found in [Miller1992] and details specific to CANDU operations in [Burnham1992].

7.1 Dose Records

Dose records are required by CNSC regulations [NSCA2000] and are maintained by the National Dose Registry, Radiation Protection Bureau, Health Canada. Routine dose monitoring is performed for nuclear energy workers (NEW) as defined under the CNSC Radiation Protection Regulations [NSCA2000], and the dose limits for workers are as defined in Table 12. Nuclear energy workers (NEW) are personnel who have been trained to work around radiation and radioactive materials, who have been educated about the risks (including fetal risks) involved in working around radiation and radioactive materials, who are monitored for dose if they are reasonably expected to be exposed to more than 5 mSv in a one-year dosimetry period and who have been informed in writing that they are designated as nuclear energy workers. Portability of dose records is ensured by coding the records to individual workers' social insurance numbers (SIN). Dose monitoring has a variety of objectives [Miller1992], as listed below:

- Compliance with regulatory limits
- Optimization of radiation protection program
- Assessing adequacy of protective controls
- Detecting changes in work practices
- Transparency in hazard assessment to management and workers.

Dose monitoring is required whenever radiation work is to be performed which can result in significant exposure or contamination. Different organizations can determine what exactly constitutes radiation work. The following are representative of radiation work [Burnham1992]:

- Work with radioactive sources and materials
- Work in external fields greater than 10 μSv/h
- Work in airborne concentrations greater than 10 μSv/h committed effective dose
- Entry into rubber areas (areas where loose contamination is expected and a temporary surface is used in the area with a clean-dirty boundary)
- Any activity in which one expects to receive a dose greater than 0.2 mSv.

Monitoring periods (time between reading dosimeters or bioassays) for CANDU stations tend to be much shorter than for other users (for example, university or medical). A typical monitoring period for a CANDU station may be two weeks or shorter, whereas other users typically submit dosimetry records every three months. Dosimetry estimates may be performed daily depending on the magnitude of the anticipated hazard of the radiation work being performed.

The detection of radiation fields or potential contamination is facilitated by air monitors, area monitors, portal monitors, and hand-held instrument readings. The principal personal exposures in CANDU stations being monitored for are:

- External dose – thermo-luminescent dosimetry (TLD)
    - Whole-body gamma (deep and shallow)
    - Whole-body neutron (could be from instrument reading instead)
- Extremity dose – extremity pack or finger (ring) TLD
    - typically for near-field and contact beta/gamma exposures
- Internal dose - bioassay
    - Tritium
    - Other (^{14}C).

The results of dosimetry estimates at any given time can be used to optimize worker exposure to ALARA levels, or if needed, to remove a worker from further exposure. Yearly reports are provided to NEWs which show their historical dose of record, dose received every year during employment, quarterly dose in the past year, and lifetime occupational dose. These data are maintained locally and are available through the National Dose Registry [NDR2013] upon request.

7.2 Radiation Work Planning

Germane to the effectiveness of work planning is that the persons developing and executing the plans be properly trained. Personnel working in CANDU stations require training to various levels depending on the jobs they will be required to perform. In CANDU operations, workers are categorized by colour scheme as described in the following subsection.

7.2.1 Radiation-worker colour scheme

- **Green**-qualified workers are personnel with sufficient experience and formal training to give them the qualifications to manage themselves and to supervise the radioactive protection of other workers.
- Yellow-qualified workers are trained to manage their own radiation-protection requirements, but not the protection of others.
    - **Orange**-qualified workers have minimal training and radiation work experience, but may perform radioactive work under the direct or indirect supervision of a **Green**-qualified person. They have sufficient qualifications to go into the various areas of the plant that are not posted with radiation hazard signage.
- **Red**-"qualified" persons have no formal radiation work qualifications; however, they can perform work in radiation areas only under the direct supervision of a **Green**-qualified person. The work has hazard limits that must be adhered to.

Whereas **Green-** and Yellow-qualified workers are typically designated as nuclear energy workers (NEWs), **Orange-** and **Red**-qualified workers typically are not designated as NEWs.

7.2.2 Zoning

In addition to categorization of workers, CANDU plants are also categorized by location and potential for radioactive contamination. Although any given CANDU station may have its own zone categories, a typical scheme is presented below [Burnham1992].

- **Unzoned** – areas with absolutely no contamination, generally not within the boundary of a reactor, but within the station boundary (for example, a switchyard).
- **Zone 1** – clean areas where absolutely no contamination is permitted; no radiation training is required to be in these areas (for example, administrative offices).
- **Zone 2** – areas that contain no radioactive systems and are normally free of contamination (although the potential exists for contamination due to ventilation and personnel traffic (for example, a turbine hall).
- **Zone 3** – areas that contain radioactive systems that may act as sources of contamination (for example, a reactor building).

Contamination monitoring stations must be used before crossing the boundary from Zone 3 to Zone 2 or from Zone 2 to Zone 1. A two-zone system is to be introduced in new CANDU reactors: a radiation-controlled area (RCA) and non-RCA spaces [Aydogdu2013]. Zone 3 becomes RCA, and Zones 1 and 2 become non-RCA areas. Work plans must take zoning into account because this impacts the level of qualification required for workers and the PPE or dosimetry that will be assigned.

7.2.3 Work planning

Planning for work in a radiation area goes far beyond categorization of workers. Work planning involves identifying hazards and planning work while acknowledging these hazards. Work planning specifications can be subdivided into approximately 11 steps [Burnham1992]:

1. Identify hazard(s)
2. Control hazard(s), which can be further subdivided into
    a. Elimination of hazard
    b. Minimization of hazard
    c. Install physical barriers
    d. Use warning devices
    e. Minimize human-error potential
    f. Establish procedures commensurate with the hazards
    g. Train and supervise personnel
    h. Accept the hazard and work with it
3. Simplify tasks
4. Verify instruments and assign as appropriate
5. Post signage appropriate to the hazard
6. Identify and issue dosimetry
7. Identify and issue appropriate PPE
8. Identify and assign personnel appropriate to task(s)
9. Perform pre-task briefings and clarify assignments
10. Monitor progress during work
11. Follow up after work.

A radiation-exposure permit (REP) is generated by a qualified person (a responsible health physicist, for example) consistent with the hazard and adhering to the “as low as reasonably achievable” (ALARA) tenet. The work plan takes into account all personnel qualifications and assigns responsibilities accordingly.

7.3 Personal Protective Equipment

- Personal protective equipment is needed for activities in Zones 2 and 3 both from an industrial hygiene and occupational health and safety perspective and a radiation-protection perspective. Normal industrial protective equipment may be required for work in Zone 1 and unzoned areas as well. In Section 6.3, control of the source (i.e., ventilation) and control of the worker (i.e., respirators) were discussed for airborne hazards. Additional routine personal protective equipment includes active work-zone clothing (in CANDU operations, historically called "browns" because the outer garment was a brown-coloured coverall). In fact, personnel working in active areas have a complete change of clothing consisting of coveralls, undergarments, socks, and safety shoes. The important provision is that the active-area clothing remains in the potentially contaminated areas and no personal clothing is allowed in Zones 2 or 3. Besides appropriate respiratory protection commensurate with the task (from cartridge respirator to plastic suit with air-line supply), a worker may also require a hard hat, safety glasses, hearing protection, lab coat, disposable coveralls, appropriate gloves, safety shoes, booties, or any combination of the above.

7.4 Contamination Control

The most effective way to control contamination is at the source, specifically by preventing it from occurring in the first place. However, there are inevitable instances where radioactive contamination will be generated in a CANDU reactor, for example from fuelling-machine visits, primary heat-transport (PHT) system leaks, and shutdown maintenance. Contamination can be present in solid, liquid, and aerosol forms, and may be found near the source or at some distance from the source. Controls are established in CANDU plants in the form of permanent control zones with defined access-control points (Section 7.2.2), and temporary control zones for *ad hoc* decontamination activities (sometimes called "rubber areas").

Personal contamination protection is achieved by proper identification and use of protective clothing (described in Section 7.3) and proper respiratory protection (described in Section 6.3). Spread of contamination is minimized by using proper work planning techniques (described in Section 7.2).

If personnel or equipment become contaminated (as determined by radiation-detection equipment), then decontamination must be performed. Decontamination of equipment can be performed using techniques similar to normal cleaning (washing and scrubbing), although more attention must be given to the waste stream generated (collection and disposal of cleanup waste).

Radiation-detection instrumentation is essential for identifying contamination and determining the efficacy of decontamination. Three distinct limits must be considered with respect to contamination monitoring and decontamination activities: (i) detection limit, (ii) action level, and (iii) regulatory limit. The detection limit is a function of the instrument used to detect radioactive material. The action (or reference) level is established by the licensee (station operator) and is typically some fraction of the regulatory limit. The purpose of this level is to identify potential problems through a comprehensive quality-assurance program and rectify them before contamination approaches the regulatory limit. The regulatory limit is set by a governing body (in Canada, the CNSC) and is a limit which should not be exceeded in normal

operations. The relative relationships, with four different measurement cases, can be seen in Figure 66.

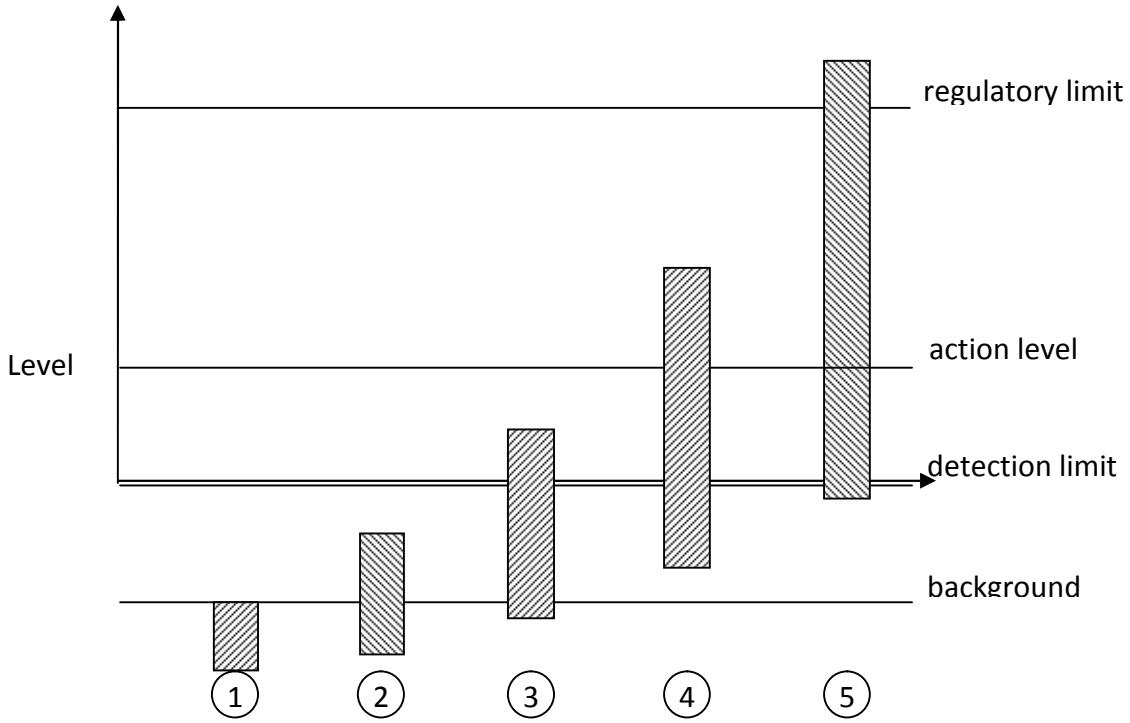

Figure 66 Relationship between background, action level, detection limit, and regulatory limit

In Case 1, the measurement lies at the background level and therefore is considered background. Note that a measurement cannot, by definition, be less than background, although the background level may vary from one place to another. In Case 2, the measurement lies between the background level and the detection limit of the instrument. In this case, with a given confidence, the measurement is below the detection limit of the contamination-survey instrument, and generally no further action is taken. In Case 3, the measurement lies above the detection limit, and radiation has been positively detected with a selected confidence level. This measurement is useful for identifying the presence of radioactive material and may assist in decontamination. In Case 4, the reading has exceeded an action level (as put in place by the station operator), and generally decontamination may be required to mitigate the hazard (although it is not strictly required by the regulations). The location of the detection limit and the action level may be exchanged. In Case 5, the measurement lies above the regulatory limit. In this case, decontamination procedures must be carried out to bring the levels of measured radioactivity to levels below the regulatory limit and as low as reasonably achievable (typically below the action level).

7.4.1.1 Waste management

Radioactive waste is generated in all CANDU plants during routine operations. Waste can originate from contamination sources (Section 7.4) and may include cleanup and laboratory wastes (paper, rags, glassware, etc.), contaminated tools or components, filters, discarded components that have been activated, and ion-exchange resins from process systems. These types of waste are typically called stored wastes. A second source often categorized as waste is

spent (or used) fuel. This material is not really waste as such because there is a strong possibility of reuse (or reprocessing) of the fuel in the future. The International Atomic Energy Agency's philosophy for long-term storage is that future generations should be afforded the same degree of protection as the current population. Spent nuclear fuel will not be considered further in this chapter.

There are two general approaches to radioactive waste management [IAEA1995]:

1. Dilute and disperse
    a. Aerosol: discharge as gases or other fine particulates
    b. Liquid: discharge into marine or fresh-water environments
2. Concentrate and contain
    a. Solid: storage (delay and decay).

Dilution and dispersion occur routinely under the CNSC licence conditions for the CANDU operator. Solid-waste management may be considered in three practical categories for CANDU operations: (i) sources (which may include activated materials or "hot" particles, (ii) loose contamination, and (iii) tritium waste, as discussed in the next subsection.

7.4.1.2 Sources

Sources, including fixed surface contamination, will normally be collected by removing the entire item to solid-waste storage. Efforts to remove sources of radioactive material from the equipment or item to which they belong (such as removing a hot particle from a section of pipe) may not be undertaken depending on the hazard analysis. Measures to ensure the integrity of the source or fixed contamination must be implemented to prevent the spread of contamination while the material is being moved.

7.4.1.3 Loose contamination

For loose contamination, it is first essential to establish contamination-control measures. An appropriate method for removing the contamination will be used. Typically this will include brushing, sweeping, wiping, cleaning, or some combination of these (as discussed in Section 7.4)

In some instances, decontamination may not be practical (e.g., in contaminated piping). Disposal of the entire item (reduced to the smallest practical volume) will then be undertaken. Decontamination efforts must continue until measurement readings are below the detection limit of the instruments, below the action level, and below the regulatory limit. Upon completion of decontamination activities, a thorough survey of the decontaminated location is performed.

7.4.1.4 Special case: tritium contamination

Tritium is a major component of the contamination associated with CANDU plants. Tritium contamination is characterized by the high mobility of tritium, resulting in ready cross-contamination, absorption, and permeation in many materials. Tritium outgassed from surfaces is primarily in the form of tritiated water. The properties of tritium make tritium decontamination more complex than decontamination of other radioisotopes. Surface tritium will generally be found as:

- Removable tritium – tritium near the surface that is easily removed by light washing.
- Fixed tritium – located deep within the surface of the material and not amenable to surface decontamination techniques; and
- Transferable tritium – tritium that is near the surface and acts as a reservoir of tritium to replenish the removable tritium if that layer is significantly diluted.

Accordingly, decontamination only of the removable layer will likely result in reappearance of the tritium after a short time (less than one month) as removable tritium. Repetitive decontamination of the surface layer will eventually reduce the inventory and hence the eventual level of removable tritium on a surface.

The most commonly used methods for tritium decontamination are washing, vacuuming, purging, thermal desorption, and isotopic exchange by adding water or hydrogen gas in the purge gas. These are generally adequate, but are time-consuming. More sophisticated techniques such as chemical or electrochemical etching and plasma can potentially provide high decontamination factors. Baking of the tritium in a contaminated surface is an alternate method for decontamination, but is not normally used.

Tritium contamination of walls and floors, particularly concrete, poses a chronic outgassing problem. Concrete decontamination is very difficult and slow. Covering the surfaces with non-porous covers may be an alternative approach to decontamination. Walls inside a CANDU reactor building are painted with reinforced epoxy-resin paint for ease of decontamination and to reduce tritium contamination.

7.5 Emergency Planning

Besides routine planning functions, health physics/radiation protection also has duties related to emergency planning. Emergencies are classified by the IAEA on the INES Nuclear and Radiological Event Scale [IAEA2009], which outlines methods for assessing releases, estimating doses, and relating these to the INES scale. Emergency planning for different categories of events is critical for protection of workers, the public, and the environment.

Nuclear emergency plans consist of on-site emergency contingency planning as well as off-site planning. In any type of emergency, the top-level management has the ultimate responsibility for emergency response. From a practical perspective, the duty shift supervisor at a CANDU reactor will be responsible for the initial response to an emergency, whether or not it involves radioactive material (for example, emergencies may involve any combination of radioactive material, fire, chemical, medical, and security issues). Emergency response involves activating an emergency response team that is specially trained for a variety of emergencies. If there is an emergency that involves, or may involve, the release of radioactive material from the confines of the station site, an off-site emergency plan will be activated which interfaces with provincial and federal assets. To ensure that emergency plans are relevant to the current plant conditions and that the various assets are prepared for an emergency, periodic training is conducted and emergency-response exercises performed.

7.6 Summary

Radiation-protection program management for CANDU operations is essential for both personnel safety and reactor safety functions. The health physics office will be involved in or responsible for various management aspects including licensing, training, personnel monitoring and

bioassay, work planning and review, surveys and equipment maintenance, inventory control, waste management, emergency preparedness, quality assurance, and audits. Well-defined and well-executed management plans at CANDU plants ensure that the stations operate in the most safe and reliable configuration possible.

8 Radiation and the Environment

CANDU plants routinely release radioactivity to the environment. This section considers releases due to routine CANDU operations and discusses the philosophy behind protection of the environment from these releases. Some historical perspective on environmental radioactivity can be found in [Moeller2005], and guidance on environmental radiological assessment can be found in [Till2008] and [Faw1999].

8.1 Radiological Environmental Monitoring Program (REMP)

There are three general reasons to sample the environment for radioactivity:

1. Establish a baseline level of radionuclides;
2. Show compliance with regulations for some activity; and
3. Track radionuclides in the event of a suspected unplanned release.

There is no essential difference in the sampling strategies for any of the above. In all cases, one is trying to determine a level of radionuclides , biological dose, or both and to define a course of action based upon measurements.

The main differences are the priorities given to various aspects of the sampling:

- For a baseline study, the priority is to determine the levels and the statistical variations to a high degree of confidence;
- For routine environmental sampling, the priority is to determine the lowest minimum detectable activity levels and compare them with what the expected releases might be and what the regulatory criteria demand; and
- For emergency response, the priority is to determine the trends and rough magnitude of the release to provide protective-action guidance to the public.

For routine and baseline sampling, a number of biota that are in the pathway between radiation source and receptor must be sampled. These biota are identified during the planning stage of the environmental monitoring program. Likewise, in a post-accident environment, a number of parameters require monitoring, but their importance has time dependencies because some pathways from radioactive source to human exposure have a longer residency time than others.

The design of an environmental monitoring program and requisite sampling strategies are individually dependent on purposes, goals, and specific locations. However, some general guiding principles for environmental monitoring exist and are presented in the following sections.

8.1.1 Design of a monitoring program

The design of an effective radiological environmental monitoring program requires a thorough understanding of the specific location and the surrounding environment, especially with regard to environmental pathways which lead to human populations.

An environmental radiation monitoring program is normally used to:

- assess independently the actual or potential dose to an average member of a Critical Group resulting from normal operation of a nuclear facility; and to
- provide an independent assessment of the effectiveness of the source and effluent control and effluent monitoring programs.

A Critical Group is made up of individuals in a population which because of habits of lifestyle and diet may be more exposed to radiation than others. Protection of members of the Critical Group ensures that the greater majority of the exposed population will be protected. For example, one such Critical Group is that of one-year-old infants around land-based nuclear facilities where dairy farming is important.

The program can also serve to:

- Validate, as far as possible, the assumptions made in any assessment of derived release (or emission) limits (DRL/DEL) for the facility; and to
- Provide assurance to the regulator and the public that human health and the environment are being monitored and protected.

Normally an operator (licensee) is regulated to develop and implement an approved radiological environmental monitoring program. Elements of an environmental monitoring program can be found in an emergency response program, although their management and deployment will be different.

The primary objective of a long-term radiological environmental monitoring program is to define a regime of both continuous and periodic sampling and analysis to assess those radionuclides which are found in the various identified and sampled pathways. The accumulation of these data over a period of years provides a baseline against which all subsequent variations can be compared and assessed.

8.1.2 CANDU facilities

CANDU facilities have conducted individual radiological environmental monitoring programs for more than 20 years. The rationale for sample collection and the selection of sampling media are discussed below.

There are three significant comparable aspects in the radiological environmental monitoring programs at Canadian utilities which have nuclear reactors. The three aspects are:

(i) source emissions monitoring and control of source emissions;
(ii) derived release (or emission) limits (DRL/DEL) specific to each facility, to define the dose impact to humans of identified radionuclides released into specific pathways; and
(iii) radiological environmental monitoring programs.

These are discussed in the following sections.

8.1.2.1 Source emissions

Source emissions occur into two major pathways from all Canadian facilities: air and water. Continuous monitoring of all airborne radioactive emissions at points of release for the gaseous effluent monitoring (GEM) program alerts station operators to radionuclide releases that may exceed release setpoints defined by derived release (or emission) limit (DRL/DEL) calculations. Radioactive liquid effluent releases from holding tanks are analyzed before pump-out to the environment and are monitored during pump-out to ensure that releases do not exceed setpoints defined (by the DEL calculations) for monthly releases in the liquid effluent monitoring (LEM) program. Such in-plant source-emissions monitoring is the first line of assessment of what is being released on a continuous or interim basis from the facility and forms the basis for control of emissions where possible, for plant shutdown, or for notification of the relevant authorities (for example, advising emergency measures organizations that the emissions exceed the DELs and could lead to exceeding the regulatory dose limits).

Source emissions are controlled at CANDU stations as follows [Aydogdu2013]:

- Airborne and waterborne releases are controlled by limiting releases to 1% of the weekly DRL (see Section 8.1.2.2) for airborne releases and 1% of the monthly DRLs for waterborne releases (these are internal targets); see Section 8.2.
- The gaseous-effluent monitoring system consists of a gaseous-effluent monitor (GEM) which continuously monitors and measures I-131, noble gases, and particulate emissions, and a tritium cartridge and a ^{14}C sampler which continuously monitor tritium and ^{14}C emissions (measurements are made in the laboratory on a daily or weekly basis).
- The reactor-building (RB) ventilation system has filter trains that contain charcoal and HEPA filters to reduce iodine and particulate releases. The off-gas management system in a CANDU reactor is used to reduce noble gas emissions.
- CANDU reactors have liquid-waste decontamination facilities (except Pt. Lepreau Generating Station) to decontaminate liquid waste in high-activity tanks (if necessary) before discharge.

8.1.2.2 Derived release (or emission) limits

Assessment of source radiation releases is gauged against derived release (sometimes called emission) limits (DRL/DEL) calculated for each radionuclide in each source pathway for each facility. The DRL for any radionuclide is derived following a process defined by the Canadian Standards Association in publication N288.1-08 [CSA2008], and guidance has also been provided by the CANDU Owners Group [COG2008]. The DRL is defined as the activity of a particular radionuclide or a group of radionuclides (i.e., noble gases, particulates) that, if released over the course of a year into defined environments, will cause the general public dose limit to be reached. DRLs are calculated in five categories (tritium, I-131, ^{14}C, noble gases, and particulates) for airborne releases and in three categories (tritium, ^{14}C, and gross β/γ) for waterborne releases. This assessment is the means by which human risk from radiation releases from any defined source can be estimated.

8.1.2.3 Radiological environmental assessment

Once releases from a facility have taken place, their impact is assessed by means of a radiological environmental monitoring program. This program is based on site-specific criteria concern-

ing release pathways leading to humans; those humans likely to be most exposed because of specific habits (one or more critical groups); sample types in these pathways; and collection frequency of these sample types. Usually, collection frequency and sample type are based on analyzing those samples that are most likely to provide evidence of specific radionuclides that would be indicative of releases from the facility into the environment. There are constraints on detection because of dispersion (dilution) from the point of release. In most airborne pathways, a dispersion factor of about one million is applied up to the first kilometre. In water effluents, the dispersion factor is a function of cooling-water flow and tidal mixing and may range from a factor of ten to thousands. In addition, radionuclides dispersed into the environment may become concentrated into various biota, making their detection relatively simple. For example, many marine biota concentrate various metals (including radioisotopes of those metals) from seawater. Much of this information is summarized in Canadian Standards Association document N288.4-10, which defines the desirable features of an environmental radiation-monitoring program, and to which Canadian nuclear utilities generally conform.

Table 19 indicates some significant sampling media and their approximate frequency of collection in the two major pathways (air and water) by the various provinces, all of which operate CANDU nuclear reactors. The most significant data outside plant boundaries are obtained from samples gathered close to the point of emission in each of the two major pathways (air and water). The air pathway leads the most directly to humans. Air is monitored by means of continuous sampling for airborne radiation particulates, radioiodines, noble gases, tritium, and carbon-14. Sampling is continuous, and measurements by the GEM of ^{131}I, noble gases, and particulate emissions are also continuous, but measurements of tritium and ^{14}C releases are not; tritium cartridges and ^{14}C samples are taken to the laboratory for tritium and ^{14}C measurements on a regular basis (daily or weekly). Samples which are not amenable to continuous collection (soils, vegetation, vegetables, fruits, milk, and potable water) are taken as grab samples on a monthly or quarterly basis or as the season permits. The effectiveness of air-pathway monitoring relies upon continuous and repetitive sampling of the given media. In the liquid effluent pathway, the primary sample is the water itself. For water which is not used for human consumption, such as seawater, grab samples may be taken quarterly from several locations. In the case of potable lake water from which communities draw their drinking water, for example, water from Lakes Ontario and Huron, daily samples (sometimes several samples per day) may be taken, primarily to monitor for tritium. Less frequently, samples of raw lake water, fish, aquatic plants and sediments may be collected, usually on a quarterly (or less frequent) basis.

Table 19 Example of samples collected by Canadian utilities [Cole1997]

Group	New Brunswick	Ontario	Quebec
Air pathway samples (continuous sampling)	Air particulates (m)	Air particulates (m)	Air particulates (m)
	Air iodine (m)	Air iodine (m)	Air iodine (6m)
(grab samples)	Air tritium (m)	Air tritium (m)	Air tritium (m)
	Air ^{14}C (m)	Air ^{14}C (m)	Air ^{14}C (m)
	Soil (q)	Forage (s)	Vegetation (y,s)
	Fruit (s)	Fruit (s)	Soil (y)
	Vegetation (s)	Vegetables (s)	Vegetables (s)
	Vegetables (s)	Drinking water (m)	Drinking water (q)
	Milk (q)	Milk (w)w	Milk (m)
	Well water (q)	Well water (m)	Farm produce (q,s)
	Surface water (q)	Greenhouse vegetables (q)	
Water pathway samples	Seawater (q)	Lake water (y)	River water (m)
	Mud (q)	Aquaculture fish (m)	Sediment (q)
	Beach sediment (q)	Water treatment plants (d)	Fish (m)
	Seafood (s)	Sediment (y)	Mollusks (m)
		Lake fish (y)	Aquatic plants (a)
Other	Rain water (m)	Rain water (m)	Rain water (m)
	Berries (s)	Honey (s)	Maple syrup (s)
	Lichen (q)	Farm samples (w,s)	TLD (c)
	TLD (c)	TLD (c)	

N.B.: collection frequency is provided in parenthesis. Key: d (daily), w (weekly), m (monthly), y (yearly), c (continuous), s (seasonal), q (quarterly).

Table 20 indicates some of the more significant radionuclides that are characteristic of potential nuclear-reactor releases into the environment because of leakage of either fission or activation

products. Selection of key indicator radionuclides (e.g., ^{131}I, ^{137}Cs, ^{95}Zr, ^{95}Nb, ^{144}Ce, ^{59}Fe, ^{54}Mn, ^{60}Co, etc.) for analysis is sufficient to define whether or not such leakage can be detected and avoids the need to address unusual and difficult-to-analyze radionuclides such as ^{90}Sr (which is a pure beta emitter and is more difficult to identify). If the indicator nuclides are not detected, then it can be assumed that an isotope such as ^{90}Sr is unlikely to be present.

Table 20 Some key fission and activation radionuclides expected from a reactor release

Argon 41	Iodine 131	Krypton 89	Thorium 228
Carbon 14	Iodine 132	Manganese 54	Thorium 234
Cerium 144	Iodine 133	Nickel 63	Tungsten 187
Cesium 134	Iodine 135	Niobium 95	Xenon 131m
Cesium 137	Iron 55	Nitrogen 13	Xenon 133m
Chromium 51	Iron 59	Ruthenium 103	Xenon 133
Cobalt 58	Krypton 83m	Ruthenium 106	Xenon 135m
Cobalt 60	Krypton 85m	Strontium 89	Xenon 135
Hafnium 181	Krypton 85	Strontium 90	Xenon 138
Helium 3	Krypton 87	Tantalum 182	Zinc 65
Hydrogen 3	Krypton 88	Thorium 227	Zirconium 95

Special Cases: ^{131}I and ^{137}Cs

Care must be taken when interpreting results for at least two of the more significant indicator radionuclides (^{131}I, ^{137}Cs) because they can enter the environment from releases other than from nuclear reactors. Iodine-131 is widely used in medical programs at larger hospitals for thyroid diagnosis and therapy. It is readily detectable from time to time in samples taken from the areas surrounding these hospitals, but its existence is not generally publicized. Routine releases of iodine-131 from such medical usage, even from one patient, is generally many thousands of times larger than typical total releases that might be associated with reactor operations. Cesium-137 and ^{90}Sr are associated with hardwood ash. Fallout from the major bomb tests conducted in the 1950s and 1960s resulted in deposition of fission radionuclides in north-eastern North American forests. These radionuclides have relatively long half-lives, are still concentrated in hardwoods in this region, and can still be detected in tree lichens. Careless or inadvertent disposal of wood ash can lead to the detection of ^{137}Cs in many sample types, including milk and soils.

8.1.3 Environmental transfer model

Routine environmental monitoring requires a comprehensive environmental-transfer model which describes the pathways from source to critical group. The principal radionuclides of investigation will be dependent on routine release criteria. For example, in CANDU reactors, ^{3}H, ^{14}C, and noble gases are important environmental radionuclides for monitoring.

By way of comparison, the principal radionuclide releases to the environment under accidental conditions would be, in order of decreasing importance in terms of dose:

- Radioactive particulates, in particular ^{137}Cs, are often released in the form of cesium iodide (CsI) following severe fuel failure and loss of containment. In very severe fuel-melt events, other radioisotopes such as actinides may also be released. However, this is unlikely, and as demonstrated following the Chernobyl accident, actinides would not be the most significant dose contributor; ^{137}Cs and ^{90}Sr would be the most significant dose contributors;
- Radioactive volatiles (mainly radio-iodines) which, along with radioactive noble gases, are the first radioisotopes released from the fuel at the onset of fuel failure; and
- Noble-gas radionuclides (mostly radio-isotopes of krypton and xenon), which may be released following failure of the reactor fuel cladding, even in the absence of reactor fuel melting.

Whether planning for routine monitoring or emergency management, the modelling analysis for pathways is similar. See Chapter 13 for a more in-depth discussion.

8.1.3.1 Modelling

A formal pathways analysis model is required to determine the significant pathways and the types of samples and measurements required for all locations around the source of release. In addition, baseline meteorological data showing long-term weather patterns, especially with respect to prevailing wind directions at different seasons, may or may not indicate that a specific geographical direction should be emphasized for sample collection at each site.

8.1.3.2 Pathways

In pathways analysis, environmental components are analyzed to determine:

- ability to accumulate environmental contaminants;
- relative residence time of contaminants in the medium;
- importance as a food source to humans; and
- suitability for inclusion in a radiation monitoring program.

A generalized environmental transfer model is depicted in Figure 67.

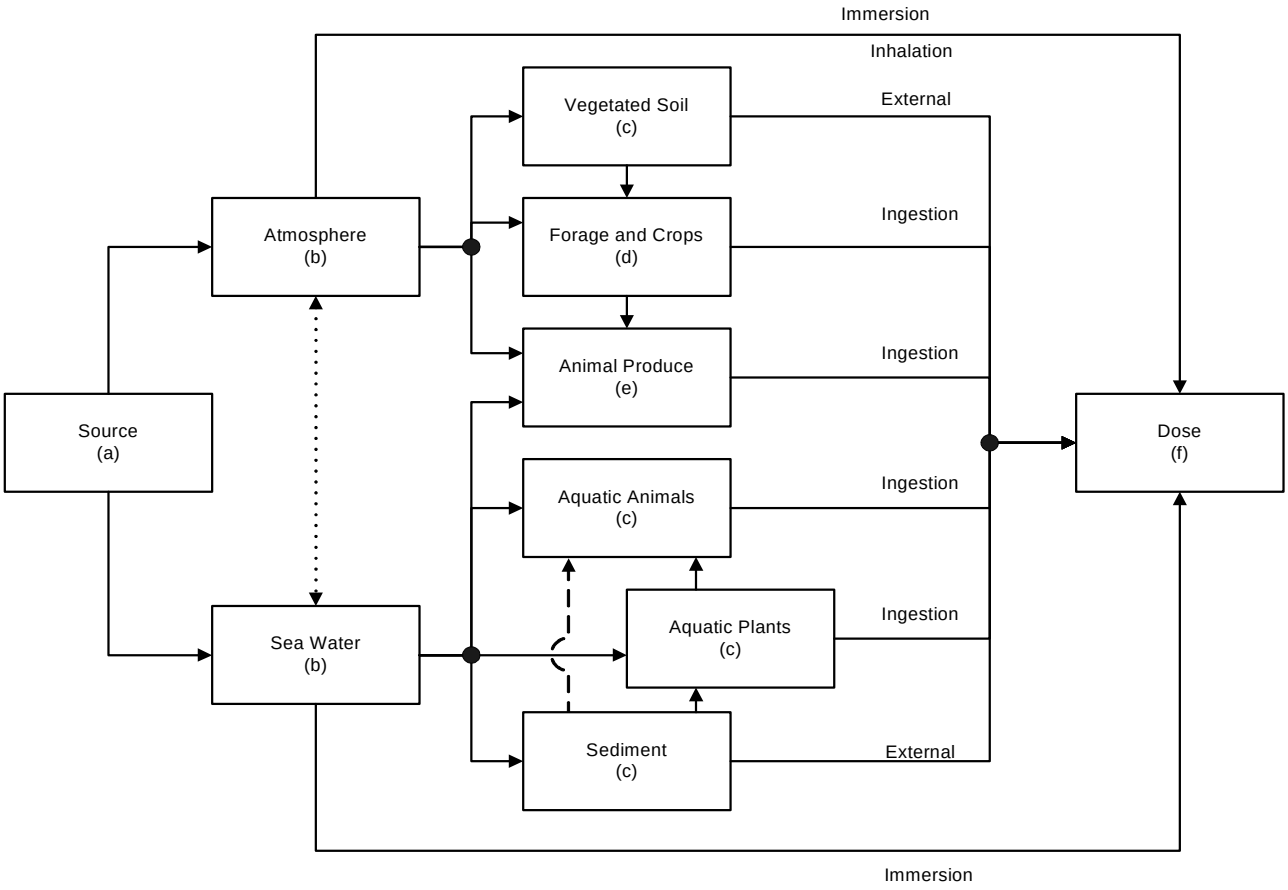

Figure 67 General environmental transfer model

The labels (a) through (f) for each component in Figure 67 may be considered as "levels of desirability" for a given sample from an environmental component. For instance, the most desirable sample for an environmental radiation-monitoring program comes from near the source, whereas dosimetry information at the end of the chain provides inconclusive information about the source of the radiation. The most useful information is, in general, obtained from components (b) through (e). Atmospheric-pathway samples are most useful if obtained from a continuous air-sampling regime because of the temporal variability of these samples. The most useful secondary indicator for the air pathway can be seen to be vegetated soil. For seawater samples, seawater, aquatic animals, plants, and sediments all provide useful data. Note that the environmental-transfer model will change depending on the location where the sampling is taking place. For example, if the sampling locations are not near sea water (as is the case in mainland Ontario), then the sea-water pathway may be replaced by a lake-water pathway. The main point is that the target group must be adequately identified and modelled in the pathways analysis.

8.1.3.3 Indicator species

It is important that food species which constitute critical pathways to humans, including terrestrial and marine organisms, are identified. The uptake through food chains is extremely complex because of the:

- Physical and chemical behaviour of radionuclides;
- Food chain being considered;
- Bio-accumulation factors in different species and foods;
- Seasonal variability; and
- Distance from the source.

The indicator species will vary depending on the individual pathways analysis performed. Some typical indicator species include:

- Cow's milk;
- Beef;
- Vegetables and fruits;
- Soil (vegetation);
- Marine plants; and
- Seafood.

The indicator species will typically be determined by close examination of the critical groups.

8.1.3.4 Seasonality

Time of year (for release and sampling) can have a significant impact on the way in which radionuclides move through the environment, particularly in their use in the various food chains. Although this feature is generally not a primary consideration when performing emergency environmental analysis, it can be extremely important for establishing baseline levels of radionuclides in the environment.

8.1.4 Practical considerations for environmental sampling

When performing sample acquisition during an emergency (such as an unplanned reactor release), care must be taken to prevent over-exposure to radiation. Dosimetry and adherence to dose restrictions will be monitored by the health physics department. A calibrated and functioning field-survey instrument must be used to monitor dose rate and total dose. There will likely be airborne hazards that are not immediately identified and that have implications for personal protective equipment. Generally, the greatest hazards will often be non-radiological hazards (as discussed below).

During routine environmental monitoring, radiation poses no threat to the person obtaining the samples because the work is being undertaken at background levels of radiation. However, there are many non-radiological threats that can pose a hazard to the sampling team. These hazards are generally of the same magnitude of hazard that a person may be exposed to daily.

The following aspects must be considered before performing any environmental survey:

- Is the area to be sampled clear of obvious threats, such as
    - Wild animals
    - Loose terrain
    - Electrical hazards
    - Slippery surfaces?
- Has the owner of the property been contacted, and has approval to sample been obtained? Hazards include:
    - *Agitated* land owners

  - Guard dogs
  - Hunters.
- Is the sampling team prepared *personally* to be out-of-doors?
  - Anti-allergens (EpiPens®)
  - Appropriate footwear
  - Appropriate clothing for time of year, terrain, and weather.

The above is not an exhaustive list, but provides examples of some of the considerations required to ensure safe environmental sampling.

8.2 CANDU Releases

CANDU reactor operators monitor both emissions of radionuclides and quantities in environmental samples. The results from these radiological environmental monitoring programs (REMPs) can be found on utility Web sites, and summaries are available for anyone to review. Detailed analyzes and reports are maintained by the health physics office and are made available to the regulator (CNSC). General emissions monitored and reported by the stations are available in these documents; a representative dataset for a CANDU is provided in Table 21 (adapted from [OPG2012]) along with the order-of-magnitude DRLs and the percentage activity of each DRL . Note that the emissions may vary from year to year, but generally are within an order of magnitude of those presented in Table 21. In addition, note that the emissions are typically much less than 1% of the derived release (or emission) limit for each radionuclide or group of radionuclides.

Environmental samples provide a basis for calculation of dose to critical-group representative persons. The doses estimated from environmental samples from CANDU plants have, up to the time of writing this document, never approached the CNSC public dose limit of 1 mSv/y and have generally been approximately three orders (0.1%) of magnitude below this limit.

Table 21 Order-of-magnitude CANDU radionuclide emissions

Emission	Magnitude of activity (Bq)	Magnitude of DRL	Activity % of DRL
Air			
HTO	10^{14}	10^{17}	0.10
Noble gas	10^{13}	10^{17}	0.01
I-131	10^{8}	10^{12}	0.01
Particulates	10^{7}	10^{11}	0.01
^{14}C	10^{12}	10^{15}	0.01
Water			
HTO	10^{14}	10^{18}	0.01
Gross β/γ	10^{10}	10^{14}	0.01
^{14}C	10^{9}	10^{15}	0.0001

It is clear from Table 21 that tritium and noble gases are the greatest contributors to the emitted activity from CANDU reactors. They are correspondingly the major contributors to

public dose as well (see, for example, [OPG2012]). However, as was previously discussed, the dose attributable to these radionuclides is approximately 1 μSv/y, which is an extremely small fraction of the average person's dose attributable to background radiation exposure.

8.3 Protection of the Environment

The radiological environmental monitoring program is designed to [OPG2012]:

i. demonstrate that radioactive materials are properly controlled at the plant, independently of effluent monitoring results;
ii. enable estimation of annual doses to the public from operation of the nuclear facility;
iii. provide data to be used for calculating derived release (or emission) limits and public doses.

There has always been an underlying assumption that if adequate protection of the public is achieved, protection of the environment and environmental-ecosystem components is *de facto* afforded. For example, in the 1990 recommendations of the ICRP [ICRP1990], it was stated that:

> "The Commission concerns itself with mankind's environment only with regard to the transfer of radionuclides through the environment, since this directly affects the radiological protection of man. The Commission believes that the standards of environmental control needed to protect man to the degree currently thought desirable will ensure that other species are not put at risk."

The underlying assumptions related to ICRP 60 are: (i) the environment is protected through the protection of humankind, (ii) reproductive capacity is the relevant endpoint, and (iii) the appropriate level of protection is to avoid endangering the existence of species or creating ecological imbalances. In the 1990 recommendations, the ICRP has not explicitly stated that the environment should be protected.

In the 2007 recommendations of the ICRP [ICRP2007], this concept is also addressed, and human protection may be considered as an indicator of environmental protection. However, the guidance suggests that it must be demonstrated directly that the environment is not affected by radiological releases. Therefore, protection of the environment shifts the focus from human to non-human biota.

Dose calculations require reference values to describe the anatomical and physiological characteristics of an exposed individual and values for tissues and organs which define a reference individual. A reference individual is not intended to describe an "average" individual, but serves to create a standard and a point of reference for dose-estimation procedures. The concept of a "reference man" is one of the cornerstones of radiological protection [ICRP2003b].

The ICRP has provided guidance on assessment of radiological impact for non-human biota [ICRP2003a]. To this end, the concept of reference animals and plants (RAPs) was introduced to provide a common reference for establishing environmental protection [ICRP2008]. ICRP initiatives include defining a reference set of dosimetric models and environmental geometries that are applied to reference animals and plants (RAP). This enables assessment of the likely consequences for individuals, the population, or the local environment. The common approach in ICRP 103 [ICRP2007] is delineated in Figure 68.

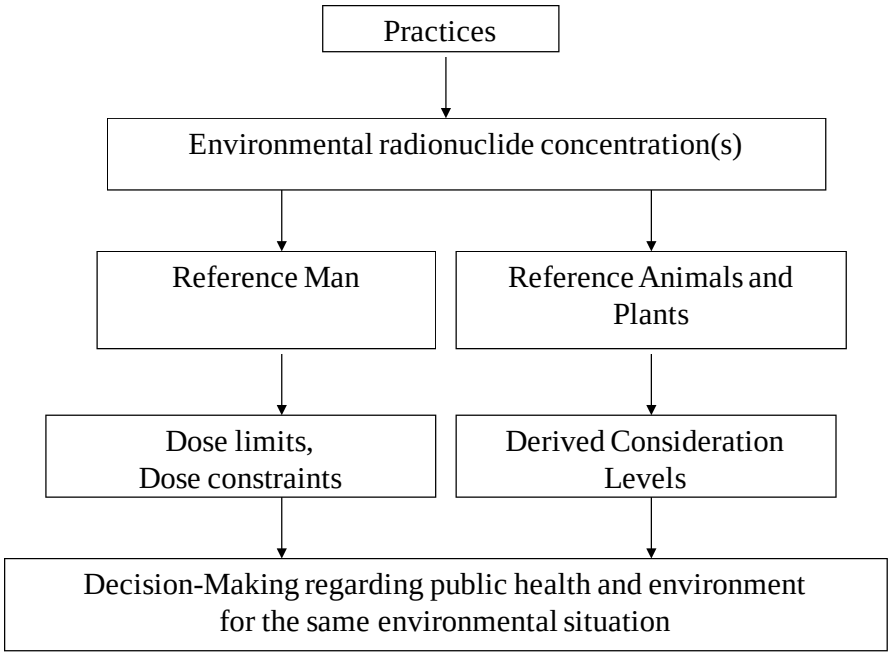

Figure 68 Common radiological assessment approach for human and non-human biota

Whereas assessment of risk for human exposure involves comparison against dose limits and constraints, non-human biota assessment involves comparison against derived consideration levels. The concept of derived consideration levels is depicted in Figure 69.

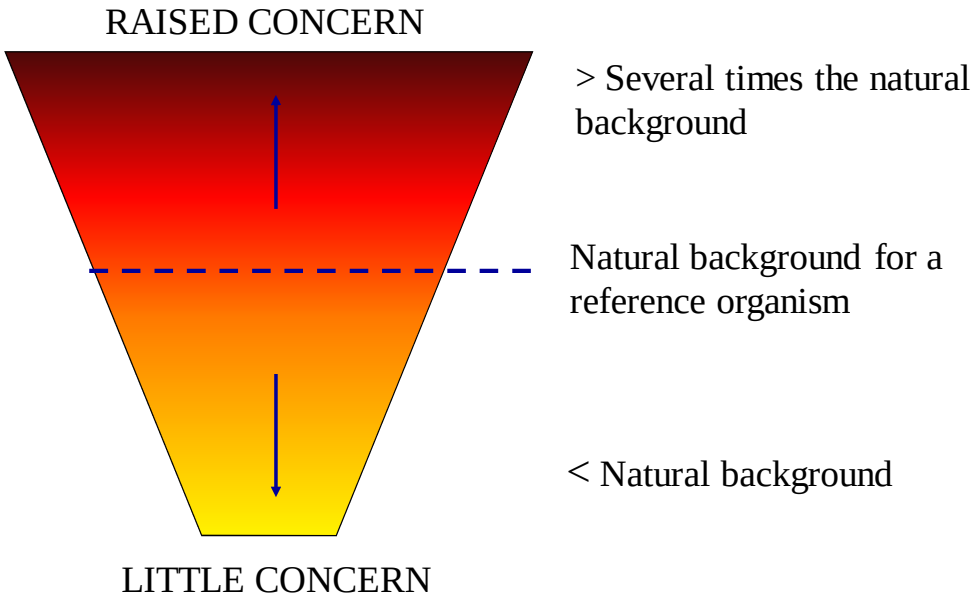

Figure 69 Derived consideration levels

In Figure 69, estimated values of dose to the reference organism that are less than the natural background are of little concern, whereas levels that are several times greater than the natural background do raise concern. The derived consideration level is a band of dose rate for a type of RAP within which there is some chance of deleterious radiation effects.

A reference animal or plant (reference organism) is a hypothetical entity with the assumed basic characteristics of a specific animal or plant, described to the taxonomic level of family,

with precisely defined anatomical, physiological, and life-history properties [ICRP2007]. RAPs are used for relating exposure to dose, and dose to effect, for that type of organism. The ICRP reference animals and plants are presented in Table 22, along with their respective wildlife groups and environments.

Table 22 Reference animals and plants (RAP)

Organism	Wildlife group	Terrestrial	Freshwater	Marine
Deer	Large terrestrial animals	X		
Rodent	Small terrestrial animals	X		
Duck	Aquatic birds	X	X	
Frog	Amphibians	X	X	
Trout	Fresh-water fish		X	X
Flatfish	Marine fish			X
Bee	Terrestrial insects	X		
Crab	Marine crustaceans		X	X
Earthworm	Terrestrial annelids	X	X	X
Pine tree	Large terrestrial plants	X		
Wild grass	Small terrestrial plants	X	X	
Brown seaweed	Seaweeds			X

It is highly likely that the future of radiological environmental monitoring programs will involve incorporating explicit consideration of RAPs to demonstrate protection of both human and non-human biota.

8.4 Summary

CANDU reactors release routine quantities of radioactive material to the environment as permitted under their licensed operating conditions. Radiological environmental monitoring programs are designed to demonstrate that there is a minimal impact of CANDU nuclear reactors as related to human exposure and that the radiological emissions are below derived

release limits established by the federal regulatory body (CNSC). Over the operating lifetime of CANDU reactors, the releases have had minimal impact on the environment, and the dose calculated for critical groups have been generally been orders of magnitude below the limits set by the Canadian Nuclear Safety Commission. The future of environmental protection around nuclear facilities in Canada may involve explicit consideration of reference animals and plants and require demonstration that a high degree of protection is afforded to non-human biota.

9 Summary of Relationship to Other Chapters

Chapter 12 is related to the following chapters in the text:

*Chapter 3 – Nuclear Processes and Neutron Physic*s

This chapter presents background data regarding particles, electromagnetic radiation, and the interactions of ionizing radiation with matter. This background information is germane to understanding the concepts developed in Sections 2 through 6.

Chapter 13 – Reactor Safety Design and Safety Analysis

This chapter discusses radiation hazards, risk, and risk models, which are related to Sections 2.3 and 2.4.

Chapter 16 – Regulatory Requirements and Licensing

This chapter is related moderately to Sections 7 and 8.

Chapter 19 – Interim Fuel Storage and Disposal

This chapter is related to Sections 7 and 8.

10 Problems

Note: Solutions to the problems in this chapter will require reference to other texts, data, materials and/or Web resources.

1. Calculate the individual electron densities and the ratio of the electron densities of a 1.5-cm-thick piece of aluminum to that of the equivalent density thickness of a piece of lead.

2. How much energy does an alpha particle require to penetrate the minimal protective epidermal layer of skin (thickness ~7 mg/cm^2)?

3. How much energy does a beta particle require to penetrate the minimal protective epidermal layer of skin (thickness ~7 mg/cm^2)?

4. Some important reactions in nuclear fuel have implications with respect to nuclear non-proliferation. Two important isotopes are Pu-239 and U-233, which are both fissionable.

a. Starting with a neutron absorption in U-238, write the decay relationship to the endpoint of Pu-239.
b. Assuming that you can have neutron absorptions in both Pu-239 and Pu-240, write down the decay relationships to the endpoint of U-233.

5. A beam of photons is incident on a slab thickness of aluminum shield on one side and exits on the other. Assume that the beam consists of photons at only two energies: 500 keV and 1000 keV, and that the incident intensities of the two energies are equal. Your goal in designing

the shield is that there be 25% intensity of the 500-keV beam component compared to the 1000-keV component on the exit side of the shield. What thickness of shield is required?

6. A primary gamma ray from the decay of K-40 is scattered twice: first, through an angle of 25 degrees, and then through an angle of 130 degrees.

a. What is the energy of the photon after the second scattering? Show intermediate energy.

b. What would the energy of the photon be if you reversed the order of scattering (i.e., first 130, then 25 degrees)? Show intermediate energy.

7. You are designing a slab shield for the outer region of a small light-water-moderated reactor. Assume that the neutron flux is 100% thermal inthe outer reactor region. What thickness of natural cadmium shield is required so that no more than 1% of the thermal flux penetrates the shield? Assume $\sigma_{capture}$ = 2450 barns.

8. Assuming that the specific heat of the body is 1 cal $g^{-1}\,{}^{o}C^{-1}$,

a. Plot the approximate temperature rise as a function of whole-body doses ranging from 100 mGy to 10 Gy;

b. What temperature rise in the body corresponds to an LD50 dose of 4 Gy? Comment on this temperature rise.

c. What whole-body absorbed radiation dose corresponds to a 3 mL sip of 60°C coffee? Show calculations for both males and females.

9. Pocket ionization chambers are built in the form of an electrical capacitor. From previous physics courses, you know that capacitance is equal to a change in charge over a change in voltage, as follows:

$$C = \frac{\Delta Q}{\Delta V},$$

where the capacitance, C, is in Farads (F), the voltage, V, is in volts (V) and the charge, Q, is in coulombs (C). You have a chamber that has a sensitive cylindrical volume with a diameter of 6.35 cm and a length of 63.5 cm. The standard density of the air in the chamber is 0.001293 g/cm^3. The chamber is calibrated so that its quartz fibre has full deflection when there is 100 µC per kg of air. The capacitance of the chamber is 1.2 nF.

a. If it takes 300 volts to charge the chamber, what is the voltage in the chamber at full-scale deflection?

b. To what exposure (in Roentgen) does full-scale deflection correspond? To what absorbed dose in air and in tissue does it correspond?

10. You have 10 GBq of each of Co-60, Cs-137, I-131, and Ir-192 (which you can assume to be point sources of radiation).

a. Compute the exposure rates, in mGy/h, at 25 cm, 50 cm, and 100 cm away from the sources (present your results in a table for full value: rows = isotope; columns = distance)

b. For each isotope at the 1-m distance, how long would it take to get to the public dose limit of 1 mSv/y (assume 1 Sv = 1 Gy)?

11. A standard-size male has an injection of 150 mCi of I-131. Assuming that 90% of this injection goes to, and is retained in, the thyroid, compute the initial dose rate (Gy/h) to the thyroid from the beta radiation.

12. You are working in a nuclear plant, and the area monitors have detected a puff release of CO_2 that has escaped from maintenance being performed on a sealed pressure tube. The only radionuclide identified in the plume is ^{14}C. You are wearing full SCBA (self-contained breathing apparatus), so that you have no possibility of getting an inhalation dose. However, it is later found that you were submerged in the plume for a total of 25 minutes. Using the assumptions listed below, determine your dose from this exposure.

Assumptions:

- ^{14}C activity detected on 1000-L air sample by area monitors = 55 kBq
- Air at STP
- Average material covering the worker has the standard thickness of a thick outer glove (*assume* that the beta attenuation coefficient for tissue will approximate this material).

13. You work in a plant that manufactures sources for radioisotope thermoelectric generators (RTGs). The source is fabricated by pressing and sintering powdered strontium titanate ($^{90}SrTiO_3$). A spill of the powdered material occurred in the plant after a shielded vessel which transports the raw material from shipping and receiving to the handling area fell over and broke open. While the person who was handling the material was trying to find a manager, one maintenance worker (Worker A), who was not wearing personal protective gear, worked near the spill (which was spread out evenly on the floor). You are the health physicist charged with determining the potential exposure to the worker. You interview the worker and co-workers and take measurements near the spill. You gather the following data to use in your analysis:

- Sr-90 contamination measured: 2 MBq cm^{-2} (assume uniform)
- Approximate distance of Worker A's hands from the spill: 1 foot
- Approximate time Worker A's hands spent in the vicinity of the spill: 12 minutes

a. Calculate the dose rate to Worker A's hands using the above data.
During your interview, you also determine that Worker A came into brief contact with the spill and contaminated her hands. She washed her hands immediately after coming in contact with the spill (two hours ago). You estimate that casual hand washing is only 65% effective in removing this type of contamination. You immediately assist her to decontaminate her hands with some strong detergent and a scrub pad.
b. Calculate the dose to Worker A's hands from this contact exposure.

14. A solution is being prepared using 0.835 GBq of Ba-133. The solution spread into a roughly circular area of diameter 50 cm. What is the maximum dose equivalent rate (SI units) 50 cm above the spill?

15. You have the following sources with associated activities:

Bq	Source

5.00E+09	Co-60
7.50E+09	Ir-192
1.00E+09	Na-22
5.00E+10	Tc-99m
7.50E+10	Au-198
1.00E+11	Ra-226

a. Compute the exposure rates, in mGy/h, at 50 cm away from these sources (present your results in a table for full value: rows = isotope; columns = distance)

b. For each isotope at the 50-cm distance, how long would it take to get to the public dose limit of 1 mSv/y (assume 1 Sv = 1 Gy)?

16. You are asked to decommission an old radioactive-waste storage room, and inside you find a large lead pig (sphere). To dispose of the source, you need to know both the radioisotope and the activity. Unfortunately, there is no documentation for this source, and there are no markings on the pig. You take a dose-rate measurement at about 4 feet away from the surface of the lead sphere and find that it is 100 μGy/h. You assume that there is a point gamma source at the centre of the sphere, and using a ruler, you determine the diameter of the sphere to be about 20 cm. Using your portable GR-135 gamma spectrometer, you determine the isotope to be Co-60 by identification of the 1.17 and 1.33 MeV peaks. What do you estimate the activity of this source to be (SI units)? (*Hint: do not neglect buildup*).

17. You are responsible for health physics at a nuclear plant. You have just been told that there are measurable levels of ^{14}C in air in the form of carbon dioxide. This has previously not been a hazard, and therefore there are no derived limits for this radioisotope. Estimate

a. the allowable intake limit (Bq), and

b. the derived air concentration (Bq/m^3) for $^{14}CO_2$ using the stochastic limit for nuclear energy workers at your plant (20 mSv/annum). *Note: Assume that after inhalation, the $^{14}CO_2$ is highly soluble and transfers directly into the blood*.

You have been told that two workers, who did not know there was an airborne $^{14}CO_2$ hazard, performed their duties in a 2-DAC environment. Worker "A" was wearing a half-face air-purifying respirator with P100 (purple) cartridges while performing his duties for three hours in the morning and four hours in the afternoon. A welder (Worker "B") was wearing an air-line, continuous-flow, half-mask respirator while performing her duties for five total hours.

c. Estimate the committed effective dose from this hazard for both Worker A and Worker B (*assume no external hazard*).

18. A standard-size male working in a nuclear plant breathes in 100 mCi of Kr-87. Assuming that 75% of this intake is retained in the lung mass, compute the initial dose rate (Gy/h) to the lungs from the beta radiation (N.B.: only use betas with a probability of emission greater than 1%).

19. Calculate the allowable limit on intake for tritium (HTO vapour) in Bq using the stochastic limit for nuclear energy workers of 20 mSv per annum. Assume that the source is whole-body and the target is whole-body.

20. The DAC value for tritiated water is typically adjusted to allow for the fact that a person working in a tritiated atmosphere will absorb half as much tritium through the skin as through inhalation (factor of 1.5). For example, if a person inhales 100 Bq of tritium, another 50 Bq will have been absorbed through the skin (i.e., 150 Bq total). Using the above, what is the DAC for tritium (Bq/m^3)?

21. A worker is performing his duties in a tritiated atmosphere of 1 DAC for eight hours wearing a positive-pressure (PP) air-purifying full-facepiece respirator. What is the worker's committed dose from his exposure to this environment?

22. You are the health physicist at a nuclear power station. The chemistry manager has asked you to review a purchase requisition that has been initiated for a nitrogen-16 calibration source. The source generates N-16 through an (α,p) reaction involving curium-244 (spontaneous fission radioisotope) and carbon-13. The source gamma-emission strength is 2.2×10^6 γ/s, and the neutron-emission strength is 2.0×10^5 n/s. Assume a gamma energy of 6.1 MeV and an average neutron energy of 2.5 MeV. Data provided are:

- Neutron flux-to-dose equivalent (at 2.5 MeV);
  20 n/cm^2-s = 25 μSv/h.

  a. Write the stoichiometric relationship for the N-16 production reaction.

  b. Calculate the total gamma dose-equivalent rate at 1 foot (μSv/h), assuming 100% emission rate from the principal gamma peak.

  c. Calculate the total neutron dose-equivalent rate at 1 foot (mRem/h).

  Both lead and polyethylene are available to shield the source.

  d. Which shielding arrangement (from the source outward) listed below would be expected to yield the lowest overall dose rate? Explain your answer.

    i. Lead only

    ii. Polyethylene followed by lead

    iii. Polyethylene only

    iv. Lead followed by polyethylene

    v. No shielding is necessary because the 12-inch air gap will sufficiently scatter or attenuate the neutrons.

  e. If the source is surrounded by 3.93 cm of lead, by what percentage will the gamma dose rate at 1 foot be reduced?

23. The radioisotope ^{140}La is a common fission product that requires shielding. For La-140, find the:

- Physical half-life (h)
- Decay scheme, and
- Average beta energy (keV).

a. Calculate the specific gamma constant for La-140 using the four (4) most dominant gamma lines.

24. List the (strongest transition) capture gamma-ray energy (MeV) and capture cross section (barns) for the following isotopes: H-1, B-10, Cd-113, Gd-157. Answer the following questions:

a. Why are cadmium and gadolinium used by control and safety systems in nuclear reactors? Where are they used?

b. With respect to B-10, discuss the principal thermal-neutron capture reactions and discuss the importance of the (n,α) reaction for shielding. What other practical use does the ^{10}B(n,α) reaction have?

25. Derive the working rule for specific gamma constant in classical units (i.e., $\Gamma = 0.5\sum_i E_i Y_i$).

Show all work and units (i.e., be explicit about where the factor of 0.5 comes from).

26. To account for the number of atoms in a material, the density of atoms in the material must be deduced. This is necessary because the interaction cross section for a material depends on the number of atoms that may potentially interact with the transiting radiation per unit path length. The terminology "mixing cross sections" refers to the proportion of the cross section assigned to each element in a material. For example, water has two atoms of hydrogen and one atom of oxygen per molecule in water, and both hydrogen and oxygen have interaction cross sections that are a function of energy and radiation type. The number density of a single-element material is typically expressed as follows:

$$N = \frac{\rho \cdot A_v}{MW},$$

where

N is the atom density (atoms/cm^3)
ρ is the mass density (g/cm^3)
A_v is Avogadro's number (6.02E23 atoms/mole)
MW is the molecular weight of the element (atoms/mole).

For compounds or mixtures, the number density of the i^{th} element in the mixture can be calculated using the weight fraction w_i of the i^{th} element in the mixture. The weight fraction can be determined as:

$$w_i = \frac{n_i A_i}{MW},$$

where

n_i is the number of atoms of element *i* in the mixture;
A_i is the molecular mass of element *i* in the mixture (g/mole); and
MW is the molecule mass of the mixture (g/mole).

The number density of the i^{th} constituent in the mixture is therefore given as:

$$N_i = w_i \frac{\rho \cdot A_v}{MW_i},$$

where N_i is the number density (atoms/cm^3) of the i^{th} constituent in the mixture (atoms/cm^3). It is common to express the number density in terms of (atoms/b-cm), and therefore:

$$N_i = w_i \frac{\rho \cdot A_v}{MW_i} \left(\frac{atom}{cm^3} \right) \cdot 10^{-24} \left(\frac{cm^2}{barn} \right) = w_i \frac{\rho \cdot A_v \cdot 10^{-24}}{MW_i} \left(\frac{atom}{b-cm} \right).$$

Note that atom density is the same thing as number density. The macroscopic cross section for the i^{th} element is:

$$\Sigma_i = N_i \cdot \sigma_i \left(\frac{1}{cm} \right).$$

This is related to material mixing in that the macroscopic cross section for the mixture is proportional to the individual number densities as:

$$\Sigma = \sum_i N_i \cdot \sigma_i \left(\frac{1}{cm} \right).$$

Note that depending on the nomenclature used, $\mu = \Sigma$.

In addition, many computer-based codes require, for material definition, the elemental compositions in terms of weight fraction or number (atomic) density, and therefore it is very useful to know how to compute these (because this is how the computer-based code will use the cross sections!).

a. Show that the weight fraction of oxygen is 0.888 and that of hydrogen is 0.112 in water.

b. Calculate the atom fractions of oxygen and hydrogen in water.

c. Calculate the number densities (atoms/b-cm) of oxygen and hydrogen in water.

27. Using the following absorption and scatter cross sections, determine the macroscopic absorption, scattering, and total cross sections for H_2O. What is the mean free path of neutrons in water based on your calculation?

28. Define and/or briefly explain the significance of the following:

a. ICRP

b. IAEA

c. NCRP

d. WHO

e. MARSSIM

f. MARLAP

g. RESRAD

h. Barriers between outer and inner body

i. Basis for different types of environmental monitoring programs and the concept of an environmental baseline study

j. Three general types of environmental samples

k. Classical method to determine whether a radioactive release has occurred from a nuclear plant accident or a weapon detonation

l. Work disciplines that make up environmental health physics.

29. It is usual to evaluate airborne derived release limits (or derived emission limits) by first evaluating the all-relevant-compartment values in units of $(Sv\ a^{-1})/(Bq\ m^{-3})$ and multiplying the sum by the airborne dispersion factor (P01, in $s\ m^{-3}$) to obtain [X9/X0] in units of $(Sv\ a^{-1})/(Bq\ s^{-1})$. The DRL is then evaluated using the annual dose limit. The applicable transfer factors are given below:

Transfer factor	Description	Value	Units
P(i)19	Dose by inhalation	1.5E-11	$Sv\ Bq^{-1}$
P(e)19	Dose by immersion	Nil	
P49, P59	Dose by ingestion	1.6E-9	$Sv\ Bq^{-1}$
P13, P34	Air to soil to forage, crops	No pathway	
P39	External dose from ground	Nil	
P14	Air to forage	2.75E+3	$m^3\ kg^{-1}$
	Air to crops	3.75E+2	$m^3\ kg^{-1}$
P45	Forage to milk	0.15	$kg\ kg^{-1}$
	Forage to meat	0.64	$kg\ kg^{-1}$
P15	Cow inhalation to milk	4.2	$m^3\ kg^{-1}$
	Cow inhalation to meat	5.1	$m^3\ kg^{-1}$
P01	Source-Atmosphere	2.0E-7	$s\ m^{-3}$

Data are provided as follows:

DRL Data

Parameter	Adult	Infant
Vegetables consumed, fraction grown locally	0.3	0.1
Meat consumed, fraction locally produced	1.0	0.1

Milk consumed, fraction locally produced	1.0	1.0
All vegetables, kg per year	200	84
All meat, kg per year	70	24
All milk, kg per year	170	220
Breathing rate, m^3 per year	8400	1400

Determine the DRL for ^{14}C in the airborne pathway, assuming an infant critical group at 1 km from the nuclear station and all produce and animals at 1 km (i.e., all data apply for 1 km).

30. A nuclear power plant located on Lake Ontario operates an environmental monitoring program which includes annual fish sampling. At the laboratory, the fish are analyzed for the following radionuclides:

- Gamma emitters
- Sr-90/Y-90
- Carbon-14
- Tritium.

One year, some fish samples (lake trout) are found to have the following radionuclide concentrations:

- Cs-137: 3.33 ± 0.19 Bq/kg (wet weight)
- ^{14}C: 250 ± 30 Bq/kgC.
- Tritium: 241 ± 6 Bq/L (free water)
- Sr-90/Y-90: Less than minimum detectable amount.

Relevant data are provided as follows:

Dose factors (Sv/Bq)

	E 10y	E 15y	E Adult
Tritium	2.4E-11	1.8E-11	1.8E-11
^{14}C	8.5E-10	5.6E-10	5.6E-10
Cs-137	1E-8	1.3E-8	1.3E-8
SR-90	5.2E-8	2.6E-8	1.6E-8
Y-90	9E-9	5.4E-9	4.2E-9

Background concentrations:
Tritium in Lake Ontario water: 1.0 Bq/L
Carbon-14 in living material: 24.0 Bq/kgC.
Cs-137 in trout from Lake Ontario: 10. Bq/kg (wet weight)
Station cooling water discharge rate: 10,000 m^3/min
Dilution factor: 30
Cs-137 bioaccumulation factor for Lake Ontario: 10,000
Assume that fish are 90% water
Assume that fish are 45% carbon

Lake Ontario fish consumption by Boy Scouts: 10 kg/y

Calculate the committed dose attributable to radioactivity from the nuclear power plant to Boy Scouts from a nearby camp eating the fish (*assume that any radionuclide concentration above background is due to releases from the nuclear power plant*).

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13 Acknowledgements

The following reviewers are gratefully acknowledged for their hard work and excellent comments during the development of this Chapter. Their feedback has much improved it. Of course the responsibility for any errors or omissions lies entirely with the authors.

Kam Aydogdu
Esam Hussein

Thanks are also extended to Diana Bouchard for expertly editing and assembling the final copy.