RADIATION SAFETY INSTITUTE
OF CANADA

INSTITUT DE RADIOPROTECTION DU CANADA

Radiation Safety Officer

(RSO-1)

Training Textbook

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Seventh Revision

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Radiation Safety Institute of Canada
Institut de radioprotection du Canada
National Office, Toronto

Training Course

Radiation Safety Officer
(RSO-1)

Structure of Matter

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Sixth Revision

Structure of Matter

CONTENTS

1.0 The Atom

All matter in our world is comprised of *atoms*.

Atoms are tiny objects, so small that they cannot be seen with the most powerful microscope. It takes vast numbers of atoms to create even the simplest object.

An atom is the smallest quantity of an element which still retains the chemical properties of that element.

Scientists have studied atoms for centuries. At present, there are approximately 100 different kinds of atoms that can form different types of *elements* (materials). For example, there are iron atoms, oxygen atoms, hydrogen atoms, and uranium atoms, just to mention a few. Each of these atoms is different in size and shape as well as in other respects. Elements are different from each other because they are made of different atoms.

Atoms are composed of *protons*, *neutrons*, and *electrons*. The protons and neutrons are contained in the *nucleus* which is located at the centre of the atom. The protons and neutrons are commonly referred to as *nucleons* because they make up the nucleus of the atom. The nucleus is surrounded by a "cloud" of electrons. The electrons circle the nucleus in orbits in much the same manner as the planets orbit the Sun. These electrons are commonly referred to as *orbital electrons*. An example of an atom is shown in Figure 1.

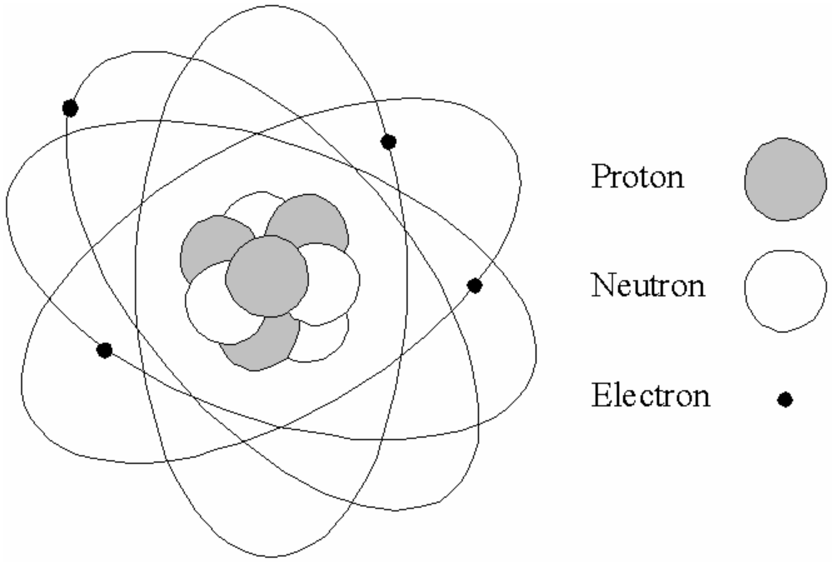


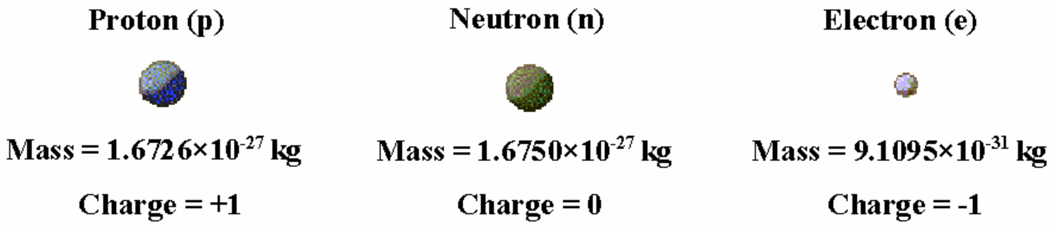
Figure 1: The atom. The atom shown is beryllium. The beryllium atom contains four protons and four neutrons in the nucleus with four electrons orbiting the nucleus.

Protons, neutrons, and electrons are the building blocks of the atom. Following is a summary of their properties (Figure 2).

Proton: The proton is a positively charged particle found in the nucleus of an atom.

Neutron: The neutron is an electrically neutral particle which has approximately the same mass as the proton. The neutron is also found in the nucleus of an atom.

Electron: The electron is a negatively charged particle which orbits around the nucleus of an atom. The electron is very light, being approximately 2000 times lighter than either the proton or the neutron.



Proton (p)	Neutron (n)	Electron (e)
Mass = 1.6726×10^{-27} kg	Mass = 1.6750×10^{-27} kg	Mass = 9.1095×10^{-31} kg
Charge = +1	Charge = 0	Charge = -1

Figure 2: Atomic building blocks.

It is interesting to note that protons and neutrons themselves are made up of other particles known as *quarks* (Figure 3). Quarks and electrons are, at present, believed to be *elementary particles*.

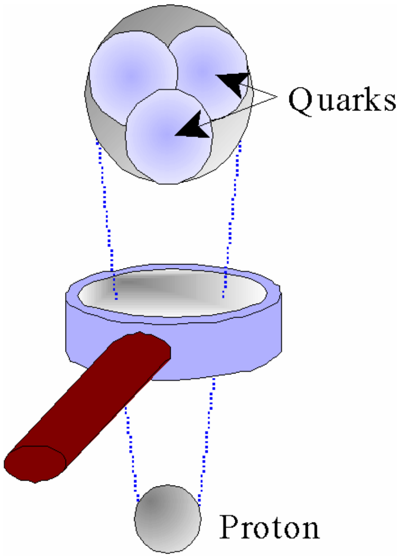


Figure 3: Quarks.

Scientists believe that these elementary particles are the fundamental building blocks of matter.

The simplest atom is the hydrogen atom. The hydrogen atom is made up of one proton in the nucleus and one orbital electron (Figure 4).

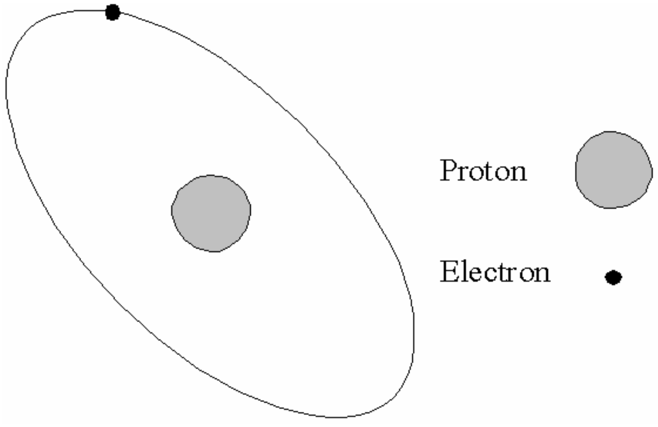


Figure 4: The hydrogen atom.

The second simplest atom is that of helium (Figure 5). The helium atom contains two protons and two neutrons in the nucleus and two orbital electrons around the nucleus.

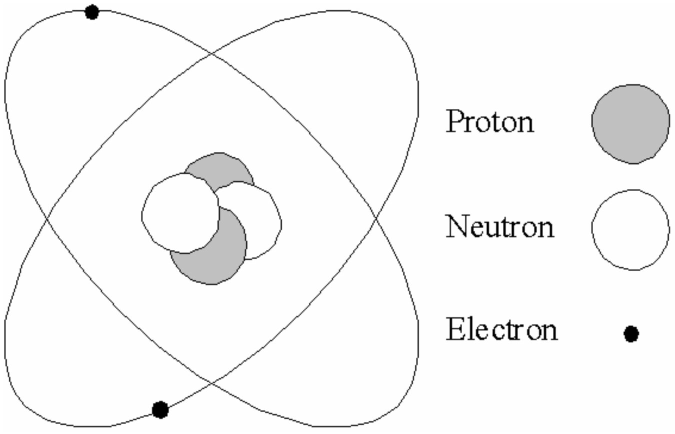


Figure 5: The helium atom.

As we will learn, the primary difference between various types of atoms is the number of protons and neutrons in the nucleus and the number of orbital electrons.

2.0 Size of the Atom

It has already been mentioned that atoms are very tiny objects. Ten million (10^7) hydrogen atoms placed side by side would form a line one millimetre in length. But as small as the hydrogen atom is, its diameter is 50,000 times greater than the diameter of its nucleus. This shows that atoms consist mostly of empty space. The relative sizes of the different components of the atom are illustrated in Figure 6.

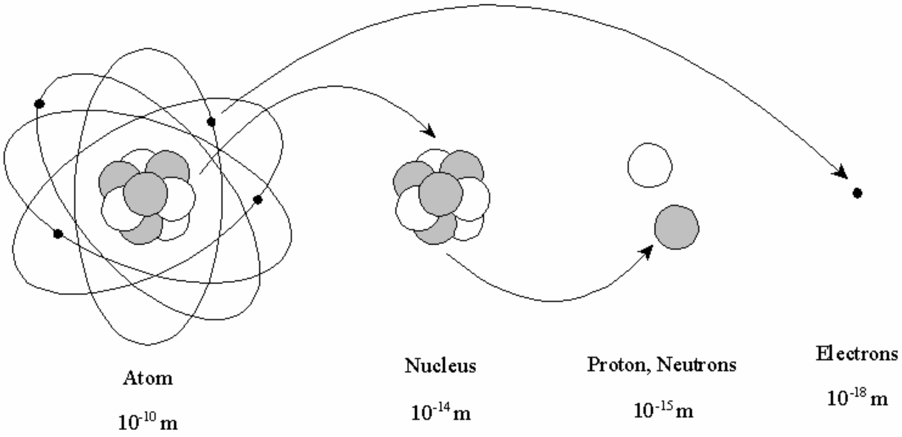


Figure 6: Size of the atom. The approximate sizes of the atom and its components are listed in units of metres (m).

Further, the nucleus of an atom constitutes approximately 99.9% of the mass of the atom.

Compare the size of the atom with something more familiar. If a hydrogen nucleus was the size of the Earth and in the position of the Sun, the electron orbiting the nucleus would be circling at a distance of 480 million kilometres. One can compare this to the distance between the Earth and the Sun which is 150 million kilometres (Figure 7).

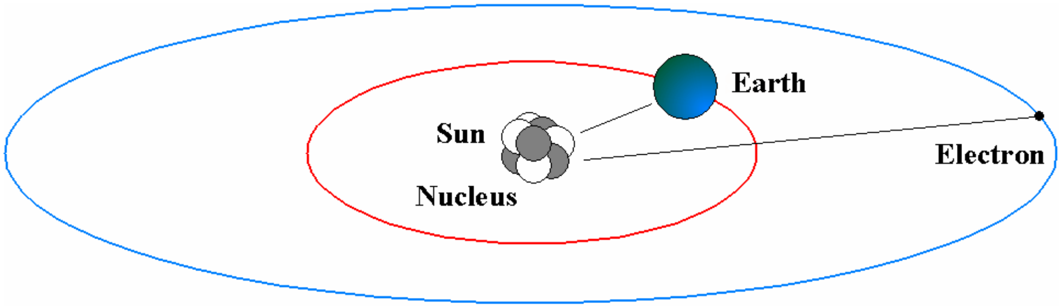


Figure 7: Atoms are mostly empty space. The distance between the nucleus of a hydrogen atom and the electron are shown in relation to the distance between the Earth and the Sun.

3.0 Electrical Balance of the Atom

There are two kinds of electric charges, positive (+) and negative (-). When two objects are both positively charged, they repel one another. Similarly, two negatively charged objects also repel one another. However, when one object has a positive charge and one object has a negative charge, the two objects will attract each other (Figure 8).

This concept can be summarized by the statement: *like charges repel and unlike charges attract.*

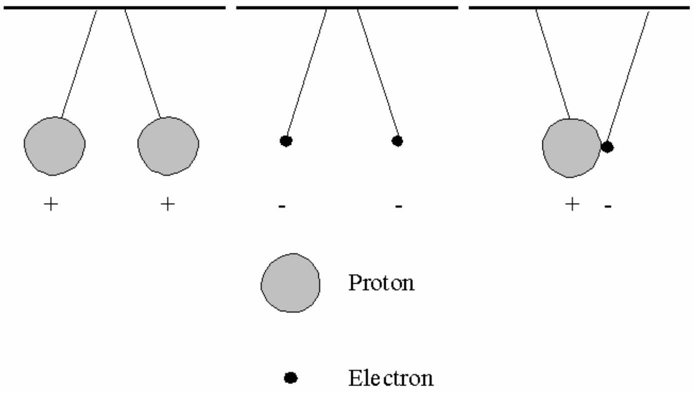


Figure 8: Electrical charges.

Protons have a positive charge and electrons have a negative charge. The negative charge of the electron is equal but opposite to the positive charge of the proton. The neutron has no electric charge and is thus electrically neutral. It is because protons and electrons have opposite charges that electrons are held in orbit around the nucleus of an atom. This phenomenon is similar to the gravitational force that holds the Earth in orbit around the Sun.

For each proton in the nucleus, there is one orbital electron. Therefore, the number of orbital electrons in an atom is equal to the number of protons in the nucleus. Because of this one-to-one correspondence, the total positive charge of the nucleus is equal to the total negative charge of the orbital electrons. The atom is, therefore, *electrically neutral*.

If an atom were short an electron for any reason, it would have a net positive charge. The positively charged atom would attract a loose electron from nearby and pull it into orbit, thereby cancelling its excess positive charge.

Question**:** *If like charges repel, what keeps the protons together in the nucleus?*

The answer to this question is still a topic of much research, but it is believed that the repulsive force resulting from the positively charged protons being together in the nucleus is overcome by an attractive force called the *nuclear force*. Protons and neutrons carry the nuclear forces that overcome the repulsive electrical forces and hold the nucleus together. The nuclear forces are very short ranged and have no direct effect beyond the boundary of the nucleus.

4.0 Atomic Mass

We have already learned that the atom is composed of protons, neutrons and electrons, and that the atom and its components are very small. So small, in fact, that it takes literally billions and billions of atoms to make up even the simplest object. It should not be surprising then to learn that the mass of a single atom or its components is also extremely small.

For example, one hydrogen atom has a mass of 1.66×10^{-27} kg. One oxygen atom is 2.66×10^{-26} kg and one uranium atom is 3.95×10^{-25} kg.

The mass of any atom is the sum of the masses of its constituent parts. The mass of the proton and the mass of the neutron are very nearly equal: 1.67×10^{-27} kg (see Figure 2). The mass of the electron, 9.11×10^{-31} kg, is approximately 2000 times less than the mass of the proton or the neutron. In fact, the mass of the electron is so small in comparison to the masses of the proton and the neutron that it can be ignored in the determination of the mass of an atom.

Because the masses of atomic particles are so small, it is impractical to deal with them in units of kilograms (kg). Instead, scientists have created a mass unit called the *atomic mass unit*, abbreviated to *amu*. One atomic mass unit (amu) is equal to 1.6605×10^{-27} kg which is very close to the masses of the proton and the neutron. As a result, the masses of these particles are approximated as 1 amu.

By knowing the number of protons and neutrons in the nucleus of an atom, one can quickly determine the mass of the atom (atomic mass).

For example, the nucleus of a hydrogen atom only contains one proton which has a mass of 1 amu. Therefore, hydrogen has an atomic mass of 1 amu. Helium atoms contain two protons and two neutrons. Each nucleon (proton or neutron) has a mass of 1 amu. Therefore, the atomic mass of helium is 4 amu. An oxygen atom contains eight protons and eight neutrons and, therefore, has an atomic mass of 16 amu.

The total number of nucleons is referred to as the *mass number*. By definition:

> *The mass number of an atom is equal to the number of protons plus the number of neutrons in its nucleus. The mass number is designated by the letter* ***A.***

The mass number can be represented by the equation:

$$A = Z + N$$

The term **Z** denotes the a*tomic number*. By definition:

> *The atomic number, **Z**, of an atom is the number of protons in its nucleus.*

The number of neutrons in the nucleus, sometimes referred to as the *neutron number*, is denoted by the term **N**.

<u>Example 1</u>:

Oxygen has eight protons and eight neutrons. What is the mass number of oxygen?

$$\begin{aligned}&A = Z + N,\quad Z = 8,\quad N = 8\\&A = 8 + 8\\&A = 16\end{aligned}$$

<u>Example 2</u>:

Carbon has a mass number of 12 and there are 6 neutrons. What is the atomic number of carbon?

$$\begin{aligned}&Z = A - N,\quad A = 12,\quad N = 6\\&Z = 12 - 6\\&Z = 6\end{aligned}$$

<u>Example 3</u>:

Uranium has a mass number of 238 and an atomic number of 92. How many neutrons are there in an atom of uranium?

$$\begin{aligned}&N = A - Z,\quad A = 238,\quad Z = 92\\&N = 238 - 92\\&N = 146\end{aligned}$$

5.0 The Elements

If a large number of identical atoms, all with the same number of protons, are assembled together, the substance formed by these atoms is referred to as an *element*.

By definition:

> *An element is a form of matter whose atoms all have the same number of protons, or the same atomic number.*

The chemical properties of an element are determined by the number of orbital electrons in the nucleus of the atom which is equal to the atomic number. Varying the number of neutrons in the atom does not change the element or its chemical properties. However, varying the number of protons in an atom does change the physical properties of atom. In fact, changing the number of protons results in a different element.

As already mentioned, there are over 100 different elements. Most of these elements occur naturally; however, some elements are man-made in nuclear reactors and particle accelerators.

A list of the known elements is given in the *periodic table* shown in Figure 9. A list of known elements is also shown in Table 1.

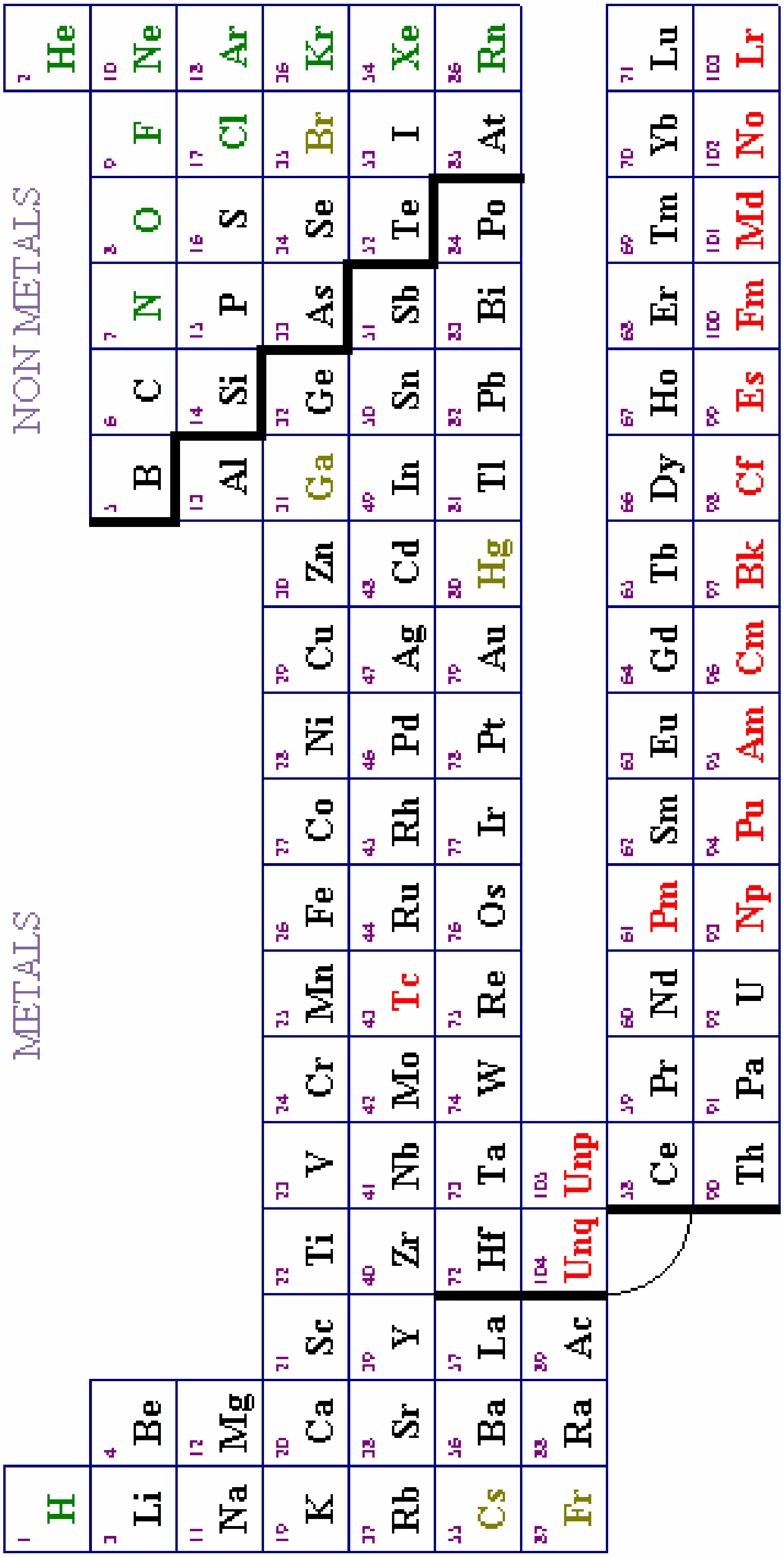


Figure 9: Periodic table.

Table 1: List of Elements.

Z	Symbol	Name	Z	Symbol	Name	Z	Symbol	Name
1	H	hydrogen	36	Kr	krypton	71	Lu	lutetium
2	He	helium	37	Rb	rubidium	72	Hf	hafnium
3	Li	lithium	38	Sr	strontium	73	Ta	tantalum
4	Be	beryllium	39	Y	yttrium	74	W	tungsten
5	B	boron	40	Zr	zirconium	75	Re	rhenium
6	C	carbon	41	Nb	niobium	76	Os	osmium
7	N	nitrogen	42	Mo	molybdenum	77	Ir	iridium
8	O	oxygen	43	Tc	technetium	78	Pt	platinum
9	F	fluorine	44	Ru	ruthenium	79	Au	gold
10	Ne	neon	45	Rh	rhodium	80	Hg	mercury
11	Na	sodium	46	Pd	palladium	81	Tl	thallium
12	Mg	magnesium	47	Ag	silver	82	Pb	lead
13	Al	aluminum	48	Cd	cadmium	83	Bi	bismuth
14	Si	silicon	49	In	indium	84	Po	polonium
15	P	phosphorus	50	Sn	tin	85	At	astatine
16	S	sulfur	51	Sb	antimony	86	Rn	radon
17	Cl	chlorine	52	Te	tellurium	87	Fr	francium
18	Ar	argon	53	I	iodine	88	Ra	radium
19	K	potassium	54	Xe	xenon	89	Ac	actinium
20	Ca	calcium	55	Cs	cesium	90	Th	thorium
21	Sc	scandium	56	Ba	barium	91	Pa	protactinium
22	Ti	titanium	57	La	lanthanum	92	U	uranium
23	V	vanadium	58	Ce	cerium	93	Np	neptunium
24	Cr	chromium	59	Pr	praseodymium	94	Pu	plutonium
25	Mn	manganese	60	Nd	neodymium	95	Am	americium
26	Fe	iron	61	Pm	promethium	96	Cm	curium
27	Co	cobalt	62	Sm	samarium	97	Bk	berkelium
28	Ni	nickel	63	Eu	europium	98	Cf	californium
29	Cu	copper	64	Gd	gadolinium	99	Ex	einsteinium
30	Zn	zinc	65	Tb	terbium	100	Fm	fermium
31	Ga	gallium	66	Dy	dysprosium	101	Md	mendelevium
32	Ge	germanium	67	Ho	holmium	102	No	nobelium
33	As	arsenic	68	Er	erbium	103	Lr	lawrencium
34	Se	selenium	69	Tm	thulium			
35	Br	bromine	70	Yb	ytterbium			

6.0 Isotopes

We have already learned that all atoms in an element have the same number of protons or atomic number (Z). If we were to vary the number of neutrons in the nucleus while keeping the number of protons the same, the resulting atoms would have a different mass number (A), but would still be the same element. Members of an element with the same atomic number but with different numbers of neutrons are referred to as *isotopes*.

An isotope of an element is an atom which has the same atomic number (Z) as other atoms of the element, but a different mass number (A). This is caused by one isotope having more (or fewer) neutrons than another isotope of the same element.

Isotopes of the same element have the same chemical properties but may have different physical properties such as boiling point and freezing point. Isotopes of an element all generally have different nuclear properties. The nuclear properties of isotopes will be discussed in more detail later.

In writing symbols for elements and the isotopes of an element, the following notation is used:

$$^{A}_{Z}\text{Symbol}$$

The word "Symbol" denotes the particular abbreviation for the element, the chemical symbol (Table 1). The superscript "A" refers to the mass number of the isotope. The subscript "Z" refers to the atomic number of the isotope.

For example, $^{12}_{6}\text{C}$ is the notation for one of the isotopes of carbon. This particular isotope of carbon has a mass number of 12 and an atomic number of 6. The isotope of uranium $^{238}_{92}\text{U}$ has a mass number of 238 and an atomic number of 92.

The simplest of all atoms is that of hydrogen. There are three isotopes of hydrogen: $^{1}_{1}\text{H}$, $^{2}_{1}\text{H}$, $^{3}_{1}\text{H}$ (Figure 10). All isotopes of hydrogen have an atomic number of one (Z = 1). The isotope $^{2}_{1}\text{H}$ has a mass number of two, whereas the isotope $^{3}_{1}\text{H}$ has a mass number of three.

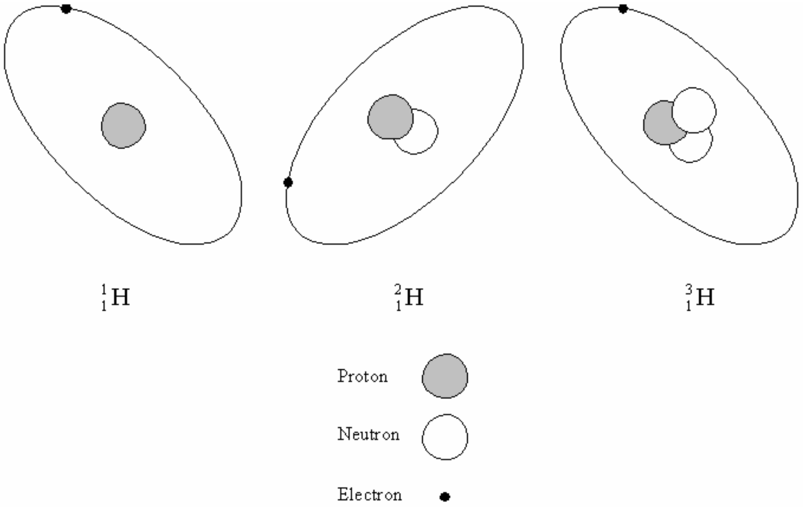


Figure 10: Isotopes of hydrogen.

Naturally occurring uranium ore contains three isotopes: $^{238}_{92}U$, $^{235}_{92}U$, and $^{234}_{92}U$. The isotope $^{238}_{92}U$ is the most abundant, comprising 99.274% of all naturally occurring uranium. The isotopes $^{235}_{92}U$ and $^{234}_{92}U$ have abundances of 0.720% and 0.006% respectively.

A partial chart of isotopes is shown in Figure 11. There are about 300 stable and naturally occurring isotopes of elements. All of the different types of isotopes are referred to collectively as *nuclides*.

By definition:

> *A nuclide is an atom characterized by the number of neutrons and protons it contains.*

Each of the 300 types of isotopes mentioned above is a separate nuclide because each has a distinctive nucleus which differs from all others in its number of protons, its number of neutrons, or both.

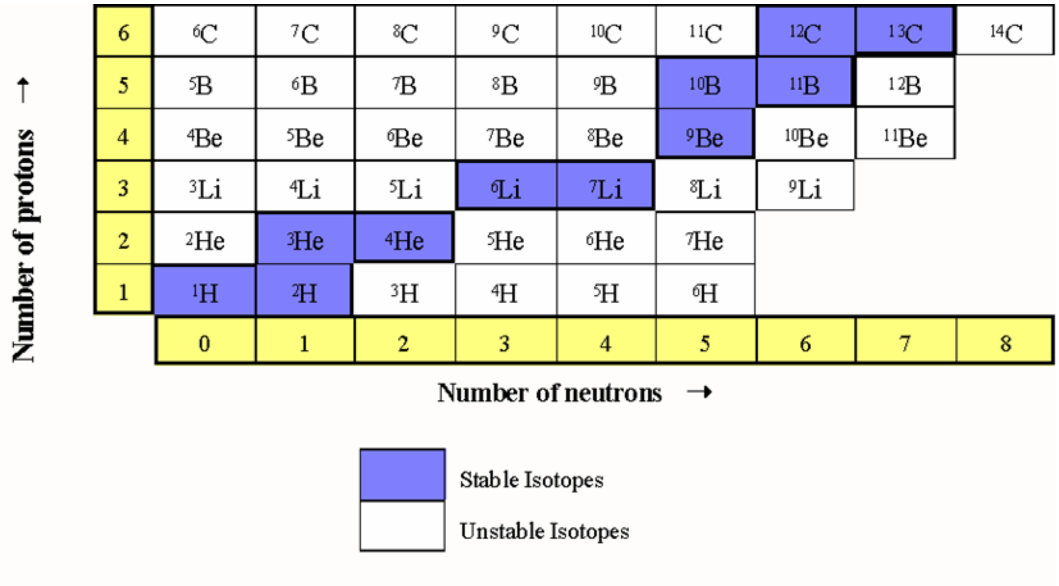


Figure 11: Partial chart of isotopes.

7.0 Stable and Radioactive Nuclides

Most of the nuclides in nature are *stable* isotopes. To understand what is meant by stable, one must first consider the *nuclear force* that holds the protons and neutrons in the nucleus of an atom together. It has already been mentioned that protons and neutrons carry certain nuclear charges and the force fields due to these charges are attractive in nature. It is this nuclear force that provides the cohesive forces that overcome the repulsive electrical forces and hold the nucleus together.

If the nuclear forces between the protons and neutrons are greater than the electric repulsive force of the protons, the nucleus will remain intact and unchanging. In other words, the nuclide is said to be *stable*. For example, $^{1}_{1}H$ is a stable isotope of hydrogen. Similarly, $^{12}_{6}C$ is a stable isotope of carbon. These stable isotopes will remain unchanged indefinitely because the nucleus is stable.

It has been found that a nuclide is stable if it has a certain number of neutrons in relation to its number of protons. For light nuclei, the number of neutrons and protons are approximately equal (for example, $^{16}_{8}O$ has 8 protons and 8 neutrons). Whereas for stable heavy nuclei, there must be approximately 1 ½ times as many neutrons as protons (for example, $^{208}_{82}Pb$ has 82 protons and 126 neutrons).

If a nuclide is not stable, it is referred to as *unstable* or *radioactive*. Radioactive nuclides are nuclides in which there is an imbalance between the number of neutrons and protons in the nucleus. In other words, the neutron/proton ratio in the nucleus is not optimum and the nuclear forces are less than the repulsive electrostatic forces.

These unstable nuclides attempt to become stable by emitting *radiation*. Radiation, simply put, is subatomic particles or energy that originates from the nucleus of an unstable nuclide. Radiation can be interpreted as a form of energy given off by the isotope. By emitting radiation, the nuclide alters the neutron/proton ratio of the nucleus in an attempt to become more stable.

In summary, substances made up of unstable atoms are said to be *radioactive*. The particles or energy emitted from an unstable atom are called *radiation*.

Radioactive nuclides are commonly referred to as *radionuclides*. A nuclide stability plot is shown in Figure 12. In the figure, the number of protons in the nucleus is plotted against the number of neutrons for all known isotopes. The jagged dark line represents proton/neutron mix for stable nuclides. Nuclides on either side of the line are unstable or radioactive.

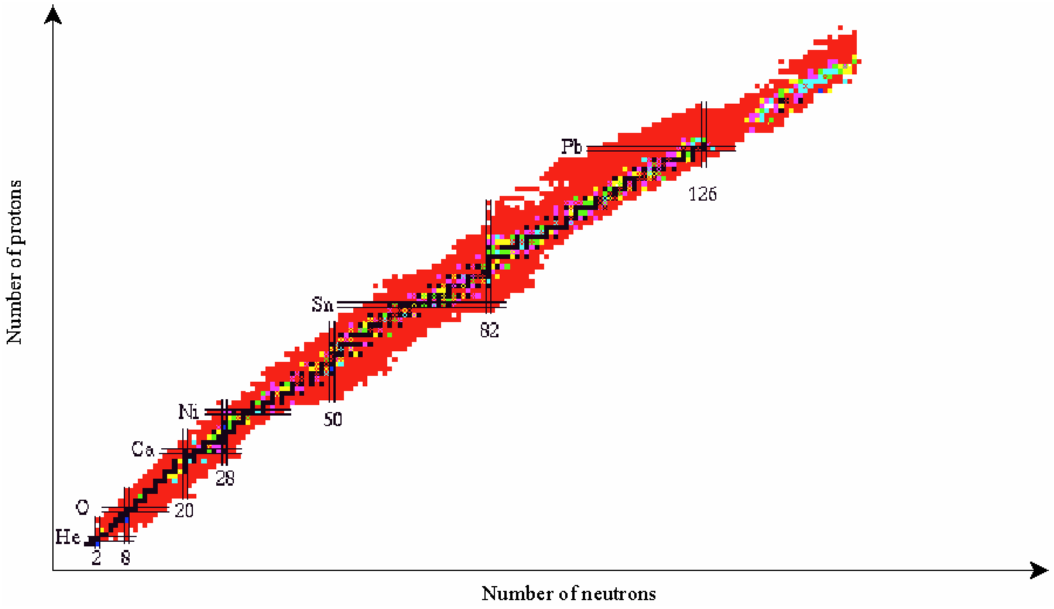


Figure 12: Nuclide stability plot.

Glossary

Atom	The atom is the smallest quantity of an element which still retains the chemical properties of that element.
Atomic Number	The atomic number of an atom is the number of protons in its nucleus. The atomic number is designated by the letter **Z**.
Electron	The electron is an atomic particle which orbits the nucleus of an atom. The electron is 1/1840 times lighter than the proton or neutron and has a negative electrical charge.
Element	An element is a form of matter whose atoms all have the same number of protons (atomic number).
Isotope	An isotope of an element is an atom which has the same atomic number (Z) as other atoms of the element, but a different mass number (A). This is caused by the atom having more (or fewer) neutrons than other atoms of that element.
Mass Number	The mass number of an atom is equal to the number of protons plus the number of neutrons in its nucleus. The mass number is designated by the letter **A**.
Neutron	The neutron is a nuclear particle with a mass similar to a proton but no electrical charge. The neutron is located in the nucleus of an atom.
Nuclear Radiation	Subatomic particles or energy which is emitted from the nucleus of an unstable isotope as it decays into something more stable.
Nucleons	Protons and neutrons are commonly referred to as nucleons as they reside in the nucleus of an atom.
Nucleus	The nucleus of an atom is composed of protons and neutrons, and is located in the central region of the atom.
Nuclide	This is a general term referring to any known isotope, either stable (about 300) or unstable (over 2,000), of any chemical element.
Proton	A proton is a nuclear particle having a positive electrical charge. The proton is located in the nucleus of an atom.
Quarks	These are elementary particles which make up protons and neutrons.

Radioactive	An unstable isotope which emits radiation is said to be *radioactive*.
Radionuclide	A term used to describe radioactive nuclides (radioactive atoms).

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Radiation and Radioactivity

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Sixth Revision

Radiation and Radioactivity

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1. Radiation

Over 2000 different nuclides have been identified, most of which are "unstable" or *radioactive*. As discussed in *Module 1: Structure of Matter,* an unstable nucleus is one which has either too many or too few neutrons for its particular number of protons.

Unstable nuclei will emit subatomic particles and/or electromagnetic waves in order to achieve a more stable neutron/proton ratio. These particles or waves emitted from the nucleus of an unstable atom are called *nuclear radiation*.

In simple terms, *radiation* is energy travelling in the form of particles or waves. Nuclear radiation is radiation which originates in the nucleus of an unstable atom.

The process in which a radioactive nucleus emits particles is called *radioactive decay* or *radioactive disintegration*.

Radioactive decay is a spontaneous nuclear transformation which results in the formation of new nuclides. These transformations are accomplished by one of several different mechanisms, including alpha-particle emission, beta-particle and positron emission, and orbital electron capture. Each of these emissions may or may not be accompanied by gamma radiation.

The type of the mechanism of radioactive transformation depends on two factors: the particular type of nuclear instability - that is, whether the neutron to proton ratio is either too high or too low for the particular nuclide under consideration, and on the mass-energy relationship among the pre-decay (parent) nucleus, the post-decay (daughter/progeny/decay product) nucleus, and the emitted particle.

2. Ionizing and Non-Ionizing Radiation

Radiation can be subdivided into two broad categories: *ionizing* and *non-ionizing*.

As radiation penetrates matter it deposits energy through collisions and other interactions in the absorbing medium. The results of these interactions are the excitation and ionization of absorber atoms. Radiation capable of producing electrically charged atoms (called *ions*) in the medium is called *ionizing radiation*. In other words, ionizing radiation possesses sufficient energy to physically remove electrons from atoms upon impact with the absorbing medium. This process of creating ions is referred to as *ionization*. *Non-ionizing radiation* does not possess sufficient energy to remove electrons from absorbing medium atoms.

Examples of ionizing and non-ionizing radiation include:

Ionizing Radiation:	*radiation from radioisotopes, cosmic radiation, radiation from x-ray machines and accelerators*
Non-Ionizing Radiation:	*visible light, microwaves, radio waves*

The only observable difference between ionizing and non-ionizing radiation is the energy of the radiation. In all other respects, they are identical.

3. Radiation and Energy

All matter has energy by virtue of its mass, even when at rest. You and I possess some intrinsic energy associated with our physical mass. The form of this mass-energy can be changed but the mass-energy is never destroyed. This is the basic principle that results from Einstein's theory of relativity and is expressed in the famous equation

$$E = mc^2$$

where E is the energy associated with the mass m and c is the speed of light in a vacuum.

When an object is moving it possesses additional energy associated with its motion. The energy of motion is commonly referred to as kinetic energy. The faster an object is moving, the more kinetic energy it has.

When radiation is emitted during radioactive decay, it possesses energy by virtue of its mass and its motion. However, because radiation is sub-microscopic in size, the energy of radiation is very small. This makes the use of traditional units of energy very cumbersome and impractical.

In the atomic world, the energies are expressed in a special unit called the *electron volt* (eV).

By definition, an electron volt is the amount of energy gained by an electron in passing through an electric potential difference of one volt (Figure 1). In Système Internationale (SI) units,

$$1\ \text{eV} = 1.602 \times 10^{-19}\ \text{joules}$$

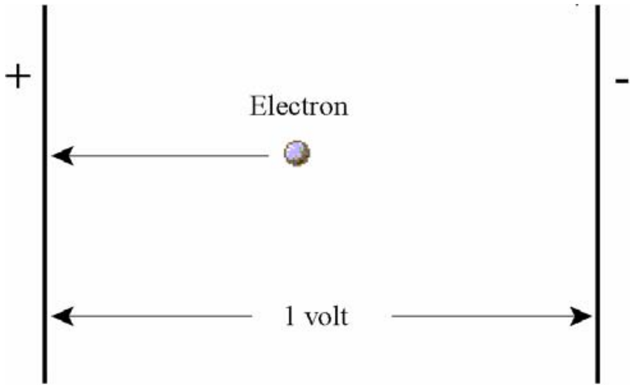


Figure 1: Electron volt.

For larger energy changes, multiple units of the electron volt are used in accordance with the SI system of units. These are:

1 thousand electron volts	(1 keV) = 10^3 eV
1 million electron volts	(1 MeV) = 10^6 eV

Radiation emitted from radionuclides typically have energies ranging from several keV to a few MeV.

The MeV can be compared to other energy units:

1 kWh = 2.25×10^{19} MeV
1 joule = 6.24×10^{12} MeV
1 joule = 6.24×10^{18} eV

where kWh refers to kilowatt-hours.

4. Types of Radiation

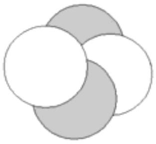
There are several types of radiation which result from the radioactive decay of unstable atoms. There are also other types of radiation such as neutron radiation and X-ray radiation which are produced under specific circumstances.

The following sections provide information on alpha radiation, beta radiation, x-ray and gamma radiation and neutron radiation.

4.1. Alpha Radiation

An alpha (α) particle is a highly energetic helium nucleus ($^{4}_{2}He$) that is emitted from the nucleus of the radioisotope when the neutron-to-proton ratio is too low. It is a positively charged, massive particle, consisting of an assembly of two protons and two neutrons (see Table 1).

Table 1: Alpha particle.

Alpha Particle 	- 2 protons and 2 neutrons - positively charged (+2e) - mass = 4 amu - monoenergetic - energy generally > 3.8 MeV - occurs in radioisotopes with Z > 82 - highly ionizing - easily stopped

In nature, alpha particles are emitted only by radioisotopes with an atomic number greater than 82. Some examples of alpha emitters include radium ($^{226}_{88}Ra$), uranium ($^{238}_{92}U$), and americium ($^{241}_{95}Am$).

When $^{226}_{88}Ra$ emits an alpha particle, it transforms into a radon ($^{222}_{86}Rn$) atom. This reaction (Figure 2) can be represented using the expression,

$$^{226}_{88}Ra \rightarrow {}^{222}_{86}Rn + {}^{4}_{2}He$$

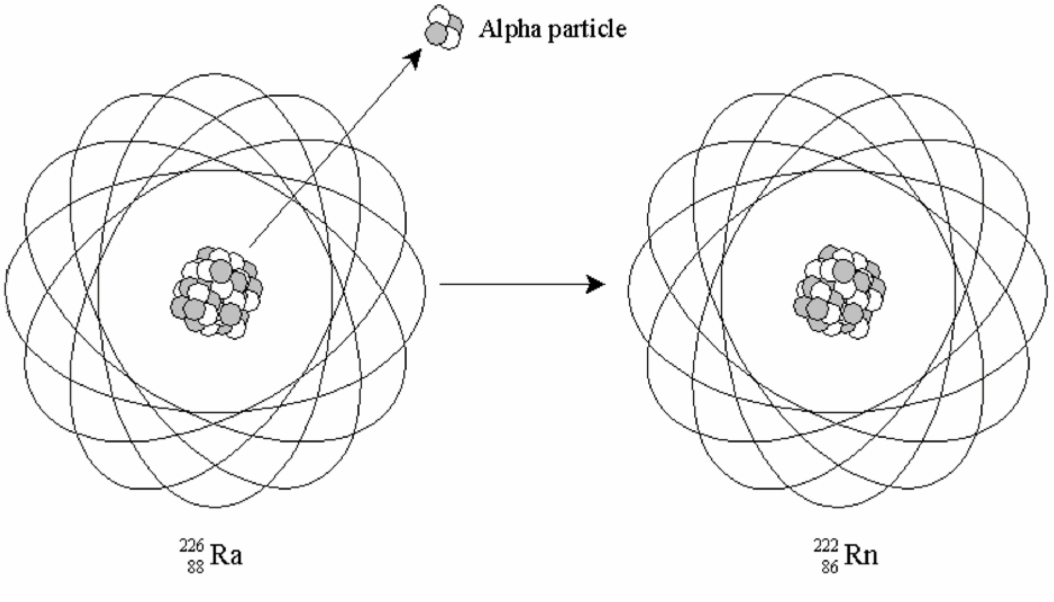


Figure 2: Alpha particle decay.

In the above expression, ${}^{226}_{88}\text{Ra}$ is commonly referred to as the “parent” radionuclide and the resulting ${}^{222}_{86}\text{Rn}$ atom, which is also radioactive, is referred to as the “progeny” or “decay product” of ${}^{226}_{88}\text{Ra}$. Progeny are also commonly referred to as “daughter products”.

Alpha particles are essentially monoenergetic. That is, alpha particles are emitted with one particular energy. For example, when ${}^{226}_{88}\text{Ra}$ undergoes radioactive decay, the alpha particle emitted always has the same energy (4.78 MeV). The uranium isotope ${}^{238}_{92}\text{U}$ emits an alpha particle with energy 4.2 MeV.

However, alpha particles do show discrete energy groupings, with small energy differences among the different groups. These small differences are attributed to differences in the energy level of the daughter nucleus. That is, a nucleus that emits one of the lower-energy alpha particles is left in an “excited” state, while the nucleus that emits the highest-energy alpha particle for any particular isotope is usually left in the “ground” state. A nucleus left in an excited state emits its energy of excitation in the form of a gamma ray. It should be pointed out that alpha particles are usually emitted with the maximum energy.

Alpha particles, because of their size and charge, are highly ionizing. As an alpha particle passes through matter, it will produce more ions per unit distance travelled than other types of radiation such as beta radiation or gamma radiation (Figure 3).

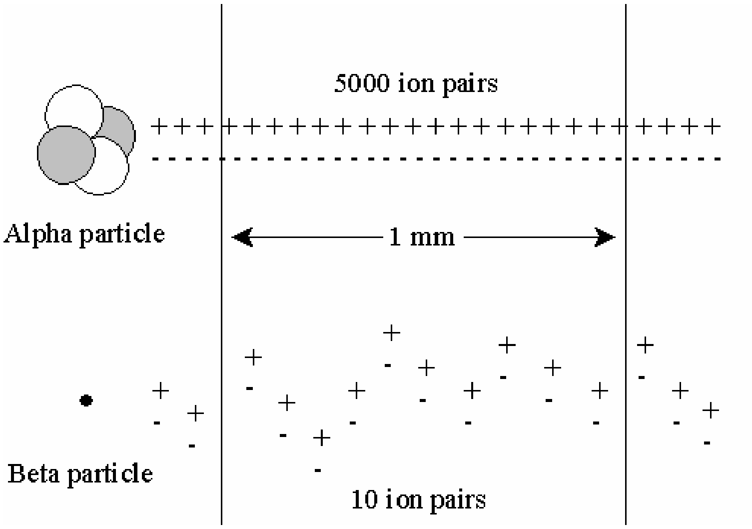


Figure 3: Alpha particle and beta particle ionization in air.

Because alpha particles are highly ionizing, they lose energy very quickly and consequently do not travel far in matter before coming to rest. Alpha particles will only travel a few centimetres in air and can be effectively stopped by a sheet of paper. Alpha particles will not penetrate the outer dead layer of our skin.

Consequently, alpha radiation is not considered an external radiation hazard. Alpha emitting radionuclides are only a hazard when taken internally. Tissues exposed to alpha radiation are highly ionized.

When an alpha particle eventually comes to rest, it will pick up a couple of orbital electrons from the free electron population around it and become an ordinary helium atom.

4.2. Beta Radiation

Beta (β) radiation consists of high speed electrons ($^{0}_{-1}e$) emitted from the nucleus of an unstable radioactive atom during radioactive decay. Beta particles are negatively charged and are significantly smaller than alpha particles. In fact, a beta particle is approximately 7,000 times lighter than an alpha particle (see Table 2).

Table 2: Beta particle.

Beta Particle 	• electron • negatively charged (-1e) • mass: ~ 1/1800 amu • continuous energy distribution • less ionizing than alpha particles • shielding: plastic, glass

As the nucleus is comprised of protons and neutrons, theoretically it is impossible for an intranuclear electron to exist. It is postulated that a beta particle is formed at the instant of emission by the transformation of a neutron ($^{1}_{0}n$) into a proton ($^{1}_{1}p$) and an electron according to the equation,

$$^{1}_{0}n \rightarrow {}^{1}_{1}p + {}^{0}_{-1}e$$

Examples of radionuclides which undergo beta decay include sodium ($^{22}_{11}Na$), phosphorus ($^{32}_{15}P$), and iodine ($^{131}_{53}I$). When phosphorus, $^{32}_{15}P$, undergoes radioactive decay, it emits a beta particle and transforms to sulphur, $^{32}_{16}S$ (Figure 4). The reaction can be described by the equation,

$$^{32}_{15}P \rightarrow {}^{32}_{16}S + {}^{0}_{-1}e$$

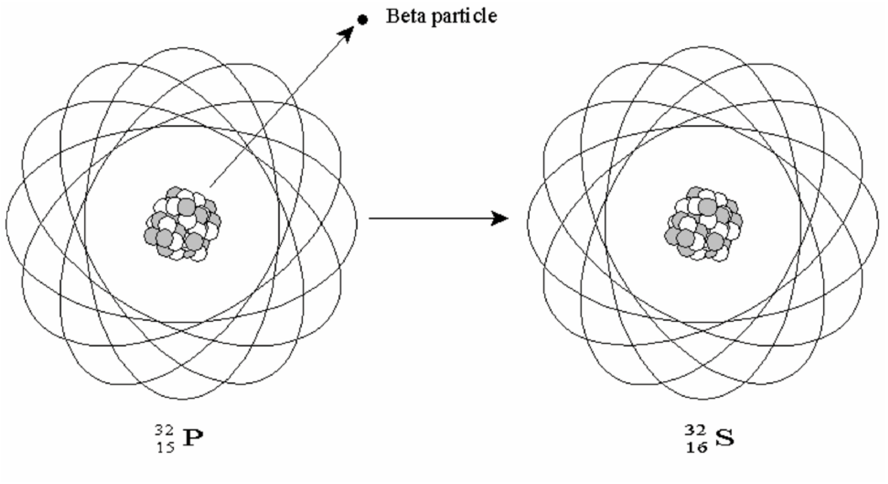


Figure 4: Beta particle decay.

Unlike alpha particles, beta particles are not monoenergetic. In fact, beta particles are emitted with a continuous energy distribution ranging from zero to a theoretically expected value based on mass-energy considerations for the particular beta emission (Figure 5).

Generally, the average energy of beta radiation from most beta emitting radioisotopes is approximately 1/3 of the maximum energy. Unless otherwise specified, when the energy of a beta emitter is given, it is the maximum energy.

Radionuclides such as $^{32}_{15}P$, $^{3}_{1}H$, $^{14}_{6}C$, and $^{90}_{38}Sr$ are known as pure beta emitters. They emit no gamma rays during radioactive decay. Often, instead of pure beta emitters, there are beta-gamma emitters. In this case, the emission of the beta particle is followed (instantaneously) by the emission of a gamma ray. The explanation for the gamma ray here is the same as that in the case of the alpha particle. The daughter nucleus, after the emission of a beta particle, is left in an excited condition and rids itself of the energy of excitation by the emission of a gamma ray.

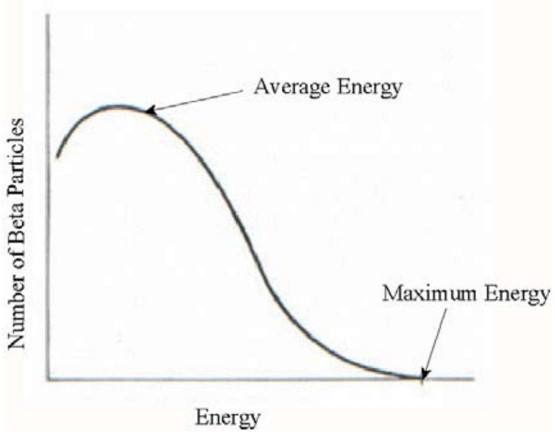


Figure 5: Beta particle energies.

Beta radiation, like alpha radiation, is ionizing. However, because of the beta particle’s smaller mass and single negative charge, it is less ionizing and can therefore penetrate matter to a greater depth than alpha particles (Figure 3).

Depending upon its energy, a beta particle can travel several hundred centimetres in air. In tissue, beta particles can travel only millimetres. Because of its penetrating ability, beta radiation can be an external and an internal radiation hazard.

Plastics or glass are very good shields for beta radiation and are commonly used for radiation protection against external sources of beta radiation.

Beta radiation also gives rise to X-ray radiation, referred to as *bremsstrahlung*, or “braking radiation”. When a fast moving beta particle is slowed down, changes direction or is stopped in an absorbing medium, energy is emitted in the form of an X-ray. To minimize the production of bremsstrahlung radiation, shielding material for beta radiation must be selected carefully. The fraction of beta radiation converted to bremsstrahlung is directly proportional to the atomic number of the absorbing medium (Z). Consequently, shielding material with high Z produce more bremsstrahlung than materials with a low Z. In practice, beta radiation shields with $Z > 13$ (aluminium) are rarely used.

4.3. Positron Radiation

In instances where the neutron to proton ratio is too low and alpha emission is not possible, the nucleus may, under certain conditions emit a positron. A positron ($^{0}_{1}\mathrm{e}$) is a beta particle whose charge is positive. In all other respects it is the same as the negative beta particle.

For example, sodium ($^{22}_{11}\mathrm{Na}$) undergoes positron emission to become neon ($^{22}_{10}\mathrm{Ne}$).

$$^{22}_{11}\mathrm{Na} \rightarrow {}^{22}_{10}\mathrm{Ne} + {}^{0}_{1}\mathrm{e} + \text{Energy}$$

In essence, a positron is a positively-charged electron. Whereas electrons occur freely in nature, positrons have only a transitory existence. Positrons occur in nature only as a result of the interaction between cosmic radiation (radiation from outer space) and the atmosphere and disappear almost instantaneously after formation. The manner in which positrons disappear will be discussed later on.

4.4. Gamma Radiation and X-Rays

Gamma radiation (γ), or gamma rays, is a type of monoenergetic electromagnetic radiation that is emitted from nuclei of excited atoms following radioactive decay.

The emission of gamma radiation provides a mechanism for ridding excited nuclei of their excitation energy.

X-rays are not produced in the nucleus of the atom. As already discussed, X-rays are produced when beta particles are slowed down, change direction or are stopped in an absorbing medium.

Gamma radiation is indistinguishable from X-ray radiation. The only difference between gamma rays and X-rays is their manner of origin.

In many respects, electromagnetic radiation, including X-rays and gamma rays, is very similar to particulate radiation such as alpha and beta particles. X-ray and gamma radiation can be theoretically described as a small bundle or packet of energy, called a *photon*.

Unlike alpha and beta radiation, X-ray and gamma radiation cannot be stopped. This type of radiation can only be attenuated. Attenuation refers to a reduction of the intensity of the radiation. When X-ray and gamma radiation pass through matter their intensity is reduced exponentially.

The extent to which X-rays and gamma radiation are attenuated depends upon the energy of the radiation and the properties of the absorbing material. In general, the denser the material is, the better it is at attenuating X-ray and gamma radiation. Lead and concrete are effective shielding materials for X-ray and gamma radiation.

X-ray and gamma radiation attenuation in matter is typically described by the property of the absorbing material known as the *Half-Value-Layer* (HVL). The HVL is defined as the thickness of the absorbing material which reduces the incident radiation intensity by one half. The HVL depends upon the energy of the incident radiation and the type of the absorbing material.

For example, the HVL for 1 MeV gamma rays in air is greater than 80 m. In water the HVL is approximately 100 mm. The HVL for 1 MeV gamma rays in concrete and lead are approximately 50 mm and 7.5 mm respectively.

X-ray and gamma radiation pose both an external and internal radiation hazard.

A summary of X-ray and gamma radiation properties is given in Table 3.

Table 3: Gamma and X-ray radiation.

Gamma Ray/X-Ray 	• photon • no charge • travel at the speed of light • monoenergetic • less ionizing than alpha particles or beta particles • highly penetrating • shielding: lead, concrete

4.5. Neutron Radiation

The only way to produce neutrons is to effectively knock them out of the nucleus. This is accomplished by bombarding a suitable target material with either alpha particles or high energy gamma rays.

A neutron source can be created by mixing americium ($^{241}_{95}\mathrm{Am}$) with beryllium ($^{9}_{4}\mathrm{Be}$). During neutron production, $^{9}_{4}\mathrm{Be}$ absorbs an alpha particle ($^{4}_{2}\mathrm{He}$) emitted from the radioactive decay of $^{241}_{95}\mathrm{Am}$ to produce a *compound nucleus* $^{13}_{6}\mathrm{C}*$.

The asterisk indicates that the nucleus is in an excited stated. The compound nucleus rids itself of its excitation energy instantaneously by emitting a neutron. The reaction can be described by the following equation.

$$^{9}_{4}\mathrm{Be} + {}^{4}_{2}\mathrm{He} \rightarrow {}^{13}_{6}\mathrm{C}* \rightarrow {}^{12}_{6}\mathrm{C} + {}^{1}_{0}\mathrm{n}$$

There are no naturally occurring radioisotopes that emit neutrons. Neutrons are emitted, for example, during the fission process, which occurs in nuclear reactors.

Neutrons have no charge and are, therefore, not repelled when they approach nuclei in an absorbing medium. Neutrons can actually enter the nucleus.

Since neutrons are electrically neutral, they do not cause ionization directly, and so they can travel great distances through matter, much like gamma rays.

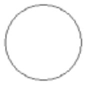
Like gamma radiation, neutrons are attenuated exponentially when passing through matter. The extent of attenuation is dependent on the neutron energy and properties of the absorbing material.

Neutrons can be effectively shielded by hydrogen rich materials, including paraffin (wax), water and concrete.

Sources of neutron radiation are most commonly found in nuclear reactors and particle accelerators.

A summary of neutron properties is given in Table 4.

Table 4: Neutron radiation.

Neutron 	• same mass as a proton • electrically neutral • mass: 1 amu • highly ionizing • highly penetrating • shielding: water, paraffin, concrete

5. Interaction of Radiation with Matter

Recall that an atom is comprised of a positively charged nucleus surrounded by negatively charged electrons which orbit the nucleus. The positive charge of the nucleus is neutralised by the negative charge of the orbiting electrons resulting in a net electrical charge of zero. In other words atoms are intrinsically electrically neutral.

When radiation passes through matter, it can interact with the nuclei of the atoms and upset the electrical balance. If the radiation possesses sufficient energy, it can ionize the atoms of the absorbing medium.

As already discussed, ionization is the process by which ion pairs are produced. We know that the positive charge of an atom resides in its nucleus and the negative charge is contained in orbiting electrons moving around it.

If this atom were to lose one of its electrons, it would be left with a net positive charge, and would be known as a *positive ion*. The liberated electron is a negative ion and the two particles together are called an *ion pair* (Figure 6). These two ions can move about quite independently of one another.

Ionization may be caused directly or indirectly through interactions with the nuclei of the absorbing material's atoms. Alpha and beta radiation produce ions directly in matter by virtue of their electric charge. As gamma rays and neutrons possess no electric charge, they produce ions in matter indirectly through collisions with the nuclei of the absorbing material's atoms.

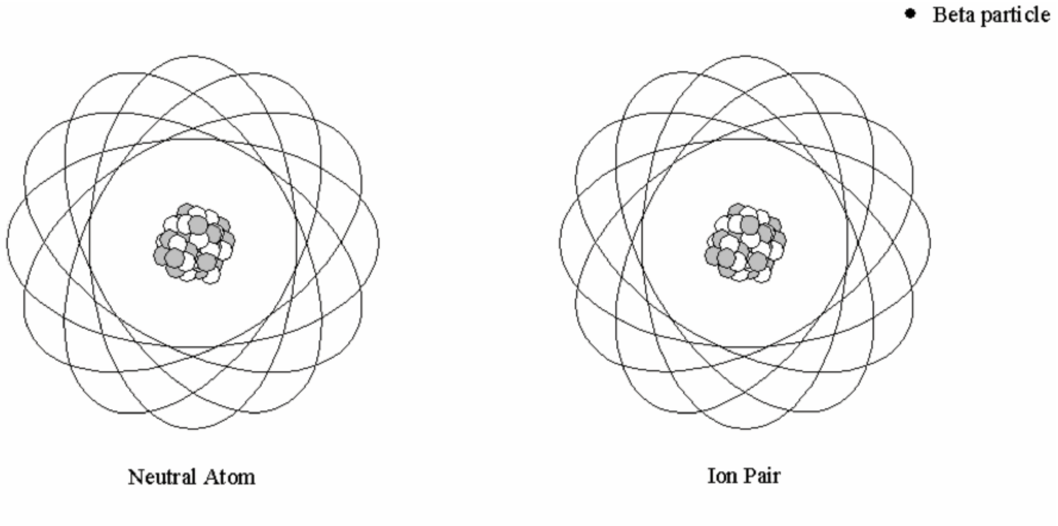


Figure 6: Ionization.

6. Ionization by Alpha and Beta Particles

A swiftly moving charged particle creates a strong electric field in its immediate neighbourhood. This "field" is similar in nature to the more familiar field of a magnet, and it has the ability to remove orbital electrons from atoms. In this way, alpha and beta particles cause the formation of ion pairs in their paths.

Because alpha particles are over 7000 times more massive than beta particles, they will move much more slowly than beta particles of the same energy. Also, they will not be as easily deflected from their path, unlike beta particles.

A slow moving particle spends more time near the atoms or molecules among which it is passing, and, therefore, is able to produce more ions per unit distance travelled. Thus, the slower moving, doubly charged alpha particles create more ions per centimetre of travel than do the swifter, singly charged beta particles.

As illustrated in Figure 3, alpha particles produce about 5000 ion pairs per millimetre of travel in air, while beta particles produce approximately 10 ion pairs per millimetre of travel in air.

However, since alpha particles produce many more ion pairs per mm than beta particles do, they lose all their energy in a relatively short distance. It is because of this, that alpha sources inside the body present a much more serious radiation hazard than beta sources inside the body. In other words, the alpha particles produce a very large number of ion pairs in a small amount of tissue.

7. Ionization by Gamma Radiation

Uncharged particles have no electric field associated with them, and cannot strip electrons from atoms the way alpha and beta particles do. Gamma rays, having no electric charge, produce ionization indirectly as a result of collisions with atoms. The major gamma ray reactions in matter include the photoelectric effect, Compton scattering and pair production.

7.1. Photoelectric Effect

A gamma ray can eject an orbital electron (any one) from an atom. The electron takes up all the energy of the photon and the photon disappears completely. This interaction is called the *photoelectric effect* (Figure 7).

The photoelectric effect is more likely to occur with heavy elements and low photon energies. The photoelectric effect occurs mainly with gamma photons below 0.5 MeV. Above this energy, it is observed that the photoelectric effect is very unlikely.

The speeding electron ejected from the atom will then cause ionization in exactly the same way as a beta particle does until its extra energy is completely dissipated. For this reason, we say that gamma photons produce *indirect* ionization. That is, only the first electron ejected from the atom was produced directly - all remaining ion pairs are produced indirectly by that electron.

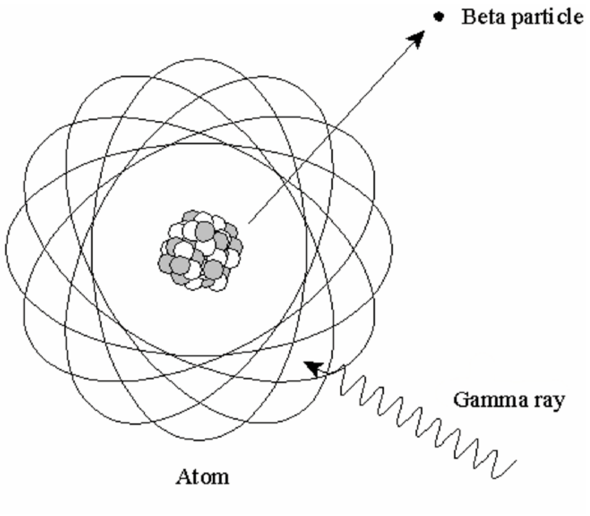


Figure 7: Photoelectric effect.

7.2. Compton Scattering

If the photon energy is large enough, the photoelectric effect is unlikely to occur, and the photon may lose only part of its energy to an electron. This electron is ejected and causes ionization as described above. The remaining photon energy is taken away by another gamma ray of lower energy. This second gamma ray is said to be scattered because it will travel in a new direction. This scattered photon may then later be absorbed by the photoelectric effect with another atom. If the scattered photon hasn't yet lost enough energy for the photoelectric effect to be probable (remember, the photoelectric effect is unlikely above 0.5 MeV), it may first lose some more energy by further scattering with other atoms.

This scattering interaction is known as *Compton scattering* (Figure 8). The likelihood of Compton scattering depends inversely on Z and occurs in all materials. Compton scattering is predominant with photons of medium energy (from about 0.5 MeV to 3 MeV).

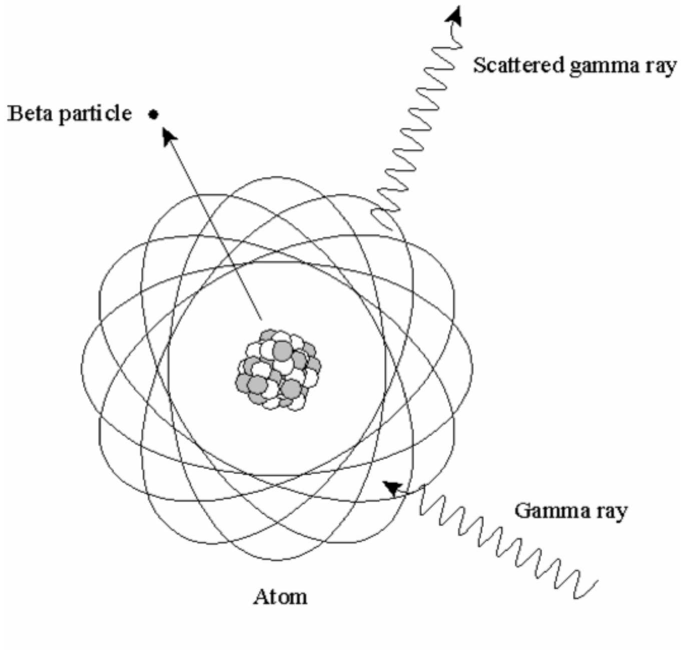


Figure 8: Compton scattering.

7.3. Pair Production

A high energy gamma ray sometimes changes into two electrons, one positively and the other negatively charged. The positive electron is called the *positron*. This interaction is known as *pair production* (Figure 9).

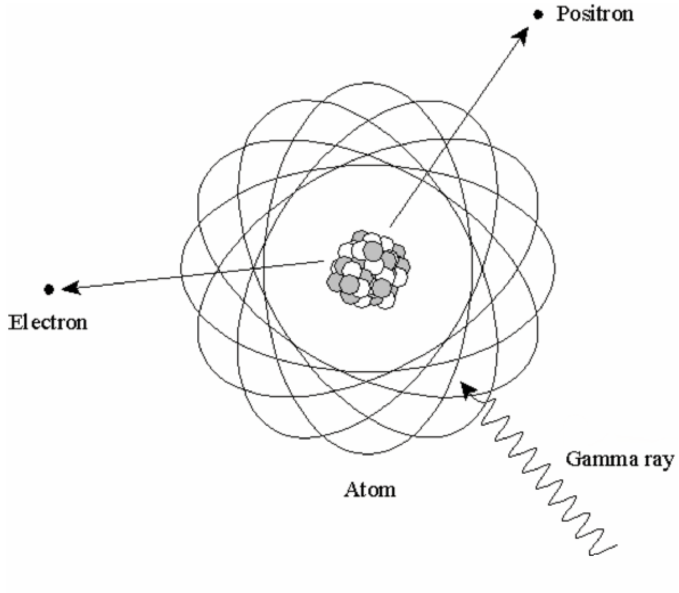


Figure 9: Pair production.

This is a good example of the conversion of energy to mass (recall: since $E=mc^2$, then $m = E/c^2$) Pair production occurs only near the nucleus of an atom. The amount of energy equivalent to the mass of the electron and positron pair is 1.02 MeV. Pair production, therefore, cannot happen when gamma rays possess energy less than 1.02 MeV. The higher the photon energy, the more likely pair production is to occur (the likelihood varies roughly as $(E-1.02)^2Z^2$). The excess energy of the photon (over 1.02 MeV) is shared equally by the electron and the positron. These high speed charged particles then cause ionization in just the same way as the electrons do in the photoelectric effect and in Compton Scattering.

When the positron comes to rest, it is captured by an electron. The pair then disintegrates to become two new photons, in a process called *annihilation*. The photons have an energy of 0.51 MeV each. This is the opposite of pair production, because now mass is converted into energy. This radiation is called *annihilation radiation* (Figure 10). The photons will be scattered by the Compton Effect or absorbed by the photoelectric effect in an identical manner to other gamma rays of similar energy.

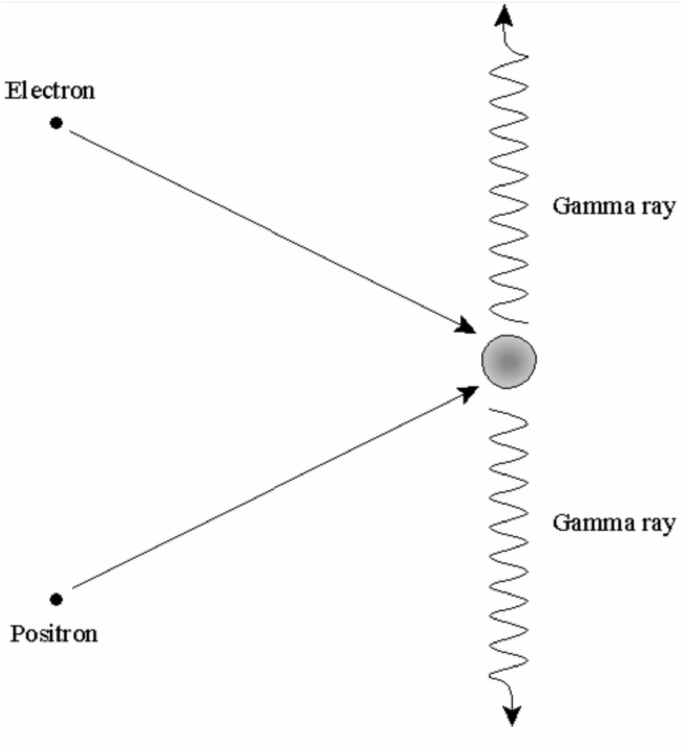


Figure 10: Annihilation radiation.

7.4. Summary of Ionization by Gamma Radiation

In conclusion, whether gamma rays interact with matter in terms of the photoelectric effect, Compton scattering or pair production, the end result will be the production of electrons and positive ions. This can happen in any material, including your body, and consequently damage to tissues may be caused by these electrons and ions.

Note that the photoelectric effect is the only one in which the gamma photons are absorbed completely (even in pair production two 0.51 MeV photons will still be produced at the end).

A probability plot of the different gamma radiation interactions with matter as a function of energy is shown in Figure 11.

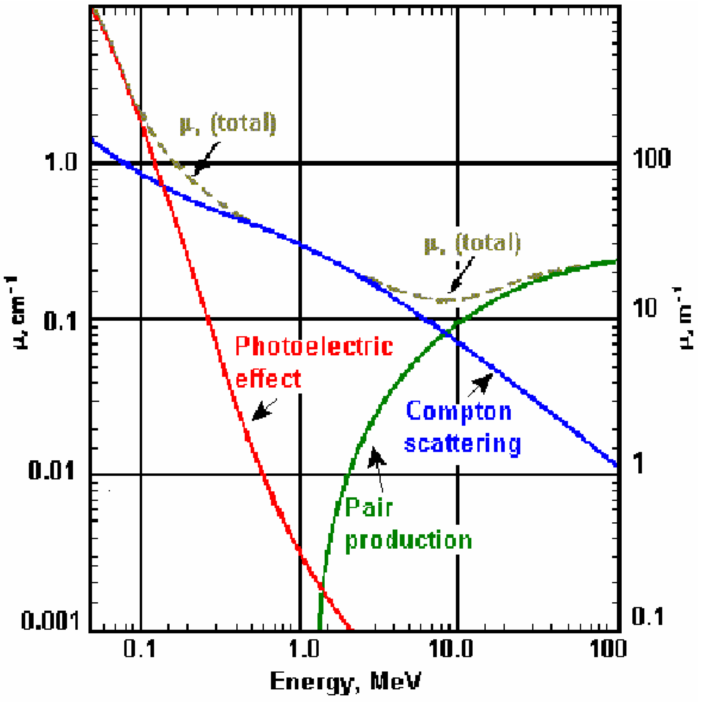


Figure 11: Probability of gamma interactions with matter. μ is directly related to the probability and exhibits the same trends.

8. Neutron Interactions with Matter

Neutrons, like gamma rays also produce ionization indirectly when passing through matter. Because neutrons have no electrical charge, they are not repelled when they approach nuclei. As already discussed, neutrons can actually enter the nucleus and split it, such as in a *fission* reaction.

Neutrons lose energy by reacting with nuclei rather than electrons. Recall that neutrons have about the same mass as protons, so they can transfer a lot of energy to hydrogen nuclei in a direct collision. Since hydrogen nuclei are nothing but protons, they will produce dense ionization in much the same way as alpha particles do. The body contains a significant portion of hydrogen (about 60% of the body is water) and this explains why living tissue gets badly punished by neutrons.

The most common neutron reactions are *fission*, *scattering* and *activation*.

8.1. Fission

Fission is the process by which the nucleus of an atom is split apart resulting in the formation of two or more new atoms.

Fission occurs naturally with uranium ($^{235}_{92}U$), but at a very low rate (approximately seven times per second per kilogram of uranium).

Fission is most commonly created in nuclear reactors when $^{235}_{92}U$ is exposed to very slow moving neutrons (thermal neutrons). It should be pointed out that the phrase “very slow moving” is a concept relative to the atomic world. Thermal neutrons possess an energy of approximately 0.025 eV and travel at a speed of 2.2 km/s. Fast neutrons (energies greater than 8 keV) can travel near the speed of light (300,000 km/s).

During a fission reaction, the thermal neutrons enter the nucleus of $^{235}_{92}U$ creating a new, unstable uranium isotope $^{235}_{92}U^*$. The unstable nucleus fissions to produce other atoms, referred to as fission fragments, one to three neutrons and a great deal of energy in the form of radiation and heat. An illustration of the fission of $^{235}_{92}U$ is shown in Figure 12.

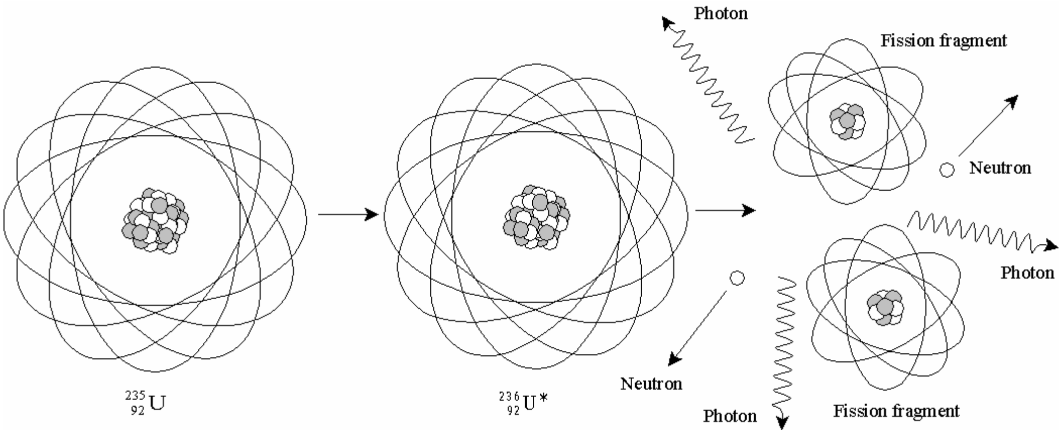


Figure 12: Fission.

The extra neutrons produced in the fission reaction can be used to react with other $^{235}_{92}U$ atoms to produce yet more fission reactions. If the conditions are right, such as in a nuclear reactor, self-sustaining fission reactions can be achieved. It is the heat generated from this self-sustaining fission process that is used to produce the steam which turns steam turbines connected to electrical generators for the commercial production of electricity.

8.2. Scattering

When neutrons pass through matter, they can collide with nuclei in a process known as scattering. Scattering can be either elastic or inelastic.

In *elastic scattering*, neutrons collide and are scattered by the nuclei of the absorbing medium in much the same way as pool balls. When a neutron strikes a heavy nucleus in an elastic collision, the neutron will bounce off the nucleus and move away with minimal energy loss (Figure 13). In elastic scattering with light nuclei, a significant portion of the neutron's energy may be transferred to the nucleus. The energy transferred manifests itself in increased speed of the nucleus which can, in turn, cause ionization. As one might conclude, materials composed of light nuclei are the most effective for slowing down neutrons.

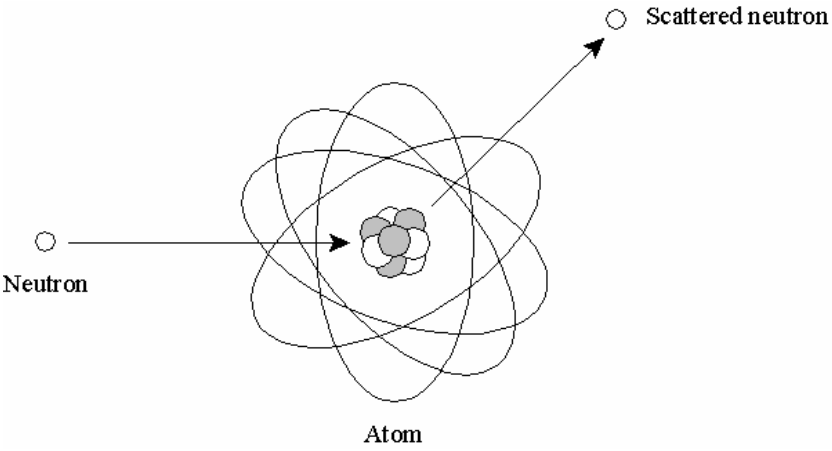


Figure 13: Elastic Scattering.

In *inelastic scattering*, the neutron is absorbed by the nuclei resulting in the creation of a *compound nucleus* which is unstable. The compound nucleus will emit a neutron of lower energy and a gamma ray (Figure 14).

Inelastic scattering generally is more predominant in collisions with heavy nuclei.

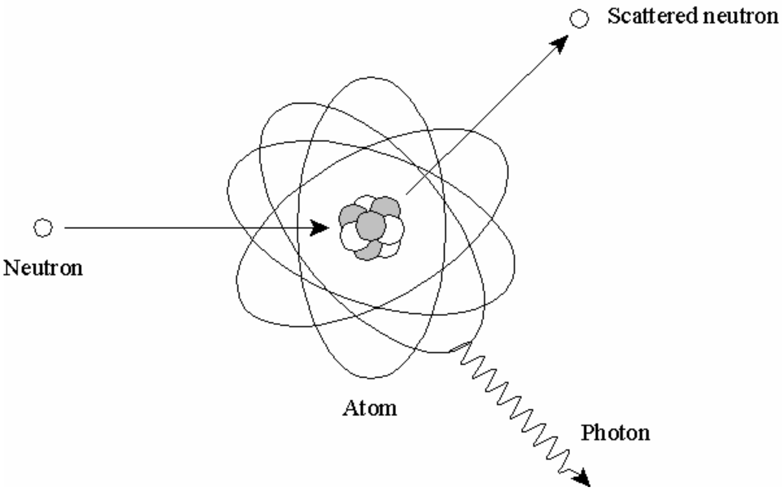


Figure 14: Inelastic scattering.

8.3. Activation

Activation is the process by which a radioisotope is produced by absorption of a neutron. The resulting radioisotope, which is unstable, may emit a proton, an alpha particle, a beta particle, a gamma ray, or a combination thereof. Examples of neutron activation reactions are presented below.

Neutron-proton reaction (n, p) with oxygen:

$${}^{1}_{0}\mathrm{n} + {}^{16}_{8}\mathrm{O} \rightarrow {}^{16}_{7}\mathrm{N} + {}^{1}_{1}\mathrm{p}$$

Neutron-alpha reaction (n, α) with boron:

$${}^{1}_{0}\mathrm{n} + {}^{10}_{5}\mathrm{B} \rightarrow {}^{7}_{3}\mathrm{Li} + {}^{4}_{2}\mathrm{He}$$

Neutron-gamma reaction (n, γ) with hydrogen:

$${}^{1}_{0}\mathrm{n} + {}^{1}_{1}\mathrm{H} \rightarrow {}^{2}_{1}\mathrm{H} + \gamma$$

Any substance that is irradiated with neutrons can be made radioactive. Therefore, neutron radiation is of great importance in radiation protection.

9. Linear Energy Transfer (LET)

Linear energy transfer (LET) refers to the rate of energy loss per unit path length of an ionizing charged particle as it traverses a medium. The energy lost by the charged particle is presumed transferred to the absorbing medium. LET is defined by the equation:

$$\mathrm{LET} = \frac{dE_L}{dl}$$

where dE_L is the average energy locally imparted to the absorbing medium by a charged particle of specified energy in traversing a distance of dl.

As one might expect, alpha radiation has a higher LET rate than beta radiation. Consequently, alpha radiation can cause more ionization and potential damage in tissue than beta radiation.

Linear energy transfer has important implications in the assessment of radiation effects on biological systems. More will be discussed on LET in later modules.

10. Mechanisms of Radioactive Decay

Recall that an atom is characterized by the number of protons and neutrons in its nucleus. Altering the number of neutrons in the nucleus does not change the chemical properties of the atom. Atoms with the same number of protons but different number of neutrons are referred to as isotopes. When the number of protons is changed, a new atom is formed.

When an atom is unstable, it may undergo radioactive decay in an attempt to become stable. As already discussed, during radioactive decay radiation is emitted from the nucleus of the atom. The most common types or radiation emitted are alpha, beta, gamma and neutrons.

When a radionuclide, commonly referred to as the parent radionuclide undergoes alpha or beta decay, a new atom is formed. This new atom is commonly referred to as the daughter radionuclide or progeny.

When a radionuclide undergoes neutron emission, a new isotope is formed.

The emission of gamma radiation only serves to rid the atom of excitation energy and does not change the fundamental character of the atom.

The different types of radioactive decay are summarized in Table 5.

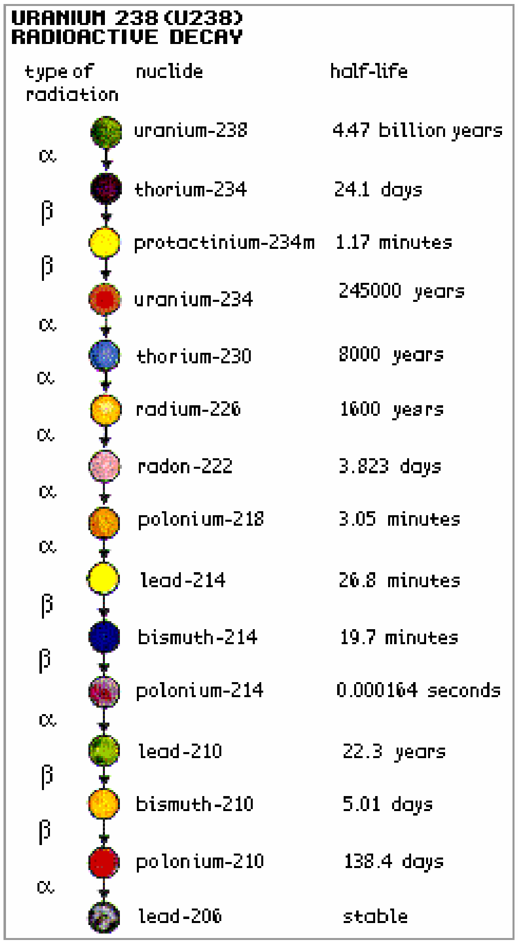
Table 5: Radioactive decay.

Decay Type	Parent Radionuclide	Daughter Radionuclide	
Alpha	Z, A	Z-2, A-4	New Nuclide
Beta	Z, A	Z+1, A	New Nuclide
Positron	Z, A	Z-1, A	New Nuclide
Neutron	Z, A	Z, A-1	New Isotope
Gamma	Z, A	Z, A	Unchanged

If the daughter nuclide is also radioactive, it too will undergo radioactive decay resulting in the creation of yet another new nuclide. This process will continue until the resulting atom is stable or, in other words, not radioactive. The number of decays will depend on the particular radionuclide.

Some decay series, such as that of uranium, are quite long while others, such as that of iodine, are short. Iodine ($^{131}_{53}\mathrm{I}$) decays to xenon ($^{131}_{54}\mathrm{Xe}$) which is a stable nuclide.

A typical example of a long decay series is the radioactive decay of uranium shown in Figure 15. In total, 14 radioactive decays occur before the original radionuclide, $^{238}_{92}\mathrm{U}$, transforms into the stable isotope of lead, $^{206}_{82}\mathrm{Pb}$.


URANIUM 238 (U238)
RADIOACTIVE DECAY

type of radiation	nuclide	half-life
α	uranium-238	4.47 billion years
β	thorium-234	24.1 days
β	protactinium-234m	1.17 minutes
α	uranium-234	245000 years
α	thorium-230	8000 years
α	radium-226	1600 years
α	radon-222	3.823 days
α	polonium-218	3.05 minutes
β	lead-214	26.8 minutes
β	bismuth-214	19.7 minutes
α	polonium-214	0.000164 seconds
β	lead-210	22.3 years
β	bismuth-210	5.01 days
α	polonium-210	138.4 days
	lead-206	stable

Figure 15: Uranium decay series.

10.1. Alpha Decay

Many of the heavy radionuclides are alpha emitters. Alpha decay only occurs for elements with atomic number Z greater than 82.

During alpha decay, the radionuclide emits an alpha particle which is simply a helium nucleus (^{4}He). Recall that the ^{4}He is composed of two protons and two neutrons. Therefore, during the process of alpha decay, two protons and two neutrons are removed from the radionuclide. This results in the creation of a new atom which may also be radioactive. The new atom will have a mass number (A - 4) and atomic number (Z-2), the parent radionuclide will have its mass number and atomic number reduced by four and two, respectively.

For example, $^{230}_{90}\text{Th}$ emits an alpha particle during radioactive decay and becomes $^{226}_{88}\text{Ra}$,

$$^{230}_{90}\text{Th} \rightarrow {}^{226}_{88}\text{Ra} + {}^{4}_{2}\text{He}$$

10.2. Beta Decay

In beta decay, a neutron transforms into a proton and a beta particle is emitted. As with alpha decay, this results in the formation of a new atom which may also be radioactive. This new atom will have the same mass number, A, as the parent radionuclide but will have a larger atomic number by one (Z + 1). See Figure 4.

For example, iodine ($^{131}_{53}\text{I}$) undergoes beta decay and transforms to xenon ($^{131}_{54}\text{Xe}$),

$$^{131}_{53}\text{I} \rightarrow {}^{131}_{54}\text{Xe} + {}^{0}_{-1}\text{e}$$

10.3. Positron Emission

During positron emission, a proton transforms into a neutron and a positron is emitted from the nucleus. The process is similar but different than that of beta decay.

The mass number A of the resulting atom is the same as the parent while the atomic number is reduced by one (Z - 1).

An example of positron decay is

$$^{22}_{11}\text{Na} \rightarrow {}^{22}_{10}\text{Ne} + {}^{0}_{1}\text{e}$$

Whereas negative electrons occur freely in nature, positrons have only a transitory existence. They occur in nature only as a result of the interaction between cosmic rays and the atmosphere and disappear in a matter of microseconds after formation. The manner of disappearance occurs when the positron combines with an electron, and the two particles are annihilated, giving rise to two gamma-ray photons whose energies are equal to the energy equivalent to the masses of the positron and the electron. This is exactly the same process as was described for pair production in Section 7.

10.4. Gamma Decay

When a daughter nucleus is formed by radioactive decay (emitting either an alpha or a beta particle), some energy is carried away from the nucleus by the particle emitted. After most disintegrations, however, the daughter still possesses more energy than is normal for that nucleus. Such a nucleus is said to be in an excited state. Any nucleus in an excited state promptly gets rid of the excess energy by emitting a gamma photon. Since there is no change in mass or electrical charge, gamma emission does not change the structure of the nucleus as does alpha and beta decay.

Gold, ${}^{198}_{79}\mathrm{Au}$, decays by beta emission to an excited state of mercury, ${}^{198}_{80}\mathrm{Hg}^*$. The symbol * is used to indicate that the atom is in an excited state. The decay scheme for ${}^{198}_{79}\mathrm{Au}$ is shown below.

$${}^{198}_{79}\mathrm{Au} \rightarrow {}^{198}_{80}\mathrm{Hg}^* + {}^{0}_{-1}\mathrm{e}$$

The ${}^{198}_{80}\mathrm{Hg}^*$ immediately emits a gamma ray to rid itself of excitation energy.

$${}^{198}_{80}\mathrm{Hg}^* \rightarrow {}^{198}_{80}\mathrm{Hg} + \gamma$$

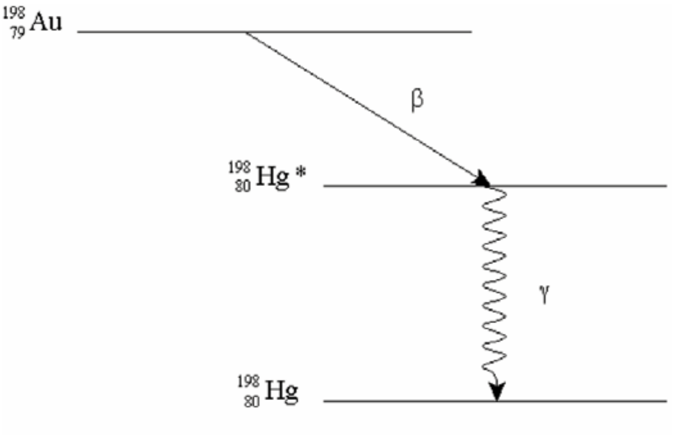


Figure 16: Gamma decay.

The gamma decay of ${}^{198}_{79}\mathrm{Au}$ is also illustrated in Figure 16.

Gamma decay can also occur in a similar manner during alpha decay such as in the case of americium (${}^{241}_{95}\mathrm{Am}$).After the emission of an alpha particle, ${}^{241}_{95}\mathrm{Am}$ transforms into neptunium (${}^{237}_{93}\mathrm{Np}^*$) which is in an excited state. To rid itself of the excitation energy, the ${}^{237}_{93}\mathrm{Np}^*$ emits a gamma ray.

11. Activity

One of the primary characteristics of a radionuclide is the rate at which radioactive decays or disintegrations occur. The rate of radioactive decay is referred to as the *activity*.

By definition, *activity* is a measure of the number of radionuclide decays per unit of time.

The SI unit of activity is the *becquerel* (Bq). A becquerel is defined as one decay per second.

1 becquerel = 1 decay/second

The historic unit for activity is the *curie* (Ci), named after the physicist Marie Curie. A curie is defined as 3.7×10^{10} decays per second. The relationship between the curie and the becquerel is shown below.

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

Consider 1 mg (0.001 g) of the radionuclide uranium ($^{238}_{92}\text{U}$) which has an activity of approximately 10 Bq. This means that there are approximately 10 decays per second occurring in the 1 mg of uranium.

The approximate activity of 1 mg of americium ($^{241}_{95}\text{Am}$) is 1×10^{8} Bq. In 1 mg of americium, 1×10^{8} decays are occurring each second.

The rate of decay of a radionuclide is exponential with time. The quantity of activity, A, remaining after any time interval T is given by the equation

$$A = A_{\circ}e^{-\lambda T} \qquad (1)$$

The term $A_{\circ}$ denotes the initial activity of the radionuclide. The parameter λ is the *decay constant*.

The *decay constant* is defined as the fractional decrease in activity per unit time of the radionuclide.

The drastic difference in activity between uranium and americium is related to a unique property of all radionuclides called the *half-life*.

12. Half-Life

The *half-life* of a radionuclide is the time required for it to lose 50% of its activity by radioactive decay. Each radionuclide has its own unique rate of radioactive decay, regardless of the quantity or form of the material (i.e., solid, liquid, gas, element or compound). The half-life of a radioisotope is an unalterable property of the radioisotope. Half-lives of radioisotopes range from microseconds (0.000001 sec) to billions of years. Some examples of half-lives are shown in Table 6.

To better understand the concept of half-life, consider the following example:

Assume we have 1000 Bq of phosphorus ($^{32}_{15}\text{P}$) which has a half-life of 14.3 days. If we wait 14.3 days and then measure the activity of the phosphorus we would find it to be 500 Bq. The activity has dropped to one half of its original value. If we were to wait another 14.3 days the resulting activity would be 250 Bq. If we were to wait yet another 14.3 days, the activity would be 125 Bq. As can be seen, the activity drops by one half for each half-life that is passed. After ten half-lives the activity of phosphorus will be reduced to one thousandth of its original value. The concept of half-life is further illustrated in Figure 17.

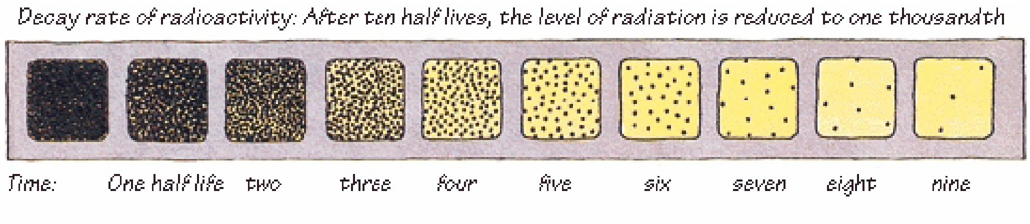


Figure 17: Half-life.

Table 6: Half-life examples.

Radionuclide	**Half-life ($t_{1/2}$)**
Nitrogen ($^{16}_{7}\text{N}$)	7.1 seconds
Argon ($^{41}_{18}\text{Ar}$)	1.8 hours
Iron ($^{59}_{26}\text{Fe}$)	45 days
Hydrogen ($^{3}_{1}\text{H}$)	12.3 years
Radium ($^{226}_{88}\text{Ra}$)	1600 years
Uranium ($^{238}_{92}\text{U}$)	4.5×10^{9} years

Mathematically, the concept of half-life can be defined by the equation,

$$\frac{A}{A_\circ} = \frac{1}{2^n} \tag{2}$$

where A denotes the activity remaining after n half-lives. The term $A_\circ$ refers to the original activity present.

The number of half-lives may be calculated using the expression

$$n = \frac{T}{t_{1/2}} \tag{3}$$

where T is the elapsed time and $t_{1/2}$ denotes the half-life of the radionuclide. In the calculation, the elapsed time and the half-life must be in the same units of time (seconds, minutes, hours, days, etc.).

The decay constant and half-life are related by the expression

$$t_{1/2} = \frac{\ln(2)}{\lambda} = \frac{0.693}{\lambda} \tag{4}$$

The first half-life period reduces the activity to 1/2 of the original activity. The second half-life reduces the activity to ½ of the remaining activity, that is ½ of ½, or 1/4 of the original activity. The fraction of the original activity remaining after succeeding half-lives is:

Activity after 1 half-life	= 1/2	of the original
Activity after 2 half-lives = ½×½ = $(½)^2$	= 1/4	of the original
Activity after 3 half-lives = $(½)^3$	= 1/8	of the original
Activity after 4 half-lives = $(½)^4$	= 1/16	of the original
Activity after 5 half-lives = $(½)^5$	= 1/32	of the original
Activity after 6 half-lives = $(½)^6$	= 1/64	of the original
Activity after 7 half-lives = $(½)^7$	= 1/128	of the original

Therefore, the activity remaining after 7 half-lives is 1/128 or 0.8% of the original activity. After this time, the number of radioactive atoms remaining will be 1/128 of the original number. A good rule of thumb to remember is that after 7 half-lives the activity has decreased to about 1% and after 10 half-lives to about 0.1% of the original activity (Figure 18).

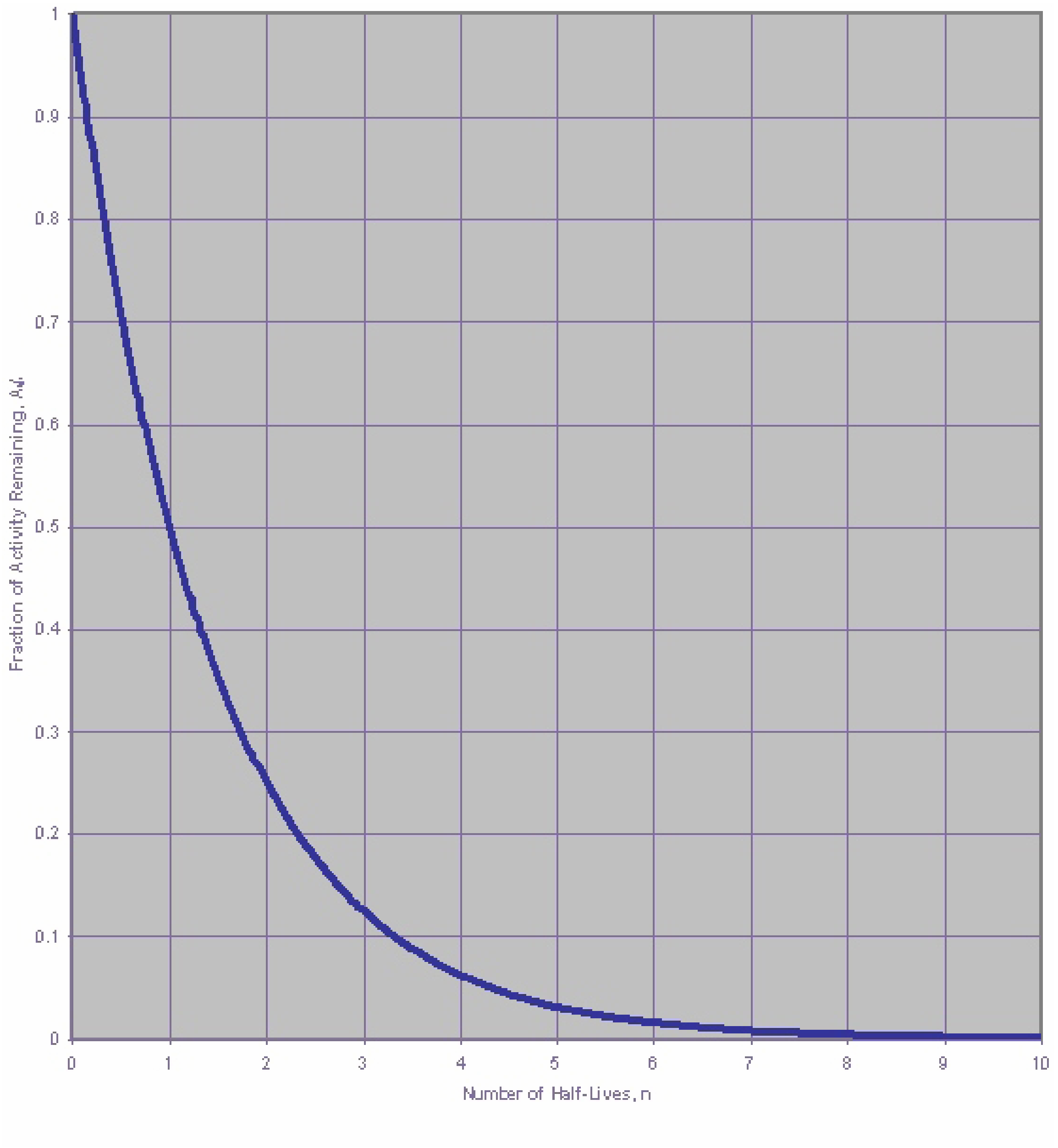


Figure 18: Radioactive decay curve.

Following are examples of half-life calculations.

Example 1

A caesium ($^{134}_{55}\mathrm{Cs}$) source has an activity of 1000 Bq. What is the activity remaining after 2 years and after 7 years? Caesium has a half-life of 2 years.

Solution: T = 2 years

The number of half-lives, n, is given by

$$\begin{aligned} n &= \frac{T}{t_{1/2}} \\ &= \frac{2}{2} \\ &= 1\ \mathrm{half-life} \end{aligned}$$

where T is the elapsed time. Therefore the activity remaining after 2 years or 1 half-life is calculated using Equation (2):

$$\frac{A}{A_{\circ}} = \frac{1}{2^n}$$

Writing this equation in terms of A yields the expression

$$A = A_{\circ} \times \frac{1}{2^n}$$

$$A = 1000 \times \frac{1}{2^1} = 1000 \times \frac{1}{2} = 500\ \mathrm{Bq}$$

The activity of caesium remaining after 2 years is 500 Bq.

The activity remaining after 7 years is calculated in the same manner.

$$n = \frac{T}{t_{1/2}} = \frac{7}{2} = 3.5\ \mathrm{half-lives}$$

$$A = A_{\circ} \times \frac{1}{2^n} = 1000 \times \frac{1}{2^{3.5}} = 88\ \mathrm{Bq}$$

After 7 years the remaining activity of caesium is 88 Bq.

Example 2

An unknown sample of material was observed to have an initial activity of 200 Bq. After 12 minutes the activity was observed to decrease to 12.5 Bq. What is the half-life of the unknown radionuclide? What is the radionuclide?

Solution:

We know the initial activity $A_\circ$ and the activity A after an elapsed time T of 12 minutes.

Solving the following equation:

$$\frac{A}{A_\circ} = \frac{1}{2^n}$$

For n, one obtains the equation

$$n = \frac{\ln\left(\frac{A}{A_\circ}\right)}{\ln(2)}$$

Substituting for A and $A_\circ$, one obtains

$$n = \frac{\ln\left(\frac{200}{12.5}\right)}{\ln(2)} = 4 \text{ half} - \text{lives}$$

The half-life of the unknown radionuclide is then

$$t_{1/2} = \frac{T}{n} = \frac{12}{4} = 3 \text{ minutes}$$

Recall that all radionuclides have a unique half-life. Therefore, an unknown radionuclide can be identified by measuring the rate at which a radionuclide decays.

To identify the unknown radionuclide, we would look in a *table of isotopes*. A table of isotopes contains data on all known radionuclides including their physical properties such as mass number and atomic number, mode of decay, decay series and half-life.

If we were to look in a table of isotopes we would find that our unknown radionuclide was an isotope of polonium ($^{218}_{84}\text{Po}$).This polonium isotope has a half-life of 3.05 minutes.

Example 3

A quantity of the iodine isotope, ${}^{131}_{53}\mathrm{I}$, is measured to have an activity of 100 Bq. The decay constant of ${}^{131}_{53}\mathrm{I}$ is 0.09 days^{-1}. Calculate the half-life of iodine and its activity after 30 days.

Solution:

The half-life of the iodine radioisotope is calculated directly using the equation

$$t_{1/2} = \frac{\ln(2)}{\lambda} = \frac{\ln(2)}{0.09} = 8\ \text{days}$$

The activity remaining after 30 days is calculated using Equation (1):

$$A = A_{\circ} e^{-\lambda T} = 100 \times e^{-0.09 \times 30} = 7\ \text{Bq}$$

We could have also calculated the iodine activity after 30 days using the equation

$$n = \frac{T}{t_{1/2}} = \frac{30}{8} = 3.75\ \text{half} - \text{lives}$$

and

$$A = A_{\circ} \times \frac{1}{2^{n}} = 100 \times \frac{1}{2^{3.75}} = 7\ \text{Bq}$$

13. Specific Activity

The specific activity (SPA) is the activity, A, per unit mass, m, of a radionuclide. It is calculated from the expression

$$\begin{aligned} \text{SPA} &= \frac{A}{m} = \lambda \frac{N}{M} \\ \text{SPA} &= \frac{\ln(2)}{t_{1/2}} \frac{N}{M} \end{aligned} \tag{5}$$

The term M denotes the atomic mass of the radionuclide and N, Avogadro's number (6.023×10^{23}). The units of specific activity are typically in Bq/g. Specific activity can also be designated in units of Bq/mole or Bq/litre.

Example

What is the specific activity of the radioisotope ^{125}I in sodium iodide (NaI)?

Solution:

The atomic mass of sodium (Na) is 23. The atomic mass of ^{125}I is 125 and its half-life is 60.2 days (5.20×10^6 s).

The fraction of iodine in NaI $= \dfrac{125}{125+23} = 0.84$

Each gram of NaI will contain 0.84 grams of I.

Using the equation for specific activity, Equation (5):

$$\mathrm{SPA}(^{125}\mathrm{I}) = \frac{\ln(2)}{5.2\times10^{6}}\frac{6.023\times10^{23}}{125}$$
$$\mathrm{SPA}(^{125}\mathrm{I}) = 6.42\times10^{14}\ \mathrm{Bq/g}$$

The specific activity of ^{125}I in NaI is then

$$\mathrm{SPA} = 0.84 \times 6.42 \times 10^{14} = 5.39 \times 10^{14}\ \mathrm{Bq/g} = 539\ \mathrm{TBq/g}$$

Glossary

Activity	The number of radioactive decays per second. The SI unit of activity is the becquerel (Bq).
Alpha particle	A highly energetic helium nucleus (^{4}He) that is emitted from the nucleus of a radioisotope whose neutron-to-proton ratio is too low.
Beta particle	A highly energetic electron that is emitted from the nucleus of a radioisotope whose neutron-to-proton ratio is too high.
Bremsstrahlung	X-ray radiation produced when a fast moving beta particle is decelerated, stopped or made to change direction. Literally, “braking radiation.”
Electron volt	The energy gained by an electron in passing through an electric potential difference of one volt. It is used as a unit of energy when dealing with particles and atoms.
Gamma radiation	Electromagnetic radiation that is emitted from the nuclei of excited atoms following radioactive decay. Differs from X-rays only in manner of origin.
Half-life	The time taken for a sample of a radioactive material to lose half of its activity by radioactive decay.
Half-Value-Layer	The thickness of an absorbing material which reduces radiation’s intensity by one half.
Ionization	Removal of one or more electrons from an atom.
Radiation	Energy traveling in the form of particles or waves.
Radiation, ionizing	Radiation capable of ionizing atoms.
Radiation, non-ionizing	Radiation which is incapable of ionizing atoms.
Radioactive decay	The process in which an unstable nucleus emits particles.
X-rays	Electromagnetic radiation that is produced when electrons or positrons are decelerated, stopped or made to change direction. Differs from gamma radiation only in manner of origin.

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Training Course

Radiation Safety Officer
(RSO-1)

Radiation Quantities and Units

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Sixth Revision

Radiation Quantities and Units

CONTENTS

1. Introduction

The depth of a snowfall is measured in centimetres. A volume of water is measured in litres. Similarly, there are various units associated with the measurement of radiation. This module introduces them.

2. Absorbed Dose

A given amount of beer will have less of an effect on a horse than it would on a man (don't laugh; English draught-horses used to be given beer on their rounds). What matters is not the amount of beer per se but the amount of beer in proportion to body weight.

Similarly, a given amount of radiation would not affect a horse as much as a man because there is a lot more of the horse. A dose of radiation is, therefore, expressed as the amount of energy deposited per unit mass. The Système Internationale (SI) unit for absorbed dose is the *gray* (Gy) and is defined as:

One gray (Gy) is one joule (J) of energy deposited in one kilogram (kg) of matter.

$$1 \text{ Gy} = 1 \text{ J/kg}.$$

Before the S.I. units were adopted, the absorbed dose was measured in rads. Rad is an acronym for "*r*adiation *a*bsorbed *d*ose" and is defined as:

One rad is one hundred ergs of energy deposited by radiation in one gram of matter.

$$1 \text{ rad} = 100 \text{ erg/g}.$$

We know that $1 \text{ J} = 10^7$ erg and that 1 kg = 1000 g, so

$$1 \text{ Gy} = 100 \text{ rad}.$$

Although the rad has been superseded by the gray, the rad is still widely used.

The gray, as we shall see later, is actually a fairly large dose. We more commonly work with milligrays (mGy) and micrograys (µGy).

$$1 \text{ Gy} = 10^3 \text{ mGy} = 10^6 \text{ µGy}$$

Note that the absorbed dose (usually given the symbol "D") is a physical quantity. It tells us how much energy has been deposited in a material by radiation but nothing about the biological consequences of that energy deposition in tissue.

3. Equivalent Dose

Different types of radiation produce different amounts of biological damage for a given energy deposition in the body. For example, alpha radiation causes much more damage than an equal absorbed dose of gamma radiation does.

The reason for this stems from the density of ionization caused by different types of radiation. An alpha particle will produce many times more ion pairs per millimetre traveled than will a gamma ray. Furthermore, it seems that the body can more easily repair damage from radiation if it is spread over a larger volume.

As 1 gray of alpha radiation is more harmful than 1 gray of gamma radiation, it is useful to have a common scale which measures the damage caused by different types of radiation. Hence we have the concept of *equivalent dose* whose S.I. unit is the *sievert* (Sv), defined as:

One sievert represents that quantity of radiation dose which is equivalent to, in terms of specified biological damage, one gray of gamma or beta radiation.

Like the gray, the sievert is a large dose and we more commonly use millisieverts (mSv) and microsieverts (μSv).

The customary unit for equivalent dose is the *rem* (*r*oentgen *e*quivalent *m*an) and it is still widely used. It is to the rad what the sievert is to the gray. Therefore

$$1 \text{ Sv} = 100 \text{ rem.}$$

The symbol used to denote the equivalent dose is “H” and we may write

$$H = D\, w_R$$

where w_R is the *radiation weighting factor*. It is the ratio of how biologically damaging 1 Gy of a type of radiation is compared with 1 Gy of gamma or beta radiation. A list of radiation weighting factors is shown in Table 1.

Note:

The electron volt (eV) is another unit of energy. It is widely used in nuclear physics and radiation science. By definition, 1 eV is the energy gained by an electron when it is accelerated across a potential difference of 1 volt (V).

Therefore, $1 \text{ eV} = 1.6 \times 10^{-19}$ J. Also, $1 \text{ MeV} = 10^6$ eV.

Table 1: Radiation weighting factors.

Type of Radiation	Radiation Weighting Factor (w_R)
Photons, all energies	1
Beta, all energies	1
Alpha, fission fragments and heavy nuclei	20
Neutrons of energy < 10 keV	5
Neutrons of energy 10 keV to 100 keV	10
Neutrons of energy 100 keV to 2 MeV	20
Neutrons of energy 2 MeV to 20 MeV	10
Neutrons of energy > 20 MeV	5
Protons of energy > 2 MeV	5

Source: Canadian Nuclear Safety Commission, Radiation Protection Regulations.

In calculating the equivalent dose from several types of radiation (called “mixed radiation”), all measurements are converted to Sv, mSv or μSv and then added together.

$$\mathrm{H} = \sum_{i} \mathrm{D}_{i} \cdot \mathrm{w}_{\mathrm{R},i}$$

<u>Example</u>

Albert is exposed to 2 mGy of gamma radiation, 0.6 mGy of neutrons of energy less than 10 keV (slow neutrons) and 0.2 mGy of neutrons of energy in the range 100 keV to 2 MeV (fast neutrons) during the course of a week. What is the equivalent dose (H) he received?

<u>Solution</u>

There are two stages to tackling this problem. First, for each type of radiation we have to convert D, the absorbed dose given in mGy, into H, the equivalent dose given in mSv.

We do this by multiplying the absorbed dose D by the w_R for that radiation. For gamma, $w_R = 1$ by definition, so 2 mGy of gamma radiation just becomes 2 mSv. For the slow neutrons, $w_R = 5$, so 0.6 mGy represents $0.6 \times 5 = 3$ mSv of equivalent dose. Similarly, the 0.2 mGy of fast neutrons is equivalent to $0.2 \times 20 = 4$ mSv. Now we just add up the individual equivalent doses to get the total: $2 + 3 + 4 = 9$ mSv.

4. Effective Dose

So far we have discussed a radiation dose to the whole body. If Cuthbert stands close to a nuclear reactor then his whole body will receive a dose of radiation. However, what if he ingested iodine-131? This is a radioactive form of iodine which emits beta particles. It is chemically identical to the more common iodine-129 and so is taken into the thyroid gland just as readily. The radiation dose in this case is only to the thyroid gland.

Another example: suppose Cuthbert took X-ray pictures of animals. Animals are not particularly inclined to hold still and so must be held while the pictures are taken. This means that Cuthbert's hands would be directly exposed to the X-rays but the rest of his body would receive a much smaller radiation dose.

How are we to handle local exposures such as these as distinct from whole-body exposures? If Cuthbert were to receive 10 mSv to his whole body from external gamma radiation and, in addition, a dose of 50 mSv to his thyroid gland, how are we to compare the relative biological importance of these two doses?

The International Commission on Radiological Protection (ICRP) believes that the relative risks associated with exposure should be equal, regardless of whether the dose applies to the whole body (H_D) or only to some tissues (H_T).

For a particular type of radiation, the equivalent dose to a single tissue is given by

$$H_T = D_T \, w_R$$

where D_T is the dose absorbed by the tissue. Hence the equivalent dose to a single tissue from mixed radiation is given by

$$H_T = \sum_i D_{T,i} \cdot w_{R,i}$$

In order to make these risks equal, the ICRP has determined *tissue weighting factors* (w_T) for doses to various tissues.

Let us return to Cuthbert, who has received 10 mSv of gamma radiation to his whole body (H_D) and 50 mSv of tissue dose (H_T) to his thyroid. The tissue weighting factor for the thyroid is 0.05. Multiplying the tissue dose by this factor gives us 50 × 0.05 = 2.5 mSv. This means that a dose of 50 mSv to the thyroid has the same chance of causing a health effect as 2.5 mSv delivered to the whole body.

We call this product of tissue dose and tissue weighting factor the *weighted dose* (H_W).

$$H_W = H_T\ w_T$$

Continuing our example with Cuthbert, we can see that 10 mSv of gamma dose to the whole body plus a 50 mSv tissue dose to the thyroid carries the same risk as a whole-body dose of 10 + 2.5 = 12.5 mSv.

This is the *effective whole-body dose* (H_{WB}) received by Cuthbert.

If more than one tissue is exposed then the total dose is given by the sum of the weighted doses (H_W):

$$H_W = \sum_i H_{T,i} \cdot w_{T,i}$$

So the effective whole-body dose (H_{WB}), in general, is given by the sum of the weighted doses (H_W) and the whole-body dose (H_D):

$$H_{WB} = H_D + \sum_i H_{T,i} w_{T,i}$$

Table 2 shows the tissue weighting factors for a number of tissues and organs.

Table 2: Weighting factors for various organs and tissues.

Organ or tissue	Tissue weighting factor (w_T)
Gonads (testes or ovaries)	0.2
Red bone marrow	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Oesophagus	0.05
Thyroid gland	0.05
Skin [1]	0.01
Bone surfaces	0.01
All organs and tissues not listed above (remainder organs and tissues) collectively, including the adrenal gland, brain, extra-thoracic airway, small intestine, kidney, muscles, pancreas, spleen, thymus and uterus [2, 3]	0.05
Whole body	1

[1] The weighting factor for skin applies only when the skin of the whole body is exposed.

[2] When the equivalent dose received by and committed to one of these remainder organs and tissues exceeds the equivalent dose received by and committed to any one of the preceding organs and tissues listed, a weighting factor of 0.025 shall be applied to that remainder organ or tissue and a weighting factor of 0.025 shall be applied to the average equivalent dose received by and committed to the rest of the remainder organs and tissues.

[3] Hands, feet and the lens of an eye have no weighting factor.

Source: CNSC Radiation Protection Regulations.

5. Exposure

One of the most common methods of measuring X-ray and gamma radiation is to measure the electric charge they produce by ionising air. This is because it is relatively easy to do. One *exposure unit* is defined as that quantity of X-ray or gamma radiation that produces in air, ions carrying 1 coulomb (C) of electric charge per kg of air.

$$1 \text{ exposure unit} = 1 \text{ C/kg air}$$

This unit is used for X-ray and gamma energies in the range from a few kilo-electron volts (keV) to a few mega-electron volts (MeV).

The electric charge on a single ion is 1.6×10^{-19} coulombs and the average energy dissipated in the production of a single ion is 34 eV. Using these facts we can relate the exposure unit to an absorbed dose:

$$\begin{aligned} 1 \text{ exposure unit} &= 1 \text{ C/kg (air)} \\ &= \left[\frac{1 \text{ ion}}{1.6 \times 10^{-19} \text{ C}}\right] \times 34 \text{ eV/ion} \times 1.6 \times 10^{-19} \text{ J/eV} \times \frac{1 \text{ Gy}}{1 \text{ J/kg}} \end{aligned}$$

Therefore:

$$1 \text{ exposure unit} = 34 \text{ Gy (in air).}$$

The strength of an X-ray or gamma-ray field is usually expressed as an exposure rate, such as coulombs per kg per hour. The total exposure, of course, is the product of exposure rate and time.

It is readily apparent that exposure, like dose, is independent of time; it is an integrated measure. This is just like photography; it doesn't matter whether one opens the shutter longer when there is little light or for less time when there is a lot of light. The amount of light to which the film is exposed is the same.

Just as the gray and the sievert superseded older units which are nevertheless still used, the S.I. exposure unit introduced above replaces the *roentgen* (R).

The roentgen was defined as that quantity of X or gamma radiation that produces ions carrying one statcoulomb (sC) of electric charge per cubic centimetre of air at a temperature of 0 ^{o}C and a pressure of 760 mm Hg.

$$1 \text{ R} = 1 \text{ sC/cm}^3$$

One ion carries an electric charge of 4.8×10^{-10} sC and so for 1 sC we have

$$(1 / 4.8 \times 10^{-10}) \text{ ions} = 2.08 \times 10^{9} \text{ ions.}$$

We know that the energy dissipated in the production of a single ion is 5.44×10^{-11} ergs and that the mass of 1 cm^3 of air at the temperature and pressure specified above is 0.001293 g. Therefore, we calculate that 1 R corresponds to an absorption of

$$2.08 \times 10^{9} \text{ ions} \times 5.44 \times 10^{-11} \text{ erg/ion} \times (1 / 0.001293 \text{ g}) = 87.7 \text{ ergs/g of air.}$$

From this, we know that 1 R corresponds to an absorption of 87.7 ergs per gram of air, or to a radiation dose to the air of 0.877 rad. Thus:

$$1 \text{ R} \equiv 0.877 \text{ rad (in air)}$$

The relationship between the exposure in S.I. unit C/kg and the roentgen may be calculated as follows:

$$1 \text{ exposure unit} \equiv 34 \text{ Gy} = 34 \text{ J/kg} = 34 \times 10^{7} \text{ ergs/kg} = 34 \times 10^{4} \text{ ergs/g}$$

We know that there are 87.7 ergs/g per roentgen, so

$$1 \text{ exposure unit (in S.I. units)} = 34 \times 10^{4} / 87.7 \text{ R}$$

i.e.

$$1 \text{ exposure unit} = 3877 \text{ R}$$

5.1. Exposure –Dose Relationship

Exposure may be an easy way to measure X-ray and gamma radiation fields but what does it mean for Derek, who has been irradiated? The important quantity to him is the dose his body tissues have received.

For our purposes, energy absorption is approximately proportional to electron density in the absorber. Now, we can consider muscle tissue to have an elementary composition of 5.98×10^{22} hydrogen atoms per gram, 2.75×10^{22} oxygen atoms per gram, 0.172×10^{22} nitrogen atoms per gram and 6.02×10^{22} carbon atoms per gram. Knowing the number of electrons in each kind of these atoms, we know that there are 3.28×10^{23} electrons per gram.

For air, whose density is 1.293×10^{-3} g/cm^3, the electron density is 3.01×10^{23} electrons per gram.

So the absorbed dose received by Derek's tissues corresponding to 1 exposure unit (i.e. 1 coulomb per kilogram of air) is (3.28 / 3.01) × 34 Gy (in tissue) = 37 Gy (in tissue). So 1 exposure unit results in an absorbed dose to tissue of 37 Gy.

$$1 \text{ exposure unit} \equiv 37 \text{ Gy (in tissue)}$$

This calculation agrees very well with experimental measurements.

Similarly, an exposure of 1 R, which corresponds to an absorbed dose of 87.7 ergs per gram of air, leads to a dose of 96 ergs per gram or 0.96 rad to tissue. This is very nearly 1 rad so we can say that 1 R of exposure results in an absorbed dose of approximately 1 rad.

$$1\ \text{R} \approx 1\ \text{rad}$$

It should be noted that it is not uncommon, although quite incorrect, for the units *roentgen* and *rad* to be used interchangeably.

6. Dose Rate

Dose is a measure of radiation energy actually deposited in an object, material or individual (or horse!) that has been exposed to ionizing radiation. *Dose rate* measures how much energy is deposited in a given amount of time. Radiation monitoring equipment is usually calibrated in Sv/hour (rem/hour if using the old units) or Gy/hour (rad/hour in old units) to describe the rate at which radiation is received.

For example, suppose Bert worked in a gamma radiation field of 3 mSv/h for 20 minutes. The total dose he received is

$$3\ \text{mSv/h} \times 20\ \text{min} \times 1\text{h} / 60\ \text{min} = 1\ \text{mSv}.$$

7. Kerma

In the case of indirectly ionizing radiation, such as X-rays, gamma rays and fast neutrons, we are sometimes interested in the initial kinetic energy of the primary ionizing particles (the photoelectrons, Compton electrons or positron-electron pairs in the case of photon radiation; the scattered nuclei in the case of fast neutrons) produced by the interaction of the incident radiation per unit mass of the interacting medium. In other words, when the incident radiation interacts with some material and produces ionizing particles, we want to know how much energy these particles have. As with radiation dose, the mass of the material is a consideration and so we want to know how much energy is given to these particles per unit mass of the material being irradiated. This quantity of transferred energy is called the *kerma* and is measured in grays (in S.I. units). For those who prefer the more traditional units, it is measured in rads.

Kerma and dose might both be measured in the same units but they are different quantities.

Kerma is defined as the *sum of the initial kinetic energies of all charged particles produced by the radiation per mass of the irradiated material*. Dose, however, is defined as the *energy absorbed per unit mass*.

It is like beer and water: both are measured in pints but they are quite different, as any draught-horse will tell you!

8. Système Internationale (SI) and Customary Units

In 1960, scientists meeting in Paris adopted a system of units, called Système Internationale (SI), as a world wide basis for the standardization of measurement units to replace both the metric and the British systems.

Table 3: Conversion of some radiation units.

S.I.		Customary
1 gray (Gy)	=	100 rads
1 sievert (Sv)	=	100 rems
1 exposure unit	=	3877 roentgens (R)

Note: Because the newer SI units are based on kilograms and metres rather than the grams and centimetres of the customary units, the SI units are much larger.

Glossary

Absorbed dose	The amount of energy deposited per unit mass.
Effective dose	A quantity which compares a dose to individual organs or parts of the body with a uniform dose to the whole body in terms of the possible health effects.
Effective whole-body dose	The sum of the uniform dose to the whole body and the weighted doses to individual tissues and organs.
Electron volt	The energy gained by an electron when it is accelerated across a potential difference of one volt. It is used as a unit of energy in nuclear physics.
Equivalent dose	A quantity which takes account of the biological effect of a radiation dose. It is defined as the product of the absorbed dose and the radiation weighting factor.
Exposure	A measurement of X-ray or gamma radiation which considers how much electric charge they produce by ionizing air.
Gray	The S.I. unit of absorbed dose, denoted as Gy. One gray is one joule of energy deposited in one kilogram of matter.
Radiation weighting factor	A measure of how biologically damaging different types of radiation are. It is the ratio of how damaging 1 Gy of a type of radiation is compared with 1 Gy of gamma or beta radiation.
Sievert	The S.I. unit of equivalent and effective dose, denoted as Sv.
Tissue weighting factor	A factor which compares the health risk of a dose to an organ or tissue with a uniform whole-body dose.
Weighted dose	The product of the equivalent dose to a tissue or organ and the tissue weighting factor.

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Training Course

Radiation Safety Officer
(RSO-1)

Biological and Health Effects of Exposure to Radiation

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Sixth Revision

Biological and Health Effects of Exposure to Radiation

CONTENTS

1. Introduction

This module deals with the effects of ionizing radiation on the body. It describes the possible and probable long-term effects of exposure to ionizing radiation.

2. Review of Ionization

Module 2, *Radiation and Radioactivity*, discussed the concept of ionization. Nuclear radiation is the emission of particles and electromagnetic waves from the nuclei of unstable atoms. This radiation has the ability to remove electrons from atoms of the target material, producing charged fragments or ions. It is therefore classified as ionizing radiation.

3. The Effect of Radiation on Water Molecules

Atoms can bond with each other to form molecules. As an atom is the smallest part of an element, so a molecule is the smallest part of a compound. Water is a simple molecule consisting of one oxygen (O) atom and two hydrogen (H) atoms, hence its chemical representation H_2O.

Approximately 60% of the human body weight is water and so it is important to consider what radiation does to water molecules.

Ionizing radiation can break apart water molecules to create free radicals.

$$H_2O \rightarrow H\cdot + OH\cdot$$

A free radical is a fragment of a compound or an element which contains unpaired electrons. Both hydrogen and oxygen normally exist as H_2 and O_2 molecules, respectively.

When radiation breaks a chemical bond in this way, it is known as *direct damage* to the body.

These free radicals can affect nearby molecules, forming undesirable products. This damage to the body is known as *indirect damage*.

The free radicals formed by breaking up water can react with each other in the following ways:

$$H\cdot + OH\cdot \rightarrow H_2O$$

This is just water, which is where we started.

$$H\cdot + H\cdot \rightarrow H_2$$

This is a hydrogen molecule. The body can tolerate these.

$$OH\cdot + OH\cdot \rightarrow H_2O_2$$

This is hydrogen peroxide, which is poisonous. Indeed, the symptoms resemble those of radiation sickness in many ways.

The damage produced by the free radicals and hydrogen peroxide is known as *indirect damage.*

4. The Effect of Radiation on People

Having seen what harm ionizing radiation can wreak on water molecules, it is natural to wonder about molecular damage to the cells of our bodies, and what this will mean for our health.

The effects of exposure to radiation can be divided into two categories: *hereditary effects* and *somatic effects*.

Hereditary effects are those which do not become apparent until future generations are born and somatic effects are those which are experienced by the people exposed to radiation, and can be subdivided into two more categories: *stochastic effects* and *deterministic effects*. These effects will be explored a little later.

5. Some Cellular Biology

The basic building block of the body is the cell. Every part of one's body is either comprised of cells or was built by them. A large number of cells of any particular type is called a *tissue*. If this tissue forms a unit with particular, specialized functions then it is called an *organ*.

In an adult organism (human, horse, yeti, etc.), most mature cells contain a set of giant molecules called *chromosomes*. These molecules contain all the information (hair colour, size and shape of the teeth, enzymes which cause raptures of excitement in biochemists, etc) required to create another individual exactly like the owner of the cell. Think of this information as a blueprint for recreating the human, horse or yeti from which it came.

In a human, the normal cell contains twenty-three pairs of chromosomes. Important exceptions are the germ cells (sperm cells and egg cells), which each contain half the usual number of chromosomes. So a human germ cell will contain twenty-three single chromosomes.

These germ cells are produced in the gonads, i.e. the male testes or the female ovaries.

Conception occurs when a sperm cell unites with an egg cell. The chromosomes from each combine so that the new cell has the full complement of twenty-three pairs. Characteristics from both parents are thereby combined in the offspring.

This single cell, the zygote, develops into a new individual by undergoing cell division. It divides to form two new cells, each containing the information held by the original. Many billions of these cell divisions take place during the formation of the new individual, the genetic information being faithfully copied each time. Note that this information includes instructions for making new germ cells and so characteristics from one individual can be passed on for many generations. Distinguishing features of certain families are an obvious illustration.

6. Mutation

Suppose that the genetic blueprint contained in the chromosomes was damaged in some way. This garbled information will then be passed on with the same remarkable accuracy when the cell divides. The alteration is known as a *mutation*. If it occurs in the germ cells then it is an *hereditary mutation*.

What happens if one of these germ cells is used in conception? It provides half of the chromosomes to the zygote, as before, and the genetic blueprint is copied faithfully. All the cells in the new individual will contain this altered information, including those cells which will later become germ cells. So just as certain characteristics are passed on through the generations, so too will be mutations.

An hereditary mutation is not necessarily harmful. A damaging effect will gradually be eliminated from a population by natural means, since such individuals are less likely to successfully reproduce than average individuals. The more severe the damage from a single mutation, the more quickly that mutation is eliminated. However, it is possible for a mutation to be beneficial and will enable an individual to reproduce more successfully than an average one.

This is how the process of evolution works. It is not so much survival of the fittest as survival of the best adapted to the environment. However, we digress.

What causes these mutations? More specifically, what can damage the genetic information in the chromosomes? An important point is that some mutations occur spontaneously. They simply happen. C’est la vie! However, we also know of various agents which can bring about mutations. They are therefore said to have *mutagenic properties*. Examples include a wide variety of chemicals, certain drugs, and physical factors such as elevated temperatures of the gonads and ionizing radiation.

7. Hereditary Effects of Radiation

7.1. Effects on Animals

We have known since 1927 that ionizing radiation can induce mutations. Large-scale experiments in more recent years examined how radiation changed the mutation rates in mice. “Large-scale” is no exaggeration: more than one million mice were needed to obtain meaningful results! Such is the nature of statistics.

In a nutshell, the findings were:

- Different types of mutation varied enormously in their sensitivity to radiation;
- 100-2000 mSv was required to double the natural mutation rate;
- The effects of a given dose were greater for the male;
- The consequences were minimized by allowing some time to elapse between irradiation and conception;
- A dose spread over a period of time produced a smaller effect than if delivered all at once.

7.2. Effects on Humans

Experimental data for humans is much sketchier. For one thing, finding volunteers is very difficult! Another difficulty is that many mutations are minor and difficult to measure. Examples include a less efficient digestive system or a predisposition to a certain disease.

The largest group of irradiated humans available for study are the descendants of those Japanese who were exposed during the nuclear bombing of Nagasaki and Hiroshima.

So far, hereditary effects such as leukemia and mental retardation have only been seen in those children who were heavily irradiated while still in their mother’s womb. Children conceived and born after the explosion have shown no change in the natural mutation rate.

While this may seem like relatively good news, the numbers involved (30,000 children born to irradiated parents) are too small to give statistically sound results. Furthermore, some mutations may not be evident for several generations.

The best we can do is to assume that radiation effects on humans are similar to those on mice.

7.3. Hereditary Risk from Radiation

So what are the numbers? How risky is a certain amount of radiation?

For the general population, the risk of serious hereditary ill-health in babies conceived after either parent is exposed to radiation is estimated to increase by 1% per 1000 mSv. This risk includes not only the first generation but also all their descendants.

This risk applies to the total exposure of the parent from birth until the final child is conceived. Obviously, any exposure thereafter cannot affect future generations.

It is assumed that the risk is proportional to the dose. For example, if the parents receive 1 mSv instead of 1000 mSv then the increase in risk of hereditary problems is only 0.001%.

Workers are treated slightly differently. Workers will only be exposed to radiation occupationally from the age of 18 to, say, 45. Members of the public, however, can be exposed from birth. For equal annual dose rates, the shorter time span for workers results in a lower risk. The hereditary risk of exposure to radiation for workers is 0.6% per 1000 mSv rather than 1% per 1000 mSv.

8. Somatic Effects

Having dealt with the risk to future generations from mutations in germ cells, let us turn our attentions to the long-term consequences of radiation exposure to the individual exposed.

A mutation in a cell in the body might affect the rate at which the cell divides or even bring about the destruction of that cell. Suppose the effect is to cause the cell to divide in an uncontrolled manner. The result will be a large number of cells which bear this fault. It is often the case that these cells will divide before they reach their mature state and so there will be an ever increasing number of cells which have no beneficial function to the body, yet are absorbing nutrients at an increasing rate.

A tissue formed from these cells is called a tumour. If the cells remain where they are and do not invade surrounding tissues then the tumour is said to be *benign*. However, if the tumour invades neighbouring tissues and causes distant secondary growth (called *metastases*), then it is said to be *malignant*.

Cancer is a malignant tumour. Whether it is fatal or not depends on where it is, how rapidly it grows and how soon it is detected.

8.1. Long Term Somatic Effects: Cases of Radiation Induced Cancer

There are many well-documented examples of radiation-induced cancers in humans. The early scientists who worked with X-rays and radioactive substances did not realize the risk. Many died from skin and bone cancers, and from leukemia.

In the 1920s, watch dials were painted with a radium-based luminous paint. The factory employees who did this work often licked their paint brushes to give them a sharp point. Each time they did this they ingested a small quantity of the paint. The radium in the paint collected in their bones and for many of them it resulted in bone cancer 8 to 40 years later.

In the United Kingdom more than 6500 patients with a certain backbone disease were treated with large doses of X-rays. The average dose was 3000 mGy (yes, this is a fair amount). 30 developed leukemia compared with the expected 7. (Expected? Remember that some people will suffer from leukemia anyway.)

Approximately 80,000 of the Japanese who survived the nuclear bomb attacks on Nagasaki and Hiroshima have been followed very closely. There have been approximately 350 more cancer deaths than would normally be expected so far. Of these, 126 were due to leukemia. This is nearly double the normal figure. It was found that there was a delay between exposure to the radiation and the onset of leukemia. This delay is known as the *latency period*. It is eight years for leukemia. For other cancers the latency period can be much longer.

Alas, these numbers are not enough to obtain solid statistical results. The best we can do is to take some educated guesses and err on the side of caution.

Although there is no scientific proof, it is assumed that there is no threshold radiation dose for cancer, i.e. there is no dose below which the ionizing radiation does not increase the risk of developing cancer. It is like smoking cigarettes; there is no number which must be exceeded before they increase the risk of lung cancer.

8.2. Somatic Risk from Radiation

Current thinking is that the risk of developing a fatal cancer as a result of exposure to radiation is 4% per 1000 mSv, i.e. for every sievert of radiation dose one receives, the chance of one developing a fatal cancer increases by 4%. This may be an overestimation of the risk, but it is better to be safe.

So, is an extra 4% per 1000 mSv chance of getting fatal cancer worth sleepless nights, and how much radiation dose is one likely to receive?

Let us consider the latter question first. The Canadian Nuclear Safety Commission (CNSC) has stated that a Nuclear Energy Worker (NEW) shall be allowed to receive not more than 20 mSv per year from their work, on average (we will discuss this more fully in a later module). A very dedicated employee who starts young and works for fifty (!) years should therefore receive no more than 1000 mSv. It is important to note, however, that the typical exposure would normally be much less than this.

Our hypothetical employee will hence have an extra 4% chance of developing a fatal cancer. Extra? Note that 25% of people will do so anyway. This employee's chance therefore becomes 29% instead of 25%.

Is it worth worrying about this increased risk of cancer? Obviously, this is a question for the individual. However, a little perspective will be of help. Other professions carry risks for their employees too. Table 1 shows the risk of dying associated with various professions and compares it with the risk associated with radiation. Table 2 compares the risk of dying in a fatal accident at home, on the road or at work, with the risk of dying from various doses of radiation.

Table 1: Average annual risk of death in Canada from fatal accidents for various professions and from radiation exposure.

Occupation	Risk of death per year
Finance	1 in 60,000
Service	1 in 40,000
Trade	1 in 20,000
2 mSv of radiation per year	1 in 12,000
Government (includes police and firefighters)	1 in 11,000
Manufacturing	1 in 11,000
Transportation	1 in 4,000
Construction	1 in 3,000
20 mSv of radiation per year	1 in 1,200
Mining	1 in 1,100
Forestry	1 in 900
Fishing and Hunting	1 in 500

Source: Canadian Nuclear Safety Commission, *Canada: Living With Radiation*. Reproduced with the permission of the Minister of Public Works and Government Services, 2001.

Table 2: Average annual risk of death in Canada from fatal accidents or from radiation exposure.

Hazard	Risk of death
Accidents on the road	1 in 5,000
Accidents at home	1 in 11,000
Accidents at work	1 in 24,000
1 mSv per year (public dose limit)	1 in 20,000
0.05 mSv per year (maximum emission from nuclear facilities)	1 in 400,000
0.001 mSv per year (average emission from nuclear facilities)	1 in 20,000,000

Source: Canadian Nuclear Safety Commission, *Canada: Living With Radiation*. Reproduced with the permission of the Minister of Public Works and Government Services, 2001.

9. Stochastic Effects

"Smoking causes lung cancer." This is the popular wisdom. But just how much tobacco smoke does it take? Will that cigar last Christmas be enough? What about a packet of five cigars every Christmas? Maybe a packet of twenty cigarettes a week for a year? "My Uncle Fred smoked sixty a day and lived to be 95!"

Some people develop lung cancer. Some of these people smoke. What we know is that more people who smoke develop lung cancer than those who don't. What we don't know is if the cancer was caused by the smoke or if it would have occurred anyway. The only conclusion we can draw is that smoking increases the likelihood of suffering lung cancer.

This is an example of a stochastic effect. They occur by chance; among both those exposed to some agent and those who are not. Exposure merely increases the likelihood of the effect occurring, greater exposures leading to greater likelihood.

Long-term low level exposure to radiation is similar in some ways. We can say what effects might arise and how a given dose of radiation affects our chances. However, we cannot be certain that an effect will occur, nor be certain that the radiation was responsible if it does.

10. Deterministic Effects

In contrast with stochastic effects, which occur by chance and which exposure simply makes more likely, a deterministic effect is one which will certainly result from exposure. There will be some minimum amount, a threshold exposure, before the effect will occur and thereafter the severity of the effect will depend on the exposure. Finally, we can look at the effect and know what caused it.

Consider the effects of alcohol. It is necessary to ingest a minimum amount before one shows signs of drinking. Thereafter, the effects will depend on the amount ingested. Finally, it is obvious to an observer if someone has been drinking!

A deterministic effect associated with exposure to radiation is cataract formation. See section 12.6 for more information.

11. Short-Term Somatic Effects

So far we have discussed the effects of exposure to low doses of radiation over relatively long periods of time, i.e. months and years. Such an exposure is said to be *chronic*.

Now let us look at the effects of *acute* exposures. An exposure can be considered acute if it is delivered within a day or so.

We have already seen that ionizing radiation can damage the molecules which comprise our cells. Huge doses can kill the cell outright. For lesser doses, the big chromosome molecules present the largest target to the incoming radiation. If they are damaged then the cells reproductive ability will be impaired or destroyed.

Thus the tissues in our bodies most affected by an acute radiation dose are those in which cells are most rapidly reproducing. These are the skin, the blood-forming tissues (bone marrow of the head and trunk, and lymph nodes), the gonads and the digestive system lining (called the gastrointestinal tract or GI tract).

If we consider the functions of these tissues, we can predict what will happen if reproduction of the cells ceases. We can then determine the symptoms of an acute exposure to radiation. These symptoms are commonly known as *radiation sickness*.

Table 3 outlines the sort of unpleasantness one might anticipate for acute doses of up to about 10 Gy. Table 4 goes into more detail about the stages of acute radiation sickness for life-threatening doses of 4-6 Gy. At higher doses, above 10 Gy, there is also severe damage to the GI tract, resulting in failure of the ability to digest food and contain bacteria. Death is certain.

Exposure to acute doses above about 15 Gy results in damage to the nervous system which can kill the victim through shock in just a few days. Above 100 Gy death is but hours away.

Table 3: Probable Effects of Acute Whole-Body Gamma Doses (mGy).

Dose (mGy)	Probable Effects
0-250	No detectable clinical effects. Delayed effects may occur but are highly unlikely.
250-1000	Slight blood changes with later recovery. Possible nausea. Serious delayed effects are possible but improbable.
1000-2000	Nausea and fatigue. Possible vomiting. Reductions in certain blood cells with delayed recovery.
2000-3000	Nausea and vomiting probable on first day. Two week latency period followed by general malaise, loss of appetite, diarrhoea, moderate loss of weight. Possible death in 2-6 weeks but for most healthy individuals recovery is likely.
3000-6000	Nausea, vomiting and diarrhoea probable in first few hours. Short latent period followed by loss of appetite and general malaise. Then haemorrhage, loss of weight, skin blotches, diarrhoea, inflammation of throat. Some deaths in first weeks, possible eventual death to 50% of individuals receiving about 3500 mGy without medical treatment.
Over 6000	Nausea, vomiting and diarrhoea in first hours. Short latent period followed by diarrhoea, haemorrhage, skin blotches, inflammation of throat and fever by end of the first week. Rapid weight loss and death as early as the second week with possible eventual death of 100% of exposed individuals.

Source: Radiation Protection. Reproduced with permission of J. U. Burnham.

Table 4: Various stages of the effects of an Acute Dose of 4 to 6 Gy.

Time from exposure	Biological effects	Symptoms observed
Stage I: 0-48 hours	Body cells killed by the radiation disintegrate, releasing irritants into the blood system. The body senses this and assumes the last meal to be at fault.	Vomiting, nausea, loss of appetite, fatigue.
Stage II: 2 days - 3 weeks	Following the removal of the irritants, there is a period during which the concentrations of all blood constituents are falling.	Symptoms disappear and patient feels well.
Stage III: After 3 weeks	There is now a severe shortage of blood constituents: Shortage of red cells - poor oxygen transport. Lack of white cells - open to infection. Lack of platelets - no clotting of damaged blood vessels.	Severe lethargy, fever, bleeding and blotchy skin. Fatalities occur here.
Stage IV: After 8 weeks	For the radiation victim to survive Stage III, he must have sufficient blood-forming tissue to sustain life, perhaps aided by medical treatment consisting of massive doses of antibiotics, massive blood transfusions and possible bone marrow transplants. The patient's condition will improve but up to six months are required before full recovery.	

Source: Radiation Protection. Reproduced with permission of J. U. Burnham.

12. Radiation Dose to Specific Organs or Tissues

Radiation exposure that is confined to only one area of the body causes much less injury and risk than does whole-body exposure. Although there may be severe damage to the affected area, the high proportion of the body which was not exposed to radiation will compensate for the loss of any blood-forming cells.

A specific organ may be exposed to external radiation or may receive radiation if radionuclides accumulate in it. An example is the thyroid gland, which absorbs radioactive isotopes of iodine as readily as the normal isotope (this is because they have identical chemical properties). These radioiodines may only give a small whole-body dose but the comparatively small thyroid gland and surrounding tissue would receive quite large doses. In extreme cases, the thyroid might be destroyed.

Some radionuclides are intense beta emitters. If they are present outside the body and a person is within range of the beta particles then they can cause skin damage leading to skin burns. Extensive skin burns from beta radiation, plus high whole-body doses, led to the deaths of several fire-fighters at the Chernobyl nuclear accident.

The following sections provide further details for specific organs.

12.1. Skin

The skin damage from high doses of gamma radiation is irrelevant because a whole-body dose sufficient to kill a person may only redden his skin and cause loss of hair. If one should suffer an acute exposure to beta radiation, only the skin is affected because the vast majority of beta particles do not have enough energy to penetrate any deeper.

In fact, skin is covered by a dead layer about 70 μm thick. The radiation must have enough energy to penetrate this dead layer of skin to do any harm. That's why even high doses from alpha emitters and low-energy beta emitters cause little skin damage (betas need at least 70 keV of energy to penetrate the dead layer).

The skin damage depends upon the dose received, and can be quite nasty. The effects range from temporary erythema (reddening of the skin at around 5 Gy) through moist desquamation (the surface layer peeling off) to necrosis (death of the skin at about 30 to 50 Gy), depending on the period of time over which the doses are received. For such acute exposures, the effects would show up about three weeks later.

The treatment required for high local skin doses is similar to the treatments given to thermal burns. However, injury caused by radiation heals slowly, owing to damage of the underlying tissue.

Hair follicles are more sensitive to radiation than the cells of the skin. An acute dose of 3 to 5 Gy leads to a temporary loss of hair, and this loss becomes permanent after an acute dose of around 7 Gy.

If the dose is spread out over a period of time (weeks), the effect is less. That is, this dose might have to be 3 or more times greater than the acute dose to produce the same effect.

12.2. Blood-Forming System

The dividing cells of the blood-forming system are among the most sensitive to radiation in the body. An acute whole-body dose of 1 Gy can bring about changes in the bone marrow and lymphoid follicles within hours. The peripheral blood count will also be affected. The maximum effect, however, will only be seen approximately 2-5 weeks afterwards.

A decrease in the number of white blood cells leads to a marked weakening of the immune system, and of resistance to infection.

Platelets are the parts of the blood which are necessary to form clots when a blood vessel is ruptured. As a result of exposure to acute whole-body doses, these too will decrease in number at the same time. Damage to blood vessels will therefore result in more bleeding.

Red blood cells transport oxygen from the lungs to all the cells in the body. A decrease in the number of these cells will therefore result in lethargy.

After whole-body doses of several Gy, infection and haemorrhage are the main causes of death. An acute whole-body dose of 3 to 5 Gy is sufficient to cause death in 50% of those exposed, although medical support can make a big difference. If the patient survives the relatively short critical period of two months then recovery is essentially complete and no long-term effects are expected.

12.3. Gastrointestinal Tract

The sensitivity of the cells of the gastrointestinal tract is similar to that of the skin cells. Ulceration, followed by fatal dysentery, is what happens if a large part of the intestine is exposed to acute doses greater than 10 Gy.

Doses greater than 0.5 Gy will cause temporary nausea, vomiting and diarrhoea. Those who have suffered food-poisoning will sympathize.

12.4. Reproductive System

The germ cells of the testes and ovaries are very sensitive to radiation.

In men, a loss of fertility could be observed several weeks after an acute exposure of as little as 0.15 Gy to the testes. This is a temporary effect and recovery will occur over the next few months. However, doses of 3.5 to 5 Gy will cause permanent sterility.

In women, an acute exposure to the ovaries of 0.7 to 1.5 Gy leads to a prompt loss of fertility. Below about 2.5 to 3 Gy, this loss is temporary. However, this number is dependant on age. Older women are more sensitive than younger ones. For example, a young woman may recover her fertility even after an acute exposure to 6 Gy while a woman over forty would almost certainly become sterile after an acute exposure to just 3 Gy.

12.5. Thyroid Gland

As stated previously, the thyroid gland cannot distinguish radioiodines from the non-radioactive isotope. Fortunately, the adult thyroid is fairly resistant to radiation. The threshold for severe impairment of its functions is about 25 to 30 Gy. Children, on the other hand, will suffer adverse effects following thyroid doses of only 7 to 14 Gy. This will result, among other things in retardation of growth.

One of the main concerns with regard to accidental releases from nuclear facilities is the intake of radioiodines. This, however, can be countered by taking a pill which contains a large amount of non-radioactive iodine. This floods the body with normal iodine and, because the thyroid takes up iodine indiscriminately, ensures that very little radioiodine is absorbed by the thyroid. The body disposes of excess iodine, eliminating the radioiodine along with it.

12.6. Eye

The lens of the eye is fairly sensitive to radiation. At high doses, lens opacities (or cataracts) develop within months, progress rapidly and eventually cloud the lens completely. At lower doses, opacities may take years to develop, remain microscopic in size and cause no notable loss of vision.

Based on several studies, it seems that a dose of more than 8 Gy of X- or gamma radiation is required to produce a vision impairing cataract under the exposure conditions typical of radiation workers (i.e., small doses spread out over long periods of time).

12.7. Developing Embryo and Foetus

This topic is covered in great detail in Module 5, *Effects of Radiation on the Foetus*.

13. Radiation Exposure from Background Radiation

Radiation is not just found near nuclear reactors, in medical diagnostic equipment or in any of the myriad devices used by industry. It is also all around us, even in the middle of the wilderness, and has been since time immemorial. The sources of this radiation, known as *background radiation*, are quite natural.

From the Sun and from space we are bombarded with a variety of charged particles and gamma rays. The atmosphere absorbs some of the energy, so the dose received varies with altitude. The average annual dose in Canada is 0.3 mSv. To give some idea of how this varies with altitude, the annual dose in the cities that are about 1 km above sea level in western Canada is approximately 0.4 mSv, while such cities as Bogota, Lhasa and Quito, which are all about 3 km above sea level, receive 1 mSv.

This also means that flying in an aircraft will result in a slightly greater exposure to cosmic radiation. A ten-hour flight at an average altitude of 8 km will result in an increased dose of about 0.02 mSv.

Various radionuclides exist naturally in the soil and rocks, exposing us to gamma radiation from the ground. The average annual dose in Canada is 0.35 mSv, but the actual amount can vary tenfold depending on where one is.

A similar annual dose, 0.35 mSv, is received from radionuclides which are incorporated in our bodies. Potassium-40 is an example.

While there is little that can be done about any of the above, the final natural source is more amenable to amelioration. Small amounts of uranium and thorium occur in the soil and one of their decay products is radon gas. This seeps continuously out of the soil everywhere and accumulates in air inside houses. Its decay products cling to dust particles and are inhaled too. The average concentration in houses in different cities in Canada varies by a factor of ten, depending on the type of soil found locally. The variation in the concentration in individual homes can vary by a factor of one thousand! Table 5 shows some average values in different Canadian cities.

Table 5: Approximate average doses received from radon decay products in household air in various Canadian cities.

City	mSv per year
Vancouver	0.2
Montreal	0.4
Toronto	0.55
Edmonton	0.8
Halifax	1.2
Regina	1.6
Winnipeg	2.2

Source: Canadian Nuclear Safety Commission, *Canada: Living With Radiation*. Reproduced with the permission of the Minister of Public Works and Government Services, 2001.

14. Occupational Doses

It is not just the obvious workers such as hospital radiographers and personnel in nuclear power stations who are exposed to ionizing radiation. Many miners, for example, are exposed to radiation from the uranium, thorium and potassium-40 in various types of rock. Pockets of radon gas can also accumulate in areas of the mine, or be released during mining. The crew of aircraft are exposed to higher doses of cosmic radiation. So too are astronauts!

Table 6 lists the doses received by various occupations. Regulatory requirements and dose limits are discussed in a later module.

Table 6: Average annual radiation doses for various occupations in Canada in 1991.

Occupation	Annual dose (mSv)
Dentist	0.31
Physician	0.73
Nurse	0.4
Veterinarian	0.42
Isotope technician	1.86
Industrial radiographer	5.28
Nuclear fuel processor	3.38
Reactor operator	2.62
Reactor mechanic	3.47
Uranium Miner (gamma)	1.79
Uranium Miner (radon)	5.15

Source: Canadian Nuclear Safety Commission, *Canada: Living With Radiation*. Reproduced with the permission of the Minister of Public Works and Government Services, 2001.

Glossary

Acute exposure	Exposure to high doses over a day or so.
Background radiation	The radiation from natural sources which is present everywhere.
Chronic exposure	Exposure to low doses over a long period of time.
Deterministic effect	An effect which will definitely occur as a result of exposure to more than a threshold amount. The greater the exposure above the threshold, the greater the effect.
Hereditary effect	A consequence of exposure to radiation which affects the descendants of the person exposed.
Latency period	The delay between exposure to radiation and the possible onset of cancer.
Radiation sickness	The term for the symptoms which manifest after an acute exposure to radiation inhibits reproduction among those cells in the body which reproduce most rapidly: the skin, the blood-forming tissues, the gonads and the gastrointestinal tract.
Somatic effect	A consequence of exposure to radiation which affects the person exposed. Somatic effects can be subdivided into stochastic and deterministic effects.
Stochastic effect	An effect which occurs by chance; among both those exposed to some agent and those who are not. Exposure merely increases the likelihood of the effect occurring.

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Training Course

Radiation Safety Officer
(RSO-1)

Effects of Radiation on the Foetus

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Sixth Revision

Effects of Radiation on the Foetus

CONTENTS

1. Introduction

The effects of radiation exposure are most pronounced in cells which are reproducing rapidly. An embryo or foetus develops by very rapid cell division and so is extremely sensitive to ionizing radiation.

2. Exposure during Pregnancy

The most sensitive tissue of the human body with respect to induction of damage by ionizing radiation is the developing embryo or foetus. Many complex processes have to happen at the right time and in the right order for the foetus to develop properly. This allows many opportunities for outside agents, such as radiation, to interfere with the development of the foetus.

Briefly, the development of the human conceptus can be divided into three phases. Firstly, there is the pre-implantation period lasting from fertilization until implantation of the embryo into the uterine wall. Secondly, there is the phase of major organogenesis, which extends until about the 8th week after ovulation. Finally, there is the phase of foetal development, continuing on until birth.

The current information on radiation-induced embryonic damage is derived from animal studies and follow-up of individuals exposed to atomic bomb explosions in Japan using statistical analyses.

Based on these studies, the United States National Council on Radiation Protection recommends that even radiation levels of 50 - 100 mGy (0.05 - 0.1 Gy) present no real danger to the embryo-foetus and the advantages gained by clinical diagnosis by far outweigh the negligible risk of embryonic damage.

For higher radiation doses, the major effect of irradiation during the first phase of development is death of the conceptus, but those that survive appear unimpaired with respect to morphology (shape), size, long-term survival and reproductive fitness. In humans, the effect would simply be noted as a temporary failure to conceive. It is interesting to note that approximately 40% of human embryos are lost after conception to spontaneous abortions, and most of these occur before the pregnancy has been diagnosed.

The possible embryonic or foetal damage from radiation exposure may be classified into two principal types. The first is *teratogenic*, or abnormal foetal development, which may occur on exposure to radiation in the first 12 weeks of pregnancy, when the embryo is in the stage of organogenesis. The second type is *carcinogenic*, or the induction of malignancy, which may occur on exposure to radiation in the second and third phases of the pregnancy; these effects are manifested in the first decade of life.

3. Risks to the Foetus of Prenatal Radiation Exposure

The following is a summary of what is presently known about the risks to the foetus as a result of prenatal radiation exposure.

3.1. Prenatal Death

The risk of prenatal death varies with the stage of pregnancy when the exposure occurs. A measure of the danger of death from exposure to radiation is the dose which kills half of the exposed foetuses. It is known as the LD_{50} (Lethal Dose for 50% of those exposed). A dose lower than this will kill fewer foetuses and a higher dose will kill more. Table 1 shows the LD_{50} for various stages of pregnancy.

Table 1: LD_{50} for various times after conception.

Time after conception	LD_{50}
1 week	1.0 Sv
5 – 7 weeks	1.5 Sv
21 weeks on	3.0 Sv

Source: Canadian Nuclear Safety Commission, *Canada: Living With Radiation*. Reproduced with the permission of the Minister of Public Works and Government Services, 2001.

3.2. Malformations

Malformations may be caused in organs of the foetus developing at the time of the exposure. In humans, there is a threshold of about 50 mGy, below which malformations will not happen.

3.3. Severe Mental Retardation

It is now recognized that the developing brain of the foetus is much more susceptible to malformation caused by exposure to radiation than most other embryonic or foetal tissue. The period of maximum vulnerability to radiation appears to be the time from week 8 through week 15 after conception. The period from week 16 to week 25 may be even more vulnerable, but high levels of radiation exposure almost always lead to spontaneous abortion. In some sense, this is, therefore, a period of lesser susceptibility.

Exposure to radiation prior to 8 weeks or after 25 weeks from conception has not been observed to induce severe mental retardation.

Within the period of maximum vulnerability, i.e., weeks 8 through 15, the frequency of severe mental retardation rises from a "normal" incidence of one case per hundred to approximately 40 cases per hundred at a dose of 1 Sv. The available data, mostly from atomic bomb survivors at Hiroshima and Nagasaki, does not rule out a threshold dose at about 0.1 Sv.

3.4. Mental Impairment of Lower Severity

Mental impairment of lower severity is also apparent in children exposed *in utero.* This mental impairment shows up as a decrease in intelligence scores, changes in the occurrence of major features of physical development, impaired school performance, susceptibility to seizures, and possibly other effects. The severity depends on the radiation dose.

These effects are still being investigated.

3.5. Childhood Cancer

From the studies of children exposed to diagnostic x-rays, it would appear that radiation has a much greater effect for inducing cancer if exposure occurs before birth than if it occurs in childhood or in adult life.

The best estimates are that the risk is substantially greater during the first phase of the pregnancy than that found in the second and third phases, but this is not firmly established and different views are also held by scientists.

Assuming a linear relationship between dose and effect, with no threshold, leads to an estimated risk of approximately 3% per Sv for fatal childhood cancer. The period of increased risk appears to begin at birth and last for 12 years for leukemia and about 10 years for solid tumours. In absolute terms, this means 25 excess fatal leukemia cases and 28 excess fatal cancer cases of other types per million foetuses exposed to 10 mSv of radiation. Because of this increased risk, x-rays are no longer used to diagnose medical problems during pregnancy but have been largely replaced by ultra-sound technologies.

4. Comparison of Risks

We have seen some numbers pertaining to the various effects of radiation on the foetus. However, it would be useful to have some idea of the effects of a given dose of radiation. Table 2 attempts to do just this. Note that it makes no mention of death of the foetus. If irradiation does cause death then it will likely happen before the pregnancy is diagnosed, just like the large number of other spontaneous abortions. Given that the foetus survives to full term, however, Table 2 shows the probability of it being properly formed and without undue risk of developing cancer.

Table 2: Chances of irradiation having no effect on the child.

Dose[*] to conceptus (mSv)	**Child with no malformation (%)**	**Child will not develop cancer (%)**	**Child will not develop cancer or have a malformation (%)**
0	96	99.93	95.93
0.5	95.999	99.927	95.928
1	95.998	99.921	95.922
2.5	95.995	99.908	95.91
5	95.99	99.89	95.88
10	95.98	99.84	95.83

[*] Refers to absorbed dose above natural background. This table assumes conservative risk estimates, and the actual risk could be much less.

Source: Canadian Nuclear Safety Commission, *Canada: Living With Radiation*. Reproduced with the permission of the Minister of Public Works and Government Services, 2001.

5. Hereditary Effects of Radiation

The gonads are one of the most radiosensitive organs in the human body. The threshold radiation dose for permanent sterility in men is 3.5-6 Gy, and for women 2.5-6 Gy. In a retrospective cohort study of survivors of cancer, it was found that radiation therapy directed below the diaphragm depressed fertility in both sexes by about 25%.

Even if fertility is preserved, there is concern regarding the induction of genetic abnormalities (germ-line mutation, chromosomal aberrations) which could potentially produce birth defects and cancer in the offspring. However, a 40-year follow-up of survivors of the nuclear blasts in Japan showed no significant increase in genetically-linked disorders in children born thereafter compared to matched controls of the general population. Based on animal experiments, the estimated dose of gonadal radiation required to produce a mutation rate equal to the baseline spontaneous rate in humans (mutation doubling dose) is between 1 and 1.5 Gy, with an approximate risk of 100 per 10,000 live-born infants per 1 Gy of radiation dose. Yet, recent studies of children of cancer treatment survivors have found no more birth defects than expected for healthy parents, although an increase in low-birth-weight infants or spontaneous abortions was noted, particularly if conception occurred less than one year after cessation of radiation.

Genetic effects were also discussed in *Biological and Health Effects of Exposure to Radiation* in the previous module.

6. Declaration of Pregnancy

According to the Canadian Nuclear Safety Commission's *Radiation Protection Regulations*, every nuclear energy worker (NEW) who becomes aware that she is pregnant shall immediately inform the licensee in writing. On being informed by the NEW that she is pregnant, the licensee shall, in order to comply with the radiation dose limits, make any accommodation that will not occasion costs or business inconvenience constituting undue hardship to the licensee.

For the remainder of her pregnancy, the NEW should receive a radiation dose of no more than 4 mSv, under the regulations.

Regulatory requirements will be covered more fully in a later module.

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Radiation Safety Instrumentation: Radiation Detection and Measurement

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Sixth Revision

Radiation Detection: Instrumentation

CONTENTS

1. Radiation Detectors

People do not possess any sense organs that can detect ionizing radiation. You cannot hear, see, feel, smell or taste ionizing radiation. As a consequence we must rely entirely on instruments for the detection and measurement of radiation.

In radiation protection, a wide variety of instrument types exist for the measurement of radiation. One finds that most instruments are designed for a specific purpose and to measure specific types of radiation.

During the last few decades, there has been relatively little change in the basic detectors. Conversely, there have been enormous advances in the electronic processing of detector signals and in the treatment of the information contained in them. Modern radiation measuring instruments are much smaller and lighter "smart" instruments. Many of these instruments have the capability of logging, manipulating, plotting and analysing data and then storing the results. The information recorded and stored in this fashion is available for downloading into computers and/or other instruments.

Despite the numerous types of instruments available, there are two general categories of radiation measuring instruments commonly used in radiation safety: *survey instruments* and *dosimeters*.

Survey instruments and dosimeters are considered portable instruments. Larger scale radiation detectors which are not portable are typically found in a laboratory setting. Non-portable instruments use the same types of detectors but generally are more sophisticated. Laboratory style instruments are generally used to perform radiation measurements under more controlled conditions.

1.1. Survey Instruments

Survey instruments are radiation detectors designed to measure the *rate* at which the radiation events occur. The rate can be interpreted as the number of particles which are registered in the detector as a function of time. For example, a survey instrument registering 100 cpm when exposed to a gamma source is measuring 100 gamma ray photons every minute.

The count rate is also a measure of the intensity of the radiation. In a high intensity gamma radiation field, the count rate measured by the survey instrument will be higher than in a low intensity gamma radiation field.

A measure of the intensity of the radiation is also a measure of the amount of radiation dose being received as a function of time. This is referred to as the *dose rate*. For example, a measured dose rate of 50 μSv/h means that you would receive a dose of 50 µSv for each hour you spent at the location of measurement. It should be noted that survey instruments measure external dose rates from radiation sources located outside the body.

Modern survey meters are multi-purpose instruments which can be used with a variety of detectors (commonly referred to as probes) to measure X-ray, gamma-ray, alpha, beta and neutron radiation.

In radiation protection, survey instruments are used to monitor and measure radioactive contamination and to ensure that radiation dose rates remain within safe limits.

1.2. Dosimeters

Dosimeters are radiation integrating instruments that record the total amount of radiation it receives in a given time rather than the rate at which it is received. Dosimeters essentially record and store the effects produced by radiation it is exposed to. The response of the dosimeter when read is an indication of the total radiation dose received by the individual while wearing the dosimeter.

Dosimeters are used to measure the total dose received from external sources of radiation including X-rays, gamma, beta and neutron radiation.

2. Principles of Radiation Detection

Although there are many types of instruments, the operating principles for most radiation measuring instruments are relatively few.

Radiation detectors can be gas, liquid or solid. In any case, passage of ionizing radiation through the detector results in energy dissipation through ionization or excitation. This ionization (or excitation) is converted into an electrical pulse which may be processed and registered in a readout device, such as a scalar or ratemeter as a radiation count.

The major types of detectors include:

- gas filled
- scintillation
- semiconductor

A summary of the different types of detector materials and their applications in radiation measurement are presented in Table 1.

Table 1: Radiation effects used in the detection and measurement of radiation.

Effect	Type of Instrument	Detector
Electrical	1. Ionization chamber 2. Proportional counter 3. Geiger counter 4. Solid state detector	1. Gas 2. Gas 3. Gas 4. Semiconductor
Chemical	1. Film 2. Chemical dosimeter	1. Photographic emulsion 2. Solid or liquid
Light	1. Scintillation counter 2. Cerenkov counter	1. Solid or liquid (scintillation) 2. Solid or liquid (scintillation)
Thermo-luminescence	Thermoluminescent dosimeter	Solid
Heat	Calorimeter	Solid or liquid

3. Gas Filled Detectors

A typical gas filled detector is a gas filled chamber with two coaxial electrodes which are electrically insulated from each other. A battery connected across the chamber makes the centre wire electrode positive (anode) and the outer cylindrical electrode negative (cathode). When a bias is applied across the anode and cathode, an electric field is setup in the chamber. In essence, the chamber acts like a capacitor. Gas filled detectors are commonly filled with dry air but may utilize, nitrogen, argon, xenon, krypton and carbon dioxide. A simple gas filled detector configuration is shown in Figure 1.

Radiation striking the gas molecules in the chamber will produce ion pairs (electrons and positive ions). The electrons will be attracted to the anode and the positive ions to the cathode. This results in the flow of an electrical current through the circuit and the generation of a voltage pulse. Each voltage pulse corresponds to the passage of a single particle through the detector.

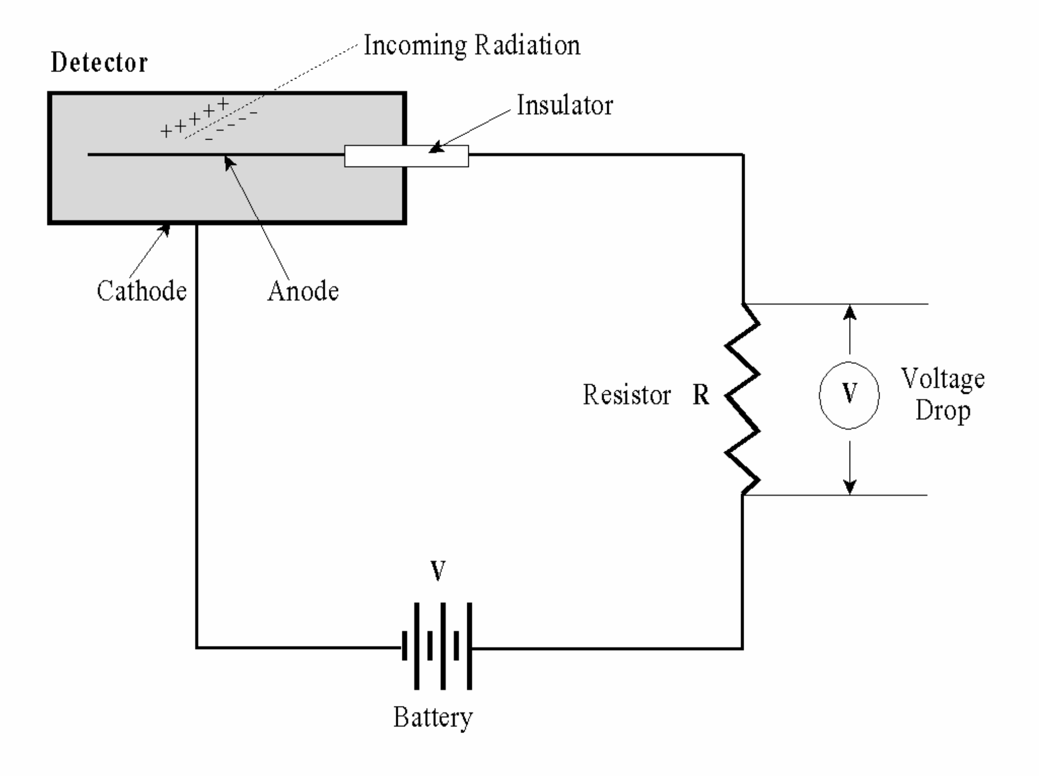


Figure 1: Basic circuit for a gas filled detector.

The detailed behaviour of gas filled detectors depends on a variety of factors including the composition and pressure of the gas, the strength of the electric field in the chamber (applied voltage) and the method of collection and measurement of the charge.

If a constant flux of radiation is permitted to pass through the gas filled detector and the voltage V to the detector is varied, several well-defined regions of importance in radiation measurement may be identified: ionization region, proportional region and Geiger Müller region (see Figure 2).

Gas filled detectors which operate in the aforementioned voltage regions are respectively called ionization chambers, proportional counters and Geiger Müller counters.

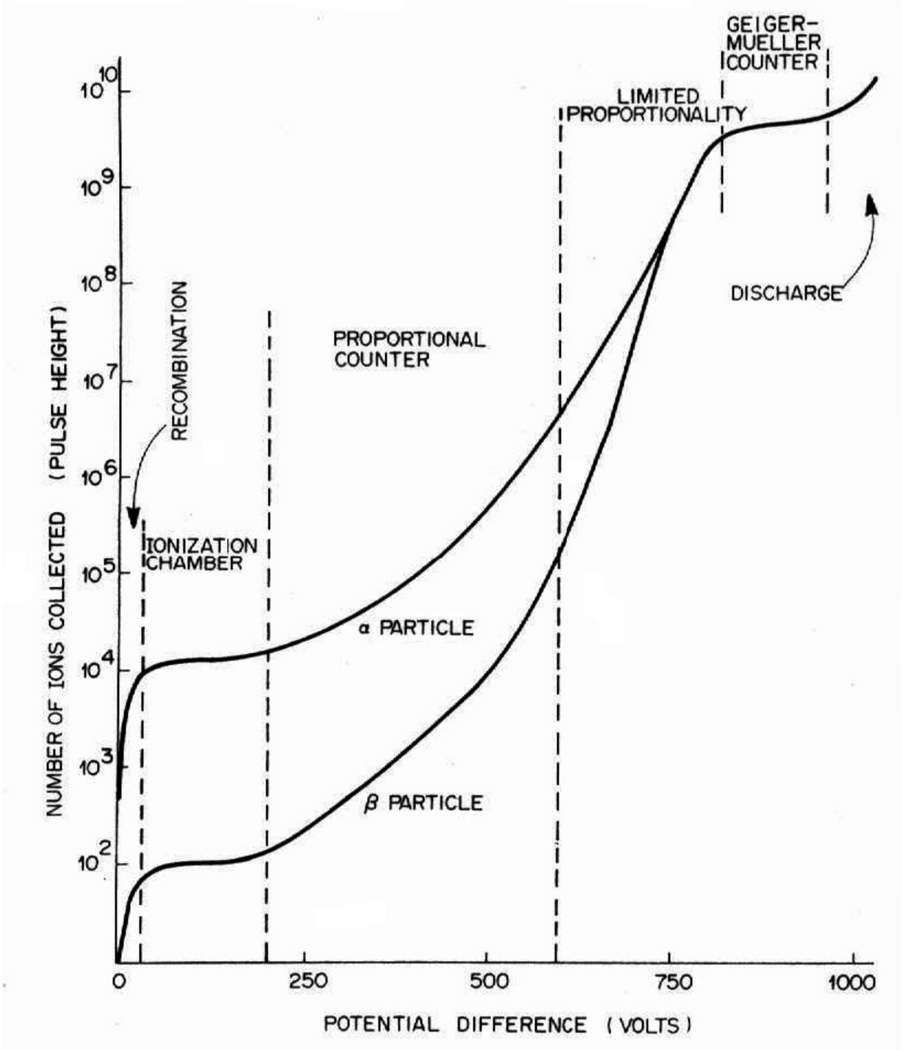


Figure 2: Regions of operation of gas filled detectors.

3.1. Ionization Chamber

As the operating voltage in a gas filled detector is increased from zero through relatively low voltages, the first region of interest is the *ionization chamber region* (see Figure 2). In the *ionization chamber region*, all the positive ions will be collected by the cathode and all electrons will be collected by the anode (see Figure 1).

In the *ionization chamber region*, the range of voltages is great enough to collect the ions before a significant fraction of them can recombine yet not great enough to accelerate the ions sufficiently to produce secondary ionization by collision with other gas molecules (see Figure 3). In this region, the number of electrons collected by the anode will be equal to the number produced by the primary ionizing particle; the gas amplification factor is equal to one. The pulse size, accordingly, will be independent of the detector operating voltage, and will depend only on the number of ions produced by the primary ionization particle during its passage through the detector.

As shown in Figure 2, the *ionization chamber region* extends to approximately 200 volts. The exact value of this voltage is a function of the type of gas, the gas pressure and the size and geometric arrangement of the electrodes.

When radiation strikes an ionization chamber at a steady rate, ion pairs will be produced at a steady rate resulting in the generation of an electric current through the detector electrical circuit (see Figure 1). These currents, albeit very small, are a measure of the rate of ionization in the chamber. The currents are measured indirectly by measuring the voltage drop across the resistor R shown in Figure 1. Because of the small currents, generally the resistor is made very large (billions of ohms) to magnify the voltage pulse because V =iR.

The fact that the pulse size from a counter operating in the ionization chamber region depends on the number of ions produced in the chamber makes it possible to use this instrument to distinguish between radiations of different specific ionization such as alpha and beta particles or gamma radiation. With the use of a discriminator (an electronic filter which blocks pulses below a specified height), beta and gamma radiation can be filtered out. This is illustrated by the two curves shown in Figure 2. Alpha particles produce much greater ionization resulting in a larger signal output than for beta radiation (or gamma radiation).

Because ionization chambers measure ionization, they are used principally to measure radiation exposure rates. Recall that exposure is a measure of the ionization per unit volume of air in units of C/kg or roentgen (R). Exposure can be used to approximate radiation dose (mSv). Ionization chambers are most commonly used to monitor gamma and X-ray radiation although they can be designed to measure beta particles and thermal neutrons.

Some examples of typical ionization chambers are shown in Figure 4. Some ionization chambers such as pocket dosimeters are not continuously powered by a voltage supply. Pocket dosimeters are charged to a specified voltage prior to use. When exposed to radiation, the voltage across the chamber decreases as a result of the production of ions and when read provides an indication of the radiation exposure. Dosimeters will be discussed in more detail later in this training module.

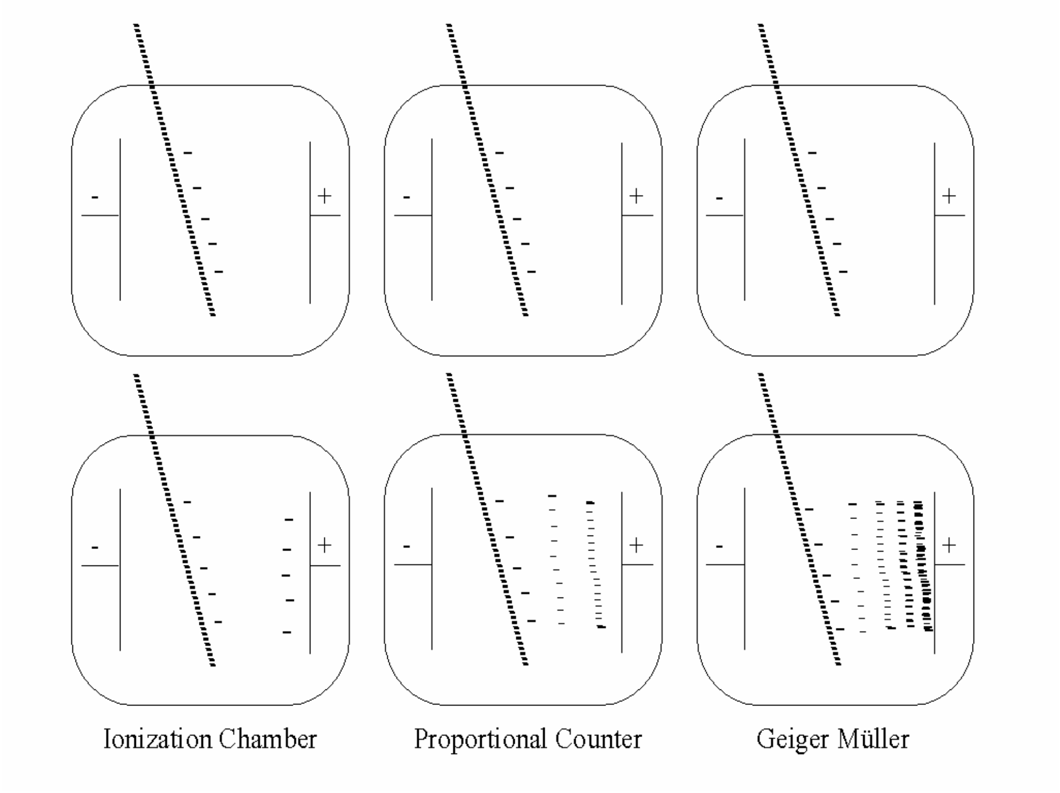


Figure 3: Gas filled detector regions of interest.

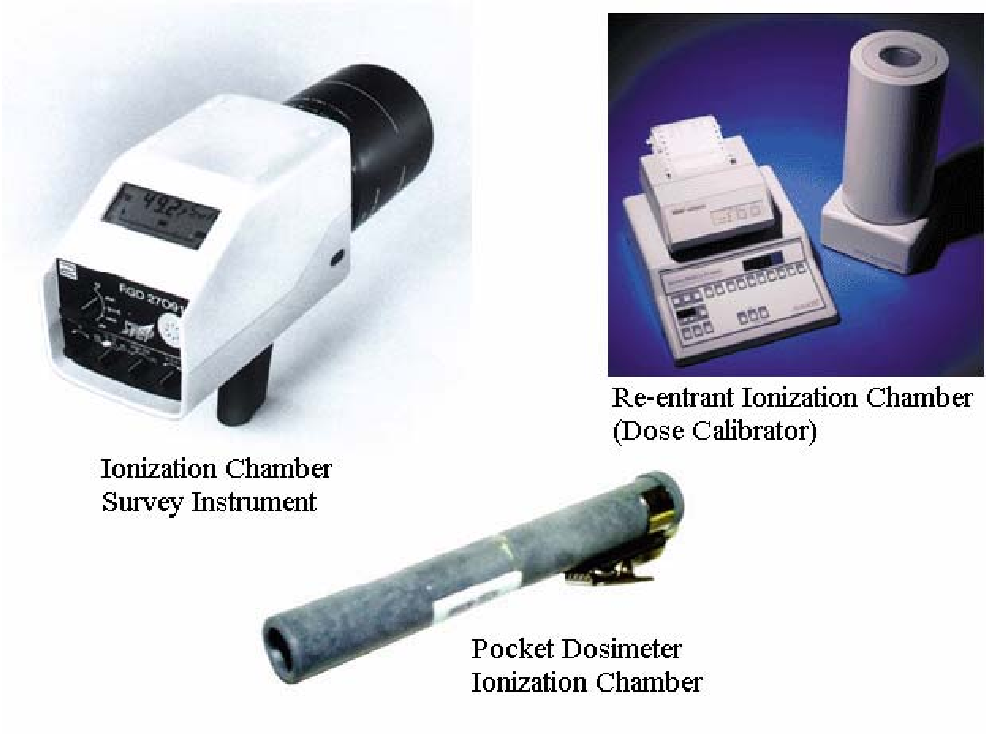


Figure 4: Example of ionization chamber detectors. (From Automess)

3.2. Proportional Counter

One of the main disadvantages of operating a counter in the *ionization chamber region* is the relatively weak output pulse, which requires either much amplification or a high degree of input sensitivity in the associated electronics. To overcome this difficulty and yet take advantage of pulse size dependence on ionization for the purpose of distinguishing between radiations, the gas filled detector may be operated as a proportional counter.

As the voltage across the gas filled detector is increased beyond the *ionization chamber region*, a point is reached where secondary electrons are produced by collision with gas molecules. This is the beginning of the *proportional counter region* (see Figure 2 and Figure 3). In the *proportional counter region*, the number of electrons collected at the anode is much greater than in the *ionization region*, but *proportional* to the original number of primary ions produced in the detector.

At first glance, the difference between an ionization chamber and a proportional counter seem inconsequential. However, upon closer investigation, one finds that the differences, subtle as they may be, are important. The current from an ionization chamber is made up of the average of many bursts of collected charge, each burst being caused by an ionizing radiation event in the chamber. Ionization chambers measure the amount of ionization produced in the chamber each second. Proportional counters amplify each of these individual bursts so that each ionizing event is detected separately. Therefore, proportional counters measure the number of ionizing events, whereas, ionization chambers measure the amount of ionization produced by these events. This is why a proportional counter is a "counter."

Proportional counters can also be used to distinguish between different types of radiation. Consider the plateau curve shown in Figure 5 for alpha and beta radiation. At point A on the plateau curve, the pulses produced by alpha particles that traverse the counter are just great enough to get by the discriminator. A small increase in voltage to the detector causes a sharp increase in counting rate because all the output pulses due to alpha particles now exceed the input sensitivity of the counter. Further increase in high voltage to the detector has little effect on the counting rate and results in a "plateau", a span of high voltage over which the counting rate is approximately independent of the operating voltage (see Figure 5). With the detector operating in the alpha plateau region, the pulses due to beta particles are still too small to get by the discriminator. However, as the high voltage is increased, point B on the plateau curve is reached, where the gas amplification is great enough to produce output pulses from beta particles that exceed the input sensitivity of the counter. This leads to another plateau where both alpha and beta particles are counted. By subtracting the alpha count rate from the alpha-beta count rate, the beta particle activity may be obtained.

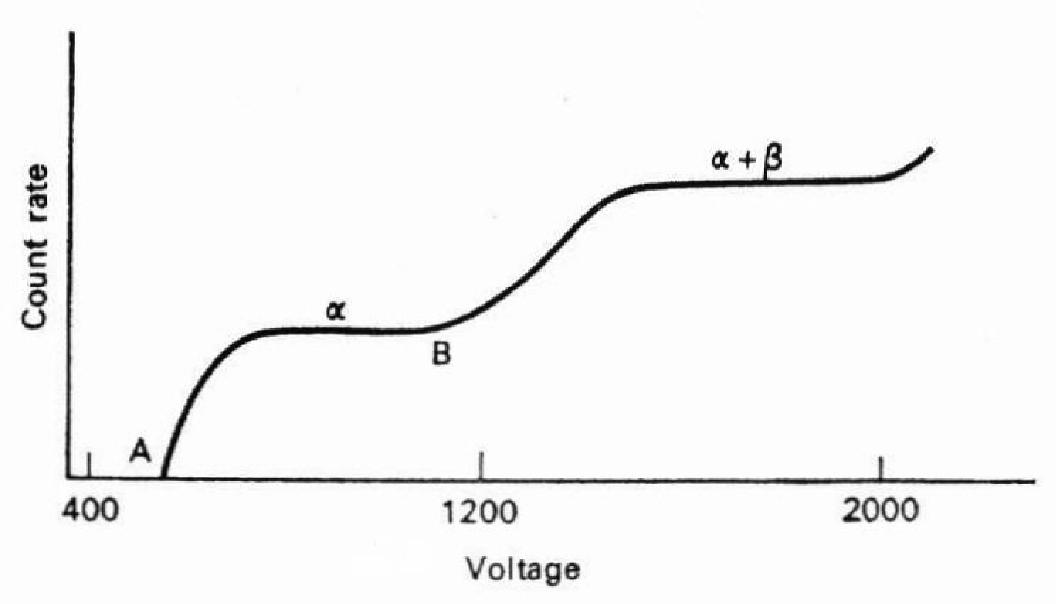


Figure 5: Alpha-beta particle plateau curve. (Source: Cember, Herman, *Introduction to Health Physics*, Third Edition, McGraw-Hill Health Professions Division, New York, 1996. Fig 9.5)

Proportional counters can be used to measure gamma, alpha, beta and neutron radiation, and are available in a number of formats including hand held survey instruments, laboratory style and larger scale detectors such as whole-body monitors (see Figure 6).

Proportional counters designed to measure alpha and beta radiation are equipped with thin Mylar windows of thickness 1 - 2 mg/cm^2 (7 - 14 µm). Without this thin entrance window, the alpha and beta radiation would be stopped before entering the detector.

Neutron proportional counters typically consist of a cadmium-loaded polyethylene sphere filled with boron trifluoride gas (BF_3). The boron isotope, B-10, in the BF_3 gas absorbs neutrons and emits alpha particles. The intensely ionizing alpha particles are detected to provide a measure of the neutron radiation intensity. With the proper discriminator settings, gamma radiation pulses can be filtered out during signal processing. The cadmium-loaded polyethylene construction of the sphere corrects the response of the instrument so that it reads directly in mSv/h.

One of the main disadvantages of gas proportional counters is that they require a very stable power supply and amplifier to ensure constant operating conditions. This is difficult to provide in a portable instrument. This is why proportional counters tend to be used more in fixed or laboratory style instruments.

Figure 6: Examples of gas proportional counters. (From Eberline, Syntrex, and Canberra)

3.3. Geiger-Müller Counter

If we continue to increase the operating voltage in a gas filled detector beyond the *proportional region*, we will enter the *Geiger-Müller region*, commonly referred to as the *Geiger region* (see Figure 2).

In the *Geiger region*, electrons produced in secondary ionizations gain so much energy that they in turn produce more ion pairs, which lead to further ion pairs again, and so on, until the detector tube is delivering the maximum number of electrons it can produce (see Figure 3). In other words, when a single ionizing event occurs, an “avalanche” of electrons is produced in the detector. This avalanche extends over the entire length of the detector anode and results in a very large output pulse.

In the *Geiger region* the size of all pulses is independent of the nature of the primary ionizing particle. Consequently, the counter cannot distinguish among the several types of radiation such as with the ionization chamber or proportional counter. The Geiger counter will deliver a pulse each time a particle causes ionization in its detector making it well suited to the detection of single individual particles.

As with the *ionization region*, the *Geiger region* too has a wide range of operating voltages over which the counting rate is approximately independent of the operating voltage (see Figure 7). This plateau in the *Geiger region* extends approximately from the voltage which results in pulses great enough to be passed by the discriminator to that which causes a rapid increase in counting rate that precedes an electrical breakdown of the detector gas.

At voltages above the *Geiger region*, the electric field in the chamber is sufficiently strong to ionize gas molecules without any need for ionizing radiation. This is commonly referred to as the *breakdown or discharge region*. In the *discharge region*, the detector discharges continuously like that of a fluorescent light. Permanent damage to the detector is likely to occur if operated in the *discharge region*.

Upon inspection of the *Geiger region* plateau shown in Figure 7, one finds that the counting rate slowly increases with increasing operating voltage resulting in a small slope in the plateau region. In the *Geiger region*, the avalanche is already extended along the entire length of the anode. Increasing the voltage causes the avalanche to spread radially resulting in increased counting rate and consequently a slight positive slope in the plateau. The typical slope for a Geiger counter is 3% per 100 V. The operating voltage is usually approximately one-third to one-half the distance from the knee of the curve of count rate versus voltage (see Figure 7). Proportional counter plateau operating voltages for alpha and beta radiation are selected in much the same manner as for a Geiger counter.

When the positive ions are collected after a pulse, they give up their kinetic energy by striking the wall of the tube. Most of this kinetic energy is dissipated as heat. Some of it, however, excites the atoms in the wall. In falling back to the ground state, these atoms may lose their excitation energy by emitting ultraviolet (UV) photons. Since at this time the electric field around the anode is re-established to its full intensity, the interaction of UV photons with the gas in the counter may initiate an avalanche, and thereby produce a spurious count. Prevention of such spurious counts is called *quenching*.

Quenching may be accomplished either electronically, by lowering the anode voltage until all the positive ions have been collected, or chemically, by using a self-quenching gas.

A self-quenching gas is one that can absorb UV photons without becoming ionized. One method of doing this is to introduce a small amount of an organic vapour, such as alcohol or ether, into the tube. The energy from the UV photon is then dissipated by dissociating the organic molecule. Such a tube is useful only as long as it has a sufficient number of organic molecules for the quenching action. In practice, an organic vapour Geiger counter has a useful life of about 10^8 counts. Organic vapours are used for quenching primarily in gas-flow-through proportional counters where there is a continuous supply of gas to the detector. Self-quenching also results when the counting gas contains a trace of a halogen. In this case, the halogen molecule does not dissociate after absorbing the energy from the UV photon. The useful life of a halogen-quenched counter, therefore, is not limited by the number of pulses that have been produced in it. Halogens are used in portable Geiger counters such as in a survey instrument.

Geiger counters are available in a number of formats and are most commonly used in survey instruments. In some survey instruments, the Geiger detector is housed inside the instrument and in others an external Geiger detector (probe) is attached. A number of different Geiger counter survey instruments are shown in Figure 8. Some models of electronic dosimeters such as the one shown in Figure 9 also use miniature Geiger detectors.

There are many types of Geiger probes (GM probes) available for survey instruments (Figure 10). The most common types include the end window, side window and pancake GM probes.

Most GM probes may be used for alpha/beta/gamma measurements although some are designed only to measure photons (X-ray and gamma radiation).

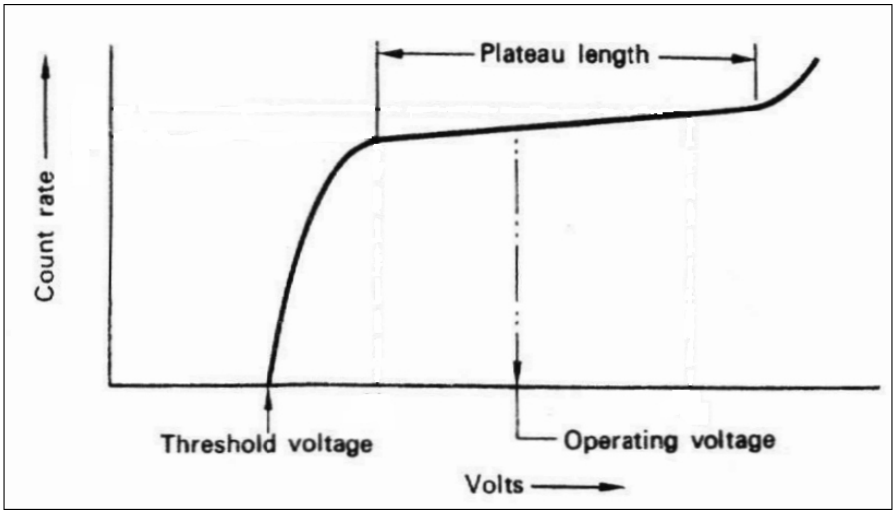


Figure 7: Operating characteristics of a Geiger counter.

Figure 8: Examples of modern Geiger counter survey instruments. (From Bircon, Automess, and Ludlum)

Figure 9: Electronic dosimeter with Geiger detector. (From Automess)

End Window GM Probes Side Window GM Probes Pancake GM Probes

Figure 10: Examples of typical Geiger Müller (GM) probes used in survey instruments. (From Ludlum, Eberline, and Victoreen)

End window and side window GM probes are equipped with a beta shield allowing for the measurement of gamma radiation and beta radiation separately (see Figure 10). With the beta shield in closed position (on), only gamma radiation is detected (very energetic beta particles ("hard betas" (>1 MeV)) may be detected as well). With the beta shield in the open position (off), the GM probe measures beta and gamma radiation simultaneously. The beta radiation field is determined by subtracting the reading with the beta shield in the closed position from the reading obtained when the beta shield is in the open position. As alpha radiation only travels a few centimetres in air, it can be ignored in beta/gamma measurements simply by having the beta shield in the closed position or by moving the detector back from the source when taking readings.

The pancake GM probe, often referred to as a "frisking probe", is most commonly used for alpha/beta/gamma contamination monitoring. The pancake GM probe has a wider detector surface area than the end window or side window GM probes and has a thin Mylar or mica window on the face of the probe to allow alpha radiation to enter the detector. As shown in Figure 10, pancake GM probes are also equipped with a metal screen on the front of the detector to provide additional protection.

When properly designed and calibrated, Geiger counters may be used to measure radiation exposure rates and dose rates. This statement seems to contradict what has been said thus far about Geiger counter. If the output pulse in the Geiger region is independent of the radiation type, how then can a GM detector measure exposure? Recall that exposure is a measure of the ionization per unit mass of air and is measured in units of C/kg or R. Also recall that 1 C/kg = 3877 R and that in air, 1 R ≈1 rad. If each incoming photon produced C C/kg, then the exposure produced by N photons would be N×C and a count per minute scale could be recalibrated to read directly in exposure rate. Most survey instruments still use R as the primary exposure unit and give exposure rates in R/h.

As one might expect, the level of ionization is dependent to some extent on photon energy. Higher energy photons produce more ionization than lower energy photons. The typical response of a Geiger counter as a function of photon energy is represented by the dotted line in Figure 11. The relative response is the ratio of the indicated exposure rate and the true exposure rate. As shown in Figure 11 by the dotted line, a Geiger counter will overestimate the exposure rate by a factor of about 6 in the low photon energy range (~50-100 keV range).

By using suitable shielding and tube wall design to attenuate the lower energy photons, the solid response line shown in Figure 11 can be achieved for a Geiger counter. It should be noted that not all commercially available Geiger counters include this additional shielding. Geiger counters designed to improve response at low photon energies are referred to as “energy compensated.” Geiger counters that are not designed to attenuate low energy photons are referred to as “non-compensated.” Examples of responses from two commercially available survey meters are shown in Figure 12.

Between photon energies of 50 keV and 3 MeV most energy compensated Geiger counters, when properly calibrated, will read to within 25% of the true dose rate.

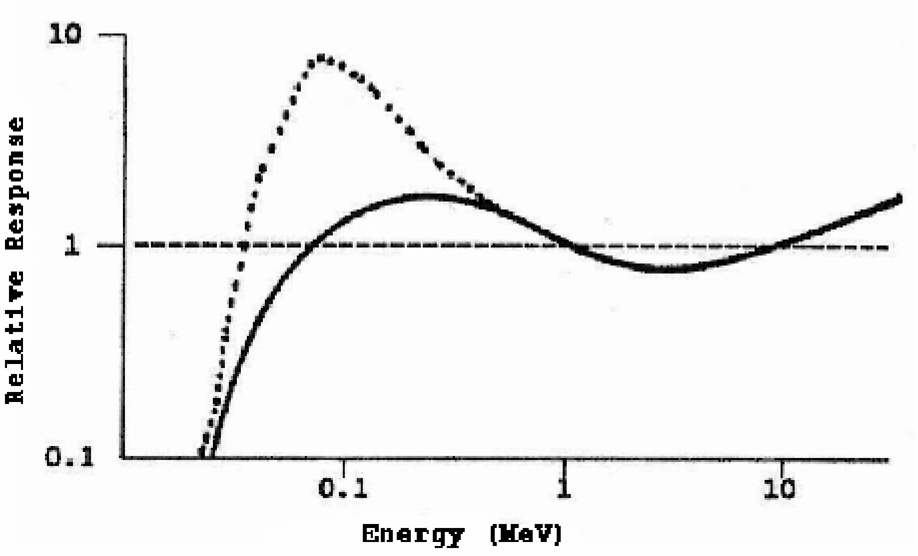


Figure 11: Geiger counter photon energy dependence (from J.U. Burnham). The dotted line shows the response of the Geiger tube when it is not energy compensated, whereas the solid line shows the response with compensation.

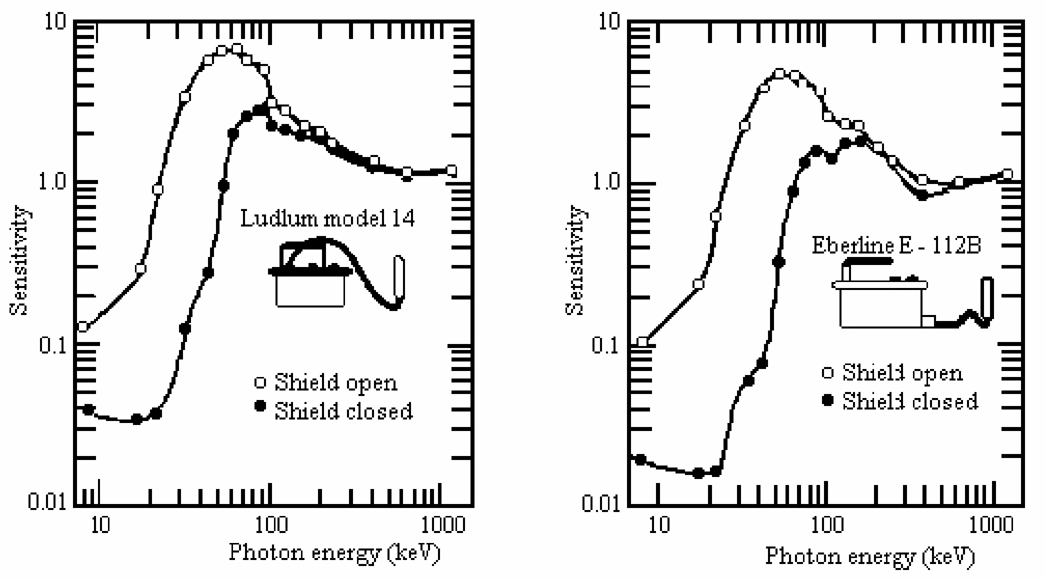


Figure 12: Geiger counter sensitivity as a function of photon energy. The sensitivity is defined as the ratio of indicated exposure over the true exposure.

4. Scintillation Detectors

4.1. Luminescence

When radiation interacts with an atom, two outcomes are possible; ionization or excitation. If ionization occurs, an atomic electron (usually the least tightly bound electron) is knocked out of the atom resulting in a negatively charged electron and the remaining positively charged atom. The resulting ions are referred to as an ion pair.

If excitation occurs, the atomic electron is not removed, but rather knocked into a higher energy level. If this occurs, the atom is said to be in an "excited" state. Because atoms seek to be stable, the electron will return to its original energy level ("ground state"). When this occurs, electromagnetic energy is emitted, often in the visible light range. The energy (frequency) of the electromagnetic radiation is precisely equal to the difference in the energy levels of the electron. This phenomenon is collectively referred to as luminescence.

Depending on the time required for the electron to return to its ground state, the luminescence process is further classified as fluorescence or phosphorescence. The phenomenon of fluorescence is independent of temperature. Phosphorescence is not. When luminescence occurs as a resulting of heating the atoms, the phenomenon is called thermoluminescence.

4.2. Scintillation Materials

Luminescent materials are referred to as scintillation materials or simply "scintillators." Scintillation detectors measure radiation by detecting the light produced in excitation reactions when radiation interacts with the detector. This is quite different from gas filled detectors which detect radiation by measuring the ionization produced in the detector gas.

Scintillation materials are also called phosphors. There are several phosphors available in solid or liquid form for use in scintillation detectors. Some of the most common phosphors are given in Table 2.

To improve light output, most solid phosphors are doped with "impurities" called activators. In Table 2, activators are given in brackets next to the phosphor. When activators are incorporated into the structure of the phosphor, they produce impurity energy levels. Electrons are more easily excited to these impurity levels, and on de-excitation, produce visible light.

Table 2: Common scintillation materials.

Phosphor	Solid/ liquid	Radiation measured	Applications
NaI (Tl)	Solid	gamma rays, X-rays	• radiation monitoring • nuclear spectroscopy • nuclear medicine • research
CsI (Tl)			
Plastics			
ZnS (Ag)	Solid	gamma rays, X-rays, alpha and beta particles	• radiation monitoring • contamination monitoring
Toluene	Liquid	low energy gamma rays, X-rays and beta particles	• radiation monitoring • bioassay
Xylene			
Pseudodocumene			
Alkyl benzene			

Liquid scintillators are used for the measurement of low energy X-ray and gamma and beta radiation which may be severely attenuated or totally stopped by the window material of solid scintillation detectors such as NaI(Tl). Liquid scintillators consist of a cocktail of an organic solvent, one or more fluorescent solutes and an emulsifier to ensure proper mixing of samples. In liquid scintillation detectors, the sample to be measured is actually mixed with the scintillator in solution.

Scintillation detectors are ideal for the measurement of X-ray and gamma radiation because of their relatively high detection efficiency (>20%). In gas filled detectors, the detection efficiency is very low (~1%). Why?

4.3. Principles of Operation

Scintillation detectors are available in many different formats, sizes and range in complexity. Despite this, all scintillation detectors have two primary components: a phosphor and a photomultiplier tube (PMT). The phosphor produces light pulses when ionizing radiation strikes it, and the PMT converts the light pulses to a current pulse.

To better understand the operation of a scintillation detector, consider the NaI(Tl) scintillation detector as shown in Figure 13. The NaI(Tl) is by far the most common type of scintillator used for X-ray and gamma radiation measurements.

The NaI(Tl) crystal is sealed in a cylindrical aluminium container to prevent it absorbing atmospheric moisture and losing its ability to transmit scintillation photons to the photocathode (see Figure 14). One face of the container is thin to permit radiation entry. For low energy X-rays and gammas, the window may be made of beryllium or Mylar. The opposite face of the container consists of a quartz window which is transparent to the near ultraviolet scintillations produced by the phosphor, and the inside surface of the container is coated with a material (aluminium oxide) which reflects this wavelength of light (see Figure 13).

The quartz window is optically coupled by a light-transmitting gel to the face of an evacuated PMT. The PMT is connected to a base designed to supply voltage to the various electrodes, and collect the signal pulse from the PMT anode (see Figure 15).

When radiation strikes the NaI(Tl) phosphor it imparts some (Compton scattering) or all (photoelectric effect) of its energy to atomic electrons. Atoms are either ionized or left in an excited state. When excited atoms return to their ground state, light is emitted. The intensity of this scintillating light is proportional to the energy dissipated by the radiation.

The light generated strikes the photocathode which is a very thin layer of caesium-antimony compound deposited on the inside of the PMT's transparent end window. The photocathode emits photoelectrons under action of the light. The number of electrons it emits is proportional to the intensity of the light striking it.

Located near the photocathode is the first series of structures called dynodes (see Figure 13 and Figure 15). Each dynode is at a potential which is approximately 100 volts more positive than that of the previous dynode. With such an arrangement, electrons released by the photocathode are promptly accelerated to the first dynode, which they strike so forcibly that more electrons are knocked out of its surface. These electrons are accelerated to the second dynode, where they release more electrons. This multiplication of electrons at each dynode is called secondary emission; the number of electrons released by a dynode is about 3 to 5 times the number striking it. The multiplication by a string of dynodes can therefore be enormous. A typical PMT with 14 dynodes would give a total multiplication of approximately 4^{14}, or about 270 million.

Beyond the last dynode there is the anode, which collects all of the electrons and sends them to the PMT's output circuit. The output circuit is designed to give a voltage pulse whose height is exactly proportional to the total number of electrons collected by the anode and thus proportional to the energy deposited in the phosphor by the photon.

The pulses delivered by the PMT are amplified, passed through a discriminator and fed to a ratemeter or scalar. Recall that a discriminator is used to block pulse heights below a certain height. A ratemeter registers the rate at which the events are occurring. A scalar simply adds up the total number of events. The schematic of a complete scintillation detector is shown in Figure 16.

These same principles of operation apply to all types of scintillation detectors whether solid or liquid.

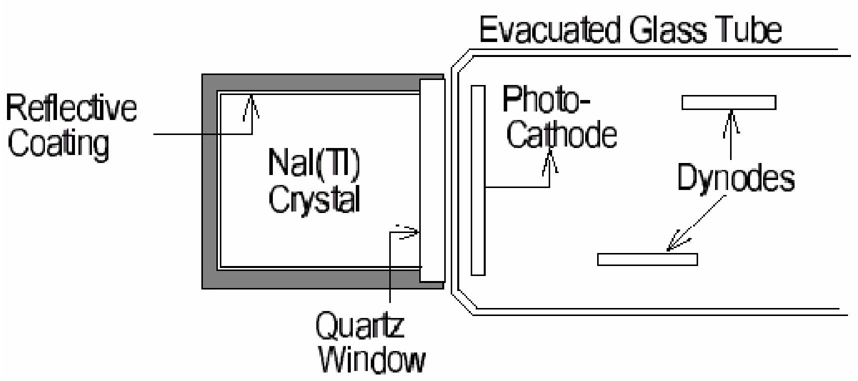


Figure 13: NaI(Tl) detector assembly. (Source: Canadian Nuclear Safety Commission, *Radiation Safety Officers Handbook Info-0718*. Reproduced with the permission of the Minister of Public Works and Government Services, 2001)

Figure 14: NaI(Tl) scintillation detector probe.

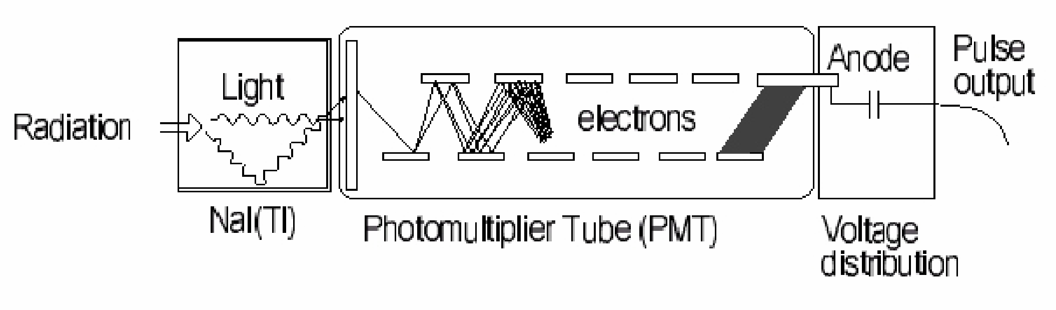


Figure 15: NaI(Tl) detector operation. (Source: Canadian Nuclear Safety Commission, *Radiation Safety Officers Handbook Info-0718*. Reproduced with the permission of the Minister of Public Works and Government Services, 2001)

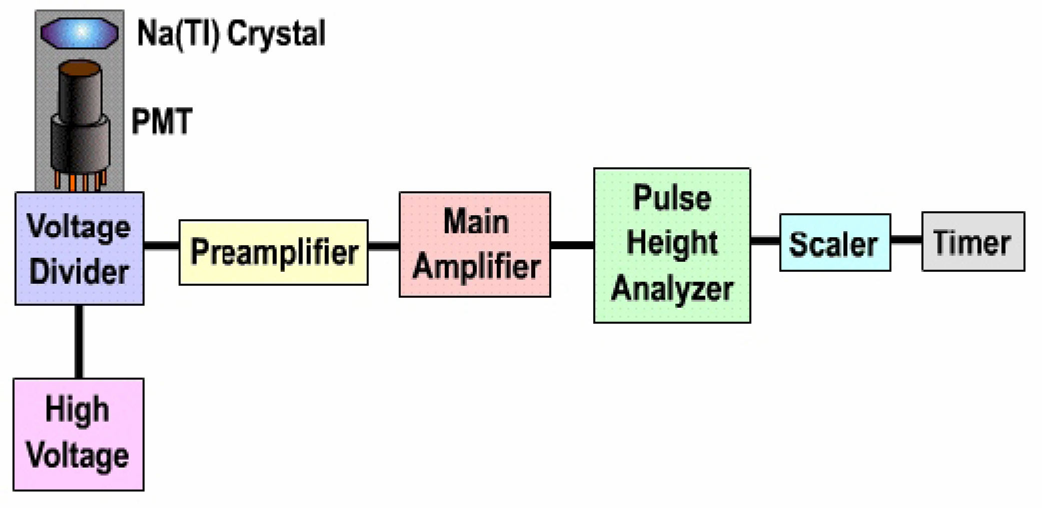


Figure 16: Scintillation detector electronics.

4.4. Types of Scintillation Detectors

4.4.1. Gamma and X-Ray Radiation Scintillation Detectors

For X-ray and gamma radiation measurement, the detector used most frequently is a sodium iodide (NaI(Tl)) crystal activated with thallium (Tl) (0.2%). The high density crystal, together with its high effective atomic number, results in a high detection efficiency. The efficiency of a NaI(Tl) scintillation detector is show in Figure 17 for various detector formats and sizes.

Other phosphors including CsI(Tl) and various plastics are also used as gamma scintillators. The selection of phosphor will depend largely on the application and the energy of the radiation being measured.

Because the signal output of a scintillation detector is proportional to the energy dissipated by the radiation, scintillation detectors can be used to discriminate between different types of radiation and between different energies of the same radiation. Consequently, scintillation detectors have applications in nuclear spectroscopy. Nuclear spectroscopy will be discussed in more detail later in this training module.

X-ray and gamma radiation scintillation detectors range in size from hand-held instruments to detectors the size of an automobile. X-ray and gamma radiation scintillation detectors are used in a wide variety applications including environmental radiation monitoring, contamination monitoring, nuclear spectroscopy, whole body counting, nuclear medicine and research, just to mention a few.

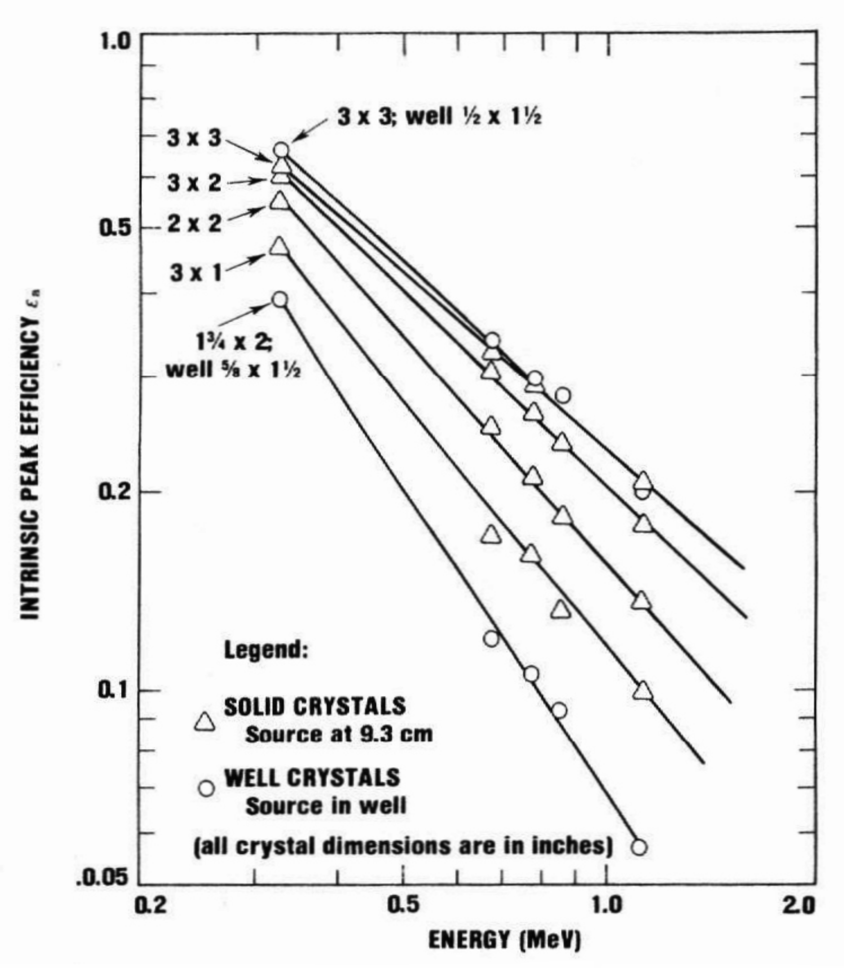


Figure 17: NaI(Tl) efficiency. (From EG&GORTEC)

Some examples of the range of scintillation materials and detectors available are shown in Figure 18 through Figure 22.

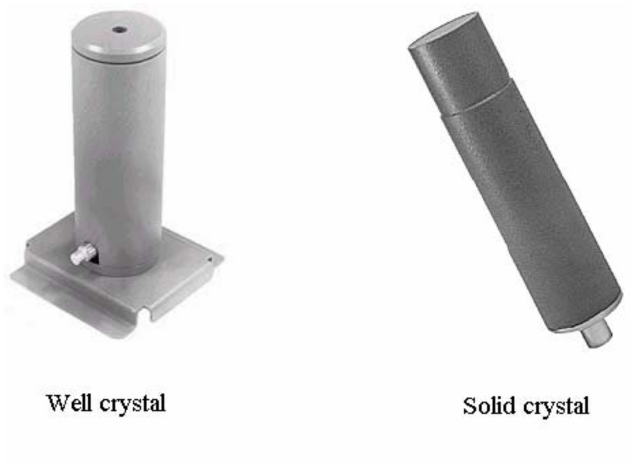
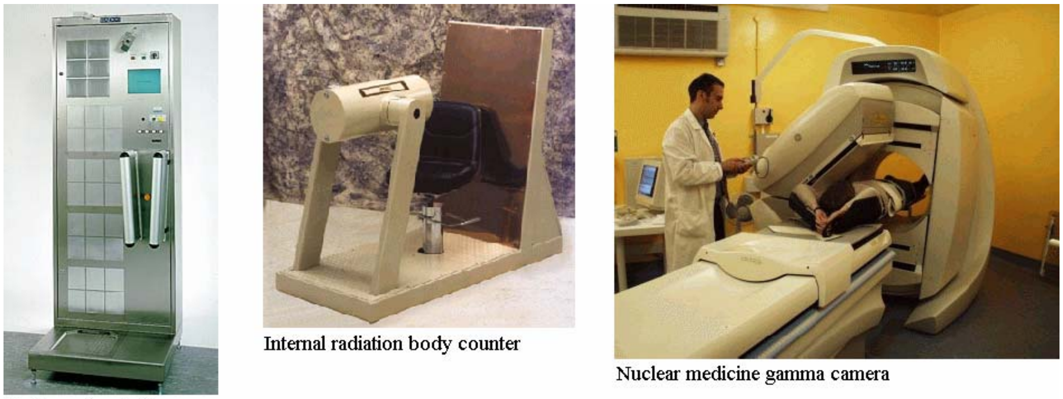


Figure 18: Scintillation materials. (From Ludlum)

Figure 19: Hand-held and portable gamma scintillation detectors. (From Automess, Bicron, and EXPLORANIUM)

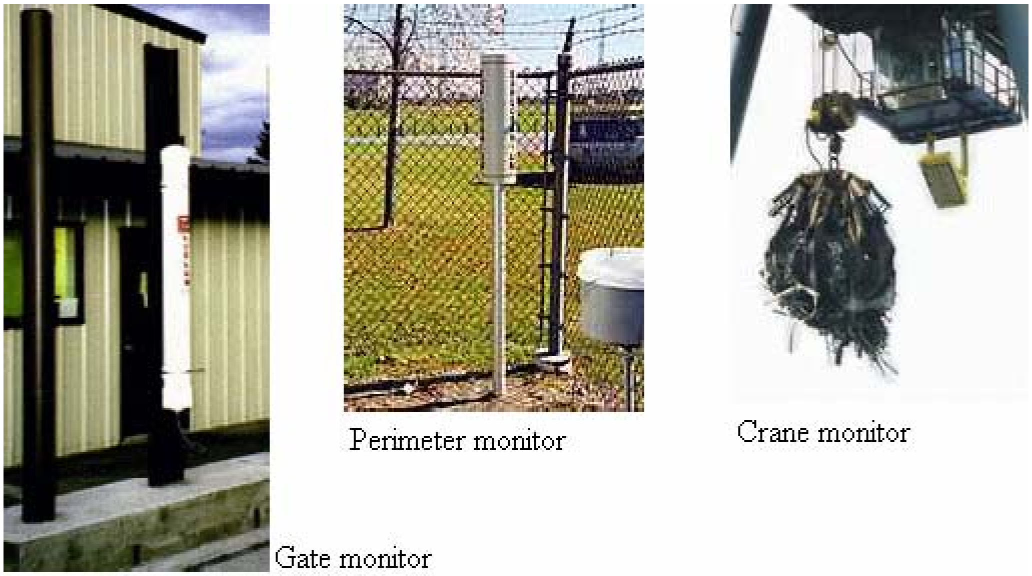


Body contamination monitor

Internal radiation body counter

Nuclear medicine gamma camera

Figure 20: Contamination, whole body, and nuclear medicine scintillation detectors. (From EG&G ORTEC)

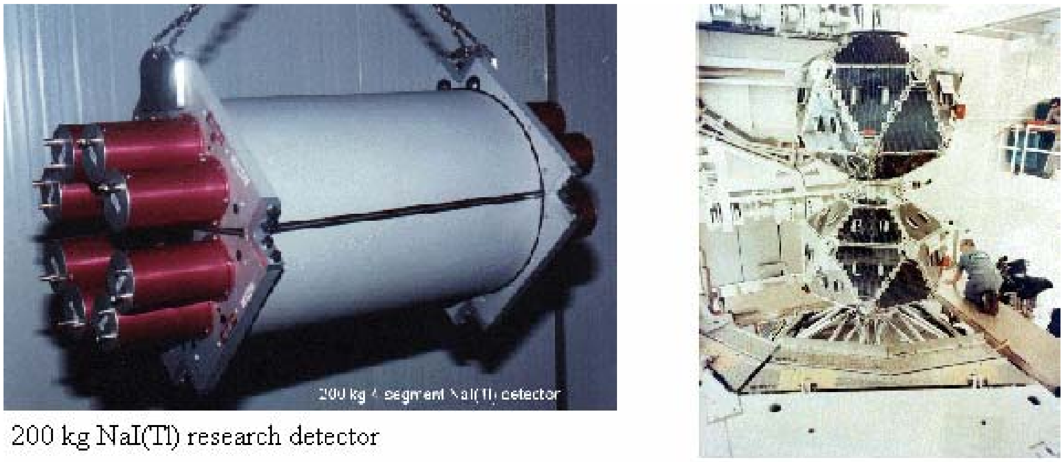


Gate monitor

Perimeter monitor

Crane monitor

Figure 21: Industrial gamma scintillation detectors. (From EXPLORANIUM)



200 kg NaI(Tl) research detector

Vela 5 A satellite uses NaI(Tl) detectors

Figure 22: Scintillation detectors used in research.

4.4.2. Alpha Radiation Scintillation Detectors

Thus far, we have focussed on scintillation detectors used for the measurement of X-ray and gamma radiation. Scintillation detectors can also be used to measure alpha radiation.

An alpha scintillation detector is designed to measure only alpha particles in the presence of gamma and beta radiation. Unlike gamma detectors which typically use NaI(Tl) phosphors, alpha scintillation detectors use a thin phosphor such as zinc sulphide (ZnS).

Because alpha radiation is much more ionizing than beta or gamma radiation, it deposits more energy in the scintillator which results in a higher output pulse. With the proper adjustment of instrument discriminators, beta and gamma signals can be filtered out leaving only the alpha signal. This is similar to a gas filled detector operated in the proportional region.

Alpha scintillation detectors are used most often with survey instruments to measure the contamination from alpha emitting radionuclides. Examples of alpha radiation scintillation detectors are shown in Figure 23.

Figure 23: Alpha scintillation detectors. (From Bicron and Ludlum)

4.4.3. Liquid Scintillation Counter

Some radionuclides such as tritium (^{3}H) and carbon (^{14}C) emit only low energy beta particles. Other radionuclides such as ^{125}I emit only low energy gamma radiation.

Low energy radiation of this type is difficult to measure with solid scintillation detectors because the radiation is usually totally stopped or severely attenuated by the detector window material. To solve this problem, the sample being analysed is mixed directly with a liquid scintillator.

Liquid scintillators consist of a cocktail of an organic solvent (see Table 2), one or more fluorescent solutes and an emulsifier to ensure proper mixing of samples. Because the sample to be measured is mixed with the liquid scintillation cocktail, the detection efficiency of liquid scintillation counters is very high.

Liquid scintillation counters (LSC) are typically non-portable laboratory style instruments designed to measure large volumes of samples automatically. A schematic of a liquid scintillation counter is shown in Figure 24.

During measurement, samples are positioned one-at-a-time between dual PMT's and counted for a preset period. Only when coincident (simultaneous) pulses are registered by each PMT, is a radiation event counted. By only counting coincident pulses, background radiation for a liquid scintillation counter is virtually eliminated.

Because of their design, liquid scintillation counters generally have a greater sensitivity and lower detection limits than solid scintillation detectors. However, there is significant sample preparation required with liquid scintillation counters. There is no sample preparation required when using solid scintillation detectors.

Liquid scintillation counters are often used for measuring the activity of biological samples (e.g. urine analysis) to assess the amount of beta emitting radionuclides ingested by the subject (bioassay). Liquid scintillation counters are also used to analyse leak and swipe test samples from nuclear instruments and gauges, and environmental samples.

Some examples of liquid scintillation counters are shown in Figure 25.

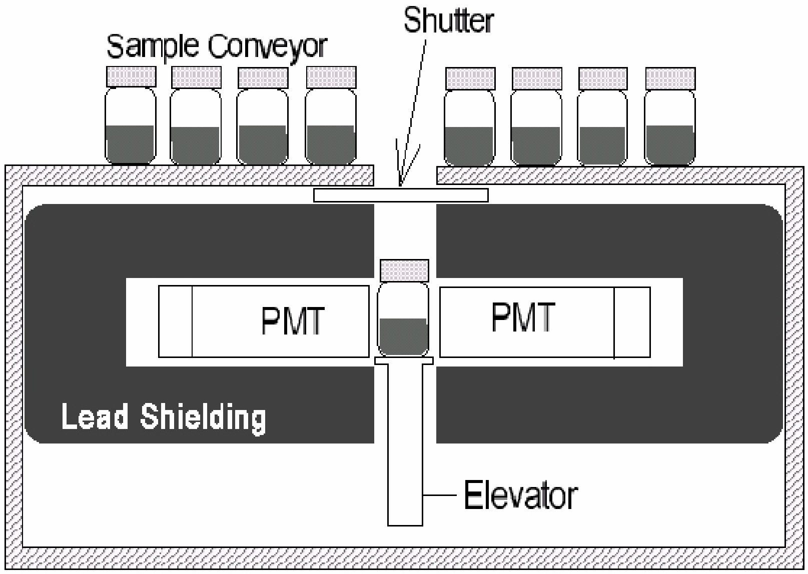


Figure 24: Schematic of a liquid scintillation detector. (Source: Canadian Nuclear Safety Commission, *Radiation Safety Officers Handbook Info-0718*. Reproduced with the permission of the Minister of Public Works and Government Services, 2001)

Laboratory liquid scintillation counter

Portable liquid scintillation counter

Figure 25: Liquid scintillation counters.

5. Neutron Measurement

5.1. Proportional Counters

Neutrons, like gamma rays, are not directly ionizing; they must react with another medium to produce a primary ionizing particle. Consequently, neutrons cannot be measured directly. Neutron detectors must rely upon a conversion process where an incident neutron interacts with a nucleus to produce a secondary charged particle. These charged particles are then directly detected and from them the presence of neutrons is deduced.

One of the most common reactions used in neutron detection is:

$$n + {}^{3}\text{He} \rightarrow p\ {}^{3}\text{H} + 765\ \text{keV}$$

where both the proton and ^{3}H are detected by a gas filled proportional counter using ^{3}He fill gas.

Another common reaction is:

$$\begin{aligned} n + {}^{10}_{5}\text{B} &\rightarrow \alpha + {}^{7}_{3}\text{Li} + 2.31\,\text{MeV}\ (94\%) \\ n + {}^{10}_{5}\text{B} &\rightarrow \alpha + {}^{7}_{3}\text{Li} + 2.79\,\text{MeV}\ (6\%) \end{aligned}$$

In this reaction, the alpha particle that is produced is detected in a proportional counter filled with boron trifluoride (BF_3) or containing thin film of BF_3 on the inside surfaces of a detector tube or sphere.

Examples of ^{3}He and BF_3 detectors are shown in Figure 26. Neutron detectors may be stand alone instruments but are often probes which are attached to multi-purpose survey instruments (like gas filled detectors and scintillation detectors).

Figure 26: ^{3}He and BF_3 neutron detectors. (From Eberline)

Gas proportional detectors such as these are efficient only for thermal neutrons (0.025 eV). For higher energy neutrons, the probability of being measured (referred to as capture cross sections) is very small, making it very unlikely that a neutron will interact with the gas and cause the necessary detection reaction. Because of this, it is necessary to slow the neutrons down to a level where the probability of interaction is statistically significant.

This is usually accomplished by surrounding the detector and/or the sample being counted by a hydrogen-rich material (moderator) such as high density polyethylene (paraffin is also used). Typically, 10 cm of polyethylene surrounds the detectors. These types of neutron detectors can measure neutrons in the range of 0.025 eV to 10 MeV.

It is interesting to note that there really is no neutron energy spectrum. Because all of the neutrons which are detected have been moderated to reduce their energy to a thermal level, all neutron energy information is lost. All events of interest fall into one peak, which is the reaction energy (765 keV for ^{3}He and 2.31 MeV for BF_3).

5.2. Bubble Detectors

Neutrons are also measured using bubble detectors. The bubble detector consists of an elastic polymer throughout which droplets of superheated liquid have been dispersed. When these droplets are struck by neutrons they form small gas bubbles that remain fixed in the polymer. The number of bubbles is directly proportional to the neutron equivalent dose.

Although bubble detectors have been used primarily in dosimetry, they are also available in survey meters (see Figure 27).

Figure 27: Neutron bubble detectors. (From Bubble Technology)

5.3. Fission Counter

Slow-neutron-induced fission of ^{233}U, ^{235}U or ^{239}Pu is utilized in fission counters. The fission pulses are extremely large, enabling slow neutron counting to be done at low levels, even in a high background. Most commonly, the fissile material is coated on the inner surface of an ionization chamber. A disadvantage of fissionable materials is that they are alpha emitters, and one must sometimes contend with the pileup of alpha particle pulses.

5.4. Activation Foils

Slow neutrons captured by nuclei induce radioactivity in a number of elements, which can be made into foils for neutron detection. The amount of induced activity will depend on a number of factors including the element chosen, the mass of the foil, the neutron energy spectrum, the capture cross section, and the time of irradiation. Examples of thermal neutron activation foil materials include Mn, Co, Cu, Ag, In, Dy and Au.

6. Practical Considerations

6.1. Introduction

Thus far, we have discussed the different types of gas filled and scintillation detectors and their principles of operation.

However, to gain a proper appreciation of these types of detectors, one must understand some of the practical considerations associated with the use of these instruments.

Some of the primary points to consider include detector resolving time, detector efficiency, background radiation and counting statistics.

To ensure the proper use of radiation measuring instruments, there are also a number of practical considerations including proper instrument selection and appropriate instrument checks and calibration.

Many of the principles to be discussed in this section apply to all types of radiation detectors.

6.2. Resolving Time

6.2.1. Gas Filled Detectors

If two particles enter a gas filled counter in rapid succession, the avalanche of ions from the first particle paralyzes the counter, and renders it incapable of responding to the second particle. Because the electric-field intensity is greatest near the surface of the anode, the avalanche of ionization starts very close to the anode and spreads longitudinally along the anode. The negative ions thus formed migrate toward the anode, while the positive ions move toward the cathode. The negative ions, being electrons, move very rapidly and are soon collected, while the massive positive ions are slower-moving and therefore travel for a relatively long period of time before being collected.

These slow-moving positive ions form a sheath around the positively charged anode, thereby greatly decreasing the electric field intensity around the anode and making it impossible to initiate an avalanche by another ionizing particle. As the positive ion sheath moves toward the cathode, the electric field intensity increases, until a point is reached when another avalanche could be started. The time required to attain this electric field intensity is called the *dead time*. After the end of the dead time, however, when another avalanche can be started, the output pulse from this avalanche is still relatively small, since the electric field intensity is still not great enough to produce a voltage pulse. As the positive ions continue their outward movement, an output pulse resulting from another ionizing particle would increase in size. When the output pulse is large enough to be passed by the discriminator and be counted, the counter is said to have recovered from the previous ionization, and the time interval between the dead time and the recovery time is called the *recovery time*. The sum of the dead time and the recovery time is called the *resolving time*. Alternatively, the resolving time may be defined as the minimum time that must elapse after the detection of an ionizing particle before a second particle can be detected. The relationship between the dead time, recovery time and resolving time is illustrated in Figure 28.

Despite the above definition of *dead time* and *resolving time*, these terms are often used synonymously, particularly by instrument manufacturers.

The resolving time of a Geiger counter is of the order of 100 μs or more. The resolving time of a proportional counter is around 10 µs. Since the avalanche in a proportional counter is limited to a short length of the anode, a second avalanche can be started elsewhere along the anode while the region of the first avalanche is completely paralyzed.

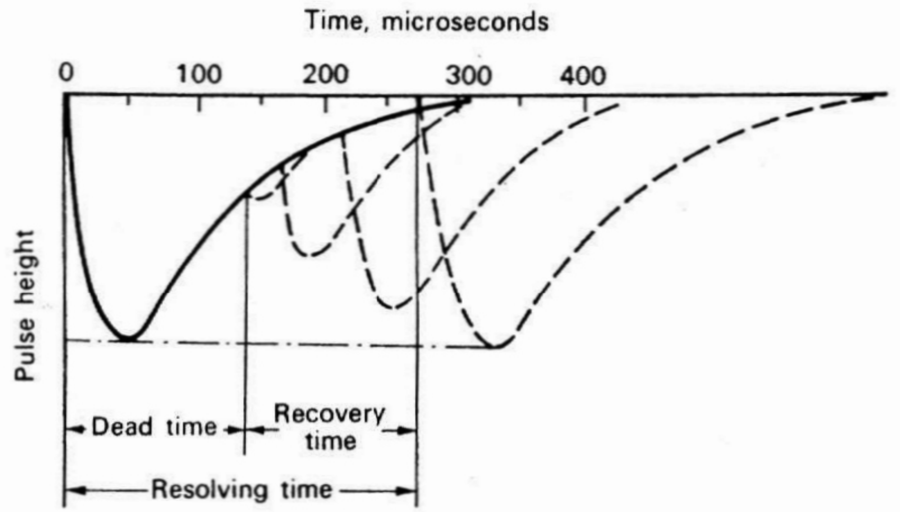


Figure 28: Relationship between dead time, recovery time, and resolving time. (Source: Cember, Herman, *Introduction to Health Physics*, Third Edition, McGraw-Hill Health Professions Division, New York, 1996. Fig 9.7)

Example:

If the dead time of a Geiger counter is 100 µs, and the counter is indicating 3000 counts per second (cps), calculate the total dead time and the actual number of counts.

Each time the Geiger counter registers a count, there is a dead time of 100 µs (100×10^{-6} s). Therefore, if the count rate is 3000 cps, the total dead time is:

$$\text{Total Dead Time} = 3000\ 1 \times 10^{-6}\ \text{s} = 0.3\ \text{s}$$

Accounting for the dead time, the 3000 cps were actually measured over a period of 0.7 s not each second as indicated by the Geiger counter.

Consequently, the actual number of counts is:

$$\text{Actual Count Rate} = \frac{3000\ \text{counts}}{0.7\ \text{s}} = 4286\ \text{cps}$$

When the Geiger counter is indicating 3000 cps, the actual count rate is 4268 cps.

Question:

What is the total dead time and actual count rate if 4000 cps is registered on a proportional counter with a dead time of 10 µs?

Answer:

Total dead time = 0.04 s Actual count rate = 4167 cps

Question:

What is the implication of dead time effects?

Answer:

In high intensity radiation fields, gas filled detectors will read low.

Fortunately, the scale on most modern survey instruments is designed to correct for dead time. Therefore, it is not normally necessary to correct for dead time when taking measurements.

It should be noted that a very intense radiation field can, in effect, "paralyse" some detectors resulting in a zero reading. In such a case, the detector is dead all of the time. This situation can be potentially hazardous if it causes a detector to read zero in a high intensity radiation field.

Most modern instruments have been designed so that if the radiation field exceeds the full-scale reading (maximum scale reading), it will continue to read off-scale until the radiation field is reduced.

One of the cardinal rules of radiation measurement is,

If the reading is off-scale, BELIEVE IT!

It is common mistake when a detector reads off-scale to conclude that there must be something wrong with the detector. However, until the reading is verified, it should be considered accurate and appropriate radiation protection steps taken.

Question:

Which is more appropriate for a high intensity radiation field; a small volume Geiger tube or a large volume Geiger tube?

Answer:

The larger the Geiger tube, the more radiation it receives. Consequently, larger Geiger tubes are more sensitive to radiation. In an intense radiation field, a large Geiger tube is more likely to read off-scale. Smaller Geiger tubes are less sensitive and are more adapted for the measurement of high levels of radiation.

It is interesting to note that some survey instruments are equipped with two Geiger tubes, a small tube for high intensity radiation fields and a large tube for low intensity radiation fields. Survey instruments of this type can measure radiation levels over very wide range of field intensities.

Most modern electronic survey instruments are “auto-ranging.” As the radiation intensity changes, the scale of the readout will automatically change. Less sophisticated survey instruments still require the user to change the scale reading manually when the instrument reads off-scale.

6.2.2. Scintillation Detectors

The resolving time, or dead time, of scintillation detectors is generally less than 10 μs. In scintillation detectors, the resolving time is more a function of the PMT, preamplifier and linear amplifiers and discriminator than the scintillator. In a scintillation detector, the preamplifier converts the current pulse to a voltage pulse and consequently will also have a resolving time, as will the linear amplifier (see Figure 16). With the appropriate selection of electronics, resolving times can be reduced to several ns (nanoseconds).

6.3. Detector Efficiency

6.3.1. Introduction

The *efficiency* of a detector is a measure of how well it detects radiation.

By definition, the *efficiency* is the percentage of the total number of radiation disintegrations occurring in a source that are actually measured by the detector.

To understand the concept of efficiency, one has to ask two questions:

- Does the radiation intercept the detector?
- Is the radiation intercepting the detector measured?

The percentage of the total number of radiation disintegrations occurring in a source that intercept the detector is referred to as the *geometric efficiency*. The geometric efficiency is governed solely by the source-detector geometry and will change as source-detector geometry changes.

Larger detectors will intercept more radiation than small detectors (see Figure 29). Further, the closer the detector is to the radiation source, the more radiation that will strike the detector.

The percentage of the total number of radiation particles intercepting the detector that are actually registered is referred to as the *intrinsic efficiency*. The intrinsic efficiency of a detector depends on the detector design and construction and the type of radiation being measured.

Mathematically, the efficiency, ε of a detector is the product of the geometric efficiency $\varepsilon_{\text{geometric}}$ and the intrinsic efficiency, $\varepsilon_{\text{intrinsic}}$.

$$\varepsilon = \varepsilon_{\text{geometric}}\,\varepsilon_{\text{intrinsic}}$$

The maximum efficiency theoretically possible is 100%, although for most detectors, it is much lower.

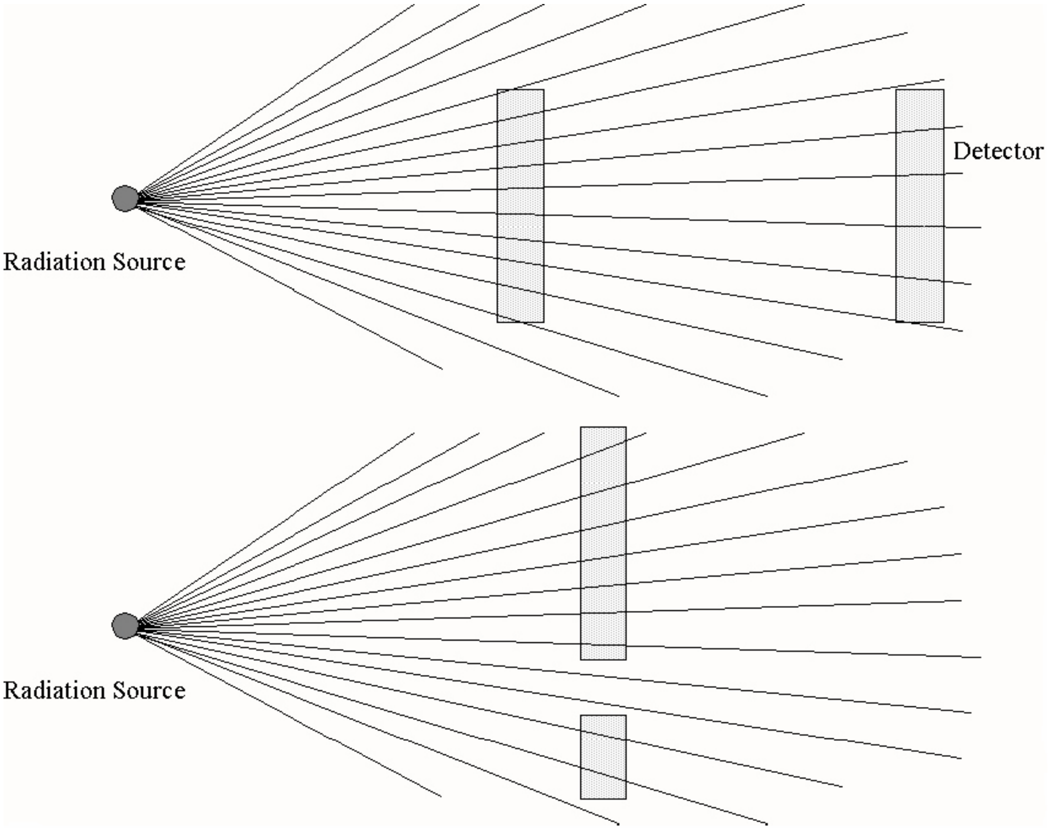


Figure 29: Factors affecting detector efficiency. The blocks represent the window of the detector. In the top picture, it is evident that if the detector is further away from the source, it will intercept less radiation. The bottom picture shows that the larger the detector window, the more radiation it will intercept.

6.3.2. Source-Detector Geometry

The source-detector geometry is affected by the following factors:

- detector size
- geometric relationship between the radiation source and the detector
- presence of material between source and detector which may absorb or scatter radiation

In a radiation field, a large detector will intercept more radiation than a small detector (see Figure 29). This is an important consideration as detector size affects the sensitivity of the detector and its efficiency. Detector size also affects the time required to collect statistically significant radiation measurements. Counting statistics will be discussed later in this training module.

Charged particles, such as alpha and beta particles, are readily absorbed or scattered in air. Therefore, their accurate measure requires that the detector be close to, or in contact with the radiation source.

X-rays and gamma rays are affected little by air and can be easily detected at some distance from the source. Intuitively, we know that the further the detector is away from the source the less radiation that will actually strike the detector and be registered (see Figure 29).

This can be better understood by considering the *Inverse Square Law*, which states that the intensity of the radiation varies as the inverse of the square of the distance from the source.

Quantitatively, the *Inverse Square Law* is described by the expression:

$$\mathrm{I} = \frac{\mathrm{S}}{4\pi\pi^2}$$

The term S refers to the radiation intensity at the source and I, to the radiation intensity at a distance r from the source. Intensity is a measure of the number of photons (in this case) passing through a unit area each second ($s^{-1}cm^{-2}$). You may have recognized that the term $4\pi r^2$ is just the equation for the surface area of a sphere of radius r centred about the radiation source (see Figure 30).

To better understand the meaning of the *Inverse Square Law*, consider the following example.

Example:

A Geiger counter positioned 10 cm from an unshielded gamma source indicates a reading of 1000 counts per minute (cpm). What is the Geiger counter reading at 20 cm and at 30 cm?

Writing the *Inverse Square Law* equation for each distance from the source, we obtain the expressions

$$I_{10\,\text{cm}} = \frac{S}{4\pi(10)^2}$$

$$I_{20\,\text{cm}} = \frac{S}{4\pi(20)^2}$$

If we take the ratio of the intensity at 20 cm and the intensity at 10 cm we obtain the result

$$\frac{I_{20\,\text{cm}}}{I_{10\,\text{cm}}} = \frac{(10)^2}{(20)^2} = \frac{1}{4}$$

This means that the intensity at 20 cm from the radiation source is 1/4 the radiation intensity at 10 cm (see Figure 30). The Geiger counter reading at 20 cm would then be

$$I_{20\,\text{cm}} = I_{10\,\text{cm}} \times \frac{1}{4} = 1000\,\text{cpm} \times \frac{1}{4} = 250\,\text{cpm}$$

At a distance of 30 cm from the radiation source the intensity will be reduced by

$$\frac{I_{30\,\text{cm}}}{I_{10\,\text{cm}}} = \frac{(10)^2}{(30)^2} = \frac{1}{9}$$

The intensity at 30 cm would be

$$I_{30\,\text{cm}} = 1000\,\text{cpm} \times \frac{1}{9} = 111\,\text{cpm}$$

As illustrated, the gamma intensity is reduced dramatically as we move away from the radiation source.

A plot of radiation intensity as a function of distance from a photon radiation source is shown in Figure 31.

Question:

During routine service of a nuclear device its ^{60}Co source is exposed. A radiation survey was conducted using a Geiger counter survey instrument. At a position 10 cm from the source, the survey instrument registered a reading of 10 000 cpm.

To control access, the area around the nuclear gauge must be roped off until the service work is completed. How far away from the source should the ropes be placed to reduce the survey instrument readout (and your radiation exposure rate!) by a factor of 1000 (i.e. to a reading of 10 cpm)?

Answer: 3.16 m

It should be noted that there are a couple of caveats to the *Inverse Square Law*.

1. The *Inverse Square Law* is only valid for gamma rays
2. The *Inverse Square Law* is only valid when the radiation source size (diameter) is much smaller than the distance r between the radiation source and the detector (d<<r). In other words, when the radiation source is a "point source."
3. When shielding is used between the source and the detector, the *Inverse Square Law* can only be applied if due allowance is made for the gamma photons absorbed in the shielding
4. If the gamma radiation is in the form of a beam, the *Inverse Square Law* is invalid.

Although the radiation intensity will always decrease with increasing distance form the radiation source, depending on the size of the source, the intensity drop may not have a $1/r^2$ dependence. In confined spaces, it is not always possible to meet the criteria d<<r. This is particularly true in nuclear facilities such as a uranium mine, mill or a nuclear power plant where the radiation sources are potentially very large and in confined areas. In such cases where the criteria d<<r is not met, the radiation intensity is more likely to drop linearly (1/r dependence).

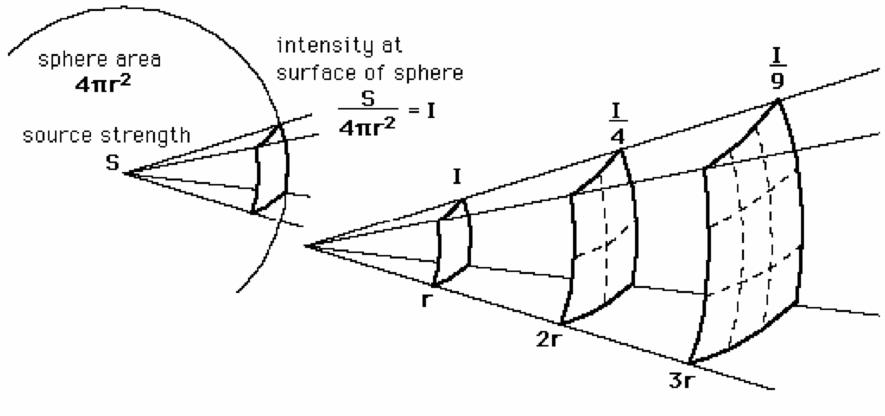


Figure 30: Inverse Square Law.

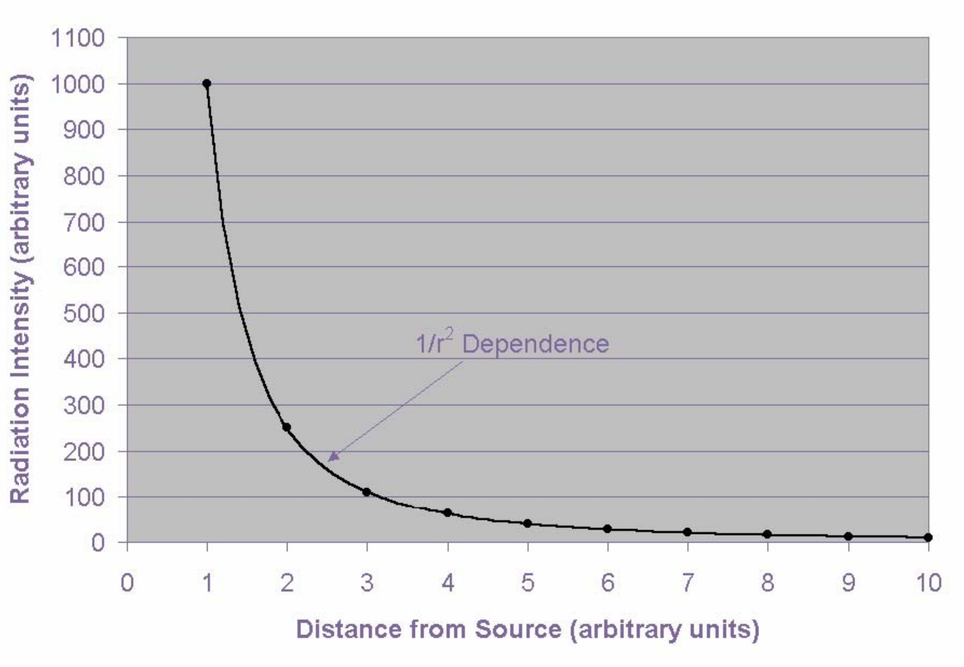


Figure 31: Plot of the Inverse Square Law.

6.3.3. Detector Construction

It is true to say that any charged particle that enters the active volume of a detector will be detected. In other words, the intrinsic efficiency for alpha and beta radiation is approximately 100% for most detectors. The difficulty in detection lies in producing a window on the detector thin enough for the alpha and beta radiation to penetrate without being absorbed.

Detectors designed to measure alpha and beta radiation have thin Mylar or mica windows. Window thicknesses typically range from approximately 2 μm to 10 μm (1 μm = 1×10^{-6} m). Instrument manufacturers normally provide window thicknesses in terms of their thickness density which range from 0.3 - 3 mg/cm^2. The thickness of the window is calculated by dividing the thickness density (mg/cm^2) by the window material density (Mylar: 1.4 g/cm^3 and mica: 2.8 g/cm^3). Detector windows are often delicate, therefore care must be taken not to damage or puncture the window. For increased protection, many survey instrument alpha/beta/gamma probes are equipped with metal screens in front of the detector window.

Unlike charged particles, X-rays and gamma rays do not require thin windows to enter the detector and be detected. However, the intrinsic detection efficiency for X-ray and gamma radiation is much less than for charged particles. In fact, a Geiger counter will normally count only 1/100 photons striking it. Why?

Gamma rays produce ionization indirectly through collisions with the detector atoms. It is possible for a photon to pass through a detector and not interact at all. This is particularly true for gas filled detectors which normally have an intrinsic detection efficiency of the order of 1% (gases are less dense than solids). The probability of photon interaction with the detector increases with atomic number.

How then are photons measured if they are likely to pass through a detector unnoticed? Gamma rays normally interact with the detector walls knocking an electron into the detector gas-filled region causing a pulse to be generated (see Figure 32).

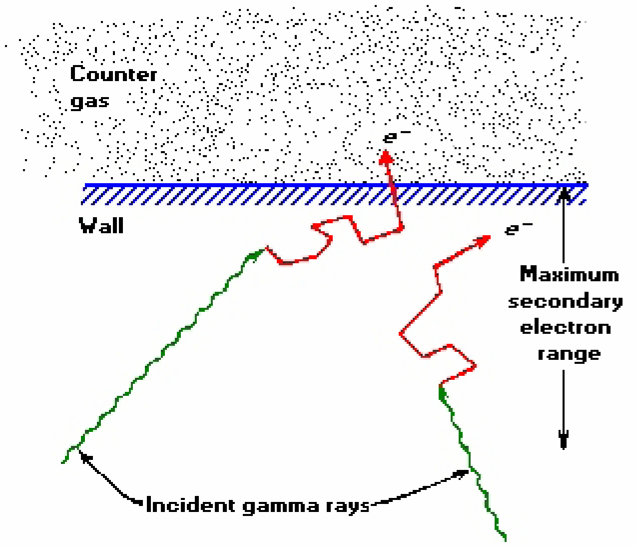


Figure 32: Photon detection in a gas filled detector.

The probability of interactions in the walls is increased by using cathodes of high atomic number (e.g. Al (Z=13) and Bi (Z=83)).

Scintillation detectors and semiconductor detectors are much more efficient at measuring X-ray and gamma radiation than gas filled detectors.

6.3.4. Determination of Detector Efficiency

Thus far we have learned that the efficiency of a detector depends on the source-detector geometry, the detector size and construction, and the type of radiation being measured.

As the detector size and construction are fixed characteristics, one soon realizes that the efficiency of a detector can and will vary as the source-detector geometry is changed and as the radiation type being measured is changed.

Fortunately, for most survey instruments the intrinsic efficiency is accounted for during calibration. In radiation surveys, it is generally not necessary to determine the efficiency of the detector.

In general, it is only necessary to determine the efficiency of laboratory style or portable instruments used to measure the activity present on a radioactive sample.

To measure the efficiency of a detector for a specific type of radiation, the following procedure should be followed.

1. Obtain a radiation source of known activity. The activity will be given in either in Bq, Ci or sometimes in dpm (disintegrations per minute).

   Check sources are commercially available in all types, activities and accuracies (±5% is typical). If you are measuring alpha radiation, the check source should be an alpha source. If you are measuring beta radiation, the check source should be a beta source. The check source should also have the same geometric characteristics as the radiation source you intend to measure (i.e. diameter).

2. Place the source in the measurement position and measure the number of counts over a predefined period such as 1 minute (cpm) (see Figure 33).

3. Calculate the efficiency, ε of the detector using the equation

$$\varepsilon = \frac{\text{measured counts per minute (cpm)}}{\text{expected counts per minute (dpm)}} \times 100\%$$

The efficiency can also be calculated using units such as counts per second (cps) and disintegrations per second (dps). It is only important the units be consistent (do not mix cps with dpm or vice-versa).

For a measured efficiency to be valid the sample activity must be measured under the same geometric conditions (source-detector distance) as the check source used to determine the efficiency.

The efficiency is applied to measured count rates to calculate activity using the expression

$$A = \frac{\text{cps}}{\varepsilon}$$

Detector efficiencies are often quoted in units of percent. However, the fractional form must be used when applied to actual calculations.

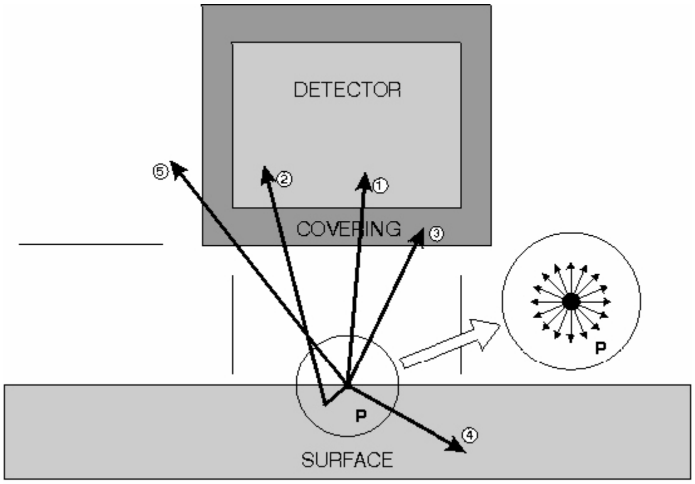


Figure 33: Detector efficiency. It should be noted that not all the radiation emitted by a source is intercepted by the detector.

Example:

A 1 kBq ^{241}Am alpha check source is used to measure the efficiency of a ZnS detector. During a 3 minute count period, the detector registered 45 000 counts. What is the efficiency of the ZnS detector for alpha particles?

Using the efficiency expression

$$\varepsilon = \frac{\left(\dfrac{45\,000\ \text{counts}}{3\ \text{min} \times 60\ \text{sec/min}}\right)}{1000\ \text{cps}} \times 100\% = 25\%$$

Is this efficiency valid if you were to measure beta radiation or gamma radiation?

Question:

A sample was measured for alpha radiation for 20 minutes using the above mentioned ZnS detector. A total count of 500 was recorded during the 20 minute count period. Ignoring the contribution of background radiation, calculate the activity of the sample in Bq.

Answer:

Using the equation for activity

$$\text{Activity} = \frac{\left(\dfrac{500\ \text{counts}}{20\ \text{min}}\right)}{0.25} \frac{1\ \text{min}}{60\ \text{sec}} = 1.7\ \text{Bq}$$

The alpha activity present in the sample is 1.7 Bq.

6.4. Background Radiation

Cosmic radiation, natural radioactivity in the earth (terrestrial radiation) and building materials, as well as sources of unnatural radiation all constitute unwanted *background radiation* which can interfere with radiation measurements.

The two common methods of dealing with background radiation are to:

- shield the detector
- measure the background separately with the radiation source removed and subtract it from the results of the measurement

It should be noted that even with detector shielding, background radiation may be reduced but will never be eliminated.

When measuring radiation, whether in the field or in the lab, background measurements should **always** be taken regardless of the type of instrument being used. In survey instruments, the background may be measured in cps or cpm or in terms of exposure rates such as µR/h or µSv/h.

<u>Question</u>:

What would be the activity in the previous question if the measured background were 15 cpm?

<u>Answer</u>:

Using the equation for activity,

$$\text{Activity} = \frac{\left(\frac{500\ \text{counts}}{20\ \text{min}}\right) - 15\ \text{cpm}}{0.25} \times \frac{1\ \text{min}}{60\ \text{sec}} = 0.7\ \text{Bq}$$

Accounting for background radiation, the alpha activity of the sample is actually 0.7 Bq not 1.7 Bq.

When measuring low activity samples, the contribution from background radiation can be significant and must be accounted for. The higher the activity the less the effect of background radiation.

All other things considered, it is the uncertainty in background radiation which governs the detection limits of most radiation measuring instruments.

Question:

In the above question, what would be the activity if the measured background were 27 cpm?

Answer:

-0.1 Bq What does this mean?

6.5. Instrument Selection

As you can now appreciate, there are a host of different types of survey instruments available to measure radiation many of which can be used to measure the same types of radiation.

For different radiation types and applications, however, some instruments are preferred. As a guide, instrument selection by radiation type and energy is presented in Table 3.

Table 3: Instrument selection by radiation type.

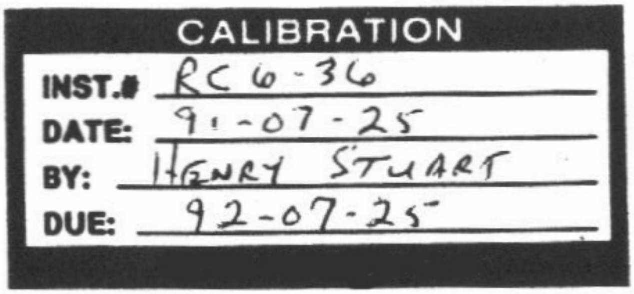
Radiation Type	Portable Survey Instrument	Laboratory Instrument
Alpha radiation (all energies)	Ionization chamber Pancake proportional counter End window or pancake GM detector ZnS scintillation detector NaI(Tl) or plastic scintillation detector	Proportional counter NaI(Tl) scintillation counter ZnS scintillation counter Semiconductor detector
Low energy beta radiation ^{3}H: 18 keV ^{63}Ni: 65 keV	Not recommended	Liquid scintillation counter
Medium energy beta radiation ^{14}C: 157 keV ^{35}S: 167 keV ^{45}Ca: 257 keV	Ionization chamber Proportional chamber End or side window or pancake GM detector	Liquid scintillation counter Proportional detector Semiconductor detector
High energy beta radiation ^{32}P: 1.7 MeV	Ionization chamber End or side window or pancake GM detector	Liquid scintillation counter NaI(Tl) scintillation counter Proportional counter
Low energy gamma and X-rays ^{125}I: 27-30 keV	NaI(Tl) or plastic scintillation detector (thin)	Liquid scintillation counter Semiconductor detector
Medium - High energy gamma rays ^{131}I: 0.30 MeV ^{137}Cs: 0.66 MeV ^{60}Co: 1.2 - 1.3 MeV	NaI(Tl) scintillation detector (thick) Ionization chamber Energy compensated GM detector	NaI(Tl) scintillation counter Semiconductor detector

6.6. Instrument Checks

It is important that radiation instruments be checked periodically to ensure they are operating correctly. These checks consist of pre-operational checks and calibration checks.

Pre-operational checks are those that you should do every time before the instrument is used. These checks will depend, in some part, on the particular instrument but typically include the following:

- Inspection: check the physical condition to verify that the instrument is in good working order (cable and connectors, detector window, etc.)
- Battery check: used to check that the battery power supply is adequate
- Audible response: check to ensure that the survey instrument's audible response is working
- Calibration check: check that the due date on the calibration sticker is not exceeded (see Figure 34)
- Source check: used to check that the instrument correctly indicates actual radiation fields - many survey instruments are equipped with check sources



CALIBRATION

INST.# RC 6-36
DATE: 91-07-25
BY: HENRY STUART
DUE: 92-07-25

Figure 34: Calibration certificate. (From J. U. Burnham)

If an instrument fails any of the pre-operational checks, it should be removed from service and sent for repair.

Calibration is a sophisticated operational test done routinely for each instrument. The instrument is checked for accuracy across its entire range and not just at one point of the range as is the case for a source check.

Under the Canadian Nuclear Safety Commission (CNSC) *Nuclear Substances and Radiation Devices Regulations* radiation survey instruments may not be used unless they have been calibrated within the last 12 months. The CNSC further requires that survey instruments be calibrated by a licensed calibration facility in accordance with the CNSC Regulatory Document R-117, *Requirements for Gamma Radiation Survey Meter Calibration.*

Non-portable or laboratory style radiation detectors should be routinely serviced in accordance with the manufacturer's instructions.

Any radiation detector suspected of malfunction should not be used.

6.7. Counting Statistics

Radioactive decay is a random process; one never knows exactly when a particular radioisotope will decay. We can only deal with the average behaviour of billions of such unstable nuclei. As a result, the counts per minute from a radiation detector are subject to statistical variation.

When a radiation source (with sufficiently long half-life) is placed near a detector and 10 one- minute counts performed, one might get the following numbers: 99, 97, 102, 80, 104, 111, 98, 116, 89, 107 cpm. The mean count rate is 100 cpm, but any single count might yield a quite different result. The more counts we do, or the longer the time we count, the more likely it is that our cpm result will be close to the "true" result.

Radioactive decay actually follows what is known as the *Poisson distribution*, (see Figure 35) a statistical theory which applies to situations where the probability of an event is low. According to the Poisson distribution, the probability p(n) of the occurrence of exactly n events per unit time, if the true average rate $\overline{n}$ is given by the expression:

$$p(n) = \frac{(\overline{n})^n \times e^{-\overline{n}}}{n!}$$

where n! can be evaluated using Sterling's approximation:

$$n! = (2\pi n)^{1/2} \times \left(\frac{n}{e}\right)^n$$

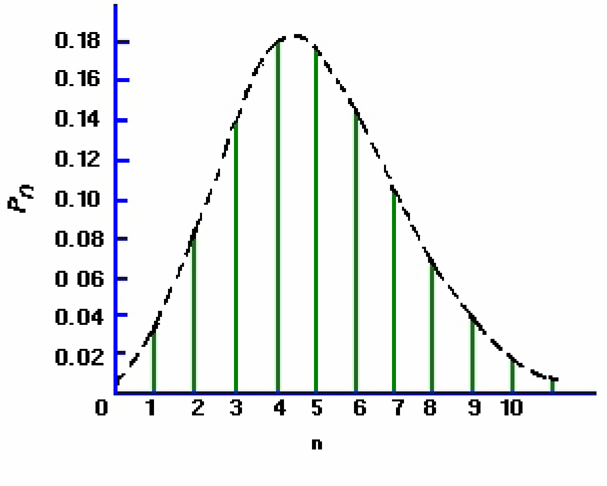


Figure 35: Poisson distribution.

<u>Example</u>:

If we have 37 Bq of activity for a source, the mean number of radioactive decays per second is 37. The probability of observing exactly 37 decays in 1 s is calculated as:

$$p(37) = \frac{37^{37} \times e^{-37}}{37!} = 0.066$$

If the probability of an event occurring is 100%, p=1. The probability of an event not occurring is simply then 1-p.

In the Poisson distribution, the mean is equal to n, the measured number of counts. The *standard deviation,* σ:

$$\sigma = \sqrt{n}$$

where n is the total number of events that occurred or were measured during the time of observation, and is thus the average rate for that time interval. The square of the standard deviation, σ^2, is called the *variance* of the mean and is equal to n. The variance is the uncertainty in the measured counts.

<u>Example</u>:

If we observe 10 000 counts during a 10 minute counting interval, the standard deviation of the observations is:

$$\sigma = \sqrt{10\,000} = 100 \text{ counts per } 10 \text{ minutes}$$

This corresponds to a 1% uncertainty in the result.

It is interesting to note that for $n \geq 20$, the Poisson distribution is indistinguishable from the Normal distribution (bell curve). Under these conditions, all statistical tests that are valid for the Normal distribution such as the *t*-test, the chi-square test and the F-test may also be used for Poisson distributions. Although all of these tests are applicable to radioactivity measurements, a full discussion of them is beyond the scope of this training module. Details and applications of these tests may be found in the suggested readings in the Reference section at the end of this module.

When measuring radiation, the number of events per unit time, t is referred to as the rate, r and is given by the equation:

$$r = \frac{n}{t}$$

The standard deviation of the count rate is:

$$\sigma_r = \frac{\sqrt{n}}{t}$$

Since

$$\sigma_r = \frac{\sqrt{n}}{t} = \sqrt{\frac{n}{t} \cdot \frac{1}{t}} = \sqrt{\frac{r}{t}}$$

the count rate and standard deviation can be written as

$$r \pm \sigma_r = r \pm \sqrt{\frac{r}{t}}$$

the count rate and standard deviation can be written as

$$r = r_g - r_b = \frac{n_g}{t_s} - \frac{n_b}{t_b}$$

where r is the net count rate. The terms r_g and r_b refer to the gross count rate (total source count rate including background) and the background count rate (no source), respectively. Similarly, the terms n_g and n_b refer to the gross number of counts and the background number of counts during the sample acquisition time t_s and the background acquisition time t_b.

When two quantities such as the gross count rate and the background count rate are subtracted (or added), the variance (σ^2) of the sum is equal to the sum of the variance for the gross count rate and the background count rate. Consequently, the standard deviation in the net count rate, r is given by:

$$\sigma_r = \sqrt{\sigma_g{}^2 + \sigma_b{}^2} = \sqrt{\frac{r_g}{t_s} + \frac{r_b}{t_b}}$$

Example:

During a 5-minute sample count 510 counts were registered. During a 1 hour background measurement 2400 counts were measured. What are the net sample counting rate and the standard deviation of the net counting rate?

The net count rate, r is calculated as

$$r = \frac{n_g}{t_s} - \frac{n_b}{t_b} = \frac{510}{5} - \frac{2400}{60} = 102\ \text{cpm} - 40\ \text{cpm} = 62\ \text{cpm}$$

The standard deviation in the net count rate r is

$$\sigma_r = \sqrt{\frac{r_g}{t_g} + \frac{r_b}{t_b}} = \sqrt{\frac{102}{5} + \frac{40}{60}} = 4.6\ \text{cpm}$$

The answer is: 62 ± 5 cpm.

Numerically, precision of a measured result is given by the *coefficient of variation* (CV), which is defined as the ratio of the standard deviation to the mean. The CV is usually expressed in terms of a percent (%CV) as:

$$\%\text{CV} = \left(\frac{\sigma_r}{r}\right) \times 100\%$$

In the above example, the %CV= 7.4%.

Question:

How would you improve the precision of a measurement?

Answer:

You can improve the precision of a measurement by reducing the standard deviation. In radiation measurements this is accomplished be counting the sample for longer periods of time. In the above example, the %CV for a 5 minute count was 7.4%. Keeping the background count time the same, if we were to count the sample for 25 minutes the %CV would be 3.5%.

6.8. Detection Limits

The *minimum detectable activity* (MDA) is important in low-level counting, when the count rate of a sample is almost the same as the count rate of the background. The MDA is defined as the smallest quantity of radioactivity that could be measured under specific conditions. The MDA depends on the *lower limit of detection* (LLD) and on the counting efficiency of the counting system.

The *lower limit of detection* (LLD) is based on a consideration of statistical errors of 5% (Type I and Type II) and is defined as the minimum amount of radiation that can be detected by a counting system. The LLD is given by the equation

$$\mathrm{LLD} = \frac{2.71}{t_s} + 3.29\sqrt{\frac{r_b}{t_b} + \frac{r_b}{t_s}}$$

It should be noted that the background radiation levels should be measured under the same conditions as the actual sample, but with no activity present.

The MDA can be calculated by dividing the LLD by the detector efficiency ε

$$\mathrm{MDA} = \frac{\mathrm{LLD}}{\varepsilon} = \frac{\frac{2.71}{t_s} + 3.29\sqrt{\frac{r_b}{t_b} + \frac{r_b}{t_s}}}{\varepsilon}$$

Example:

A portable survey instrument is used to monitor the fixed phosphorus-32 (^{32}P) contamination in an active laboratory. The survey instrument is operated in count rate mode using a pancake GM probe. The efficiency of the GM probe is 25% for ^{32}P beta particles (1.71 MeV) and the measured background count rate, r_b is 50 cpm.

Assuming, the background was measured for 1 minute and the radiation survey was performed over 3 minutes, calculate the LLD for this survey instrument in this scenario. What is the MDA?

Using the LLD equation:

$$\mathrm{LLD} = \frac{2.71}{3} + 3.29\sqrt{\frac{50}{1} + \frac{50}{3}} = 28\,\mathrm{cpm}$$

The MDA is

$$\mathrm{MDA} = \frac{\mathrm{LLD}}{\varepsilon} = \frac{28}{0.25} = 112\,\mathrm{dpm}$$

7. Survey Instruments: Basic Controls and Functions

As already indicated, modern survey instruments are available in many styles and with a range of features. Some survey instruments come with built-in detectors and can only be used for their prescribed purpose. Other survey instruments such as ionization chambers proportional counters, GM detectors, and scintillation detectors are “multi-purpose” devices which can be used with a variety of different detectors (probes).

More sophisticated survey instruments have non-volatile memories allowing for automatic data logging and analysis (much like a mini computer). This is particularly useful when taking field measurements on a large scale as the instrument records measurements which can be retrieved at some later time, either directly from the instrument (display) or via connection to a computer. Some very sophisticated survey instruments incorporate bar code readers for site location identification and global positioning systems (GPS).

With the proper probes and calibration, a single multi-purpose survey meter can be used to measure almost all radiation types.

Because the range of survey instruments and features can vary dramatically and depend to a large extent on the manufacturer, it is beyond the scope of this training module to describe all possible functions and features of modern survey instruments. However, following, is a brief discussion of the controls and functions of a basic survey instrument as shown in Figure 36.

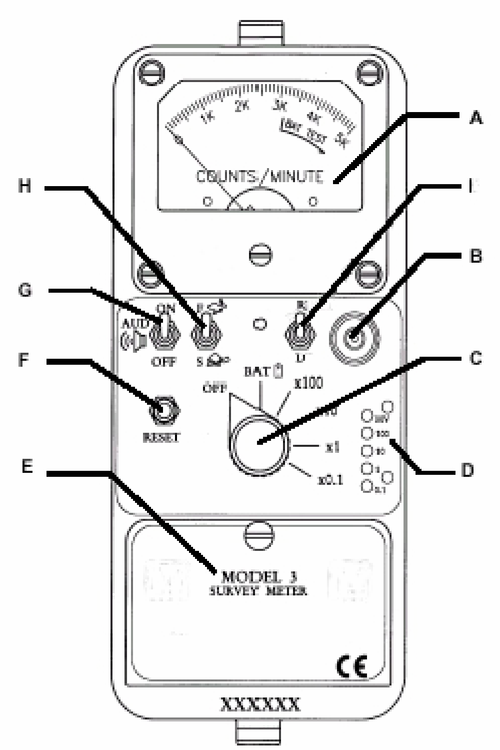


Figure 36: Basic survey instrument (from Ludlum). See the following pages for a description of the features.

A **Meter display**:

Typical scales are dual allowing for the measurement of count rate in cpm (or multiples of) and dose rate in mR/h, mSv/h (or multiples of).

B **Connector**:

Used to connect the probe to the instrument. Typical connections are BNC (*British Naval Connector or Bayonet Nut Connector or Bayonet Neill Concelman*), MHV (*Medium High Voltage*) or UHF (*Ultra-High Frequency*).

C **Instrument Selector Switch**:

A multi-position switch used to turn the instrument ON/OFF, test the battery (BAT) and adjust the meter scale the instrument will operate in (×0.1, ×1, ×10, ×100). Some instruments are auto-ranging and will adjust the scale automatically to the measured radiation intensity.

D Calibration Controls

Recessed potentiometers which are used to calibrate the individual range selections and allow for high voltage adjustment. On many instruments, calibration potentiometers are located inside to minimize user tampering.

Many modern survey instruments perform self-diagnostic tests when first turned on to confirm that the instrument is functional. If the instrument fails any of the self-diagnostic tests, an error message is displayed.

E Battery Compartment

Sealed compartment to house the batteries. In some survey instruments, the battery compartment is located on the back of the instrument.

F Reset Button

When depressed, the Reset Button returns the current meter reading to zero.

G Audio Button

This switch allows the operator to turn the instrument audio ON or OFF. When ON, the survey instrument will give a clicking sound each time a radiation event is detected.

H Response Time

This switch changes the response time of the instrument from fast to slow. In the F position there is a fast meter response and large meter deviation. In the S position there is a slow response and less meter deviation.

I Count Rate/Dose Rate

This switch changes the instrument from count rate mode to dose rate mode. In count rate mode, the survey instrument measures the number of events per unit time (cpm). In dose rate mode, the survey instrument measures dose rate in mR/h, μSv/h, etc. It should be noted that for this switch to be used, the probe must be calibrated for count rate and dose rate measurements.

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Radiation Safety Institute of Canada
Institut de radioprotection du Canada
National Office, Toronto

Training Course

Radiation Safety Officer
(RSO-1)

Radiation Safety Instrumentation: Dosimeters

Developed and Prepared by:

Brian Bjorndal, MSc, PPhys
Reza Moridi, PhD, FInstP, FIEE, CEng

Sixth Revision

Radiation Detection: Dosimetry

CONTENTS

1. Introduction

People who work with radiation sources or in the vicinity of radiation sources may routinely be exposed to radiation from these sources. The amount of radiation received is called the *radiation dose* and the measurement of such doses is known as *radiation dosimetry*. In SI units, radiation dose is measured in mSv or a multiple thereof.

Doses from radiation fall into two general categories: external radiation dose and internal radiation dose.

External radiation dose is the dose received from sources of radiation located outside the body. Internal radiation dose is the dose received when radioactive material is taken into the body via inhalation, ingestion or absorption through the skin. As you might have surmised, internal dosimetry is very complicated and usually involves *bioassay* measurements and/or the use of specialized dosimeters. This is beyond the scope of this training module and will not be discussed further. The reader is referred to the suggested readings in the Reference section at the end of this module.

External radiation doses are measured using a device called a *dosimeter*.

A dosimeter is a device which measures the total radiation dose over some time interval. Dosimeters are integrating devices; that is, they add up all the doses received by the person wearing the device to yield a cumulative dose result. This is fundamentally different from radiation survey instruments which measure the instantaneous radiation count rate or dose rate.

External radiation dosimeters can be used to measure the doses received from X-rays, gamma-rays, beta particles and neutrons.

Because dosimeters measure radiation dose, they are designed to be worn by the individual being monitored during work activities.

There are several types of external radiation dosimeters in use. Following is a list of the most common types.

- Thermoluminescent Dosimeter (TLD)
- Film Dosimeter
- Direct Reading Dosimeter (DRD)
- Electronic Dosimeter

In Canada, TLD’s are the most common dosimeter used to measure external radiation dose. Further, TLD’s are currently the only device certified by the Canadian Nuclear Safety Commission (CNSC) for external dosimetry.

It should be noted that the personal alpha dosimeter (PAD) is the only dosimeter approved by the CNSC for the measurement of worker exposure to radon and thoron progeny and long-lived radioactive dust.

2. Thermoluminescent Dosimeters

2.1. Principles of Operation

Many different crystals emit light if they are heated after having been exposed to radiation. This effect is called *thermoluminescence*, and dosimeters based on this effect are called *thermoluminescent dosimeters* (TLD). Some of the most common thermoluminescent crystals include:

- LiF
- CaF_2:Mn
- $CaSO_4$:Tm
- $Li_2B_4O_7$:Cu
- LiF:Mg
- Ti

The elements listed after the semicolons are called *activators*. Activators are added to the crystals to enhance thermoluminescence.

Absorption of energy from radiation excites the atoms in the crystal, which results in the production of free electrons and holes. These electrons are trapped at energy levels created by the activators or by imperfections in the crystalline lattice, thereby locking the excitation energy in the crystal (see Figure 1). Heating the crystal releases the excitation as visible light. This process is called thermoluminescence (“thermo” meaning heat and “luminescence” meaning light).

Thermoluminescent materials are manufactured in several formats including loose powder, disks, squares and rods (see Figure 2).

For personnel monitoring, one or more small pieces of thermoluminescent material (called elements) are placed into a small holder, called a badge, which is worn by the person being monitored. Depending on the design, some badges are also placed in hangers. The resulting device is called a thermoluminescent dosimeter (TLD).

Most TLD’s contain several crystals made of different materials. Each crystal may be differently filtered in order to take advantage of the differing energy dependence of the filters and to make the TLD as a whole approximately energy independent. One such TLD, for example contains four TLD crystals, two each of $Li_2B_4O_7$:Cu and two each of $CaSO_4$:Tm. With the appropriate filters, this dosimeter is approximately energy-independent from 30 keV to 10 MeV X-ray or gamma radiation, 0.5 MeV to 4 MeV beta radiation and 0.025 eV to 10 MeV neutrons. It can measure doses from ^{60}Co from 10 µSv to 10 Sv. With the proper filtration, multi-element TLD's are capable of distinguishing between X-rays and gamma rays, beta radiation and neutrons.

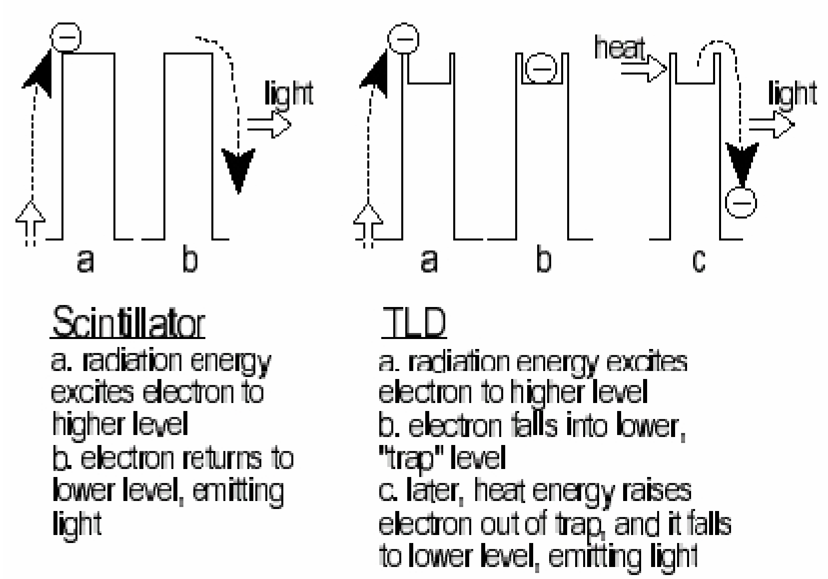

Figure 1: Electron traps. (Source: Canadian Nuclear Safety Commission, *Radiation Safety Officers Handbook Info-0718*. Reproduced with the permission of the Minister of Public Works and Government Services, 2001)

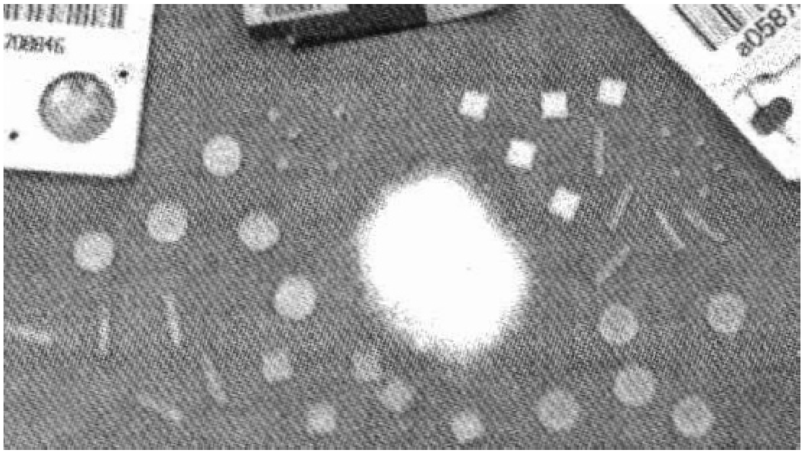

Figure 2: TLD crystal formats.

Thermoluminescent dosimeter crystals containing LiF and $Li_2B_4O_7$:Cu (LiBO for short) are widely used because LiF and LiBO crystals are very nearly tissue-equivalent, and almost energy independent from about 100 keV to 1.3 MeV gamma radiation (see Figure 3). Because the ^{6}Li isotope in natural lithium (natural lithium contains the isotopes ^{6}Li, and ^{7}Li) is sensitive to neutrons, LiF and LiBO can also be used for neutron monitoring.

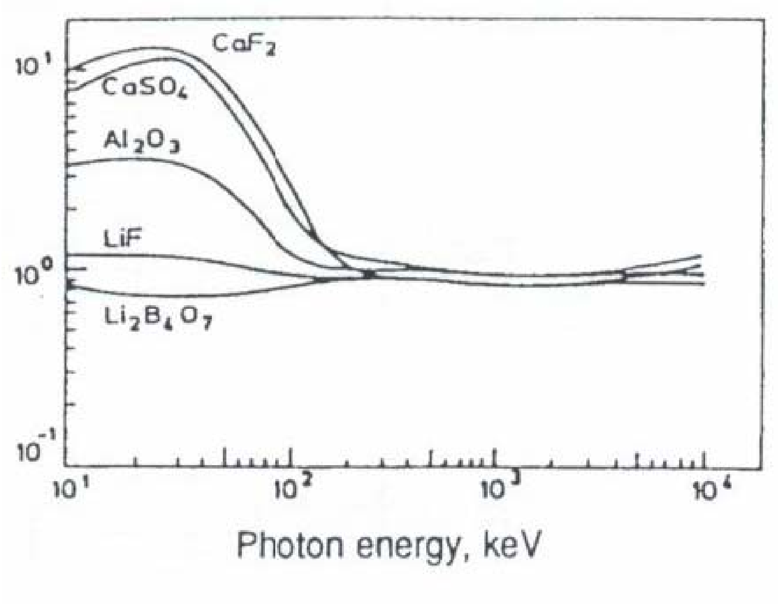

Figure 3: Energy response of several TLD crystals.

TLD's are available in two formats: whole body dosimeter and extremity dosimeter. Whole-body TLD's are badge-like and used to measure whole body *deep dose* and *shallow dose*. Deep dose is the radiation dose received at depth of 10 mm in the body and shallow dose is the radiation dose received at 0.07 mm (skin thickness). Extremity TLD's come in ring and bracelet format and are used to measure the shallow dose to extremities such as the hands and feet.

Because most beta particles will not penetrate deeper than the skin thickness, these particles deliver only shallow doses to the exposed person, while deep doses are delivered from exposure to X-ray, gamma and neutron radiation. If you have received only an X-ray or gamma radiation dose, the shallow dose and deep dose should be the same.

Each type of TLD crystal is characterized by what is called a *glow curve*. A glow curve is obtained by heating the irradiated crystal at a uniform rate and measuring the emitted light as the temperature increases. The temperature at which the maximum light output occurs is a measure of the binding energy of the electron or hole in the trap. More than one peak on a glow curve indicates different trapping levels, each with its own energy. The total light output is proportional to the number of trapped electrons, which in turn is proportional to the amount of energy absorbed from the incident radiation. The light output is directly proportional to the amount of thermoluminescent material. The characteristic glow curve of LiF is shown in Figure 4.

During the heating process, there are three regions of interest, *preheat*, *acquisition* and *anneal* (see Figure 4). In the preheat region, electrons at low energy trap levels, which are responsible for short term fade, are released. The TLD element response is measured in the acquisition region (~270°C). The element response is used to calculate the radiation dose. In the anneal region, electrons trapped at high energy levels are released. The purpose of the anneal region is to re-condition the TLD element for re-use.

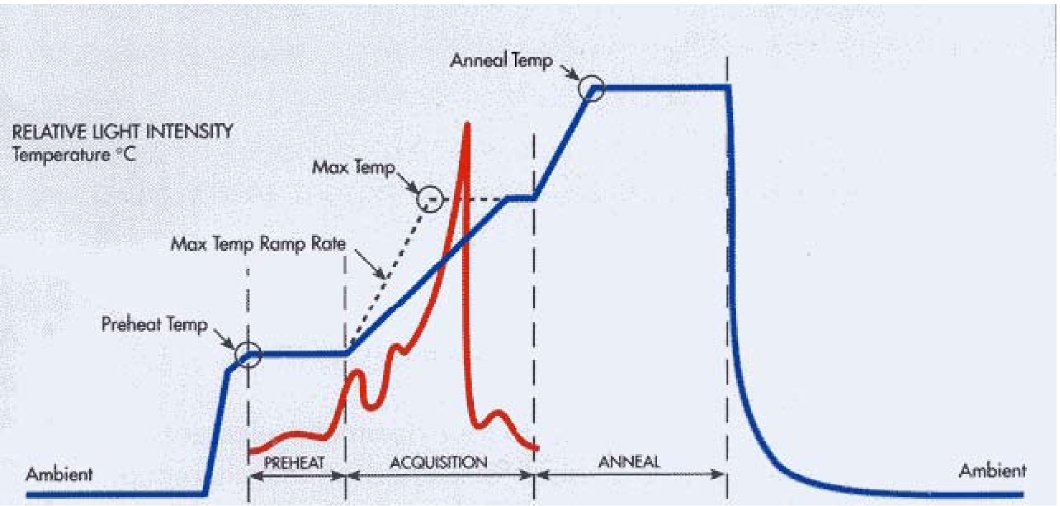

Figure 4: Glow curve for LiF. The flatter line indicates the temperature. The curvy line represents the relative light intensity.

After being worn for the prescribed period (typically monthly or quarterly), the TLD elements are heated electrically and the intensity of the resulting luminescence measured. The light pulses are measured with a photomultiplier tube (PMT), much like in a scintillation counter. The most common techniques used for heating TLD crystals include:

- heating through direct contact
- heating with hot gas
- heating with infrared lamps

The instruments used to read TLD's are collectively referred to as *TLD readers*. Each TLD manufacturer will have its own TLD reader design. Most TLD readers are automated and computer controlled, and include comprehensive TLD data management and quality assurance software.

The radiation dose from a TLD is determined through a complex set of steps which are summarized below:

1. Integrate the light output under the glow curve acquisition peak for each TLD element (using the TLD reader). The light output is summed either by counting light pulses or light frequency pulses.

2. Convert the light output to a TLD response (exposure).

3. Convert the TLD responses to an equivalent radiation dose using a *dose algorithm*.

A dose algorithm, is a set of instructions which are used to determine the type of radiation from the element responses (via element response ratios), and to apply the appropriate correction factors to convert the TLD element responses to an equivalent dose. A sample dose algorithm for a four-element TLD (E1, E2, E3 and E4) is shown in Figure 5. Each type of TLD will have its own dose algorithm. Dose algorithms are often optimized for the particular radiation environment at a facility. Because of their complexity, dose algorithms are usually computer based. The details of dose algorithms are beyond the scope of this training module.

Like other detectors, the response of each TLD element must be measured and this is done by exposing each TLD element to known doses of radiation.

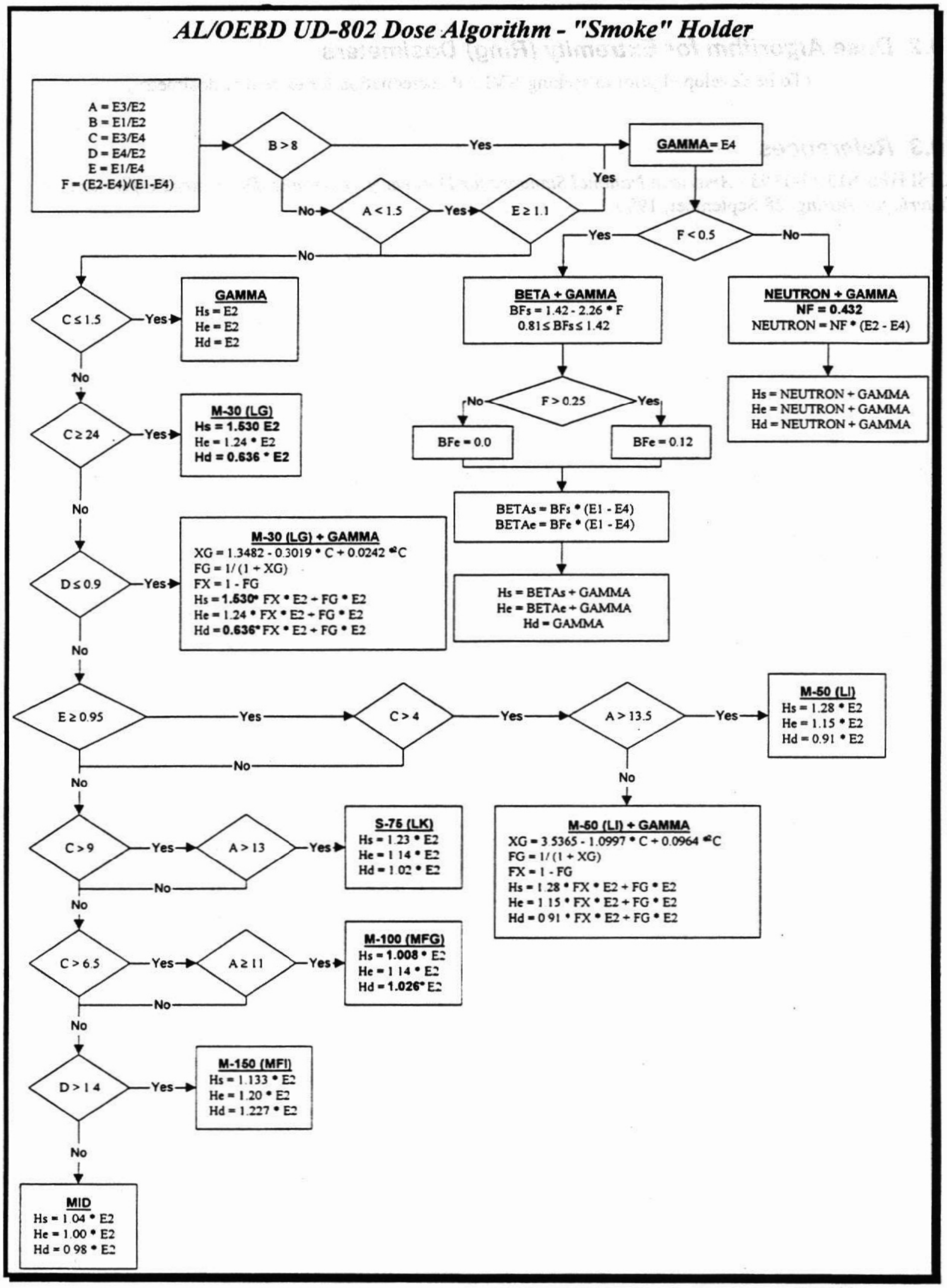

Figure 5: Typical TLD dose algorithm. (From Panasonic)

2.2. TLD Dosimetry System Suppliers

The major manufacturers of TLD equipment in North America are:

- National Dosimetry Services (NDS)
  (Radiation Protection Bureau of Canada)
- Landauer
- Harshaw
- Panasonic

The major commercial dosimetry service providers in North America are:

- National Dosimetry Services (NDS)
- Landauer
- Global Dosimetry Solutions (formerly ICN)

Examples of the various types of TLD's and TLD readers used in North America are shown in Figure 6 through Figure 11.

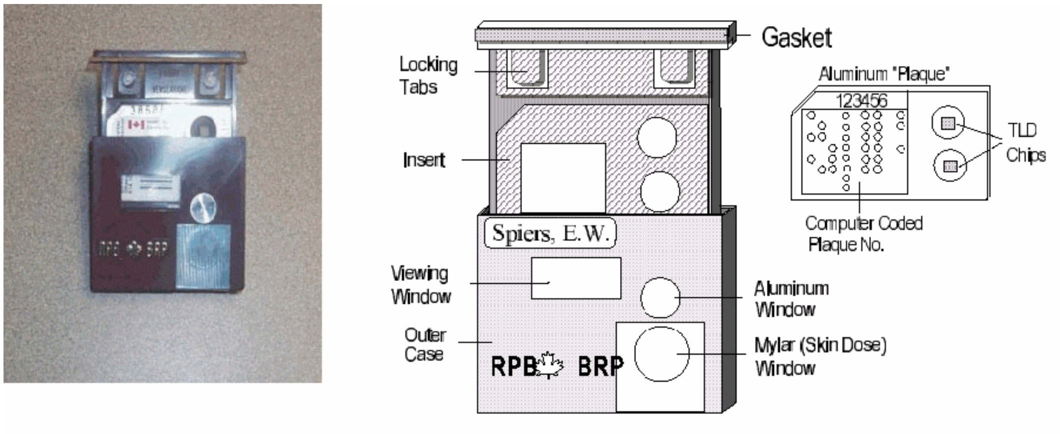

Figure 6: NDS TLD.

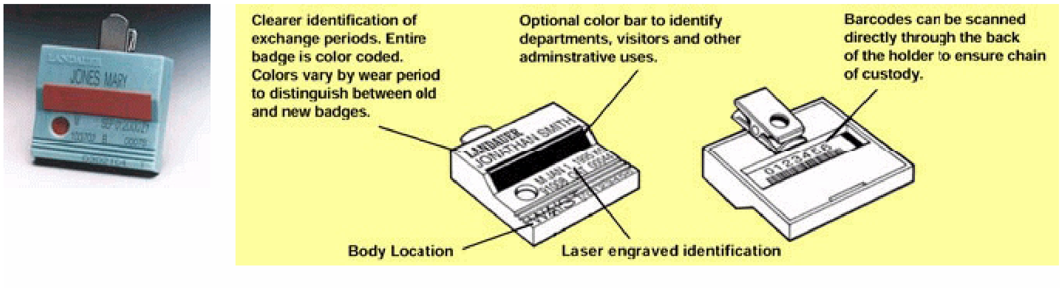

Figure 7: Landauer TLD.

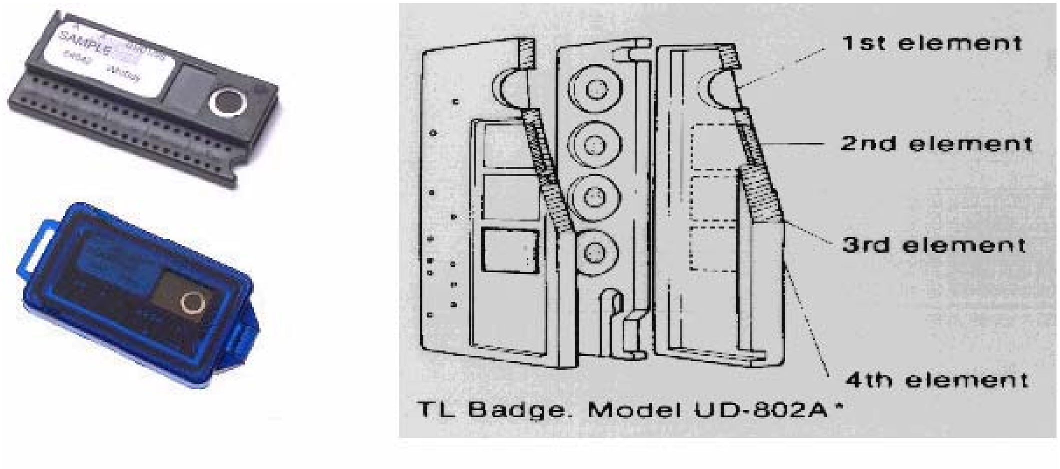

Figure 8: Panasonic TLD.

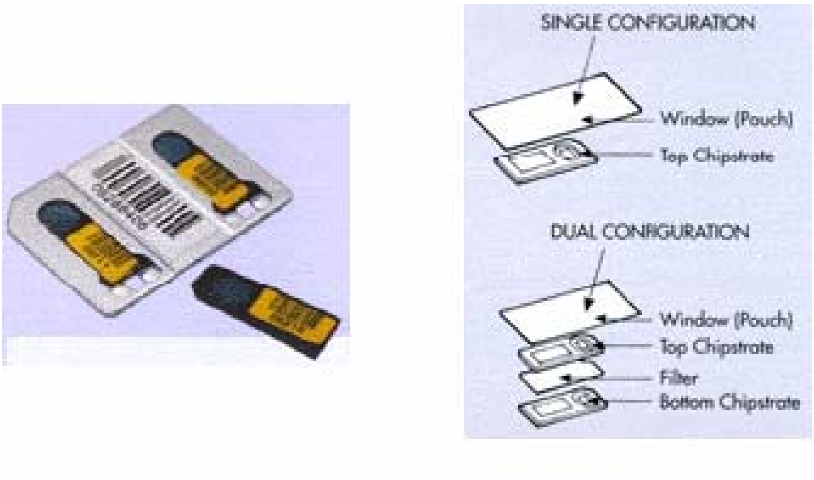

Figure 9: Global Dosimetry Solutions (formerly ICN) TLD.

Figure 10: Landauer and NDS extremity TLDs.

Figure 11: Harshaw and Panasonic TLD readers.

The Landauer Luxel dosimeter shown in Figure 12 is actually not a TLD but an optically stimulated luminescence (OSL) dosimeter. The Luxel dosimeter utilizes aluminium oxide crystals which when optically stimulated using a laser (not heat) release energy of trapped electrons that were created by exposure to radiation. The Luxel is the latest in luminescent dosimeter technology and will eventually replace the Landauer TLD. The major advantage of the Luxel dosimeter is that it has a ten-fold lower detection limit and can be re-analysed.

Figure 12: Landauer Luxel dosimeter.

In Canada, TLD’s are the most common dosimeter used to measure external radiation dose. Further, TLD’s are currently the only device certified by the Canadian Nuclear Safety Commission (CNSC) for external dosimetry. The Landauer Luxel dosimeter has not yet been certified for use in Canada. Typical specifications of TLD’s are given in Table 1.

Table 1: TLD characteristics.

Dose Range	Personnel TLD's: 0.1 mSv - 10 Sv
Radiation type	X-ray/gamma/beta/neutron (with appropriate TLD crystals)
Accuracy	approximately ±10%
Stability (fading)	Negligible loss of stored dose when read out one month after exposure (< 5% for 3 months) - depends on crystal type
Linearity	Light output is directly proportional to dose up to 100 mSv.
Repeatability	< 5 percent coefficient of variation
Environmental Effects	Operating range: -10°C - +40°C Humidity: <80% RH
Re-Use	approximately 300 cycles

TLD's have several advantages for personnel monitoring including:

- TLD's can be used with radiation fields of widely varying intensity
- TLD's can store information for long periods of time without fading
- TLD's are reusable, and can be used for many applications because of their small size
- TLD's are less energy dependent than film dosimeters or pocket ion chambers
- TLD's are inexpensive

However, TLD's do suffer from some disadvantages:

- They cannot be analysed immediately.
- Environmental factors such as humidity and heat may affect results.
- They can only be analysed once.

2.3. Proper Care and Use of TLD's

To ensure the proper care and use of TLD's, the following guidelines should be followed.

<u>Handling TLD's</u>

- ‣ Do not expose the TLD to high temperatures, water, direct sunlight or fluorescent light.

- ‣ Change the TLD's in a clean, dry area away from direct light and avoid direct skin contact.

Wearing TLD's

- Whole-body TLD's should be attached firmly to clothing between the waist and the neck.
- Extremity TLD's (rings, bracelets) should be worn facing the radiation source. If gloves are used, the extremity TLD should be worn under the gloves.
- Never share your TLD or wear another person's TLD. Each dosimeter is intended to be worn by only the designated person.
- Do not intentionally expose TLD's to radiation. Intentional tampering with dosimeters is a very serious matter.
- If you lose or damage your TLD, or suspect that it has been contaminated, you should stop working with radiation until you receive a replacement TLD.

TLD Storage

- Store TLD's in a holder or rack when not in use.
- TLD's are best stored in a low radiation background area away from direct light and heat.

It is good practice to keep extra TLD's as replacements for lost or damaged ones.

3. Film Badges

Another very commonly used personal monitoring device is the film badge. The film badge consists of a packet of two or more pieces of film wrapped in light proof paper or Mylar and worn in a suitable plastic or metal container (see Figure 13 and Figure 14). Film badges are primarily used for X-ray, gamma and beta radiation dosimetry, but can also be used for neutron radiation dosimetry.

Film badge dosimetry is based on the fact that the silver halide in the photographic emulsion exposed to ionizing radiation, results in a darkening of the film. The degree of darkening (which is called optical density of the film) can be precisely measured with a photoelectric densitometer. By comparing the degree of darkening of the film worn by an exposed individual to that of films exposed to known amounts of radiation, the exposure to the individual's film may be determined. Small variations in emulsions greatly affect their quantitative response to radiation.

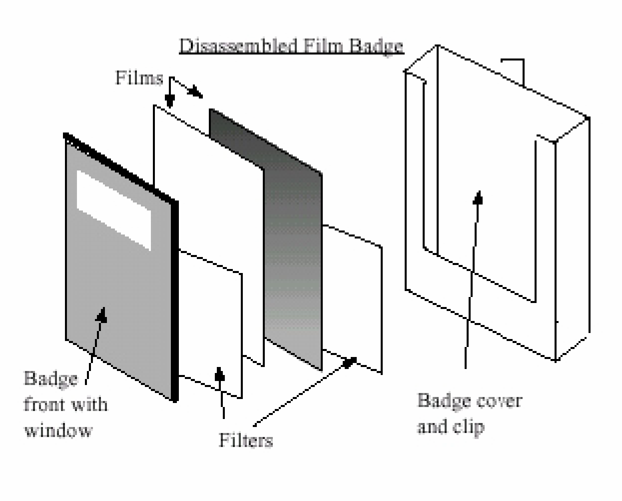

Figure 13: Typical film badge assembly. (From Global Dosimetry Solutions, formerly ICN)

Figure 14: Global Dosimetry Solutions (formerly ICN) film badge.

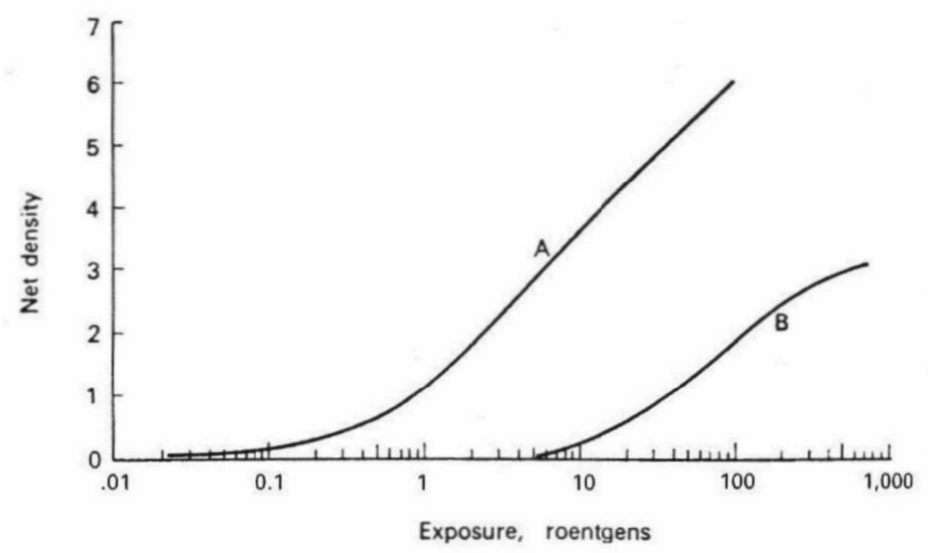

Figure 15: Relationship between radiation exposure and optical density of two different film badges. (From Cember: Fig. 9.21)

The films used in film badge dosimeters are highly energy dependent in the low energy range from about 0.2 MeV gamma radiation downward (see Figure 16). This energy dependence arises from the fact that the photoelectric cross section for the silver in the emulsion increases much more rapidly than that of air or tissue as the photon energy decreases below about 200 keV. A maximum sensitivity is observed at about 30-40 keV. Below this energy, the sensitivity of the film decreases because of the attenuation of the radiation through the badge casing. As a result of this very strong energy dependence, film badges are useless for X-rays less than 200 keV unless the film is calibrated with radiation of the same energy distribution as the radiation being monitored or the energy dependence of the film is accounted for. This allowance for energy dependence is made by selective filtration (like the TLD). The film badge is designed so that radiation may reach the film directly through an open window, or the radiation may be filtered by the film badge holder or by one of several different filters, such as aluminium, copper, cadmium, tin, silver and lead. The exact design and choice of filter is governed by the type of radiation to be monitored. The evaluation of the exposure is then made by considering the ratio of the film densities under each of the various filters.

Beta radiation dose is determined from the ratio of the open window film reading to that behind the filters. If exposure was beta radiation only, then film darkening would be seen only in the open window area of the film. To help distinguish between low energy gamma rays and beta rays, a comparison is made between the darkening in the open window and under two thin filters, such as aluminium and silver, which are of the same density thickness and therefore equivalent beta particle absorbers. The different atomic numbers, however, result in much greater low energy X-ray filtration by the silver filter than by the aluminium filter, thereby giving different degrees of darkening under the two filters. Interpretation of mixed beta-gamma radiation with a film badge is difficult because of the greatly different penetrating powers of beta and gamma radiation. For this reason, information from beta radiation monitoring with film badges is used mainly in a qualitative or semi-quantitative manner to evaluate exposure. Consequently, film badges are normally used for X-ray and low energy gamma radiation dosimetry.

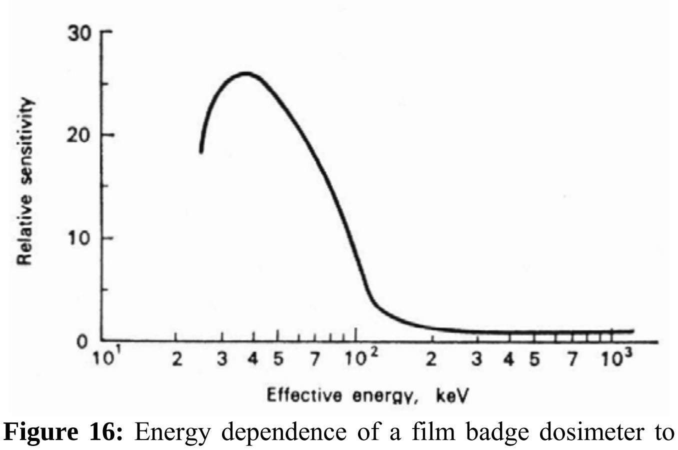

Figure 16: Energy dependence of a film badge dosimeter to X-rays. (From Cember: Fig. 9.22)

If nuclear track film (CR-39 is most commonly used) is added to the badge, fast neutrons (>0.5 MeV) can be measured. Irradiation of the film by fast neutrons results in proton recoil tracks due to elastic collisions between the hydrogen nuclei in the film's paper (or Mylar) wrapper, in the emulsion, and in the film base. Because the concentration of hydrogen atoms in the film and its wrapper is not very much different from that of tissue, the response of the film to fast neutrons is approximately tissue-equivalent, and the number of proton tracks per unit area of the film is therefore proportional to the absorbed dose. Fast neutron exposures are estimated by scanning the developed film with a high-powered microscope and counting the number of proton tracks per square centimetre of film.

Film badges can measure X-ray and gamma radiation dose between 0.1 mSv and 15 Sv and beta radiation between 0.5 mSv and 10 Sv. For fast neutrons, film badges can measure approximately 1 mSv/week. Dose rates higher than this may result in a saturation of the track etch film.

Some examples of film badges are shown in Figure 17. One of the major suppliers of film badge and film badge dosimetry services in North America is Global Dosimetry Solutions (formerly ICN). Figure 18 shows the first commercially manufactured film badge in the world (1905).

Figure 17: Examples of film badge dosimeters from ICN (now Global Dosimetry Solutions).

Figure 18: First commercially manufactured film badge (1905).

Some of the advantages of film badges include:

- Film badges can measure a wide range of doses
- Film badges can distinguish between penetrating radiation (gamma radiation) and non-penetrating radiation (beta and X-ray radiation less than 20 keV)
- Film badges provide a permanent record of the exposure
- Film badges are practical, small, lightweight and relatively inexpensive

Some of the disadvantages of film badges are:

- The response of the film to radiation is highly energy dependent
- The films cannot be read immediately
- Environmental conditions such as heat and humidity will affect the film's response to radiation
- Wear periods are generally less than TLD's
- The film cannot be re-used

The guidelines for proper use and care of film badges are the same as for TLD's. When using film badges it is important to wear the badge with the emulsion side facing outward.

4. Pocket Dosimeters

To measure radiation dose, the response of the instrument must be proportional to absorbed energy. A basic instrument for doing this is the free air ionization chamber or "air wall" ionization chamber. Ionization chambers of this type are often called *pocket dosimeters*, and are widely used for personnel monitoring. Pocket dosimeters are designed to be clipped to a person's lapel or pocket and unlike TLD's or film badges can provide an immediate reading of radiation exposure. Two common types of pocket dosimeters are the condenser-type pocket dosimeter and the direct reading dosimeter.

4.1. Condenser-Type Pocket Dosimeter

The condenser-type pocket dosimeter is of the indirect reading type which requires an auxiliary device to read the measured dose. This device is in reality an electrostatic voltmeter that is calibrated in roentgens and is called a charge-reader. An example of a condenser-type pocket dosimeter is shown in Figure 19.

Commercially available condenser-type pocket dosimeters measure integrated X-ray and gamma radiation exposures up to 200 mR with an accuracy of about ±15% for energies between 0.05 and 2 MeV. For radiation energies outside this range, correction factors are supplied by the manufacturer. These dosimeters can also be used to measure beta radiation whose energies exceed 1 MeV. By coating the inside of the chamber with boron, the pocket dosimeter can also be made sensitive to thermal neutrons. The standard pocket dosimeter however, is designed for measuring X-ray and gamma radiation only. Pocket dosimeters measure deep dose.

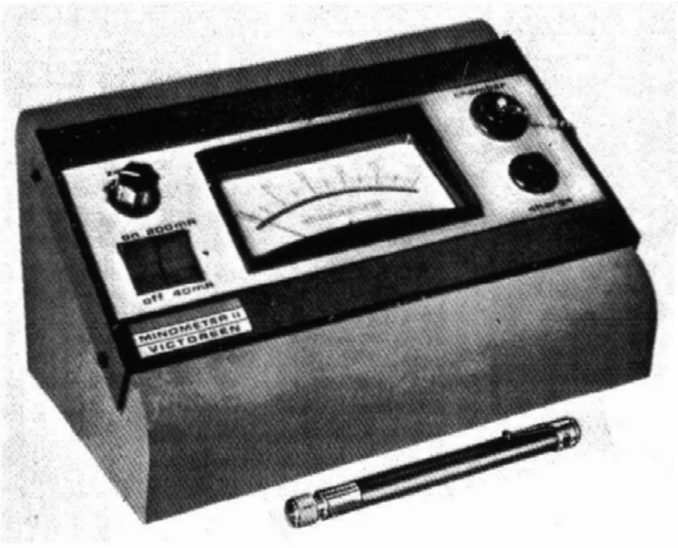

Figure 19: Condenser-type pocket dosimeter and its charge-reader.

4.2. The Direct Reading Dosimeter

The direct reading pocket dosimeter (DRD) is a quartz fibre electroscope-type dosimeter. In a DRD, a quartz fibre is displaced electrostatically by charging, via a battery operated charger, to a potential of approximately 200 V. A schematic diagram of a DRD is shown in Figure 20. An image of the fibre is focussed on a scale and is viewed through a lens at one end of the instrument. Exposure of the dosimeter to radiation discharges the fibre, thereby allowing it to return to its original position.

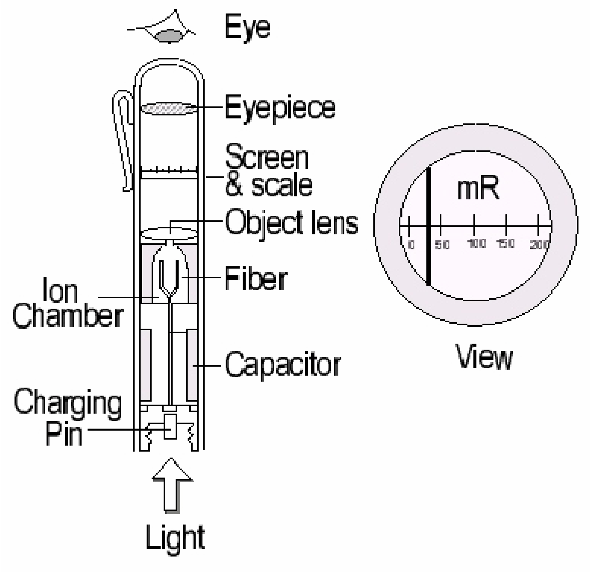

Figure 20: DRD schematic. (Source: Canadian Nuclear Safety Commission, *Radiation Safety Officers Handbook Info-0718*. Reproduced with permission of the Minister of Public Works and Government Services, 2001.)

The amount discharged, and consequently the change in position of the fibre, is proportional to the radiation exposure. An advantage of the DRD is that it does not have to be recharged after being read.

Commercially available DRD's have a measurement range of 0-200 mR and read within about ±15% of the true exposure for energies from about 0.05 - 2 MeV. The photon energy response for a DRD is shown in Figure 21.

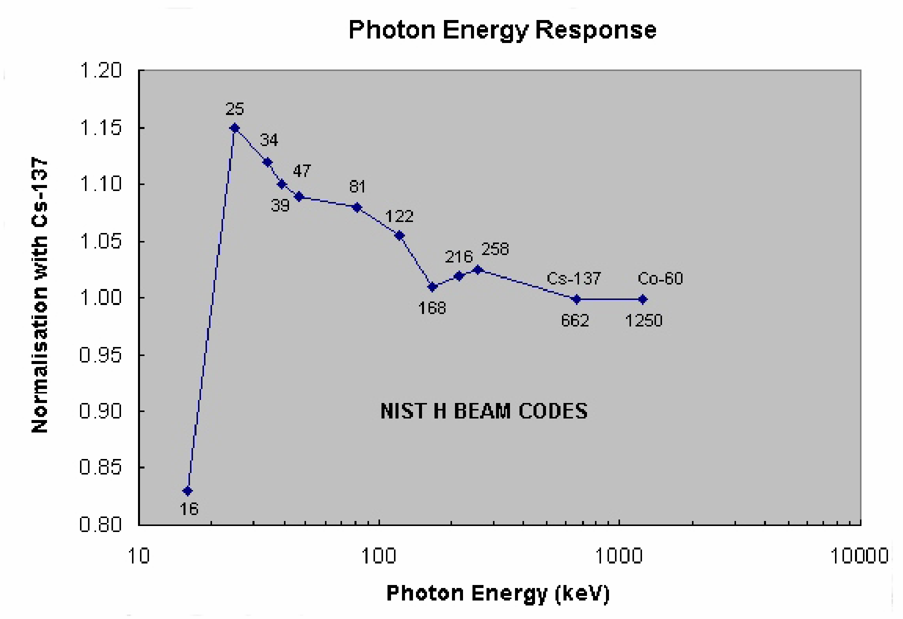

Figure 21: Photon energy response for a DRD.

Before and after use, the DRD is charged using a charger. The charger is equipped with a control knob to regulate the amount of charge deposited, so that the fibre can be set to the zero position on the scale.

Some of the major manufacturers of pocket dosimeters include:

- Ludlum Instruments
- Victoreen
- Automess

A typical DRD and charger is shown in Figure 22. The oldest known DRD in world (1932) is shown in Figure 23.

Figure 22: A typical DRD and charger. (From Victoreen)

Figure 23: World's oldest known DRD (1932).

Pocket dosimeters discharge slowly even when they are not in a radiation field because of background radiation and because charge leaks across the insulator that separates the central electrode from the outer electrode. A dosimeter that leaks more than 5% of the full-scale reading per day should not be used. Because of leakage and possibility of malfunction due to being dropped, pocket dosimeters are usually worn for no more than one day at a time and recharged and zeroed each time they are used.

Some of the advantages of pocket dosimeters include:

- A cumulative exposure can be read at any time or location without ancillary equipment (TLD's and Film badges require processing)
- Pocket dosimeters can be used repeatedly by recharging or re-zeroing
- Pocket dosimeters have a long shelf life with little to no maintenance requirements
- Pocket dosimeters measure gamma exposure accurately
- Pocket dosimeters are relatively insensitive to environmental conditions

Some of the disadvantages of pocket dosimeters are:

- Pocket dosimeters are sensitive to mechanical shock such as being dropped
- Pocket dosimeters slowly discharge even when not exposed to a radiation field
- The initial cost of pocket dosimeters is quite high.

Pocket dosimeters are often used in conjunction with TLD's (or film badges) to provide real-time estimates of radiation dose. This is particularly advantageous in nuclear power generating stations or in uranium mines where there is a potential for high acute radiation exposures. With TLD's and film badges, a high radiation dose would not be identified until long after the exposure has occurred.

5. Electronic Dosimeters

The continuing evolution of solid-state electronics and electronic data processing has led to ever smaller, more reliable and more sophisticated instruments. This miniaturization has made possible small lightweight, accurate electronic dosimeters that are more useful for personal dosimetry and dose tracking than either TLD's or film badges.

Electronic dosimeters measure and display instantaneous dose rate and integrate over time to obtain, store and display the cumulative radiation dose. They are also designed to alarm at any desired dose rate or cumulated radiation dose. The data is stored in a non-volatile memory and can be downloaded into a computer for dose-tracking and record-keeping purposes.

Electronic dosimeters employ solid-state semiconductor detectors (silicon diodes) to detect gamma and beta radiation over a very wide range of dose rates and doses. With accuracy of about ±10% for ^{137}Cs gammas, electronic dosimeters measure dose rates from 1 μSv/h to 10 mSv/h and store doses from 1 μSv to 10 Sv.

Electronic dosimeters typically incorporate several detectors, each one responding to radiation that had passed through different filters in order to account for the inherent energy dependence of the detectors. When the outputs on the several detectors are added together, the resulting output is approximately proportional to the absorbed dose rate. By this technique, the response of electronic dosimeters is approximately energy independent from 60 keV to 1.5 MeV to within ±25%. Electronic dosimeters are capable of measuring deep and shallow dose.

There are several manufacturers of electronic dosimeters including:

- Bicron
- Automess
- Rados
- Siemens
- Ludlum
- Aptec
- Health Physics Instruments

Some examples of electronic dosimeters are shown in Figure 24.

Figure 24: Examples of commercially available electronic dosimeters. (From Automess, Rados, Health Physics Instruments, and Siemans)

Figure 25: Apollo mission electronic dosimeter.

Electronic dosimeters share many of the advantages of pocket dosimeters including:

- Electronic dosimeters give instantaneous dose rate and accumulated dose readings
- Electronic dosimeters can be programmed to alarm when specified dose rates and/or cumulative doses are exceeded
- Electronic dosimeter can accurately measure X-rays and gamma rays over a wide range of dose rates and doses

Some of the disadvantages of electronic dosimeters are:

- Electronic dosimeter require regular calibration (at least annually)
- Electronic dosimeters can be easily damaged if dropped
- Electronic dosimeters may be sensitive to environmental conditions
- Strong electromagnetic fields can affect performance
- Electronic dosimeters are battery operated
- Electronic dosimeters are very expensive
- Electronic dosimeters are expensive to repair

Like pocket dosimeters, electronic dosimeters are usually used in conjunction with TLD's or film badges.

6. Regulatory Requirements

Under the Canadian Nuclear Safety Commission (CNSC), only dosimetry service providers licensed under the CNSC Regulatory Standard S-106 *Technical and Quality Assurance Standards for Dosimetry Services in Canada* can provide dosimetry services in Canada.

Radiation doses measured using licensed dosimetry services are referred to as “legal dose” or “actual dose” which is to be distinguished from dose estimates obtained from other instruments or techniques. All legal doses generated by a licensed dosimetry service must be submitted to the National Dose Registry (NDR).

CNSC licensees are required by law to use a licensed dosimetry service when there is a likelihood that a worker(s) may receive an annual radiation dose from any single pathway in excess of 5 mSv. When radiation doses are below 5 mSv, approved dose estimate techniques may be used. Although not stated in the regulations, it is implied that, as part of an effective radiation protection program, dosimetry services be provided well before this dose limit is reached (usually for doses greater than 2 mSv).

Currently, only NDS and Landauer are licensed in Canada to provide commercial external dosimetry services using TLD’s. All of the nuclear power generating facilities in Canada are also licensed but only to provide dosimetry services to their employees (they also use TLD’s). It should be noted that Landauer, Harshaw and Panasonic TLD systems have met S-106 technical specifications, performance testing and accuracy requirements.

The Radiation Safety Institute of Canada operates the only dosimetry service in Canada that is licensed by the CNSC to measure worker exposure to radon and thoron progeny and long-lived radioactive dust. This is done with the use of the Institute’s personal alpha dosimeters (PAD’s) which record the presence of those substances on a filter and film within a measuring head in the PAD (Figure 26).

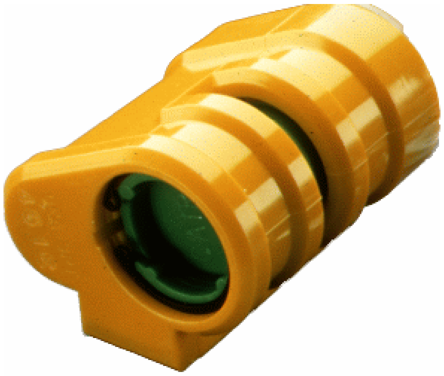

Figure 26: Personal alpha dosimeter. (From the Radiation Safety Institute of Canada)

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Radiation Safety Institute of Canada

Institut de radioprotection du Canada
National Office, Toronto

Training Course

Radiation Safety Officer
(RSO-1)

Controlling Radiation Exposure

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Sixth Revision

Controlling Exposure to Radiation

CONTENTS

1.0 Introduction

The fundamental aims of radiation protection are:

- To avoid fatalities
- To avoid disabilities
- To keep radiation exposures as low as reasonably achievable (ALARA)

Protection against radiation is concerned with two separate situations:

- External radiation exposure - from sources outside the body
- Internal radiation exposure - from sources inside the body

The general biological effects of ionizing radiation from external and internal sources are the same. However, precautions taken against the one hazard are of little use in protecting against the other, and the corresponding methods of estimating dose are different. Therefore, external and internal radiation are treated separately, with the methods of controlling external exposure being considered first.

Before discussing controlling external and internal radiation in detail, the ALARA concept needs to be explained.

2.0 The ALARA Principle

The concept of the maximum permissible dose is misleading in that it implies a safe level of exposure. An analogy may be made to the highway speed limit where travelling at 110 km/h, for example, does not imply danger. However, as speed is increased, the risk of injury due to an accident also increases. Management of radiation doses is similar except that there is emphasis placed on keeping the radiation exposure as low as reasonably achievable. Where it may be permissible to travel at the posted speed limit on a highway, in radiation protection the limit is the upper bound and all doses must be kept as far below this value as practicable. This is the ALARA principle.

The concept of ALARA is to keep all exposures “as low as reasonably achievable,” social and economic factors taken into consideration. Although the federal regulatory agencies may prescribe a certain radiation dose limit, it is incumbent on persons working with radioactive materials and their supervisor to ensure that all radiation doses are in accordance with the ALARA principle.

3.0 External Radiation Exposure Control

External radiation exposure can be decreased by:

- Reducing the *time* spent near a source;
- Allowing the source to *decay* before approaching it;
- Increasing the *distance* between oneself and the source;
- Using *shielding* material placed between oneself and the source.

3.1. Time

Radiation exposure can be lowered by simply limiting the time a person spends near the source. For example, if the radiation level in an area is 150 μGy/h and it takes 6 hours to complete a job, the dose received would be

$$\text{Dose Rate} \times \text{Time} = \text{Dose}$$

$$150\ \mu\text{Gy/h} \times 6\ \text{h} = 900\ \mu\text{Gy}$$

If you wish to limit the dose received by a person to a certain value, and you know the radiation dose rate, you may calculate the maximum exposure time by using the formula:

$$\text{Time} = \text{Dose} \div \text{Dose Rate}$$

<u>Example 1</u>

If you want to limit your dose rate to 750 μGy, how long could you work in a dose rate of 500 μGy/h?

Solution:

$$\text{Time} = \text{Dose} \div \text{Dose Rate} = 750\ \mu\text{Gy} \div 500\ \mu\text{Gy/h} = 1.5\ \text{hours}$$

<u>Example 2</u>

The gamma radiation field in an area is 150 μGy/h. If you are to receive a dose no greater than 200 μSv, how long can you work there? Assume that there are no other hazards.

Solution:

We notice that the units are not the same. However, we know that for gamma radiation $W_R = 1$, and a 200 μGy absorbed dose would result in a 200 μSv equivalent dose.

Time = Dose ÷ Dose Rate = 200 μGy ÷ 150 μGy/h = 1.33 hours = 1 h 20 min.

3.2. Decay

A second way of reducing exposure is to wait for the source to decay before you start work. The radiation level will drop by a factor of 2 for every half-life you wait. This is a good approach when work has to be done in a radiation field that will decrease very quickly with time (half-lives of minutes or hours).

Example 3

You have to work for half an hour near the snout of the fuelling machine of a nuclear reactor. The radiation field is 3.2 mGy/h and the activation products causing this field have an apparent half-life of 2 hours. How long should you wait before doing the job, so that the dose you receive will be 0.1 mGy or less?

Solution:

Initially, the dose rate will be 3.2 mGy/h or 1.6 mGy per half hour. This will decrease by a factor of 2 for every 2 hours, delay owing to the radioactive decay of the activation products. Therefore, after 2 hours the dose received will be 0.8 mGy; after 4 hours 0.4 mGy; after 6 hours 0.2 mGy; and after 8 hours 0.1 mGy. Therefore, if you wait 8 hours, you will receive only 0.1 mGy during the half an hour you are working.

Incidentally, this problem assumed a constant half-life of 2 hours. In practice, the effective half-life might well increase because the short-lived activation products would decay faster than the longer-lived. This means that you might get less decay than you would expect. Therefore, before you start the job, you must check the radiation field again.

A useful rule of thumb when determining the decrease of radiation fields with time is that they reduce by a factor of 100 for every 7 half-lives that elapse.

3.3. Distance

If you increase your distance from a small radiation source, there will be a marked reduction in the radiation field. This is due to the fact that a small (point) gamma source emits photons equally in all directions. Therefore, as you move away from the source, the gamma radiation field decreases due to the spreading out of the emitted photons.

The distance criteria for gamma exposure is based on the fact that the exposure rate of radiation at any point is inversely proportional to the square of the distance from the source (assuming negligible absorption between the source and the point of interest) and directly proportional to the amount of activity.

$$\text{Exposure Rate} = \frac{\tau A}{d^2}$$

where:

A = the activity of the source in mC_i
d = the distance from source to dose point in cm.
τ = the specific gamma constant in $R.cm^2/(h)\cdot(C_i)$.

Specific gamma constant for various radionuclides are given on page 37 of this module.

Note: The above formula applies only to point-source gamma radiation.

Example 4

Find the exposure rate 2 cm from a 10 mCi source of ^{60}Co.

Solution:

For ^{60}Co, $\tau = 13.2\ R.cm^2/(h)\cdot(mC_i)$

$$\text{Exposure Rate} = \frac{\tau A}{d^2} = 13.2\frac{R\cdot cm^2}{h\cdot mCi}\times\frac{10\ mCi}{4\ cm^2} = 33\ R/h$$

3.3.1. Inverse Square Law

The intensity of *gamma radiation* varies inversely with the square of the distance from the source; that is, doubling the distance drops the exposure rate to 1/4, tripling the distance drops it to 1/9, and so on.

It is important to realize that the inverse square law applies to gamma radiation only, and then only for point sources and not to beams. In practice, a point source is considered to be one whose largest dimension is smaller than 1/5 the distance between it and oneself.

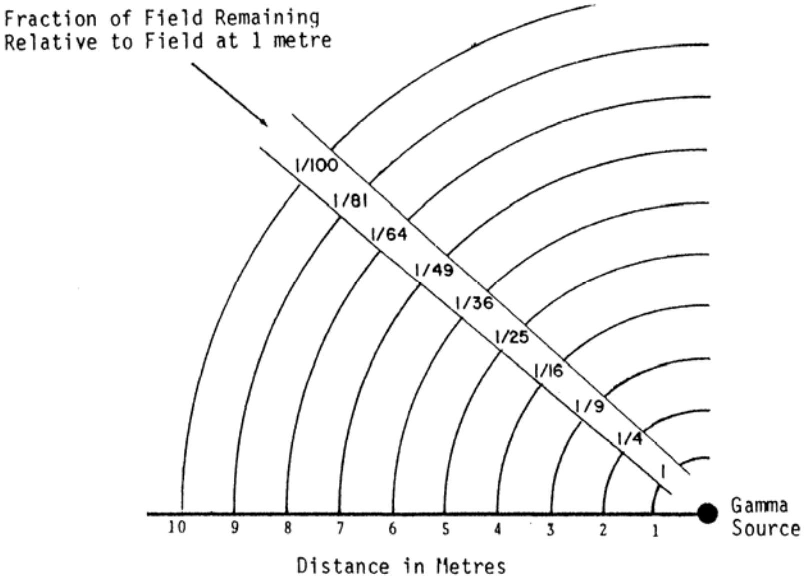


Figure 1: Illustration of the inverse square law (from J.U. Burnham).

Example 5

If the gamma radiation field at 1 metre from a source is 320 μGy/h, what is it at 4 metres?

Solution:

Figure 1 indicates that the radiation field has been reduced to 1/16 of its level at 4 metres, i.e.

$$(1/16) \times 320\ \mu\text{Gy/h} = 20\ \mu\text{Gy/h}$$

or,

$$\text{Exposure Rate at 1 m} = 320\ \mu\text{Gy/h} = \tau A/1^2\text{, therefore } \tau A = 320\ \mu\text{Gy m}^2\text{/h}$$

Hence,

$$\text{Exposure Rate at 4m} = \tau A/4^2 = \tau A/16 = 320/16 = 20\ \mu\text{Gy/h}$$

Example 6

If the gamma radiation field at 10 m from a source is measured to be 20 μGy/h, what radiation field would you expect at 4 m from the source?

Solution:

Figure 1 shows that the gamma field at 1 m would be 100 times greater than at 10 m, and that the field at 4 m would be 1/16 of that at 1 m. Hence,

$$\text{Gamma field at 4 m} = (100/16) \times 20\ \mu\text{Gy/h} = 125\ \mu\text{Gy/h}$$

This second example leads us into an easy method to use if you want to calculate gamma radiation fields at various distances from the source.

It is based on common sense applied to a *distance factor*. Divide the larger distance by the smaller distance and square the result; this is your distance factor. Now multiply or divide the known dose rate by this distance factor. Multiply if the dose rate should increase and divide if it should decrease.

Look again at example 6. The larger distance is 10 m and the smaller distance is 4 m. The distance factor is $(10/4)^2 = 2.5^2 = 6.25$. Since we are moving towards the source, the radiation field will increase. Therefore, we multiply the dose rate of 20 μGy/h by 6.25 to get 125 μGy/h for the dose rate at 4 m.

More generally, if we know the exposure rate, ψ_a, at distance a from the source, then

$$\psi_a = \tau A/a^2$$

and so

$$\tau A = \psi_a\, a^2$$

Therefore, the exposure rate, ψ_b, at distance b from the source will be

$$\psi_b = \tau A/b^2 = \psi_a\, a^2/b^2$$

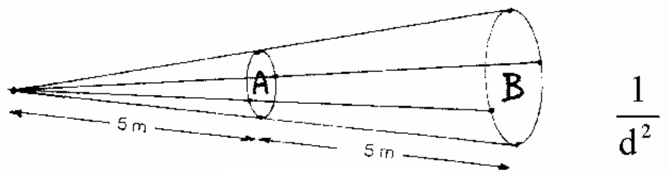
3.3.1.1. Limitations of the Inverse Square Law

- The inverse square law is valid only for gamma rays because you can neglect any shielding by the air. This is not true for alpha and beta radiation. Alpha particles have only a small range in air (less than 10 cm for the highest energy alpha particles). Once the distance from the source becomes greater than this, no alpha particles are detected. The range of beta particles is usually much greater, but the same reasoning applies. Furthermore, beta particles all have different ranges, even if they all come from the same source. Because of scattering, they won't travel in straight lines anyway.

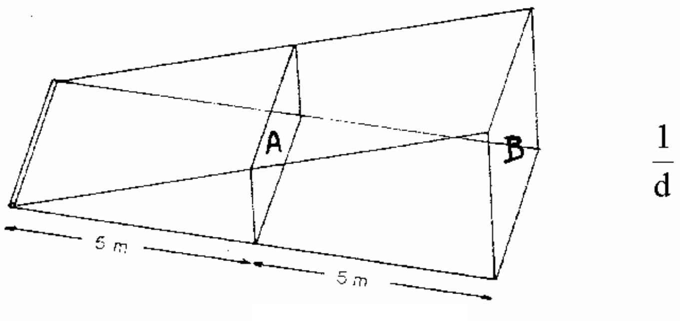
- In theory, the inverse square law holds true only for “point” gamma sources. In practice, a source may be quite large and the detector will still “see” it as a point source when the detector is sufficiently far away from it. We assume that the inverse square law won’t hold until the detector is about five source diameters from the source.
- When shielding is placed between the source and the detector, the inverse square law can only be applied if due allowance is made for the gamma photons absorbed by the shielding.
- If the gamma radiation is in the form of a beam, the inverse square law won’t apply either.

3.3.2. Line Sources and Plane Sources

It has been pointed out that the inverse square law applies only to point sources. For a line source, e.g., a pipe carrying radioactive fluid, the gamma field does not drop as rapidly as you move further away from it as it would for a point source.



Point Source



Line Source

Figure 2: Point and line sources (from J.U. Burnham).

For a point source, all the gamma photons in the cone passing through area A will also pass through area B. B is twice as far away as A, and, therefore, has four times the area of A (twice as high and twice as wide). The number of photons passing through one square metre located at B will therefore only be one quarter the number passing through one square metre at A. In other words, increasing the distance by a factor of 2 reduces the gamma field by a factor of 4. That's the inverse square law.

For the line source, Area B is only twice as large as Area A, and the radiation field, therefore, will be half of what it is at A. For example, if you go ten times as far away, the field will only drop by a factor of 10, and not by the factor of 100 that would apply to a point source.

For a large plane (or area) source, there will be only very little reduction in the gamma fields as you move away from it. You would not be able to measure a significant drop in gamma radiation intensity until you moved far enough away from the plane source for the spreading out of the gamma photons to become significant.

3.3.3. Beta Emitters as External Sources

Exposed (unshielded) beta emitters can cause some surprising external radiation fields. Due to the limited range of beta particles, we would expect beta radiation to drop off more quickly with distance than gamma radiation; on the flip-side, beta radiation will increase much more rapidly than gamma radiation as you approach the source.

Usually beta-gamma sources have enough shielding to absorb all the beta radiation. But if they don't, beware! A 10 MBq beta source delivers a dose rate of 200 mGy per hour at 2 cm. At the point of contact, the dose rate would be greater than 1000 Gy/h. Therefore, always use tongs when handling small beta sources.

3.4. Shielding

In some cases, the only practical way of reducing radiation exposures to an acceptable level is to install shielding between the source and yourself. Radiation shielding is a very complex subject, and, therefore, only a few basic points are discussed here.

3.4.1. Alpha Shielding

Alpha particles have a relatively small penetrating ability - even in air the most energetic alpha particles cannot travel more than 10 cm. The dead layer of one’s skin will stop them completely. Because of this, alpha sources outside the body do not present an external hazard and shielding against alpha particles is therefore quite unnecessary. However, alpha particles with a radiation weighting factor, W_R, of 20 are a very serious internal hazard and great care must be exercised to ensure that alpha sources are kept out of the body.

3.4.2. Beta Shielding

Beta (including positron) emitters present two potential external radiation hazards, namely, the beta rays themselves and the bremsstrahlung they produce. In addition, annihilation photons are always present with positron sources. Beta particles can be stopped in a shield surrounding the source if it is thicker than their range. To minimize bremsstrahlung production, this shield should have a low atomic number. It can be enclosed in another material (preferably of high atomic number) that is thick enough to attenuate the bremsstrahlung intensity to the desired level. For a shielded beta emitter, bremsstrahlung may be the only significant external radiation hazard.

3.4.3. Gamma-Ray Shielding

Gamma rays will penetrate to great depths in materials and no amount of shielding will stop all the radiation. The effectiveness of gamma ray shielding is frequently described in terms of the half-value layer (HVL), which is the thickness of absorber required to reduce the gamma radiation to half its initial intensity.

The first HVL reduces the radiation field by one half. The second HVL reduces the radiation by one half again, i.e., to one quarter of the initial level. The radiation levels after successive HVLs are:

Radiation after 1 HVL $= (1/2)^1 = 1/2$ of the original
Radiation after 2 HVLs $= (1/2)^2 = 1/4$ of the original
Radiation after 3 HVLs $= (1/2)^3 = 1/8$ of the original
Radiation after 4 HVLs $= (1/2)^4 = 1/16$ of the original
Radiation after 5 HVLs $= (1/2)^5 = 1/32$ of the original

After n HVLs, a gamma radiation field will be reduced to $(1/2)^n$. This leads to the following useful guides:

7 HVLs reduce the radiation field to about 1/100 or 1%
10 HVLs reduce the radiation field to about 1/1000 or 0.1%

Gamma photons interact with electrons. Therefore, those materials that have a large number of electrons per atom and a large number of atoms per unit volume will be the most effective gamma shields. Heavy metals like uranium, tungsten, gold and lead are good examples of such materials.

Concrete is a good structural material. Lead is not, because large lead shields require some sort of supporting frame. On the other hand, lead shields will be thinner than shields made of less dense materials, so for semi-permanent shielding we often use lead blocks, lead blankets or bags of lead shot. With these, the shield can be built up to fit the requirements. Lead blankets are convenient for the shielding of awkwardly shaped objects. They can be wrapped around pipes, for example.

Water may be used where it is necessary to see the source of radiation or to work on the source with remote tools. Spent reactor fuel is usually stored in deep bays filled with water. This water also absorbs the heat being generated by the fuel elements, and the water is easily cooled and purified.

Figure 3 shows the HVLs of various materials for a range of gamma energies. The HVLs are not constant for a given material, because the relative probabilities of the three gamma absorption processes (photoelectric effect, Compton effect, pair production) vary with gamma energy. In the range of energies that interest us, the HVL usually will increase with energy. In other words, for higher energy gamma photons you will need more shielding than for lower energy photons.

For gamma radiation with energies in the range where Compton scattering is the predominant absorption process, the weight of material required for a shield is generally about the same, regardless of the material used. For instance, the HVL of iron for 1 MeV gamma radiation is about 15 mm as compared with 30 mm for heavy concrete. Since iron is about twice as dense as heavy concrete, the total mass required for a shield will be roughly the same for both materials.

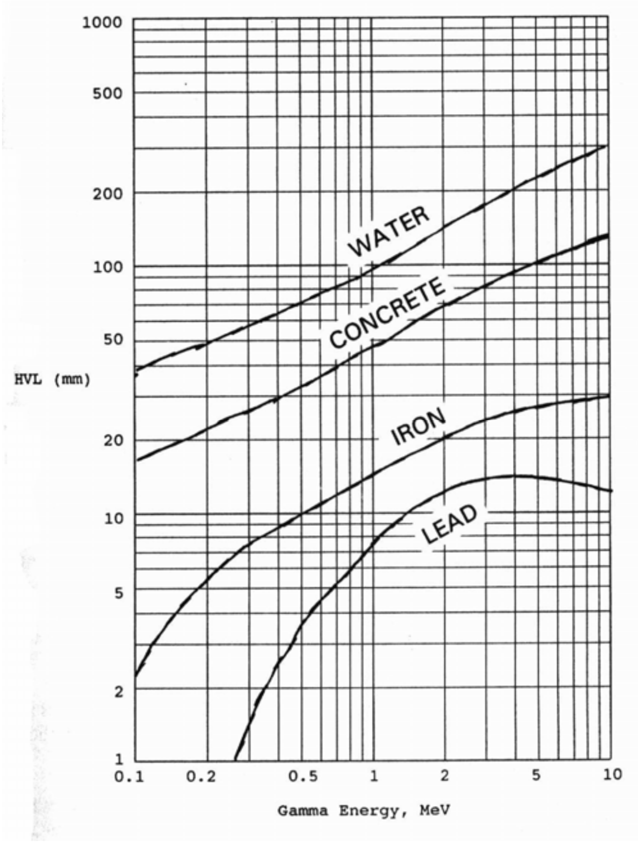


Figure 3: Half value layer for various materials (from Radiation Protection, reproduced with permission from J.U. Burnham).

Example 7

A beam of gamma radiation is found to be coming up through a small hole in the floor. A gamma survey meter shows a reading of 200 mGy/h at the hole. The gamma energy is assumed to be around 1 MeV. The radiation field has to be reduced to less than 50 µGy/h. How many lead blocks that are 50 mm thick would you need to block the beam?

Solution:

The radiation field has to be reduced from 200 mGy/h or 200,000 µGy/h to 50 µGy/h, which corresponds to a factor of 200,000/50 = 4000. We know that 2 HVLs will reduce the field by a factor of 4, and another 10 HVLs will reduce that by a factor of 1000, so we'll need 12 HVLs. Figure 3 tells us that the HVL for 1 MeV gammas is 8 mm of lead. Therefore, we'll need 12 × 8 mm, or 96 mm of lead. It seems that two 50 mm lead blocks should suffice.

In practical work situations, you should *never* base your radiation protection plans on shielding calculations alone, but you should *always* measure the radiation field after the shielding has been put in position.

3.4.3.1. Gamma Build-Up

The HVL values given so far apply only to narrow beams, and where the number of HVLs is not too large. In practice, we are more likely to come across gamma radiation that is not in the form of a narrow beam, but that is spreading out in all directions.

In this case, 3 HVLs would not reduce the field by a factor of 8, but rather less. The reason is that some photons that would have missed the detector if there had been no shielding are now scattered inside the shield (Compton effect) to interact with the detector. In other words, the gamma field will be greater than you would expect. The thicker the shield, the greater will be the chance of these scattering effects. It is possible to work out what the extra field would be, but in practice, you would just keep on adding to the shielding until your instrument measured a radiation field that you considered to be tolerable.

Some jobs require the use of temporary shielding while work is being done. You must ask yourself whether the extra dose you will absorb while you place the shield into position and then remove it is going to be off-set by the reduction in absorbed dose caused by the shielding. If the job is a long job, temporary shielding will be worthwhile, but for a short job it may not be.

3.4.4. Neutron Shielding

Shielding against neutrons is based on slowing down fast neutrons and absorbing thermal neutrons, principally by elastic scattering. Most effective in slowing down neutrons are the light elements, particularly hydrogen.

Many hydrogenous materials, such as water and paraffin, make efficient neutron shields. However, water shields have the disadvantage of needing maintenance; also, evaporation can lead to a potentially dangerous loss of shielding. Paraffin is flammable and may be a risk for that reason.

Concrete (ordinary or heavy aggregate) or earth is the neutron shielding material of choice in many applications. Often temporary neutron shielding must be provided in experimental areas around a reactor or an accelerator. Movable concrete blocks are convenient for this purpose. One must exercise care to assure that cracks, access ports, and ducts in such shielding do not permit the escape of neutrons. Vertical cracks should be staggered. (Natural sagging of concrete blocks under the force of gravity usually precludes the existence of horizontal cracks). In designing shielding against neutrons, it must be borne in mind that absorption of neutrons can lead to induced radioactivity and to the production of gamma radiation. Generally, surveys are desirable to check temporary neutron shielding before and during extensive use.

4.0 Internal Radiation Exposure Control

4.1. Behaviour of Internal Sources

When dealing with external exposure, protective means are fairly straightforward; usually the radiation level can be measured easily with an instrument, and the exposure can be stopped at any time by walking away from the source. However, the hazard presented by internal sources demands more elaborate precautions for the following reasons:

i. Only some of the radiation emitted by external sources is directed towards the body; the rest will not be able to interact with it. However, if such sources are taken into the body, all the radiation emitted is capable of interacting with the body.

ii. The chemical properties of some radionuclides cause them to concentrate in certain body organs or tissues rather than be spread throughout the body. This means that all the energy of alpha and beta emissions and part of the energy of the gamma emissions are absorbed in these tissues, causing them to receive much more radiation dose than the body as a whole.

Reasons (i) and (ii) explain why the dose rate from sources within the body is much greater than from the sources outside the body.

iii. Internal sources irradiate the body 24 hours a day, seven days a week, until they have been eliminated from the body by excretion and decay. In other words, you can't walk away from them as you can with external sources.

iv. While some radionuclides are eliminated fairly rapidly, there are others that remain in the body for years. And in many cases it is difficult to increase their rate of elimination from the body.

v. Once the radionuclides have entered the body, it is often difficult to estimate the dose they will deliver.

4.1.1. Entry of Radionuclides into the Body

Radioactive materials may occur in many physical or chemical forms just as other materials do. They may appear as solids, powders, dusts, liquids, gases, vapours or solutions. Internal contamination can result from the careless handling of such radioactive materials. It may enter the body in three different ways:

- Inhalation
- Ingestion
- Absorption through unbroken skin or through wounds

Inhaled material cleared from the lungs often enters the gastrointestinal tract (GI tract) and then a secondary ingestion type of exposure occurs. The behaviour of inhaled radionuclides in the lungs depends on whether they are soluble or insoluble in lung fluids.

The term used by the International Commission on Radiological Protection (ICRP) to describe solubility in body fluids is “transportability.” Transportable radionuclides will readily enter the blood stream from the lung and deposit in body organs. Elimination from the body is mainly through urinary excretion.

For non-transportable radionuclides, the lung is usually the target organ because it retains these radionuclides for a long time. A small and uncertain fraction is eliminated in the urine, which means that urine analysis is unreliable as a useful bioassay technique for these radionuclides. The greater part of such non-transportable radionuclides is slowly expelled up the respiratory passage by ciliary action, subsequently swallowed and excreted in the faeces.

The amount of radionuclide taken up by the body depends on the magnitude of the intake, solubility in body fluids (which in turn depends on the particular chemical form of the radionuclide), and whether the intake is via inhalation, ingestion or absorption through the skin.

Transportable materials which are inhaled or ingested go largely to the blood stream, while non-transportable materials irradiate the respiratory tract or the gastrointestinal tract.

4.1.2. Distribution of Radionuclides in the Body

A large percentage of radionuclides that enter the body are eliminated in the first few days. However, a portion will be absorbed in various organs depending on the type of radionuclide. The body deals with elements and compounds on a chemical basis. For example, normal inactive iodine (I-127) concentrates in the thyroid gland. If radioactive I-131 is taken in, the body cannot differentiate between the two isotopes because they are chemically identical, and the radioactive I-131 will also concentrate in the thyroid.

Some elements are so closely chemically related that the body cannot always differentiate effectively between two different elements. For example, chemists say that calcium, strontium, barium and radium are in the same group. Calcium present in the body is largely deposited in the bone, and any radioisotopes of strontium, barium and radium that enter the body will, therefore, also be collected to a considerable extent in the bone. Such radioisotopes (called bone-seekers) are excreted at a very slow rate once they have been deposited in the bone. If their radioactive half-life is long, they may, therefore, irradiate the sensitive bone marrow, as well as the bone, for many years.

The tissue or organ in which radioactive material is absorbed is known as *target tissue*. For exposures to radioactive iodine, the main target tissue would be the thyroid gland; for inhaled strontium it is the lung, but for ingested strontium it is the bone.

4.1.3. Elimination of Radionuclides from the Body

In addition to the tendency for a particular element to be taken up by a particular organ or tissue, the main consideration in determining the hazard of a given radioisotope inside the body is the total radiation dose delivered to the target tissue. The most important factors determining this dose are:

- The amount of radioactive material deposited
- The length of time for which it is active in the body
- The type and energy of the radiation emitted

The time depends on two factors: one is the *radioactive half-life*, T_r**,** the other is the *biological half-life*, T_b.

The biological half-life is the time taken for the amount of a particular element in the body to decrease to half its initial value due to elimination by biological processes alone. The biological half-life depends on the rate at which the body normally uses a particular compound of an element.

The combination of the radioactive half-life and biological half-life gives rise to the *effective half-life*, T_e. The effective half-life is the time taken for the amount of a specified radionuclide in the body to decrease to half its initial value as a result of both radioactive decay and biological elimination.

The effective half-life, T_e, is given by the equation:

$$\frac{1}{T_e} = \frac{1}{T_r} + \frac{1}{T_b} \Rightarrow T_e = \frac{T_r T_b}{T_r + T_b}$$

Serious internal hazards are presented by those radionuclides that have long effective half-lives, such as Ra-226 and Pu-239 with effective half-lives of 45 and 100 years respectively. Once deposited in bone, they remain there essentially unchanged in amount during the lifetime of the individual. The continued action of the emitted alpha particles can cause significant injury because they deposit their energy over a period of years in a limited region.

4.2. Bioassay

Bioassay is the determination of the type, quantity and location of radioactive material in the body by direct measurement or by analysis of materials excreted from the body. There are two methods:

- Excretion Analysis (urine and faeces)
- Whole-Body Counting

4.2.1. Urine Analysis

A fraction of the body content of a radionuclide is excreted in urine each day. By analyzing the radionuclide content of urine, we can estimate the body content for some radionuclides if we know the fraction that is excreted each day. Unfortunately, this fraction is usually not well known. The main problem with urine analysis is that only transportable radionuclides are excreted in substantial amounts and in a predictable manner in urine.

Many inhaled radionuclides are non-transportable. For these, a very small and highly variable daily fraction reaches the blood stream from the lung. This makes it very unreliable and sometimes impossible to estimate body contents from urine analysis. However, it is possible to confirm the presence of the non-transportable radionuclides and to identify them.

Tritium represents a special case because it is uniformly distributed in body water. The concentration in urine is the same as in all the other body fluids, so the daily dose rate and dose commitment may be estimated directly from the urine concentration.

Tritium-in-urine analysis is done by liquid scintillation counting. The normal sampling period for people routinely exposed (i.e., those with a committed dose greater than 0.01 mSv) is one week. This rather short sampling period is necessary because the effective half-life of tritium is only 10 days.

Tritium-in-urine analysis is advantageous because the presence of other beta-gamma emitters in urine can also be detected. Tritium has a rather low beta energy (E_{max} = 18 keV), and spare channels of the pulse height analyzer in the liquid scintillation counter are used to monitor the beta energy region above 18 keV.

The presence of transportable radionuclides such as C-14, I-131, Sr-90, Cs-137 and Cs-134 is looked for in these “high energy” channels. If anything other than tritium is detected, a whole-body count will likely be done.

On some occasions, large volume urine samples may be analyzed by gamma spectrometry or radiochemistry to identify and estimate radionuclide contents with much better sensitivity. But even so, this will still only measure what comes out of the body, not what's in it. Radionuclides that are excreted very slowly can be very hard to detect.

4.2.2. Faecal Analysis

Faecal analysis is more useful for ingested material. Inhaled transportable nuclides will largely be eliminated in urine; inhaled non-transportable nuclides will be deposited in the lung. From the lung they are eventually brought up as phlegm to be swallowed and show up in faeces.

Normally, it is much better to look for non-transportable radionuclides in the lungs with a whole-body counter. Faecal analysis is superior when the gamma energy is so low that one cannot detect the gamma emitter in the lungs, or when it is suspected that non-transportable pure beta emitters are present (such as C-14 particulates).

4.2.3. Whole-Body Counting

In whole-body counting the amounts of gamma emitting radionuclides in the whole body are measured directly using a sodium iodide scintillation detector. To reduce the background radiation and hence increase the sensitivity of the measurement, both the detector and the subject are shielded, usually with lead.

Three versions of a whole-body counter are common: a scanning type, a chair type, and a stand-up type.

In the older scanning type, the person to be monitored lies on a motor-driven bed which passes slowly under a single large sodium iodide detector enclosed in a lead-shielded housing. The time for a complete scan from head to toe is typically around 10 minutes. The counting data are analyzed by gamma spectrometry.

A chair counter has three detectors; one looks at the thyroid, one at the lungs and one at the abdomen. The subject sits in the chair, and the counts seen by each detector are accumulated in separate spectra. All the data are processed by computer. After a 5 minute count, radionuclides can normally be identified and their quantities in the thyroid, lung and abdomen assessed often to much better than 1% of the quantity that would cause one to be irradiated to the annual dose limits. If activity is detected, the results from the whole-body counter are used to calculate the doses to the target tissue. These doses are then entered into the individual's dose records.

The stand-up type of body counter is available with sodium iodide or germanium detectors. The subject stands in front of the shielded detectors for a minute and then the computer analyses the spectrum and produces the results.

4.2.3.1. Pros and Cons of Whole-Body Counting

The advantages of whole-body counting are:

- Whole-body counting measures body contents directly, as contrasted with indirect inferences from excreta measurements (urine and faecal analyses).

- Non-transportable radionuclides in the lung do not pose a problem for whole-body counting as they do for urine analysis, provided that the gamma photons are sufficiently energetic to be able to reach the detector.

The disadvantages of whole-body counting are:

- It cannot be used in general for radionuclides that do not emit gamma radiation (e.g. pure beta emitters).

The whole-body counter cannot distinguish between radioactivity inside the body (genuine internal contamination) and external contamination on the skin surface or in hair. This can lead to counting results that somewhat overestimate the amount of internal contamination. Also, it is difficult to calibrate a whole-body counter because of differences in organ sites, body size and shape, etc. However, a whole-body counter has high sensitivity so that negative results (i.e., no counts above background) definitely indicate the absence of internal contamination.

5.0 Contamination Control

Radioactive contamination is the presence of radioactive material in any place where it is not desired (or known to be there). It can be in solid, liquid or gaseous form.

5.1. Radiation and Contamination Surveys

A *survey* is a systematic set of measurements made in order to determine one or more of the following:

- An unknown radiation source;
- Dose rate;
- Surface contamination; or
- Atmospheric contamination.

In order to make these determinations, the surveyor must choose the appropriate instruments and must use them properly.

5.1.1. Choosing an Appropriate Instrument

The choice of a radiation surveying instrument for a specific application depends on a number of factors. Some general requirements include portability, mechanical ruggedness, ease of use and reading, ease of servicing, ease of decontamination, and reliability. In addition to these general requirements, radiation survey instruments must be calibrated for the radiation that they are designed to measure, and they must have certain other characteristics that are discussed below.

Ability to respond to the radiation being measured

This point can be clarified with a practical example: a commonly used side window beta-gamma probe has a window thickness of 30 mg/cm^2. This probe would be worse than useless if one wished to survey for low-energy beta radiation, such as ^{14}C or ^{35}S, or for an alpha contaminant such as ^{210}Po. Each of these radioisotopes emits only radiation whose range is less than 30 mg/cm^2. Thus, the radiation cannot penetrate the window of the probe.

Incorrect use of this probe, therefore, may falsely indicate safe conditions when, in fact, there may be severe contamination. Similarly, incorrect inferences may be drawn if a neutron monitor is used to measure gamma radiation or if an instrument designed to measure gamma rays is used for neutrons. It is essential that radiation survey instruments be used only for the radiations they are designed to measure.

Sensitivity

The instrument must be sufficiently sensitive to measure radiation at the desired level. Thus, an instrument to be used in a search for a lost radium needle should be more sensitive than a survey meter used to measure the radiation levels inside the shielding of an accelerator.

In the latter case, where the radiation levels may reach hundreds of milligrays per hour, an ionization chamber whose sensitivity is about 0.01 mGy/h is suitable.

In searching for the lost radium needle, on the other hand, a sensitivity of 0.01 mGy/h would greatly limit the area that could be covered in the search; a Geiger counter survey meter that has a sensitivity of about 0.5 µGy/h is much more useful. For example, if a 1-mg radium needle were lost, the distance within which it could be detected with the ionization chamber is about 90 cm, while the Geiger counter will respond to the lost radium at a distance of 400 cm. The Geiger counter can thus cover a search area about 20 times greater than the ionization chamber.

Too great a sensitivity, on the other hand, may be equally undesirable. The range of radiation levels over which the instrument is to be used should be matched by the range of radiation levels for which the instrument is designed.

Response Time

The response time of a survey instrument may be defined as the time required for the instrument to attain 63% of its final reading in any radiation field. This time is determined by the product of the input capacitance (in farads) of the detector and the shunting resistance (in ohms) across the detector. The response time is usually expressed in seconds.

A low value for the response time means an instrument that responds to rapid changes in radiation level - such as would be experienced when passing the probe rapidly over a small area of contamination on a bench top or over a small crack in a radiation shield.

A fast response time, however, may mean a decrease in sensitivity. Furthermore, a fast response time may result in rapid fluctuations of the meter reading, thus making it difficult to obtain an average level. In practice, the response time of a survey instrument is designed to optimize these divergent factors. Many instruments offer a range of response times, the appropriate one being selected by the surveyor, who turns the time constant selector switch to the desired value.

Energy Dependence

Most radiation measuring instruments have a limited span of energy over which the radiation dose rate is accurately measured. One of the figures of merit of a radiation dosimeter is the energy range over which the instrument is useful. This information must be known by the health physicist in order to choose a proper instrument for a particular application or to interpret the measurements properly. The energy dependence is usually specified by the manufacturer as “Accurate to ±10% of the true value from 80 keV to 2 MeV” or by means of an energy dependence curve.

5.1.2. Surface Contamination

If radioactive material has been deposited on surfaces (such as walls, floors, benches) it is called *surface contamination*. It may be loosely deposited, much like ordinary dust, or it may be quite firmly fixed by chemical action. This distinction is important, and we classify surface contamination on the basis of how easily it can be removed:

Loose contamination

This is surface contamination that can easily be removed with simple decontamination methods. Removable surface contamination is of great concern as it is transferable to other surfaces. This can result in wide spread contamination and may lead to internal contamination of workers.

There are two ways in which loose contamination can enter the body. One is by transfer to exposed skin from which it may find its way into the body. The other is by inhalation as a result of *resuspension*. This means that the work activity and traffic in the area has stirred up the loose contamination so that some of it becomes airborne and can be inhaled.

Radioactive surface contamination is primarily caused by poor handling techniques and housekeeping. Accidental spills or leaks of radioactive material are another source of surface contamination.

Fixed contamination

Surface contamination that is quite firmly attached to the surface and cannot be removed by normal housekeeping methods is called fixed contamination. Depending on the radioisotope and activity, fixed contamination may pose an external radiation hazard. When contamination is significant, the equipment with contaminated surfaces may have to be disposed of or, if the radionuclide has a short half life, it may be stored until the radiation field is no longer a problem.

5.1.2.1. Measuring Surface Contamination

There are two methods used for monitoring surface contamination. One is the *Direct Method*, which measures the activity of fixed and loose surface contamination, and the other is the *Indirect* or *Smear Method*, which measures loose contamination only.

Since the levels of contamination that we are trying to measure are usually quite low, sensitive instruments are required. For beta-gamma contamination measurements, the “pancake” Geiger detector may be used. It has a very thin mica window (0.01 mm), which most beta particles are able to penetrate. The counting efficiency of the pancake Geiger detectors depends on how close to the contaminated surface you hold the detector.

5.1.2.1.1. The Direct Method

The following procedure is used to monitor for surface contamination directly:

i. Monitor locations marked on a plan of the working area by slowly passing the detector over each area.
ii. Keep the detector face towards the surface being monitored and keep the distance between the detector and surface as small as possible without touching (and possibly contaminating) the detector.
iii. If contamination is detected, stop and obtain a measurement.
iv. Clean the area until the reading is below the licence criteria.
v. Identify and mark the contaminated area on the plan.
vi. Record the highest measurement for each area and the final measurement after decontamination.

Disadvantages of the Direct Method:

i. It isn't possible to make a direct measurement of surface contamination when a high gamma background is present. This would completely swamp any increase in count rate from the contamination.
ii. A direct measurement cannot tell you whether the contamination is fixed or loose.

5.1.2.1.2. Indirect Method – Smears

A smear survey is a systematic series of smears used to detect transmissible (loose) contamination. A smear test consists of wiping the suspected area with a piece of filter paper several centimetres in diameter and then measuring the activity in the paper. It is often done in a work area that is subject to contamination, where the background due to radiation sources is high enough to mask the activity due to contamination, or when detection with a survey meter is difficult, as is the case with ^{3}H, ^{14}C, or ^{35}S.

It should be emphasized that a smear test is a qualitative or at best a semi-quantitative determination whose chief purpose is to allow an estimate to be made of the degree to which surface contamination is fixed.

The method for taking a smear is as follows:

i. Select the surface to be smeared.
ii. Number or otherwise identify the smear paper to be used on each surface.
iii. Lightly moisten the smear paper with alcohol or water. Hold the smear on the edge with your thumb and index finger and rub it lightly over the surface.
iv. Smear an area of 100 cm^2.
v. Put the smears in separate envelopes, or use some other method to separate them so that they don't cross-contaminate each other.

vi. Take them to a *Smear Counting Station* that is located in a low-background area. Smears may be counted using either ratemeters or scalers.
vii. Check that the smear holder is empty and measure the background count accumulated in 60 seconds.
viii. Insert the smear into the shielded smear holder (dirty side up), and restart the counter to measure the count accumulated in 60 seconds.
ix. The difference in counts is due to activity picked up on the smear.
x. Remove the smear from the smear holder and discard it.
xi. Clean any contaminated areas and re-monitor. Record results before and after cleaning (decontamination).

<u>Advantages of Smears</u>

i They measure loose contamination only, not fixed plus loose.
ii They are not plagued by high background problems because the smear counting stations are in low background areas. In addition, the smear holder is shielded to reduce the background even further.

<u>Disadvantages of Smears</u>

i They disintegrate on rough surfaces.
ii This method assumes that an area of about 100 cm^2 is smeared. This inaccuracy is rarely a problem in practice because usually we just want to know whether loose contamination is present or not. The smear method will certainly tell you that.
iii If you monitor wet surfaces for loose contamination, the smears should be dried before they are counted. Water on the smear will absorb a lot of beta particles, so that they can't reach the detector. This will cause a serious underestimate of the loose contamination.
iv A smear taken from a very highly contaminated area could give rise to such a high count rate that dead time losses in the detector would cause the indicated count to be low. For practical purposes, you can ignore dead time losses if the count is less than 100,000 in 60 seconds.

5.1.2.1.3. Indirect Method – Masslinn Mop

This method is used to look for loose contamination on large areas, like floors. The Masslinn Mop is a foam-covered, plastic mop head swivel-mounted to a broom handle. A clean Masslinn dust cloth is wrapped around the mop head and then large sections of floor (typically 3 m^2 or so) are mopped with individual dust cloths. All cloths are labelled and taken to a low background area where they are checked for contamination.

This is a very quick method of finding loose contamination. For surveying areas where you would not expect loose contamination, you could smear the whole area with one cloth. If you haven't picked up any contamination, you know that the entire area is clean. On the other hand, if you find something, the area must be divided into sectors and the procedure described above followed.

5.1.2.1.4. Relating Measurement Readings to Regulatory Criteria

The readings from radiation contamination meters and non-portable instruments can be related to regulatory criteria if the efficiency of the instrument for a specific radioisotope is known. Instrument efficiencies for specific radioisotopes can be obtained from the manufacturer or determined using an appropriate standard radioactive source of known activity.

For mixtures of radioisotopes, do all calculations using the radioisotope for which the instrument has the lowest detection efficiency. Using the following equation, calculate the measurement results in $\mathrm{Bq/cm^2}$

$$\text{Removable Activity} = \frac{N - NB}{E \times 60 \times A \times F}$$

Where:

N = the total count rate in counts per minute (cpm) measured directly or on the wipe

NB = the normal background count rate (in cpm) from the survey instrument or on the blank

E = the instrument efficiency factor, expressed as a decimal, i.e., for 26% efficiency, E = 0.26 for the radioisotope being measured.

60 = sec/min

A = area wiped (not to exceed 100 $\mathrm{cm^2}$) or area of the detector in $\mathrm{cm^2}$ (for direct measurement)

F = the collection factor for the wipe (used ONLY when calculating indirect wipe monitoring results). If F is not determined experimentally, a value of F = 0.1 (i.e. 10%) shall be used.

Consult the manufacturer to find out the instrument efficiency factor or calculate it using a standard radioactive source with a known amount of activity in a counting geometry similar to that used when surveying for contamination. See equation below for how to calculate the instrument's efficiency.

$$\text{Efficiency} = \frac{\text{detector count rate} - \text{background count rate}}{\text{known activity of standard source}}$$

5.1.3. Leak Testing of Sealed Sources

Sealed gamma-ray, beta, bremsstrahlung, and neutron sources are used in a wide variety of applications in medicine and industry. In all cases, the radioactive material is permanently enclosed either in a capsule or another suitable container. Before being shipped from the supplier, all such sources must pass inspection for freedom from surface contamination and leakage.

Either during transport from the supplier or in the course of time, however, the capsule may develop faults through which the radioactive material may escape into the environment. Because of the serious consequences of such an escape, a sealed source must be tested before being put into use and periodically thereafter for surface contamination and leakage. The testing cycle depends on the nature of the source and on the kind of use to which it is put. However, it is usually recommended that such tests be performed at least once every 6 months. The following techniques may be employed to perform these tests:

i Wipe the source with either a piece of wet filter paper or a cotton swab. Repeat at least 7 days later. If less than 200 Bq of activity was wiped off each time, the source is considered free of leaks.

ii For high-activity sources such as those used in teletherapy, where wiping the source might be hazardous, accessible surfaces of the housing port or collimator should be wiped while the source is in the "off" position.

iii Immerse the source in ethanediol and reduce the pressure on the liquid to 100 mmHg for a period of 30 seconds. A leak is indicated if a stream of fine bubbles issues from the source. This method is reliable only for such sources where enough gas would be trapped to produce a stream of fine bubbles.

5.2. Air Sampling

Since no system of containment or control can be 100% effective all of the time, it is inevitable that some radioactive material will be released to the atmosphere during normal operations involving unencapsulated radioactivity. Atmospheric radioactivity is a matter of concern because the inhalation pathway is a major avenue for the entry of contaminants into the body. We take in a larger mass of air than either food or water; the daily intakes by a reference person are only 1.9 kg of food and 2.2 kg of water, but 26 kg of air! Additionally, the area of interface in the lungs - between the body's internal milieu and the outside atmosphere - is 50 to 100 m^2. This large interface facilitates the transfer of noxious agents from the inhaled air into the body fluids. Therefore, if the quantity of radioactivity being handled is great enough to pose a significant inhalation hazard in case of an accidental release of the radioactivity to the air, an air sampling program is required to assess the radioactivity content of the air accurately.

Air sampling programs are implemented in order to meet regulatory requirements, to verify the effectiveness of engineering and administrative methods for control of airborne radioactivity, and to supply data for public information purposes. Sampling strategies are determined by several different factors, including the reasons for sampling; the physical and radiological characteristics of the contaminant; environmental considerations (such as airflow patterns, dust, and radon); operational considerations, including who, when and where to sample; type of sample (area or personal); sample size, and so on.

For example, general area sampling is useful for planning protective measures and detecting releases, but it is not suitable for determining workers' “actual exposure.” To determine actual exposure, we must use breathing zone samples or individual personal samplers (called *lapel samplers*) that are worn by the workers near their noses. These are among the many factors that must be considered in the design of an air sampling program that will yield useful information.

5.3. Contamination Prevention

5.3.1. Personal Protective Clothing

The philosophy of radiation protection advocates the restriction of radiation exposure to levels as far below the recommended maxima as is reasonably achievable. When working with unsealed sources there is a possibility of an accidental spill or release to the environment of radioactivity, so it is customary to require radioisotope workers to wear protective clothing in order to prevent contamination of the skin.

Such protective clothing, whose use may be restricted to the radioactivity area, may include laboratory coats, coveralls, caps, gloves, and shoes or shoe covers. Protective clothing is always assumed to be contaminated and, therefore, must be removed when the worker leaves the radioactivity area. To be most effective, the protective clothing should be designed so that the worker can remove it easily and without transferring contamination from the clothing to his or her skin or to the environment. To this end, the worker should be instructed in the proper sequence of removal of the protective clothing before stepping out of the contaminated area into a clean area. Workers should always be monitored before leaving the radioactivity area.

Protective clothing, by its very nature, must become contaminated; its main function is to intercept radioactivity that would otherwise contaminate the worker's skin or the clothing worn outside the radioactivity area. The degree of allowable contamination on the protective clothing varies with the type of work that the wearer does. For this reason, the degree of contamination permitted on protective clothing is determined by the individual installation.

For most isotope laboratories, the simplest method for dealing with contaminated protective clothing is to rent the protective clothing from a commercial supplier and to return the contaminated clothing to the supplier.

For those installations that do their own laundry, ordinary laundering procedures, using sodium hexametaphosphate or sodium ethylenediaminetetraacetic acid (Na-EDTA) added to the wash water may facilitate the removal of the contaminants. After laundering, the protective clothing should be monitored to ascertain that it has, in fact, been decontaminated to some previously determined limit. If a piece of protective clothing is unusually or very severely contaminated, it may be simpler to dispose of the item as low-level radioactive waste rather than to try to decontaminate it. Unless the wash water meets regulatory requirements for discharge into the sanitary sewer system, it must be treated as low-level radioactive waste.

5.3.2. Respiratory Protection

When a worker is likely to be exposed to airborne radioactivity, respiratory protection must be considered. According to ALARA principle, the effective whole body dose must be minimized. This principle must be considered when a worker is in an area of high radiation and high airborne radioactivity.

Wearing a respirator can offer protection against inhalation of the airborne radioactive materials. However, wearing a respirator decreases the worker's efficiency and increases the time necessary to complete a job by about 20 to 25%. Thus, the decision about whether or not to use respiratory protection depends on the actual levels of atmospheric activity and radiation. These relationships are illustrated by the following example:

Example 8

A worker is in an area where he is exposed to radiation and to an airborne radioactivity concentration of 2.5 DAC[1] (derived air concentration). The estimated time for completion of the job without wearing a respirator is 2 hours. Time and motion studies showed a 20% reduction in worker efficiency if the worker wears a half-mask respirator but is not subject to heat stress. This means a

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[1] 1 DAC.h = 10 μSv

25% increase in time taken to do the job. Calculate the worker's total dose commitment if he works in radiation fields of 50, 90 and 150 μSv/h and either

a) wears a respirator with a PF[2] (protection factor) of 10, or
b) does not wear a respirator

Solution:

For regulatory purposes, exposure to a radioactive atmosphere of 1 DAC for an hour corresponds to a dose of 10 μSv. Therefore, the dose to the worker while wearing a half-mask respirator whose protection factor (PF) is given as 10, in a radiation field of 50 μSv/h is

$$\left(\frac{2.5\,\mathrm{DAC}}{10}\times\frac{2}{0.8}\,\mathrm{h}\times 10\,\frac{\mu\mathrm{Sv}}{\mathrm{DAC}\cdot\mathrm{h}}\right)+\left(50\,\frac{\mu\mathrm{Sv}}{\mathrm{h}}\times\frac{2}{0.8}\,\mathrm{h}\right)=130\,\mu\mathrm{Sv}$$

Without a respirator, his dose is

$$\left(2.5\,\mathrm{DAC}\times 2\mathrm{h}\times 10\,\frac{\mu\mathrm{Sv}}{\mathrm{DAC}\cdot\mathrm{h}}\right)+\left(50\,\frac{\mu\mathrm{Sv}}{\mathrm{h}}\times 2\mathrm{h}\right)=150\,\mu\mathrm{Sv}$$

By performing similar calculations for the cases where the worker is in external radiation fields of 90 and 150 μSv/h, we find the total doses to be:

Table 1: Radiation doses with and without the use of a respirator.

Radiation Field (μSv/h)	Dose (μSv)	
	With Respirator	Without Respirator
50	130	150
90	230	230
150	350	380

The example above shows that the decision regarding the use of a respirator is governed by the relationship between the dose when wearing a respirator and when not wearing a respirator. Keep in mind that wearing the respirator increases the amount of time spent doing the job.

5.3.2.1. Medical Assessment

It must be emphasized that a worker must be medically approved for respirator use before being allowed to put on a respirator or being fitted for one. Wearing a

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[2] Each respirator has a protection factor, this is the factor by which the DAC is reduced.

respirator effectively increases the volume of the upper respiratory tract, thereby decreasing the volume of air that reaches the deep respiratory tract (where gas exchange occurs).

To compensate for this decreased air supply, the body's homeostatic mechanisms increase the respiratory rate and the rate of blood flow through the lungs. These rate increases lead to increased demands on the heart muscle. If the worker's cardiovascular and respiratory systems are in good health, then these increased cardiac demands are safely met. On the other hand, if the worker has an impaired cardiovascular system, the heart may not be capable of meeting this increased demand and a heart attack may ensue.

For this reason, a worker who may have to wear a respirator on the job must be tested and approved for respirator use by a qualified physician before he or she is allowed to use a respirator.

5.3.2.2. Respiratory Protective Devices

The exact type of respiratory protective device that may be required depends on the nature of the airborne contaminant. It must be emphasized that respiratory protective devices may be used only for those hazards for which they are designed. Half-mask or full-mask facepieces must not leak and must fit properly. Accordingly, the wearer of a respirator must be fitness-tested before a respirator is assigned.

Air-purifying respirators remove a contaminant, either by use of a filter for aerosols or by chemical cartridges that remove gases. Because of the specific action of the chemical agents on the contaminant, different canisters must be used for different gases. For this reason, gas masks are not recommended for use against radioactive gases.

Supplied air respiratory protective devices may be used against either or both radioactive gases and radioactive aerosols. In this category of protective device, there are two subcategories.

Firstly, there are air line hoods, which utilize uncontaminated air under positive (with respect to the atmosphere) pressure supplied from a remote source. Secondly, there are self-contained breathing apparatus (SCBA), in which breathing air is supplied either from a bottle carried by the user or from a canister containing oxygen-generating chemicals.

The advantage of the supplied air devices is that the pressure in the breathing zone is higher than atmospheric pressure. As a consequence, leakage is from the inside out. When using a supplied air device, it is imperative that the time limitation on the air supply is known.

5.4. Regulatory Limits

5.4.1. Contamination Control Limits

Contamination can be measured and is expressed in terms of *contamination control limits* or CCLs.

$$1 \text{ CCL} = 50 \text{ kBq/m}^2$$

Basically, contamination levels below 1 CCL aren't much of a problem, but above 1 CCL we want to impose some controls. 1 CCL of contamination gives about 15 μSv/h to the underlying skin. Activity at the 1 CCL level is low enough to present a minimal radiation hazard, but yet high enough to be detectable with portable contamination meters.

5.4.2. Where no Licensed Activity is being Carried on

According to the CNSC's *General Nuclear Safety and Control Regulations,* the prescribed limit of contamination for a place or vehicle where no licensed activity is being carried on is any quantity of a radioactive nuclear substance that may, based on the circumstances, increase a person's effective dose by 1 mSv or more per year in excess of the background for the place or vehicle.

5.4.3. CNSC Radioisotope Licence Criteria for Radioactive Contamination

The licensee shall ensure that:

i On all normally accessible working surfaces in any location where a radioactive prescribed substance is used or stored, loose contamination does not exceed 0.5 Bq/cm^2 of alpha activity or 5 Bq/cm^2 of beta or gamma activity, averaged over an area not exceeding 100 cm^2.

ii On all other surfaces, and prior to decommissioning any location where a radioactive substance has been used or stored, loose contamination does not exceed 0.05 Bq/cm^2 of alpha activity or 0.5 Bq/cm^2 of beta or gamma activity, averaged over an area not exceeding 100 cm^2.

iii The dose rate due to fixed contamination does not exceed 0.5 μSv/h at 0.5 metres from any surface; and

iv Records of all contamination measurements shall be maintained for at least three years.

5.5. Decontamination Techniques

The methods for removal of radioactive contamination are similar to methods used to remove ordinary dirt, but the required degree of cleanliness is much greater. Small amounts of contamination, which would be completely negligible from an ordinary cleanliness point of view, may still present a radiation hazard.

Apart from this distinction, the processes used in decontamination are the same as those used for industrial cleaning. Namely, soil or contamination is removed by chemicals such as detergents, or by physically removing the contaminated surface (e.g., by abrasives).

Contamination on the surface of an object may be attached either loosely or firmly, hence the terms loose and fixed. The contamination may be attached chemically or it may be mechanically trapped in cracks or small holes. The cleaning process must break the union or bond between the surface and the contamination. The contaminant must then be removed from the surface and prevented from re-attaching itself. When decontaminating a large area, it is usually best to start from the outside and work your way to the middle.

5.5.1. Chemical Decontamination

The preferred method for most decontamination operations is to clean with water to which one or more suitable chemical cleaning agents have been added.

<u>Detergents</u>

The cleaning agents must be capable of removing the contamination from the contaminated surface and then holding it in solution. Various chemical compounds are available for special purposes but, generally speaking, commercial detergents work just as well.

<u>Chemical Solvents</u>

Chemical solvents are normally used only after efforts with detergent solutions have failed. Varsol, trichloroethylene (used in dry-cleaning), and paint removers have all been used successfully. The problem is that they are often toxic and flammable.

<u>Acids</u>

Application of acids to surfaces is an excellent way of removing contamination, but it has limited usefulness because of the damage that may result due to the corrosive action of the acids. Acids should be used with caution.

Alkalis

Alkalis such as caustic soda solutions are occasionally used as decontamination agents. These are particularly useful for washing rubber articles such as gloves or rubber overshoes.

5.5.2. Physical Decontamination

There are decontamination methods in which the outer layer of the contaminated surface is removed by physical force. Such methods are effective, but they are somewhat crude and destructive, and it may not be possible to use them on delicate objects.

Some physical decontamination techniques are:

- Scrubbing with abrasive powder like Ajax or with steel wool.
- Sand-blasting.
- Cleaning with high pressure steam jets.
- Vacuum cleaning.
- Ultrasonic cleaning.

These methods have particular applications. A method useful for certain objects may be unsuitable for others. All these techniques are useful but the most common ones are washing with detergent by hand and scrubbing.

Scrubbing and Sand-Blasting

The waste suds or sand will be contaminated and must be treated as radioactive waste.

Steam Cleaning

This is usually used on large decontamination jobs. One advantage is that the residue (condensed steam) has a much smaller volume than would be left if high pressure hot water had been used instead. Therefore, it is easier to dispose of as radioactive waste.

Vacuum Cleaning

In principle, this is the same method as is used to clean a carpet. However, the dust in the dust bag will be radioactive. This bag should be treated as a radioactive waste. It may be very highly radioactive after removing high level surface contamination. It is possible for vacuum cleaners to create radiation and airborne hazards.

Ultrasonic Cleaning

This involves the use of a special cleaning tank and ultrasonic (i.e. very high frequency) sound. The sound waves bombard the surface of dirty equipment and tear the dirt from it. The cleaning tank is filled with a suitable decontamination solution to prevent re-deposition of the contamination on the equipment.

5.5.3. Decontamination of Skin

People doing radioactive work sometimes get contamination on their skin, especially on their hands. However, contamination of the face and hair also happens.

For hand decontamination, a good wash with soap and warm water is generally all that's needed. (hot water opens the pores and you might drive the contamination deeper. Cold water closes the pores and you may not get the contamination out.) This should remove all loose contamination and reduce the residual contamination to below 1 CCL.

If contamination is under your fingernails, a soft bristle brush should be used with soap and warm water. You should be careful not to scrub or rub the skin to the point where it reddens because there is a risk of the contamination entering your blood stream directly.

If soap and water aren't effective, use a chemical hand cleanser. This skin cleanser contains small plastic granules to help loosen contamination. This step may be repeated once, but only if there is no evidence of skin damage from the decontamination.

For decontamination of the face and hair, showering once or twice is the most effective. Keep your mouth and eyes shut while doing this.

If you have minor cuts or scratches you should be very sure that:

- They are covered with surgical dressings, and
- You do not work in areas where you are likely to contaminate yourself.

Glossary

ALARA	As Low As Reasonably Achievable.
Bioassay	The determination of the type, quantity and location of radioactive material in the body by direct measurement or by analysis of materials excreted from the body.
Contamination	The presence of radioactive material in any place where it is not desired.
Half-life, biological	The time taken for the amount of a particular element in the body to decrease to half its initial value due to elimination by biological processes alone.
Half-life, effective	The time taken for the amount of a specified radionuclide in the body to decrease to half its initial value as a result of both radioactive decay and biological elimination.
Half-life, radioactive	The time taken for the amount of a particular radionuclide to decrease to half its initial amount due to radioactive decay.
Half-value layer (HVL)	The thickness of *shielding* that is required to reduce gamma radiation to half its former intensity. It varies with material and with energy of the gamma radiation.
Radiation, external	Radiation from sources which are outside the body.
Radiation, internal	Radiation from sources which are inside the body.
Shielding	Any material or obstruction that absorbs radiation and thus tends to protect personnel or materials from the effect of ionizing radiation.
Survey	A systematic set of measurements made in order to determine an unknown radiation source; dose rate; surface contamination or atmospheric contamination.
Whole-body counting	Direct measurement of the amounts of gamma-emitting radionuclides in the body using a sodium iodide scintillation detector.

Appendix I

Table of Specific Gamma Constants for Various Radioisotopes

Nuclide	τ*	Nuclide	τ*	Nuclide	τ*
Actinium-227	~2.2	Gold-198	2.3	Potassium-43	5.6
Antimony-122	2.4	Gold-199	~0.9	Radium-226	8.25
Antimony-124	9.8	Hafnium-175	~2.1	Radium-228	~5.1
Antimony-125	~2.7	Hafnium-181	~3.1	Rhenium-186	~0.2
Arsenic-72	10.1	Indium-114m	~0.2	Rubidium-86	0.5
Arsenic-74	4.4	Iodine-124	7.2	Ruthenium-106	1.7
Arsenic-76	2.4	Iodine-125	~0.7	Scandium-46	10.9
Barium-131	~3.0	Iodine-126	2.5	Scandium-47	0.56
Barium-133	~2.4	Iodine-130	12.2	Selenium-75	2
Barium-140	12.4	Iodine-131	2.2	Silver-110m	14.3
Beryllium-7	~0.3	Iodine-132	11.8	Silver-111	~0.2
Bromine-82	14.6	Iridium-192	4.8	Sodium-22	12
Cadmium-115m	~0.2	Iridium-194	1.5	Sodium-24	18.4
Calcium-47	5.7	Iron-59	6.4	Strontium-85	3
Carbon-11	5.9	Krypton-85	~0.04	Tantalum-182	6.8
Cerium-141	0.35	Lanthanum-140	11.3	Tellurium-121	3.3
Cerium-144	~0.4	Lutecium-177	0.09	Tellurium-132	2.2
Cesium-134	8.7	Magnesium-28	15.7	Thulium-170	0.025
Cesium-137	3.3	Manganese-52	18.6	Tin-113	~1.7
Chlorine-38	8.8	Manganese-54	4.7	Tungsten-185	~0.5
Chromium-51	0.16	Manganese-56	8.3	Tungsten-187	3
Cobalt-56	17.6	Mercury-197	~0.4	Uranium-234	~0.1
Cobalt-57	0.9	Mercury-203	1.3	Vanadium-48	15.6
Cobalt-58	5.5	Molybdenum-99	~1.8	Xenon-133	0.1

Nuclide	τ*	Nuclide	τ*	Nuclide	τ*
Cobalt-60	13.2	Neodymium-147	0.8	Ytterbium-175	0.4
Copper-64	1.2	Nickel-65	~3.1	Yttrium-88	14.1
Europium-152	5.8	Niobium-95	4.2	Yttrium-91	0.01
Europium-154	~6.2	Osmium-191	~0.6	Zinc-65	2.7
Europium-155	~0.3	Palladium-109	0.03	Zirconium-95	4.1
Gallium-67	~1.1	Platinum-197	~0.5		
Gallium-72	11.6	Potassium-42	1.4		

[*] Unit of τ: (R.cm^2) ÷ (h.mCi)

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Training Course

Radiation Safety Officer
(RSO-1)

Operating and Emergency Procedures

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Sixth Revision

Operating and Emergency Procedures

CONTENTS

1.0 Introduction

This module outlines a variety of procedures for work involving radiation and radioisotopes, and for coping with accidents and incidents. They can be expanded and tailored to particular circumstances, as necessary.

2.0 Laboratory Procedures

2.1. General Safety

- Keep unauthorized persons out of the laboratory.
- Keep the laboratory locked when unoccupied.
- Keep external radiation exposure as low as reasonably achievable (ALARA) by using the principles of time, distance and shielding.
- Minimize internal radiation exposure by limiting loose contamination and preventing personal contamination.
- Use the minimum quantity of radioactivity possible.
- Use tongs or other remote-handling equipment where appropriate.
- Dosimeters are to be worn at all times if required by the Radiation Safety Officer.
- Dosimeters must be stored away from sources of radiation.
- In the case of a radioactive spill or accident, follow emergency procedures and notify the Radiation Safety Officer.
- Comply with the Nuclear Safety and Control Regulations, and the conditions of the radioisotope licence.

2.2. Contamination Control

- Do not eat, drink, store food, smoke or apply cosmetics while in the laboratory.
- Do not pipette radioactive materials by mouth.
- Wear a laboratory coat at all times.
- Wear gloves when handling radioactive materials. Change gloves often.
- Never work with unprotected cuts or with breaks in the skin.
- Clearly identify work surfaces used for handling radioactive materials. Ideally, maintain room(s) which are used solely for radioactive work.
- Work over a spill tray lined with absorbent paper.
- Work in a fume hood when working with dry powders or volatile substances.
- Monitor the laboratory for contamination at least weekly. Decontaminate as necessary. Keep a record of monitoring and decontamination results.

- Glassware and other equipment used for radioactive work must be segregated until it has been monitored and, if necessary, decontaminated.
- Remove gloves, wash hands and monitor oneself before leaving an active area.

2.3. Usage, Storage, and Disposal

- Store radioisotopes in a locked room or enclosure.
- Supervise radioisotopes at all times when in use.
- Place radiation safety warning symbols on radioactive containers and at the entrance to storage locations.
- Maintain up-to-date inventory, usage and disposal records of all radioisotopes.
- Liquid waste should be poured carefully into containers containing celite. A dust mask should be worn to ensure that the celite powder is not inhaled. Used containers should be placed in the radioactive-waste bin.
- Use different containers for different isotopes. Each container should be clearly marked with the isotope it contains.
- Solid waste should be deposited in double plastic bags, and the bags sealed and placed in the radioactive-waste bin.
- Store radioactive waste in a secure area. Exhaust waste storage areas when a risk of radioactive vapours or dust exists.

3.0 Emergency Procedures: Spills

3.1. General Precautions

- Alert everyone in the area that a spill has occurred.
- Keep people away from the contaminated area.
- Confine the problem or accident. If the material is a liquid, use some absorbent material to prevent its spread.
- Limit the movement of anyone who may have been contaminated until they have been monitored.
- Summon aid. If first aid or an ambulance is required, summon one.
- If the spill is major, call the Radiation Safety Officer for assistance.

3.2. Classification of Radioactive Spills

A minor spill typically involves less than 100 times the exemption quantity of the radioisotope which has been spilled. For information on exemption quantities refer to Nuclear Substances and Radiation Devices Regulations in the Nuclear Safety and Control Act

A major spill involves more than 100 times the exemption quantity of the radioisotope which has been spilled.

3.3. Minor Spill

- If personal contamination has occurred, deal with it first.
- Wear disposable gloves.
- Ensure that a laboratory coat is worn and properly fastened to prevent contamination of personal clothing.
- If the spill is wet, drop pads of dry, absorbent material on it.
- If the spill is dry, either use dampened pads of absorbent material or dampen the spilled material with either the appropriate organic solvent or water.
- Used pads should be disposed as radioactive waste.
- Use either normal cleaning agents or a special decontamination agent to clean the area of the spill.
- Avoid spreading contamination. Work from the outside of the spill towards the centre.
- Discard all cleaning materials as radioactive waste.
- Wipe test or survey for residual contamination as appropriate.
- Repeat decontamination as necessary.
- Check hands, clothing and shoes for contamination.
- Record the details of the spill and the contamination monitoring results.
- Notify the Radiation Safety Officer.

3.4. Major Spill

- Clear the area. Persons not involved in the spill should leave the immediate area.
- Call the Radiation Safety Officer.
- Limit the movement of anyone who may have been contaminated until they are monitored.
- Close off and secure the spill area to prevent entry. Post warning signs.
- Do not attempt to clean the spill but do take actions to prevent its spread.

3.5. Contaminated Personnel

During skin decontaminations, it is important to proceed from mild treatments to harsher ones only if necessary. Abrasion or any other breaks in the skin must be avoided as these will allow rapid penetration of radioactive material.

- Contact the Radiation Safety Officer.
- Monitor with a survey meter to determine which areas of the body have been contaminated.
- If the skin is intact:
    - Flush the contaminated area with warm water.
    - Apply mild soap or detergent. Lather well with plenty of water.
    - Work lather into contaminated area by rubbing gently for 3 minutes.
    - Take care not to spread the contamination to other areas of the body.
    - Rinse thoroughly.
    - Monitor contaminated area.
    - Repeat washing and rinsing several times, if necessary. A soft brush can be used. Be careful not to scratch the skin.
- In case of minor wounds:
    - Clean the affected area with swabs. Brush away from the wound, taking care not to spread contamination to other parts of the body or into the wound.
    - Wash the contaminated wound with copious amounts of warm water. Encourage *minor* bleeding.
    - Wash wound with mild soap and warm water, as noted above.
    - Monitor contaminated area.
    - After decontamination, apply a first aid dressing.
- In case of serious injuries:
    - The treatment of serious injuries takes priority over all other considerations.
    - Providing assistance to seriously injured persons should not be delayed because of concerns about radioactive contamination.
    - Call 911, advising of the nature of the hazard, the amount of material, the chemical form of the material and any other pertinent information.
    - Direct someone to meet the emergency personnel.
    - Ensure that the casualty cannot be further contaminated by radioactive material.
- If ingestion has occurred, seek medical advice and/or assistance immediately.

3.6. Contaminated Clothing

- If personal clothing or a laboratory coat becomes contaminated by radioactive material, remove it as soon as possible in order to reduce the exposure to radiation.
- Contaminated clothing must be sealed in plastic bags.

- The bags should be labeled with the owner’s name, the isotope and the suspected amount of activity.
- The Radiation Safety Officer should be contacted for further assistance.

4.0 Monitoring for Radioactive Contamination

4.1. General Considerations

Radioactive contamination may be measured directly or indirectly. Direct measurement means the use of portable radiation detection instruments to detect both fixed and removable contamination. Direct measurement may be used when background radiation is negligible compared with licence criteria. Indirect measurement only detects removable contamination by means of wipes.

The choice of detection instrument should be carefully tailored to the radioisotopes of interest. See the appropriate module.

The locations that are to be monitored should be numbered on a plan of the radioisotope work area. These locations should include working surfaces, such as benches, counter tops, fume hoods, etc., storage areas, and non-working surfaces such as floors, instruments and equipment, door handles, light switches, sink taps and telephone receivers. Several random locations should also be monitored. Too rigid a set of locations may overlook problem areas.

Non-portable instruments used for counting wipes, such as liquid scintillation counters, semiconductor gamma spectrometers, etc., should be routinely serviced according to the manufacturer’s instructions. Keep a record of servicing information and dates.

Before monitoring for contamination, portable instruments should be given operation checks as specified by the manufacturer (i.e. battery check, high-voltage check, response check, etc.) and the amount of background radiation should be measured. Record the details. Similarly, non-portable instruments used to count wipes should count and record a blank and a standard source with each set of wipes.

Instruments which are not operating within the parameters of the operational checks or which show anomalous background, blank or standard measurements, should not be used until their proper operation can be verified.

4.2. Direct Measurement with a Portable Meter

- Monitor the locations marked on the plan of the working area by slowly passing the detector over each area.
- Keep the detector face towards the surface being monitored and keep the distance between the detector and surface as small as possible without touching (and possibly contaminating) the detector.
- If contamination is detected, stop and obtain a measurement.
- Clean the area until the instrument measurement is below the licence criteria. A reading in excess of licence criteria after repeated cleaning is an indication of fixed contamination or a high background radiation.
- Identify and mark the contaminated area on the plan.
- Record the highest measurement for each area and the final measurement after decontamination.

4.3. Indirect Measurement with Wipes

- Wipe each of the locations shown on the plan of the working area with a filter paper, wipe or cotton swab lightly moistened with alcohol or water.
- Use one numbered wipe per location. (A screening wipe can be used to monitor several locations. If contamination is found, the contaminated area must be identified and decontaminated.)
- Wipe an area of 100 cm^2. Using uniform and constant pressure, ensure the entire area is wiped.
- If necessary, carefully dry the wipe to prevent loss of activity. (Since the contamination may be absorbed into the wipe material, the use of a wetting agent may lead to a significant underestimate of alpha and low-energy beta contamination with some counting methods.)
- Count the wipes in a low-background area and record all results.
- If the wipes are to be counted on a contamination meter, the wipe should be smaller than or equal to the sensitive area of the detector.
- Clean any contaminated areas and re-monitor. Record results before and after decontamination.

5.0 Identifying and Opening Radioactive Packages

5.1. General Considerations

The packaging and transport of radioactive materials is regulated by the Canadian Nuclear Safety Commission’s Transport and Packaging of Nuclear Substances Regulations. Radioactive material is shipped as “Excepted Packages,” “Low Specific Activity (LSA),” “Type A” or “Type B” packages.

On Excepted Packages, the safety mark “RADIOACTIVE” must be visible on opening the package and the radiation level at any point on the external surface of the package must not exceed 5 μSv/h. All other packages must be classified by radiation level and display the corresponding radiation warning labels as follows:

- I-WHITE: does not exceed 5 μSv/h at any location on external surface of the package.
- II-YELLOW: does not exceed 500 μSv/h at any location on the external surface of the package and the transport index does not exceed 1.
- III-YELLOW: does not exceed 2 mSv/h at any location on the external surface of the package and the transport index does not exceed 10.

The transport index (TI) for a package is the maximum radiation level in microsieverts per hour at one meter from the external surface of the package, divided by ten. Example: 1 μSv/h at 1m gives a TI = 0.1.

Upon receipt, packages containing radioactive material should be delivered to the recipient by placing them on a cart or other device to increase the distance between the transporter and the package in order to minimize radiation exposure. The recipient must, as soon as practical and before opening the package, examine it for damage or leakage. If the package is found to be damaged or leaking, the recipient must comply with Section 19 of the Transport and Packaging of Nuclear Substances Regulations. The package should be contained and isolated to minimize radiation exposure.

5.2. Procedure for Opening Packages Containing Unsealed Sources

- Wear a lab coat and disposable gloves while handling the package.
- If an appropriate survey meter is available, monitor the radiation fields around the package and compare with the units stated on the package. Note any discrepancies.
- If the material is volatile (unbound iodine, tritium, radioactive gases, etc.) or in powder form, place the package in a fume hood.
- Open the outer package and check for possible damage to the contents, broken seals or discoloration of packing materials. If the contents appear to be damaged, isolate the package to prevent further contamination and notify the Radiation Safety Officer.
- If no damage is evident, remove the inner package or primary container, and wipe test the container.
- If contamination is detected, monitor all packaging and, if appropriate, all areas coming into contact with the package for contamination.
- Contain the contamination, decontaminate and dispose in accordance with the conditions of the radioisotope licence.
- Avoid unnecessary direct contact with unshielded containers.

- Verify the radioisotope, the activity and other details with the information on the packing slip and purchase order.
- Log the radioisotope, activity, date received and any anomalies in the inventory record.
- Report any anomalies to the Radiation Safety Officer.

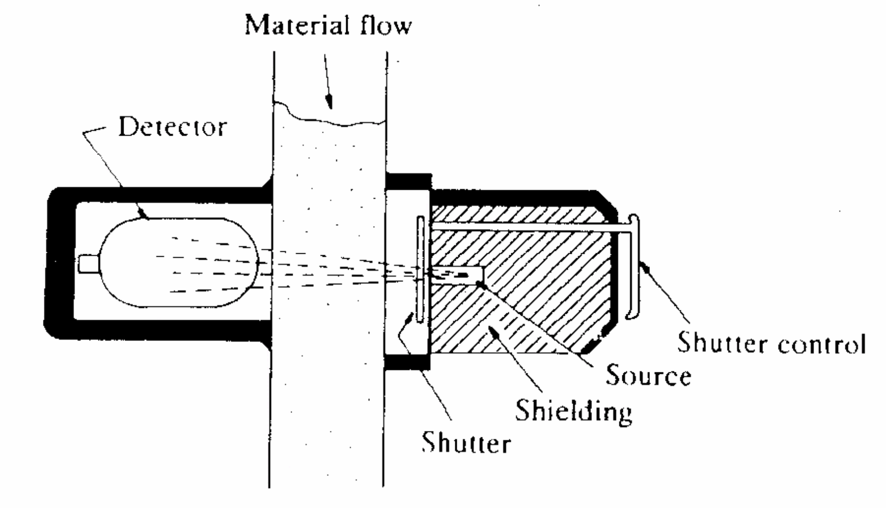
6.0 Nuclear Gauges

6.1. Use of Nuclear Gauges in Industry – Some Examples

- Industrial Radiography
  - Fracture and discontinuities detection of components or parts
- Paper and pulp industry
  - Thickness measurement
  - Moisture measurement
- Oil industry
  - Flow measurement
  - Density measurement
  - Chemical analysis
  - Level indicator
  - Environmental monitoring
  - Oil exploration
- Steel industry
  - Thickness measurement
  - Density measurement
  - Level indicator
- Glass industry
  - Thickness measurement
  - Density measurement
  - Level indicator
- Bottling industry
  - Level indicator
- Rubber industry
  - Thickness measurement
  - Density measurement.
- Chemical industry
  - Flow measurement
  - Chemical analysis
- Construction industry
  - Soil compaction (soil density)
  - Moisture content of soil
  - Density of asphalt in a paving mix

- Hospitals
    - Medical diagnostics
    - Medical treatment.

6.2. Schematic Diagram of a Nuclear Gauge



6.3. Accidents Involving Nuclear Gauges

The following are general procedures to be followed in the event of an accident involving a nuclear gauge. This procedure should also be followed in the event that a gauge is found to be leaking by means of a swipe test.

- Establish a zone with a radius of at least 5 m from the gauge.
- Prohibit entry into the zone.
- If it is necessary to leave the area, post a guard to ensure no-one enters the prohibited zone.
- If the gauge was involved in a road accident, contact the local police.
- Call the Radiation Safety Officer.
- Call the Canadian Nuclear Safety Commission.
- Identify and isolate anyone who may have been exposed until they have been surveyed.
- Record the names of anyone who may have been exposed to radiation.
- Access to the zone will be allowed only when it is deemed safe by the Radiation Safety Officer.
- The Radiation Safety Officer will make a report to the Canadian Nuclear Safety Commission, in accordance with the Regulations.

7.0 Examples of Accidents Involving Radioactive Materials

Kjeller, Norway, 1982

A worker at an industrial radiation facility was exposed in an irradiation room to a 2.4 PBq cobalt-60 source. He died of acute kidney failure with underlying aplasia. The dose he received was estimated to be more than 10 Gy.

France, 1991

Three workers in a linear accelerator unit were irradiated during repeated operations on the machine, in which the electron source was cut off but the accelerator voltage was maintained to save time. The residual dose rate was a few Gy/s. They suffered skin lesions. One of them deteriorated seriously and spent a whole year in hospital.

Indiana, USA, 1992

A cancer patient was given high dose rate brachytherapy treatment with a 160 GBq iridium-192 source. She was returned to her post-treatment unit without the source being removed. Four days later, a nurse threw a catheter discharged by the patient - and containing the source - into the waste. The next day, the patient died. She was estimated to have absorbed 16,000 Gy! As the waste was transported through Ohio, more than 90 people were exposed, although no acute effects were observed. The alarm was raised by a monitoring device at the entrance to a waste processing facility.

Goiâna, Brazil, 1987

A highly soluble caesium-137 chloride source of 50.9 TBq had been left in a teletherapy unit at a derelict hospital. The source assembly was scavenged and an attempt at dismantling it ruptured the source capsule. The assembly was then sold to a scrapyard. The owner noticed that it glowed blue at night, and invited friends and relatives to see the phenomenon. 20 people required emergency hospitalization, 10 of whom had received acute doses between 3 Gy and 7 Gy. The town and its surroundings were not considered acceptably free of contamination until March 1988.

Figure 1: Skin injury from X-ray fluoroscopy, 18-21 months following multiple coronary angiography and angioplasty (From: Radiation Induced skin injuries from fluoroscopy, Thomas B. Shope, US FDA web site)

Figure 2: Skin necrosis 21 days after exposure. (From: How to recognize and initially respond to an accidental radiation injury. IAEA leaflet)

8.0 Duties and Responsibilities of the Radiation Safety Officer (RSO)

The RSO will be responsible for the following tasks as required:

- Periodic inspections and stopping of unsafe work practices.
- Ensuring the proper use of radioactive materials.
- Ensuring the proper use and maintenance of radiation devices.
- Training staff.
- Managing incidents, responses as required, investigations and reporting.
- Controlling security and storage of radioactive materials and radiation devices.
- Providing waste disposal procedures in accordance with conditions of the licence.
- Disposal and decommissioning of nuclear gauges and radiation devices (in cooperation with the manufacturer and the CNSC).
- Maintaining complete records.
- Conducting internal audits.
- Ensuring compliance with all regulatory requirements including CNSC licence conditions, the *Nuclear Safety and Control Act* and all associated regulations.
- Adhering to the principles of ALARA.
- Designating as necessary employees as Nuclear Energy Workers (NEW).
- Providing proper equipment and instrumentation to employees.
- Ensuring that approved procedures are followed and maintained including operating and emergency procedures.
- Maintaining radioisotope licence(s) with the CNSC.

The RSO will be the primary contact person for the CNSC regarding all questions relating to radiation safety and licensing matters.

9.0 A List of Radioisotopes and Examples of Their Uses

Americium-241	Used in many smoke detectors for home and business - to measure levels of lead in dried paint - to ensure uniform thickness in rolling processes for steel and paper - to determine optimum locations for oil well drilling.
Cadmium-109	Used to analyze metal alloys for checking stock or sorting scrap.
Calcium-47	Important aid for biomedical researchers studying cell formation and bone structure in mammals.

Californium-252	Used to inspect airline baggage for hidden explosives - to gauge the moisture content of soil in construction activities - to measure moisture in material stored in silos.
Carbon-14	Used to ensure that new drugs under development are metabolized without forming harmful byproducts - used in archaeology and paleontology for dating historical artefacts.
Caesium-137	Used to treat cancer tumors - to measure correct patient dosage of radioactive pharmaceuticals - to measure and control liquid flow in pipelines - to tell researchers whether oil wells are plugged with sand - to ensure that containers are filled to the proper level with food, drugs or other consumables.
Chromium-51	Used in research on red blood cells.
Cobalt-57	Used in nuclear medicine to help doctors interpret diagnostic scans of patients' organs.
Cobalt-58	Used as a tracer to diagnose pernicious anaemia.
Cobalt-60	Used to sterilize surgical instruments - used to improve safety of industrial fuel-oil burners.
Copper-67	Helps antibodies bind with and destroy cancer tumors.
Curium-244	Used in mining to analyze material excavated from pits or slurries taken from drilling operations.
Iodine-123	Used to diagnose thyroid cancers.
Iodine-129	Used to check radioactivity counters in in-vitro diagnostic testing laboratories.
Iodine-131	Used to test the integrity of pipe welds, boilers and aircraft parts.
Iron-55	Used to analyze electroplating solutions.
Krypton-85	Used in indicator lights in appliances such as washing machines, dryers, stereos and coffeemakers - used to gauge the thickness of thin plastics, sheet metal, rubber, textiles and paper - used to measure dust and pollutant levels.

Nickel-63	Used to detect explosives - used in voltage regulators and current surge protectors in electronic devices.
Phosphorus-32	Used in molecular biology and genetics research.
Plutonium-238	Has been used to power NASA spacecraft since 1972.
Polonium-210	Reduces static discharge in the making of photographic film and phonograph records.
Promethium-147	Used in electric blanket thermostats - to gauge thickness of metals, plastics, textiles and paper.
Radium-226	Makes lightning rods more effective.
Selenium-75	Used in protein studies for life-science research.
Sodium-24	Used to locate leaks in pipelines.
Strontium-85	Used to study bone formation and metabolism.
Technetium-99M	Widely used in radiopharmaceutical studies - many different chemical forms used for organ imaging and blood flow studies.
Thallium-204	Measures pollutant levels on filter paper - used to measure thickness of metal, rubber, textiles and paper.
Thorium-229	Used to make fluorescent lights last longer.
Thorium-230	Provides colour and fluorescence in glazes and glassware.
Tritium	Used for medical studies - for self-luminous watch dials, and commercial and aircraft exit signs - to make luminous paint.
Uranium-234	Used in dental fixtures such as crowns and dentures to provide natural colour.
Uranium-235	Fuel for nuclear power plants - used to produce fluorescent glassware and coloured glazes.
Xenon-133	Used in nuclear medicine for lung ventilation and blood flow studies.

Radiation Safety Institute of Canada

Institut de radioprotection du Canada
National Office, Toronto

Training Course

Radiation Safety Officer
(RSO-1)

Appendix

Sixth Revision

Relevant Web Sites

CANADA

Radiation Safety Institute of Canada
www.radiationsafety.ca

Canadian Nuclear Safety Commission
www.nuclearsafety.gc.ca

Radiation Protection Bureau, Health Canada
www.hc-sc.gc.ca/hecs-sesc/rpb/

Canadian Centre for Occupational Health and Safety (CCOHS)
www.ccohs.ca

Canadian Radiation Protection Association
www.crpa-acrp.ca

Canadian Nuclear Society (CNS)
www.cns-snc.ca

INTERNATIONAL

International Commission on Radiological Protection (ICRP)
www.icrp.org

International Radiation Protection Association (IRPA)
www.irpa.net

International Atomic Energy Agency (IAEA) (UN)
www.iaea.org

World Health Organization (WHO) (UN)
www.who.int/en/
www.who.dk

World Nuclear Association
www.world-nuclear.org

USA

National Institute of Environmental Health Sciences
www.niehs.nih.gov/odhsb/
www.niehs.nih.gov/emfrapid/

Environmental Protection Agency (EPA)
www.epa.gov/radiation/
www.epa.gov/iaq/radon/

Centers for Disease Control and Prevention
www.cdc.gov

National Institute for Occupational Safety and Health (NIOSH)
www.cdc.gov/niosh/
www.cdc.gov/niosh/ocas/

Health Physics Society
www.hps.org

UK

National Radiological Protection Board
www.nrpb.org.uk

AUSTRALIA

Uranium Information Centre
www.uic.com.au

Australasian Radiation Protection Society
www.arps.org.au

A Guide to SI Units in Radiation Protection

EXPOSURE

Roentgen (R), the charge produced in air by x or gamma rays. The SI unit is in terms of coulombs per kilogram of air ($C\ kg^{-1}$).

$1\ R = 2.58 \times 10^{-4}\ C\ kg^{-1}$ $1\ C\ kg^{-1} = 3,876\ R$

KERMA
(Kinetic Energy Released in Material)

The SI unit is the gray.

1 gray (Gy) = 100 rad 1 rad = 0.01 Gy

An exposure of 1 R ($2.58 \times 10^{-4}\ C\ kg^{-1}$) corresponds to an air kerma of about 0.87 rad (8.7 mGy) or a tissue kerma of about 0.97 rad (9.7 mGy).

RADIATION ABSORBED DOSE

The SI unit is the gray (Gy).

1 gray = 100 rad 1 rad = 0.01 Gy

RADIATION DOSE EQUIVALENT

The SI unit is the sievert (Sv).
1 sievert (Sv) = 100 rem 1 rem = 0.01 Sv

ACTIVITY

The SI unit of activity is the becquerel (Bq).

1 becquerel (Bq) = 1 disintegration per second

$1\ Bq = 2.7 \times 10^{-11}$ Curie (Ci)
$1\ Ci = 3.7 \times 10^{10}\ Bq = 37$ Gbq

ADDITIONAL USEFUL CONVERSIONS

1 μCi = 37 kBq
1 mCi = 37 MBq
1 Bq = 27 pCi
370 MBq = 10 mCi
1 μSv = 0.1 mrem

COMMON PREFIXES FOR SI UNITS

Submultiples			*Multiples*		
10^{-3}	milli	m	10^{3}	kilo	k
10^{-6}	micro	μ	10^{6}	mega	M
10^{-9}	nano	n	10^{9}	giga	G
10^{-12}	pico	p	10^{12}	tera	T

SYSTÈME INTERNATIONAL (SI) UNITS

The *rad* (rad) is replaced by the *gray* (Gy)

1 kilorad (krad) = 10 grays (Gy)
1 rad (rad) = 10 milligrays (mGy)
1 millirad (mrad) = 10 micrograys (µGy)
1 microrad (µrad) = 10 nanograys (nGy)

The *gray* (Gy) replaces the *rad* (rad)

1 gray (Gy) = 100 rad (rad)
1 milligray (mGy) = 100 millirad (mrad)
1 microgray (µGy) = 100 microrad (µrad)
1 nanogray (nGy) = 100 nanorad (nrad)

The *rem* (rem) is replaced by the *sievert* (Sv)

1 kilorem (krem) = 10 sieverts (Sv)
1 rem (rem) = 10 millisieverts (mSv)
1 millirem (mrem) = 10 microsieverts (µSv)
1 microrem (µrem) = 10 nanosieverts (nSv)

The *sievert* (Sv) replaces the *rem* (rem)

1 sievert (Sv) = 100 rem (rem)
1 millisievert (mSv) = 100 millirem (mrem)
1 microsievert (µSv) = 100 microrem (µrem)
1 nanosievert (nSv) = 100 nanorem (nrem)

The *curie* (Ci) is replaced by the *becquerel* (Bq)*

1 kilocurie (kCi) = 37 terabecquerels (TBq)
1 curie (Ci) = 37 gigabecquerels (GBq)
1 millicurie (mCi) = 37 megabecquerels (MBq)
1 microcurie (µCi) = 37 kilobecquerels (kBq)
1 nanocurie (nCi) = 37 becquerels (Bq)

The *becquerel* (Bq)* replaces the *curie* (Ci)

1 terabecquerel (TBq) = 27 curies (Ci)
1 gigabecquerel (GBq) = 27 millicuries (mCi)
1 megabecquerel (MBq) = 27 microcuries (µCi)
1 kilobecquerel (kBq) = 27 nanocuries (nCi)
1 becquerel (Bq) = 27 picocuries (pCi)

*1 Bq = 1 disintegration/second = 1 s^{-1}

Table 4.1 Range of Alpha Particles Versus Energy For Al, Ni, Cu, Ag, Ag and Mylar (Taken from Reference 7)

Range in Units of mg/cm^2

Energy (MeV)	Al	Ag	Au	Cu	N	O	Ne	Mylar	He
.050	.142	.371	.746	.479	.123	.132	.149	.097	.081
.080	.190	.487	.976	.525	.170	.181	.204	.133	.113
.128	.252	.634	1.269	.605	.232	.247	.277	.180	.158
.201	.327	.812	1.620	.726	.312	.330	.367	.238	.219
.400	.497	1.199	2.362	.968	.490	.513	.560	.366	.364
.500	.574	1.369	2.675	1.085	.563	.588	.639	.421	.425
.640	.680	1.597	3.086	1.300	.654	.684	.739	.493	.500
.800	.802	1.854	3.539	1.440	.750	.783	.844	.570	.579
1.000	.959	2.178	4.099	1.688	.863	.900	.969	.665	.671
1.60	1.475	3.209	5.825	2.503	1.193	1.246	1.346	.959	.937
2.00	1.86	3.955	7.039	3.107	1.428	1.493	1.620	1.174	1.121
2.401	2.283	4.758	8.323	3.758	1.687	1.765	1.925	1.412	1.320
2.80	2.745	5.622	9.687	4.455	1.979	2.073	2.269	1.678	1.538
3.20	3.244	6.544	11.126	5.196	2.308	2.418	2.655	1.973	1.778
4.000	4.349	8.549	14.205	6.808	3.079	3.227	3.564	2.658	2.331
5.00	5.920	11.339	18.411	9.057	4.236	4.444	4.931	3.682	3.156
6.40	8.460	15.759	24.929	12.610	6.189	6.487	7.222	5.422	4.556
8.00	11.835	21.512	33.237	17.280	8.851	9.277	10.321	7.829	6.504

Table 4.2 dE/dx in $MeV/mg/cm^2$ for Various Materials (Taken from Reference 7)

Energy (MeV)	Al	Ag	Au	Cu	N	O	Ne	Mylar	He
.050	.532	.223	.112	.311	.564	.530	.471	.713	.851
.080	.673	.282	.141	.401	.684	.644	.576	.889	.989
.128	.852	.358	.180	.670	.817	.777	.710	1.107	1.102
.201	1.040	.444	.226	.643	.956	.930	.865	1.346	1.211
.400	1.28	.575	.307	.764	1.303	1.262	1.201	1.757	1.572
.500	1.317	.604	.331	.780	1.442	1.389	1.327	1.888	1.745
.640	1.323	.622	.349	.781	1.604	1.546	1.466	2.021	1.955
.800	1.299	.622	.357	.770	1.734	1.665	1.574	2.096	2.117
1.000	1.248	.610	.357	.740	1.816	1.748	1.632	2.114	2.239
1.60	1.086	.555	.338	.660	1.776	1.688	1.529	1.947	2.237
2.00	.996	.520	.322	.630	1.640	1.561	1.396	1.786	2.112
2.401	.904	.480	.302	.580	1.458	1.385	1.238	1.593	1.926
2.80	.832	.448	.285	.550	1.292	1.227	1.098	1.427	1.756
3.20	.773	.422	.271	.520	1.152	1.098	.982	1.287	1.593
4.00	.682	.380	.250	.460	.949	.900	.802	1.073	1.329
5.00	.598	.340	.228	.410	.792	.756	.672	.895	1.113
6.40	.512	.297	.204	.360	.655	.626	.561	.730	.907
8.00	.442	.262	.183	.320	.556	.530	.479	.610	.750

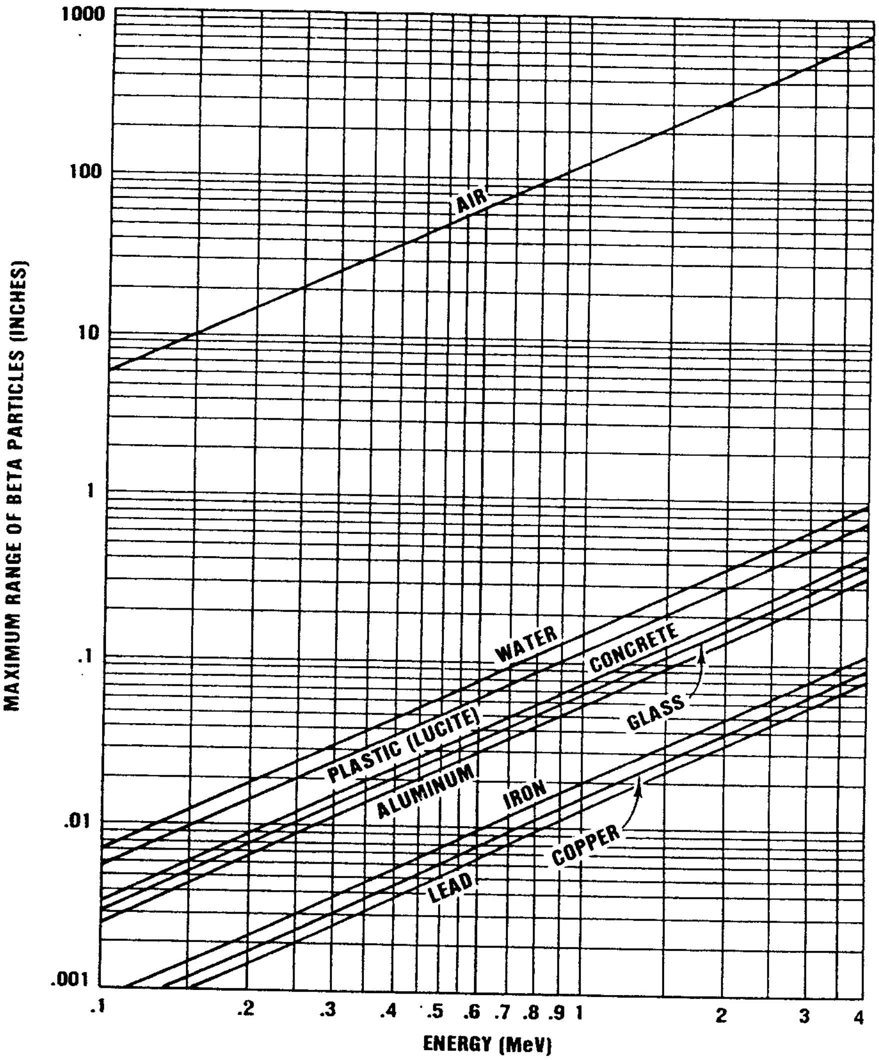


Figure 19.5. Beta ranges in common shielding materials and air. (Taken from the NBS Radiological Health Handbook, now out of print.)

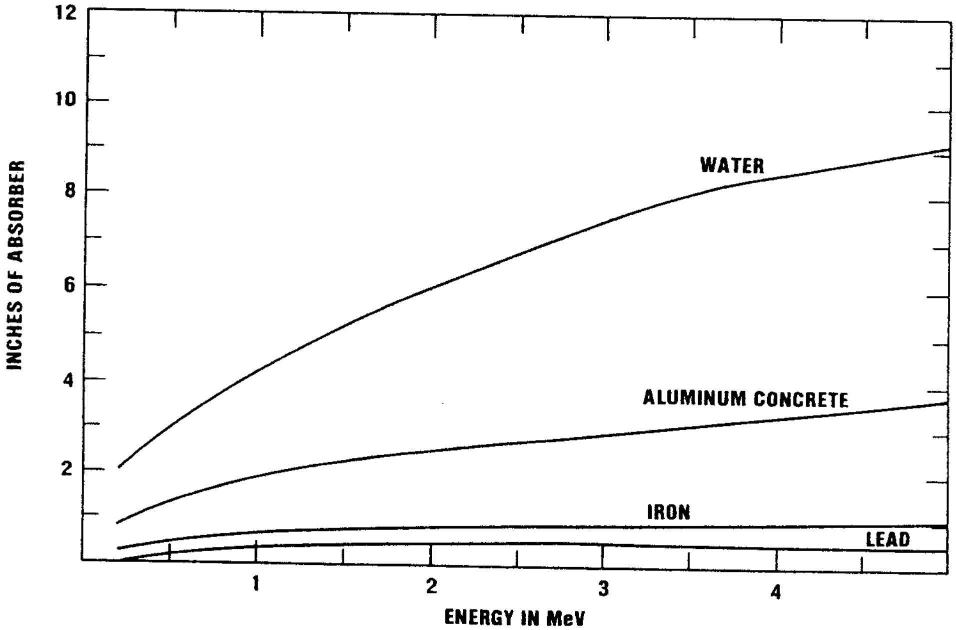


Figure 19.6. Half Value Layers (HVL) for various shielding material for gammas or x-rays.

Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire

INFO-0728-1

BASIC LEVEL

Use of Unsealed Nuclear Substances

This room has been classified as “basic level” for the use of unsealed nuclear substances in accordance with Canadian Nuclear Safety Commission guidelines. Below is a list of safe work practices to be followed when working in this room.

24-hour emergency contact (name and phone number)	Room identification

- Do not eat, drink, store food, or smoke in this room.
- In case of a spill or incident involving a nuclear substance, follow emergency procedures and notify the Radiation Safety Officer.
- Clearly identify work surfaces used for handling nuclear substances.
- Use protective clothing and equipment when working with nuclear substances.
- Check all packages containing nuclear substances for damage upon receipt.
- Store nuclear substances in a locked room or enclosure when not in use.

A room is classified as “basic level” for the use of unsealed nuclear substances when more than one exemption quantity is handled and where the largest quantity (in becquerels) of a substance handled by any worker does not exceed 5 times its corresponding annual limit of intake (in becquerels). Contact your Radiation Safety Officer for a list of annual limits of intake.

For more information, contact: Canadian Nuclear Safety Commission, Directorate of Nuclear Substance Regulation, P.O. Box 1046, Station B, Ottawa, Ontario, K1P 5S9. Telephone: 1-888-229-2672. Facsimile: (613) 995-5086.

Canada

Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire

INTERMEDIATE LEVEL

Use of Unsealed Nuclear Substances

This room has been classified as “intermediate level” for the use of unsealed nuclear substances in accordance with Canadian Nuclear Safety Commission guidelines. Below is a list of safe work practices to be followed when working in this room.

24-hour emergency contact (name and phone number)	Room identification

- Do not eat, drink, store food, or smoke in this room.
- Wear appropriate dosimeter at all times.
- In case of a spill or incident involving a nuclear substance, follow emergency procedures and notify the Radiation Safety Officer.
- Clearly identify work surfaces used for handling nuclear substances.
- Use protective clothing and equipment when working with nuclear substances.
- After working with nuclear substances, monitor work area for contamination.
- Wash hands regularly and monitor them for contamination frequently.
- Check all packages containing nuclear substances for damage upon receipt.
- Store nuclear substances in a locked room or enclosure when not in use.

A room is classified as “intermediate level” for the use of unsealed nuclear substances where the largest quantity (in becquerels) of a substance handled by any worker does not exceed 50 times its corresponding annual limit of intake (in becquerels). Contact your Radiation Safety Officer for a list of annual limits of intake.

For more information, contact: Canadian Nuclear Safety Commission, Directorate of Nuclear Substance Regulation, P.O. Box 1046, Station B, Ottawa, Ontario, K1P 5S9. Telephone: 1-888-229-2672. Facsimile: (613) 995-5086.

Canada

Canadian Nuclear
Safety Commission

Commission canadienne
de sûreté nucléaire

INFO-0728-3

HIGH LEVEL

Use of Unsealed Nuclear Substances

This room has been classified as “high level” for the use of unsealed nuclear substances in accordance with Canadian Nuclear Safety Commission guidelines. Below is a list of safe work practices to be followed when working in this room.

24-hour emergency contact (name and phone number)	Room identification

- Do not eat, drink, store food, or smoke in this room.
- Restrict access to authorized workers only.
- Wear appropriate dosimeter at all times.
- In case of a spill or incident involving a nuclear substance, follow emergency procedures and notify the Radiation Safety Officer.
- Work in a fumehood when required by the Radiation Safety Officer.
- Clearly identify work surfaces used for handling nuclear substances.
- Wear protective clothing and equipment at all times.
- After working with nuclear substances, monitor work area for contamination.
- Wash hands regularly and monitor them for contamination frequently.
- Check all packages containing nuclear substances for damage upon receipt.
- Store nuclear substances in a locked room or enclosure when not in use.

A room is classified as “high level” for the use of unsealed nuclear substances where the largest quantity (in becquerels) of a substance handled by any worker does not exceed 500 times its corresponding annual limit of intake (in becquerels). Contact your Radiation Safety Officer for a list of annual limits of intake.

For more information, contact: Canadian Nuclear Safety Commission, Directorate of Nuclear Substance Regulation, P.O. Box 1046, Station B, Ottawa, Ontario, K1P 5S9. Telephone: 1-888-229-2672. Facsimile: (613) 995-5086.

Canada

Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire

INFO-0728-4

NUCLEAR MEDICINE

Use of Unsealed Nuclear Substances

This room has been classified as a “nuclear medicine” room for the use of unsealed nuclear substances in accordance with Canadian Nuclear Safety Commission guidelines. Below is a list of safe work practices to be followed when working in this room.

24-hour emergency contact (name and phone number)	Room identification

- Do not eat, drink, store food, or smoke in this room.
- Wear appropriate dosimeter at all times.
- In case of a spill or incident involving a nuclear substance, follow emergency procedures and notify the Radiation Safety Officer.
- Work in a fumehood when required by the Radiation Safety Officer.
- Wear protective clothing and equipment at all times.
- Wash hands regularly and monitor them for contamination frequently.
- Perform thyroid screening or bioassay when required.
- Check all packages containing nuclear substances for damage upon receipt.
- Store nuclear substances in a locked room or enclosure when not in use.

A room is classified as “nuclear medicine” for the use of unsealed nuclear substances where *in vivo* related diagnostic or therapeutic nuclear medicine is performed.

For more information, contact: Canadian Nuclear Safety Commission, Directorate of Nuclear Substance Regulation, P.O. Box 1046, Station B, Ottawa, Ontario, K1P 5S9. Telephone: 1-888-229-2672. Facsimile: (613) 995-5086.

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Canadian Nuclear Safety Commission
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Canadian Nuclear Safety Commission

P.O. Box 1046, Station B
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Tel: (613) 995-5894 Fax: (613) 995-5086
24 Hour Emergency Hotline: (613) 995-0479

Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Am	Common Names:	Americium
Atomic Weight:	241	Atomic Number:	95

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 432.2 years
CNSC Exemption Quantity (in Bq): 1 x 10^3 (1 kBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	n/a	n/a	n/a
Gamma & X-rays	0.05954	n/a	0.085	
Beta*		n/a	n/a	n/a
Alpha	5.49	n/a	n/a	n/a

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	Np-237(2.14E6 y), Pa-233(27.0 d), U-233(1.585E5 y), Th-229(7340 y), Ra-225(14.8 d), Ac-225(10.0 d), Fr-221(4.8 m), At-217(0.0323 s), Bi-213(45.65 m), Po-213(4.2 µs), Pb-209(3.253 h)

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: ZnS scintillation counter
Dosimetry:
External: TLD (whole body & skin) ✓ Extremity ✓ Neutron ______
Internal: Whole body ✓ Thorax ✓ Urine analysis ✓ Other (specify) faeces

Part 4 - PREVENTATIVE MEASURES

Low energy gamma radiation from sealed sources. No protective clothing is necessary for work with sealed sources.

Optimize time, distance, shielding. Manipulate sealed sources remotely to minimize extremity doses. Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	**Inhalation**
Compound Type	All compounds	All compounds
Annual Limit on Intake (Bq)	1×10^5	7×10^2

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques

- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control

- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements

- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

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Canada

Canadian Nuclear Safety Commission

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24 Hour Emergency Hotline: (613) 995-0479

Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	C	Common Names:	Carbon
Atomic Weight:	14	Atomic Number:	6

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 5730 years
CNSC Exemption Quantity (in Bq): 1 × 10^8 (100 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	n/a	n/a	n/a
Gamma & X-rays	-	n/a	n/a	n/a
Beta*	0.04945	0.156	n/a	n/a
Alpha	-	n/a	n/a	n/a

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	n/a

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: Thin end window Geiger-Mueller detector. Wipes counted by liquid scintillation.

Dosimetry:
External: TLD (whole body & skin) ______ Extremity ______ Neutron ______
Internal: Whole body ______ Thorax ______ Urine analysis T Other (specify) breath (carbon dioxide)

Part 4 - PREVENTATIVE MEASURES

Hazards: Carbon Monoxide: Chemical anoxia and asphyxiation. Carbon Dioxide: asphyxiation. Generation of carbon dioxide which could be inhaled.

Recommended protective clothing: Disposable lab coat, gloves (select gloves appropriate for chemicals handled) and wrist guards. Some organic compounds can be absorbed through gloves therefore wear two pairs of gloves and change the outer layer frequently.

Optimize time, distance, shielding. Be careful not to generate carbon dioxide and handle potentially volatile or dusty compounds in a fume hood. Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	**Inhalation**		
Compound Type	labelled organic compounds	Vapour	Dioxide	Monoxide
Annual Limit on Intake (Bq)	3E+07	3E+07	3E+10	3E+09

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques

- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control

- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements

- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

Canadian Nuclear Safety Commission

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Ca	Common Names:	Calcium
Atomic Weight:	45	Atomic Number:	20

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 163 days
CNSC Exemption Quantity (in Bq): 1 × 10^6 (1 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	n/a	n/a	n/a
Gamma & X-rays	0.01247	n/a	n/a	n/a
Beta*	0.07723	0.257	~0	1 cm Plexiglas
Alpha	-	n/a	n/a	n/a

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	n/a

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: End-window Geiger-Mueller detector

Dosimetry:
External: TLD (whole body & skin) T Extremity T Neutron ______
Internal: Whole body ______ Thorax ______ Urine analysis ______ Other (specify) Feces ______

Part 4 - PREVENTATIVE MEASURES

Calcium-45 is considered highly radiotoxic because of its affinity for the bone. Radiocalcium has a long biological half-life and can cause damage to the blood forming organs. Calcium reacts with water, producing hydrogen. If concentrated, the gas becomes a fire and explosion hazard. Calcium also poses a fire and explosion hazard when heated or when in contact with strong oxidizing agents.

Recommended protective clothing: No protective clothing is necessary for work with sealed sources. When working with unsealed sources wear appropriate protective clothing, such as laboratory coats, coveralls, gloves, safety glasses/goggles and a suitable mask, if the radioactive material is in the form of a dust, powder or if it is volatile.

Optimize time, distance, shielding. Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	**Inhalation**
Compound Type	All compounds	All compounds
Annual Limit on Intake (Bq)	3E+07	9E+06

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
C Wash well with soap and water and monitor skin
C Do Not abrade skin, only blot dry
C Decontamination of clothing and surfaces are covered under operating and emergency procedures
Spill and Leak Control
C Alert everyone in the area
C Confine the problem or emergency (includes the use of absorbent material)
C Clear area
C Summon Aid
Emergency Protective Equipment, Minimum Requirements
C Gloves
C Footwear Covers
C Safety Glasses
C Outer layer or easily removed protective clothing
C Suitable respirator selected

Revision number: 0 Date of revision: 23 Oct. 2003

Canadian Nuclear Safety Commission   Commission canadienne de sûreté nucléaire   Canada

Canadian Nuclear Safety Commission

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Cd	Common Names:	Cadmium
Atomic Weight:	109	Atomic Number:	48

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 464 days
CNSC Exemption Quantity (in Bq): 1 × 10^6 (1 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	n/a	n/a	
Gamma & X-rays	0.08803	n/a	0.045	
Beta*	-	-	n/a	
Alpha	-	n/a	n/a	

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	n/a

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: NaI scintillation counter

Dosimetry:
External: TLD (whole body & skin) T Extremity T Neutron ______
Internal: Whole body ______ Thorax ______ Urine analysis _______ Other (specify) Feces

Part 4 - PREVENTATIVE MEASURES

Cadmium and its compounds are toxic by ingestion and inhalation. The oral toxicity of Cd and its compounds is high. However, ingestion usually causes a strong emetic action, little Cd is therefore absorbed and fatal poisoning rarely occurs. Cadmium and some compounds are suspected carcinogens. Flammable in powder form.

Recommended protective clothing: No protective clothing is necessary for work with sealed sources. When working with unsealed sources wear appropriate protective clothing, such as laboratory coats, coveralls, gloves, safety glasses/goggles and a suitable mask, if the radioactive material is in the form of a dust, powder or if it is potentially volatile.

Optimize time, distance, shielding. Manipulate sealed sources remotely to minimize extremity doses.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	Inhalation		
Compound Type	All inorganic compounds	Oxides, hydroxides	Sulphides, halides, nitrates	Unspecified compounds
Annual Limit on Intake (Bq)	1E+07	2E+06	4E+06	5E+06

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
C Wash well with soap and water and monitor skin
C Do Not abrade skin, only blot dry
C Decontamination of clothing and surfaces are covered under operating and emergency procedures
Spill and Leak Control
C Alert everyone in the area
C Confine the problem or emergency (includes the use of absorbent material)
C Clear area
C Summon Aid
Emergency Protective Equipment, Minimum Requirements
C Gloves
C Footwear Covers
C Safety Glasses
C Outer layer or easily removed protective clothing
C Suitable respirator selected

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Co	Common Names:	Cobalt
Mass Number:	57	Atomic Number:	27

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 270.9 days
CNSC Exemption Quantity (in Bq): 1×10^5 (0.1 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	Average Energy (MeV)**	Maximum Energy (MeV)***	Dose Rate at 1m Distance (mSv/h•GBq)	Recommended Shielding
Neutrons	-	-	-	-
Gamma & X-rays	0.1221	-	0.041	2 mm Pb
Beta*	-	-	-	-
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	n/a

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: NaI scintillation counter
Dosimetry:
External: TLD (whole body & skin) ✓ Extremity ✓ Neutron ______
Internal: Whole body ✓ Thorax ✓ Urine analysis ✓ Other (specify) Faeces

Part 4 - PREVENTATIVE MEASURES

Health hazards associated with cobalt (metal, fume and dust) include cumulative lung damage and dermatitis. May emit radioactive fumes containing Co-57 when heated to decomposition. Cobalt dust is flammable.

Recommended protective clothing: Disposable plastic, latex, or rubber gloves. Wear a lab coat, which must be monitored before leaving the laboratory. Also wear safety glasses.

Keep handling time to a minimum. Use syringe shields and tongs.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion		Inhalation	
Compound Type	Unspecified compounds	Oxides, hydroxides, inorganic compounds	Unspecified compounds	Oxides, hydroxides, halides, nitrates
Annual Limit on Intake (Bq)	1E+08	1E+08	5E+07	3E+07

EMERGENCY PROCEDURES

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Personal Decontamination Techniques
- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control
- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements
- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

Revision number: 1 Date of revision: 18 May 2004

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Co	Common Names:	Cobalt
Atomic Weight:	58	Atomic Number:	27

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 70.8 days
CNSC Exemption Quantity (in Bq): 1 × 10^5 (0.1 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	-	-	-
Gamma & X-rays	0.8108	-	0.17	30 mm Pb
Beta*	-	-	-	-
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	n/a

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: Geiger-Mueller counter, NaI scintillation counter

Dosimetry:
External: TLD (whole body & skin) T Extremity T Neutron ______
Internal: Whole body T Thorax T Urine analysis T Other (specify) Faeces

Part 4 - PREVENTATIVE MEASURES

Health hazards associated with cobalt (metal, fume and dust) include cumulative lung damage and dermatitis. Cobalt dust is flammable.

Recommended protective clothing: No protective clothing is necessary for work with sealed sources. When working with unsealed sources wear appropriate protective clothing, such as laboratory coats, coveralls, gloves, safety glasses/goggles and a suitable mask, if the radioactive material is in the form of a dust, powder or if it is potentially volatile.

Optimize time, distance, shielding.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion		**Inhalation**	
Compound Type	Unspecified compounds	Oxides, hydroxides, inorganic compounds	Unspecified compounds	Oxides, hydroxides, halides, nitrates
Annual Limit on Intake (Bq)	3E+07	3E+07	1E+07	1E+07

EMERGENCY PROCEDURES

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Personal Decontamination Techniques
C Wash well with soap and water and monitor skin
C Do Not abrade skin, only blot dry
C Decontamination of clothing and surfaces are covered under operating and emergency procedures
Spill and Leak Control
C Alert everyone in the area
C Confine the problem or emergency (includes the use of absorbent material)
C Clear area
C Summon Aid
Emergency Protective Equipment, Minimum Requirements
C Gloves
C Footwear Covers
C Safety Glasses
C Outer layer or easily removed protective clothing
C Suitable respirator selected

Revision number: 0 Date of revision: 23 Oct. 2003

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Co	Common Names:	Cobalt
Mass Number:	60	Atomic Number:	27

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 5.271 years
CNSC Exemption Quantity (in Bq): 1 × 10^5 (0.1 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h•GBq)**	**Recommended Shielding**
Neutrons	-	-	n/a	-
Gamma & X-rays	1.25	-	0.37	40 mm Pb
Beta*	0.09577	0.344	~0.05	-
Alpha	-	-	n/a	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	n/a

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: Geiger-Mueller detector

Dosimetry:
External: TLD (whole body & skin) ✓ Extremity ✓ Neutron ______
Internal: Whole body ✓ Thorax ✓ Urine analysis ✓ Other (specify) Faeces

Part 4 - PREVENTATIVE MEASURES

Health hazards associated with cobalt (metal, fume and dust) include cumulative lung damage and dermatitis. Cobalt dust is flammable. Cobalt-60 sealed sources presents an external gamma hazard.

Recommended protective clothing: No protective clothing is necessary for work with sealed sources. When working with unsealed sources wear appropriate protective clothing, such as laboratory coats, coveralls, gloves, safety glasses/goggles and a suitable mask, if the radioactive material is in the form of a dust, powder or if it is potentially volatile.

Optimize time, distance, shielding. Manipulate sealed sources remotely to minimize extremity doses. Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion		**Inhalation**	
Compound Type	Unspecified compounds	Oxides, hydroxides, inorganic compounds	Unspecified compounds	Oxides, hydroxides, halides, nitrates
Annual Limit on Intake (Bq)	6E+06	8E+06	3E+06	1E+06

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques

- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control

- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements

- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Cr	Common Names:	Chromium
Atomic Weight:	51	Atomic Number:	24

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 27.704 days
CNSC Exemption Quantity (in Bq): 1×10^6 (1 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	n/a		
Gamma & X-rays	0.3201	n/a	0.0063	10 mm Pb
Beta*	-			
Alpha	-	n/a		

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	n/a

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: Scintillation detector

Dosimetry:
External: TLD (whole body & skin) T Extremity T Neutron ______
Internal: Whole body T Thorax ______ Urine analysis ______ Other (specify) ____________________

Part 4 - PREVENTATIVE MEASURES

Chromium and chromate salts are suspected carcinogens of the lungs, nasal cavity and paranasal sinus, also experimental carcinogen of the stomach and larynx. Skin exposure to chromate salts may result in dermatitis. Sodium chromate (Cr-51) solution may emit radioactive fumes containing Cr-51 when heated to decomposition.

Recommended protective clothing: Disposable plastic, latex, or rubber gloves. Lab coat. Safety glasses.

Minimise handling time. Use syringe shields and tongs to handle unshielded sources and potentially contaminated vessels.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	Inhalation		
Compound Type	All compounds	Unspecified Soluble compounds	Halides, nitrates	Oxides, hydroxides
Annual Limit on Intake (Bq)	5E+08	7E+08	6E+08	6E+08

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control
- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements
- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

Revision number: 0

Date of revision: 23 Oct. 2003

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Ga	Common Names:	Gallium
Atomic Weight:	67	Atomic Number:	31

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 78.26 hours
CNSC Exemption Quantity (in Bq): 1×10^6 (1 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	-	-	-
Gamma & X-rays	0.09331	-	0.03	6 mm Pb
Beta*	-	-	-	-
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	n/a

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: Geiger-Mueller detector, Nal(TI) detector, Liquid scintillation

Dosimetry:
External: TLD (whole body & skin) T Extremity T Neutron ______
Internal: Whole body ______ Thorax ______ Urine analysis T Other (specify) ____________________

Part 4 - PREVENTATIVE MEASURES

Rare occurrences of allergic reactions, skin rash, and nausea have been reported in association with Ga-67 use. Gallium 67 citrate solution may emit radioactive fumes containing Ga-67 when heated to decomposition.

Recommended protective clothing: Disposable plastic, latex, or rubber gloves. Wear a lab coat, which must be monitored before leaving the laboratory. Also wear safety glasses.

Handling time should be kept to a minimum and syringe shields and tongs should be used.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	**Inhalation**	
Compound Type	All compounds	Unspecified compounds	Oxides, hydroxides, nitrates, halides, carbides
Annual Limit on Intake (Bq)	1E+08	2E+08	7E+07

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
C Wash well with soap and water and monitor skin
C Do Not abrade skin, only blot dry
C Decontamination of clothing and surfaces are covered under operating and emergency procedures
Spill and Leak Control
C Alert everyone in the area
C Confine the problem or emergency (includes the use of absorbent material)
C Clear area
C Summon Aid
Emergency Protective Equipment, Minimum Requirements
C Gloves
C Footwear Covers
C Safety Glasses
C Outer layer or easily removed protective clothing
C Suitable respirator selected

Revision number: 0

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Ge	Common Names:	Germanium
Atomic Weight:	68	Atomic Number:	32

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 288 days
CNSC Exemption Quantity (in Bq): 1×10^4 (10 kBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	n/a		
Gamma & X-rays	0.00924	n/a	0.0163	2 mm Pb
Beta*	-			
Alpha	-	n/a		

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	Ga-68(68.0 m)

Part 3 - DETECTION AND MEASUREMENT

Method of Detection:

Dosimetry

External: TLD (whole body & skin) ✓ Extremity ✓ Neutron _____

Internal: Whole body _____ Thorax _____ Urine analysis ✓ Other (specify) __________

Part 4 - PREVENTATIVE MEASURES

No protective clothing is necessary for work with sealed sources. When working with unsealed sources wear appropriate protective clothing, such as laboratory coats, coveralls, gloves, safety glasses/goggles and a suitable mask, if the radioactive material is in the form of a dust, powder or if it is potentially volatile. Optimize time, distance, shielding.

Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	Inhalation	
Compound Type	All compounds	Unspecified compounds	Oxides, sulphides, halides
Annual Limit on Intake	2E+07	2E+07	3E+06

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control
- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements
- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	H	Common Names:	Tritium
Atomic Weight:	3	Atomic Number:	1

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 12.35 years
CNSC Exemption Quantity (in Bq): 1×10^9 (1 GBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	n/a	n/a	n/a
Gamma & X-rays	-	n/a	n/a	n/a
Beta*	0.0057	0.0186	n/a	n/a
Alpha	-	n/a	n/a	n/a

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	n/a

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: Wipes counted by liquid scintillation

Dosimetry:

External: TLD (whole body & skin) ______ Extremity ______ Neutron ______

Internal: Whole body ______ Thorax ______ Urine analysis __T__ Other (specify) ____________________

Part 4 - PREVENTATIVE MEASURES

Tritium is not a radiation hazard unless it enters the body. Once in the body, tritium water is uniformly distributed in the body water and can then irradiate live tissue. Inhaled tritium gas will irradiate the lungs. Tritiated water is much more (10,000 ×) radiotoxic than gaseous tritium. Tritiated thymidine will be concentrated in the nuclei of DNA synthesising cells and may result in chromosome damage. Tritiated water can be absorbed through the surface of skin, leading to an internal exposure. Gaseous tritium is a fire and explosion hazard when exposed to heat or flame and can react vigorously with oxidizing materials.

Recommended protective clothing: Lab coat. PVC gloves (0.5 mm thick) are preferred because of this material's low permeability to tritiated water. Many tritium compounds readily penetrate gloves and skin. Handle these compounds remotely, wear two pairs of gloves and change the outer layer at least every twenty minutes. Plastic aprons provide added protection especially against tritiated water. Plastic suits may be necessary for work at TBq levels or in an atmosphere contaminated with tritiated water.

Handle tritiated water, gases and volatile liquids in ventilated enclosures. Store tritiated water at room temperature as it is known to contaminate the frost in freezers. Use glass containers to store tritium compounds because tritiated water and tritiated organic solvents will pass through plastic.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	Inhalation	
Compound Type	Tritiated water	Tritiated water	Elemental tritium gas
Annual Limit on Intake (Bq)	1E+09	1E+09	1E+13

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
C Wash well with soap and water and monitor skin
C Do Not abrade skin, only blot dry
C Decontamination of clothing and surfaces are covered under operating and emergency procedures
Spill and Leak Control
C Alert everyone in the area
C Confine the problem or emergency (includes the use of absorbent material)
C Clear area
C Summon Aid
Emergency Protective Equipment, Minimum Requirements
C Gloves
C Footwear Covers
C Safety Glasses
C Outer layer or easily removed protective clothing
C Suitable respirator selected

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	I	Common Names:	Iodine
Atomic Weight:	123	Atomic Number:	53

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 13.2 hours
CNSC Exemption Quantity (in Bq): 1×10^7 (10 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	-	-	-
Gamma & X-rays	0.1590	-	0.075	1 mm Pb
Beta*	-	-	-	-
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	Te-123(1E13 y), Te-123m(119.7 d)

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: NaI scintillation counter

Dosimetry:
External: TLD (whole body & skin) T Extremity T Neutron ______
Internal: Whole body ______ Thorax ______ Urine analysis T Other (specify) Thyroid

Part 4 - PREVENTATIVE MEASURES

Exposure to significant amounts of radioiodine increases risk of developing thyroid cancer.
Iodine is toxic by ingestion and inhalation, and a strong irritant to eyes and skin. Iodine can be absorbed through the skin. Heating sodium iodide 123 capsules to decomposition may emit in radioactive fumes containing I -123.

Recommended protective clothing: Disposable plastic, latex, or rubber gloves. Wear a lab coat, which must be monitored before leaving the laboratory. Also wear safety glasses.

Keep handling time to a minimum. Syringe shields and tongs should be used.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	**Inhalation**
Compound Type	All compounds	All compounds
Annual Limit on Intake (Bq)	1×10^8	2×10^8

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control
- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements
- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

Revision number: 0

Date of revision: 24 Oct. 2003

Canadian Nuclear Safety Commission

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	I	Common Names:	Iodine
Atomic Weight:	125	Atomic Number:	53

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 60.14 days
CNSC Exemption Quantity (in Bq): 1×10^6 (1 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	Average Energy (MeV)**	Maximum Energy (MeV)***	Dose Rate at 1m Distance (mSv/h∃GBq)	Recommended Shielding
Neutrons	-	-	-	-
Gamma & X-rays	0.03549	-	0.074	-
Beta*	-	-	-	-
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	

Part 3 – DETECTION AND MEASUREMENT

Method of Detection: Scintillation detector

Dosimetry:
External: TLD (whole body & skin) ✓ Extremity ✓ Neutron ______
Internal: Whole body ______ Thorax ______ Urine analysis ✓ Other (specify) Thyroid

Part 4 - PREVENTATIVE MEASURES

Exposure to significant amounts of radioiodine increases risk of developing thyroid cancer. Iodine is toxic by ingestion and inhalation and a strong irritant of eyes and skin. Iodine can be absorbed through the skin. When iodinated (I -125) albumin injection is heated to decomposition, radioactive fumes containing I -125 may be emitted.
Recommended protective clothing: Disposable plastic, latex, or rubber gloves. Wear a lab coat, which must be monitored before leaving the laboratory. Also wear safety glasses. Some iodine compounds can penetrate surgical rubber gloves. Wear two pairs, or polyethylene gloves over rubber.

Minimise handling time. Use syringe shields and tongs. When possible handle iodine compounds in a fume hood.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	Inhalation
Compound Type	All compounds	All compounds
Annual Limit on Intake (Bq)	1×10^6	1×10^6

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
X Wash well with soap and water and monitor skin
X Do Not abrade skin, only blot dry
X Decontamination of clothing and surfaces are covered under operating and emergency procedures
Spill and Leak Control
X Alert everyone in the area
X Confine the problem or emergency (includes the use of absorbent material)
X Clear area
X Summon Aid
Emergency Protective Equipment, Minimum Requirements
X Gloves
X Footwear Covers
X Safety Glasses
X Outer layer or easily removed protective clothing
X Suitable respirator selected

Revision number: 1 Date of revision: 1 September 2004

Canadian Nuclear Safety Commission   Commission canadienne de sûreté nucléaire   Canada

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	I	Common Names:	Iodine
Atomic Weight:	131	Atomic Number:	53

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 8.04 days
CNSC Exemption Quantity (in Bq): 1×10^4 (10 kBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	-	-	-
Gamma & X-rays	0.3645	-	0.076	24 mm Pb
Beta*	0.1915	0.606	1.2	-
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	< 1% to ^{131m}Xe (11.8 d)

Part 3 – DETECTION AND MEASUREMENT

Method of Detection: Scintillation detector

Dosimetry:
External: TLD (whole body & skin) T Extremity T Neutron ______
Internal: Whole body ______ Thorax ______ Urine analysis T Other (specify) Thyroid

Part 4 - PREVENTATIVE MEASURES

Exposure to significant amounts of radioiodine increases risk of developing thyroid cancer. Iodine is toxic by ingestion and inhalation and a strong irritant of eyes and skin. Iodine can be absorbed through the skin. Heating Hippuran (I-131) or sodium iodide -131 to decomposition may result in radioactive fumes containing I -131 to be emitted.
Recommended protective clothing: Disposable plastic, latex, or rubber gloves. Wear a lab coat, which must be monitored before leaving the laboratory. Also wear safety glasses. Fluoroscopy aprons provide no protection against the radiation from I -131. Always wear disposable plastic when working with I-131 and use instruments to handle I-131.

Minimise handling time. Use syringe shields and tongs. Store volatile iodine -131 in a refrigerator to reduce the production of radioactive vapour.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	**Inhalation**
Compound Type	All compounds	All compounds
Annual Limit on Intake (Bq)	9×10^5	2×10^6

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control
- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements
- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

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Canadian Nuclear Safety Commission Commission canadienne de sûreté nucléaire

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	In	Common Names:	Indium
Atomic Weight:	111	Atomic Number:	49

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 2.83 days
CNSC Exemption Quantity (in Bq): 1 × 10^5 (100 kBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	Average Energy (MeV)**	Maximum Energy (MeV)***	Dose Rate at 1m Distance (mSv/h·GBq)	Recommended Shielding
Neutrons	-	n/a		
Gamma & X-rays	0.2454	n/a	0.136	3 mm Pb
Beta*	-			
Alpha	-	n/a		

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: NaI scintillation counter

Dosimetry:
External: TLD (whole body & skin) T Extremity T Neutron ______
Internal: Whole body T Thorax ______ Urine analysis ______ Other (specify) ____________________

Part 4 - PREVENTATIVE MEASURES

Indium metal and its compounds are toxic by inhalation, causing cumulative organ damage. Suspected teratogen. When Indium 111 chloride is heated to decomposition, radioactive fumes may be emitted.

Recommended protective clothing: Disposable plastic, latex, or rubber gloves. Wear a lab coat, which must be monitored before leaving the laboratory. Also wear safety glasses.
Keep handling time to a minimum. Use syringe shields and tongs. Always use the principles of time, distance and shielding to minimize dose.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	Inhalation	
Compound Type	All compounds	Unspecified compounds	Oxides, hydroxides, halides, nitrates
Annual Limit on Intake (Bq)	7×10^7	9×10^7	6×10^7

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control
- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements
- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

Revision number: 0

Date of revision: 24 Oct. 2003

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Ir	Common Names:	Iridium
Atomic Weight:	192	Atomic Number:	77

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 74.02 days
CNSC Exemption Quantity (in Bq): 1×10^4 (10 kBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	-	-	-
Gamma & X-rays	0.3094	-	0.16	19 mm Pb
Beta*	0.2089	0.672	~0	-
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: End or side window Geiger-Mueller counter, NaI scintillation counter

Dosimetry:
External: TLD (whole body & skin) ✓ Extremity ✓ Neutron ______
Internal: Whole body ______ Thorax ______ Urine analysis ______ Other (specify) ____________

Part 4 - PREVENTATIVE MEASURES

Recommended protective clothing: No protective clothing is necessary for work with sealed sources. When working with unsealed sources wear appropriate protective clothing, such as laboratory coats, coveralls, gloves, safety glasses/goggles and a suitable mask, if the radioactive material is in the form of a dust, powder or if it is potentially volatile.
Always use the principles of time, distance and shielding to minimize dose.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	**Inhalation**		
Compound Type	All compounds	Unspecified compounds	Halides, nitrates, metallic iridium	Oxides, hydroxides
Annual Limit on Intake (Bq)	1×10^7	9×10^6	5×10^6	4×10^6

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control
- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements
- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

Revision number: 0 Date of revision: 29 Sept. 2004

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Mo	Common Names:	Molybdenum
Atomic Weight:	99	Atomic Number:	42

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 66 hours
CNSC Exemption Quantity (in Bq): 1×10^4 (10 kBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	n/a		
Gamma & X-rays	0.7395	n/a	0.031	20 mm Pb
Beta*	0.4426	1.23	6.30	
Alpha	-	n/a		

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	Tc-99m(6.02 h), Tc-99(2.13E5 y)

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: End or side window Geiger-Mueller counter

Dosimetry:
External: TLD (whole body & skin) ✓ Extremity ✓ Neutron ______
Internal: Whole body ______ Thorax ______ Urine analysis ______ Other (specify) ____________________

Part 4 - PREVENTATIVE MEASURES

Molybdenum dust or powder is flammable.

Recommended protective clothing: When working with unsealed sources wear appropriate protective clothing, such as laboratory coats (which must be monitored before leaving the laboratory), coveralls, gloves, safety glasses/goggles and a suitable mask, if the radioactive material is in the form of a dust, powder or if it is potentially volatile.

Optimize time, distance, shielding. Laboratory equipment used for radioactive work must not be used for other purposes. Monitor equipment and supplies for loose contamination before removing from laboratory. Use disposable absorbent liners on trays.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion		**Inhalation**	
Compound Type	Unspecified compounds	Sulphide	Unspecified compounds	Oxides, hydroxides, MoS_2
Annual Limit on Intake (Bq)	3×10^7	2×10^7	6×10^7	2×10^7

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques

- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control

- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements

- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

Revision number: 0

Date of revision: 29 Sept. 2004

Canadian Nuclear Safety Commission   Commission canadienne de sûreté nucléaire   Canada

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	P	Common Names:	Phosphorus
Atomic Weight:	32	Atomic Number:	15

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 14.3 days
CNSC Exemption Quantity (in Bq): 1×10^4 (10 kBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	Average Energy (MeV)**	Maximum Energy (MeV)***	Dose Rate at 1m Distance (mSv/h·GBq)	Recommended Shielding
Neutrons	-	-	-	-
Gamma & X-rays	-	-	-	-
Beta*	0.6947	1.710	9.17	1 cm Plexiglas
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	n/a

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: Geiger-Meuller detector, Gamma survey meters with sodium iodide crystal detectors

Dosimetry:
External: TLD (whole body & skin) ______ Extremity ______ Neutron ______
Internal: Whole body T Thorax ______ Urine analysis T Other (specify) ____________________

Part 4 - PREVENTATIVE MEASURES

Chromic acid and its salts have a corrosive action on the skin and mucous membranes. Sodium phosphate is a mild irritant. Phosphocol and Sodium Phosphate (P-32) solutions may emit radioactive fumes containing P-32 when heated to decomposition.

Recommended protective clothing: Disposable plastic, latex, or rubber gloves. Lab coat. Safety glasses.

Keep handling time to minimum. Use syringe shields (aluminium or lead foil) and tongs to avoid direct skin contact. When possible work behind a Plexiglas screen. Finger dosimeters should be worn if using quantities greater than a few tens of MBq (~a mCi). Vial should be encased in Lucite.

Always use the principles of time, distance and shielding to minimize dose.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	Inhalation
Compound Type	All compounds	All compounds
Annual Limit on Intake (Bq)	8×10^6	8×10^6

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
C Wash well with soap and water and monitor skin
C Do Not abrade skin, only blot dry
C Decontamination of clothing and surfaces are covered under operating and emergency procedures
Spill and Leak Control
C Alert everyone in the area
C Confine the problem or emergency (includes the use of absorbent material)
C Clear area
C Summon Aid
Emergency Protective Equipment, Minimum Requirements
C Gloves
C Footwear Covers
C Safety Glasses
C Outer layer or easily removed protective clothing
C Suitable respirator selected

Revision number: 0 Date of revision: 5 April 2004

Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	S	Common Names:	Sulphur
Atomic Weight:	35	Atomic Number:	16

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 87.44 days
CNSC Exemption Quantity (in Bq): 1×10^8 (100 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	-	-	-
Gamma & X-rays	-	-	-	-
Beta*	0.04883	0.167	n/a	1 cm Plexiglas
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	n/a

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: Thin end window Geiger-Mueller detector, Liquid scintillation counter
Dosimetry:
External: TLD (whole body & skin) ______ Extremity ______ Neutron ______
Internal: Whole body ______ Thorax ______ Urine analysis T Other (specify) ____________

Part 4 - PREVENTATIVE MEASURES

Sulphur dioxide: irritant to eye, nose, throat, lungs; bronchoconstriction; mutagen, suspect reproductive effects.
Hydrogen sulphide: moderate irritant to eye (conjunctivitis), lung; acute systemic toxicity; CNS effects. Sulphur is combustible.
Recommended protective clothing: Wear disposable lab coat, gloves and wrist guards for secondary protection. Select appropriate gloves for chemicals handled.
Handle potentially volatile compounds in ventilated enclosures. Take care not to generate sulphur dioxide or hydrogen sulphide which could be inhaled.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	Inhalation
Compound Type	All compounds	All compounds
Annual Limit on Intake (Bq)	1×10^8	2×10^8

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
C Wash well with soap and water and monitor skin
C Do Not abrade skin, only blot dry
C Decontamination of clothing and surfaces are covered under operating and emergency procedures
Spill and Leak Control
C Alert everyone in the area
C Confine the problem or emergency (includes the use of absorbent material)
C Clear area
C Summon Aid
Emergency Protective Equipment, Minimum Requirements
C Gloves
C Footwear Covers
C Safety Glasses
C Outer layer or easily removed protective clothing
C Suitable respirator selected

Revision number: 0 Date of revision: 5 April 2004

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Sb	Common Names:	Antimony
Atomic Weight:	124	Atomic Number:	51

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 60.2 days
CNSC Exemption Quantity (in Bq): 1×10^4 (10 kBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	-	-	-
Gamma & X-rays	1.691	-	0.288	42 mm Pb
Beta*	0.1943	2.31	2.98	-
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: End or side window Geiger-Mueller counter, NaI scintillation counter

Dosimetry:
External: TLD (whole body & skin) ✓ Extremity ✓ Neutron ______
Internal: Whole body ✓ Thorax ______ Urine analysis ______ Other (specify) ____________________

Part 4 - PREVENTATIVE MEASURES

Antimony trichloride is a corrosive liquid or solid that is very irritating to eyes and skin. Antimony is combustible. Soluble salts of antimony are toxic. Antimony trichloride fumes slightly in air.

Recommended protective clothing: No protective clothing is necessary for work with sealed sources. When working with unsealed sources wear appropriate protective clothing, such as laboratory coats, coveralls, gloves, safety glasses/goggles and a suitable mask, if the radioactive material is in the form of a dust, powder or if it is potentially volatile. Optimize time, distance, shielding.

Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	Inhalation	
Compound Type	All compounds	Unspecified compounds	Oxides, hydroxides, halides, sulphides, sulphates, nitrates
Annual Limit on Intake (Bq)	8×10^6	1×10^7	4×10^6

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control
- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements
- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

Revision number: 0

Date of revision: 29 Sept. 2004

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24 Hour Emergency Hotline: (613) 995-0479

Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Sc	Common Names:	Scandium
Atomic Weight:	46	Atomic Number:	21

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 83.83 days
CNSC Exemption Quantity (in Bq): 1×10^5 (100 kBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	-	-	-
Gamma & X-rays	1.121	-	0.32	40 mm Pb
Beta*	0.1120	0.357	~0	-
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: NaI scintillation counter

Dosimetry:
External: TLD (whole body & skin) ✓ Extremity ✓ Neutron ______
Internal: Whole body ✓ Thorax ______ Urine analysis ______ Other (specify) ____________________

Part 4 - PREVENTATIVE MEASURES

Scandium reacts rapidly with acids.
Recommended protective clothing: No protective clothing is necessary for work with sealed sources.
When working with unsealed sources wear appropriate protective clothing, such as laboratory coats (which must be monitored before leaving the laboratory), coveralls, gloves, safety glasses/goggles and a suitable mask, if the radioactive material is in the form of a dust, powder or if it is potentially volatile.

Optimise time, distance and shielding. Laboratory equipment used for radioactive work must not be used for other purposes. Monitor equipment and supplies for loose contamination before removing from laboratory. Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	**Inhalation**
Compound Type	All compounds	All compounds
Annual Limit on Intake (Bq)	1×10^{7}	4×10^{6}

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques

- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control

- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements

- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

Revision number: 0

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Radiation Safety Data Sheet

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Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Se	Common Names:	Selenium
Atomic Weight:	75	Atomic Number:	34

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 119.8 days
CNSC Exemption Quantity (in Bq): 1×10^5 (0.1 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	n/a		
Gamma & X-rays	0.1360	n/a	0.232	3 mm Pb
Beta*	-			
Alpha	-	n/a		

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: NaI scintillation counter

Dosimetry

External: TLD (whole body & skin) ____ Extremity ✓ Neutron ____

Internal: Whole body ✓ Thorax ____ Urine analysis ✓ Other (specify) ____

Part 4 - PREVENTATIVE MEASURES

No protective clothing is necessary for work with sealed sources. When working with unsealed sources wear appropriate protective clothing, such as laboratory coats (which must be monitored before leaving the laboratory), coveralls, gloves, safety glasses/goggles and a suitable mask, if the radioactive material is in the form of a dust, powder or if it is potentially volatile.

Optimize time, distance and shielding. Laboratory equipment used for radioactive work must not be used for other purposes. Monitor equipment and supplies for loose contamination before removing from laboratory. Use disposable absorbent liners on trays.

Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion		Inhalation	
Compound Type	Unspecified compounds	Elemental selenium & selenides	Unspecified inorganic compounds	Elemental selenium, oxides, hydroxides, carbides
Annual Limit on Intake	8E+06	5E+07	1E+07	1E+07

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control
- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements
- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
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Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Tc	Common Names:	Technetium
Atomic Weight:	99m	Atomic Number:	43

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 6.02 hours
CNSC Exemption Quantity (in Bq): 1 × 10^7 (10 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h□GBq)**	**Recommended Shielding**
Neutrons	-	-	-	-
Gamma & X-rays	0.1405	-	0.033	1 mm Pb
Beta*	-	-	-	-
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	Tc-99(2.13E5 y)

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: End or side window Geiger-Mueller counter, NaI scintillation counter

Dosimetry:
External: TLD (whole body & skin) T Extremity T Neutron ______
Internal: Whole body T Thorax ______ Urine analysis T Other (specify) ____________

Part 4 - PREVENTATIVE MEASURES

May emit radioactive fumes containing Mo-99 or Tc-99m when heated to decomposition.

Recommended protective clothing: Waterproof gloves should be worn during elution. Disposable plastic, latex, or rubber gloves. Lab coat (which must be monitored before leaving the laboratory). Safety glasses.
During and subsequent to elution, the eluate collection vial should be kept in a lead shield. Use syringe shields (tungsten, 1.4 mm lead equivalent) and tongs.
Laboratory equipment used for radioactive work must not be used for other purposes. Monitor equipment and supplies for loose contamination before removing from laboratory. Use disposable absorbent liners on trays.
Always use the principles of time, distance and shielding to minimize dose.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	Inhalation	
Compound Type	All compounds	Unspecified compounds	Oxides, hydroxides, halides, nitrates
Annual Limit on Intake (Bq)	9×10^8	1×10^9	7×10^8

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control
- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements
- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

Revision number: 0

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Radiation Safety Data Sheet

This data sheet presents information on radioisotopes only.
For information on chemical compounds incorporating this radionuclide, see the relevant Material Safety Data Sheet.

Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Th	Common Names:	Thorium
Atomic Weight:	232	Atomic Number:	90

Note: There will always be some ^{228}Th and a variable amount of ^{228}Ra present in ^{232}Th.

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 1.405 × 10^{10} years
CNSC Exemption Quantity (in Bq): 100

A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	Average Energy (MeV)**	Maximum Energy (MeV)***	Dose Rate at 1m Distance (mSv/h·GBq)	Recommended Shielding
Neutrons		-	-	-
Gamma & X-rays	0.059	-	0.0185	-
Beta*	-	-	-	-
Alpha	4.010	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	Ra-228(5.75 y), Ac-228(6.13 h), Th-228(1.9131 y), Ra-224(3.66 d), Rn-220(55.6 s), Po-216(0.15 s), Pb-212(10.64 h), Bi-212(60.55 m), Po-212(0.305 µs), Tl-208(3.07 m)

Part 3 – DETECTION AND MEASUREMENT

Method of Detection: ZnS scintillation counter

Dosimetry:

External: TLD (whole body & skin) ✓ Extremity ✓ Neutron ____

Internal: Whole body ✓ (Measurement of progeny) Thorax ____ Urine analysis ✓ Other (specify) Faeces, ^{220}Rn in breath, personal air sampler

Part 4 - PREVENTATIVE MEASURES

Thorium and its decay products are toxic by ingestion and inhalation. Thorium is attracted to the bones, lungs, lymphatic glands and parenchymatous tissues. Thorium remains in the body for a long time and is known to cause changes to blood forming, nervous and reticuloendothelial systems, and functional and structural damage to lung and bone tissue. Long after the initial exposure, neoplasms may occur and immunological activity of the body impaired. Always use the principles of time, distance and shielding to minimize dose.

Thorium is flammable and explosive in powder form. Thorium dusts have very low ignition points and may ignite at room temperatures.

No protective clothing is necessary for work with sealed sources. When working with unsealed sources wear appropriate protective clothing, such as laboratory coats (which must be monitored before leaving the laboratory), coveralls, rubber or plastic gloves.

When handling thorium oxide or thorium nitrate take care not to generate dust. Handle unsealed sources in glove boxes. Optimize time, distance and shielding. Laboratory equipment used for radioactive work must not be used for other purposes. Monitor equipment and supplies for loose contamination before removing from laboratory. Use disposable absorbent liners on trays.

Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion		**Inhalation**	
Compound Type	Unspecified compounds*	Oxides & hydroxides*	Unspecified compounds*	Oxides, hydroxides*
Annual Limit on Intake (Bq)	7×10^4	2×10^5	4×10^2	5×10^2

Note: * Values are in Bq ^{232}Th activity for intakes of natural thorium, *i.e.*, ^{232}Th + ^{228}Th in equilibrium.

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
- Wash well with soap and water and monitor skin
- Do Not abrade skin, only blot dry
- Decontamination of clothing and surfaces are covered under operating and emergency procedures

Spill and Leak Control
- Alert everyone in the area
- Confine the problem or emergency (includes the use of absorbent material)
- Clear area
- Summon Aid

Emergency Protective Equipment, Minimum Requirements
- Gloves
- Footwear Covers
- Safety Glasses
- Outer layer or easily removed protective clothing
- Suitable respirator selected

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Radiation Safety Data Sheet

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Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	Tl	Common Names:	Thallium
Atomic Weight:	201	Atomic Number:	81

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 3.044 days
CNSC Exemption Quantity (in Bq): 1 × 10^6 (1 MBq)
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h·GBq)**	**Recommended Shielding**
Neutrons	-	-	-	-
Gamma & X-rays	0.1667	-	0.024	-
Beta*	-	-	-	-
Alpha	-	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.

Progeny	

Part 3 - DETECTION AND MEASUREMENT

Method of Detection: Geiger-Mueller detector, NaI (tl) detector, Liquid scintillation counter

Dosimetry:
External: TLD (whole body & skin) T Extremity T Neutron ______
Internal: Whole body T Thorax ______ Urine analysis T Other (specify) Thyroid

Part 4 - PREVENTATIVE MEASURES

Thallium forms toxic compounds on contact with moisture. Avoid skin contact. Thallous chloride is poisonous. Thallous 201 chloride solution may emit radioactive fumes containing Tl-201 when heated to decomposition. Thallium readily forms toxic soluble compounds when exposed to air or water.
Recommended protective clothing: Disposable plastic, latex, or rubber gloves. Lab coat (which must be monitored before leaving the laboratory). Safety glasses.

Keep handling time to a minimum. Use syringe shields and tongs. Laboratory equipment used for radioactive work must not be used for other purposes. Monitor equipment and supplies for loose contamination before removing from laboratory. Use disposable absorbent liner on trays.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE

	Ingestion	Inhalation
Compound Type	All compounds	All compounds
Annual Limit on Intake (Bq)	2×10^8	3×10^8

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
C Wash well with soap and water and monitor skin
C Do Not abrade skin, only blot dry
C Decontamination of clothing and surfaces are covered under operating and emergency procedures
Spill and Leak Control
C Alert everyone in the area
C Confine the problem or emergency (includes the use of absorbent material)
C Clear area
C Summon Aid
Emergency Protective Equipment, Minimum Requirements
C Gloves
C Footwear Covers
C Safety Glasses
C Outer layer or easily removed protective clothing
C Suitable respirator selected

Revision number: 0
Date of revision: 24 Oct. 2003

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Radiation Safety Data Sheet

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Part 1 - RADIOACTIVE MATERIAL IDENTIFICATION

Chemical Symbol:	U	Common Names:	Uranium
Atomic Weight:	238	Atomic Number:	92

Note: Uranium will always contain ^{238}U with varying amounts of ^{235}U and ^{234}U.

Part 2 - RADIATION CHARACTERISTICS

Physical Half-Life: 4.468×10^{9} years
CNSC Exemption Quantity (in Bq): Dispersible (*e.g.*, powder) 1×10^{4} Non-dispersible: $1 \text{ x } 10^{7}$
A CNSC license is not required if the amount of radioactive nuclides possessed is less than one Exemption Quantity.

Principal Emissions	**Average Energy (MeV)****	**Maximum Energy (MeV)*****	**Dose Rate at 1m Distance (mSv/h□GBq)**	**Recommended Shielding**
Neutrons	-	-	-	-
Gamma & X-rays	0.1438, 0.1857[†]	-	0.021[‡]	-
Beta*	-	-	-	-
Alpha	4.174	-	-	-

* Where beta radiation is present, bremmstrahlung radiation will be produced. Shielding may therefore be required.
** Average energy of most abundant emission.
*** Maximum of most abundant emission.
[†] Both gamma rays emitted by ^{235}U.
[‡] Dose rate from natural uranium.

Progeny	Th-234(24.1d), Pa-234m(1.17 min), U-234(2.4E5 y), Th-230(7.7E4 y), Ra-226(1600 y), Rn-222 (3.8 d), Po-218(3.1 min), Pb-214(26.8 min), Bi-214(19.9min), Po-214(164 us), Pb-210(22.3y), Bi-210(5d), Po-210(138d).

Part 3 – DETECTION AND MEASUREMENT

Method of Detection: ZnS scintillation counter
Dosimetry:
External: TLD (whole body & skin) T Extremity T Neutron ______
Internal: Whole body T Thorax T Urine analysis T Other (specify) Faeces, personal air sampler

Part 4 - PREVENTATIVE MEASURES

Soluble uranium compounds are quickly removed from the blood and deposit in the kidney, liver and bone - primarily where they cause damage. Insoluble uranium compounds may affect the lungs because the deposited radioactive particles are cleared slowly. The inhalation of insoluble compounds may lead to osteocarcoma and lung cancer because of the prolonged irradiation of the thorax.

Uranium reacts with both air and water, and may form a pyrophoric surface when stored in the presence of moist air. Uranium dust is easily ignited and uranium in powder form will ignite spontaneously in air.
No protective clothing is necessary for work with sealed sources.

When working with unsealed sources wear appropriate protective clothing, such as laboratory coats (which must be monitored before leaving the laboratory), coveralls, gloves, safety glasses/goggles and a suitable mask, if the radioactive material is in the form of a dust, powder or if it is potentially volatile.

Optimize time, distance and shielding. Handle uranium and its compounds in well ventilated areas. Use metal containers for storage. Store uranyl nitrate away from organic and combustible substance. Laboratory equipment used for radioactive work must not be used for other purposes. Monitor equipment and supplies for loose contamination before removing from laboratory. Use disposable absorbent liners on trays.
Consult CNSC license for requirements concerning engineering controls, protective equipment, and special storage requirements.

Part 5 - ANNUAL LIMIT ON INTAKE*

	Ingestion		**Inhalation**		
Compound Type	Unspecified compounds	Most tetravalent compounds, e.g., UO_2, UF_4	Most hexavalent compounds, e.g., UF_6	UO_3, UF_4, UCl_4	Insoluble compounds, e.g., UO_2
Annual Limit on Intake (Bq)	5×10^5	3×10^6	3×10^4	1×10^4	4×10^3

*Note: Values are for natural uranium.

EMERGENCY PROCEDURES

The following is a guide for first responders. The following actions, including remediation, should be carried out by qualified individuals. In cases where life threatening injury has resulted, **first** treat the injury, **second** deal with personal decontamination.

Personal Decontamination Techniques
C Wash well with soap and water and monitor skin
C Do Not abrade skin, only blot dry
C Decontamination of clothing and surfaces are covered under operating and emergency procedures
Spill and Leak Control
C Alert everyone in the area
C Confine the problem or emergency (includes the use of absorbent material)
C Clear area
C Summon Aid
Emergency Protective Equipment, Minimum Requirements
C Gloves
C Footwear Covers
C Safety Glasses
C Outer layer or easily removed protective clothing
C Suitable respirator selected

It should be noted that ^{238}U is a component of natural uranium, consisting of ^{238}U, ^{235}U and ^{234}U in mass percentages of 99.2837%, 0.7110% and 0.0053%, respectively, and with ^{234}U and ^{238}U present in equal activities per gram of natural uranium.