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ANL-88-26
Corrected

Operational Health Physics Training

by H. J. Moe

E. J. Vallario
DOE Project Manager

Argonne National Laboratory, Argonne, Illinois 60439
operated by The University of Chicago
for the United States Department of Energy under Contract W-31-109-Eng-38

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OPERATIONAL HEALTH PHYSICS TRAINING

by

H. J. Moe

Environment, Safety, and Health Division

E. J. Vallario
DOE Project Manager

June 1992

Prepared for the
U. S. Department of Energy
Assistant Secretary for Environment, Safety, and Health

FOREWORD

For many years the Radiation Safety Technician Training Manual, ANL-7291 (affectionately referred to as the "Moe Handbook"), has provided the basis for technician training throughout the nuclear industry. Though a sound document, changes in radiation protection standards and measurement technology since its publication in the early 1970's suggested that a revision would be timely. Due to our keen interest in radiation protection training, the Office of Nuclear Safety, U.S. Department of Energy, was pleased to provide support for such a revision. The end result, Operational Health Physics Training, should provide a useful reference for applied health physicists and technician training courses for years to come.

We want to express our sincere appreciation to Harold Moe for his dedicated efforts in the revision to this document. It is truly a contribution to the nuclear industry.

Edward J. Vallario
Acting Director
Radiological Controls Division
Office of Nuclear Safety

TABLE OF CONTENTS

TABLE OF CONTENTS - (cont'd.)

TABLE OF CONTENTS - (cont'd.)

TABLE OF CONTENTS - (continued)

TABLE OF CONTENTS - (continued)

TABLE OF CONTENTS - (continued)

TABLE OF CONTENTS - (continued)

TABLE OF CONTENTS - (continued)

TABLE OF CONTENTS - (continued)

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

TABLE OF CONTENTS - (continued)

TABLE OF CONTENTS - (continued)

TABLE OF CONTENTS - (continued)

TABLE OF CONTENTS - (continued)

TABLE OF CONTENTS - (continued)

LIST OF FIGURES

LIST OF FIGURES - (continued)

LIST OF FIGURES - (continued)

LIST OF FIGURES - (continued)

LIST OF FIGURES - (continued)

LIST OF FIGURES - (continued)

LIST OF TABLES

OPERATIONAL HEALTH PHYSICS TRAINING

H. J. Moe

ABSTRACT

This revised publication updates a previous report (ANL - 7291), initially published in 1965, entitled " Radiation Safety Technician Training Course" which was intended to complement on-the-job monitoring training for Health Physics technicians. In addition to updated information in all the sections, additional topics and/or new or expanded sections have been incorporated. Reference sections have been extensively revised, expanded and updated. Questions and problems following the sections have been reviewed and rewritten.

The initial four sections treat basic information concerning atomic structure and other useful physical quantities, natural radioactivity, the properties of α, β, γ, x rays and neutrons, and the concepts and units of radiation dosimetry (including SI units).

Section 5 deals with biological effects and the risks associated with radiation exposure. Background radiation and man-made sources are discussed next. The basic recommendations of the ICRP concerning dose limitations: justification, optimization (ALARA concepts and applications) and dose limits are covered in Section seven.

Section eight is an expanded version of shielding, and the internal dosimetry discussion has been extensively revised to reflect the concepts contained in the MIRD methodology and ICRP 30.

The remaining sections discuss the operational health physics approach to monitoring radiation. Individual sections include radiation detection principles, instrument operation and counting statistics, health physics instruments and personnel monitoring devices.

The last five sections deal with the nature of, operation principles of, health physics aspects of, and monitoring approaches to air sampling, reactors, nuclear safety, gloveboxes and hot cells, accelerators and x ray sources. Decontamination, waste disposal and transportation of radionuclides are added topics.

Several appendices containing constants, symbols, selected mathematical topics, and the Chart of the Nuclides, and an index have been included.

SECTION 1 - BASIC INFORMATION

A. Atomic Structure

Beginning in the decade before 1900, a series of discoveries led to a profound change in the manner in which the structure of matter was viewed. Up to that time, the general view was that all matter consisted of a combination of one or more units. This view, proposed by John Dalton, assumed that each of the constituents was a small indivisible unit, called an atom (Greek: atomos-indivisible). A compound material consisted of chemical combinations of atoms, always in the same proportion of different fundamental substances. These fundamental substances were called elements, and elements formed, or were the constituents of, all matter. The term molecule was used to designate a combination of atoms, of one or more of the elements, to form compounds.

The discovery of the electron and measurement of its properties by J. J. Thomson and others, began to reveal a structure within the atom. The electron was determined to be a very small, light particle with a basic negative charge. The presence of electrons in atoms implied that the atom must also contain an equal number of positive charges-to balance the negative charges. Moreover, because the electron was so light, a large fraction of the mass of the atom must be associated with the positive charges. Thomson proposed a new concept of the atom-enough electrons embedded in a sea of positive charge to balance the charge on the atom. However, Ernest Rutherford showed from scattering experiments that all of the positive charge on the atom (and most of the mass) was concentrated at the center of the atom in a core-later called the nucleus. Furthermore, it was soon recognized that the charge of the hydrogen nucleus represented the basic unit of positive charge. That is, the positive charge of the nucleus of any other element was some integral multiple of the charge of the hydrogen nucleus. Because of the significance of the hydrogen nucleus, it was given the name proton (Greek: protos-first).

The discovery of the electron, and the subsequent recognition of the proton, showed that "sub-atomic" particles existed. The atom was not an indivisible entity. An additional sub-atomic particle was discovered by James Chadwick in 1932. His experiments showed the existence of a neutral particle (no electrical charge) with a mass about the same as a proton. He called this particle a neutron. In addition, his experiments showed that this particle was contained in the nucleus. With the discovery of the neutron, proton and electron, the classical picture of indivisible atomic structure was abandoned.

With the advent of high energy accelerators, extending the ability to observe "sub-nuclear" reactions, investigation of the nucleus has revealed levels of structure within the nucleus itself. So that, even the particles in the nucleus (neutrons and protons) have revealed certain sub-nuclear structures. Neutrons and protons have been found to be composite particles, made up of different combinations of fundamental units which are called quarks.

Our present state of understanding with respect to atomic structure has experienced a number of changes during the past century, brought on by increased technology. It is reasonable to expect that as the ability to probe deeper into the recesses of nuclear structure increases, our future understanding of nuclear structure will also improve.

For our purposes, we make use of the following features of atomic structure. An atom is the smallest unit of an element. A listing of the different elements by chemical symbol is shown in Figure 1.1. There are 90 naturally occurring elements. In addition to these, modern science has increased the number to more than 100 by artificially producing elements.

The elements are usually designated by their chemical symbols, shown in the periodic chart (Figure 1.1). Hydrogen, for example, has the symbol H; oxygen has the symbol O; etc. A listing of the elements by name, as well as their chemical symbols can be found in Appendix G.

An atom consists of a nucleus surrounded by a sufficient number of electrons to provide electrical neutrality. The nucleus is the massive, central, positively charged portion of the atom, composed of neutrons and protons. The neutron and the proton are particles of about the same mass, but the proton has a positive charge and the neutron is electrically neutral. A method of denoting atomic structure which is often used is ${}^{A}_{Z}X$, where X is the chemical symbol of the element. The quantity A is the mass number which gives the number of protons and neutrons in a

PERIODIC CHART OF THE ELEMENTS

1 H																	2 He
3 Li	4 Be											5 B	6 C	7 N	8 O	9 F	10 Ne
11 Na	12 Mg											13 Al	14 Si	15 P	16 S	17 Cl	18 Ar
19 K	20 Ca	21 Sc	22 Ti	23 V	24 Cr	25 Mn	26 Fe	27 Co	28 Ni	29 Cu	30 Zn	31 Ga	32 Ge	33 As	34 Se	35 Br	36 Kr
37 Rb	38 Sr	39 Y	40 Zr	41 Nb	42 Mo	43 Tc	44 Ru	45 Rh	46 Pd	47 Ag	48 Cd	49 In	50 Sn	51 Sb	52 Te	53 I	54 Xe
55 Cs	56 Ba		72 Hf	73 Ta	74 W	75 Re	76 Os	77 Ir	78 Pt	79 Au	80 Hg	81 Tl	82 Pb	83 Bi	84 Po	85 At	86 Rn
87 Fr	88 Ra		(104)	(105)	(106)												
			57 La	58 Ce	59 Pr	60 Nd	61 Pm	62 Sm	63 Eu	64 Gd	65 Tb	66 Dy	67 Ho	68 Er	69 Tm	70 Yb	71 Lu
			89 Ac	90 Th	91 Pa	92 U	93 Np	94 Pu	95 Am	96 Cm	97 Bk	98 Cf	99 Es	100 Fm	101 Md	102 No	103 Lr

Figure 1.1 Periodic Chart - Elements with a similar chemical properties are shown in the same column; elements of increasing atomic number Z are arranged in rows.

nucleus, and Z is the atomic number of the element, which gives the number of protons (or positive charges) in the nucleus, and the number of surrounding electrons. The number of neutrons in the nucleus is then given by A-Z. Inasmuch as the proton and the neutron each have approximately unit mass on the scale of atomic weights, the mass number is also that integer closest to the atomic weight. See Appendix A for the mass of the neutron, proton and electron, as well as for other useful constants.

The radius of the nucleus, in meters, is given approximately by

$$r = 1.3 \times 10^{-15} A^{1/3}. \qquad 1.1$$

The diameter of the nucleus is less than 1/10,000 that of the atom itself. Almost the entire mass of the atom is concentrated in the nucleus. The nuclear density is about 10^{17} kg/m^3.

The atomic mass of an element may be expressed in unified atomic mass units (u). One u, on the physical scale, is 1/12 of the mass of a carbon-12 atom. The mass of any atom, in grams, is equal to its atomic mass divided by Avogadro's number N_a (6.022×10^{23} molecules/mole). The reciprocal of Avogadro's number ($1/N_a$) is equal to the mass, in grams, of 1 u, i.e., $1\ u = 1.6606 \times 10^{-24}\ g = 1.6606 \times 10^{-27}$ kg.

The number of atoms in a certain mass of an element is given by the expression:

$$N = \frac{m}{A} N_a \qquad 1.2$$

where m is the mass of the element in grams. For example, the number of atoms in one gram of ^{238}U is calculated to be:

$$N = \frac{m}{A} N_a = \frac{1}{238}(6.022 \times 10^{23}) = 2.53 \times 10^{21} \text{ atoms.}$$

The gram-atomic weight or gram-molecular weight is often used in chemistry. These quantities simply mean a weight in grams of an element which is equal to its atomic or molecular weight. A gram-atomic weight of ^{238}U is 238 g of ^{238}U; a gram-molecular weight (often called a mole) of H_2O is 18 g. A gram-atomic weight of an element contains 6.022×10^{23} atoms; a gram-molecular weight contains 6.022×10^{23} molecules.

Note, that in SI units, the mole is equal to the number of molecules in a kg-molecular weight. A mole of ^{238}U would then be 0.238 kg of ^{238}U. In equation 1.2, if we express m in kg, then A must be divided by 1000 to express the gram-molecular weight as kg-molecular weight.

With respect to the arrangement of electrons within the atom, Niels Bohr provided a theory which was based upon the observed light from hydrogen atoms. He assumed that electrons revolved around the central nucleus in circular orbits. However, only certain fixed, or quantized, orbits were possible. These stationary orbits were described by a quantum number n, which could have integral values from 1 to 7. While an electron remained in a stationary orbit, it existed in a definite energy state and no light would be emitted. However, if an electron moved from one stationary orbit to another, light would be emitted. Moreover, the amount of light emitted was related to the difference in the energy states of the two orbits.

The Bohr theory explained spectra from hydrogen, but was inadequate when applied to other elements. Arnold Sommerfeld showed that elliptical orbits were also possible, and this introduced another quantum number, now called l. Two more quantum numbers, m and s, were needed to completely describe the energy state of an electron. In addition, the application of mathematical theory, by use of quantum mechanics, led to successful interpretation of many other atomic properties. From the quantum mechanics standpoint, the atom may be viewed as consisting of a nucleus surrounded by a cloud of electrons. Nevertheless, the simple physical picture of the atom provided by Bohr still serves as a useful representation.

The atom may be pictured as a planetary system of electrons in motion about the nucleus. An example, in this case of the fluorine atom, is shown in Figure 1.2. The electrons rotate in elliptical and circular orbits about the nucleus. These well-defined orbits comprise a series of shells which are called energy or quantum levels. The diameters of the orbits, and consequently of the shells, are large in comparison with the

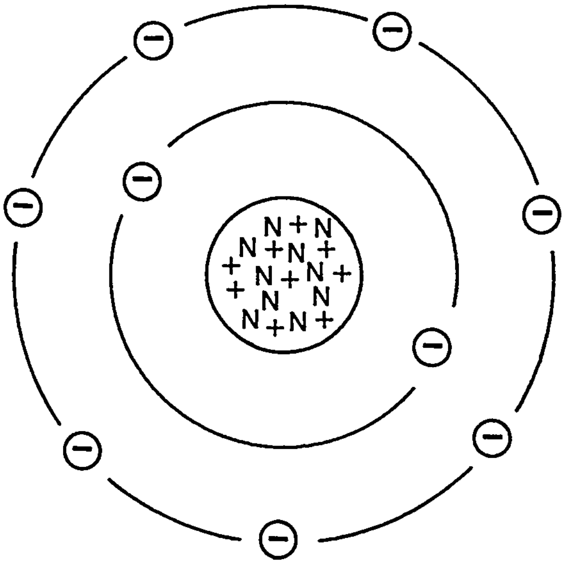

Figure 1.2
The stable fluorine atom ($^{19}_{9}F$).
The nucleus contains 9 protons
and 10 neutrons.

nuclear diameter. The shells are denoted by either a letter (K, L, M, N, O, P, Q) or quantum number (1, 2, 3, 4, 5, 6, 7). The energy state of each electron in a shell is completely described by four independent quantum numbers (n, l, m, and s). The Pauli Exclusion Principle states that no two electrons in any one atom can be identical, that is, have all four quantum numbers the same. This principle sets an upper limit to the number of possible electrons in each level. This limit is given by $N_e = 2n^2$, where n is the principal quantum number of the shell, which may take on integral values 1, 2, 3, 4, 5, 6, 7 for the corresponding shell of the atom.

The electrons in an atom influence the chemical, mechanical and electrical properties of the atom. The outer electrons exert the forces which hold atoms together in chemical compounds. In metals, loosely bound outer electrons aid in the conduction of electricity.

B. Ionization

An electron can be removed fairly easily from an atom or molecule.

One or more electrons may also be added to such entities. The resulting component has an electric charge because the number of protons exceed the number of electrons. The term ion is used to define atoms or groups of atoms carrying an electric charge. The charge each ion carries depends upon the number of electrons which have been added (negative) or removed (positive). Ionization is a process whereby electrons may be removed from, or added to, a neutral atom or molecule. The removal of an electron from an atom or molecule (see Figure 1.3 for the process of electron removal from an atom) results in an ion pair: the free electron and the positively charged residue.

The normal state of an atom, in which it is stable, is the ground state. If energy is supplied to the atom, the atom is said to be in an excited state. If enough energy is added to remove electrons, the atom will be ionized. The electron binding energy, or work function, is the amount of energy needed to remove an electron from the atom. Ionization

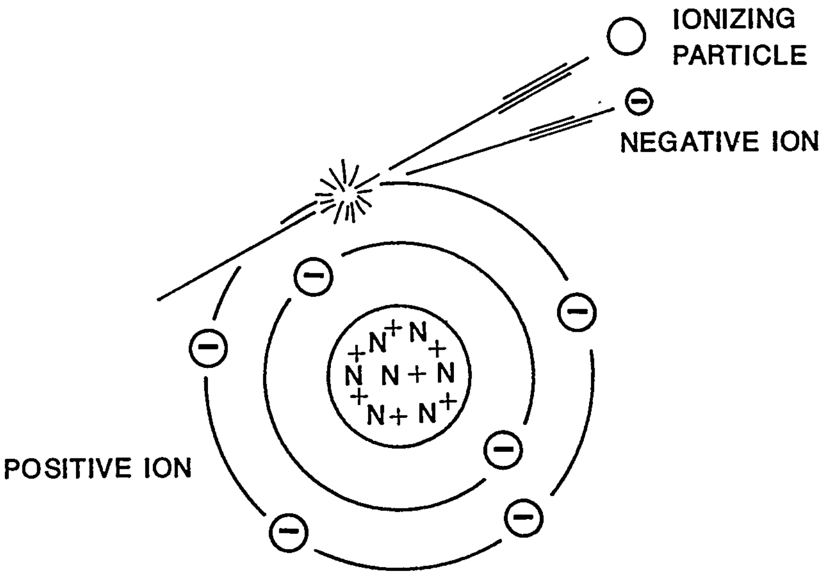

Figure 1.3 The ionization of an atom.

and excitation of atoms can be produced directly or indirectly by radiation interacting with matter. Excited atoms lose energy through emission of light and return to the ground state.

C. Isotopes, Nuclides, Isobars

It was mentioned that over 100 elements are known to man. Each element is characterized by its atomic number. However, an element may have atoms possessing different masses. The term isotope (Greek: is topos-same place) is used to denote elements with the same atomic number but different atomic masses. Since the atomic number is the same, isotopes have the same number of protons within the nucleus, but the number of neutrons vary. An element may have one or more isotopes. Isotopes exhibit similar chemical properties.

The term nuclide is used to generally designate an atomic species containing neutrons and protons. The neutrons and protons are often referred to as nucleons. The Chart of the Nuclides can be found in Appendix G.

There are nuclides which have the same atomic mass number, but possess different atomic numbers. These nuclides are referred to as isobars (Greek: is baros-same weight). Isobars have similar nuclear properties.

Some nuclides are unstable, releasing energy in the form of particles and/or electromagnetic radiation by a process of decay or disintegration called radioactivity.

Within the nucleus, a repulsive force, the Coulomb force, exists between the protons, which tends to push the nucleus apart. Such forces are relatively long-range forces. In addition, there are very short-range attractive forces between the nucleons which tend to hold the nucleus together. Similar to the atomic binding energy for electrons, there is a nuclear binding energy which holds the nucleons. As the number of nucleons changes, the relative effectiveness of each type of force varies. The net effect of these forces will determine whether a certain combination or ratio of neutrons and protons is stable.

The neutron to proton ratio (n/p) varies from about 0.5 - 1.6 for most nuclides, except for ^1H. In the lighter nuclides, for even mass number A, when n = p, the nuclide is generally stable. For odd A, stability is found for one excess n and instability for one excess p. When Z>20, an excess of neutrons are needed for stability. Above Z=84, all nuclides exhibit instability, generally attributed to the effect of the very large nuclear mass.

D. Particle Dynamics

A particle has the property of inertia, that is, a resistance to change in its velocity. The mass of a particle is a measure of this inertia. The greater the resistance to a change in velocity, the greater will be the inertia, and thus the mass of the particle. The unit of mass in the SI system is the kilogram (kg). There are about 0.454 kg to a pound.

The velocity of a particle is a vector quantity in which both speed and direction must be specified. The magnitude of the velocity (that is, the speed) is given in m/s in the SI system. The acceleration of a particle is defined as the time rate of change of velocity.

The force F which acts on a particle is defined in terms of the product of the acceleration a and the mass m of the object:

$$F = ma. \qquad 1.3$$

If the acceleration is constant,

$$F = \frac{m(v - v_o)}{t} = \frac{mv - mv_o}{t}. \qquad 1.4$$

This equation is known as the impulse equation. The quantity mv is called the momentum of the particle, so that one can measure the magnitude of a force from the time rate of change of the momentum. The unit of force in the SI system is the newton (N): a newton is the force which will give to one kg an acceleration of 1 m/s^2, that is 1N = 1 kgm/s^2. The quantity mv is often called the linear momentum (p). When a particle is rotating at a radial distance r from a central point, it will possess an angular momentum (L),

$$L = mvr. \quad 1.5$$

An important concept in physics is Conservation of Momentum. When no external force acts on a system, the total momentum of the system remains constant.

E. Work and Energy

Work W is usually defined as the product of a force F and the distance d through which an object moves under the action of the force:

$$W=Fd. \quad 1.6$$

The term energy is defined as the capacity for doing work. An object has energy by virtue of its position and by virtue of its motion. Potential energy is energy of position; e.g., a book resting on the floor has potential energy equal to zero if we consider the floor as a reference level. In order to give the book potential energy we must do work on it to change its position. For example, we might pick it up and put it on a table. In raising the book to the table, the potential energy of the book will now be equal to the work we have done in raising it to that height. This work can be recovered if the book falls off the table and returns to its original position on the floor.

Kinetic energy is energy of motion. It is given by the equation

$$E_k = 1/2mv^2. \quad 1.7$$

The unit of work or energy in the SI system is the joule (J). The joule= newton meters = kgm^2/s^2. When we have the book on the floor or table, its kinetic energy will be zero. If the book falls, it will gain kinetic energy as it approaches the floor. During its flight, it will lose potential energy. However, at any point in its flight, the sum of the two energies will be constant and equal to the work done in raising the book to the table.

Another important concept in physics is Conservation of Energy. If no work is done on the system or by the system, the sum of the potential and kinetic energies of the system of objects in motion remains constant. In the case of nuclear reactions, the sum of the mass and energy of the particles before, and after, the reaction must remain constant. This is a consequence of the interchangeability of mass and energy as shown by Albert Einstein.

F. Charged Particles

If one looks at the forces exerted by charged particles, two types of forces may be noted, depending upon whether the charges are at rest or in motion. The electrostatic force is the force of attraction or repulsion between charges at rest. The electromagnetic force is produced by charges in motion. The moving charges, which we refer to as an electric current, produce a changing electric field which gives rise to a magnetic field which can exert magnetic forces on other charges.

The expression which gives the force of attraction or repulsion between two electric charges of magnitudes q_1 and q_2, separated by a distance r, is called Coulomb's law and is expressed as:

$$F = \frac{Kq_1q_2}{r^2}, \qquad 1.8$$

where K is a constant.

Charges which are alike, repel; charges which are unlike, attract. The constant K depends upon the units in which F, q_1, q_2 and r are expressed.

The unit of charge in the SI system is the coulomb (C). Stated in terms of Coulomb's law: one coulomb is that charge which will repel an exactly equal charge with a force of 9 x 10^9 newtons when the two are one meter apart in a vacuum. The magnitude of the fundamental charge on the proton and electron is 1.602 x 10^{-19} C. The coulomb is a large unit; it takes 6.24 x 10^{18} electrons or protons to equal a charge of one coulomb.

When two parallel plates are connected to a battery (see Figure 1.4), a negative charge appears on one plate and an equal positive charge on the other plate. In the region between the plates, an electric field exists which will exert a force on a charge placed in the field. The field strength E is defined by the force exerted on a unit charge in the field:

$$E = F/q, \text{ and } F = Eq \qquad 1.9$$

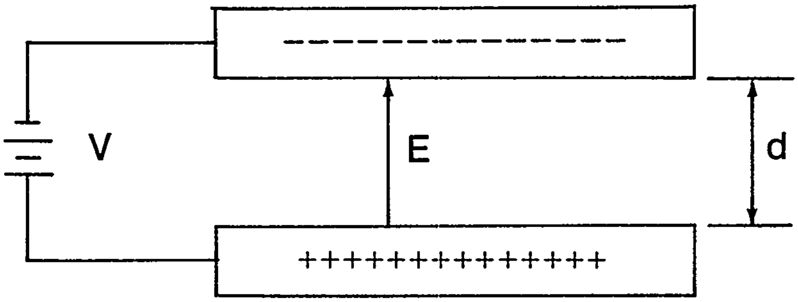

Figure 1.4 The electric field E is perpendicular to the plates. (R.E.Lapp / H.L.Andrews, NUCLEAR RADIATION PHYSICS, 2/e 1954, p18. Reprinted by permission of Prentice-Hall, Inc.,Inglewood Cliffs,NJ)

The term potential difference, symbolized by V, is defined as the work per unit charge done in moving a charge from one plate to the other:

$$V = \frac{W}{q} \qquad 1.10a$$

and $$W = Fd = Eqd \qquad 1.10b$$

so $$V = Ed \qquad 1.11$$

If W is in joules and q in coulombs, then V will be in volts. Note that the electron field strength E may also be expressed by V/d (volts/m). This is a more commonly used designation.

In the parallel-plate example, the electric field E is uniform, that is, the force on a charge in the field is constant, both in direction and magnitude. In the case of cylindrical electrodes, the field is not uniform, but varies with radius r from the central wire. In this type of arrangement, one will usually have a central wire of radius r_1, and an inner radius of the cylinder r_2. The expression for the field is then:

$$E = \frac{V}{r\ln(r_2/r_1)} \qquad 1.12$$

where V is the voltage across the electrodes. The field gets stronger as one approaches the central wire, so that the force on a charge increases as the charge nears the wire.

Devices used to detect radiation have utilized both parallel plate and cylindrical electrode designs. Parallel plate chambers require very high field strengths in order to achieve gas amplification. The latter occurs when the ions formed by interacting radiation acquire enough energy to create more ions before collection occurs. In cylindrical designs, high field strengths can be reached without the use of very high voltages.

The velocity of an electron is related to the potential difference V by the expression:

$$Ve = \frac{1}{2}mv^2 \qquad 1.13$$

where Ve is expressed in joules. This relationship is valid if the velocity of the electron is much less than that of light and the scattering of electrons is negligible (high vacuum). When the speed of the electron becomes greater than 1/10 that of light, the mass of the electron will increase and the above relationship will not hold.

The usual SI unit of energy is the joule. In nuclear physics, however, the electron volt (eV) is used as the energy unit. By definition, the electron volt is the amount of work done in accelerating an electron by a potential difference of one volt. It is related to the joule by: 1 eV = 1.6×10^{-19} J. Common multiples are: 1 keV = 10^3 eV, 1 MeV = 10^6 eV and 1 GeV = 10^9 eV.

As was pointed out earlier, the movement of a charged particle produces a current. One may say that a current is simply a flow of charge. This flow can occur between parallel plates connected to a battery (see Figure 1.4) or be induced by some other voltage source (such as a generator - a device which converts mechanical energy into electrical energy). This flow may also take place in a wire. In metals, some outer electrons are loosely bound and can be easily removed. When an electric field is applied, these electrons move under the influence of the applied force. One may say that electrons can be pumped, very much like water is pumped. The voltage, which is applied, produces a pressure of electrons at one end, and the electrons tend to flow towards the other end to equalize the pressure. Generally, the flow of electrons begins from the source of voltage, moves through various components to perform work and returns to the source. For this reason, the route taken by the flow of electrons is referred to as a circuit.

The current (I) in the wire is a measure of the flow of electrons. It may be expressed as the charge (Q) per unit time which passes a fixed point:

$$I = \frac{Q}{t} \qquad 1.14$$

The unit of current is the ampere (A), which is one coulomb per second, 1A = 1 C/s. The direction of the current is defined as the direction in which a positive charge would move. Note that the flow of electrons will actually be in the opposite direction.

Another factor which affects electron flow in a circuit is called resistance. The resistance in the circuit influences how much current will flow. George Ohm found that in many materials, the current was directly proportional to the applied voltage. This is expressed as Ohm's Law:

$$I \propto V \quad \text{or} \quad I = \frac{V}{R} \qquad 1.15$$

in which $\frac{1}{R}$ is the constant of proportionality, and R is called the resistance and is measured in units called Ohms (Ω): 1 Ω = 1 V/A. As is seen from Ohm's Law, resistance impedes current.

Going back to Figure 1.4, the plates, on which buildup of the charges occur, represent two conducting surfaces separated by a non-conducting surface (the gap). This arrangement defines a capacitor, which has the ability to store charge. The amount of charge (Q) which builds up on the plates is directly proportional to the applied voltage (V). This may be written

$$Q = CV, \qquad 1.16$$

in which C, called the capacitance, is the constant of proportionality. Capacitance is measured in units of farad (F), which is the coulombs per volt of stored electricity.

Complex circuits consist of combinations of resistors, capacitors and transistors (semiconductor devices - made from silicon or germanium-used to control and amplify current). These circuits can now be put together on one small piece of silicon. An integrated circuit (IC) is a complete electronic circuit. These circuits may contain transistors, resistors and capacitors, along with the interconnecting conductors, contained in a silicon chip as small as 1/16 of an inch square.

G. Relativistic Corrections

Among the new concepts which Einstein contributed to modern physics in his special theory of relativity are:

1) No measured velocity can exceed the speed of light c in a vacuum.

2) A body does not have a constant mass m. The mass of an object depends upon its speed v:

$$m = \sqrt{\frac{m_o}{1 - \beta^2}} \tag{1.17}$$

where $\beta = v/c$, and m_o is the rest mass.

3) Mass and energy are interchangeable: $E = (\Delta m)c^2$, or, for our purposes,

$$E = mc^2. \tag{1.18}$$

The relativity theory considers a rest mass m_o and a mass m which an object has by virtue of its speed v. From statements 2) and 3), we get

$$E = mc^2 = m_o c^2 / \sqrt{1 - \beta^2}, \tag{1.19}$$

where E represents the total energy content of an object of rest mass m_o moving with a speed v. The energy content of the rest mass is found by setting $v = 0$, namely, $E_o = m_o c^2$. The kinetic energy of the object is then:

$$E_k = E - E_o = m_o c^2 \left[\frac{1}{\sqrt{1 - \beta^2}} - 1\right] \tag{1.20}$$

and the momentum is

$$p = \frac{m_o v}{\sqrt{1-\beta^2}}. \qquad 1.21.$$

The concept that mass is another form of energy, and that each may be converted into the other, means that even a small amount of mass contains a very large amount of energy. The correctness of this concept has been amply demonstrated in the release of energy in fission (splitting a nuclide) and fusion (forming one nuclide from two smaller nuclides). If one adds up the masses of the neutrons and protons in a given nucleus, all nuclei (except hydrogen) have masses less than the sum of the nucleon masses. The difference in mass can be related to the nuclear binding energy which is about 8 MeV/nucleon for most nuclides. However, for the very light nuclides, the binding energy is much smaller than this. The value peaks at about 8.5 MeV/nucleon for mass numbers 40-120 and decreases to about 7.6 MeV/nucleon for uranium. This indicates that combining two light nuclei (fusion) will result in a new heavier nucleus which has a lighter mass than the sum of the original masses. The mass difference would appear as released energy. Combining the mass of two nuclides heavier than iron to form uranium results in a heavier mass than the sum of the masses of the two nuclides. This indicates that energy would be released if the heavy nucleus could be split up (fission) into two smaller nuclei.

H. Charged Particle in a Magnetic Field

A charged particle which moves through a magnetic field will experience a force acting upon it if the magnetic field has a component at right angles to the direction of motion of the particle. In particular, if the entire field is perpendicular to the direction of motion of the particle, the force acting on the particle will be:

$$F = Bev, \qquad 1.22$$

where B is the magnetic field strength, e is the charge on the particle, and v its velocity. The force will be perpendicular to B and to the direction of v, and the particle will tend to move in a circle of radius r (see Figure 1.5).

In considering circular motion, the centrifugal force exerted on the particle of mass m is given by

$$F = mv^2/r. \qquad 1.23$$

Accordingly, for a magnetic field acting on a charged particle moving at right angles to the field, the radius r may be found from

$$Bev = mv^2/r \qquad 1.24$$

$$r = \frac{mv}{Be}. \qquad 1.25$$

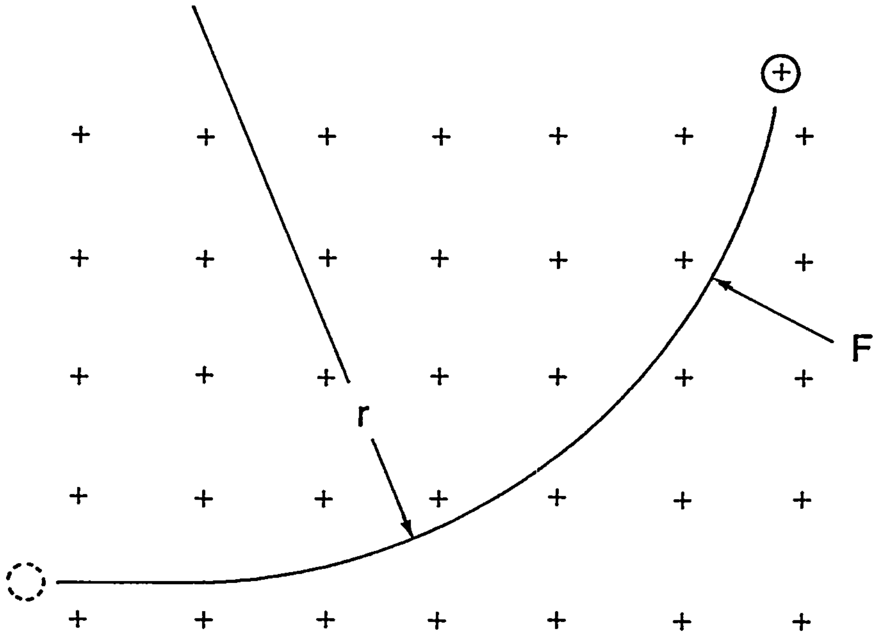

Figure 1.5 A charged particle traveling perpendicular to a magnetic field which is directed into the plane of the paper.

For B in tesla (T), e in coulombs, m in kg and v in m/s, r will be given in m. The tesla is the SI unit for magnetic field strength, 1T = 10^4 Gauss.

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QUESTIONS

1.1 Where does the nucleus reside in an atom?

1.2 In the method $^{A}_{Z}X$, used to denote an atom, what are A, Z and X?

1.3 What does (A-Z) indicate?

1.4 How many times larger is the atom than its nucleus?

1.5 Upon what atom is the unified mass unit u based?

1.6 What is gram atomic weight? Gram molecular weight?

1.7 How is Avogadro's number related to gram atomic weight?

1.8 What are the two methods to designate the electron shells?

1.9 State Pauli's exclusion principle.

1.10 What is an ion?

1.11 What is an isotope? Isobar?

1.12 What force, due to electrical charge on the protons, tends to push the nucleus apart?

1.13 What are the units of work, energy, and force? Which two units are the same and why?

1.14 What term indicates (a) the energy due to position? and (b) the energy due to motion?

1.15 Consider the formula $F = K \frac{q_1 q_2}{r^2}$

what does K represent? What are its units?

1.16 What is the unit of potential?

1.17 At what point between oppositely charged, concentric, cylindrical electrodes is the electric field greatest?

1.18 What is rest mass?

1.19 Assume that an observer is looking into a horizontal magnetic field directed toward him.

a) In what direction will a proton curve if it enters the field in front of the observer in a vertically downward direction?

b) In what direction will an electron curve if it enters the field in front of the observer moving from right to left?

1.20 A person pushes a stalled truck. The truck does not move. Is work done on the truck? Why not?

PROBLEMS

1.1 Compute the nuclear radii of

a) ${}^{1}_{1}H$ b) ${}^{3}_{1}H$ c) ${}^{131}_{53}I$ d) ${}^{238}_{92}U$

Answers:
a) 1.30×10^{-15} m
b) 1.87×10^{-15} m
c) 6.60×10^{-15} m
d) 8.06×10^{-15} m

1.2 Find the number of atoms in 0.006 kg of lead, atomic weight 207.

Answer: 1.746×10^{22} atoms

1.3 Find the gram molecular weight (mole) of sulphuric acid, H_2SO_4. The atomic weights are: H 1.00, S 32.06, O 16.00.

Answer: 98.06 g.

1.4 Find the mass in kilograms of a single atom of thorium, atomic weight 232.

Answer: 3.85×10^{-25} kg

1.5 Find the maximum number of electrons that an N shell can accommodate without violating the Pauli Exclusion Principle.

Answer: 32

1.6 The mass of a ^{12}C atom is experimentally found to be 19.92637×10^{-27} kg. Assuming the carbon nucleus to be spherical, compute the nuclear density of $^{12}_{6}C$.

Answer: 1.8×10^{17} kg/m^3

1.7 Find the value of Avogadro's number using the mass of $^{12}_{6}C$ given above and the definition of unified mass unit.

Answer: 6.022×10^{23} mol/mole

1.8 What force must be applied to a helium ion, atomic mass, 4.00u, to cause it to accelerate 100 m/s^2. Ignore Einstein correction.

Answer: 6.64×10^{-25} N.

1.9 Assuming a constant force of 10^{-15} N, how long would it take to accelerate a lithium ion of 6.0150 u, from rest to 10^4 m/s.

Answer: 9.99×10^{-8} s.

1.10 A neutron ($m = 1.675 \times 10^{-27}$ kg) passes two points 10 m apart in a time interval of 2×10^{-4} s. Assuming its speed is constant, what is its kinetic energy in joules?

Answer: 2.1×10^{-18} J.

1.11 In the formula $F = K \frac{q_1 q_2}{r^2}$, it is a standard practice to write the constant K in the form $K = \frac{1}{4\pi\epsilon_o}$, where ϵ_o is called the permitivity of space. Compute the value of the permitivity of space.

Answer: $\epsilon_o = 8.84 \times 10^{-12}$ C^2/N.m^2

1.12 Find the magnitude of the electrostatic force of attraction between the proton and electron in the hydrogen atom. Assume the charge on proton to be 1.6×10^{-19} C and the charge on electron to be -1.6×10^{-19} C and the distance of separation, is 5.3×10^{-11} m.

Answer: -8.2×10^{-8} N.

1.13 A potential difference of 1000 V exists between two parallel plates separated by 0.004 m. What is the electric field between the plates?

Answer: 2.5×10^{5} V/m.

1.14 The inside diameter of a cylindrical Geiger-Müeller tube is 0.02 m and the diameter of the center electrode is 5×10^{-4} m. Find the electric field at a distance of 0.001 m from the centerline of the tube when the applied potential is 900 V.

Answer: 2.44×10^{5} V/m.

1.15 Find the velocity gained by an electron upon passing from one plate to another between which the potential difference is 100 V. Assume the charge on the electron to be 1.6×10^{-19} C and the mass to be 9.11×10^{-31} kg.

Answer: 5.93×10^{6} m/s

1.16 The rest mass of an electron is 9.1095×10^{-31} kg. What is its rest energy in MeV? (Use 2.9979×10^{8} m/s as the speed of light.)

Answer: 0.511 MeV

1.17 The rest mass of a proton is 1.6726×10^{-27} kg. What is its rest energy in MeV?

Answer: 938.3 MeV

1.18 Express 1 u (unified mass unit) in MeV.

Answer: 931.6 MeV

1.19 Using the relativistic correction due to Einstein, find the mass of a proton when it gains a velocity of $2x10^{8}$ m/s from rest. Use $3x10^{8}$ m/s as the speed of light and $1.6726x10^{-27}$ kg as proton rest mass.

Answer: $2.244x10^{-27}$ kg

1.20 An electron is moving at 1/10 of the speed of light.

a) What is the difference in kinetic energy calculated with Einstein's relativistic formula and the non-relativistic formula?

b) What is the percent difference?

Answers: a) 19 eV
b) < 1%

1.21 Show that the binding energy of a deuteron is approximately 2.2 MeV. Assume the following:

deuteron mass:	2.014102 u
proton mass:	$1.672649x10^{-27}$ kg
neutron mass:	$1.674954x10^{-27}$ kg
electron mass:	$9.109534x10^{-31}$ kg

1.22 An electron moving perpendicularly through a magnetic field of 0.05 T has a velocity of $6x10^{6}$ m/s. What is the radius of the circular path of the electron?

Answer: $6.83x10^{-4}$ m

SECTION 2 - RADIOACTIVITY AND ITS PROPERTIES

A. Early History

Henri Becquerel is credited with the discovery of natural radioactivity. While working with uranium salts and photographic plates in 1896, he found that the uranium emitted a penetrating radiation which seemed similar to that which Roentgen had produced a year earlier in experiments with a gas discharge tube. Madame Curie called this phenomenon radioactivity. Further investigation by her and others showed that this property of emitting radiation is a specific property of the given element and that certain quantitative relationships exist. They also found that the radiations are continually emitted so that energy is constantly lost. These radiations, and therefore the energy, come from the nucleus of the atom. Also, the energy released by radioactive materials is about a million times more intense than the energy released in standard chemical reactions. Moreover, the atoms producing these radiations are unstable and emit radiation at characteristic rates to form new atoms. These, in turn, may transform to other substances, but the end product will be more stable.

By the use of magnetic fields, it was shown that there are three distinct types of radiation (see Figure 2.1). These were given the designations: alpha (α), beta (β), and gamma (γ) radiation. The first two types can be deflected in magnetic fields, indicating that they are charged particles. Gamma radiation cannot be deflected in magnetic fields. It is not a charged particle but a form of electromagnetic radiation, similar to light.

Since α particles were deflected in a different direction in the magnetic field than were β particles, the two were oppositely charged. From the direction in which the α particle was deflected, it was shown to be positively charged, whereas the β particle is negatively charged. Further investigation revealed that the α particle is actually a helium nucleus ($^{4}_{2}He$ - two positive charges, mass number 4) and the β particle is actually an electron (e^{-} - one negative charge, mass of 0.000549 u).

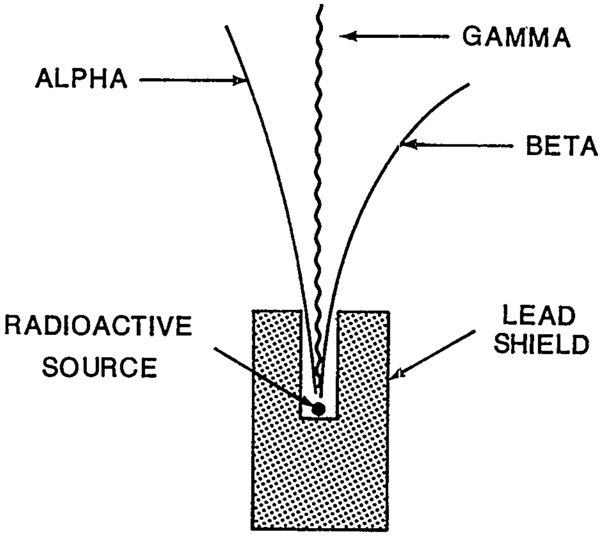


Figure 2.1 Radiation deflection in a magnetic field. The field is perpendicular to and directed into the plane of the paper. (R.E.Lapp / H.L.Andrews, NUCLEAR RADIATION PHYSICS, 2/e 1954, p.73. Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, NJ).

Following Becquerel's discovery, further studies revealed that three naturally occurring radioactive series existed. The parent, or first member, of these series, or chains, are ^{232}Th, ^{238}U and ^{235}U and the series are called the thorium, uranium and actinium series, respectively. These chains have certain common properties: each parent has a very long half life (time for half the initial nuclei to decay - see 2.D); each has a gaseous member which is a radon (Rn) isotope; and each ends up as a stable lead (Pb) isotope. An artificially occurring series, called the neptunium series, beginning with ^{241}Pu but named after the longest-lived radionuclide in the series, ^{237}Np, was later produced by neutron bombardment of ^{238}U.[1]

With the discovery in 1934, that radioactivity can be induced in a stable atom, a number of additional decay modes, have emerged (such as electron capture and isomeric transitions). In addition, a number of new particles, associated with radioactive decay, have also been uncovered. These are the neutrino and the anti-neutrino. These particles have been useful in the explanation of β decay. The neutrino and anti-neutrino are almost massless particles ($< 10^{-3}$ of the electron mass) which carry away energy in decays. Also, the anti-particle to the electron, the positron, was discovered, as well as radioactive decay in which positrons are emitted. In addition, the advent of high energy accelerators has increased the complexity of the number of known particles and their decay mechanisms.

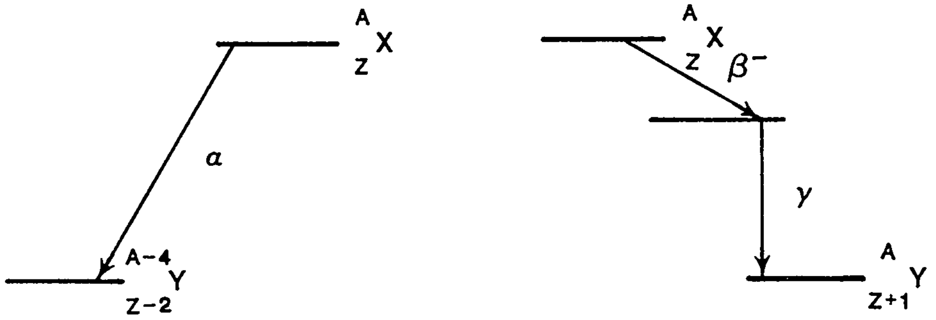
Other naturally occurring radionuclides, such as ^{40}K, ^{147}Sm and ^{187}Re, etc., which are not members of long series, or decay chains, have also been found.

B. Radioactive Transformations

When an atom undergoes a transition, or decays, by the emission of an alpha particle, the atomic number Z decreases by 2 and the mass number A decreases by 4, so that

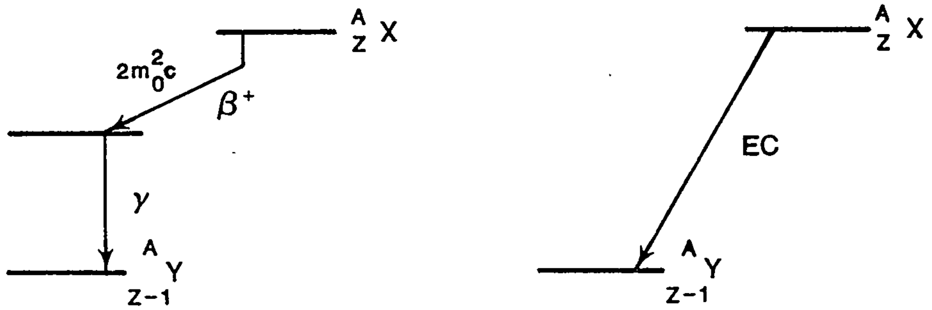
$$^{A}_{Z}X \rightarrow {}^{A-4}_{Z-2}Y + {}^{4}_{2}He$$

This can be shown by a diagram called a decay scheme (see Figure 2.2.a). In this diagram, different energy states are indicated by a vertical scale. In the figure, the ground state of the parent atom is shown at a higher vertical position (or higher energy level) than the ground state of the decay product (called the daughter product). In the decay scheme layout, the emission of positively charged radiation is shown by an arrow to the left; negatively charged radiation by an arrow to the right. An example of the α decay is $^{238}U \rightarrow {}^{234}Th + {}^{4}He$, in which ^{238}U is the parent, ^{234}Th is the daughter and He is the α particle. In the diagram shown, the α is emitted with a discrete energy which equals the net energy difference between the energy level of $^{A}_{Z}X$ and that of $^{A-4}_{Z-2}Y$ (accounting for the masses of the parent, daughter and the α particle as well as the recoil energy of the daughter). In some cases, the emission of the α will not bring the decaying nuclide to the ground state of the daughter, so that emission of a γ ray may follow.



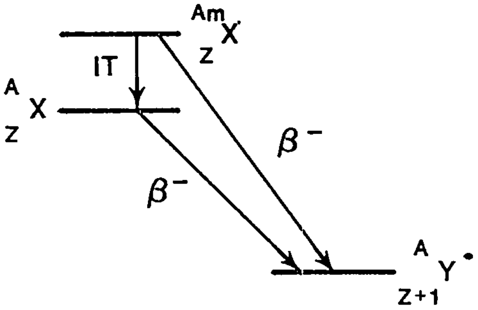
a) Alpha Decay

b) $\beta^- + \gamma$ Decay



c) $\beta^+ + \gamma$ Decay

d) EC Decay



e) IT Decay

Figure 2.2 Decay schemes for various modes of decay: α, β^-, β^+, γ, EC and IT.

When the transformation is by negative β emission, the atomic number Z increases by 1, but the mass number A remains the same. The antineutrino $\bar{\nu}$ carries off excess energy. Often, a γ ray is emitted following a β^- decay (see Figure 2.2.b). Negative β emission is likely to occur if the n/p ratio is too high. The decay is given by

$${}^{A}_{Z}X \rightarrow {}^{A}_{Z+1}Y + \beta^- + \bar{\nu}$$

in which β^- represents the electron, ${}^{0}_{-1}e$. The emission of a gamma ray (as shown in Figure 2.2b) does not affect either the atomic number or the mass number. It is indicated by a vertical straight line in the figure.

Positron (β^+ or ${}^{0}_{1}e$) emission is shown in Figure 2.2c. The reaction is

$${}^{A}_{Z}X \rightarrow {}^{A}_{Z-1}Y + \beta^+ + \nu$$

in which ν is the neutrino that carries away excess energy. As in the case of β^- decay, a γ ray may be emitted after positron emission. For positron emission to occur, the parent mass must exceed the daughter mass by more than two electron masses ($2\ m_oc^2$). An alternate mode, which often competes with positron decay is electron capture (EC), pictured in Figure 2.2d. It should be noted that some diagrams picture β^+ decay in the same manner as electron capture. Of course, if the condition for positron emission stated above is not met, then only electron capture may occur. For this reaction,

$${}^{A}_{Z}X + {}^{0}_{-1}e \rightarrow {}^{A}_{Z-1}Y + \nu.$$

The electron which is captured is most often a K shell electron, although L and M capture are also possible. X rays which are emitted following electron capture will be those from element Y. Positron decay and/or electron capture are likely to occur if the n/p ratio is too low.

Sometimes a nucleus in an excited state will emit energy as a γ ray, rather than emit a particle. This is called an isomeric transition. As stated before, this emission does not affect the atomic number or the mass number, so

$$ {}^{A}_{Z}\overset{*}{X} \rightarrow {}^{A}_{Z}X + \gamma . $$

The asterisk indicates that the nucleus is in an excited state. In Figure 2.2e, the excited state is indicated by ${}^{Am}X$, where the m stands for metastable. In this decay scheme, two paths are available. A photon may be emitted from the excited level to the ground state of ${}^{A}X$, followed by β^- emission. The alternate mode involves emission of a higher energy β^- in the transition to ${}^{A}_{Z+1}Y$. The branching ratio gives the fraction of the total transitions which proceed by each given mode.

An alternate mode of decay for an excited nucleus is called internal conversion. Instead of the emission of the γ, the nucleus transfers this energy to one of the orbital electrons, which is then ejected with a discrete energy (see 3.2).

Another mode of decay observed in high mass number substances (A>230) is spontaneous fission. This process often competes with α decay. The reaction is

$$ {}^{A}_{Z}X \rightarrow {}^{A'}_{Z'}Y + {}^{A-A'-k}_{Z-Z'}W + k\,{}^{1}_{0}n $$

In the above, the original nuclide A_X splits into two fragments Y and W, and a number, k, of neutrons are emitted. The same number of neutrons will not be released in each fission, so k is an average value. The total number of n released in spontaneous fission varies with the mass and the atomic number. Nuclides with even Z and A seem to have the shorter spontaneous fission half lives.[2] Since fission releases neutrons, fragments, prompt and delayed γ rays and β particles, this complex mode is some times indicated by simply a vertical arrow marked SF.[3]

If a nuclide has enough energy in an excited state, it may also decay by neutron emission.[4] Since the process itself is rare, and the emission time very short, practical neutron sources, such as those for α, β or γ, are not readily found.

Included with the diagram of the decay scheme is other useful information about the nature of the process. Sometimes, the complex nature of the process does not allow all of the information to be displayed. Shown in Figure 2.3, is the decay scheme for ^{24}Na, adapted from Reference 5, to point out some of the data which is supplied for simple decay schemes.

In the figure, ^{24}Na is indicated as a β^- emitter with a half life of 15 h. For the two β^- groups which are emitted, the fraction of transitions in which that β^- is emitted (often called the intensity), as well as the maximum energy (in MeV) of the emitted β is shown. That is, β^-_1, is emitted in 99.9% of the transitions and the maximum energy of this β is 1.39 MeV. The emission of the

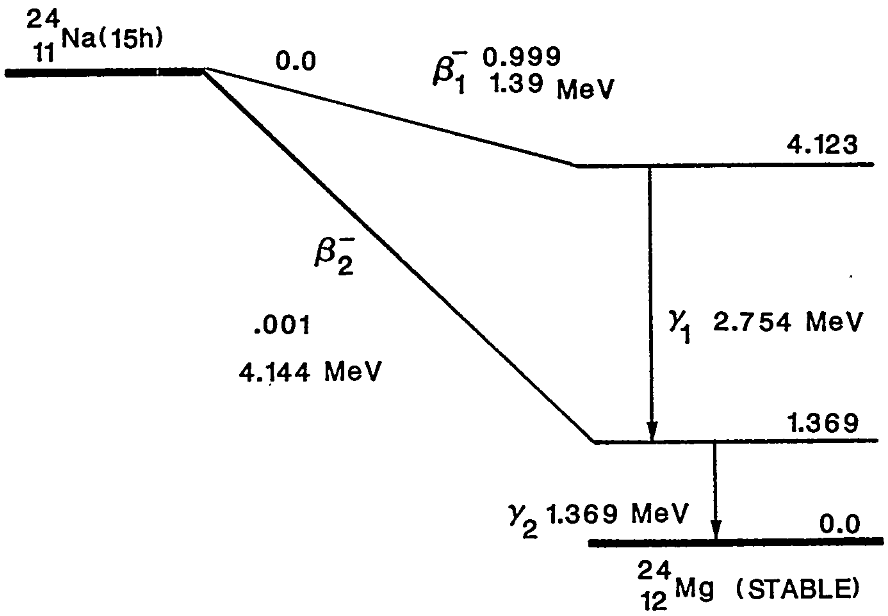


Figure 2.3 Decay scheme for ^{24}Na.

β produces ^{24}Mg but in an excited state, either 4.123 MeV above the stable ^{24}Mg ground state, or 1.369 MeV, if β_2^- is emitted. The excited nucleus returns to the ground state by emission of γ radiation. For the majority of decays (99.9%), two γ rays, one of 2.754 MeV immediately followed by one of 1.369 MeV, are emitted. For the small number of transitions involving β_2^- (0.1%), only one 1.369 MeV γ is emitted.

Transformation or decay schemes can be found in diagram form in References 3,5-7, along with additional information concerning the transition and other nuclear properties. Reference 3, which has been prepared specifically for use in internal dosimetry calculations, gives the average β transition energy, rather than the maximum energy. Other sources of useful decay scheme data, in tabular form rather than diagrams, can be found in References 8-10.

Another useful source of information about unstable (or radioactive) nuclides is the Chart of the Nuclides (Appendix G). This chart lists well over 2,000 nuclides, of which only about 300 are stable. The general features of the chart, of interest at this point, are shown in the excerpt, Figure 2.4. Each horizontal row represents an element, i.e., H = hydrogen, He = helium, and the individual filled spaces in a row are the known isotopes of that element. Note that hydrogen has three known isotopes. The atomic number Z increases vertically with each row. The neutron number N = A-Z increases horizontally. The number of each column is the total number of neutrons (n) for each nucleus in that column.

The heavily bordered space at the far left side of each row gives the chemical atomic weight (in atomic mass units) of the element as found in nature (a combination of ^{1}H and ^{2}H). Shaded spaces in the row are stable isotopes. A black rectangle at the top of the space indicates a radioactive isotope that is found in nature. White spaces represent radionuclides which are artificially produced.

For stable nuclides, such as ^{1}H, the first line of the square gives the symbol and the mass number. The second line gives the isotopic abundance. This is the percent of ^{1}H which is present in the element

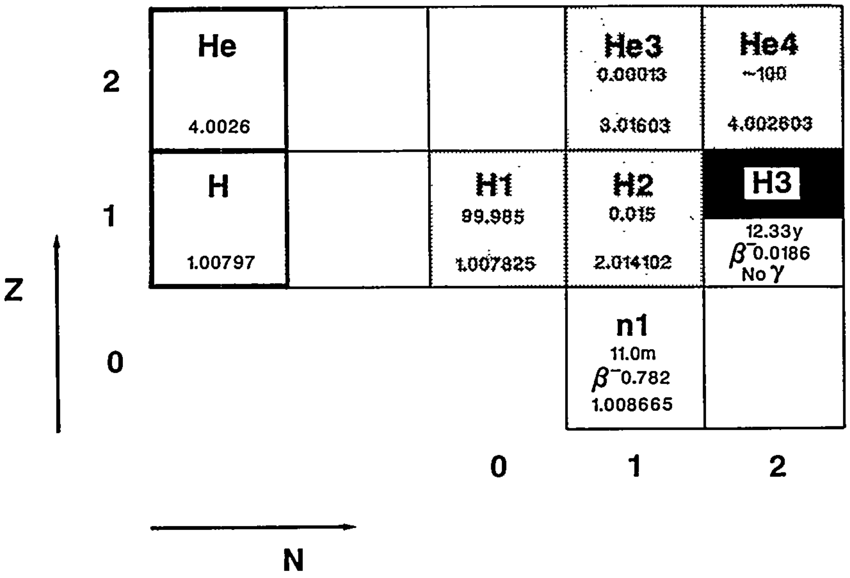


Figure 2.4 Excerpt from Chart of Nuclides. (Knolls Atomic Power Laboratory. Schenectady, New York. Operated by the General Electric Co. for Naval Reactors, The U. S. Department of Energy).

hydrogen as it is found in nature. The number at the bottom of the space is the atomic mass of the neutral atom ^{1}H.

For the unstable nuclides, such as ^{3}H, the first line gives the symbol and mass number. The second line gives the half life. Additional lines present decay modes and energies (in MeV). For ^{3}H, the listing indicates a β^- emitter of maximum energy 0.0186 MeV which is not followed by any γ emission. In the case of very long-lived, naturally occurring radionuclides, the second line gives the abundance. Subsequent lines give the data for the decay modes.

For certain radionuclides, a long-lived excited nuclear state is possible. These are called <u>isomeric</u> states. Each state has its own half life, decay mode, and energy of emissions. Isomers are shown on the chart by a divided square for the given radionuclide.

To trace the product of a radioactive decay, or to find a product nucleus when a target is bombarded, the scheme in Figure 2.5 may be used. It is designed to allow easy location on the Chart of the Nuclides. For example, let the original nucleus have an atomic number Z and neutron number N, and decay by α emission. The product nucleus will be found

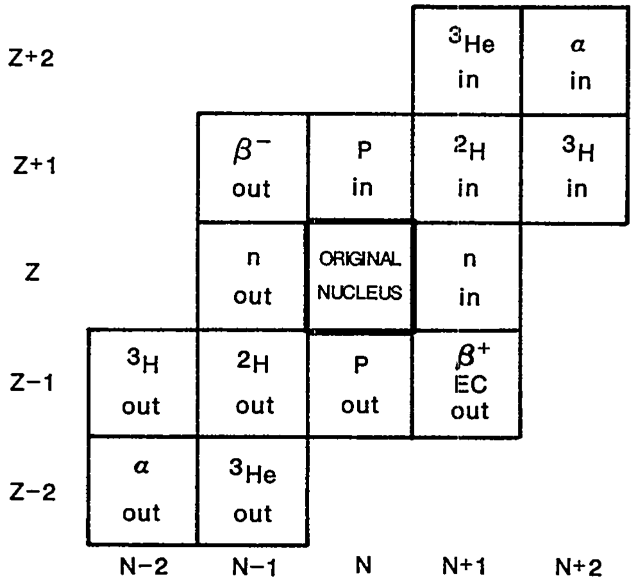


Figure 2.5 Location chart for nuclear products. (Knolls Atomic Power Laboratory, Schnectady, New York. Operated by the General Electric Co. for Naval Reactors, the U.S.Department of Energy.)

on the chart at the location Z-2 and N-2. In the case of ^{238}U which is an α emitter, Z=92 and N=146, so the daughter product will be found at Z=90 and N=144. From the Chart in Appendix G, this location corresponds to ^{234}Th. For tracing a radioactive chain, the first daughter can then be considered the original nucleus to find the second daughter, and so on.

C. Decay Law

When one deals with large numbers of atoms, it is found that all radioactive substances follow the same general decay pattern. For example, assume one has a radioactive source and some means of counting the number of atoms which decay in a given time interval. The number of atoms decaying in a given time interval is called the activity of the sample. If one plots the percentage ratio of the activity at some later time (t) to the activity at time t=0, on linear graph paper (Figure 2.6) versus the time t, the curve obtained indicates an exponential or logarithmic relationship. If the same ratio is plotted on semilog paper (Figure 2.7), a straight line is obtained. This indicates that radioactive decay is an

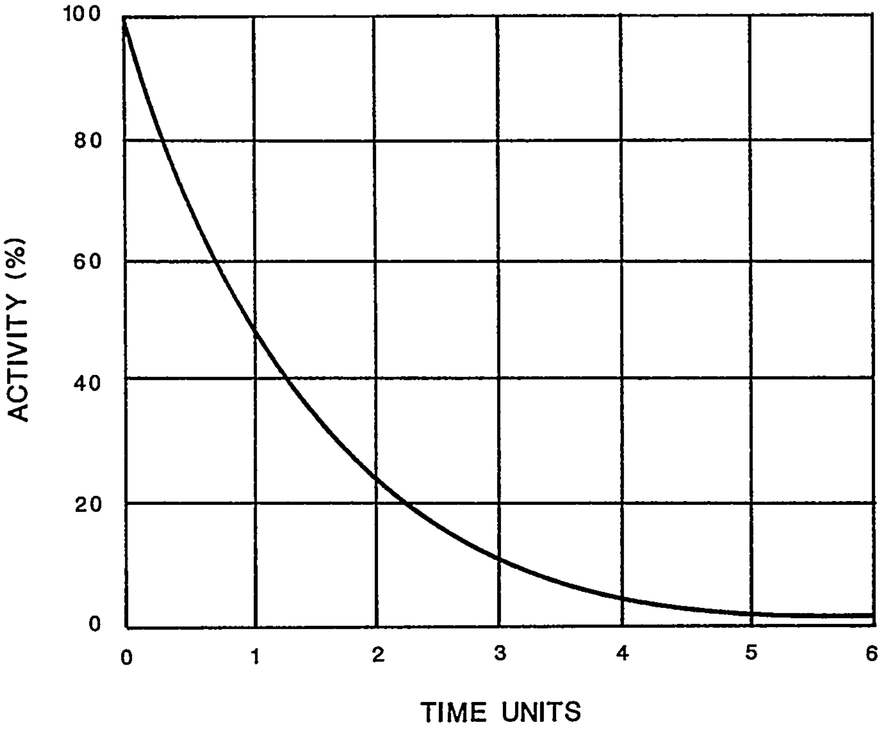


Figure 2.6 Radioactive Decay, linear plot.

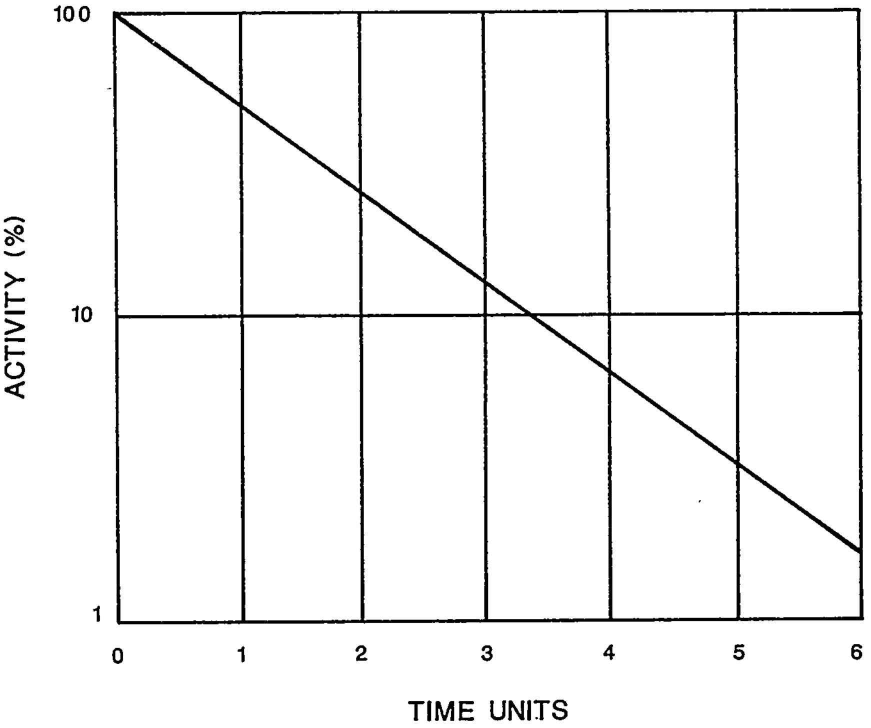


Figure 2.7 Radioactive decay, semilog plot.

exponential (logarithmic) process, that is, there is a constant fractional decrease in the decay rate during equal units of time. Although the same number of atoms do not disintegrate during each unit of time, the same fraction of the atoms present decay during a unit of time. The disintegration rate is proportional to the number of atoms present:

$$\frac{\Delta N}{\Delta t} = \frac{N_1 - N_o}{t_1 - t_o} \propto N, \qquad 2.1$$

where N is the number of atoms present at any time t. By including a constant of proportionality, the expression becomes

$$\frac{\Delta N}{\Delta t} = -\lambda N , \qquad 2.2$$

where λ, called the transformation constant (also the decay constant), is the constant of proportionality and the minus sign indicates a decrease in the decay rate as time increases. When the expression is integrated, we get the general exponential relationship for radioactive decay:

$$N = N_o e^{-\lambda t} \qquad 2.3$$

where N is the number of atoms left at time t from a sample of N_o radioactive atoms initially present.

If we take the logarithm of each side of this equation, we get

$$\ln N = \ln N_o - \lambda t. \qquad 2.4$$

This is similar to the slope-intercept form of a straight line, i.e., $y = a + bx$. Thus, if one uses semilog paper and plots the values of N on the logarithmic scale versus the time on the linear scale, the slope of the resulting straight line will be $-\lambda$ and the intercept (the value of N at t = 0) will be N_o.

An an example, consider the data:

Number of Atoms	Time (s)
80,000	0
29,432	60
10,824	120
3,984	180
1,464	240
536	300
200	360

Plot a curve on semilog and on linear paper, and determine the decay constant for the radioactive element.

To plot on semilog paper: Take three-cycle semilog paper (this will cover the ranges 100-1000, 1000-10,000 and 10,000-100,000) and plot the number of atoms on the logarithmic scale and the corresponding value of the time on the linear scale. The plot will be a straight line.

To find the slope of the line ($-\lambda$): Take the ratio of the number of atoms for two times; find the natural log of this ratio, and divide this log by the difference in time for the values chosen. For example, choose the readings for $t = 0$ and $t = 120$ s. Take the ratio of the number of atoms at the later time to the number of atoms at the earlier time. This ratio is 0.1353. The natural log of this ratio is $\ln 0.1353 = \ln (1.353 \times 10^{-1}) = \ln 1.353 + \ln 10^{-1} = \ln 1.353 - \ln 10 = 0.3026 - 2.3026 = -2$. The time difference for the two values is 120 s. We now obtain

$$\ln N = \ln N_o - \lambda t;$$
$$\ln N - \ln N_o = -\lambda t;$$
$$\ln N/N_o = -\lambda t;$$
$$-2 = -\lambda(120 \text{ s});$$
$$1.67 \times 10^{-2} \text{ s}^{-1} = \lambda.$$

To plot on linear paper: Plot number of atoms as ordinate (y axis) and the time as abscissa (x axis).

The decay constant λ expresses the probability that a single atom will decay in a unit of time. The larger the value of λ, the more rapidly a radioelement decays.

One is usually more interested in the decay rate of a given sample rather than the number of atoms present. The activity of a sample is expressed as:

$$\frac{dN}{dt} = A_t = \lambda N. \qquad 2.5$$

In other words, the activity A_t of the sample can be found by multiplying N, the number of atoms present at any time, by the decay constant λ of the element. Accordingly:

$$N = N_o e^{-\lambda t};$$

$$\lambda N = \lambda N_o e^{-\lambda t};$$

$$A_t = A_o e^{-\lambda t}, \qquad 2.6$$

where A_o is the activity (decay rate) of the sample at t = 0. Combining equation 1.2 and 2.5 above, the expression for the activity of a known mass m of a radionuclide is given by:

$$A_t = \lambda N = \frac{\lambda m}{A} N_a \qquad 2.7$$

D. Half Life and Mean Life

The decay constant λ is closely related to the concept of the half life of a radionuclide. The half life $T_{1/2}$ is defined as the time required for the activity of a given nuclide to decay to one-half of its initial value. Consider an initial activity A_o. At some time $t = T_{1/2}$, the activity will be equal to $A_o/2$. From our general relationship, then,

$$A_t = A_o e^{-\lambda t};$$

$$\frac{1}{2} A_o = A_o e^{-\lambda T_{1/2}}$$

$$\frac{1}{2} = e^{-\lambda T_{1/2}}$$

$$\ln 1 - \ln 2 = -\lambda T_{1/2}$$

$$\lambda T_{1/2} = \ln 2;$$

$$T_{1/2} = \frac{\ln 2}{\lambda} = \frac{0.693}{\lambda}, \qquad 2.8$$

substituting for λ in equation 2.7 gives an alternate expression for the activity in terms of the half life,

$$A_t = \frac{\lambda\ m\ Na}{A} = \frac{0.693\ m\ Na}{AT_{1/2}} \qquad 2.9$$

Example: Find the activity (dis/s) of a 10^{-3} kg (1 g) sample of ^{60}Co ($T_{1/2}$ = 5.27 y).

In SI units, a mole of ^{60}Co = 0.06 kg, and

$$A_t = \frac{0.693\ (10^{-3}\ \text{kg})\ 6.022\text{x}10^{23}\ \frac{\text{molecules}}{\text{mole}}}{0.06\ \frac{\text{kg}}{\text{mole}}\ 5.27\ \text{y}\ 3.1536\text{x}10^{7}\frac{\text{s}}{\text{y}}} = 4.19\text{x}10^{13}\text{dis/s}$$

The actual length of time any one atom survives may be anything from 0 to infinite time, theoretically. Suppose we were able to sum up all the lifetimes for the entire sample of atoms. Now, divide by the total number of atoms, and we arrive at the average lifetime of an atom. The average, or mean life, T is given by:

$$T = \frac{1}{\lambda} = \frac{T_{1/2}}{\ln 2} = 1.443\ T_{1/2} \qquad 2.10$$

Given the activity A_t of a sample, the total number of transitions which will occur in the sample may be obtained from:

$$\text{Trans.} = A_t T = 1.443\ T_{1/2} A_t \qquad 2.11$$

in which the time units must be consistent. For the ^{60}Co example above, the total number of transitions would be:

$$\text{Trans.} = 1.433(4.19\text{x}10^{13}\frac{\text{dis}}{\text{s}})(5.27\ \text{y})(3.1536\text{x}10^{7}\ \frac{\text{s}}{\text{y}})$$

$$= 1.005\text{x}10^{22}.$$

Each radioactive atom has its own unique pattern of decay. Three aspects which characterize the radioactive decay of a given nuclide are:

1) half life of emission,
2) energy of the emission, and
3) type of emission

The identification of a particular radionuclide will normally depend upon how well one can determine these three factors. Many radionuclides have half lives which are nearly the same, but the energy of their emissions differs greatly. On the other hand, many radionuclides have similar energy of emission, but their half lives differ greatly.

Oftentimes, a number of techniques or combination of techniques will be required for the identification of an unknown radionuclide. Sometimes, when one knows the types of radioactive material being used in an area, it is possible to pinpoint the identity of the radionuclide by the half life alone. For long-lived radionuclides, however, this approach may not be feasible. Also, for faster identification, it may be easier to analyze for the energy of the emissions or to search for the type of emission.

E. Activity Units

The SI unit for the activity of a radioactive substance is the becquerel (Bq). A becquerel is the activity of a radionuclide whose transformation rate is 1 dis/s. This unit replaces the curie (Ci), the historical activity unit originally taken as the decay rate of 1 g of ^{226}Ra, but later defined as a transformation rate of $3.7\text{x}10^{10}$ dis/s.[4] The two units are related by

$$1\ \text{Bq} = 2.703\text{x}10^{-11}\ \text{Ci}. \qquad (1\ \text{Ci} = 3.7\text{x}10^{10}\ \text{Bq})$$

The becquerel is a rather small unit, whereas the curie is a rather large unit. To aid in designating the large range of values that may be experienced, SI prefixes are shown in Table 2.1 (adapted from Reference 11).

Table 2.1 - SI Prefixes

Prefix	Symbol	Factor	Prefix	Symbol	Factor
deka	da	10^1	deci	d	10^{-1}
hecto	h	10^2	centi	c	10^{-2}
kilo	k	10^3	milli	m	10^{-3}
mega	M	10^6	micro	μ	10^{-6}
giga	G	10^9	nano	n	10^{-9}
tera	T	10^{12}	pico	p	10^{-12}
peta	P	10^{15}	femto	f	10^{-15}
exa	E	10^{18}	atto	a	10^{-18}

Common smaller units of the curie encountered in health physics work are the pCi, nCi and μCi. In terms of SI units, these become

$$1 \text{ pCi} = 10^{-12} \text{ Ci} = 3.7\text{x}10^{-2} \text{ Bq} = 37 \text{ mBq}$$

$$1 \text{ nCi} = 10^{-9} \text{ Ci} = 3.7\text{x}10^{1} \text{ Bq} = 37 \text{ Bq}$$

$$1 \ \mu\text{Ci} = 10^{-6} \text{ Ci} = 3.7\text{x}10^{4} \text{ Bq} = 37 \text{ kBq}$$

Some laboratory sources in use will be more conveniently expressed in MBq.

The definition of activity refers to the transformations (disintegrations) per unit time. Oftentimes, the number of transformations will differ greatly from the number of radiations emitted, as for example, in the case of ^{60}Co. For each ^{60}Co atom which decays, a beta particle and two γ rays are emitted. In this case, then 1 MBq of ^{60}Co would emit $1.0\text{x}10^6$ β/s and $2.0\text{x}10^6$ γ/s. If the activity of a sample is to be accurately calculated from experimental data, the decay scheme of the radionuclide must be taken into account.

If counting a β^- emitter, some of the γ rays may also be counted. If one merely takes the counting rate and converts this to a transformation rate, without considering the decay scheme, one will usually overestimate the activity of the source. For example, consider a counter which counts all the β (1 count/dis) and 2% of the gammas. In counting a 1 MBq ^{60}Co source, the observed counting rate would be $1.0\text{x}10^6$ counts/s β + 0.02 $(2.0\text{x}10^6)$ counts/s γ = $1.04\text{x}10^6$ counts/s. If one converts this activity to MBq without regard to the decay scheme, the activity would come out as 1.04 MBq, a slight overestimate. However, if one counted this same source in a counter, which did not count the β, a serious overestimate would occur. Suppose such a counter counted all of the γ rays (1 count/dis). Then, the result would be 2x 10^6 cts/s, which converted to activity would be 2 MBq, double the true source activity.

In the case of electron capture decay, a serious underestimate could occur when counting, say ^{7}Be, in such a counter. ^{7}Be emits about 0.1 γ per transformation. This means a 1 MBq source will only give about 1x10^5 cts/s, which for this counter could be erroneously interpreted as 0.1 MBq.

In counting an α emitter, the situation is usually different. Since a proportional counter is normally used, the γ rays associated with an α emitter would not be counted, and thus the activity can be determined from the counting rate. Nevertheless, if the source being counted contains a radioactive chain, there may be more than one α emitted per transformation, since daughter products may also emit α particles. This would subject one to the potential of overestimating the source activity.

When the radionuclide that one is counting is known, a proper choice of counting equipment can be made. A suitable calibration of this equipment will then allow one to obtain a reasonable activity estimate. When the radionuclide is unknown, as occurs frequently in health physics work, one must exercise care in the interpretation of any counting data.

F. Specific Activity

The specific activity is defined as the activity per unit mass of a substance. It has usually been expressed as Ci/g of the radioelement before SI units. The shorter the half life of the emitter, the greater is found to be its specific activity. The specific activity can be calculated from equation 2.5, now expressed as

$$SP.A. = \lambda N. \qquad 2.12$$

where λ is the transformation constant and N is now the number of atoms in one kg of the radioelement. ^{60}Co has a half life of 5.27 y.

Calculate the specific activity:

$$SP.A. = \lambda N = \frac{0.693}{T_{1/2}} \frac{1 \text{ kg}}{A} N_a, \text{ and } A = .060 \text{ moles}$$

$$= \frac{0.693 \; (1) \; 6.022\text{x}10^{23}}{5.27 \text{ y } (3.1536\text{x}10^{7} \frac{\text{s}}{\text{y}}).06} = 4.19\text{x}10^{16} \frac{\text{dis}}{\text{s-kg}}$$

$$= 4.19\text{x}10^{16} \frac{\text{Bq}}{\text{kg}}$$

$$[SP.A.=4.19\text{x}10^{16} \frac{\text{Bq}}{\text{kg}} (2.703\text{x}10^{-11} \frac{\text{Ci}}{\text{Bq}})10^{-3} \frac{\text{kg}}{\text{g}} = 1.13\text{x}10^{3} \text{ Ci/g}]$$

To convert a tabulated value of SP.A. in Ci/g to SI units, use

$$SP.A \text{ (Bq/kg)} = 3.7\text{x}10^{13} \; SP.A \text{ (Ci/g)}$$

G. Decay Chains

In general, most radioactive substances do not decay to form a stable nuclide, that is to say, the daughter nucleus is also radioactive and decays with its own characteristic half life. The problem of determining the amount of the daughter present at any time depends, therefore, upon both half lives. The daughter will be produced at a certain rate from the parent, but will decay with its own rate.

Suppose that at a given time there are N_1^0 parent atoms, with decay constant λ_1, and no daughter atoms. After a certain interval of time, Δt, the increase in the number of daughter atoms, ΔN_2, is given by

$$\Delta N_2 = (\text{decay rate of parent - decay rate of daughter}) \; \Delta t. \qquad 2.13$$

The decay rate of the parent is actually the formation rate of the daughter, since whenever a parent atom decays, it becomes a daughter atom. The decay rate of the parent is $\lambda_1 N_1$, where N_1 is the number of parent atoms present at any time. Similarly, the decay rate of the daughter is $\lambda_2 N_2$. The expression for the rate of change of daughter atoms per unit time becomes

$$\frac{\Delta N_2}{\Delta t} = \lambda_1 N_1 - \lambda_2 N_2. \tag{2.14}$$

From the general relationship, the number of parent atoms at any time is

$$N_1 = N_1^0 e^{-\lambda_1 t}. \tag{2.15}$$

Substituting this into the above expression and integrating the equation, one arrives at

$$N_2 = \frac{N_1^0 \lambda_1}{\lambda_2 - \lambda_1} (e^{-\lambda_1 t} - e^{-\lambda_2 t}). \tag{2.16}$$

where N_2 is the number of daughter atoms present at any time.

When the parent half life is long compared with that of the daughter, then $\lambda_1 < \lambda_2$, and the term $e^{-\lambda_2 t}$ becomes negligible compared with $e^{-\lambda_1 t}$ after a sufficiently long time. Then equation 2.16 reduces to

$$N_2 = \frac{\lambda_1 N_1^0}{\lambda_2 - \lambda_1} e^{-\lambda_1 t}. \tag{2.17}$$

However, since

$$N_1 = N_1^0 e^{-\lambda_1 t}.$$

we find

$$N_2 = \frac{\lambda_1 N_1}{\lambda_2 - \lambda_1}. \tag{2.18}$$

A condition is thus reached in which the ratio N_2/N_1 remains constant. This state is called transient equilibrium. In this case, the daughter activity decays at the same rate as the parent activity. Another way of stating this is that the formation rate of daughter atoms equals the decay rate of the daughter atoms. In each unit of time then, there is the same fractional decrease in the parent and daughter activities, yielding a constant ratio.

When the parent activity is extremely long-lived, as is the case with ^{238}U ($T_{½}$=4.5x10^9 y), then $\lambda_1 << \lambda_2, e^{-\lambda_1 t}$ ~1, and $\lambda_2 - \lambda_1$ ~λ_2, so that the original expression for N_2 reduces to

$$N_2 = \frac{\lambda_1 N_1^0}{\lambda_2}(1 - e^{-\lambda_2 t}). \qquad 2.19$$

After a sufficient time, $e^{-\lambda_2 t}$ approaches zero, and the amount of the daughter product is then given by

$$N_2 = N_1^0(\lambda_1/\lambda_2). \qquad 2.20$$

This state is known as secular equilibrium.

If the parent has a shorter half life than the daughter, $\lambda_1 > \lambda_2$. The daughter rises to a maximum and decays with its own characteristic half life. In this case, no equilibrium is reached.

The above relationships have been presented for cases in which no daughter atoms are initially present. If there are daughter atoms, we may add a term $N_2 e^{-\lambda_2 t}$ to equation 2.16, which gives

$$N_2 = \frac{\lambda_1 N_1^0}{\lambda_2 - \lambda_1}(e^{-\lambda_1 t} - e^{-\lambda_2 t}) + N_2^0 e^{-\lambda_2 t} \qquad 2.21$$

in which N_2^0 is the number of daughter atoms present at t = 0. In most instances, one is interested in the <u>activity</u> of the sample, rather than the number of atoms. The activity can be obtained by use of equation 2.5,

$$A_t = N\lambda = N_2\lambda_2 \text{ for this case,}$$

$$(A_t)_2 = N_2\lambda_2$$

$$= \frac{\lambda_2 A_1^0}{(\lambda_2 - \lambda_1)}(e^{-\lambda_1 t} - e^{-\lambda_2 t}) + A_2 e^{-\lambda_2 t}$$

When a radioactive chain is present, the method outlined above can be extended. The original relationships were developed by H. Bateman in 1910. A general equation for the quantity of the nth member of a series can be found in Reference 12.

H. Decay Curve of a Mixture

The problem of identifying the half life of a radionuclide from a decay curve can be complicated by the fact that the experimental curve may be a composite of the contributions from a mixture of radionuclides. When the activity values are plotted on semilog paper, the curve obtained from such a mixture will contain: 1) an initial straight portion, followed by 2) a curved portion, which is then followed by 3) a final straight portion. This type of decay curve is explained by the presence of at least two radioactive products in the sample being counted.

The initial straight portion of the curve represents the sum of the activities of all the components, whereas the final straight portion is due only to the activity of the longest-lived component of the mixture. The shorter-lived products contribute greatly to the decay rate initially; but as they decay out, the total activity will not decrease as rapidly as in the beginning. This accounts for the curvature in the plot.

Since the final straight portion represents the activity of the long-lived component, this line can be extrapolated back to zero time. The initial activity, of the long-lived component, is the intercept value at $t = 0$, of the extrapolated line. In addition, this line gives the decay curve of the long-lived component. When a series of values along this curve are subtracted from corresponding values on the original curve, the remainders will represent the activity of the short-lived components, at the corresponding times. If the mixture contains only two components, then a plot of the remainder values will give a straight line. As before, the activity value at $t = 0$ for this new straight line will represent the initial activity of the short-lived component.

The half life for each component can then be determined graphically from each of the straight lines.

If three or more components are present, this method can be extended provided the original data are sufficiently accurate.

Example: Using the data in the table, plot a curve on semilog (see Figure 2.8) and determine the half lives for all the activities in the sample.

cps	t (h)	cps	t (h)
10,000	0	370	14
5,788	1	258	16
4,078	2	181	18
3,148	3	127	20
2,516	4	89	22
1,666	6	62	24
1,125	8	44	26
770	10	31	28
531	12		

Plot the data on 3-cycle semilog paper. From the shape of the curve one would surmise that there are at least two components. Extrapolate the final straight portion of the curve back to zero time. This line intersects the t = 0 axis at 4000 cps. The half life is determined directly from the graph: find the time at which the activity has dropped to 2000 cps. This occurs at 4 h. The half life can be determined mathematically by choosing two points on the curve, finding the ratio of these two points, and dividing the natural log of this ratio by the difference in time between these two points. This gives the value of λ. Take the points: t = 0, cps = 4000; t = 12 h, cps = 500.

$$\ln\frac{N}{N_o} = \ln\frac{1}{8} = -\ln 8 = -12\ \lambda;$$

$$\lambda = \frac{\ln 8}{12} = \frac{3\ \ln 2}{12} = \frac{\ln 2}{4}\ h^{-1}$$

But

$$T_{\frac{1}{2}} = \frac{\ln 2}{\lambda} = \frac{\ln 2}{\ln 2}(4)\ h = 4\ h.$$

Take points on the extrapolated curve and subtract their values from the corresponding points on the experimental curve. Plot these differences

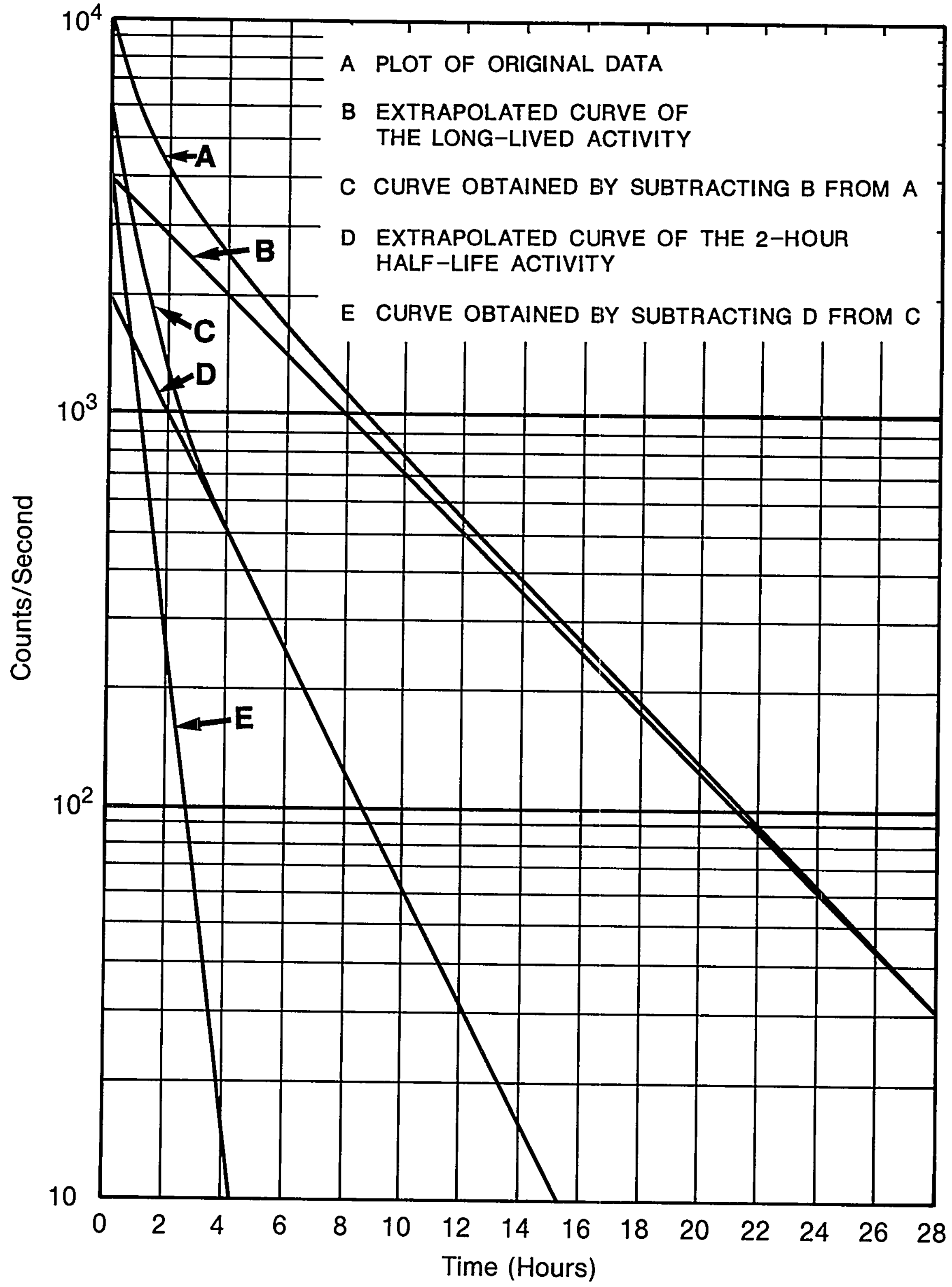


Figure 2.8 Decay curve using example data in text.

versus the corresponding times. Consider the points for t = 0; the value on the experimental curve is 10,000 cps, the value on the extrapolated curve is 4,000 cps. The difference, 6,000 cps, is plotted for t = 0. Similarly, for the point t = 1 h, the value on the experimental curve = 5,778 cps, and the value on the extrapolated curve = 3,400 cps. The difference, 2,378 cps, is plotted for t = 1 h. Proceeding in this way, one can determine enough points to draw the curve.

In this case the curve is not a straight line, it is a curved line. Once again extrapolating the final straight portion back to t = 0 gives an initial activity of 2,000 cps. The half life of this activity as determined from the graph is 2 h.

When points on this extrapolated curve are subtracted from the corresponding points on the second curved line and the differences are plotted, a straight line is again obtained. The initial activity of this component is 4,000 cps, and its half life as determined from the graph is 30 minutes.

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QUESTIONS

2.1 When and by whom was radioactivity discovered?

2.2 Name the three distinct types of naturally occurring radiation.

2.3 With what nucleus are α particles identical?

2.4 With what atomic particle are β particles identical?

2.5 When an α particle is emitted, how many nuclear particles are emitted?

2.6 How are the A and Z numbers of a radioactive atom affected when:

a. an α particle is emitted?
b. a β particle is emitted?

2.7 When a β particle is emitted,

a what nuclear particle was converted to release the β particle?

b. what is the new nuclear particle that results from the conversion?

2.8 How does the emission of a γ ray affect the atomic number and the mass number?

2.9 What is the relationship between the number of atoms present in a radioactive source at any particular time t to the time t?

2.10 To what value is the disintegration rate of radioactive atoms proportional?

2.11 What does the symbol λ (lambda) represent in radioactive decay schemes?

2.12 Describe the plot of radioactive decay rate versus time:

a) on linear (common) graph paper and,
b) on semilogarithmic paper.

2.13 How many cycles should semilogarithmic paper have to plot

a) 700 to 1300 atoms?
b) 20 to 95 atoms?
c) 300 to 120,000 atoms?

2.14 What is the base of

a) common logarithms?
b) natural logarithms?

2.15 What are the common logarithms of

a) 1 b) 100
c) 10^2 d) 10^{-3}
e) $10^{-4.7}$?

2.16 What are the natural logarithms of

a) e b) e^3
c) e^{-1}?

2.17 What value results when the number of radioactive atoms present (N) is multiplied by the decay constant (λ)?

2.18 What value is defined by the time required for a given isotope to decay to one-half its original value?

2.19 If a radioactive substance loses half its activity in 3 hours, what will its half life be 6 hours later?

2.20 In the conversion formula $T_{1/2} = \frac{0.693}{\lambda}$,

a) what does the symbol $T_{1/2}$ represent?
b) what does the symbol λ represent?
c) of what number is 0.693 the natural logarithm?

2.21 What is the unit of source activity based upon the number of disintegrations per unit of time?

2.22 Why may β emitters appear to have a higher becquerel value than their true value?

2.23 What does the activity of one kilogram of a radioactive substance define?

2.24 What terms identify

a) a radioactive material that, upon decay, results in another radioactive material?

b) the resulting radioactive material?

2.25 Under what condition(s), does the ratio of the number of daughter atoms to the parent atoms remain constant? What is the name given to such a state?

2.26 Under what conditions is secular equilibrium reached between the daughter and parent?

2.27 Under what condition is no equilibrium reached?

2.28 Assuming that good data has been obtained, what does a curved decay plot on semilogarithmic paper indicate?

2.29 What term is given to the projection of a mathematical curve beyond known data?

2.30 What value is obtained by taking natural logarithm of the ratio of two points on a radioactive decay curve and dividing that ratio by the intervening time?

PROBLEMS

2.1 Complete the following decay schemes

a. ${}^{226}_{88}\mathrm{Ra} \rightarrow \mathrm{Rn} + \mathrm{He}$

b) ${}^{14}_{6}\mathrm{C} \rightarrow \mathrm{N} + \beta^{-} + \bar{\nu}$

c. ${}^{107m}_{47}\mathrm{Ag} \rightarrow {}^{107}? + \gamma$

d) $\mathrm{Au} \rightarrow {}^{198}_{80}\mathrm{Hg} + \beta^{-} + \bar{\nu}$

2.2 If the number of radioactive atoms at time t is 2×10^{6}, and 2×10^{4} atoms disintegrate in 5 minutes, what is the approximate radioactive constant?

Answer: ~ 2×10^{-3} minutes

2.3 If the radioactive constant is 0.1/day and there are 3×10^{6} radioactive atoms, approximately how many atoms will disintegrate in 1 minute?

Answer: 208/min

2.4 From the formula $A_t = A_o e^{-\lambda t}$, find the activity of a sample at 4:00 P.M. when its activity was 1000 disintegrations per minute at 10:00 A.M. The decay constant λ of the sample is 0.2/day.

Answer: 951/min

2.5 The half-life of radon is 3.8235 days. What is the decay constant?

Answer: $0.1813\ d^{-1}$

2.6 In problem 2.5, what percentage of a freshly separated sample of radon will disintegrate in 1 day? In 2, 3, 4, 5, 10 and 20 days?

Answer:

Time(d)	% of radon decayed
1	16.6
2	30.4
3	41.9
4	51.6
5	59.6
10	83.7
20	97.3

2.7 The activity of 10^{-7} kg of ^{230}Th is found to be $7.2\text{x}10^4$ Bq (dis/s). What is the half life of $^{230}_{90}$Th?

Answer: $8.0\text{x}10^4$ years

2.8 The activity of a radioactive sample is 25 Bq. What was the activity of the sample 1 hour earlier if the half life is 25 minutes?

Answer: 132 Bq

2.9 Carbon-14, ^{14}C, has a half life of 5730 years. What is the specific activity of $^{14}_{6}$C?

Answer: $1.65\text{X}10^{14}$ Bq/kg

2.10 Krypton-88, ^{88}Kr has a half life of 2.8 hours and its daughter rubidium-88, ^{88}Rb has a half life of 18 minutes. If the krypton-88 has decayed to $5\text{x}10^{10}$ atoms over a period of several weeks, how many daughter atoms are present? Hint: use the short form formula since krypton-88 has a much longer half life compared with the half life of rubidium-88.

Answer: $6\text{x}10^{9}$ atoms

2.11 Thorium-232, ^{232}Th, has a half life of $1.41\text{x}10^{10}$ years and its daughter radium-228, ^{228}Ra has a half life of 5.76 years. If 10^{16} atoms of thorium-232 are found in a lump of natural ore, how many atoms of radium-228 should be present? Hint: thorium-232 has an extremely long half life when compared with the half life of radium-228.

Answer: $4.085\text{x}10^{6}$ atoms.

2.12 The first two members of a radioactive series have half lives of 6 minutes and 12 minutes respectively, while the third member is stable. Starting with 10^{6} atoms of the first member and none of the second and third, plot the number of atoms of the three members as a function of time. Determine from the graph (or otherwise), the time at which the second member reaches its maximum.

Answer: 12 minutes

2.13 A smear obtained from a graphite block bombarded by a 50 MeV proton beam was counted with a germanium detector. The following data was recorded

Time (Minutes) Elapsed	Counts Per Minute
5	8400
10	7104
15	6008
20	5102
25	4300
30	3600
40	2591
50	1820
60	1310
70	932
80	661
100	330

Plot the data as (a) linear plot and as a (b) semilog plot. Determine the half life. Can you guess what could be the radionuclide?

2.14 The half life of ^{238}U is 4.4683×10^9 years. How many kilograms of ^{238}U is needed for an activity of 3.7×10^{10} Bq (this is 1 Ci in the old system of units)?

Answer: 2973 kg

2.15 The data in the following table represents the decay curve of a mixture of two radionuclides

a) Plot the data on semilog paper

b) Determine the half lives of the two radionuclides graphically.

t (min)	cpm	t (min)	cpm
0	60,000	40	1406
5	34,000	45	953
10	20,000	50	664
15	12,000	55	457
20	7,500	60	323
25	4,750	65	222
30	3,140	70	152
35	2,062	75	111

Answer: ~ 9.5 min and ~ 4.5 min

SECTION 3 - PROPERTIES OF ALPHA, BETA, GAMMA, X RAYS AND NEUTRONS

There are four main types of radiation (we include γ and x rays as one type) which must be dealt with in health physics. The properties of these radiations are important in the determination of the relative hazard which they present. These properties will be treated in the following discussions. Other radiations that result primarily from high energy reactions will be dealt with in the later sections as appropriate.

A. Alpha Particles

The α particle is a helium nucleus (^4_2He) emitted with a discrete energy and a characteristic half life from each α emitter. Rutherford and Royds showed that the α particle is a helium nucleus by collecting α particles from radon in a glass tube with electrodes. When a discharge in the tube occurred, the light spectrum obtained showed characteristic helium lines.

Examination of cloud-chamber photographs (see Figure 3.1) have revealed these properties of α particles:

1) Most alphas have the same range in a given gas.
2) The alphas travel along straight tracks.
3) Some are scattered near the end of their path.
4) A few are scattered at distances closer to the source.

The first property mentioned above shows that almost all the particles emitted by a given radionuclide have about the same discrete energy. The other properties imply that scattering of the alphas occurs infrequently, most often near the end of the path of the particle.

Alpha particles are ejected from naturally occurring radioactive atoms with speeds of the order of one-twentieth that of the speed of light. Because of their large mass and relatively high speed, they have large kinetic energies. Alpha particles from naturally radioactive nuclei usually have energies in the range from 4 to 9 MeV.

The energy of an α particle is lost mainly by ionization and excitation of the atoms of the traversed substance. Ionization occurs by two processes: (1) direct collision with an orbital electron, and (2)

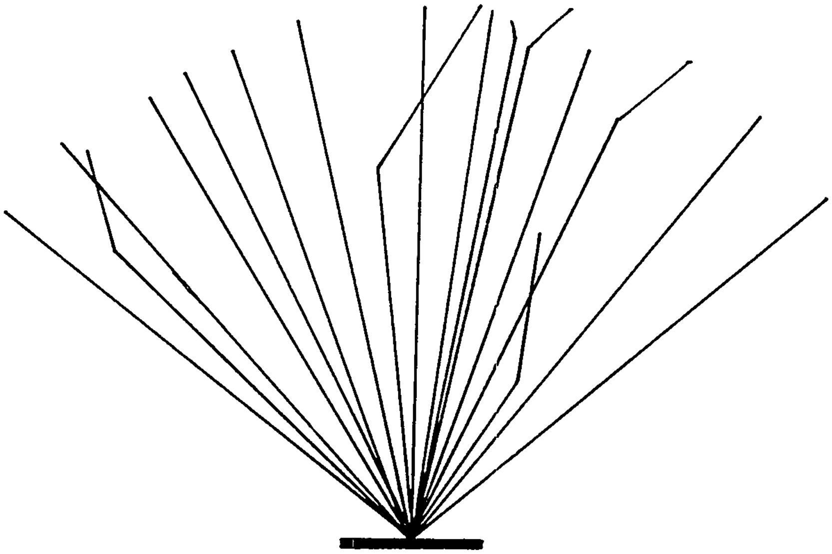


Figure 3.1 Illustration of alpha tracks as seen in a cloud chamber. (From F.Rasetti, Elements of Nuclear Physics. New York: Prentice-Hall, Inc., 1947; Fig. 53, p. 303)

interaction of the electrostatic fields of the particle and the orbital electrons of the absorbing medium (see Figure 3.2). Excitation occurs when the energy transferred to the electron is not enough to remove it from the atom (see Figure 3.3). In this case an ion pair does not result from the loss of energy by the α. Consequently, the average energy expended by an α particle in creating an ion pair in a given substance is usually greater than the ionization potential of the medium.

Other energy loss processes are available to α particles, but these generally do not occur very frequently except at higher energies.[1] For example, nuclear reactions can occur with α, but their range is so short in most media, that unless the α emitter is incorporated into a light element (such as Be,B), no reactions can take place. When α particles are accelerated to $E > 20$ MeV, then neutron production by reactions with nuclei become feasible.

1. Specific Energy Loss

The W value is defined as the mean energy needed to create an ion pair in a given substance. This value is different for α parti-

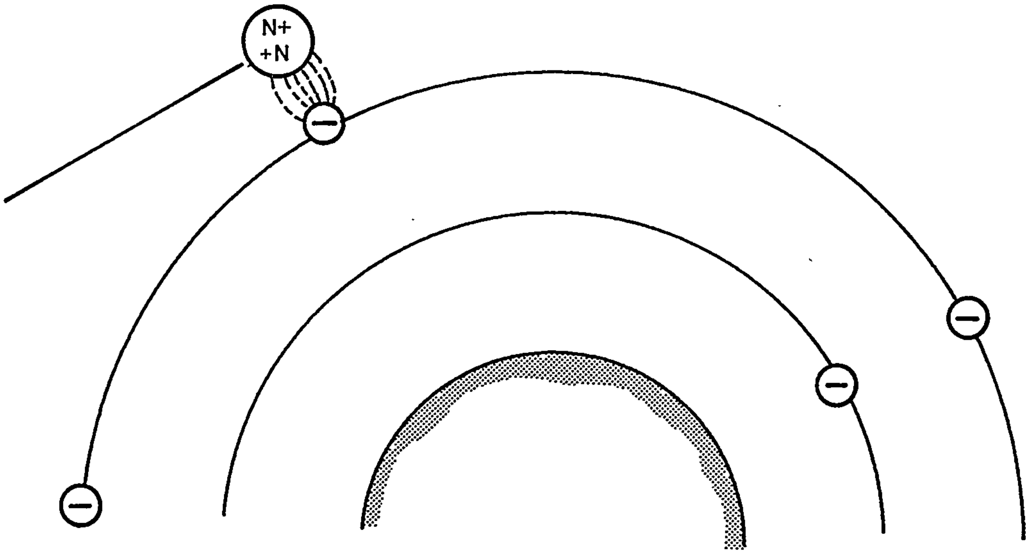


Figure 3.2 Electrostatic interaction.

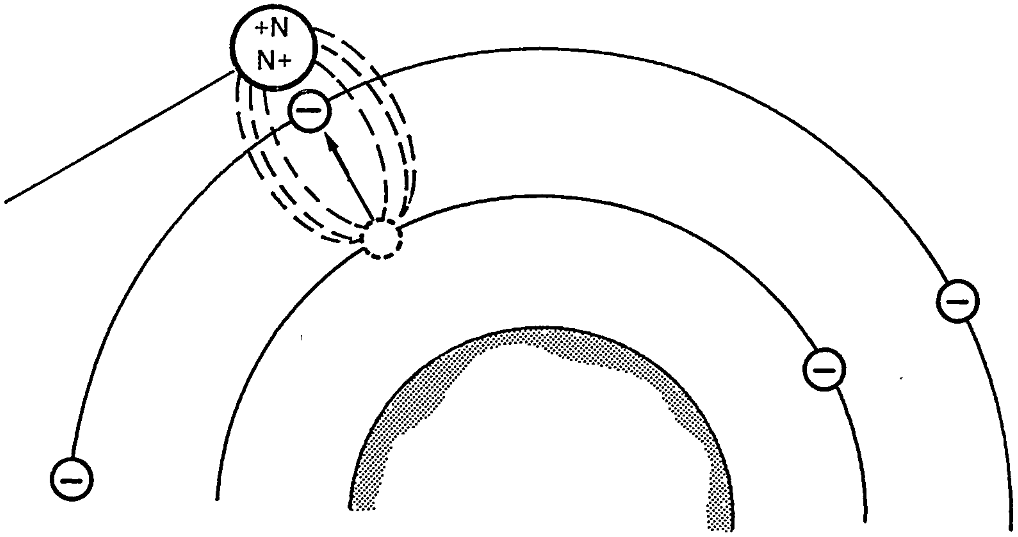


Figure 3.3 Excitation.

cles in various gases, ranging from 22 in xenon to 46 eV/ion pair (ip) in pure helium, but is relatively constant in a given gas for different α energy. The W value for α particles in air is 35.08 eV/ip (5.6×10^{-18} J/ip).

Because of the double positive charge and its large mass, an α particle forms a rather high number of ion pairs per unit path length. The specific energy loss along the path of the α is very high.
In air, an α may produce > 10^6 ip/m. So that, one may get a relative idea of energy loss by measuring the ionization produced along the path.

If the ionization produced by an α particle is plotted against the distance of penetration in a substance, the ionization gradually increases as the particle loses energy (see Figure 3.4). Finally, the ionization reaches a peak value and drops to zero as all the α energy is lost. An α whose energy is entirely lost in a substance will pick up two electrons and become a neutral helium atom.

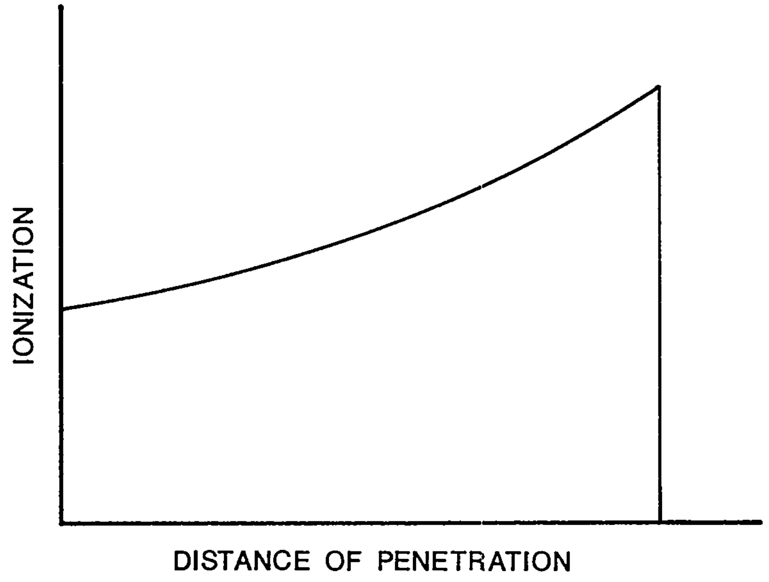


Figure 3.4 The ionization increases as the velocity of the alpha particle decreases with the depth of penetration. (R.E.Lapp/H.L.Andrews, NUCLEAR RADIATION PHYSICS, 2/e, 1954, p.139. Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, NJ)

2. Stopping Power

The linear stopping power S is related to the specific energy loss in the material. It is seen from Figure 3.4 that the energy loss per unit path length increases as the α loses energy. One may define S, the linear stopping power, as

$$S = -\frac{dE}{dl} \qquad 3.1$$

in which $\frac{dE}{dl}$ gives the energy transferred as ionization and excitation along the path of the charged particle. The mathematical expression for the linear stopping power fails for α particles of very low energy. Therefore, one cannot express the ionization loss at the very end of the path of the α particle.

The linear stopping power for heavy particles of charge ze is given by the expression

$$\frac{dE}{dl} = \frac{4\pi e^4 z^2}{m_o v^2} NB = K_\alpha NB \qquad 3.2$$

in which K_α will be a constant for a given charged particle of velocity v, N is the atoms/cc of the medium, m_o is the rest mass of the electron, and B the atomic stopping number (which is a function of the speed v of the particle, and the atomic number [Z] and the mean ionization potential [I] of the medium). The linear stopping power is expressed in SI units as J/m. B is a slowly varying function of the velocity, so that the linear stopping power varies inversely with the velocity squared. As the α loses energy, and its velocity decreases, the linear stopping power will increase which results in increased ionization near the end of the path.

The mass stopping power, S/ρ, is the linear stopping power divided by the density ρ of the substance:

$$\frac{S}{\rho} = \frac{1}{\rho}\frac{dE}{dl}\frac{J\ m^2}{kg}\left(6.25\times10^{13}\ \frac{MeV\ cm^2}{g}\right) \qquad 3.3$$

In health physics work, a quantity of importance is the ratio of two stopping powers or the relative stopping power. For α particles, air is often taken as the reference substance, and other media are compared to air. The relative mass stopping power $(S/\rho)_m$ of a medium m with respect to air is then:

$$(S/\rho)_m = \frac{S/\rho \text{ of the medium}}{S/\rho \text{ of air}} = \frac{(B/A)_m}{(B/A)_{air}} \qquad 3.4$$

where A is the effective atomic weight.

3. Range

Because of the large specific ionization of α particles, they use up their energy in short distances. For instance, the range in air will be only a few centimeters. Most of the alphas from a given source lose their energy in about the same distance. The range R of α particles in air at 15°C and 760 mm Hg can be estimated with the aid of the relationships:

$$R_a(\text{in m}) = 5.6\text{x}10^{-3}\ E \text{ (for } E<4 \text{ MeV)} \qquad 3.5$$

$$= 3.18\text{x}10^{-3}\ E^{3/2} \text{ (for } 4<E<7 \text{ MeV).}$$

A convenient Rule of Thumb for roughly estimating α range in air is

$$R_{air} \sim \left(\frac{E\alpha}{80} - .025\right) \text{ m for } 4<E<8 \text{ MeV} \qquad 3.6$$

The range of α in media other than air can be found approximately by the Bragg-Kleeman relationship:

$$R_m = \frac{\rho_a R_a}{\rho_m (S/\rho)_m} \qquad 3.7$$

where $(S/\rho)_m$ is the relative mass stopping power of the medium with respect to air. Bragg showed that for a number of substances:

$$B_m/B_{air} \sim \frac{\sqrt{A_m}}{\sqrt{A_a}}. \qquad 3.8$$

From this relationship, then, since from equation 3.4,

$$(S/\rho)_m = \frac{B_m A_a}{B_a A_m} \sim \frac{\sqrt{A_m}}{\sqrt{A_a}} \frac{A_a}{A_m} = \frac{\sqrt{A_a}}{\sqrt{A_m}} \qquad 3.9$$

we find for the range in any substance

$$R_m \text{ (in m)} \sim \frac{0.32\sqrt{A_m}}{\rho_m} R_a, \qquad 3.10$$

where $\sqrt{A_a} = 3.82$ and $\rho_a = 1.226$ kg/m³ at 15°C and 760 mm of Hg. This relationship is usually good with ± 15%. The quantity $\sqrt{A_m}$ for a compound or a mixture can be found from the expression[2]:

$$\sqrt{A_m} = \frac{n_1A_1 + n_2A_2 + n_3A_3 + \ldots\ldots\ldots}{n_1\sqrt{A_1} + n_2\sqrt{A_2} + n_3\sqrt{A_3} + \ldots}, \qquad 3.11$$

where n_i is the atomic fraction of the element i of atomic weight A_i.

Since the mass stopping power in tissue, (S/ρ), for alphas is almost equal to the mass stopping power in air, $(S/\rho)_t = 1$, and

$$R_{air}\, \rho_{air} = R_{tissue}\, \rho_{tissue}. \qquad 3.12$$

Because there is such a great difference in density between a solid and a gas, the range of alphas in solids will be very small. Find the range of ^{238}U α particles (E = 4.198 MeV) in UO_2 ($\rho = 1.09 \times 10^4$ kg/m³).

$$\sqrt{A_m} = \frac{\frac{1}{3}(238) + \frac{2}{3}\, 16}{\frac{1}{3}\sqrt{238} + \frac{2}{3}\sqrt{16}} = \frac{90}{7.81} = 11.52.$$

$$R_a = 3.18\text{x}10^{-3}E^{3/2} = 3.18\text{x}10^{-3}(4.198)^{3/2}$$

$$= 2.74\text{x}10^{-2} \text{ m in air (2.74 cm).}$$

$$R_{UO_2} = \frac{.32(11.52)2.74\text{x}10^{-2}}{1.09\text{x}10^4} = 9.27\text{x}10^{-6} \text{ m}(9.27\ \mu\text{m})$$

4. Relative Hazard

The fact that α particles have short ranges in dense substances makes the α particle much less of a hazard to humans than other external radiations. The α particle of highest energy emitted by natural radioactive substances will just penetrate the "dead" layer of skin on the human body. So, little living tissue will be affected. With the source on the skin surface, an α particle needs approximately 7.5 MeV to get through the dead layer. For most α emitters then, no living tissue will be damaged when the source is external to the body.

Once inside the body, the short range and high specific ionization of the α particle becomes very important. In this case, the α source will be surrounded by living tissue. This means that the damage will be highly localized near the point of origin of the alphas. Thus, great damage can be done to small essential organs of the body if an α source is lodged in them, since all the α energy will be absorbed in that particular organ and not spread out over a larger volume of tissue. In addition, the more dense energy deposition along the path of the α has been found to be more effective in producing damage.[3] For these reasons, α particles are a very real concern as an internal hazard.

B. Beta Particles

Beta particles were found to be high-speed electrons emitted from the nucleus of the unstable atom. Further investigation has shown that either positive electrons (positrons) or negative electrons may be emitted by a radionuclide. The concept of β emission has been extended to either emission of positrons and electrons. The rest mass of these particles is the same; they have equal but opposite charges.

Unlike alphas, betas are not emitted with discrete energies but show a continuous energy spectrum (see Figure 3.5). The atom emits electrons of all energies up to some maximum value. This maximum value for the β spectrum may be found in nuclide tables and is characteristic of that atom. Most known β sources emit particles with maximum energy up to about 4 MeV.

The average energy ($\bar{E}$) of a source is about 1/3 E_{max} in the case of electrons (β^-) and 0.4 E_{max} in the case of positrons (β^+). In 1934, Enrico Fermi developed the theory of β^- and positron decay. His theory was able to predict the shape of the β spectrum. He assumed that when a β^- was emitted, a neutron converted to a proton and an anti-neutrino was also emitted. In Figure 3.5, the sum

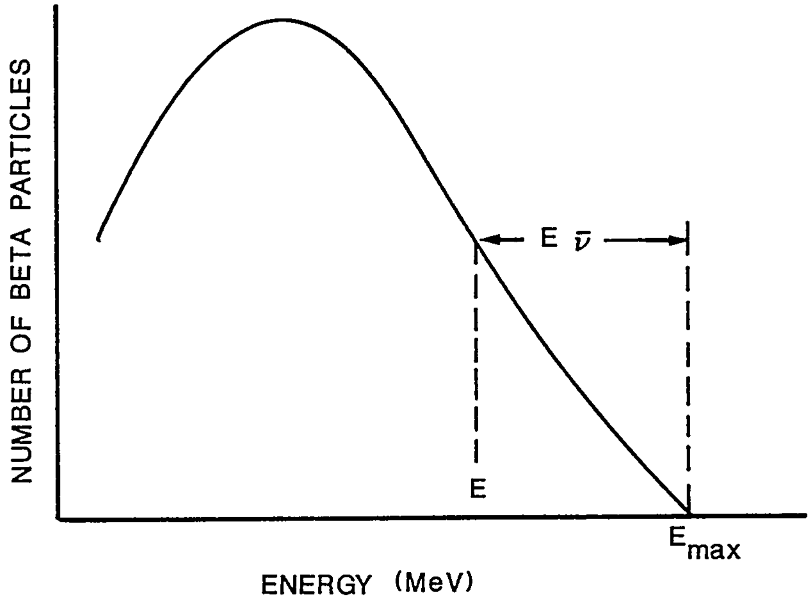


Figure 3.5 Typical beta spectrum.

of the energy of the β (E) and the anti-neutrino ($E_{\bar{\nu}}$) equals E_{max}. In the case of positron emission, the sum of the energy of the positron and the emitted neutrino is equal to E_{max}.

Electrons lose energy in a number of ways as they pass through matter. Loss by collisions with orbital electrons, leading to ionization and excitation, is the most frequent mechanism. But interactions with the nucleus do occur and lead to the emission of x rays (bremsstrahlung). The latter process occurs predominantly for the more energetic electrons, E> 1 MeV (see 3B.3).

When an electron reaches an energy of 10-15 MeV, high energy photons are produced which can interact with the nucleus, and neutrons are released. Since this energy is well above the highest known emission energy of betas, these reactions take place mainly in electron accelerators.

An important reaction for positrons (see 3E.4) is annihilation. In this process, the combination of matter (e^-) and anti-matter (e^+) results in the release of energy (2 γ rays of 0.511 MeV each).

In a given substance, the electron will make a large number of collisions until it eventually loses all of its kinetic energy. The processes leading to ionization in the substance are the same for the electron as those for the α. But because of its smaller mass (approximately 1/7300 that of an α) and lower charge (1/2 that of an α), the interactions take place at less frequent intervals. Consequently, electrons do not produce as many ions per meter of path as do alphas.

1. Specific Energy Loss

The W value for electrons in air is 33.85 eV/ip (5.42×10^{-18} J/ip). The ionization values for electrons are much lower than those for alphas. Roughly, the specific energy loss for electrons varies inversely as the square of the speed of the electron for energies up to 10 MeV. Ionization values in air for electrons range from approximately 10^4-10^5 ip/m.

At low energy, the specific energy loss decreases as energy increases (see Figure 3.6). It reaches a minimum value above 1 MeV. Then,

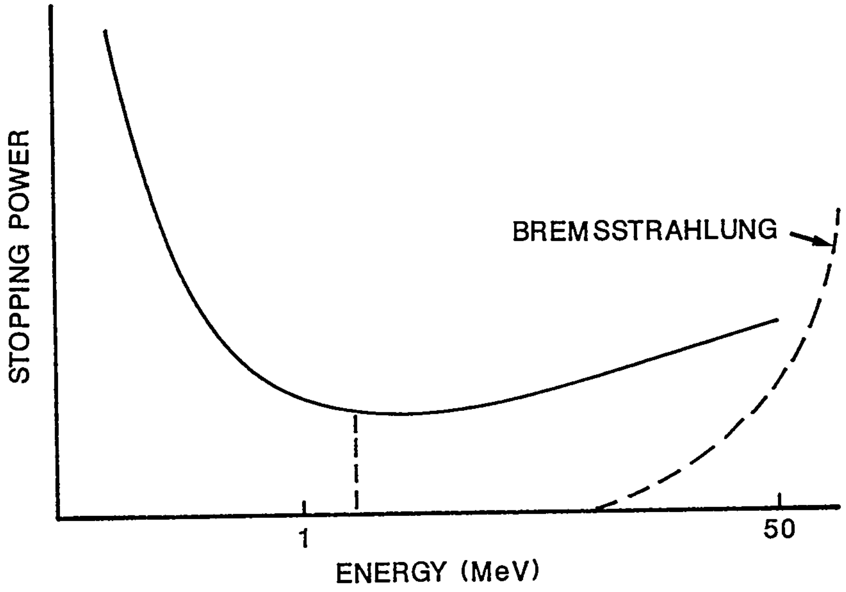


Figure 3.6 The general shape of the stopping power vs. energy curve for beta particles.

the total energy loss increases as the energy increases due to relativistic effects and bremsstrahlung production.

The specific energy loss for betas should be higher than the value for monoenergetic electrons of energy equal to the maximum of the beta spectrum. This is expected because most of the betas emitted by a given source will have lower energies than the maximum value of the spectrum. Since the specific energy loss for electrons increases as the energy decreases, the value obtained for the spectrum should be higher than that for monoenergetic electrons.

2. Stopping Power

The linear stopping power for electrons due to ionization and excitation is given by:

$$\frac{dE}{dl} = \frac{2\pi e^4 NZ}{m_o v^2} B' = K_e NZB' , \qquad 3.13$$

where N is the atoms/cc of the medium of atomic number Z. The electron stopping number, B′, is a function of the speed of the electron and the mean ionization potential of the substance. This expression gives the rate of energy loss along the actual path of the electron. Since electrons are greatly scattered in passing through a given substance, the actual path traveled is often much greater than the straight-through path (about 1.2-4 times the straight line path).

3. Radiation Production - Bremsstrahlung

In addition to that energy lost in ionization and excitation by an electron in passing through matter, some energy may be given up in the form of x rays produced in the substance as a result of interactions between high-speed electrons and nuclei of the absorbing substance. The interactions leading to the production of bremsstrahlung are inefficient for electron energies below 1 MeV. As the energy increases above 1 MeV, the production of x rays becomes increasingly more important. However, this process is the source of x ray production from commercial tubes used in both dentistry and medicine.

When a charged particle is either accelerated or decelerated in an electric field, electromagnetic radiation may be given off. If an electron passes close to an atom while traversing a substance, the charge Z on the nucleus will exert a force on the electron. This will cause its path to be bent (see Figure 3.7). During this acceleration, the electron may radiate energy of any amount from zero up to its total kinetic energy E_k. The total bremsstrahlung per atom is roughly proportional to $(Z/m)^2$, where Z is the atomic number of the absorbing matter and m is the mass of the charged particle. Because of a $1/m^2$ dependence, the amount of bremsstrahlung is almost completely negligible for all particles except electrons, unless the particle energy is in the GeV range.

For electrons in thin targets, the maximum intensity of the bremsstrahlung will be at right angles to an incident beam of low-energy electrons. As the beam energy increases to very high values, the maximum shifts to the forward direction.[4-6]

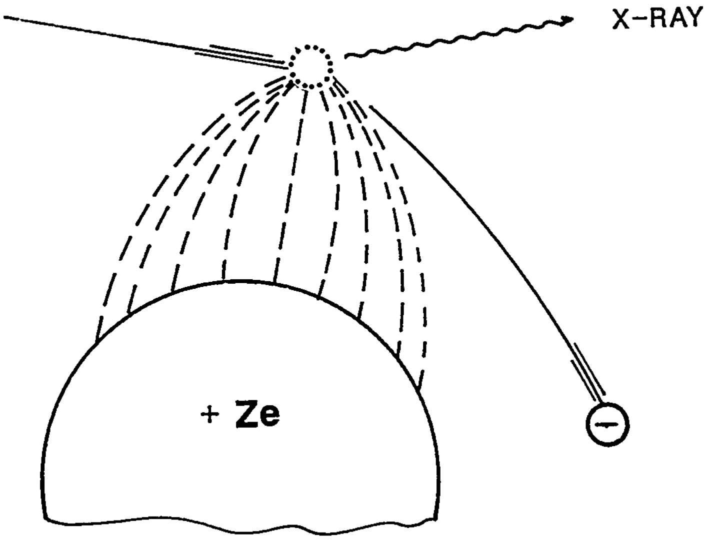


Figure 3.7 Bremsstrahlung.

For very high energy, the bremsstrahlung is very highly peaked in the forward direction. The angular distribution is mostly confined to a cone of half-angle θ, given by[7]

$$\theta = \frac{m_o c^2}{E} \sim \frac{1}{2E} \text{ (rad)}, \qquad 3.14$$

in which E is the total energy of the electron (MeV).

For an electron beam incident on a thick target, the fraction of its initial energy which is converted to x rays is given approximately by[2]

$$F \sim 7\text{x}10^{-4}\ ZE_k, \qquad 3.15$$

where Z is the atomic number of the absorbing substance and E_k is the energy in MeV of the beam. Thus, a 1 MeV beam will lose about 6% of its energy as x rays when the beam is absorbed in lead (Z=82).

In the case of a β source, the fraction of β energy converted to x rays is approximately[8,9]

$$F \sim 3.33\times10^{-4}\ ZE_{max}, \qquad 3.16$$

where E_{max} is the maximum energy of the β spectrum. In this case, a 1 MeV β will lose about 3% of its energy as bremsstrahlung when absorbed in lead. The spectral distribution will contain x rays of energy from zero up to E_{max}. In most cases, one may assume that the x rays which are emitted correspond to 1/3 the maximum energy of the β for purposes of shielding design for thin shields. This will normally result in a safety factor in the calculations, if one uses the attenuation coefficient for $\frac{E_{max}}{3}$.

Returning to Figure 3.6, the effect of bremsstrahlung on the total energy loss is shown by the dotted line. For medium Z materials, the linear stopping power and the linear bremsstrahlung loss become comparable around 50 MeV. Above this energy, the radiation loss dominates. One may roughly estimate the ratio of radiative energy loss (bremsstrahlung) to collision energy loss (ionization and excitation) by[9]

$$\frac{S_{rad}}{S_{col}} = \frac{\left(\frac{dE}{dl}\right)_{rad.}}{\left(\frac{dE}{dl}\right)_{col}} \sim \frac{EZ}{700} \qquad 3.17$$

in which E is the kinetic energy of the electron (MeV) and Z is the atomic number of the absorbing medium.

EXAMPLE - What is the ratio of the radiative energy loss to the collision energy loss of a 1 GeV e^- in copper (Z=29)?

$$\frac{S_{rad}}{S_{col}} = \frac{(1000)(29)}{700} = 41.4.$$

The result indicates that the electron loses >97% of its energy by bremsstrahlung production at this energy!

4. Range

The stopping power for electrons decreases as the atomic number Z of the absorber increases. This occurs because substances of high Z have fewer electrons per gram and these are more tightly bound. Consequently, the range tends to increase as Z increases. But as Z increases, the multiple scattering of the electrons increases. The effect of multiple scattering is to increase the actual path of the electron in a substance. This tends to decrease the range which is the linear distance through the medium. These two effects act to balance each other, so that the density of a substance gives one a good idea of its relative ability to stop electrons. It is common to express the range of electrons in terms of density-thickness, measured in kg/m^2, i.e., $t(m)\rho(kg/m^3)$ = Range (kg/m^2). The range is then nearly independent of the type of absorbing substance.

Katz and Penfold[10] have expressed the relationship between the range and the energy (MeV) of the electron in mathematical form. Expressed in SI units, this becomes

$$R(kg/m^2)=4.12\ E^{1.265-0.0954\ln E}(0.01<E<2.5\ \text{MeV}) \qquad 3.18$$

or

$$\ln E = 6.63 - 3.2376\ (5.6093-\ln R)^{1/2}, \qquad 3.18a$$

and

$$R(kg/m^2) = 5.30E - 1.06\ (E>2.5\ \text{MeV}). \qquad 3.18b$$

These expressions can be used to find the range of electrons in any substance in terms of kg/m^2. As far as a continuous β spectrum is concerned, it turns out that the maximum range of the β is equal to the range of a monoenergetic electron, whose energy is the same as the maximum energy of the β spectrum. Thus, one can use the same relationship to find the range of betas in matter.

What is the range of a 1.5 MeV β in kg/m^2 of any substance?

$$R = 4.12\ (1.5)^{1.265-0.0954\ln 1.5} = 4.12(1.5)^{1.265-0.0954(0.40547)}$$
$$= 4.12\ (1.5)^{1.265-0.0386} = 4.12\ (1.5)^{1.226};$$
$$\ln R = \ln 4.12\ (1.5)^{1.226} = \ln 4.12 + 1.226\ \ln 1.5$$
$$= 1.41585 + 0.49710 = 1.91295$$
$$R = 6.77\ \mathrm{kg/m^2}\ (677\ \mathrm{mg/cm^2}).$$

From a practical standpoint, the average range of a β source turns out to be much less than this. Most of the particles have energies much less than the maximum energy. The average range of a β source is taken as about 0.2 R.[11]

A Rule of Thumb which can be used to estimate the maximum range of β in any medium is:

$$R \sim 5E\ (\frac{\mathrm{kg}}{\mathrm{m^2}}) \qquad 3.19$$

in which E is the maximum β energy (MeV). The approximation is good above 1 MeV but greatly overestimates the range for lower energies. For β radiation in air, the range is about 3.5 m/MeV.

5. Exponential β-particle Absorption

Suppose β particles from a source traverse absorbers of various thicknesses. The ionization produced before and after passing through a given thickness t of the substance is measured. It is found that a plot of the ratio of ionization after passage through an absorber to the initial ionization versus absorber thickness on semilog paper yields a straight line. This is similar to the plot one gets for radioactive decay. It tells us that beta absorption is an exponential process. The slope of the straight line gives the constant fractional decrease in the ionization per unit thickness of the substance. This constant, denoted by μ, is called the attenuation coefficient. The equation describing β absorption is

$$I = I_o e^{-\mu t}, \qquad 3.21$$

where I_o is the initial intensity, I is the intensity after passing through a thickness t of absorber (in m), and μ is the attenuation coefficient (in m^{-1}). Experiments have shown that the mass attenuation coefficient μ/ρ (m^2/kg), where ρ is the density of the substance in kg/m^3, is almost independent of the atomic weight of the absorber. Thus, one can express the mass attenuation coefficient by[12]

$$\frac{\mu}{\rho} \sim 1.7\ E_{max}^{-1.14}\ m^2/kg \qquad 3.22$$

for most substances, where E_{max} is in MeV. In terms of the mass attenuation coefficient

$$I = I_o e^{-(\mu/\rho)x}, \qquad 3.23$$

where x is expressed as kg/m^2, i.e., $t\rho$. This relationship is valid provided the distance of penetration is well within the range of the β.

6. Relative Hazard

Depending upon the particle energy, a β^- or positron source may constitute an external hazard. Because of their greater range, β particles of 70 keV are able to penetrate the dead layer of skin and hence deposit their energy in living tissue. Since most vital organs are at a depth of 50 mm or more, they will be unaffected by β particles from sources having tissue ranges less than this, which is the case for most β^- and positron emitters. This factor reduces an external β source to primarily a skin-exposure problem. Since most β particles are easily absorbed by a small amount of metal or plastic, the hazard will usually exist only very near the source.

An external hazard may result from x rays produced when a high-energy β particle is stopped in an absorber (bremsstrahlung radiation). The amount of such x ray production can be minimized by using low-Z-number absorbers. For positron emitters, the annihilation γ rays require thicker absorbers to reduce the external hazard.

As an internal hazard, β particles are not as significant as α particles. The greater tissue range of β particles means that damage will not be as localized as for α particles. A much larger tissue volume about the deposited β source will be involved. Some of the energy may actually be lost outside a small organ of concern. Annihilation γ rays or bremsstrahlung may not even interact in the tissue mass, which means that some of the available energy will not be absorbed by the tissue. Also, the less dense energy deposition along the path of β particles is less effective in producing damage than that for α particles.[3] Nevertheless, damage may still be significant, so that β emitters are of concern as an internal radiation source.

C. Wave Properties

Up to this point, we have been dealing with particles. We turn our attention now to waves. A wave may be defined as a disturbance in a medium. The wave moves through matter with a definite velocity and transmits energy. One sees a wave motion moving across a smooth body of water. These waves show the advance of a form of motion which has a definite velocity. From the movement of water, one can infer that energy is being propagated. In this case, the wave disturbs the surface of the water. However, only the surface moves to any extent. In this example, the disturbance moves through the entire medium but the medium itself does not move greatly.

If one fastens a rope to a rigid body and moves the free end up and down, a wave motion will be set up in the rope. The wavelength is the distance between two adjacent particles in the same phase or displacement (Figure 3.8). The symbol λ is often used to represent wavelength. The frequency of a wave, ν, is the number of waves which pass a point in unit time. The velocity, v, of the wave is given by

$$v(\text{in m/s}) = \nu\lambda, \qquad 3.24$$

where ν is the number of waves/s and λ is the m/wave.

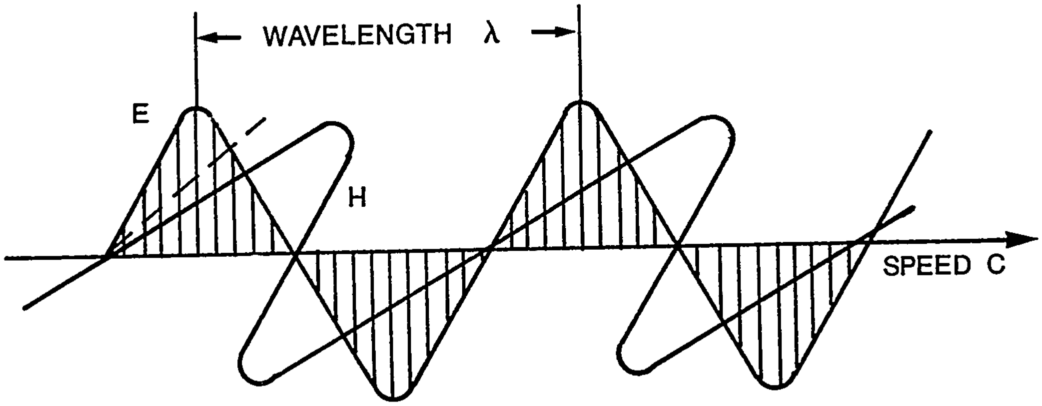


Figure 3.8 Electromagnetic wave. (R.E.Lapp / H.L.Andrews, NUCLEAR RADIATION PHYSICS, 2/e, 1954, p.139. Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, NJ)

The velocity of the wave is a property of the medium. That is, the same wave moving through different media will exhibit different velocities. The change in velocity when a wave travels from one medium to another is called refraction. The frequency of the wave is a property of the source of the disturbance. A related property, the period τ, is the time for 1 wave ($\tau = 1/\nu$ s/wave).

Other wave properties of some interest in health physics work are diffraction, reflection and interference. The term diffraction is used to denote scattering of waves. Reflection of waves occurs at the interface of two dissimilar media and is usually referred to as backscatter. Reflection denotes that the wave is deflected to some degree. Interference is a term used to denote the combination of waves to form a new wave motion.

D. Electromagnetic Waves

Electromagnetic waves are of main concern to the health physicist. When an electromagnetic wave moves through matter, there is set up an electric field E perpendicular to both the electric field and the direction of motion (see Figure 3.8). As the E field changes it gives rise to a magnetic field. Since the magnetic field is also changing it gives rise to an electric field. In this way, each "feeds" the other. Electromagnetic waves are transverse waves. This means the disturbance is transmitted perpendicular to the direction of propagation. Microwaves, heat, light, ultraviolet x rays and γ rays are electromagnetic waves which differ only in wavelength. Our interest is the properties of x and γ rays.

The speed of an electromagnetic wave in a vacuum is $3\text{x}10^{8}$ m/s. The letter c is used to denote this speed.

X and γ rays have very short wavelengths. (It is common to denote wavelengths in terms of the angstrom unit, A^{o}; one angstrom unit equals 10^{-10} m). X and γ rays have wavelengths in the range from 10^{-8} to 10^{-13} m.

Up to 1900, the classical concept pictured radiant energy as being emitted and absorbed in a continuous manner. In an attempt to derive the radiation law theoretically, Max Planck abandoned the classical concept. He assumed that energy was radiated as quanta of energy which have come to be called photons. These small packets (quanta) were given up in integral units by an emitter. He stated that the quantum of energy, E, is directly proportional to the frequency ν of the radiation. These two terms are related by a fundamental constant, h. Planck called h the action constant, but it is now known as the Planck Constant and has a value of 6.626 x 10^{-34} Js. The energy E of radiation of frequency ν is given by

$$E = h\nu = hc/\lambda. \qquad 3.25$$

The intensity of a wave gives the time rate at which the wave transmits energy. By definition, the intensity is the average time rate at which energy is transported by the wave per unit area across a surface which is perpendicular to the direction of motion of the wave. The intensity of x and γ rays may be given in units of MeV/m^2 s. For an isotropic point source of radiation, the intensity at a point varies inversely as the square of the distance of that point from the source. Given a point source of radiation, the intensity at point A distant r_A from the source is given by

$$I_A \propto 1/r_A^2 \tag{3.26a}$$

or

$$I_A = k/r_A^2 \tag{3.26b}$$

At point B, the intensity is similarly given by

$$I_B = \frac{k}{r_B^2} \tag{3.26c}$$

From these two equations we get

$$\frac{I_A}{I_B} = \frac{r_B^2}{r_A^2} \tag{3.26d}$$

or

$$I_A r^2 = I_B r^2_B. \tag{3.27}$$

This expression is the mathematical form for the inverse-square law. The law is subject to two conditions:

1) The attenuation of the radiation in the intervening space must be negligible.

2) The dimensions of the source and the detector at the point must be small compared with the distance between them.

A useful Rule of Thumb, in applying condition 2 above, states that if the distance between is greater than 3x the larger dimension of either the source or receptor, a point source may be assumed. The error will generally be <5%.

The inverse square law expresses the general relationship between the source and the point of interest. If one knows the value of the intensity at A, he or she can find the value at B. The intensity at A is given in terms of the energy emitted by the source and the distance r_A between A and the source. Let S be the power (MeV/s) emitted by the source. This energy is emitted in all directions about the point source. If the point source is assumed to be at the center of a sphere of radius r(m), all the energy emitted per unit time will pass through the surface of this sphere. The intensity (energy fluence rate) about point A is then

$$I \text{ (in MeV/m}^2\text{s)} = S/\text{Surface area of Sphere} = S/4\pi r^2. \qquad 3.28$$

EXAMPLE: A point source of ^{137}Cs emits $2\text{x}10^{10}$ MeV/s. Find the intensity at a distance of 1 m from the source. Use the inverse square law to compute the intensity at 3 m.

$$I = \frac{S}{4\pi r^2} = \frac{2 \times 10^{10} \frac{\text{MeV}}{\text{s}}}{4\pi(1)^2(\text{m}^2)}$$

$$= 1.59\text{x}10^9 \frac{\text{MeV}}{\text{m}^2\text{s}} \left(1.59\text{x}10^5 \frac{\text{MeV}}{\text{cm}^2\text{s}}\right),$$

from equation 3.27

$$I_A r_A^2 = I_B r_B^2$$

$$I_B = \frac{1.59\text{x}10^9 (1)^2}{(3)^2} = 1.77\text{x}10^8 \frac{\text{MeV}}{\text{m}^2\text{s}} \left(1.77\text{x}10^4 \frac{\text{MeV}}{\text{cm}^2\text{s}}\right)$$

E. X and γ Radiation

Roentgen discovered x rays in 1895, so termed because of their unknown nature. From his experiments the following properties of this new radiation were determined:

1) Most substances are transparent to x rays.
2) Many substances glow when exposed to x rays.
3) X rays produce ionization in gases.
4) X rays are produced when energetic electrons strike solids.
5) Photographic plates are affected by x rays.
6) X rays are not deflected by electric or magnetic fields.

The same properties were found for γ rays by other workers later. Further work showed that x and γ rays were electromagnetic waves. The only distinction made at the present time is one of origin: gamma rays refer to radiation emitted from the nucleus of the atom; x rays refer to radiation which is produced outside of the nucleus of the atom.

When a beam of energetic electrons is stopped in any dense substance, x rays (bremsstrahlung) are produced. The spectrum of x rays has a continuous distribution from zero up to the energy E of the electrons. Some lines of much greater intensity than the bremsstrahlung also appear (see Figure 3.9). These are called characteristic x rays. The wavelengths of these lines are a property of the target substance. Characteristic x rays appear when electrons from the inner shells (K, L, M) undergo transitions.

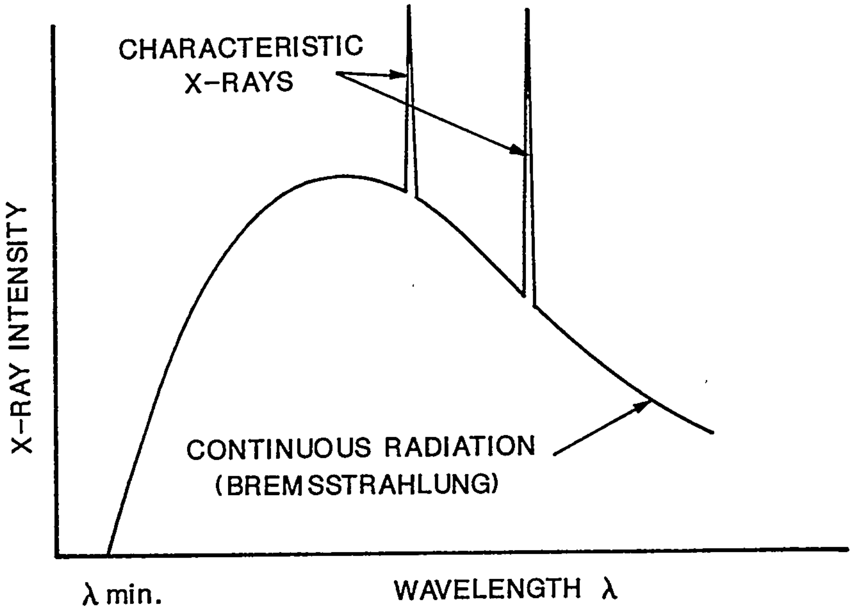


Figure 3.9 Typical x-ray spectrum. (R.E.Lapp / H.L.Andrews, NUCLEAR RADIATION PHYSICS, 2/e, 1954, p.102. Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, NJ)

The characteristic lines may be excited by any method which will remove electrons from the inner shells. If a K-shell electron is removed, all characteristic lines may appear. If the electrons in the outer orbits of the atom undergo transitions, the characteristic radiation may be in the visible range. The visible spectrum given off by a substance will have lines in the range from 4000 to 8000 A^o. The only difference between x rays and the optical spectrum is to be found in the amount of energy associated with the lines. X rays have shorter wavelengths, i.e., higher energy, than the lines of the optical spectrum.

As a low energy electron beam strikes a target, a large portion of its energy is lost as heat. For example, for a 300 keV e^- beam striking a tungsten (Z=74) target, the ratio of radiative loss to collision loss is given by equation 3.17:

$$\frac{S_{rad}}{S_{col}} \sim \frac{EZ}{700} = \frac{0.3(74)}{700} = 3.2x10^{-2}$$

The total E loss is $S_{rad} + S_{col} = (3.2x10^{-2} + 1)S_{col}=1.032\ S_{col}$

so that $S_{col} \sim 97\%$ of the total E loss.

Part of the energy is lost in x ray production (~ 3%), and there is a probability that all the kinetic energy may be converted to a photon. The wavelength of this photon will be given by equation 3.25

$$E = h\nu = hc/\lambda_{min},$$

where E represents the energy of the electron. The electron energy is also given by

$$E = Ve, \qquad 3.29$$

where V is the accelerating voltage and e is the charge on the electron. Thus, we obtain

$$\lambda_{min}(\text{in m}) = \frac{hc}{Ve} = \frac{(6.626x10^{-34})(3x10^{8})}{V\ (1.6x10^{-19})} = \frac{1.24x10^{-6}}{V} \qquad 3.30$$

or

$$\lambda_{min} \text{ (in } \overset{o}{A}) = 1.24x10^4/V. \qquad 3.31$$

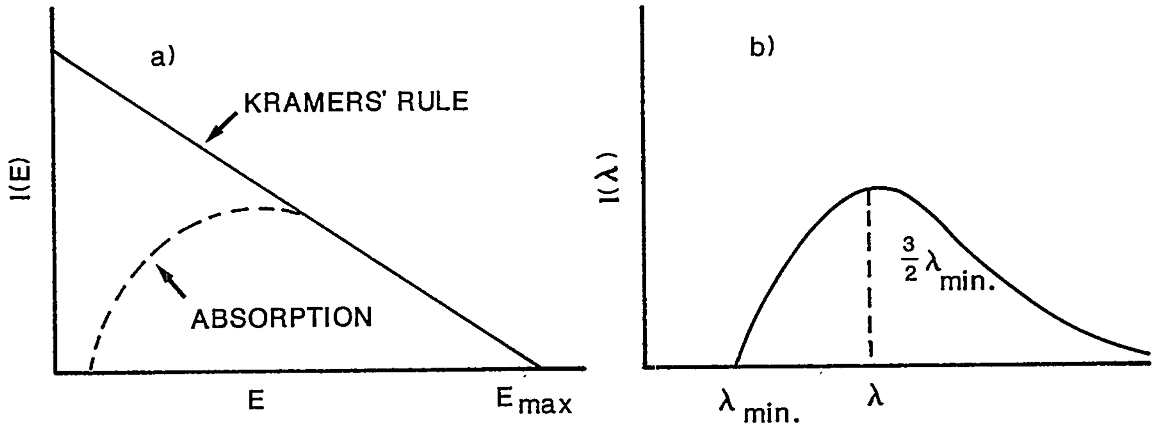
Here, λ_{min} is the wavelength of the highest-energy x ray produced for a given accelerating voltage V (volts). If V is increased, a higher-energy x ray can be obtained. Moreover, the shape of the response curve in Figure 3.9, remains about the same. A simple relation, useful for roughly predicting the shape of the response curve for thick targets, was developed by Kramers.[9,13] This is shown in Figure 3.10a, as the solid line, and is represented by

$$I = A(E_{max}-E). \qquad 3.32$$

Because of absorption in the window, and/or by filters, the lower end of the spectrum looks more like the dotted portion. The intensity is a function of the high voltage and beam current in an x ray tube. Depending upon the absorbing media and the filtration, as well as variations in the pulsating voltage, the shape of the spectrum may differ from that predicted by equation 3.32. As seen from both equation 3.30 and Figure 3.10a, the spectrum of x rays contains a small number of these higher-energy x rays. The highest intensity of x rays produced will occur at roughly 1.5 λ_{min} for the continuous portion (as shown in Figure 3.10b, which is the same distribution plotted as a function of the wavelength, λ.

Further details concerning x ray production equipment can be found in Chapter 4 of Reference 4.

Gamma rays are emitted from the nucleus of a radioactive atom. Most atoms, which decay by β emission, give off γ rays. Many atoms which decay by α emission also give off γ rays. Gamma rays are emitted when the nucleus of an atom has an excess energy above its lowest energy state after emitting either an α or β. In fact, these rays may be emitted from any nucleus in an excited state. Typical γ ray emissions from radioactive sources range from 10 keV - 7 MeV, but mostly < 4 MeV.



a) Intensity versus energy for Kramers' rule.

b) Intensity versus wavelength, λ.

Figure 3.10 X ray intensity.

1. Interactions With Matter

The transfer of energy to matter by α and β radiation occurs mainly by ionization. Each of these particles produces a rather large number of primary ions in passing through a substance. In the case of x and γ rays, the ionization which is produced is almost all secondary, that is, when an x or γ ray interacts with matter, only a small number of primary ions are formed. These ions in turn produce most of the ionization which occurs in the substance.

Particle radiations, such as α and β, are called directly ionizing radiation for the reason cited above, they produce ions directly along their respective paths. Photons, x and γ rays, are called indirectly ionizing radiation. This follows since most of the ionization occurs after the photon has interacted. That is, the electrons released by photon interactions lose their energy by producing ion pairs.

Three main ways in which x and γ rays from radionuclides interact with matter are by means of: the photoelectric effect, the Compton effect, and pair production. All three processes yield electrons which then ionize or excite other atoms of the substance. Other photon reactions require more energy to be feasible. When the photon energy exceeds about 10 MeV, photon-nuclear reactions can release photoneutrons. Notable exceptions, are γ reactions in ^{2}H and ^{9}Be, which are possible for much lower energies.[14]. For very high energy photons, the process of photon-induced fission (photofission) can also occur.

2. Photoelectric Effect

In the Planck concept, each x or γ ray is a photon with energy $E=h\nu$. The photon retains all of this energy until it interacts. The photon may interact with an electron in an orbit of an atom of the substance. In the photoelectric effect (see Figure 3.11), all of the

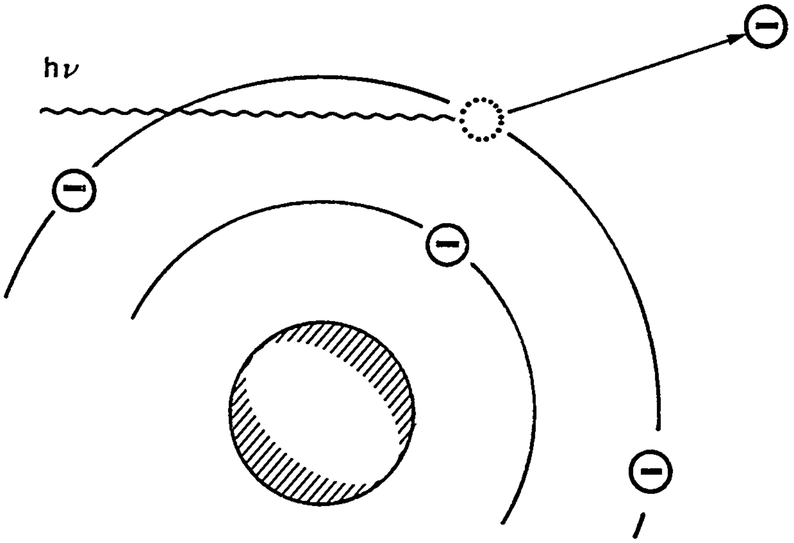


Figure 3.11 The photoelectric effect. (R.E.Lapp / H.L.Andrews, NUCLEAR RADIATION PHYSICS, 2/e, 1954, p.114. Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, NJ)

photon energy is given up in this process. Part of the photon energy is used in removing the electron from the atom: this is the work function ϕ. The rest of the photon energy is carried off as the kinetic energy of the electron. This electron will then cause ionization and excitation. The energy relationship between the photon and the electron is

$$E = h\nu = E_k + \phi, \tag{3.33}$$

where E_k is the kinetic energy given to the electron.

The photoelectric effect is important when the energy of the photon is low, i.e., less than 1 MeV. However, the photon energy must be greater than ϕ for the process to occur at all. For small values of E, the photons will interact with electrons in the outer shells of the atom. As E increases, more of the inner electrons are released. Also, the photoelectric effect is more likely to occur in materials with a high atomic number Z; the effect occurs to a greater extent in lead (Z=82) than in copper (Z=29). The photoelectric effect is approximately proportional to Z^{4-5}/E^{3-4}.

When the electron is removed from an inner orbit, the vacancy will be filled by one of the outer electrons. When such a transition occurs, a photon may be emitted. If the photoelectron is removed from the K shell, then all of the characteristic x rays may appear. These x rays are sometimes called fluorescent radiation.

Sometimes when the electron vacancy is filled by an electron from an outer shell, the energy is released in the form of an ejected electron from one of the outer shells (see Figure 3.12). These emitted electrons have kinetic energies which are equal to the energy of the characteristic x ray minus the binding energy of the electron. Such electrons are called Auger (oh-zhay) electrons.

Many x rays and Auger electrons may result when the photoelectric effect occurs in a heavy atom. These processes take place as the atom tries to return to its original ground state.

Another process which may occur for a radioactive substance which emits γ rays is called internal conversion. This is an alternative mode of decay, that is, the excited nucleus may emit a γ

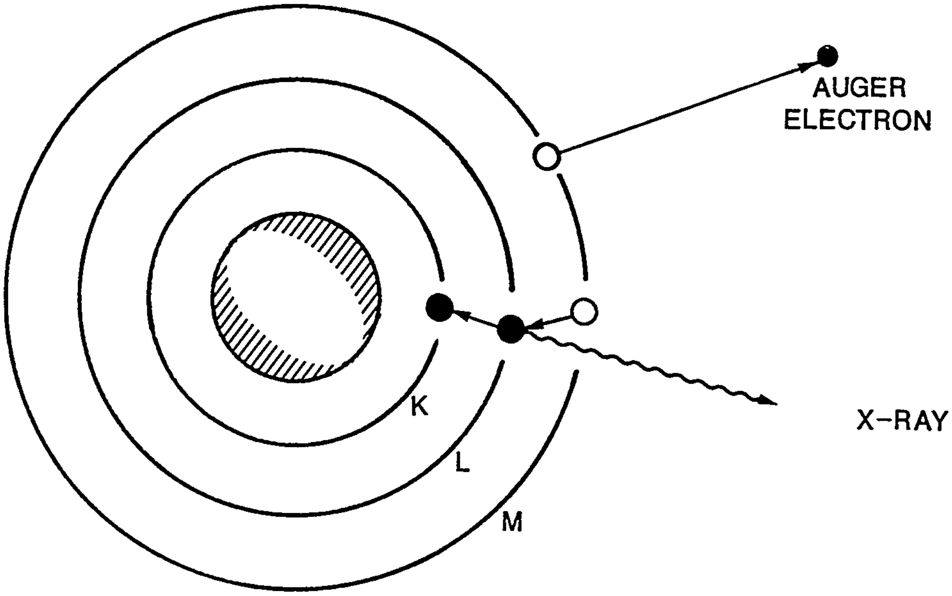


Figure-3.12 Auger electron effect. An L shell e^- fills a K shell vacancy, and an Auger e^- is simultaneously emitted from the M shell.

ray or it may eject an electron from one of the inner shells. In other words, the nucleus interacts with the electron to get rid of excess energy. If the nucleus emits the γ ray, the energy of the photon is given by $E=h\nu$. If an electron is ejected, its kinetic energy will be $E_k=h\nu-\phi$, where ϕ is the binding energy of the electron in the given shell. This process most often occurs with K-, L-, and M-shell electrons and in the higher-Z emitters.

One may define the <u>internal conversion coefficient</u> α_i as the ratio of the number of electrons from the i^{th} shell to the number of unconverted γ rays. Then, α_K is number of K electrons/number of unconverted γ rays. Thus, the total coefficient α is given by

$$\alpha = \sum_i \alpha_i \qquad 3.34$$

In the decay scheme literature, the conversion coefficient may be presented in a number of ways:

$$\alpha_K = \frac{\text{no. of K-shell } e^-}{\text{no. of unconverted } \gamma} = \frac{e_K}{\gamma} = \frac{K}{\gamma} \qquad 3.35$$

In addition, sometimes conversion coefficients are given as a ratio, i.e. $\frac{K}{L+M} = 4$.

Internal conversion is important in the measurement of activity. Serious error can result if one does not take this process into account. Suppose a γ counter has a measured efficiency of 0.1 cts/dis assuming 1 γ per dis. Consider a source of $1\text{x}10^8$ Bq, in which $\alpha_K=1.6$ and

$\frac{\alpha_K}{\alpha_L+\alpha_M}=2$. The number of disintegrations will be given by:

$$K + L + M + \gamma = N.$$

Remembering, $\frac{\alpha_K}{\alpha_L+\alpha_M} = \frac{K}{L+M} = 2$; $L + M= \frac{K}{2} =$ and $\frac{K}{\gamma} = 1.6$

then, by substituting

$$1.6\ \gamma + 0.8\ \gamma + \gamma = N$$

$$3.4\ \gamma = N$$

$\frac{\gamma}{N} = \frac{1}{3.4} = .294\sim0.3$ is the fraction of disintegrations

which actually yield a γ. So that, for 1×10^8 Bq, the γ/s will be $3\text{x}10^7$. This will result in $3\text{x}10^6$ cts/s, since only 1 out of 10 photons are counted. However, without paying attention to the decay scheme, $3\text{x}10^6$ would be interpreted as $\frac{3\text{x}10^6}{0.1} = 3\text{x}10^7$ dis/s, which is only 30% of the actual source strength.

3. Compton Effect

In the case of the Compton effect (see Figure 3.13) a γ ray ($h\nu_0$) impinges upon an electron and gives up only part of its energy. The result is that a photon of lesser energy ($h\nu$) is scattered at an angle θ with the initial direction of the photon. The electron is scattered at an angle ϕ with the initial direction of the γ ray. This process occurs in such a manner that both energy and momentum are conserved. The electron has a kinetic energy equal to the difference in energy between the incident and scattered photon. The electron will lose this energy by ionization of the atoms in the substance.

The change in wavelength of the photon in a Compton process is given by

$$(\lambda_1 - \lambda_0) = \Delta\lambda = 0.0242\ (1-\cos\theta) \qquad 3.36$$

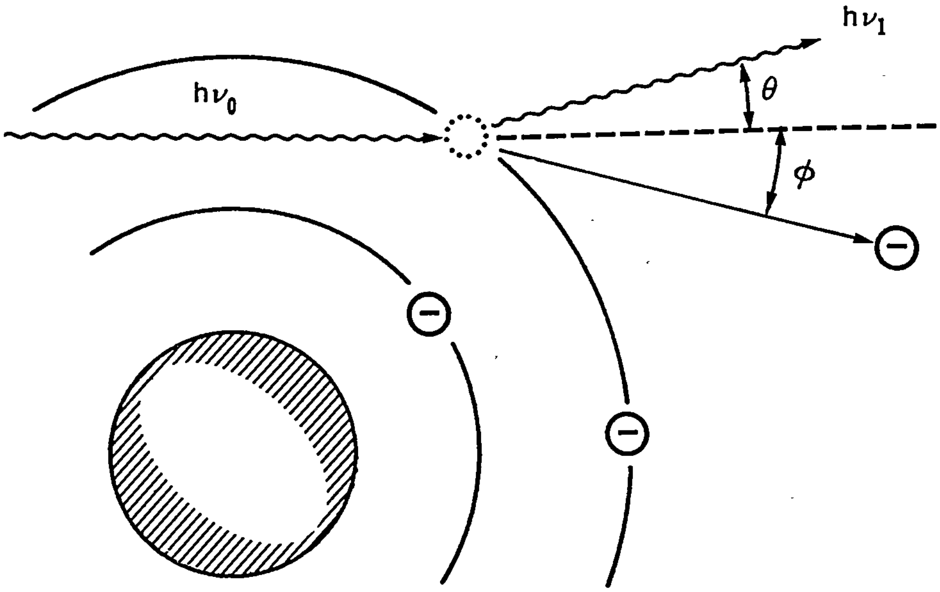


Figure 3.13 The Compton effect. (R.E.Lapp / H.L.Andrews, NUCLEAR RADIATION PHYSICS, 2/e, 1954, p. 117. Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, NJ)

(in Angstroms), where λ_0 and λ_1 are the wavelengths of the initial and scattered photons, respectively. From this, one can see that the change in wavelength of the scattered radiation does <u>not</u> depend upon the energy of the incident photon.

The maximum energy which an electron can acquire in a Compton process is attained when the photon is backscattered (θ = 180°). In this case, the energy of the electron is given by

$$E_e = \frac{E_\gamma^2}{E+\frac{m_oc^2}{(1-\cos\theta)}} = \frac{E_\gamma^2}{E_\gamma+\frac{0.511}{2}} = \frac{E_\gamma^2}{E_\gamma+0.256} \qquad 3.37$$

where E_γ is the initial energy of the photon in MeV. The energy E_1 of the scattered photon is given by

$$E_1 = \frac{E_\gamma}{1 + 1.96\, E_\gamma(1 - \cos\theta)} \qquad 3.38$$

The Compton effect is important for γ ray energies between 200 keV and 5 MeV in most light elements. The Compton effect decreases with increasing γ energy, but not as quickly as the photoelectric effect. The Compton process depends upon the number of electrons in the substance. Compton scattering will then be proportional to Z/A. The process is predominant in substances of intermediate Z.

For the gamma rays of interest to health physicists, the predominant interaction is the Compton effect. This presents a problem in shield design since the scattered photons are not truly removed or absorbed from a beam of radiation. In a wide beam of radiation and a thick shield, some gamma rays, which are at first scattered out of a beam, may later be scattered back into the beam. This leads to a greater amount of radiation reaching a point of interest than one would expect. For this reason, a correction must often be made to take into account this buildup effect.

4. Pair Production

The process in which a photon of sufficient energy gives up all its energy and forms two particles, an electron and a positron, is called pair production. This process is pictured (see Figure 3.14) as occurring near the nucleus of an atom, for it is only in this way that momentum can be conserved. The minimum energy needed for pair production is given by $E = 2m_0c^2$, where m_0 is the rest mass of an electron or positron. Since the rest mass of an electron is equivalent to 0.511 MeV, the photon must have an energy ≥ 1.022 MeV for pair production to occur.

When this process does occur, energy of the photon beyond 1.022 MeV is imparted as kinetic energy to the electron-positron pair and a portion to the nucleus in order to conserve momentum. Both electron and positron lose kinetic energy through ionization of atoms in the substance. Eventually the positron interacts with an electron in the substance in a

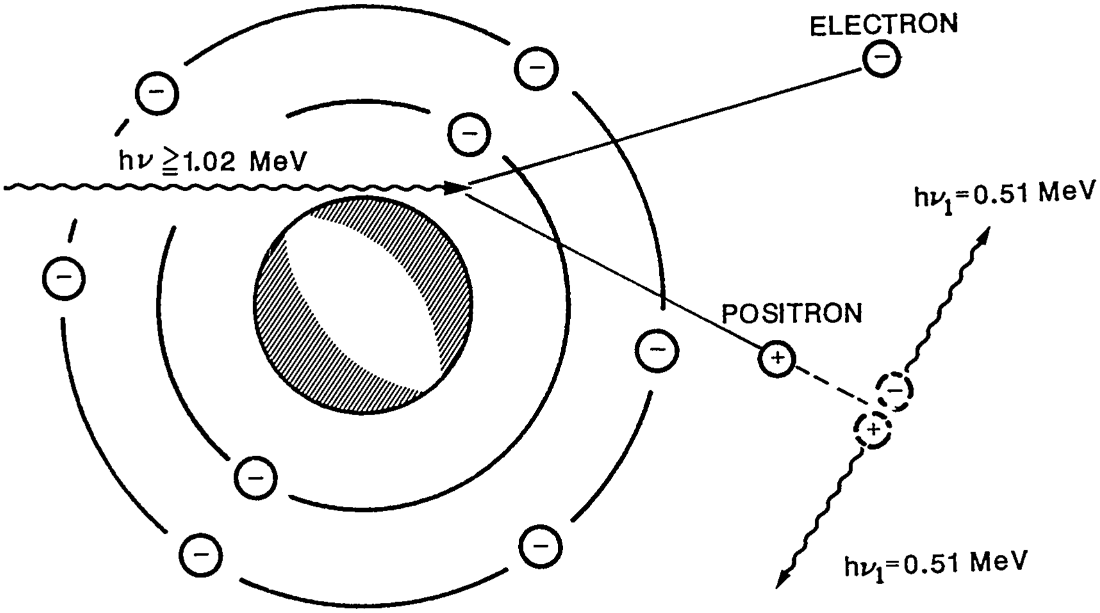


Figure 3.14 Pair production and annihilation. (R.E.Lapp / H.L.Andrews, NUCLEAR RADIATION PHYSICS, 2/e, 1954, p.120. Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, NJ)

process called annihilation. In this process, the mass of the particles is changed into two photons of 0.511 MeV each, emitted in nearly opposite directions. These may interact further in the substance through the photoelectric or Compton effects.

Since this process does not occur at all unless the γ energy is $\geq$1.022 MeV, pair production is important only for gammas of high energy. The process is also proportional to Z^2 of the absorber. Thus, pair production predominates for substances of high Z.

When the photon energy is very high, cascade reactions may take place. That is, a pair production occurs but the energy given to the pair is so large that more interactions with the nucleus are likely. The accompanying photon-electron multiplication creates a shower or cascade.

The three processes: photoelectric effect, Compton effect, and pair production, account for the main photon interactions with matter for $E < 10$ MeV. Both the photoelectric and Compton effects decrease with increase of γ energy. Pair production increases with γ energy. The result of these effects is that each substance will have minimum absorption at some γ energy. This is shown graphically in Figure 3.15, which illustrates the relative probability of the occurrence of the specific process (shown individually marked) and the total probability of a reaction occurring in iron.

In health physics, we are concerned with the absorption of radiation in matter so that we may employ procedures and protective barriers to protect people and equipment from `these rays. The three processes produce electrons which then lose energy when they ionize the absorbing matter. In a dense substance, the range of the electrons will be very short. The more interactions which occur in a given substance will determine how well one is absorbing the radiation. For this reason, high Z materials, which contain a large number of electrons, are the best shielding materials for photons.

5. Absorption of x and γ Rays

Suppose a narrow beam of monoenergetic photons is sent through a substance. Photon interactions may occur in any of the three ways

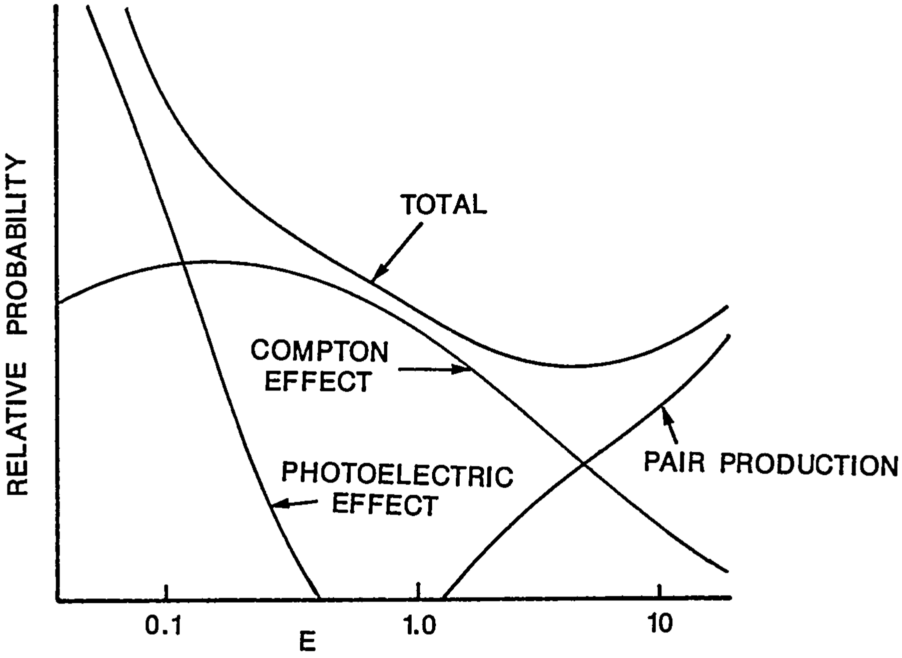


Figure 3.15 Relative probability of photon interactions in iron (Fe) versus energy E.

already mentioned. Energy will be removed from the beam by Compton scattering and by absorption, absorption occurring in the photoelectric effect and pair production. Since we assume a narrow beam, a Compton scattering process will be the equivalent of an absorption, that is, a scattered photon will be removed from the beam.

Let us measure the relative intensity of the beam as it passes through various thicknesses x of the absorber. A plot of the relative intensity versus absorber thickness on semilog paper will give a straight line (see Figure 3.16). As is the case for radioactive decay, x and γ ray absorption is an exponential process: there is a constant fractional decrease in intensity per unit thickness of the substance. The value of this constant, denoted by μ and called the total linear attenuation coefficient, is the slope of the straight line on the semilog plot.

For a narrow beam of monoenergetic photons, x and γ ray attenuation is described by an equation similar to equation 3.21,

$$I = I_0 e^{-\mu x}, \qquad 3.39$$

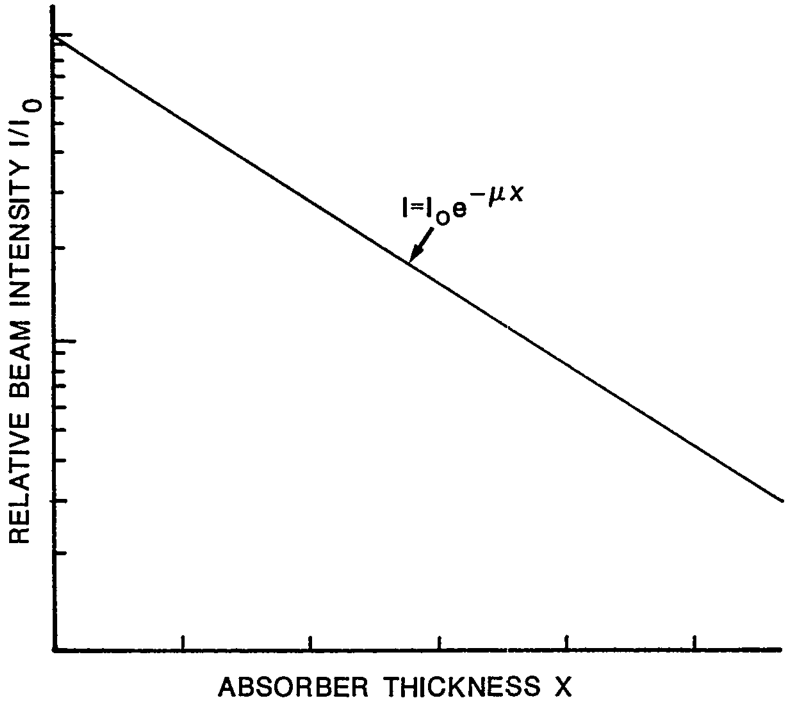


Figure 3.16 Absorption of X and γ rays. (R.E.Lapp / H.L.Andrews, NUCLEAR RADIATION PHYSICS, 2/e, 1954, p.109. Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, NJ)

where I_0 is the intensity at zero thickness, I is the intensity after passing through a thickness x (in m), and μ is the total linear attenuation coefficient (in m^{-1}).

The total linear attenuation coefficient μ represents the probability of photon interaction per unit path length. The quantity μ is the sum of the probability for each of the three processes, that is,

$$\mu = \tau + \sigma + \pi, \qquad 3.40$$

where τ is the linear attenuation coefficient for the photoelectric effect, σ is that for the Compton effect, and π is that for pair production. The coefficient μ is a function of the energy of the γ and the absorbing substance (see Figure 3.15). Thus, μ for a given material is a constant only for a given energy of x and γ radiation. One must consult attenuation coefficient tables or curves in order to find the value of μ for a given energy of x or γ rays in a certain substance. Such data can be found in the literature (References 15-17).

In these curves or tables, one will more often find the mass attenuation coefficient given in units of cm^2/g. These may be converted to m^2/kg by dividing the value in cm^2/g by 10.

The mass attenuation coefficient μ_m is simply the linear attenuation coefficient divided by the density ρ of the absorber:

$$\mu_m(\text{in } m^2/kg) = \mu/\rho, \qquad 3.41a$$

where ρ is in kg/m^3. Thus,

$$\mu = \mu_m\rho(m^{-1}). \qquad 3.41b$$

What is the linear attenuation coefficient for a 1-MeV gamma in lead? The density of lead is $1.134\times10^4 \frac{kg}{m^3}$ and μ_m=0.0046 m^2/kg:

$$\mu = \mu_m\rho = (0.0046)(1.134\times10^4)=52.2\ m^{-1}(.522\ cm^{-1}).$$

In the case of a compound substance, the coefficient for the compound can be found from the equation

$$\mu = \sum_{i=1}^{n} \left(\frac{\mu}{\rho}\right)_i \rho_i \qquad 3.42$$

where $(\mu/\rho)_i$ is the mass attenuation coefficient of the i^{th} element of the compound. The symbol ρ_i represents the density of the i^{th} element as it appears in the mixture. It is given by the product of the weight fraction of the i^{th} element and the density of the compound. The weight fraction of an element in a compound is found from:

$$(\text{Weight Fraction})_i = k_iA_i/A, \qquad 3.43$$

where k_i is the number of i^{th} atoms of atomic mass A_i, and A is the molecular weight of the compound.

For example, find the attenuation coefficient for 1-MeV gamma rays in H_2O. The mass attenuation coefficients for H and O are 0.0126 and 0.00635 m^2/kg, respectively. Then,

$$\mu=\sum_{i=1}^{2} \left(\frac{\mu}{\rho}\right)_i \rho_i = \left(\frac{\mu}{\rho}\right)_O \rho_O + \left(\frac{\mu}{\rho}\right)_H \rho_H$$

$$= 0.00635\frac{k_O A_O}{A_{H_2O}} \rho H_2O + 0.0126 \frac{k_H A_H}{A_{H_2O}} \rho H_2O$$

$$= 0.00635\frac{(1)(16)}{18}(10^3)+0.0126\frac{(2)(1)}{18}(10^3) = 7.04 \text{ m}^{-1}(.0704 \text{ cm}^{-1})$$

A graph of μ/ρ for selected materials can be found in Appendix E.

6. Half Value Layer - Tenth Value Layer

A quantity of usefulness in quick shielding estimates is the half value layer (HVL). This is the thickness of a substance which reduces the intensity of a beam of radiation to one-half of its initial value. This concept is similar to the half life in radioactivity. The half value layer is found from the equation

$$x_{1/2} = 0.693/\mu. \qquad 3.44$$

Closely allied with the concept of HVL is the tenth value layer or TVL. This is the thickness which reduces the initial intensity to one-tenth its value. It is related to the HVL by:

$$x_{1/10} = \frac{\ln 10}{\mu} = \frac{\ln 10}{\ln 2} x_{1/2} = 3.32\ x_{1/2} \qquad 3.45$$

Given the half value layer, one can quickly estimate the attenuation factor (AF) for a shield of thickness x, from

$$AF = 2^n \qquad 3.46$$

in which $n = x/x_{1/2}$ is the number of HVLs. For example, the HVL in lead for a 1 MeV γ ray is 0.0133 m (1.33 cm), find the attenuation factor for a lead thickness of 0.15 m:

$$n = \frac{.15\ \text{m}}{.0133\text{m}} = 11.28$$

$$AF = 2^{11.28} = 2487.$$

Similarly, if the TVL is known, then the attenuation factor is

$$AF = 10^n, \qquad 3.47$$

in which $n = x/x_{1/10}$.

The concepts HVL and TVL are useful for conditions in which scattered radiation may be ignored. If the shield is not too thick and/or the photon beam is narrow, significant scattered radiation may not reach the point of interest. For these cases, the total attenuation coefficient μ may be used to obtain an estimate of the HVL or TVL. For thick shields and/or wide photon beams (Figure 3.17), scattered radiation may increase the number of photons at the point over that estimated by use of the HVL or TVL.

7. Mean Free Path

The mean free path λ is the average distance which a photon, of initial energy E, travels before interacting. The mean free path (mfp) is equal to the reciprocal of the attenuation coefficient:

$$\lambda = 1/\mu. \qquad 3.48$$

If the thickness of an absorber is equal to one mean free path, then the intensity of the γ beam will be reduced by a factor of e. For a narrow beam of γ rays, the mean free path is sometimes called the relaxation length. But note that the mfp is a property of the initial photon of energy, E. When a photon interacts, its energy changes and so will the attenuation coefficient in that medium. So, when scattered radiation is important, one may make a distinction between the mean free path and the relaxation length. The mfp refers to the thickness of absorber which reduces the <u>uncollided</u> fluence by a factor of e. The <u>uncollided</u> fluence is a term used to signify those photons which have <u>not</u> interacted in traveling the distance $\lambda = \frac{1}{\mu}(\mathrm{m})$. Whereas, the relaxation length gives the thickness in which the total beam (uncollided plus scattered photons) is reduced by the factor e. In many situations, the relaxation length changes with penetration through the shield, as the energy spectrum of the photon beam changes.[18] Eventually, the relaxation length reaches a constant value which then can be used to predict attenuation in the rest of the absorber.

8. μ_{en}/ρ

Up to this point, all the concepts have been based upon a <u>narrow</u> beam of γ rays. In this case, if a γ is scattered out of the beam, it is assumed to be lost from the beam. The use of the total coefficient μ is thus valid. This coefficient is often called the narrow beam coefficient.

If one treats a <u>wide</u> beam of radiation (see Figure 3.17), then along a path well within the beam, some of the rays will be scattered away from this path. On the other hand, rays from other parts of the beam may be scattered into this path. Added to these may be photons produced by positron annihilation, fluorescence radiation, and bremsstrahlung formed elsewhere in the beam. For this complex case, the true attenuation will be less than that given by using μ in equation 3.39.

One approach to this problem is to use the mass energy absorption coefficient, μ_{en}/ρ, in place of μ. The attenuation of the beam is then given by

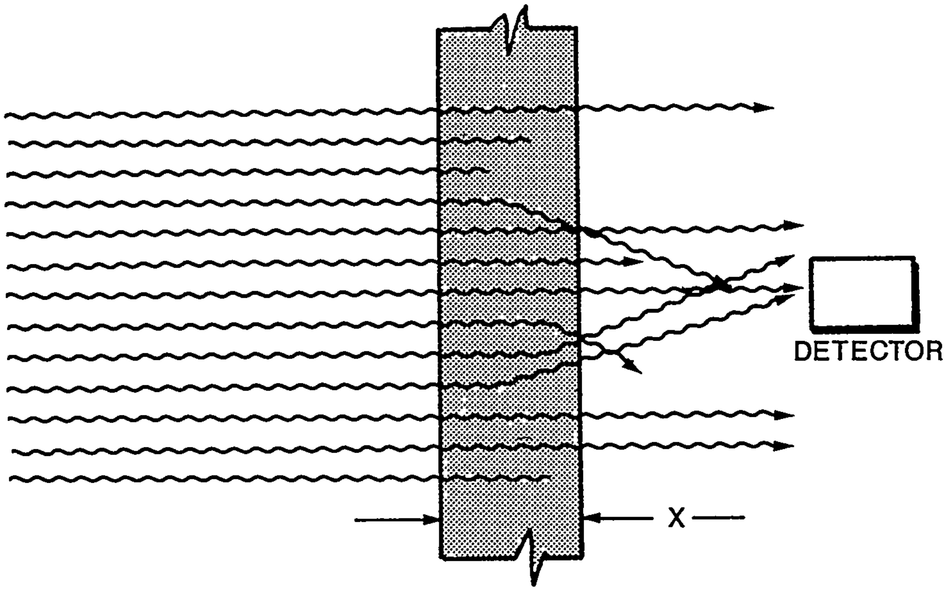


Figure 3.17 Wide beam of gamma rays in an absorber.

$$I=I_o e^{-\frac{\mu_{en}}{\rho}x'} \qquad 3.49$$

in which μ_{en}/ρ is in units of m^2/kg and x' in units of kg/m^2, to achieve a dimensionless quantity. The coefficient, μ_{en}/ρ, gives the probability of energy being <u>absorbed</u> as the photon interacts along the beam path. Corrections for the energy carried away by photons have been made to arrive at μ_{en}/ρ.

As a wide beam of photons passes through a substance to a detector, the attenuating processes result in some energy loss. The lower energy photons which survive interactions will be more likely to be absorbed in the next event. So, using μ_{en}/ρ based on the initial photon energy will lead to an overestimation of the fluence at the detector. To use μ/ρ leads to an underestimation, since some degraded photons will reach the detector. In health physics work, μ_{en}/ρ

is preferred since it will introduce a safety factor in calculations. A graph of μ_{en}/ρ for some selected absorbers can be found in Appendix E.

9. Buildup Factor

The buildup factor b is defined as the ratio of the actual gamma fluence to that which would be calculated by use of the narrow beam coefficient:

$$b = \frac{\text{true fluence}}{\text{calculated fluence}} = \frac{I}{I_o e^{-\mu x}} \qquad 3.50$$

or

$$I = I_o b e^{-\mu x}. \qquad 3.51$$

In this case one can calculate the true fluence I if the value of the buildup factor is known. The buildup factor corrects for the underestimation of scattered radiation reaching the detector. Tables of buildup factors are available in the literature.[17,19] The use of the buildup factor concept will be discussed in Section 8.

10. Relative Hazard

X and γ ray sources are significant as external hazards. The great range of photons in air means the hazard may be significant at large distances from the source of radiation. To reduce the hazard, extensive absorbing matter (shielding) is often required. Also, scattered photons may contribute to the hazard. Direct-source radiation may be absorbed to ensure no hazard in one direction, while scattered photons may cause a hazard to exist in adjacent areas. Any cracks or breaks in an otherwise adequate shield may lead to streaming or beams of radiation.

Because x and γ rays do penetrate to such a high degree, the damage to tissue will extend throughout the body as photons pass through it. The deeper and more radiosensitive tissues will be exposed in an external γ field. These tissues are more vital than skin, resulting in damage that affects the well-being of the body to a greater extent. This makes x and γ rays of greater concern as external hazards than either α or β particles.

From the standpoint of internal hazards, x and γ rays are not as significant as α or β particles. The longer tissue range of photons means less energy loss in a small tissue volume than for either α or β particles. Since photon energy loss occurs only at interaction sites, energy loss is not continuous along the photon path. In a small tissue mass, few interactions occur, since the path is small. For those that do occur, the density of energy deposition is similar to that for beta particles. Photons that do not interact in the organ carry away energy that is not deposited and therefore not effective in producing damage.

F. Neutrons

In previous sections, we have discussed the properties of the three types of natural radiation. These occur as the result of the natural decay of a nucleus. In this section, we will discuss the neutron. The neutron is found mainly as the result of nuclear reactions.

The work of Bothe and Becker showed that a very penetrating radiation was emitted when beryllium was bombarded with alphas from polonium. They assumed that the radiation consisted of gamma rays. Curie and Joliot showed that this radiation ejected protons from a sheet of paraffin. Chadwick applied the concepts of conservation of energy and momentum to show that the gamma ray assumption would not hold. He assumed that the radiation consisted of particles of zero charge and mass about equal to the proton, which he named neutrons. Thus, neutrons are emitted when beryllium absorbs an α according to the reaction:

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

This work by Chadwick in 1932 indicated that the neutron came from the nucleus. This view helped to form the present concept of a nucleus composed of protons and neutrons. Later work revealed that neutrons are emitted from almost any element when bombarded with high-energy particles. Also, neutrons are produced by cosmic ray bombardment of the earth's atmosphere.

Studies of the properties of the neutron confirmed that it is slightly larger than a proton in mass. The atomic number Z of a neutron is zero, since it has no net charge. Free neutrons are unstable; they decay by β emission (E_{max} = 0.782 MeV) with a half life $T_{½}$ = 10.5 minutes.

1. Sources of Neutrons

Since the discovery of the neutron, studies have shown that neutrons can be produced in a variety of ways. The use of radioactive sources and certain target substances may yield neutrons by either the (α,n) or (γ,n) reactions. Accelerators produce neutrons when high-energy particles strike suitable targets. Neutrons also result from the fission process in a reactor. Some of the transuranic elements can be used as neutron sources since they undergo spontaneous fission. Neutrons are also emitted during fusion reactions.

Some of the properties of these sources are:

a. (α,n) reactions - These sources are often prepared by making an intimate mixture of α emitter and finely divided powder of the target substance. ^{226}Ra, ^{210}Po, and ^{239}Pu are common α emitters used in the past. Recently, ^{241}Am and ^{238}Pu sources have been used. B, Be, Li, Na, and F are used as target materials. Be is most commonly used since it gives the highest yield. These sources emit neutrons which have a spectrum of energies, up to about 10 MeV. Source strength is most often expressed in terms of the activity of the α emitter.

Because of the high toxicity of Ra, Po, Pu, and Be, these sources must be sealed in metal containers. Ra-Be sources provide a strong neutron source but have a high gamma-ray background. Pu-Be sources have low γ activity, but give a lower yield than Ra-Be. Po-Be sources also have a low γ activity, but the ^{210}Po half life (138 days) is short. Radium sources have the added danger of radon production; since radon is a radioactive gas, one must be sure that the container is adequate to contain the pressure buildup.

The emission of neutrons is often anisotropic. This results from a loss of spherical symmetry in the source. Each source should be checked for anisotropy. This effect can have a magnitude of 10-30%.

The characteristics of some (α ,n) sources are given in the literature (References 1, 9, 17 and 20).

b. (γ,n) reactions - Some nuclei will emit neutrons when exposed to γ rays. Most sources do not emit gamma rays with energy greater than 4 MeV. The target substances are thus limited mainly to beryllium and deuterium which have low neutron binding energies (1.67 and 2.23 MeV, respectively).

The γ background is always high in these sources. A further drawback is the short half life of the γ emitters which are used. The emission rate for these sources is closely proportional to the amount of the γ emitter. These sources emit nearly monoenergetic neutrons, generally below about 1 MeV.

The characteristics of some (γ,n) sources are given in the literature (References 1, 9, 17, 20).

c. Accelerator sources - Nuclear reactions caused by high-speed charged particles impinging on a suitable target yield neutrons. Types of reactions used are: (α,n), (α,2n), (p,n), (γ,n), and (d,n).[6] Accelerator sources are useful for producing monoenergetic neutrons over a wide energy range. Since these machines produce very high-energy charged particles, many new target substances can be used. Proton and deuteron accelerators of 10 MeV can produce monoenergetic neutrons at all energies up to 27 MeV.[14]

d. Reactors - The fission process in reactors produces neutrons with a spectrum of energy, extending up to approximately 17 MeV.

A reactor provides a significant source of neutrons since, on the average, 2.5-3.0 n are emitted per thermal fission in presently known nuclear fuels.

e. Spontaneous fission sources - The neutron spectra from these sources are similar to that produced by fission in reactors. These sources are desirable as calibration sources since they have a fission-like spectrum. ^{252}Cf is at present the most notable of these types of sources. The characteristics of these sources are discussed in References 9, 13, and 20.

f. Fusion neutrons - When light elements are fused to form a heavier nucleus, neutrons are released. Two reactions of interest in fusion research are[9]

$$^{2}_{1}\mathrm{H} + {}^{2}_{1}\mathrm{H} \rightarrow {}^{3}_{2}\mathrm{He} + {}^{1}_{0}\mathrm{n}\ (2.45\ \mathrm{MeV})$$

$$^{2}_{1}\mathrm{H} + {}^{3}_{1}\mathrm{H} \rightarrow {}^{4}_{2}\mathrm{He} + {}^{1}_{0}\mathrm{n}\ (14.1\ \mathrm{MeV})$$

In these reactions, nearly monoenergetic neutrons of the indicated energies will be produced.

2. Neutron Energy

The types of interactions which neutrons undergo depend quite strongly on the energy of the neutron. As in the case of γ rays, certain effects predominate depending on the energy of the neutrons. It is often advantageous to treat neutron interactions in terms of the energy range in which they predominate. The National Council on Radiation Protection and Measurements (NCRP)[14] has classified neutrons according to the following energy scheme:

a. Thermal neutrons - When neutrons are slowed down so that they are in thermal equilibrium with matter, they have speeds comparable to gas molecules at room temperature. In this case, the most probable velocity is about 2.2×10^{3} m/s. The energy of thermal neutrons is then, from equation 1.7,

$$E = \frac{1}{2}mv^{2} = \frac{1}{2}(1)(1.66\times10^{-27})(2.2\times10^{3})^{2}$$

$$= 4.02\times10^{-21}\ \mathrm{J}(4.02\times10^{-14}\ \mathrm{erg}) = 0.025\ \mathrm{eV}.$$

One may use 0.5 eV as the upper limit of this energy range.

b. Intermediate neutrons - This energy region of neutrons is arbitrary, let us use 0.5 eV to 10 keV. Neutrons in this interval are often called "resonance neutrons."

c. Fast neutrons - Fast neutrons range in energy from approximately 10 keV to approximately 10 MeV. Again, this is an arbitrary designation but of general descriptive use.

d. Relativistic neutrons - This range includes neutrons of energy greater than 10 MeV. Although the NCRP does not use this designation, it is useful to describe reactions in this range.

3. Interactions With Matter

The study of neutron interactions with matter still forms a large part of the present experimental work in the field of physics. There are a number of processes which a neutron can enter into while passing through matter. The particular effect which occurs depends upon the properties of the substances and the energy of the neutron.

The binding energy of a neutron in any element is about 8 MeV, except for the lighter nuclei. This is the energy which holds a neutron in the nucleus. To remove the neutron one must supply at least this much energy. For a relativistic neutron, then, nuclear reactions are the most important interaction processes. For neutrons with energy ranging from 9 to 20 MeV, the (n,2n) reaction occurs. As the neutron energy increases, more complex processes, such as spallation, can occur. Spallation is a term used to describe a process in which a number of light fragments are emitted from an excited nucleus.

For <u>fast neutrons</u>, <u>elastic scattering</u> (n,n) is the predominant mode of interaction. Because the neutron has no charge, it does not interact with the electric field of a nucleus and can pass very close to charged particles. For this reason, one can view neutron interactions as collisions with nuclei. An elastic collision may be pictured as a billiard-ball type of collision in which energy and momentum will be conserved (Figure 3.18). If the scattering is isotropic, the energy transfer in the collision will not depend on the scattering angle. Isotropic scattering is a good assumption for most nuclides when the neutron energy is in the thermal to intermediate region. For hydrogen elastic scattering is isotropic up to about 14 MeV.

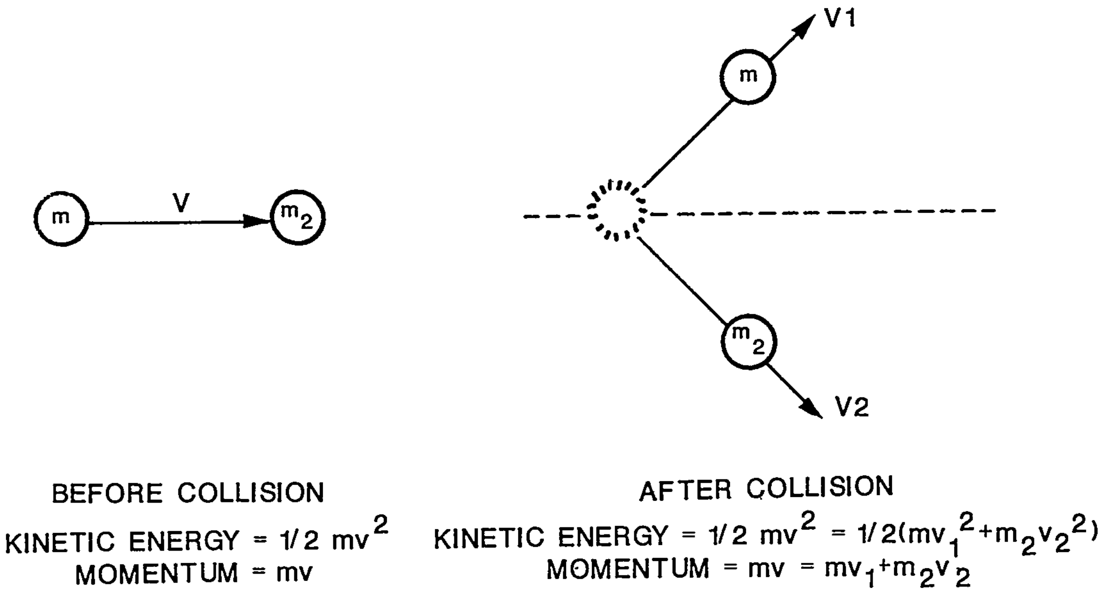


Figure 3.18 Elastic collision. Energy and momentum are conserved.

The loss of energy by a neutron in an elastic collision is greatest when the colliding particles have equal mass. Thus, a neutron will lose the most energy in a collision with a proton (hydrogen nucleus). The average fraction of energy lost is about the same for each collision.

The energy which is lost appears as the kinetic energy of the recoil nucleus. For a neutron beam of energy E_n, the <u>average</u>, over all interacting n, of the energy lost in a single collision with hydrogen is approximately En/2. As the mass of the nucleus becomes very large, the neutron will lose hardly any energy in elastic collisions. For this reason, substances with high Z are poor for use in slowing down neutrons.

In the energy range above 0.5 MeV, <u>inelastic scattering</u> (n,n'γ) begins to occur. This process is pictured as a putty-ball type of collision (see Figure 3.19). In the process, the neutron raises the nucleus to an excited state and comes off with a loss in energy. The

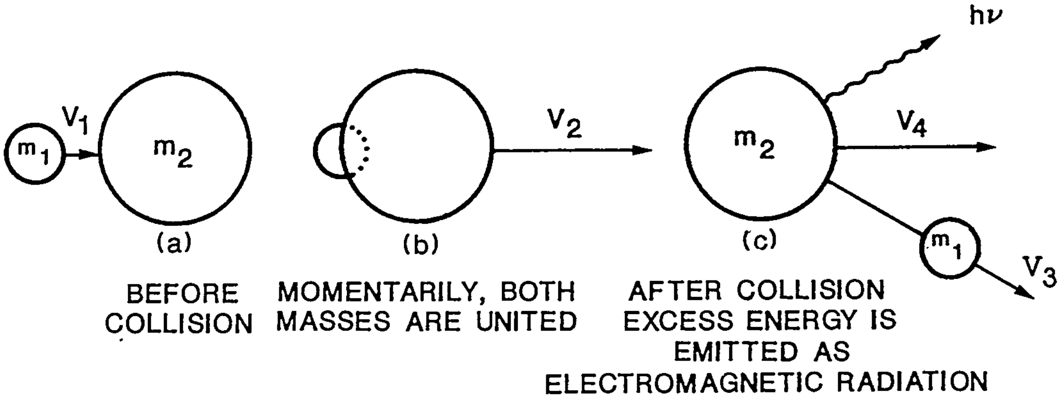


Figure 3.19 Inelastic collision.

excited nucleus often returns to the ground state by emitting the excess energy as a γ ray. For this process to occur, the neutron must supply the excitation energy to the nucleus. In heavy elements, the first excitation state is about 100 keV above the ground state. For light elements, this state may be at 3 or 4 MeV. This process, then, is more important for heavier nuclei and neutrons of higher energy (> 1 MeV).

Near the upper end of the fast neutron range, inelastic scattering as well as nuclear reactions begin to occur as frequently as elastic scattering.

The process of <u>elastic scattering</u> is still dominant for <u>intermediate</u> neutrons. The phenomenon of resonance absorption also occurs in this region. When the neutron energy, combined with the energy of the absorbing nucleus, produces an energy state nearly equal to a nuclear energy level of the product nucleus, the probability of absorption increases greatly. Thus the substance will absorb more neutrons at this energy than at an energy either slightly higher or lower. When the neutron energy becomes less than about 100 eV, capture becomes an important process.

As the energy of the <u>neutron</u> is further decreased, it approaches the thermal neutron range. The dominant process for <u>thermal neutrons</u> is capture, in which the neutron combines with the absorbing nucleus. The compound nucleus thus formed must then get rid of excess energy, usually by emission of γ rays. This is called radiative capture or an (n,γ) reaction. For some light nuclei, the (n,p) reaction can occur. In this case, the capture of a neutron leads to the emission of a proton. For slow-neutron capture in B and Li, the (n,α) reaction occurs. The capture of thermal neutrons in certain heavy nuclei such as uranium and plutonium may lead to fission.

Ionization which is produced as the result of neutron interaction with matter is almost totally secondary in nature. The neutron gives up energy to a recoil nucleus (p or heavier nuclei) which causes ionization in the medium. In the case of capture or nuclear reactions, ionization occurs due to the emission of charged particles (n,p) from the excited nucleus. If an (n,γ) capture takes place, the resulting ionization will depend upon the γ interacting in the medium. The loss of energy by neutrons, as well as by γ rays, is not a continuous process as it is for alpha and beta particles. The neutron, or the γ ray, will move through matter with little interference until an interaction takes place. When this occurs, then energy will be lost. So, neutrons are also called indirectly ionizing radiation.

4. Cross Sections

As a beam of neutrons moves through matter, certain interactions occur. From the preceding section we know that the nature of the substance and the energy of the neutron will make certain processes more likely to occur. In discussing neutron interactions, the term cross section, denoted by σ, is used to express the probability that a neutron will interact with a given substance. For any given process, if the probability is high, the cross section will be large. The cross section is expressed by means of a unit called the barn (b), equal to 10^{-28} m^2. The barn expresses the probability in terms of an area. In a sense, then, the atom may be viewed as presenting an effective target area to a neutron. If a neutron passes through this area, the reaction occurs.

The cross section σ is called the microscopic cross section since it expresses the probability per atom. Each possible interaction has its own probability. The total microscopic cross section, σ_t, is the sum of the separate cross sections for all processes which may occur:

$$\sigma_t = \sigma_{scatter} + \sigma_{capture} + \sigma_{fission} + \ldots \qquad 3.52$$

One should not construe that the cross section is simply a measure of the geometrical area of the nucleus. The effective target area which a nucleus presents to a neutron is often much greater than the impact area. For example, the radii of a proton and a neutron are about the same: approximately 1.3×10^{-15} m. For an impact to just occur, the particle centers cannot be more than $2.6\text{x}10^{-15}$ m apart (see Figure 3.20). The impact area would then be a circle of area $\pi(2.6\text{x}10^{-15})^2 = 2.12\text{x}10^{-29}$ m². But the total cross section in hydrogen for 0.01 eV neutrons is 48 b, or $480\text{x}10^{-29}$ m². It is clear that the effective area is much greater than the impact area. One can see from this that σ_t is much more than simply a geometric area. Curves for σ_t in a number of materials can be found in the literature (References 17, 21, 22).

In dealing with neutron penetration through a substance, one may wish to express the probability of a reaction in terms of the thickness of the substance. In such a case, the probability per unit path length, called the <u>macroscopic cross section</u> Σ is desired. The total macroscopic cross section Σ_t is related to the total microscopic cross section σ_t by the equation:

$$\Sigma_t = N\sigma_t \quad \text{m}^{-1} \tag{3.53}$$

where N is the number of atoms per cubic meter of the substance ($=\rho Na/A$).

Let us picture a cubic volume of a substance in terms of a cross section area of 10^{-4} m² and a thickness 10^{-2} m. Each atom of the substance has a cross section σ_t. The total reaction cross section will be the sum of the contributions from each atom. But N is the total number of atoms. Thus, $N\sigma_t$ is the total reaction cross section in the 10^{-2} m.

The macroscopic cross section Σ for neutrons is similar to the attenuation coefficient μ for gammas, since both quantities express the probability per unit path length. Sometimes the older symbol Σ is replaced by μ. By analogy, the mean free path λ for neutrons is given by

$$\lambda = 1/\Sigma_t. \tag{3.54}$$

In this case, λ gives the average distance a neutron of a given initial energy travels before interacting in a substance.

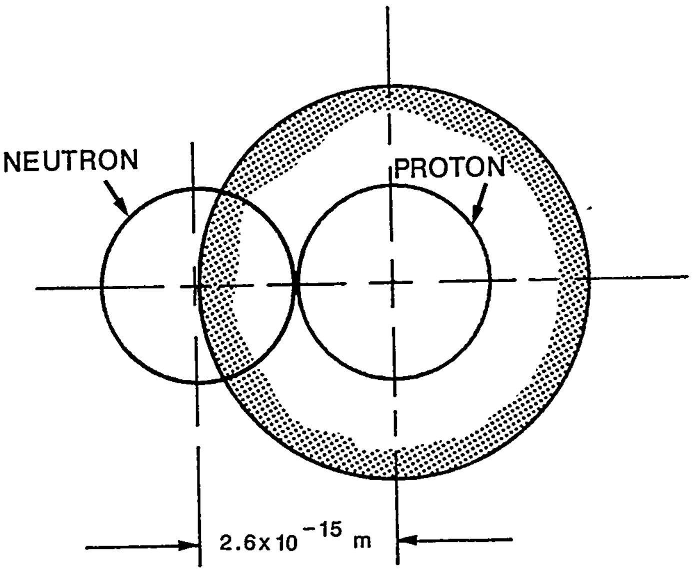


Figure 3.20 Impact area.

As was the case with photon radiation, one must interpret the meaning of the mean free path carefully. We have already seen that when several energies are present, or if a spectrum of energies are present, the concept of mean free path must be modified. Since most practical neutron sources will not be monoenergetic, the mfp, determined from $1/\Sigma_t$ will probably underestimate the neutron fluence reaching a point. This happens because the mfp value is based on one given initial energy of the n beam, not a spectrum.

5. Neutron Absorption

Assume a narrow beam of neutrons of a certain energy E, is passing through a substance. Let k be the number of neutrons per cubic meter in the beam. Assume further, that each neutron has a speed v. Then the neutron fluence rate ϕ is given by

$$\phi = kv \text{ (in n/m}^2\text{ s)}. \qquad 3.55$$

The fluence rate gives the number of neutrons passing through a sphere of cross sectional area of one square meter per unit time. In terms of a picture of 1 cubic meter of a substance, the probability of a reaction in passing through 1 m of the substance is given by Σ_t. The number of reactions per unit time in this meter of the substance will be

$$\frac{\text{number of reactions}}{\text{m}^3\ \text{s}} = \phi\Sigma_t. \qquad 3.56$$

The decrease in the fluence rate as the beam moves through a thickness Δx will be

$$\Delta\phi = -\ \phi\Sigma_t\Delta x. \qquad 3.57a$$

In this case, we assume that if a neutron is scattered, it is removed from the beam. This may be written as a differential equation in which

$$\frac{\Delta\phi}{\Delta x} \sim \frac{d\phi}{dx} \text{ and}$$

$$\frac{d\phi}{dx} = -\Sigma_t\phi. \qquad 3.57b$$

This form is the well known expression for an exponential relationship, as seen before. When integrated, this gives

$$\phi = \phi_o e^{-\Sigma_t x}, \qquad 3.58$$

where ϕ_o is the initial fluence rate of neutrons. If this equation is plotted on semilog paper, one will get a straight line whose slope will be the value of Σ_t, similar to the case for radioactive decay. In recent times, common usage has been to use the symbol μ for the attenuation coefficient, whether for γ or n.

As is the case with γ rays, this type of relationship is valid only if one can consider the scattered neutrons are being removed from the beam. For a narrow beam, this condition may be assumed for a monoenergetic n beam. For a wide beam, some of the monoenergetic neutrons scattered out of the beam will be replaced by neutrons scattered into the beam. The use of the total cross section Σ_t may thus lead to an overestimation of the effectiveness of an absorber. Also, since most n sources are not monoenergetic, even for narrow beam conditions, the relationship will not hold. That is, a spectrum of neutron energies will exist and the attenuation will not adequately be given by a single value of $\Sigma_t(\mu_t)$.

6. Removal Cross Section $(\frac{\mu_R}{\rho})$

Because of the large amount of energy lost by a fast neutron in a collision with hydrogen, this process in effect removes the neutron from the beam. One can then view neutron attenuation in terms of a removal concept. The removal cross section μ_R is then used to estimate neutron attenuation. The removal cross section can be viewed as giving the probability of large angle scattering (both elastic and inelastic), which would tend to remove n.

The mass macroscopic removal cross section is given by[22]

$$\frac{\mu_R}{\rho} = 0.0206\ A^{-1/3} Z^{-.294} \frac{m^2}{kg}. \qquad 3.59$$

Generally, the removal cross section is about 2/3 of the average value of the total attenuation coefficient for E between 6-8 MeV.

EXAMPLE: Compute the removal cross section for the element iron (Z=26, A=55.847).

$$\frac{\mu_R}{\rho} = 0.0206(55.847)^{-1/3}(26)^{-.294} = 0.0021\ \frac{m^2}{kg}(.021\ \frac{cm^2}{g})$$

Note that $\frac{cm^2}{g}$ divided by 10 equals $\frac{m^2}{kg}$.

The concept of removal cross section is based upon the presence of hydrogen in the absorber. It can also be applied to other substances backed up by hydrogen absorbers (provided there be at least 60 kg/m^2 of H). The concept is valid for fast neutrons in the range 2-12 MeV. In that energy range, neutrons which suffer inelastic collisions in a heavy substance will lose a large part of their energy. In that case, they may then be captured by hydrogen. If they suffer an elastic collision with a heavy nucleus, they lose hardly any energy, but they may be scattered over a large angle. However, an elastic collision with hydrogen causes a large transfer of energy. In these cases the neutron will also be effectively removed.

Without the presence of hydrogen in the absorber or backing up the absorber, the removal concept does not apply. In the neutron energy range treated, elastic or inelastic scattering with heavy nuclei will occur. Again, hardly any energy is lost in an elastic collision. When an inelastic collision occurs, energy will be lost. As the energy of the neutrons drops below about 1 MeV, inelastic scattering will become less probable and the neutrons will still have too much energy to be captured. In this case, the substance is transparent to the neutrons, which stream through. Since these neutrons have not been removed from the beam, the use of the removal cross section will lead to poor results for the calculated attenuation.

Keeping the above discussion in mind, we may rewrite equation 3.57 to obtain

$$\phi = \phi_o e^{-\left(\frac{\mu_R}{\rho}\right) x} \qquad 3.60$$

where ϕ_o is the fluence rate in the hydrogenous material, $({}^{\mu}R/\rho)$ is the removal cross section, and the thickness x must now be expressed in kg/m^2. One further restriction applies to the use of the removal cross section. The region of validity is limited to

$(^{\mu}R/\rho)\ x \leq 5.$[9] A plot of removal cross section values (in cm^2/g) versus the atomic mass can be found in Reference 14. See Section 8.E.1 for shielding approaches used for neutrons.

The removal cross section for a compound can be calculated if one knows the removal cross sections for each of the constituent elements. The removal cross section of substances mixed together is assumed to be additive, so that

$$\mu_R \text{ (compound)} = \sum_{i=1}^{n} (^{\mu}R/\rho)_i \rho_i \qquad 3.61$$

where $(^{\mu}R/\rho)_i$ is the value obtained from equation 3.59 for the i^{th} element, and ρ_i is the density of the i^{th} element as it appears in the compound (see equations 3.42 and 3.43 for evaluation of ρ_i).

7. Neutron Activation

One of the processes by which a neutron is attenuated is absorption by a nucleus. For many substances, this process produces a compound nucleus which is unstable. The substance then becomes radioactive. Many artificially radioactive substances are produced when other substances are exposed to the neutron fluence in a reactor. Absorption occurs mainly for thermal neutrons. For this reason, most artificial radionuclides have been produced in reactors. When a nucleus absorbs a neutron, the term activation is used to imply that the product becomes radioactive. The thermal activation cross section, σ_{act}, expresses the probability that a certain substance will be activated if it absorbs a thermal neutron. However, activation by neutrons of energy greater than thermal energy also occurs.

Assume a sample containing N atoms is exposed to a thermal neutron fluence rate ϕ. If σ_{act} is the probability of activation per atom, the total probability is $N\sigma_{act}$. The rate of formation of radioactive atoms is then formation rate = $\phi\sigma_{act}N$. 3.62a

During the time that radioactive atoms are being formed, some of the atoms will decay. The decay rate of the radioactive atoms N^1 will be given by

$$\text{decay rate} = \lambda N^1, \tag{3.62b}$$

where N^1 is the number of radioactive atoms present and λ is the transformation constant. The net rate of growth is then

rate of growth = formation rate - decay rate.

The corresponding differential equation is

$$\frac{dN^1}{dt} = \phi\sigma_{act}N - \lambda N^1. \tag{3.63}$$

here N^1 gives the number of radioactive atoms present after an irradiation time t. The activity A_t (in dis/s) of the sample at the time t that the irradiation is stopped is expressed by

$$A_t = \lambda N^1 = \phi\sigma_{act}N(1 - e^{-\lambda t}). \tag{3.64}$$

The activity at time t_1 after a sample is removed, is given by:

$$\begin{aligned} A &= A_t e^{-\lambda t_1} \\ &= \phi\sigma_{act}N(1 - e^{-\lambda t})e^{-\lambda t_1} \end{aligned} \tag{3.65}$$

The activity A_t of the sample in the neutron field will increase until the formation rate is equal to the decay rate. At this point, no further increase in activity will occur. For a given sample, the maximum value that the activity will reach is $\phi\sigma_{act}N$, which is called the saturation activity A_s. The term $(1 - e^{-\lambda t})$ varies from 0 for irradiation time $t = 0$ to 1 for $t = \infty$. For any given irradiation time t, this term gives the fraction of the saturation activity A_s which the irradiated sample attains.

The equation is valid if all target atoms are in the same neutron field. In effect, this means that the samples must be thin. Also, the relationship assumes that the number of target atoms remains about constant. This means that there should be no significant decrease in the number of target atoms during the irradiation.

For thick samples, the neutron field is reduced by the prescence of a neutron absorber. This loss is called the flux depression. In addition, the fluence rate at the inner central portion of the sample will be reduced due to absorption of neutrons by the outer layers of atoms. This is referred to as the self-shielding effect. Both of these perturbing effects need to be accounted for in thermal neutron absorption. As the energy of the neutron increases, these perturbation effects become less severe and the corrections are reduced in magnitude.

Let us work an example, to illustrate the use of equation 3.65. A 10^{-3} kg tungsten sample is exposed for three days at an average thermal-neutron fluence rate of 10^{15} n/m^{2}s in a reactor. The cross section σ_{act} is 40 b and the half life of W-187 is 24h. What is the activity of the sample 12 h after irradiation? Also, determine A_s, the saturation activity.

$$A = \phi\sigma_{act}N(1 - e^{-\lambda t}).$$

$$N = (m/A)N_a = (10^{-3}/0.186)6.022\text{x}10^{23};\ t = 3\text{d},\ t_1 = 0.5\text{d}$$

$$\sigma_{act} = 40\ \text{b} = 40\text{x}10^{-28}\ \text{m}^2.$$

$$\lambda = 0.693/1\ \text{day} = 0.693\ \text{d}^{-1}$$

$$\phi = 10^{15}\ \text{n/m}^2\text{s}.$$

Therefore:

$$A = 10^{15}(40\text{x}10^{-28})10^{-3}\frac{6.022\text{x}10^{23}}{0.186}(1 - e^{-0.693(3)})e^{-0.693(0.5)}$$

$$= 8.01\text{x}10^{9}\ \text{Bq}\ (0.216\ \text{Ci})$$

and

$$A_s = \phi\sigma_{act}N = 10^{15}(40\times10^{-28})10^{-3}\ \frac{(6.022\times10^{23})}{0.186}$$

$$= 1.295 \times 10^{10}\ \text{Bq}\ (0.35\ \text{Ci}).$$

8. Relative Hazard

Neutron sources are significant as external hazards. Since the neutron moves rather freely through air and other matter, the neutron field may still be hazardous far from the source. Absorbing material can reduce the hazard, but the required amount is often large. Scattered neutrons produced in the shielding material can also create concern in adjacent areas. In addition, radiative capture and inelastic scattering in the shield may result in enough gamma radiation to cause concern.

As neutrons travel through the body, tissue interactions extend throughout the body. The deeper, more radiosensitive, tissues are exposed to the external neutron field. The amount of damage done is related to the neutron energy. Fast neutrons give up about 80-95% of their energy in elastic scattering processes with hydrogen in tissue. Intermediate and thermal neutrons lose their energy mainly in the absorption reactions: $^{1}H(n,\gamma)^{2}H$ and $^{14}N(n,p)^{14}C$.[14] For a large mass of tissue, the $^{1}H(n,\gamma)$ process is more important than the $^{14}N(n,p)^{14}C$ process. For a small mass, the reverse is true. Since a photon has a much greater range in tissue than a proton, the damage in a small organ will be greater for a recoil proton than for a γ ray. For a large organ, the γ dose could be greater

Since neutron interactions produce recoil particles, protons, and γ rays, the transfer of neutron energy to tissue will vary. The energy deposition from protons and recoil nuclei is more dense than that of electrons. This means that neutrons should be more effective in producing tissue damage than γ rays. Depending upon the neutron energy, neutrons are 4-20 times more effective. On this basis, neutrons are more hazardous than γ rays, although both are significant external hazards.

Neutron sources are not normally considered as an internal hazard. The lack of natural sources that emit neutrons and the physical properties of neutron sources make the chance of a neutron source being fixed in the body quite remote. This view could change if substances with high spontaneous fission rates become more readily available in larger quantities.

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QUESTIONS

3.1 Name the four main types of radiation which must be dealt with in health physics. What two radiations may be considered as a single type?

3.2 What term is used to indicate that the energies of α particles emitted from a given radioisotope are almost identical?

3.3 Although scattering of α particles is infrequent, at what point along their path is scattering most likely to occur?

3.4 What is the energy range of most α particles emitted by naturally occurring radioactive atoms?

3.5 By what two processes do α particles lose their energies? Describe the processes.

3.6 What value is defined by the average energy required to create an ion pair in a given substance? What is that value for air?

3.7 What happens to an α particle whose energy is entirely lost in a substance?

3.8 Why does an α particle produce a high number of ion pairs per unit path length?

3.9 Why should an increased ionization occur for a heavy charged particle near the end of its path?

3.10 What term is used to indicate the loss of energy by a particle per unit of path length as it passes through a specific material?

3.11 What term is used to indicate the linear stopping power of a substance divided by the density of that substance?

3.12 What term is used to indicate the ratio of the stopping powers of two substances divided by the density of that substance?

3.13 What term indicates the distance that a radiation particle will travel in matter?

3.14 The following symbols are used in formulas relating to the distance that an α particle will travel in a medium:

E_α, W, R, ip, ze, S, ρ, $(S/\rho)_m$, $(B/A)_m$, R_a, ρ_a, $(B/A)_a$, ρm

Make up a table to indicate the meaning of these terms.

3.15 Explain the relative hazard of α particles as an external hazard and as an internal hazard.

3.16 How is the energy of β particles from a particular radioisotope different from the energy of α particles?

3.17 What high speed particle is identical to β^-?

3.18 Name three ways in which β^- particles lose their energy in passing through matter.

3.19 Why is the actual path travelled by an electron or β particle usually much greater than the straight-through path?

3.20 With what part of an atom does a β particle or an electron interact to produce bremsstrahlung?

3.21 For a particular energy of β particle or electron, how does the Z or atomic number of an absorbing material affect the production of bremsstrahlung?

3.22 Explain why the stopping power for β particles or electrons in materials decreases as the Z number increases yet their relative stopping ability is essentially a function of their density.

3.23 What does the constant fractional decrease in the ionization per unit of thickness by β particles or electrons define?

3.24 Explain the relative hazard of β particles as an external hazard and as an internal hazard.

3.25 How is the velocity of a wave related to its wavelength?

3.26 How is the period of a wave related to the frequency?

3.27 Name some familiar electromagnetic radiations. Which of these constitute a hazard because of their ionizing power?

3.28 What term is given to a single quantum of an electromagnetic radiation?

3.29 What is Planck's constant? What are its units?

3.30 What is the name of the statement relating the radiation intensity at a specific distance from an isotropic source?

3.31 What does the term "isotropic" mean?

3.32 What are the two conditions under which inverse-square law is valid?

3.33 Identify the sources of

a) the continuous x ray spectrum, and
b) the characteristic x ray

3.34 Upon what does the highest energy x ray produced by an x ray tube depend?

3.35 What part of the atom is responsible for the production of γ rays?

3.36 Name the three common interactions with matter that result when x rays or γ rays are absorbed.

3.37 Which electromagnetic interactions with matter take place when:

a) all of the radiation energy is given to the work function ϕ and to the kinetic energy of an ejected orbital electron?

b) The radiation energy is given to the production and kinetic energy of an electron and a positron?

c) the radiation energy is given partially to the ejection of an orbital electron and the remaining energy continues as a lower energy photon?

3.38 What are Auger electrons? When are they produced?

3.39 What is internal conversion? Why is it important in the measurement of activity?

3.40 What is the minimum energy needed for pair production? How is it related to the rest mass of an electron?

3.41 Name the constant fractional decrease of x ray or γ ray intensity per unit thickness of an absorbing substance.

3.42 What value is obtained by dividing the linear attenuation coefficient by the density ρ of the absorber?

3.43 What term is used to indicate the average distance which a photon travels before interacting?

3.44 What corrections must be made to the total attenuation coefficient when computing the absorption of a wide beam of x rays or γ rays? Why?

3.45 Explain why x rays and γ rays constitute a greater external than internal hazard.

3.46 What is half value layer? What is tenth value layer?

3.47 What is buildup factor?

3.48 What are the principal sources of neutrons?

3.49 How is the source strength of an (α,n) neutron source frequently rated? What are the hazards associated with these sources?

3.50 Explain the term anisotropy in relation to an (α,n) neutron source.

3.51 What type of neutron sources will produce

a) a highly monoenergetic neutron?
b) a large supply of a spectrum of neutrons?
c) neutrons most conveniently and economically for a small laboratory?

3.52 Make a chart indicating:

a) neutron terminology based upon neutron energy and
b) interactions of neutrons with matter.

3.53 Why are high Z materials poor for slowing down neutrons?

3.54 What is resonance absorption?

3.55 What is meant by radiative capture of neutrons? At what energies, is it predominant?

3.56 Explain why ionization produced as the result of neutron interaction in matter is secondary.

3.57 What term is defined by the probability that a neutron will interact with a nucleus in terms of an area?

3.58 What is:

a) the name of the unit, and
b) the size of the unit that would be used for the definition in the preceding item?

3.59 What term is defined by the probability that a neutron will interact with a particular material in terms of the thickness of the substance?

3.60 Name the three principal cross sections for neutron interaction.

3.61 What does the reciprocal of the macroscopic cross section Σ_t define?

3.62 What does the number of neutrons per unit time passing through a 1 m^2 cross section of area define?

3.63 What error occurs in calculating neutron absorption in the case of a broad neutron beam that could lead to an overestimation of the effectiveness of an absorber?

3.64 What unit for neutron attenuation is based upon the large amount of energy lost by fast neutrons colliding with hydrogen?

3.65 What is the approximate relationship of neutron attenuation versus atomic number Z for fast neutrons?

3.66 What term is used to indicate that when a nucleus absorbs a thermal neutron that the product becomes radioactive?

3.67 What two rates will determine the net rate of growth of the number of radioactive atoms in neutron activation?

3.68 Discuss the difference between the activity of an activation sample A_t and the term indicated by the symbol A_s.

3.69 Of what is the damage that neutrons do to the human body a function of?

3.70 By what process do recoil protons lose their energies?

3.71 Explain the following reactions:

a) $H(n,\gamma)D$ b) ${}^{14}_{7}N(n,p){}^{14}_{6}C$

3.72 In the reactions given above, which is most important for:

a) small organs? b) large organs?

Why?

3.73 Why are neutrons not considered as an internal hazard?

3.74 What radiation may accompany neutrons and must be considered as an added hazard?

3.75 Why are neutrons more hazardous than γ rays?

PROBLEMS

3.1 An α particle emitted by polonium-214, ^{214}Po, has a range of 0.069 m in air and an energy of 7.68 MeV. If the W value for α particles in air is 35 eV/ip, what is the average ion pairs/m formed?

Answer: 3.18×10^6 ip/m

3.2 The mass stopping power of a 50 MeV proton in lead (density: 1.13×10^4 kg/m^3) is given as 5.67 MeV cm^2/g.

a) convert it to SI units and
b) what is the linear stopping power?

Answer:
a) 9.072×10^{-14} Jm^2/kg
b) 1.03×10^{-9} J/m.

3.3 a) Find the range in air of the 5.15 MeV α particle emitted by plutonium-239, $^{239}_{94}Pu$.

b) Using the "Rule of Thumb" equation, find the range.

c) What is the % error between the two values?

Answer:
a) 3.72×10^{-2} m
b) 3.94×10^{-2} m
c) ~ 6%

3.4 Find the range in air of the 3.16 MeV α particle emitted by platinum-190, $^{190}_{78}Pt$.

Answer: 1.77×10^{-2} m

3.5 Find the range of a 6 MeV α particle in aluminum, atomic weight 26.98 and density of 2.7×10^3 kg/m^3.

Answer: 2.88×10^{-5} m

3.6 Find the range of a 5 MeV α particle in common water, H_2O. Use 1000 kg/m^3 for density and the atomic weights of 1 and 16 respectively for hydrogen and oxygen.

Answer: 3.41×10^{-5} m.

3.7 The mass stopping power of 1.32 MeV β particles emitted from potassium-40, ^{40}K is given as 0.168 MeV m^2/kg in air. The density of air is 1.29 kg/m^3. If the W value for betas in air is 33.85 eV/ip, how many ion pairs per meter are formed?

Answer: 6.4×10^3 ip/m

3.8 What is the range of a 1.32 MeV β^- in the preceding problem?

Answer: 5.81 kg/m^2 or 4.5 m of air

3.9 What percent of energy of the above β^- is converted to bremsstrahlung if the absorber is

a) Al (Z=13) and
b) Pb (Z=82)?

Answer: a) 0.57% b) 3.6%

3.10 What percent of energy of an electron beam of 5 MeV is converted to x rays if it is absorbed by lead (Z=82)?

Answer: 28.7%

3.11 The collision mass stopping power of 50 MeV electrons in lead (density 1.13×10^4 kg/m^3, Z=82) is 0.138 MeV m^2/kg and the total mass stopping power is 0.83 MeV m^2/kg.

a) Find the ratio of radiative energy loss (bremsstrahlung) to collision loss.

b) What % of electron energy is lost in the bremsstrahlung production?

c) If one uses the approximation EZ/700 for the ratio of radiative to collision loss, what would be the %.

Answer: a) 5.01 b) 83.4%
c) 85.4%

3.12 Find the range of a manganese-56, $^{56}_{25}Mn$, β particle that has a maximum energy of 2.86 MeV.

Answer: 14.1 kg/m^2

3.13 The original intensity of a beam of β particles is $3x10^4$ cpm. Upon passing through a foil of 0.25 kg/m², the intensity is $1x10^4$ cpm. Find the energy of the β particle. Hint: First, find the mass attenuation coefficient.

Answer: 0.435 MeV.

3.14 Find a) the wavelength of an 1150 kHz radio signal, and

b) the frequency of an x ray whose wavelength is $2x10^{-12}$ m. The velocity of electromagnetic radiation in the atmosphere is $3x10^8$ m/s.

Answer: a) $2.61x10^2$ m
b) $1.5x10^{20}$ Hz.

3.15 Find the energy of the radio signal and the x ray in the preceding problem, in MeV ($h = 6.626x10^{-34}$ Js).

Answers: a) $4.75x10^{-15}$ MeV
b) 0.62 MeV

3.16 If the wavelength region of visible light is between 400 nm and 700 nm, what is the corresponding photon energy region in eV?

Answer: 3.1 eV and 1.8 eV

3.17 Find the wavelength and frequency of a 0.662 MeV γ ray from $^{137}_{55}Cs$.

Answer: Frequency: $1.6x10^{20}$ Hz
Wavelength: $1.88x10^{-12}$ m

3.18 The γ radiation intensity, 0.06 m from an isotropic point source, is $5x10^{11}$ MeV/m² s. What is the intensity at 0.4 m distance?

Answer: $1.125x10^{10}$ MeV/m² s

3.19 Find the power of the isotropic source in problem 3.18.

Answer: $2.26x10^{10}$ MeV/s

3.20 A point source of ^{137}Cs emits $2x10^4$ MeV/s.

a) Find the intensity at a distance of 1 m from the source.

b) At what distance will the intensity be reduced to half the initial intensity?

c) At what distance will intensity drop by a factor of 10?

Answers: a) $1.59\text{x}10^3$ MeV/m^2 s
b) 1.414 m
c) 3.162 m

3.21 What is the shortest wavelength of x ray (m) that can be produced by an electron accelerated by 60,000 volts?

Answer: $2.07\text{x}10^{-11}$ m

3.22 If the photoelectric work function is 4 eV, what is the energy of the emitted photoelectron when the photon energy was 206 eV?

Answer: 202 eV

3.23 A 2 MeV photon causes the emission of a Compton electron and scatters at an angle of 90° from its path.

a) What is the wavelength of the scattered photon?
b) What is the energy of the Compton electron?
c) What is the energy of the scattered photon?

Answer: a) $3.04\text{x}10^{-12}$ m
b) 1.593 MeV
c) 0.407 MeV

3.24 In problem 3.23, if the Compton electron is found to have an energy of 1.0 MeV, in what direction is the scattered photon emitted?

Answer: ~ 42°.

3.25 Find the linear attenuation coefficient when the intensity of a narrow gamma beam is reduced to ¼ of its original intensity upon passing through 0.05 m of a substance.

Answer: 27.7 m^{-1}

3.26 Find the mass attenuation coefficient, μ_m, when the linear attenuation coefficient is 50/m and the density of the substance is $4.1\text{x}10^3$ kg/m^3.

Answer: $1.22\text{x}10^{-2}$ m^2/kg

3.27 Find: a) half value layer and
b) tenth value layer

when the linear attenuation coefficient is 50 m^{-1}

Answers: a) $1.386\text{x}10^{-2}$ m
b) $4.6\text{x}10^{-2}$ m

3.28 Find the mean free path λ when the linear attenuation factor is 15 m^{-1}.

Answer: 6.67×10^{-2}

3.29 What is the buildup factor when the true reading is 6×10^4 cpm and the calculated one is 5×10^4 cpm?

Answer: 1.2

3.30 Assuming a buildup factor of 1.2, a linear attenuation coefficient of 20 m^{-1} and a thickness of material as 0.1 m, find the emerging beam fluence rate when the original fluence rate is 9×10^8 γ/m^2 s.

Answer: 1.46×10^8 γ/m^2 s

3.31 Find the kinetic energy in electron volts of a neutron moving at 10,000 m/s. Assume the mass of a neutron to be 1.6747×10^{-27} kg.

Answer: 0.523 eV

3.32 Find the speed of a 100 MeV neutron. Hint: Use relativistic equations.

Answer: 1.283×10^8 m/s

3.33 Find the macroscopic cross section Σ_{act} for natural magnesium, atomic weight 24.3, microscopic cross section for activation σ_{act} 0.063 barns, and density 1741 kg/m³. Hint: use same form used for finding Σ_t.

Answer: 0.272 m^{-1}

3.34 If the macroscopic cross section of a material is 80 m^{-1} and the thickness of the material is 2×10^{-3} m, what fluence rate will result when the original fluence rate is 5×10^{12} n/m² s?

Answer: 4.26×10^{12} n/m² s

3.35 A 1×10^{-3} kg carbon sample is bombarded with a 50 MeV proton beam for 6 hours. What activity is to be expected from ^{11}C at the end of irradiation? The reaction cross section is 24 mb and the irradiating flux is 10^{12} particles/m² s? The half life of ^{11}C is 20.4 m.

Answer: 1.2×10^5 Bq

3.36 A thin gold foil weighing $1x10^{-5}$ kg was irradiated 6 days by a flux of $9x10^9$ n/m^2 s. What was its activity 3 days after irradiation? Data for Gold: atomic weight 197, activation cross section 98.8 barns, and $T_{\frac{1}{2}}$ = 2.7 days.

Answer: $9.89x10^2$ Bq or $5.934x10^4$ dis/min.

SECTION 4 - CONCEPTS OF RADIATION QUANTITIES AND APPROACHES TO DOSE DETERMINATIONS

A. Definition of Terms

The organization which selects and defines the quantities and units of radiation is the International Commission on Radiological Units and Measurements (ICRU). This group, formed in 1925, gathers the most recent data, evaluates these data, and then issues its recommendations. Reports which are particularly relevant to our present discussion are listed in the References (Refs. 1-6). Other ICRU reports containing useful information are listed in the Bibliography. Much of the information found in this section is discussed in more detail in ICRU Report 33 (Reference 1).

A number of useful terms are used in health physics work to describe radiation quantities. The definitions of value to this discussion are:

1. Mean Energy Imparted, $d\bar{\epsilon}$

The result of radiation interactions is that energy is imparted to matter. This occurs either by charged ionizing radiation (α, β) or by uncharged ionizing radiation (x,γ,n). The mean energy imparted, $d\bar{\epsilon}$, equals the sum of the energies (excluding rest energy) of all ionizing radiation which enter a given volume, minus the sum of all the energies (not rest energy) of all ionizing radiation leaving the volume, plus the sum of all rest mass energy changes due to nuclear transformations. Imparted energy means energy retained in matter and that is the important quantity. This energy may appear as ionization, excitation, heat or changes in chemical bonds.

2. Absorbed Dose

The absorbed dose, D, is the quotient of $d\bar{\epsilon}$ by dm, where $d\bar{\epsilon}$ is the mean energy imparted by ionizing radiation to matter of mass dm.

$$D = d\bar{\epsilon}/dm \qquad 4.1$$

The SI unit for absorbed dose is Jkg^{-1} and the absorbed dose rate ($\dot{D}$ or $\frac{dD}{dt} = Jkg^{-1}s^{-1}(\frac{J}{kgs})$. The special name for the unit of absorbed dose is the gray (Gy): 1 Gy = 1 Jkg^{-1}. The previously used unit was the rad. This unit is no longer recommended, but still can be found in the literature. For ease in converting units, one may use 1 Gy=100 rad. Also, since often the absorbed dose rate will be important: 1 Gy/s = $3.6x10^5$ rad/h. For a discussion of the rad and other historical units, see Reference 7. For a summary of the SI units and their relationship to former units, consult Appendix B.

We may note these factors about the absorbed dose:

a. the absorbed dose is the quantity, the gray (Gy) is the special unit;

b. the quantity refers to all ionizing radiation;

c. the specification of D in <u>any</u> medium is appropriate;

d. the absorbed dose is a measure of the energy imparted, not the ionizing ability of the radiation type;

e. the absorbed dose is the physical quantity most suitable for designating dose, since the imparted energy more closely relates to the types of biological damage produced, although the degree of the effect is related to other factors.

3. Kerma

The kerma, K, is the quotient of dE_{tr} by dm, where dE_{tr} is the sum of the initial kinetic energies of all the charged ionizing particles liberated by uncharged ionizing particles in a material of mass dm,

$$K = \frac{dE_{tr}}{dm}. \qquad 4.2$$

The unit for kerma is the same as that used for absorbed dose: Jkg^{-1}. The kerma rate ($\dot{K}$ or $\frac{dK}{dt}$) is $Jkg^{-1}s^{-1}$. Again, the older unit rad for the kerma and rad/h for the kerma rate will still be found in the literature. For kerma: 1Gy=100 rad and for kerma rate, one may use the conversion: $1\ Gy/s = 1 \frac{J}{kgs} = 3.6x10^5$ rad/h. The term kerma was based on kinetic energy released in material. This concept was proposed[7] to adequately explain the two-step process involved in n and γ energy absorption.

The following points should be noted about kerma:

a. The concept refers to liberation of charged particles by uncharged ionizing particles, so that kerma is a property of n and γ;

b. only the charged particles which originate in the mass element are part of the kerma;

c. the kerma is the first step in photon or neutron energy absorption--namely, energy transfer;

d. the concept refers to transfer of E to the medium, not to the absorption in the medium, so the kerma is not directly measurable, although it may be calculated;

e. the definition mentions all charged particles, thus even Auger electrons are included;

f. kerma is the sum of the kinetic energy released so K includes any energy secondaries may later lose by collisions, by radiation production (bremsstrahlung) or annihilation.[8,9]

g. the second step, which is not the kerma, is the absorption of energy, if energy is carried away by the processes in f. above, then the kerma will not equal the absorbed dose.

If the medium of the mass element differs greatly from the surrounding medium, an appreciable disturbance of the field occurs. Thus, in these instances, attempted calculation of the kerma will lead to errors. It is possible, however, to refer to the kerma for specified material in free space or at a point inside some other substance. These situations are evaluated by assuming a small amount of the specified material is placed at the given point.

The use of the kerma allows one to conveniently describe a field of uncharged ionizing particles in any medium. For photons up to a few MeV, air is a suitable substance and tissue will suffice for all radiations.

4. Exposure

The exposure, X, is the quotient of dQ by dm, where dQ is the absolute value of the total charge of the ions of one sign produced in air when all the electrons (e^- and e^+), liberated by photons in air whose mass is dm, are completely stopped in air:

$$X = dQ/dm. \qquad 4.3$$

The SI unit of exposure is Ckg^{-1}, where C stands for coulomb. The unit of exposure rate ($\dot{X}$ or $\frac{dX}{dt}$) is $Ckg^{-1}s^{-1}$. There is no special name for the SI unit of exposure. The historical exposure unit, the roentgen (R), is no longer recommended. In terms of SI units, 1 R = 2.58×10^{-4} Ckg^{-1}. Certain features of the exposure concept that should be remembered are:

a. the exposure is the quantity, the SI unit of exposure is Ckg^{-1}.

b. the quantity refers only to photons, no other type of radiation;

c. air is the only defined medium;

d. the quantity is a measure of the ions--thus related to the ionizing ability of photons;

e. electrons and positrons are completely stopped--expend all their energy in air;

f. ionization arising from absorption of bremsstrahlung emitted by electrons is not included;

g. the exposure is difficult to measure when the photon energy is greater than a few MeV or below a few keV.

An alternate definition of exposure, given by the ICRU,[1] is

$$X = \Psi \left(\frac{\mu_{en}}{\rho}\right) \frac{e}{W}, \qquad 4.4$$

in which Ψ is the energy fluence (the energy of particles incident on a sphere of unit cross sectional area, Jm^{-2}), e is the electron charge in coulombs (1.6×10^{-19} C), W is the mean energy expended in air per ion pair formed (5.42×10^{-18} J) and $\frac{\mu_{en}}{\rho}$ is the mass energy absorption in air (m^2/kg).

5. Dose Equivalent

The dose equivalent, H, is the product of D, Q and N at the point of interest in tissue, where D is the absorbed dose, Q is the quality factor (see 4.B) and N is the product of all other modifying factors,

$$H = DQN. \qquad 4.5$$

In general, the value assigned in the past, for N, has always been 1; so in its report on quality factor, the ICRU[11] recommends that N now be dropped from the definition. The SI unit for the dose equivalent H is Jkg^{-1}. The special name for the unit of dose equivalent is the sievert (Sv); 1 Sv = 1 Jkg^{-1}. The SI unit is related to the previous dose equivalent unit, rem, by 1 Sv = 1 Jkg^{-1} = 100 rem.

The concept of dose equivalent was established for use in protection applications. Biological research has proven that equal absorbed doses of different types of radiation produce different degrees of biological effect. This indicated that the relative hazard depended on other factors, than simply the physical dose, D. It turns out, that the physical dose, D, when multiplied by certain suitable modifying factors, equates, in a numerical way, the effectiveness of different radiation types. It allows a common scale (dose equivalent) to be used for all ionizing radiation to compare the relative hazard of different types of radiation.

These points should be remembered:

a. the dose equivalent is not a physical quantity so it cannot be measured directly;

b. the defining equation implies that one can arrive at a different value of H based upon irradiation conditions;

c. the concept is useful for radiation protection work, not experiments, since it allows for a numerical scaling of absorbed doses.

B. Quality Factor

The quality factor (Q) is an assigned factor for radiation protection applications that denotes the modification of the effectiveness of a given absorbed dose.

The numerical value of Q is based partly on biological effects and partly on judgment.[11] The object of the assigned value Q is to attempt to equate the risks from exposure to different types of radiation. It is defined as a function of the collision stopping power (L_∞) in water at the point of interest.[12] Note, that L_∞ is equivalent to S_{col}, the linear collision stopping power (linear energy transfer).[1] The relationship between L_∞ and Q, as given in ICRP Publication 26,[12] is shown in Table 4.1. The International Commission on Radiological Protection (ICRP) is an advisory body which reviews and discusses basic protection principles and prepares reports recommending safe practices

Table 4.1 - L_∞-Q Relationship*

L_∞ (keV/μm of H_2O)	Q
$\leq$ 3.5	1
7	2
23	5
53	10
$\geq$175	20

*(Reprinted with permission from Annals of the ICRP 1, No. 3, ICRP Publication 26, Recommendations of the ICRP, Copyright 1977, Pergamon Press, Ltd.)

The values in the table apply to the linear energy transfer (LET) of a particle of initial energy E. When the particle moves through the medium, the particle energy is altered and the LET will change. Moreover, for an initial radiation field composed of a number of different energy components (or a spectrum of radiation), one must use an effective value of Q, which is averaged over all the components. This is called $\overline{Q}$. When the radiation spectrum is unknown, the approximate values of $\overline{Q}$ recommended by the ICRP[12] are shown in Table 4.2.

Table 4.2 - Recommended Q Values

(Reprinted with Permission from Annals of the ICRP 1, No. 3, ICRP Publication 26, Recommendations of the ICRP, Copyright 1977, Pergamon Press, Ltd.)

Radiation Type	Q
X rays, γ rays and electrons	1
Neutrons, protons and particles (Z=1, A>1) of unknown E	10*
α and particles with Z>1 of unknown energy	20
Thermal neutrons	2.3

*The ICRP has recommended an increase in the fast neutron quality factor by a factor of 2.[13] The new values for fast neutrons should be 20.

Let us examine some of the features of Q:

a. the quality factor, Q, is not experimentally determined, it is an <u>assigned</u> value based on dose modifying factors;

b. Q is for radiation protection applications--that is, for relatively low dose rates, not for use in radiobiology or experiments, and not for acute accidental exposures (> .25 Sv = 25 rem);[12]

c. the modification of effectiveness by LET--implies that one of the main factors determining the Q value is the consideration of the LET influence in biological changes;

d. the nature of the modification--the relationship of Q - LET is stated in terms of the L_∞, the stopping power in water, an indicates that high LET radiation is much more effective in producing damage.

C. Exposure

As a beam of photons passes through air, the interactions which take place produce electrons, which then lose energy by creating ion pairs. The exposure is measured by collecting these ion pairs. Thus, the exposure concept is based on the ability of photons to produce ionization.

There is very little attenuation of photons in air. What does occur is due mainly to the photoelectric effect, the Compton effect, and pair production. If a wide beam of photons is used, then well within this beam (that is, not near the edge) the field will be uniform. In this case, one may say that in each small mass, dm, of air, the effects should be the same, that is, about the same number and type of interactions should occur. Then, the number of electrons which are produced in these processes will be about equal.

In the consideration of a given mass of air then, the electrons which are produced will give up part of their energy outside this mass before being stopped. However, this loss of electrons may be compensated by electrons entering the given region from other parts of the field. If the field is uniform over a large enough volume, the compensation will be complete. In this case, a state of <u>charged particle equilibrium</u> is reached, that is, the energy imparted to a given small mass of air by all the electrons which traverse this region equals the energy imparted to the medium along the total path of all electrons produced in the mass by the photons.

The collection of the ions which are produced in a given small mass of air under equilibrium conditions is then a measure of the exposure. The exposure, X, is expressed in SI units as coulombs/kg. Since one electron has a charge of 1.6×10^{-19} C, one coulomb represents the collection of 6.25×10^{18} electrons or positrons.

D. Exposure Rate - Isotropic Point Source

Assume an isotropic point source of C Bq of a γ emitter of energy E at the point 0 (see Figure 4.1). Our interest is to obtain an expression for the exposure rate at the point A, a distance r from the

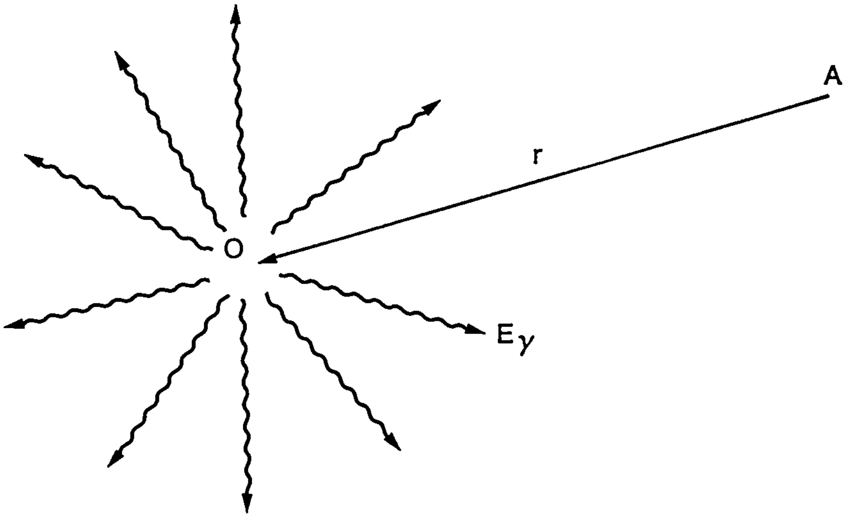

Figure 4.1 Isotropic point source.

source, in terms of C/kg per unit time. The fluence rate, ϕ_A, at the point A, is the number of photons per second (N_γ), incident on a sphere of unit cross sectional area, at distance r(m):

$$\phi_A = \frac{N_\gamma}{4\pi r^2}\left(\frac{ph}{m^2 s}\right). \qquad 4.6$$

The fluence rate is often called the flux density and, in the past, was expressed in units of ph/cm²s. Since the source is monoenergetic, the energy fluence rate ψ_A at the point A will be given by

$$\psi_A = \phi_A E = N_\gamma E_\gamma / 4\pi r^2 = S/4\ \pi r^2 \qquad 4.7$$

where S is the total energy per second emitted by the source. If S is expressed in J/s,

$$S = C(Bq)\left(\frac{1\ dis}{Bq}\right) n_\gamma \left(\frac{fraction\ of\ \gamma}{dis}\right) E_\gamma \left(\frac{MeV}{\gamma}\right)$$

$$x1.6x10^{-13}\ (J/MeV) \qquad 4.8$$

$$= 1.6x10^{-13}\ n_\gamma C E_\gamma$$

Then:

$$\psi_A = 1.6x10^{-13}\ n_\gamma C E_\gamma / 4\pi r^2$$

$$= 1.273x10^{-14}\ n_\gamma C E_\gamma / r^2 (J/m^2 s) \qquad 4.9$$

The exposure X is defined in terms of the ionization produced in a mass (dm) of air. The exposure rate is given by

$$\text{Exposure rate} = dX/dt \text{ or } \dot{X} \qquad 4.10$$

To find the ions produced in air, the energy given up by the beam must be known. The term $\dot{\psi}_A$ gives the energy delivered per unit time about the point A. Some of this energy will be given up in interactions which produce electrons. These electrons in turn produce many ion pairs. The mass energy absorption coefficient (μ_{en}/ρ) expresses the probability of energy being absorbed by the substance. The product

ψ_A (μ_{en}/ρ) gives the amount of energy absorbed per unit time per unit mass. If one assumes that all this energy is used in producing ion pairs, then, adapting equation 4.4,

$$\dot{X} = \frac{dX}{dt} = \psi_A\left(\frac{\mu_{en}}{\rho}\right)\frac{e}{W}\left(\frac{C}{kgs}\right) \qquad 4.11$$

where $W=5.42\times10^{-18}$ J/ip for electrons in air, μ_{en}/ρ (m^2/kg) is the absorption coefficient introduced in Section 3, and $\psi_A = d\Psi/dt$.[1]

This leads to:

$$\dot{X} = \frac{(1.273\times10^{-14})(1.6\times10^{-19})n_\gamma CE_\gamma}{5.42\times10^{-18}\ r^2}\left(\frac{\mu_{en}}{\rho}\right)$$

$$= 3.764\times10^{-16}\frac{n_\gamma CE_\gamma}{r^2}(C/kgs), \qquad 4.12$$

from an isotropic point source of a γ emitter of photon activity $n_\gamma C$ (in Bq), energy E_γ (in MeV), and mass energy absorption coefficient μ_{en}/ρ (m^2/kg) at a point a distance r(m) from the source.

For air, the attenuation is very small for short distances, so that one can neglect the attenuation factor. For large distances in air, or for the case of an absorber between the source and the point A, an attenuation factor of the form $e^{-(\mu_{en}/\rho)x}$ must be included. Such a factor is needed for each such absorber and in the general case is

$$e^{-\sum_{i=1}^{k}(\mu_{en}/\rho)_i x_i} = e^{-(\mu_{en}/\rho)_i x_i} \ldots e^{-(\mu_{en}/\rho)_k x_k}$$

where x_i is expressed in kg/m^2. Note the use of μ_{en}/ρ here, instead of μ/ρ, to compensate for any scattered radiation.

The general expression becomes:

Exposure rate (in C/kgs)

$$=3.764\text{x}10^{-16}\frac{n_\gamma CE_\gamma}{r^2}\left(\frac{\mu_{en}}{\rho}\right)e^{-\sum_{i=1}^{k}(\mu_{en}/\rho)_i x_i} \qquad 4.14$$

If the only substance of concern is air, then certain simple assumptions can be employed. For example, the attenuation in a few meters of air is negligible and no attenuation factor need be included. Also, μ_{en}/ρ can be taken as 0.00275 m^2/kg in the range $0.2<E\gamma<2$ MeV, so that the exposure rate is

$$\dot{X} \text{ or } \frac{dX}{dt}(\text{in C/kgs}) \sim 1.04\text{x}10^{-18}n_\gamma CE_\gamma, \qquad 4.15$$

at 1 m and $0.2<E_\gamma<2.0$ MeV. For $\dot{X}$ in C/kgh, multiply equation 4.15 by 3600.

Example - Compute the exposure rate in air at 1 m from a $3.7\text{x}10^{10}$ Bq (1 Ci) source of 1 MeV photons (n_γ = 1 ph/dis).

$$\dot{X}=1.04\text{x}10^{-18}\ n_\gamma CE_\gamma=1.04\text{x}10^{-18}(1)(3.7\text{x}10^{10})(1)$$
$$= 3.85\text{x}10^{-8}\ \frac{C}{kgs}$$

Now, if we multiply our answer by $\frac{3600\ s/h}{2.58\text{x}10^{-4}\ C/kgR}$, the result is

$$\dot{X} = 3.85\text{x}10^{-8}\ \frac{3600}{2.58\text{x}10^{-4}} = 0.537\ R/h.$$

Note, this is the way in which the approximation was formerly expressed- the R/h at 1 m in air from a γ source of C Ci~0.5 $n_\gamma CE_\gamma$. Let us use the inverse square law to compute the exposure rate at 1 ft (.3 m).

$$\dot{X}_1(1)^2 = \dot{X}_{0.3}(.3)^2$$

$$\dot{X}_{0.3} = \frac{3.85\text{x}10^{-8}\ (1)^2}{(.3)^2} = 4.28\text{x}10^{-7}\ \frac{\text{C}}{\text{kgs}}$$

Again, if we transform this as above, the result is

$$\dot{X}_{0.3} \sim 4.28\text{x}10^{-7}\ \frac{(3600)}{2.58\text{x}10^{-4}} = 5.97\ \text{R/h}.$$

This is the same as the other previously used approximation-the R/h at 1 ft in air from a gamma source of C Ci~6 $n_\gamma CE_\gamma$. The literature contains a number of listings which give what has been called the specific gamma ray constant, Γ. This is generally expressed as either $\frac{\text{R m}^2}{\text{hCi}}$, or $\frac{\text{R ft}^2}{\text{hCi}}$, depending upon the literature source. To

convert to SI units,

$$\frac{\text{R m}^2}{\text{h Ci}}\ (1.94\text{x}10^{-18}) = \frac{\text{C m}^2}{\text{kgs Bq}} \qquad 4.16$$

or

$$\frac{\text{R ft}^2}{\text{h Ci}}(1.8\text{x}10^{-19}) = \frac{\text{C m}^2}{\text{kgs Bq}} \qquad 4.17$$

As an example, the R/h at 1 ft from 1 Ci of ^{60}Co is given as 14.2. Find the C/kgs at 1 m from 1 Bq.

$$14.2(1.8\text{x}10^{-19}) = 2.55\text{x}10^{-18}\ \frac{\text{C}}{\text{kgs}}$$

The simple formula, equation 4.15, may often be used to advantage in health physics applications to estimate quickly the exposure rate. In cases for which more accuracy is desired, the general formula should be used and the actual value of $^{\mu}\text{en}/\rho$ for the γ obtained.

E. Exposure Rate - Fluence Rate

Along the same line, one may wish to express the exposure rate in terms of the fluence rate of photons in a beam of radiation. If one assumes ϕ photons/m^{2}s of energy E_γ in the beam, the resulting expression is

$$\text{Exposure rate:} \quad \dot{X} = 4.73\text{x}10^{-15}\phi E\gamma(\mu_{en}/\rho)\ \frac{C}{kgs} \qquad 4.18$$

where E_γ is in MeV and μ_{en}/ρ is in m^2/kg. In this case, no attenuating substance is included, and the exposure rate as given must be well within the beam dimensions to avoid edge effects. In Figure 4.2, the fluence rate, ϕ, and energy fluence rate, ψ, for an exposure of 1 C/kgs are shown versus photon energy in MeV. To estimate the fluence rate at a point in space, based on the previously used R/h unit, divide the reading (R/h) by 1.395x10^7 to convert to C/kgs. Multiply the results by the fluence rate conversion factor for the appropriate energy to obtain the fluence rate (ph/m^{2}s).

F. Exposure - Kerma in Air Relationship

The concept of kerma, which defines the energy transferred to matter can be related to the exposure, which defines the ionization produced in air. The kerma may also be determined for other media and can be related to the absorbed dose in a given medium. We may relate the mass energy absorption coefficient to the mass energy transfer coefficient by

$$\frac{\mu_{en}}{\rho} = \frac{\mu_{tr}}{\rho}(1-g) \qquad 4.19$$

in which g represents the fraction of energy carried away as bremsstrahlung. The fraction g is negligible except for high energy photons (g is

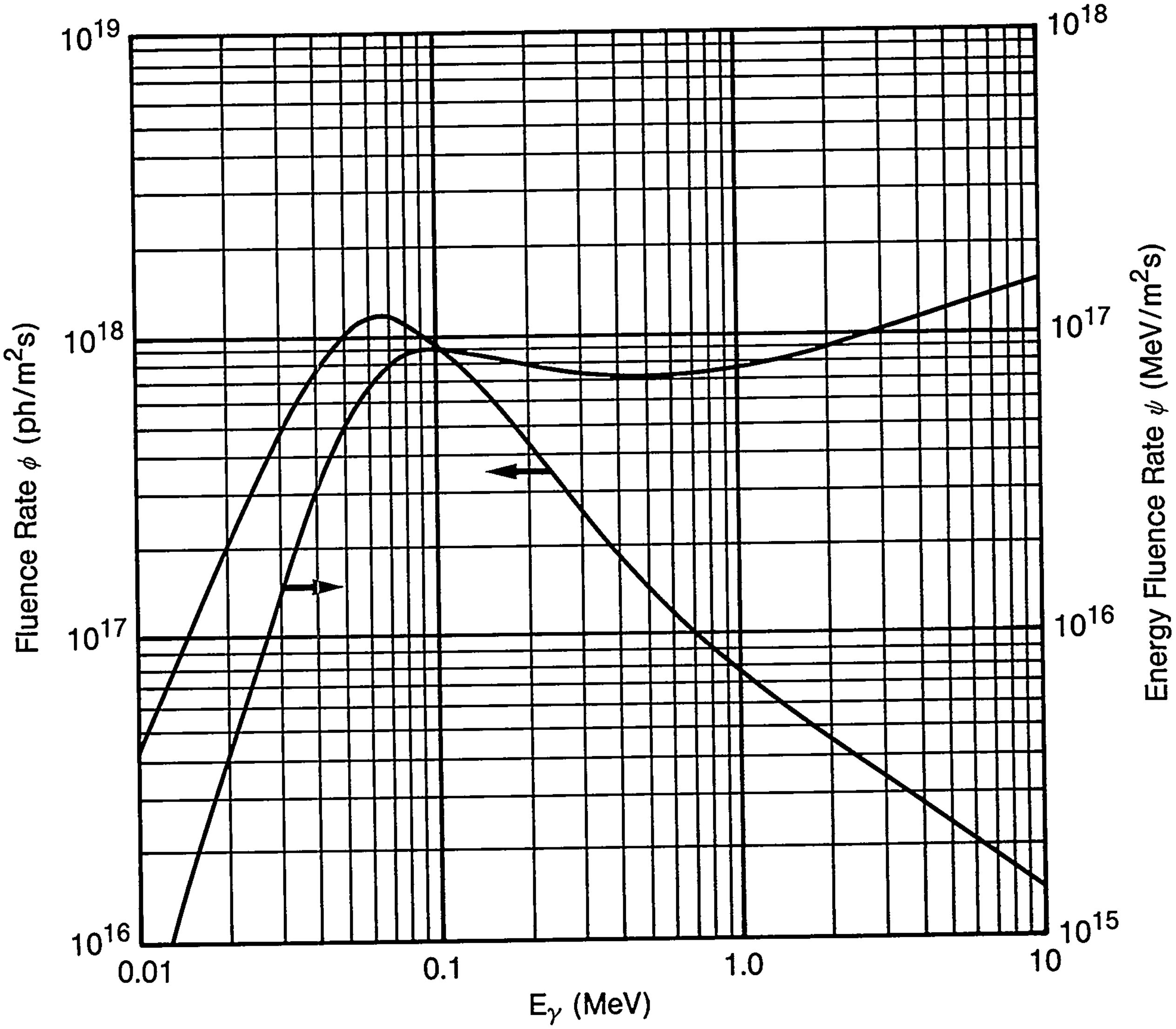

Figure 4.2 Fluence rate and energy fluence rate for an exposure rate of 1C/kgs. Mass energy absorption coefficients were obtained from Reference 15.

about 0.3% for e^- released in air by ^{60}Co radiation).[14] In many practical situations then, $\mu_{en}/\rho \sim \mu_{tr}/\rho$. The kerma rate in $\dot{K}_{air}$ for an isotropic point source (Figure 4.1) is

$$\dot{K}_{air} = \psi_A \left(\frac{\mu_{tr}}{\rho}\right)_{air} \qquad 4.20a$$

in which ψ_A is given by equation 4.9. Substituting for ψ_A in equation 4.11 gives

$$\dot{X} = \psi_A \left(\frac{\mu_{en}}{\rho}\right) \frac{e}{W} = \dot{K}_{air} \left(\frac{\mu_{en}/\rho}{\mu_{tr}/\rho}\right)_{air} \frac{e}{W} \qquad 4.20b$$

and

$$\dot{K}_{air} = \frac{\dot{X}W}{e}\left(\frac{\mu_{tr}/\rho}{\mu_{en}/\rho}\right)_{air} = \frac{\dot{X}W}{e(1-g)} = \frac{33.85\ \dot{X}}{1-g} \qquad 4.21$$

For the condition that g is negligible, $\mu_{tr}/\rho = \mu_{en}/\rho$ and the exposure rate $\dot{X}$ is the ionization equivalent of the kerma rate in air. We note that equation 4.21 is valid when charged particle equilibrium is established. Also, from the simple relationship of 4.21, the kerma rate is easily found from the exposure rate $\dot{X}$ for a given fluence rate ϕ. The kerma rate $\dot{K}$ is then simply 33.85 $\dot{X}$(Gy/s). Also, in air, the kerma rate equals the absorbed dose rate, under the same equilibrium conditions.

G. Absorbed Dose

The exposure concept is concerned with the production of ions in air. The concept of absorbed dose focuses attention on the energy imparted to matter in a mass element, so energy imparted always refers to a finite region. Since one can presume that the biological effects result from the energy imparted to matter, the concept of absorbed dose is more closely associated with these effects than is the exposure. Also, the absorbed dose is not specified for any given substance, so that the unit applies for any medium.

Whereas the exposure refers to photons only, the absorbed dose may be used for any ionizing radiation. Thus the concept can be used for α, β, γ, and neutrons.

H. Absorbed Dose Rate - Isotropic Point Source of Photons

Consider again an isotropic point gamma-emitting source. As we saw in Section 4.D above, $\psi_A(\mu_{en}/\rho)$ gives the amount of energy absorbed per unit time per unit mass. If this absorbed dose rate in air is given in J/kgs, we may write

$$\text{Absorbed dose rate: } \dot{D} \text{ or } \frac{dD}{dt} = \psi_A\left(\frac{\mu_{en}}{\rho}\right)\frac{J}{kgs} \qquad 4.22a$$

Using equation 4.9, we may express equation 4.22a as

$$\dot{D} = 1.273\times10^{-14}\frac{n_\gamma CE_\gamma}{r^2}\left(\frac{\mu_{en}}{\rho}\right)\frac{J}{kgs} \text{ or } \frac{Gy}{s} \qquad 4.22b$$

This gives the absorbed dose rate from an isotropic point source of a γ emitter of activity $n_\gamma C$ (Bq), energy E_γ (MeV), and mass energy absorption coefficient μ_{en}/ρ (m²/kg), at a point a distance r(m) from the source. The general expression for the absorbed dose rate in Gy/s is then, similar to equation 4.14,

$$\dot{D}=1.273\times10^{-14}\frac{n_\gamma CE_\gamma}{r^2}(\mu_{en}/\rho)e^{-}\sum_{i=1}^{k}(\mu_{en}/p)_i x_i \qquad 4.23$$

If the only substance of concern is air, then we may ignore attenuation, assume μ_{en}/ρ is approximately 0.00275 m^2/kg in the range $0.2 < E_\gamma < 2$ MeV, and the absorbed dose rate is:

$$\dot{D} = 3.50\text{x}10^{-17}\ n_\gamma C E_\gamma \frac{Gy}{s} \qquad 4.24$$

at 1 m in air for $0.2 < E_\gamma < 2.0$ MeV. For $\dot{D}$ in Gy/h, multiply equation 4.24 by 3600.

Returning to equation 4.20b, we may rewrite this in terms of

$$\psi_A\left(\frac{\mu_{en}}{\rho}\right)$$

to give

$$\psi_A\left(\frac{\mu_{en}}{\rho}\right) = \dot{X}\frac{W_{air}}{e} \qquad 4.25a$$

and from equation 4.22a, $\dot{D} = \psi_A\left(\frac{\mu_{en}}{\rho}\right)$ we arrive at

$$\dot{D} = \dot{X}\frac{W}{e} = 33.85\ \dot{X}\ \left(\frac{Gy}{s}\right) \qquad 4.26$$

and the absorbed dose rate in air is simply related to the exposure rate in air, again under the conditions of electronic equilibrium. For the absorbed dose rate in materials other than air,

$$\dot{D} = 33.85\left[\frac{(\mu_{en}/\rho)_{med}}{(\mu_{en}/\rho)_{air}}\right]\dot{X}\quad \frac{Gy}{s}. \qquad 4.27$$

In the literature, the factor f (rad/R), which was used to convert exposure to absorbed dose, will be found. The numerical value of f is about .96 over a large energy interval. This gave the result that the exposure in air was approximately equal to the absorbed dose in water (approximating tissue). As seen from equation 4.27, this equivalence no longer holds for absorbed dose and exposure. However, if we substitute the kerma rate, from equation 4.21, into equation 4.27, we arrive at

$$\dot{D} = \dot{K}_{air}\frac{(\mu_{en}/\rho)_{med}}{(\mu_{tr}/\rho)_{air}}. \qquad 4.28$$

Both $\dot{D}$ and $\dot{K}_{air}$ are in the same units, and the attenuation coefficient ratio, in the case of water to air, is approximately 1.08 between 0.1-10 MeV.[14] So, the absorbed dose rate in water is approximately equal to the air kerma rate. Or, putting it another way, a measure of the kerma rate in air can be interpreted as approximately equal to the absorbed dose rate in a small tissue mass.

I. Absorbed Dose Rate - Fluence Rate of Photons

The absorbed dose rate in terms of the fluence rate of photons in a beam of radiation is given by

$$\dot{D} = 1.6\mathrm{x}10^{-13}\phi E_{\gamma}\mu_{en}/\rho \ \text{(Gy/s)} \qquad 4.29$$

where ϕ is the fluence rate in photons/m²s, E is the energy of the photons in MeV, and μ_{en}/ρ is in m²/kg. This expression is valid well within the beam dimensions. For $\dot{D}$ in Gy/h, multiply equation 4.29 by 3600.

J. Absorbed Dose Rate - Isotropic Point Source - Alpha and Beta Radiation

If an isotropic point source of C(Bq) is an α emitter, then the absorbed dose rate at the point A (see Figure 4.1) is given by

$$\dot{D}\left(\frac{Gy}{s}\right) = \frac{7.96\mathrm{x}10^{-2} n\, C_p}{r^2}\left(\frac{S}{\rho}\right) \qquad 4.30$$

where n_p is the fraction of particles of initial energy E (in MeV) and S/ρ is the mass stopping power for a particle of residual energy E in Jm^2/kg. Since α radiation does not have a very long range in any substance, this expression is valid only when the distance r to the point A is well within the range of the particle (generally < range/2).

For purposes of calculation, the mass stopping power S/ρ for use in the previous formula is given in Figure 4.3, in terms of Jm^2/kg of air for α particles from 1-8 MeV. To use equation 4.30, one must estimate the residual energy of the α particle. Suppose the initial energy of the α particle is 5 MeV and the distance of interest is r = 0.018 m. From the range equations 3.5, the estimated energy loss is $E = \frac{.018}{.0056} = 3.2$ MeV. The residual energy would then be 1.8 MeV and the

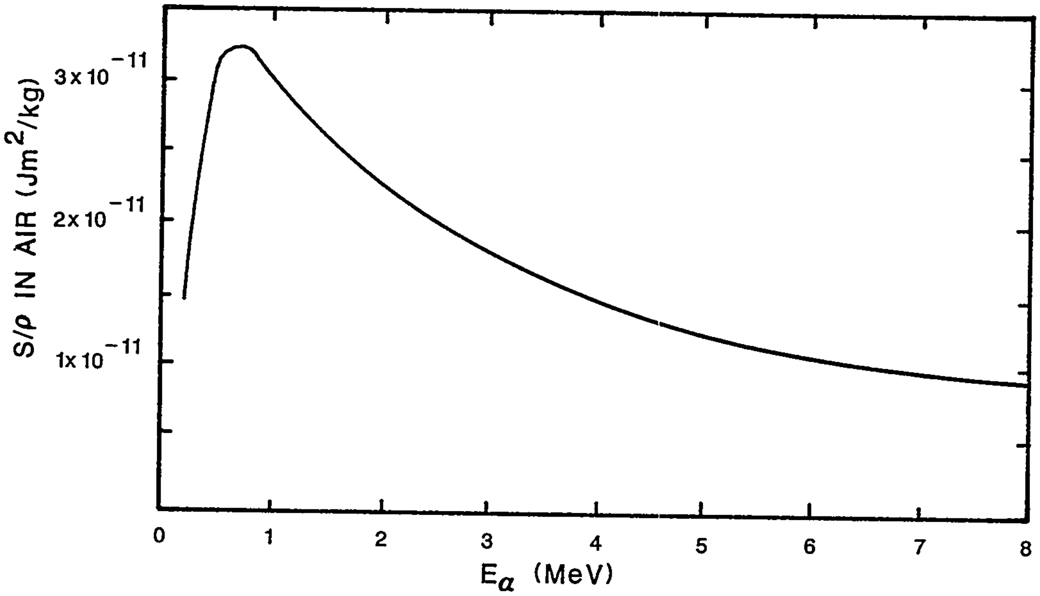

Figure 4.3 Mass stopping power of alpha radiation in air as a function of energy (based upon information in reference 16).

S/ρ value to use in equation 4.30 would be approximately 2.4×10^{-11} Jm^2/kg. The absorbed dose rate in tissue can be roughly estimated by use of the relative mass stopping power $(\frac{S}{\rho})_t$ of tissue with respect to air, which can be taken as approximately 1.2.

For substances other than tissue, use equation 3.9,

$$(S/\rho)_m \sim \sqrt{A_{air}}/\sqrt{A_{medium}}$$

This information allows one to calculate roughly the absorbed dose rate from an α source, but only well within the range of the particle in the medium. The reason for this is that the specific energy loss of the particle does not change too much until near the end of its range, and also the scattering can be neglected. Near the end of the range of an α particle, the specific energy loss, as well as the stopping power, changes very rapidly, and the above information is no longer valid.

In the case of β radiation, a further difficulty is encountered, namely, the existence of a spectrum of energy. Also, the scattering of β particles is quite extensive. Because of the spectrum of energy, the stopping power of the β distribution is not a constant to begin with. As the particles move through a medium, the stopping power will be continually changing as the betas lose energy. Since betas travel such a tortuous path due to scattering, it is difficult to predict the number of electrons at any depth in the medium. For the previously stated reasons, the spectrum characteristics of β radiation change continuously with depth in the medium. As a consequence, detailed information about β spectra at a given depth in a medium are generally lacking.

Early efforts at β dose determination involved an empirical approach by Loevinger[17] and a moments-method calculation by Spencer.[18] Later work by Cross[19] and Berger[20] have led to tabulations of beta dose rate functions for point sources. Both of the latter determinations used the earlier work of Spencer to help develop their tabulations. A discussion of the approaches used in the development of β point source functions can be found in Loevinger.[21]

For the estimation of β dose rates, the tabulations of Berger[20], for monoenergetic electrons and 75 β emitters, or the updated work of Cross[22] will be found useful for point source determinations. For extended sources, one may consult the discussion by Berger.[22]

K. Absorbed Dose Rate - Fluence Rate of Particles

The absorbed dose rate for a fluence rate of charged particles in a beam of radiation is given by

$$\dot{D}(\mathrm{Gy/s}) = \phi_p \, (S/\rho), \qquad 4.31$$

where ϕ_p is the particles/m²s and S/ρ is the mass stopping power (Jm²/kg) for a particle of energy E (in MeV). In this case, the value of S/ρ in air for α particles can be found from Figure 4.3. If the stopping power given in the literature is in units of MeV cm²/g, it must be multiplied by 1.6×10^{-14} to convert the units to Jm²/kg.

For electrons or positrons, the mass stopping power can be found in the Tables of ICRU Report 37.[24] One may obtain either the mass stopping power for air or tissue in this report.

For other applications, one may use the dose conversion factors computed by Kocher.[25-27] These are factors relating dose delivered for immersion in air and water, and for radioactive sources on a plane surface.

L. Absorbed Dose and Kerma for Neutrons

The measurement of the absorbed dose from a neutron source is very difficult, because the energy transfer depends so much on the neutron energy as well as the properties of the struck nucleus. In Section 3, the neutron interactions were discussed. For fast neutrons (energies from 10 keV to 10 MeV), elastic scattering is the important process, especially for the light elements. For thermal neutrons (energy of 0.025 eV), capture is the important process. In these processes, it is the secondary products which directly impart the energy to the substance. The problem increases in magnitude when one deals with a spectrum of energy rather than a monoenergetic neutron source. Also, if one is dealing with a compound, the presence of many elements complicates the picture.

The absorption of neutron energy is a two-step process; namely, the transfer of energy to the medium followed by absorption of radiation by the medium. The kerma describes step one; the absorbed dose describes step two. In general, the transfer of energy to the charged particles occurs at a different site in the medium than the absorption of energy by the medium. The absorption of radiation becomes more complex, as the nature of the secondaries, both charged and uncharged particles, changes with depth of penetration. Since the ranges of the charged secondaries will vary, the absorbed dose will vary with depth. In general, the kerma is a quantity which is more easily calculated than the absorbed dose.

If the neutron fluence or fluence rate and the neutron energy are known at a <u>point</u> in a medium, then the kerma may be computed. From equation 4.20a, the kerma rate for a monoenergetic field is

$$\dot{K} = \left(\frac{\mu_{tr}}{\rho}\right)\psi = \left(\frac{\mu_{tr}}{\rho}\right)\phi E.$$

Values of the kerma in different materials for monoenergetic neutrons from ~ thermal up to 30 MeV are listed in ICRU Report 26.[28] Figure 4.4 is a plot of air kerma rate and tissue kerma rate factors ($\dot{K}/\phi$). For the case of a spectrum of neutron energy, one must integrate the above expression to obtain the kerma rate:

$$\dot{K} = \int \frac{\mu_{tr}(E)}{\rho}\psi(E)\,dE = \int \frac{\mu_{tr}(E)}{\rho}E\phi(E)\,dE \qquad 4.32$$

in which $\frac{\mu_{tr}}{\rho}$, $\psi(E)$ and $\phi(E)$ are functions of the neutron energy.

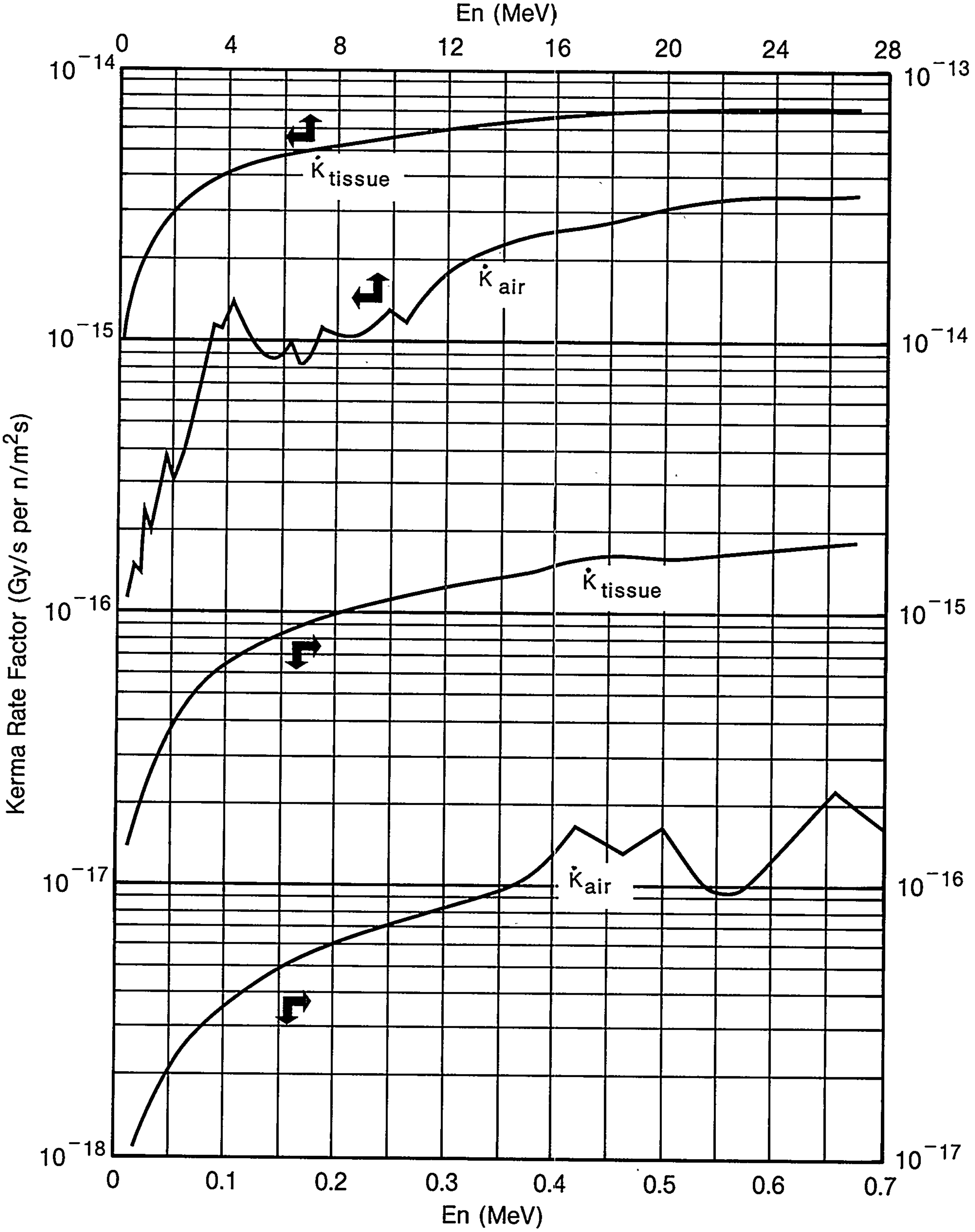

Figure 4.4 Neutron kerma rate factors for air and tissue. (Data adapted from Reference 28)

In some cases, a crude spectrum is known, or has been measured, which gives the fluence rate of neutrons in energy groups. It is then possible to estimate the kerma rate from

$$\dot{K} = \sum_{i=1}^{n} \bar{\dot{K}}_i(E)\phi_i(E) \tag{4.33}$$

in which $\dot{K}_i(E)$ is the average kerma rate factor from Figure 4.4 for the i^{th} energy group and $\phi_i(E)$ is the neutron fluence rate for the same group.

As in the case of photons, when charged particle equilibrium is obtained in the medium, the neutron absorbed dose rate at some point will be approximately equal to the kerma rate. For this case, the kerma rate is constant within a distance equal to the maximum range of the charged secondaries.[28] In large objects, the multiple scattering which occurs, makes the determination of the absorbed dose more difficult to compute. Computer calculated depth dose distributions of absorbed dose for neutrons in the range thermal-14 MeV in cylindrical phantoms and from 0.5-400 MeV in tissue slabs can be found in Reference 29. In addition, this reference also contains dose equivalent depth distributions.

M. Neutron Dose Equivalent

The dose equivalent calculations of NCRP Report No. 38, when compared to the absorbed dose calculations, allow the determination of an effective quality factor $\bar{Q}$. This value varies at each point within a phantom because of the changing energy spectrum of the neutrons. The NCRP[29] recommends that for protection purposes, the highest value of $\bar{Q}$ be applied. When this is done, one may arrive at a table of fluence rate factors versus neutron energy for a given dose equivalent rate. Such a compilation is shown in Table 4.3, which has been adapted from NCRP Report No. 38. A review of experimental evidence has indicated that the previously used values of quality factors for fast neutrons may be low by a factor of 2.[13] The NCRP has recommended that the quality factors in the table be multiplied by two for the case of known energies.[30]

TABLE 4.3*

Mean Quality Factors, $\overline{Q}$, and Neutron Fluence Rates for a Dose Equivalent Rate of 25 μSv/h (2.5 mrem/h)

Neutron Energy, MeV	$\overline{Q}$	Neutron Fluence Rate, $m^{-2}s^{-1}$
2.5x10^{-8} (thermal)	2	6.8 x10^{6}
1x10^{-7}	2	6.8 x10^{6}
1x10^{-6}	2	5.6 x10^{6}
1x10^{-5}	2	5.6 x10^{6}
1x10^{-4}	2	5.8 x10^{6}
1x10^{-3}	2	6.8 x10^{6}
1x10^{-2}	2.5	7.0 x10^{6}
1x10^{-1}	7.5	1.15x10^{6}
5x10^{-1}	11	2.7 x10^{5}
1	11	1.9 x10^{5}
2.5	9	1.6 x10^{5}
5	8	1.6 x10^{5}
7	7	1.7 x10^{5}
10	6.5	1.7 x10^{5}
14	7.5	1.2 x10^{5}
20	8	1.1 x10^{5}
40	7	1.0 x10^{5}
60	5.5	1.1 x10^{5}
1x10^{2}	4	1.4 x10^{5}
2x10^{2}	3.5	1.3 x10^{5}
3x10^{2}	3.5	1.1 x10^{2}
4x10^{2}	3.5	1.0 x10^{5}

*Adapted from NCRP Report No. 38, Protection Against Neutron Radiation.

If one has a measure of the neutron fluence rate, then this table can be used to find the dose equivalent rate which this fluence rate would represent. When little information on the neutron energy is available, one should assume a quality factor of 20,[30] which would reduce the fluence rate number by a factor of 2.

The values of fluence rate in Table 4.3 are sometimes referred to as ϕ_{AVG}, meaning the fluence rate which will result in an average level of exposure of 1 mSv (0.1 rem) in a work week (40 h). The dose equivalent rate from a fluence rate of ϕ n/m^{2}s of a given energy can be found by use of the table. In this case, a simple proportion would exist:

$$\frac{\dot{H}}{25(\mu Sv/h)} = \frac{\phi}{\phi_{AVG}} \qquad 4.34$$

This gives the dose equivalent rate of the fluence rate ϕ. The dose equivalent, H, would then be

$$H(\mu Sv) = \dot{H}t = \frac{25\ \phi t}{\phi_{AVG}}, \qquad 4.35$$

in which t is the time of exposure, in h.

In general, the suggested procedure will overestimate the dose equivalent when the energy of the neutron flux density is not known, but for protection work the added safety factor is useful.

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QUESTIONS

4.1 Name the organization which selects and defines the units and quantities of radiation.

4.2 What is defined by the sum of the electrical charges on all ions of one sign produced in air when all the electrons, liberated by photons in a volume element of air whose mass is Δm, are completely stopped in air?

4.3 What is the unit of exposure?

4.4 What does the energy imparted by ionizing radiation to matter in a unit volume element containing mass Δm, define?

4.5 What is the unit of absorbed dose?

4.6 What is defined by the product of the absorbed dose and the quality and distribution factors?

4.7 What is the unit used to express the dose equivalent H?

4.8 Upon what is the exposure concept for photons based?

4.9 What does exposure per unit of time define?

4.10 What special unit is used to express exposure per unit of time for gamma and x rays?

4.11 Make a chart that indicates each symbol and its meaning in the general expression for exposure rate.

4.12 What assumption is made in the simplified exposure rate formulae?

4.13 What is defined by the number of ionizing photons passing through a unit area per unit of time?

4.14 How is kerma defined? What is its unit?

4.15 How are exposure and kerma related?

4.16 Explain why the factor $e^{-(\mu_{en}/\rho)x}$ would not be sufficient in the general dose rate formula.

4.17 Explain why it is not expedient for one to have exact expressions for the mass attenuation coefficient and the mass stopping power of α and β particles in radiation safety work.

4.18 Identify the following:

a) The fraction of a radiation beam removed per unit of thickness of a material.

b) The fraction of a radiation beam passing through a unit of area that is removed per unit of mass.

c) The fraction of energy removed from a beam of radiation passing through a unit of area per unit of mass.

4.19 Present some reasons for the difficulties in establishing satisfactory measurements for neutron protection.

4.20 What does the symbol ϕ_{AVG} signify?

PROBLEMS

4.1 A charged cylindrical ionization chamber having a diameter of 0.03 m and a length of 0.1 m is drained of 1 μcoulomb as the result of exposure to an x ray beam. What was the exposure?

Answer: 0.011 C/kg

4.2 a) Kr-85 emits a photon of energy 0.514 MeV with an intensity of 0.434%. Find the gamma-exposure rate at 1 m, at 0.5 m and 0.01 m in air from a source of Kr-85 of 3.7×10^{10} Bq. The linear energy attenuation coefficient (μ_{en}) is 0.00385 m^{-1} and the density of air is 1.293 kg/m^3.

Answer: 3.32×10^{-7} C/kgh @ 1 m, 1.33×10^{-6} C/kgh @ 0.5 m,
3.33×10^{-3} C/kgh @ 0.01 m

b) Use the approximate formula (4.15) of the text and compute the exposure rates in part a and compare the results. Is the difference significant?

Answer: 3.09×10^{-7} C/kgh @ 1 m, 1.24×10^{-6} C/kgh @ 0.5 m,
3.09×10^{-3} C/kgh @ 0.01 m

4.3 a) Find the γ ray exposure rate at 0.3 m for a 1.85×10^{8} Bq Cobalt-60, ^{60}Co source. Cobalt-60 emits two γ photons 1.33 MeV and 1.17 MeV, each of which occurs essentially 99% per disintegration.

μ_{en}/ρ for 1.33 MeV is 0.00259 m^2/kg and for
1.17 MeV is 0.00267 m^2/kg

b) Use the approximate formula and compare with the result of a).

Answer: a) 1.81×10^{-5} C/kgh.
b) 1.90×10^{-5} C/kgh.

4.4 Find the γ ray exposure rate at 10 m for a 3.7×10^{10} Bq, Sodium-22, ^{22}Na, source. ^{22}Na emits a 1.28 MeV photon essentially 99% per disintegration. Use the approximate formula. Ignore air attenuation.

Answer: 1.76×10^{-6} C/kgh.

4.5 ^{137}Cs emits a γ ray of 0.6616 MeV with an intensity of 84.62%. Find the γ fluence rate at 0.3 m, 0.5 m and 1 m from a point source (10^9 Bq). Assuming μ_{en}/ρ to be 2.939×10^{-3} m^2/kg, calculate the exposure rates.

Answer: 7.48×10^8 ph/m^2 s @ 0.3 m and 2.48×10^{-5} C/kgh
2.69×10^8 ph/m^2 s @ 0.5 m and 8.91×10^{-6} C/kgh
6.73×10^7 ph/m^2 s @ 1 m and 2.23×10^{-6} C/kgh

4.6 Calculate the absorbed dose rates in the problem above.

Answer: 8.39×10^{-4} Gy/h @ 0.3 m
3.02×10^{-4} Gy/h @ 0.5 m
7.55×10^{-5} Gy/h @ 1 m

4.7 A 50 MeV proton beam from a linear accelerator has a measured fluence rate of 10^{12} protons/m^2s. The mass stopping power in air is 1.76×10^{-13} Jm^2/kg. Calculate the absorbed dose rate in the beam.

Answer: 0.176 Gy/s or 633.6 Gy/h.

4.8 Using the graph, "Neutron Kerma Rate Factors for Air and Tissue," find the tissue kerma rate for 10^{10} n/m^2 s of 4 MeV.

Answer: 4.3×10^{-5} Gy/s.

4.9 If the maximum allowable dose limit for a week (40 hours) is 1 mSv, how long can a person work in a thermal neutron flux of 10^8 n/m^2 s such that he would not exceed the limit?

Answer: ~ 163 min.

4.10 In the above problem, if the quality factor is increased by a factor of 2, how long can the person work in the same neutron flux?

Answer: ~ 80 min.

SECTION 5 - BIOLOGICAL EFFECTS AND RISKS OF RADIATION

A. The Cell - Basic Unit of Structure

This section will be devoted to the effects of ionizing radiation on the human body. The structure of the body is quite complex, and it is often of value to deal with effects at certain levels of organization within the body. Thus, the human body contains many organs, each of which is composed of two or more types of tissue. In turn, a tissue is composed of similar cells, there being four types of tissues in the body: epithelial, connective, muscle, and nerves. The job of a tissue is to perform a special body function. The cell, which makes up the tissues of the body, is the basic unit of structure and function in the body. A cell contains many elements; hydrogen, oxygen, carbon and nitrogen are the main components. Human cells vary in size from approximately 3-100 μm.

A cell is normally composed of a nucleus surrounded by cytoplasm, both encased in membranes. Although highly complex in structure, the nucleus and cytoplasm contain about 70 percent water.[1] The vital part of the cell is the nucleus, usually an oval body located near the center of the cell. From a chemical standpoint, the nucleus is quite active. The normal growth of the cell is controlled by the nucleus. Also, it initiates cell division and controls the repair of injured cells. The cytoplasm is a more or less colorless liquid substance. It secretes enzymes, and controls absorption and excretion in the cell.

Among the complex structures found in the cell are carbohydrates, fats, proteins and nucleic acids. Carbohydrates serve as food for the cell and as structural units. Fats store chemical energy. Proteins, which differ from each other by the number, sequence, length and arrangement of amino acid chains, are involved in all the metabolic activities (chemical reactions) which sustain cell life. Amino acids are composed of NH_2 groupings and about 20 amino acids are found in the proteins of mammals. The nucleic acids, DNA (deoxyribonucleic acid) and RNA (ribonucleic acid) function together to produce the cell's proteins. Most of the DNA is found in the nucleus. The DNA molecules are thought to carry the genetic code necessary for proper cell reproduction.[2] The RNA is distributed throughout the cell and is thought to be the messenger which translates the information contained in the DNA into instructions for protein production.

Many body cells have only a limited life span. In order that their functions be carried on, cells divide at a certain stage in their life (mitosis). The daughter cell then takes over the functions of the parent cell. As a cell divides, there appear in the nucleus threadlike structures called chromosomes. The chromosome number is fixed for a given species (23 pairs in human cells). Arranged linearly along the chromosomes are the genes which determine hereditary characteristics. Prior to cell division, the number of chromosomes is temporarily doubled. When a cell divides, the daughter cell receives a duplicate set of chromosomes from the parent as well as identical genes. If the process is normal, no alterations or changes occur. However, any changes which do occur in the chromosomes and/or genes are called mutations. These changes or mutations can then affect either the daughter cell or future cells.

The development of an organ then proceeds from mitosis, the term used to designate cell division. In a bisexual species, the union of two cells (gametes) - one from the male and the other from the female - produces an original cell from which the species will be reproduced.

This cell then undergoes a number of divisions which increase its number. In the embryonic state, all the cells look alike. However, changes in the structure of the cells begin to take place. The changes enable the cells to perform specialized functions. This process of change is referred to as differentiation. The result of this process is the development of differential cell types or lines. One of these lines is the germ line, the rest are called somatic. The germ line gives rise to either male or female gametes. The somatic lines develop into the tissues of the individual.

Since only the gametes can be transmitted to future generations of the species, damage to somatic cells is limited to the individual. Damage to the offspring of an individual may occur when there is damage to the cells of the germ line.

A great many agents can cause injuries to the cells. When such injury occurs, the effects are the same regardless of the agent which caused the damage. Ionizing radiation produces damage to cells, but in a mostly nonspecific way; that is, other physical and chemical substances cause the same effects because the body responds the same to certain cell damage regardless of the cause.

Radiation passing through living cells will directly ionize or excite atoms and molecules in the cell structure. These changes affect the forces which bind the atoms together into molecules. If the molecule breaks up (dissociates), some of the parts will be charged. These fragments are called free radicals and ions, and are not chemically stable. Free radicals are electrically neutral structures with one unpaired electron.[3] Because the cell has a higher water content, the most important free radicals are those formed from water molecules. For example, an excited H_2O^* molecule may dissociate into

$$H_2O^* \rightarrow H^\circ + OH^\circ,$$

in which the hydrogen radical H° has an unpaired e^- and the OH° radical will have nine e^-, one of which will be unpaired. The free radicals are very reactive chemically, and when combining can produce hydrogen peroxide (H_2O_2), which is a chemical poison, and the HO°_2 radical which is more damaging than peroxide. Further effects are produced when the radicals and ions interact with other cell material. In this way, damage is caused in a direct and indirect manner.[2] The role that each type of action plays in the total damage to the cell is still an unsolved problem. Of the damage which is done, the effects are greatest in the nucleus of the cell, but injury to the cytoplasm can also cause serious effects in the cell.

The total effect on cell processes is a function of the dose of radiation. The cell processes will be affected in varying degrees up to the ultimate result-cell death. Some damage to the cell may be repaired. This can be accomplished by action of the cell itself, or by replacement of badly injured cells in a given tissue through mitosis of healthy cells. On the other hand, if the extent of the damage to an organ is quite large, the organ may not be able to repair itself. That is, damaged cells may show confused growth but eventually be unable to divide. Or the cells may begin to exhibit uncontrolled growth. Although many factors are important in assessing the total damage, it seems likely that most cell functions and structures are somewhat impaired by radiation.

B. Radiosensitivity

Since the cells which make up the tissues of the body differ both in appearance and function, one might suspect that their response to radiation would also differ. Such is the case, and this property is known as the radiosensitivity of the cell. The first statement about this property was given by Bergonié and Tribondeau. They found that the radiosensitivity of a tissue is directly proportional to the reproductive capacity and varies inversely with the degree of differentiation. Since then, other factors have been found which affect the radiosensitivity. Among these are the metabolic state of the cell, the state of cell division, and the state of nourishment. It turns out that to produce a given effect, the necessary radiation dose varies inversely as the relative sensitivity of the given tissue.

Thus, cells which are most active in reproducing themselves, cells which have a high metabolic rate (rate of chemical changes) in the cell, and those cells which are more nourished than others are more sensitive to radiation.[4] Also, there is evidence that cells are more susceptible to radiation at certain stages of division than at others. Moreover, cells not fully mature will also be more harmed by radiation than mature cells. In the body, bone marrow, lymphoid tissues, and the reproductive organs rank among the most radiosensitive. Muscle and bone cells are the least radiosensitive.

C. Radiation Damage

As has been pointed out, damage to somatic cells is limited to the individual whereas damage to germ cells may result in damage to the offspring of an individual. One may broadly classify biological effects in man as somatic or hereditary. Somatic effects include any and all types of damage which affect only the individual; hereditary effects are those which can be transmitted to a future generation[1,5]. Thus, damage to the genes of a somatic cell may produce damage to a daughter cell, but this would be a somatic effect, not hereditary. The term genetic damage refers to effects caused by chromosome and/or gene mutations. This may lead to hereditary effects only when the damage affects the germ line since only then can these effects be transmitted to a future generation.

D. Factors Influencing Radiation Effects

The radiation effects in man and animals are usually discussed in terms of total body and partial body irradiation, and with reference to damage to an organ. Because of the importance of some organs in the body, certain damage to these can induce effects in other organs. A number of physical factors are important in the determination of radiation effects:

1) sensitivity of the individual-for a select group, the effects may differ greatly from those in a heterogeneous group;

2) nature or type of radiation-some types of radiation are more effective in producing damage;

3) the absorbed dose-this is a function of the energy absorbed per gram of tissue;

4) time distribution or fractionation-a lethal dose given in a short time may not be lethal if protracted over a long time;

5) dose distribution-is the total body involved or only a specific organ?

6) age at irradiation-response is altered during growth in some systems.

The nature, severity and duration of biological effects depend upon the above and other factors. The combination of factors makes the effects on different organs differ for changes in the number of relevant parameters.

1. Individual Sensitivity - Dose-Effect Curve

Because a number of factors enter into the response of individuals to radiation, one might expect an effect to be seen in some at a lower dose than others. This may be due to certain host factors in the individual[6], such as general health, previous exposure to other agents, etc. Such a variation is seen, so that, if one studies the fraction of individuals which exhibit a given effect as the dose is varied, a dose-effect curve (Figure 5.1) is obtained. In the figure, for very low doses, no effects are seen. As dose is increased, the % affected increases with increased dose. Above 50% affected, the rate of increase slows, and larger doses are required to produce the effect in the remaining population. In the study of radiation effects, the dose D at which 50% of the population is affected is generally used as the reference dose. If the effect is death, the symbol LD_{50} called the median lethal dose (lethal

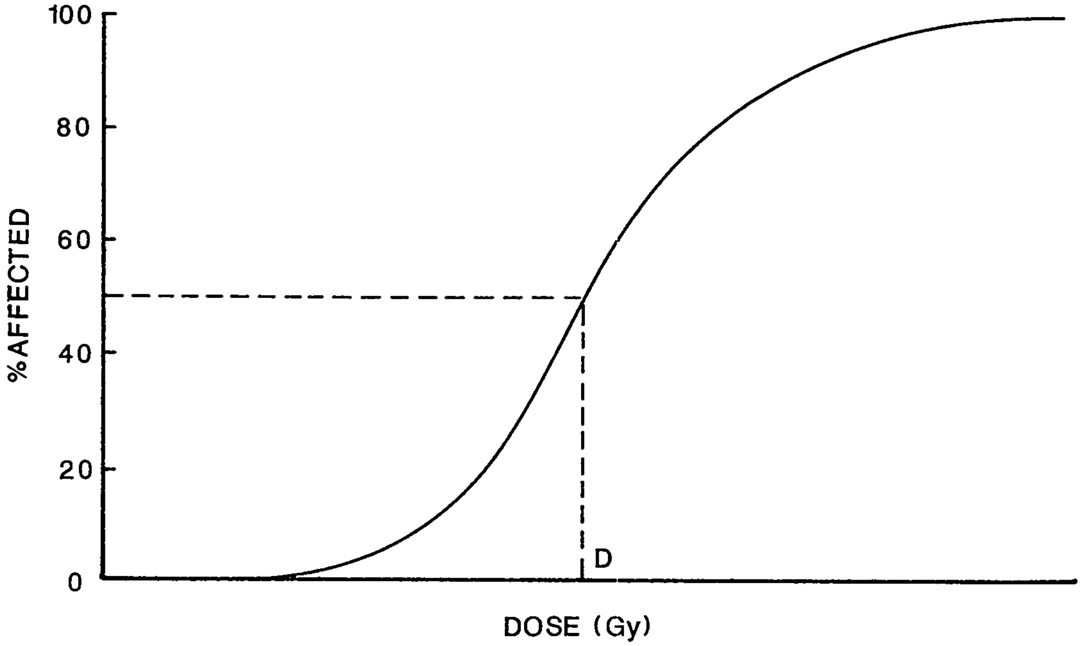

Figure 5.1 Dose-effect relationship.

dose to 50% of the exposed), is used. Moreover, since the time for an effect to show up may also vary, the symbol $LD_{50/30}$ (lethal dose to 50% of the exposed within 30 days of irradiation) may be used.

Getting back to the dose-effect curve, we note that at D_{50}, half of the exposed would show the particular biological effect. This means that a single value of the dose cannot adequately describe the probability for <u>all</u> individuals. That is, almost half of those exposed show the effect for less than D_{50}. On the other hand, half have not yet shown the effect at D_{50}. This type of statistical behavior is typical of the response to radiation seen in mammals and implies a normal distribution of sensitivity with respect to dose.[2]

2. Type of Radiation

As discussed in Section 4.B, the damage done by radiation depends upon the type of radiation. Not all radiation types are equally effective in producing biological damage. In radiobiology, the effectiveness is determined by the relative biological effectiveness (RBE). This quantity can be determined if one can establish the effect and control the conditions of the exposure. For this case,

$$\text{RBE} = \frac{\text{Dose of 200-250 kV x rays to produce the effect of interest}}{\text{Dose of comparison radiation to produce the same effect}}$$

In radiation protection work, one cannot control the conditions of exposure or concentrate on one effect. Thus, the quality factor, Q, an assigned value, is used to denote that the degree of response is modified for the type of radiation. Both the quality factor and the RBE are related to the LET. In general, for high values of LET one finds high values of RBE, and high LET radiation is more effective in producing damage then low LET.

For low LET radiation (electrons), the ion density is low so that recombination of ions and radicals is less likely. This allows ions and radicals to diffuse through the medium easily to form other products. This would increase the contribution of indirect effects. In addition, the

the small electrons colliding with large DNA macromolecules tend to produce only small bond breaks. This tends to lessen the effect of direct action.

In the case of high LET radiation (protons and other heavy recoils), the ion density is greater promoting more recombination and less diffusion. This would work to reduce indirect action. The much larger protons colliding with DNA would tend to produce many large fragments, and even damage the macromolecule beyond repair. This would enhance the effect of direct action.

If one looks at cells exposed to both low LET and high LET radiation, the survival curve will look something like Figure 5.2. In curve A, a similar set of cells are irradiated by high LET particles. The simple exponential implies that cells become inactivated following a hit. That is, single events are important. In curve B, cells of both high and low sensitivity are irradiated. The initial dropoff suggests the rapid

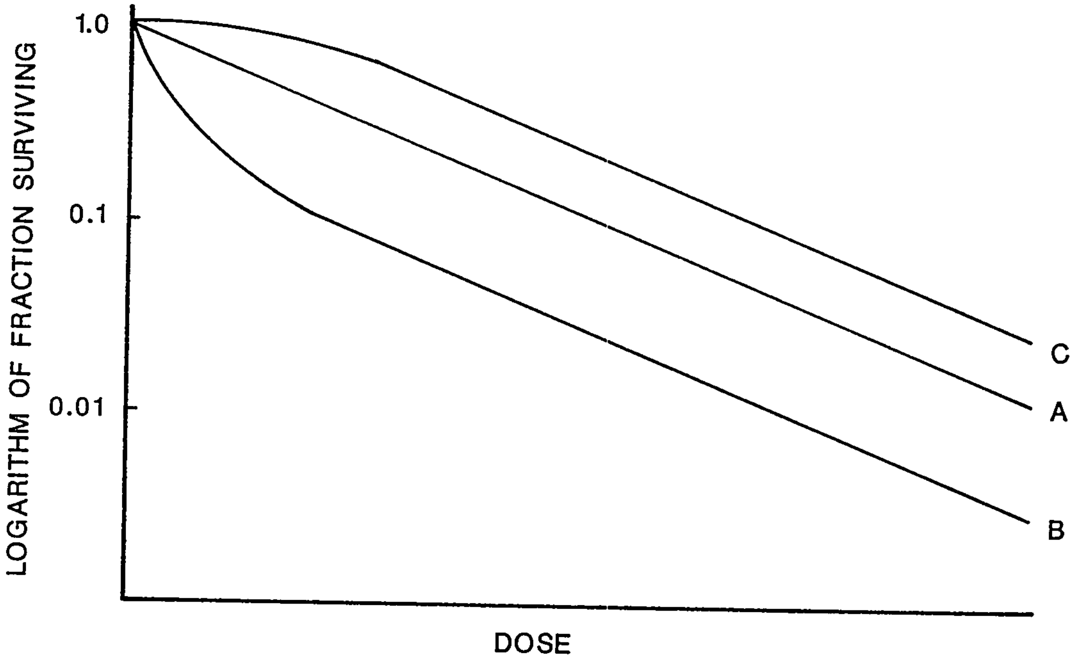

Figure 5.2 Survival curves for low LET and high LET radiation.

inactivation of the high sensitivity cells. The latter straight portion represents the removal of the low sensitivity cells. Curve C is typical of most types of similar cells irradiated by low LET radiation (^{60}Co).[7] The initial shoulder portion implies that cells sustain non-lethal damage at first, followed by more damage which inactivates the cell.

Many models have been suggested to explain the shape of the survival curves.[8] The simplest model explains curve A by assuming an exponential in which the slope of the line is a constant=1/D. This model is compatible with the assumption of direct action (or single hits) being the important reactions. Curve C is better explained by assuming a multi-target, single hit model.[7] In this view, a number, n, of targets must all be hit at least once to kill the cell. This is consistent with the shoulder portion of the curve C, which indicates more interactions are necessary to inactivate cells. That is, sublethal damage is followed by lethal damage.

Some qualitative results which apply to high LET radiation are:

1) a smaller dose is required to achieve a given degree of effect,
2) the exponential survival curve implies little recovery from sublethal damage,
3) fractionation of dose seems to have little effect, and
4) the degree of damage is not affected greatly by dose rate.

For low LET, the following apply:

1) there is some recovery of sublethal damage,
2) fractionation of dose results in less effect, and
3) a dose rate dependence is seen, the effects of low dose rate are 1/3 to 1/6 those at high dose rate.

3. Absorbed Dose

The basic parameter, which can be associated with biological damage, is the energy absorbed per unit mass. However, we have seen that the damage produced is also related to the type of radiation. But, in a given system, the absorbed dose for a given radiation type will acquire added significance in a small tissue mass. Moreover, for a very large energy deposition per unit mass, the absorbed dose will override the LET considerations.

4. Time Distribution or Fractionation

If a dose of some value is delivered in a number of smaller fractions, the effectiveness of the radiation is often reduced. This sparing effect is attributed to repair of sublethal damage and replacement of cells suffering lethal damage. For low LET damage, it is assumed that sublethal damage occurs before cell inactivation results. Since successive low LET doses, separated in time, are less effective than their sum delivered in a single, short dose, sublethal damage is somewhat repairable.[7] When irradiation is given over many doses, repair occurs after each dose. In addition to repair, cell replacement is also able to offset cell death. Although little is known about the dose rate at which cell replacement counterbalances cell death, it is known that this effect acts to mitigate the effect of cell killing.

5. Dose Distribution

When the whole body is irradiated, the total effect is more severe for a given dose than if that dose were given to a specific organ. This indicates that when the tissue volume is reduced, the severity of the response is also reduced. The magnitude of this sparing effect has not been adequately measured.

6. Age At Irradiation

In addition to the fact that effects are more pronounced in the young, there is also an increased sensitivity in the unborn.[4] Certain organs, such as bone and cartilage, show a definite response during growth but are relatively radioresistant when mature.[7] Whereas, for the ovary, sensitivity for sterility decreases with age. Damage to the germ cells is of concern only during the period of child bearing. Some effects are irrelevant when the life expectancy of the individual is much less than the latent period required for the effect to show up.

E. Early Somatic Effects - Acute Radiation Syndrome

The somatic effects which are observed can be loosely divided into early and late effects. Although quite an arbitrary grouping, early effects are classed as those which appear within a few weeks after the exposure. The range of these effects as well as their duration depend upon the dose. For very high whole body doses, there are three basic forms of early or acute damage. In the range above about 20 Gy (2000 rad), the dose is fatal within a day or two.[2] The same symptoms appear when the head suffers severe irradiation, which points to a breakdown of the central nervous system. This type of acute radiation syndrome is thus referred to as central nervous system death (CNS death).

For the range 5-20 Gy (500-2000 rad), symptoms may appear within hours. Death often occurs within a week or so. In this mode, the damage to the lining of the intestinal tract is the most severe. This form is called gastrointestinal tract death (GI death). At the lower end of this dose range, it is possible for one to survive this mode of death only to fall victim to the effects which prevail at lower doses.

At doses $<$ 5 Gy (500 rad), the most important effect is damage to the blood forming organs. Since these centers are located in the bone marrow, this mode of death is often called bone-marrow death. The first signs may appear within a few days, depending upon the dose, and the total effect may not develop for a few weeks. Severe changes occur when the dose is $>$ 2 Gy (200 rad). In the range above 3 Gy (300 rad), the damage is severe enough so that death becomes more and more probable.

In the preceding sections, the results are based on x and γ ray data. As such, the conversion to rem for other types of radiation is not justified in this instance.

The main clinical effects which follow acute exposure to total body doses of ionizing radiation are nausea, vomiting, loss of appetite and fatigue for doses > 1 Gy (100 rad) or so.[3] As indicated above, the time of onset, the severity and the duration of the effects depend on the dose and the exposed organs. A summary of clinical effects for acute doses is given in Table 5.1 and may be used as a rough guide. Note the virtual absence of any symptoms in the dose range below 1 Sv (100 rem). Some people would be expected to have mild symptoms in the range 0.5-1 Sv (50-100 rem), because of differences among individuals. Below 0.5 Sv (50 rem), however, no symptoms at all are expected. In fact, special techniques are needed to detect doses this low.[4]

Death occurs in a larger fraction of cases as the dose increases. If the dose becomes large enough, all cases of exposure result in death. In the range where survival is possible, the concept of the median lethal dose (LD_{50}) is used. For man, the best estimate places the LD_{50} in the range of 3 to 5 Gy (300-500 rad). Of course, in this range all would have severe symptoms. Note that this dose refers to short-term total body radiation.

F. Late Somatic Effects

The problem in the study of late effects resulting from exposure to radiation is that the elapsed time may be rather long, thus making it hard to relate the cause to the effect. Since the late effects may be caused by many other agents besides radiation, there can be no positive assignment of the cause in most cases. At best, it can be shown that radiation increases the incidence of these nonspecific injuries.

The main late effects are discussed in the following.

1. Cancers

In its 1980 report,[6] the Committee on the Biological Effects of Ionizing Radiation (BEIR) indicated that cancer was considered to be the most important somatic effect of low-dose ionizing radiation. Cancer

		Therapeutic Range – 1 to 10 Sv			Lethal Range – Over 10 Sv	
Range:	Subclinical 0 to 1 Sv	1 to 2 Sv	2 to 6 Sv	6 to 10 Sv	10 to 50 Sv	Over 50 Sv
Incidence of Vomiting:	None	1 Sv: 5% 2 Sv: 50%	3 Sv: 100%	100%	100%	100%
Delay Time:	-	3 Hours	2 Hours	1 Hour	30 min	30min
Leading Organ:	None	Hematopoietic Tissue			G.I.Tract	Central Nervous System
Characteristic Signs:	None	Moderate Leukopenia	Severe Leukopenia, Hemorrhage, Infection, Purpura, Epilation Above 3 Sv		Diarrhea, Fever, Disturbance of Electrolyte Balance	Convulsions, Tremor, Ataxia, Lethargy
Critical Period Post-Exposure:	-	-	4-6 Weeks		5 to 14 Days	1 to 48 Hours
Therapy:	Reassurance	Reassurance, Hematologic Surveillance	Blood Transfusions, Antibiotics	Consider Bone-Marrow Transplantation	Maintenance of Electrolytic Balance	Sedatives
Prognosis:	Excellent	Excellent	Good Therapy Effective	Guarded Therapy Promising	Hopeless Therapy Palliative	
Convalescent Period:	-	Several Weeks	1 to 12 Months	Long	-	
Incidence of Death:	-	None	0 to 80% (Variable)	80 to 100% (Variable)	90 to 100%	
Death Occurs Within:	-	-	2 Months		2 Weeks	2 Days
Cause of Death:	-	-	Hemorrhage, Infection		Circulatory Failure	Respiratory Failure Brain Edema

Table 5.1 Summary of Clinical Effects of Acute Ionizing Radiation Doses.[9]

may be induced by radiation in nearly all human tissue. The main sites of solid tumors are the breast in women, thyroid, lung and some digestive organs. These tumors have long latent periods (approximately 10 to > 30 years) and occur in larger numbers than leukemia.[6] Leukemia (abnormal increase in white blood cells) has a much shorter latent period. The incidence peaks within a few years of exposure and returns to normal levels after about 25 years.[10] Reference 6 contains an extensive discussion on radiation induced cancers and presents data on the incidence in specific organs. The ICRP[10] has estimated that the fraction of all cancers, for both sexes and all ages, is 1.25×10^{-2} Sv^{-1} (1.25×10^{-4} rem^{-1}).

2. Tissue Effects

Among the more prominent late effects in tissues are cataracts (see 5.I.7) and sterility (see 5.I.4). Radiation-induced cataracts are slowly progressive over a period of time, but may stop or even regress.[1] Sterility is a late effect which may be either permanent or temporary. In some cases, fertility will return in a few years.[6]

3. Life-span

Information on life-shortening effects in man is still inadequate. The effects of long-term, low-level irradiation on longevity cannot be predicted. With the exception of tumor induction, there is no evidence of life shortening.[6,10]

4. Growth and Development

Effects on the embryo depend upon the dose as well as the age of the embryo. The younger the embryo, the more it is affected.[1] Here, as in the case of other late effects, the results of damage are the same as those caused by other agents. The effects on the fetus are so much more important since minor damage may be amplified during growth into a major anomaly. Relatively high doses can cause death, malformation, growth retardation and impairment of function.[6] Susceptibility to certain cancers appears to be higher during prenatal and childhood periods.[6,10]

The incomplete status of these problems at the present time reflects the lack of needed information. Such data are very hard to obtain, and require careful work and analysis. In each case of late effects, extensive data are needed before any real firm conclusions may be drawn. As the dose rate is reduced, one approaches background levels and any effects produced by the applied dose rates may be masked by effects produced by other agents or background radiation.

G. Hereditary Effects

The study of hereditary effects attempts to discover the traits which can be transmitted from generation to generation in a given species. The genes are the determinants of the inherited traits. Any change or mutation of a gene, which is usually quite stable, can result in an altered trait. Such changes can be produced by radiation, as well as other agents. The study of radiation-induced mutations is thus hampered by the fact that other substances also act to produce the same effects. Since the change is not unique, radiation only serves to increase the frequency of the effect. Increases in the rate are small even for high doses. Thus, the study requires the use of large numbers of subjects studied over many generations. In the case of man, the study is very difficult, since large numbers are seldom available and the time between generations is so long. To date, there has been no demonstration of radiation induced mutations in man.[10]

For this reason, much of the present knowledge is based on work with animals.[6] At all doses and dose rates used up to the present, radiation is known to induce mutations in all species studied. Because of this, any increase of radiation to humans should bring about an increase in the mutation rate. Sometimes the application of animal data to man can result in error. However, in this case, the effects on some other species are similar enough to those in man.

Studies have shown that some hereditary effects in man are caused by chromosome damage.[1] Up to the present time, the lack of knowledge about the harmful traits has made estimates of the magnitude of these effects uncertain. When more diseases and effects can be related to certain types of mutations, then better estimates can be made.

Genes may be dominant or recessive. When genes differ, the trait which results may be determined by either gene, or some intermediate trait may occur. If the trait is determined by one or the other of the genes, this gene is then a dominant gene. The other is a recessive gene. Mutations in dominant genes give rise to damage in the offspring of the first generation. Damage to offspring caused by recessive genes occurs only if the same altered gene is received from each parent. Unless these changes occur frequently, recessive damage won't show up for many generations.

It is estimated that already about 10% of human liveborn offspring suffer from serious disorders of genetic origin. In evaluating radiation induced genetic effects in humans, mouse data must be used.[6] The estimates of the increased serious genetic disorders in humans due to radiation exposure are small relative to the natural incidence cited above. That is, the ICRP[10] has estimated that the fraction of effects to be expected, in the first two generations, can be taken as about $4\text{x}10^{-3}$ Sv^{-1} ($4\text{x}10^{-5}$ rem^{-1}).

H. Stochastic and Nonstochastic Effects-Risk

For some of the effects that have been mentioned, the relative damage or severity of the effect increases with increased dose. Higher doses produce a greater degree of that type of damage. On the other hand, for some effects, increasing the dose increases the chance, or probability of the effect occurring. The terms "nonstochastic" and "stochastic" effects have been employed by the ICRP[10] to distinguish between these. Nonstochastic effects are those in which the severity of the effect varies with the dose. For these types of effects, a threshold dose may exist. That is, if the dose is kept below the threshold dose, the effect will not be observed. Nonstochastic effects are considered to result from the collective injury of a substantial number of cells in the tissue.[7]

Examples of such effects are cataracts, skin ulcerations or burns, depletion of blood-forming cells in bone marrow, and impairment of fertility.

Stochastic effects are those in which the probability of the effect occurring increases with dose, without threshold. Any dose, therefore, has a certain probability, however low, of causing the effect. Stochastic effects may result from injury to a single cell or a small number of cells.[8] Carcinogenic (cancer) and heritable effects are examples of stochastic effects. In these, once the effect is induced, the severity is already determined by the nature of the effect.

From a protection standpoint, preventing nonstochastic effects may be achieved if the dose limit is set low enough so that the threshold dose is not exceeded. In this case, the risk of producing nonstochastic effects could be reduced to zero. The ICRP has recommended dose limits intended to prevent detrimental nonstochastic effects.[10] However, since stochastic effects have some chance of occurring no matter how low the dose, the ICRP limits intend to limit the probability of stochastic effects occurring to an acceptable level. That is, any exposure to radiation involves a risk, and no risk should be undertaken without the expectation of a net benefit.[10] Since the setting of limits involves judgments that cannot wholly be based on scientific knowledge, the concept "acceptable risk" has evolved. The basis for an acceptable level of risk in radiation work is comparison to the risks in other occupations considered to have high safety standards.[10] The <u>average</u> annual mortality in these occupations is 10^{-4}. The risk factor of 10^{-4} represents 1 chance in 10,000 that an accidental death due to occupational hazards will occur during the year. The risk factors used by the ICRP are the estimated probability of occurrence of stochastic effects per unit dose, that is, the sum of the risk factor for all radiation induced cancers ($1.25\text{x}10^{-2}$ Sv^{-1}) and the average risk factor for hereditary effects ($4\text{x}10^{-3}$ Sv^{-1}). The average annual dose equivalent in cases of occupational exposure is about 5 mSv (.5 rem).[10] So, the <u>average</u> annual risk would be ($1.65 \text{ x } 10^{-2}$ Sv^{-1}) ($5\text{x}10^{-3}$ Sv) = $8.25\text{x}10^{-5}$, approximately $1\text{x}10^{-4}$ or comparable to other safe industries.

For more discussions on risk and risk factors, for both radiation workers as well as risks in everyday living, one may consult References 11 and 12, respectively.

The ICRP makes a further point in their recommendations. They recommend that all necessary exposures be kept as low as reasonably achievable, taking into account economic and social factors.[10] This part of the recommendations is referred to as optimization. That is, achieving the optimum net benefit such that the increased protection cost to reduce exposures is balanced by the reduction in the expected harm (detriment).

I. Biological Responses of Specific Organs

With respect to specific damage in various organs of the body, some effects may be stochastic or nonstochastic. For the stochastic effects, the risk factors from Reference 10 will be given. For nonstochastic effects, the threshold doses from Reference 7 are quoted. The sensitivity of the method for detecting damage is an important factor. For this reason, the threshold dose is defined as the amount of radiation required to cause a particular effect in at least 1-5% of those exposed.[7] With the exception of tissues such as skin, which respond for short-term irradiation, data is lacking for reactions to long-term protracted exposures in all organs. Knowledge of the radiosensitivity of different tissues regarding nonstochastic effects is based mainly on therapy experience. The thresholds are based on therapy irradiation conditions; typically, 20-35 exposures to x or γ rays over a 4-7 week period, not occupational exposure conditions. The following summary points out some of the specific aspects of biological response in specific tissues and organs of the body:

1. Blood and Bone Marrow

The blood is composed of three major types of cells, namely, the erythrocytes (red cells), the leukocytes (white cells), and platelets, suspended in a fluid called plasma. The red cells supply other body cells with food and oxygen, and remove waste products. White cells help to combat infections, and the platelets aid in blood-clotting action. Plasma is a viscid liquid which contains water, proteins, salts, and free ions. Blood contains about 45% red cells, approximately 1% white cells and platelets, and 54% plasma.

Bone centers are filled with marrow, either red or yellow. Red marrow is found in the skull, breastbone, ribs, pelvis, and spine of adults. The red marrow provides the blood-forming function. Yellow marrow provides fat storage.

White cells are the first to be affected by radiation. Although there are subtypes of white cells which differ in their sensitivity, the net effect of irradiation is to reduce the number of white cells. This lack of white cells is known as leukopenia. For acute whole body doses > 1 Gy (100 rad), the maximum drop in the leukocytes is seen within 2-5 weeks. The platelets drop in number somewhat more slowly. A few weeks later, a loss of red cells (anemia) occurs.

The loss of white cells affects resistance to infection. The drop in the platelet number affects clotting action, so that open wounds may not heal. Anemia causes a general weakness in the individual. Recovery will take place if the damage to the bone marrow is not too great. The marrow will regenerate and produce new blood cells to replace the cells which were lost. When the damage to the red marrow is too great, the effects are likely to be permanent. The $LD_{50/60}$ is not known precisely, but is estimated to lie in the range 2.5-5 Gy (250-500 rad).

For fractionated or protracted doses, the effects are not as severe due mainly to replacement and some repair. Although dose rate influence on damage is not well known for humans, it is believed that the blood forming system can withstand 3-10 Gy (300-1000 rad) if protracted over several months. The threshold dose for nonstochastic effects is 2 Gy (200 rad).

With respect to leukemia, the red bone marrow is believed to be the main tissue involved. The risk factor is taken as $2\text{x}10^{-3}$ Sv^{-1} ($2\text{x}10^{-5}$ rem^{-1}).

2. Lymphatic System

The lymphatic system is a network of small tubes which permeate the body tissues. A fluid called lymph, somewhat like plasma but with less proteins, drains from tissues into the lymphatic system. The lymph picks up waste products from the tissues. Along the course of a lymph vessel are oval-shaped glands (lymph nodes) which filter out foreign substances from the lymph. Thus purified, the lymph is passed back into the blood stream.

The spleen contains the largest mass of lymphatic tissue in the body. The spleen filters dead blood cells from the blood and is a source of white blood cells. Also, the spleen stores red blood cells.

The lymph nodes show the first signs of hemorrhaging and infection after acute irradiation. The spleen may exhibit weight loss and damage to lymphocytes (a subtype of white blood cells). A complication in the function of lymphoid tissues is a drop off in the body's immune response to infection.

3. Digestive Tract (GI Tract)

The digestive tract, or alimentary canal, consists of the mouth, pharynx, esophagus, stomach, and the small and large intestines. This system is often called the GI tract. In an adult man, the canal may be as much as 9 m in length. The cells which line the walls of the intestines secrete substances which act on food to make absorption into the blood stream possible. The stomach is the reservoir in which the major chemical phases of digestion occur. The radiosenstivities of the many sections of the canal vary greatly. The small intestine is quite radiosensitive, whereas the stomach and esophagus are much less radiosensitive.

The symptoms of damage to the canal are nausea and vomiting. The initial effects are impaired secretion and discontinued cell production. When cell breakdown occurs, the dead cells are released from the walls of the tract. This debris clutters up the intestine. The exposure of tissues under the surface layer may lead to ulcers. The threshold dose for ulceration is estimated as 45 Gy (4500 rad). In fatal cases, infection, failure of food absorption, and dehydration from diarrhea are the causes. As indicated earlier, the LD_{50} for acute whole body exposures is in the range 5-20 Gy (500-2000 rad).

4. Reproductive Organs

Since the reproductive organs are the source of germ cells, damage to these cells can result both in somatic and hereditary effects. For the present, our concern is only with the somatic effects. The response of germ cells to radiation differs slightly in the male and female. These cells are highly radiosensitive, while other cells of the reproductive system are relatively radioresistant. Radiation can produce sterility in both sexes, but the degree depends upon the dose delivered. In man, partial sterility can be induced at doses as low as 0.15 Gy (15 rad). Based on dog experiments, it is inferred that the human testes can tolerate 1 mGy d^{-1} (0.1 rad/d) for an indefinite period of time without fertility impairment. It requires a larger dose to produce permanent sterility in the male than in the female. Acute exposure of the ovaries to 0.65-1.5 Gy (65-150 rad) may cause prompt impairment of fertility and acute doses > 3 Gy (300 rad) will cause permanent sterility. Threshold doses for testes of 5-15 Gy (500-1500 rad), and for ovaries of 2-3 Gy (200-300) have been estimated.

Germ cells which survive damage can transmit any genetic changes caused by the radiation. For this reason, the total effect of radiation on the gonads may not be seen for several generations. The estimated risk factor for the first two generations is $4x10^{-3}$ Sv^{-1} ($4x10^{-5}$ rem^{-1}).

5. Nervous System

The nervous system, composed of the brain, spinal cord, and the peripheral nerves, acts to coordinate body activity. The spinal cord and peripheral nerves are highly radioresistant, but the brain is more sensitive than often supposed. Lesions and functional impairment have been observed for brain doses above 10 Gy (1000 rad) in adults. In children, doses from 1-6 Gy (100-600 rad) have produced detectable physical and functional changes.[8] Necrosis of the brain has an estimated threshold dose of 50 Gy (5000 rad), the damage occurring directly or through lack of blood supply due to blood vessel damage. Inflamation of the spinal cord producing numbness, tingling, weakness or paralysis can be caused by radiation damage depending on dose, irradiated tissue volume and nerve location. The estimated threshold dose is 50 Gy (5000 rad). Peripheral nerves have been damaged at doses > 60 Gy (6000 rad) during conventional radiotherapy.

6. Thyroid Gland

The thyroid, a gland located at the base of the throat, secretes a hormone known as thyroxine, which helps to control basal metabolism. The action of the thyroid seems to be closely connected with the functions of the pituitary and adrenal glands. Thyroxine contains about 65 percent iodine, and is essential for growth and development. Damage to the thyroid, or to the other two glands, have marked effects in the body.

The thyroid is radioresistant from the standpoint of external radiation. It can be severely damaged if radioiodine is inhaled, since iodine will concentrate in the thyroid. Damage causes a decrease in production of thyroxine (hypothyroidism) which produces a lower metabolism rate. Muscle tissue may then fail to absorb enough oxygen and health can be badly impaired. The threshold dose is estimated to be 45 Gy (4500 rad).

The sensitivity of the thyroid to cancer induction appears to be higher than that for leukemia induction. However, the tumors are slowly progressive so that treatment is generally successful. So, the mortality (death rate) for thyroid cancers is much lower than that for leukemia. The mortality risk factor is taken as 5×10^{-4} Sv^{-1} (5×10^{-6} rem^{-1}).

7. Eyes

The lens of the eye is highly susceptible to irreversible damage by radiation. The lens cells of the eye are not replaced by regrowth. The retina is much less sensitive than the lens.

When the cells of the lens become damaged, the cells lose their transparency. The opacities which may occur will appear after a latent period. The term cataract is applied to these lens opacities. The formation of cataracts is generally a late somatic effect. Acute effects in other eye structures occur only after high doses. At high doses, cataracts may develop within months, while at low doses, the latent period may be years.

In the early stages of development, radiation induced cataract may be distinguished from that due to other causes.[1,7] The initial opacity appears as almost a dot near the center of the lens, whereas spontaneous cataracts tend to begin at the periphery of the lens. The central opacity grows larger, developing a clear center, so that it resembles a doughnut. As it continues to progress it becomes similar in appearance to other types of cataracts, and therefore can no longer be distinguished. It is possible for the lesion to grow for a time and then remain stationary. The lowest dose observed to cause a progressive cataract in radiotherapy patients was 5 Gy (500 rads). For the case of occupational exposure, extrapolation suggests that > 8 Gy (800 rads) of low LET radiation would be necessary to produce a vision impairing opacity.

For high energy neutrons (7.5 MeV) in fractionated doses, the threshold for visual impairment appeared to be 3-5 Gy (300-500 rad). On this basis, the ICRP has recommended that no change is required for the neutron quality factor relative to cataract formation. In relation to the lens, other parts of the eye are radioresistant. In terms of the threshold dose, the estimate is 5 Gy (500 rad) with respect to cataract production.

8. Lungs

The lungs are cone-shaped organs made up of very small air sacs called alveoli. When a person breathes, the air is directed down the trachea (wind pipe). From there, two large tubes (bronchi) direct the air toward each lung section. Many small tubes (bronchioles) branch out from the bronchi to connect with the alveoli in the lungs.

During breathing, each air sac is expanded and compressed by lung muscles, and is thus filled and emptied. Air passes through the walls of the alveoli into tiny blood vessels (capillaries).

The effects produced in the lung by radiation are the result of damage to the air sacs. The lungs are not normally affected by external radiation. As in the case of the thyroid, the greater hazard occurs from internal radiation from inhaled dust and vapors. However, when a major portion of the lungs are irradiated at high dose, a fatal pneumonia may result. The LD_{50} in man for acute exposure is about 8-10 Gy (800-1000 rad) for gamma rays. The damage which occurs is to the alveoli and the lung blood vessels. The tissues of the upper respiratory tract are relatively less radiosensitive. The threshold dose for nonstochastic damage in the lung is 40 Gy (4000 rad).

Cancer of the lung has been observed in miners exposed to high radon concentrations. There is also evidence that external irradiation can induce lung cancer in man. The risk factor for lung cancer is $2x10^{-3}$ Sv^{-1} ($2x10^{-5}$ rem^{-1}).

9. Liver and Gall Bladder

The liver is radioresistant as compared with other organs. The liver, the largest gland in the body, secretes bile for digestion. The gall bladder stores and concentrates the bile secreted by the liver. When bile is needed, it passes from the gall bladder to the intestine.

External radiation is not too effective in causing damage to these organs. Most damage is caused by internal exposure from radionuclides which concentrate in the liver. Impairment of liver function occurs for exposure of the entire liver to 30 Gy (3000 rad) of conventional therapeutic irradiation. The damage is indicated by decreased liver function and fluid accumulation (ascites). The threshold dose for nonstochastic effects in liver is 35 Gy (3500 rad).

10. Kidneys

The kidneys help to control the concentration and content of the blood by excreting water and waste products. The waste products pass from the kidneys through small tubes (ureters) into the bladder (this system is called the urinary tract). Impairment of renal functions does not add to mortality in the case of total body radiation. Damage to the kidney is indicated by an increase in amino acids in the urine. These effects occur mostly from internal radiation. The appearance of blood in the urine is an indication of severe renal damage. The dose to cause injury in the system, is lowest for the kidneys, highest for the ureters. The threshold dose for a fatal kidney infection (nephritis) is estimated to be approximately 23 Gy (2300 rad) when delivered in fractionated doses. The tolerance of the bladder is higher than that of the kidney, with a threshold of 55-60 Gy (5500-6000 rad) when fractionated over 4 weeks.

11. Circulatory System

The heart and blood vessel system are damaged seriously only for very high doses of radiation. The heart is not highly radiosensitive, but can be damaged by doses in the range 40-60 Gy (4000-6000 rad). The threshold for inducing inflammation of the lining surrounding the heart (pericarditis) is 40 Gy (4000 rad). Blood vessels show damage after 40-60 Gy (4000-6000 rad). In many cases, vascular damage in advance of tissue effects suggests that this is an important factor in tissue injury.

12. Skin

The degree of skin damage varies with the dose and the species of animal. Skin is easily damaged but has a tremendous capacity for repair. Various structures of the skin show quite different sensitivities. The damage seems to be greater for less penetrating radiations.

Slight damage to the skin may result in an erythema (reddening). With increasing dose, loss of hair (epilation), dry, scaly skin and death of tissue in the epidermis (outermost skin layer) may occur. For increased damage, ulceration may result. The threshold dose for skin ulceration is estimated to be 55 Gy (5500 rad).

Skin cancer is a late effect of chronic irradiation at very high dose rates. However, skin is less likely to develop fatal cancer.[10] The ICRP feels that the nonstochastic limit for skin will prevent the occurrence of skin cancer.

13. Hair

Irradiation can lead to temporary baldness (epilation). This condition may last for a few weeks. The hair begins to return, but the new hair may have different characteristics, such as a new color.

With respect to hair follicles, an acute dose of low LET of 3-5 Gy (300-500 rad) can cause temporary epilation. The threshold for permanent epilation for acute exposure is about 7 Gy (700 rad).

14. Bones

Bone is composed of living cells which are distributed in a matrix of fibers and bone salts. Although the marrow of the bone is very radiosensitive, the bone cells, fibers, and salts are relatively radioresistant.

When radionuclides, such as strontium or plutonium, are internally deposited in the bone marrow or bone tissue, then great damage can be done. These effects again are late effects since the damage may take years to show up.

In children, the developing bone cells and cartilage show a greater response. For doses as low as 1 Gy (100 rad), some retardation of growth may be seen. Mature bone in adults can withstand fractionated doses of 65 Gy (6500 rad) in a 6-8 week period.

In adults, the developing bone cells lining the bone cavity are the radiosensitive cells at risk with respect to cancer. The risk factor is taken as $5x10^{-4}$ Sv^{-1} ($5x10^{-6}$ rem).

15. Muscle

Mature muscle is relatively radioresistant, but in children when muscle is growing, the radiation response is greater. Contraction and delayed healing show up for doses of approximately 60 Gy (6000 rad) fractionated. The threshold dose is estimated to be > 100 Gy (10,000 rad).

16. Breast

The breast in females is one of the more radiosensitive organs with respect to cancer induction.[6,10] The latent period seems to be related strongly to age at exposure. An estimate of the latent period is 5 years for women 25 years or older.[6]. The risk factor is taken as $2.5x10^{-3}$ Sv^{-2} ($2.5x10^{-5}$ rem^{-1}).

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QUESTIONS

5.1 What is the basic unit of structure in the body?

5.2 What are main component elements of the basic unit of structure in the body?

5.3 What are the two principal parts of a cell? What does each part do?

5.4 What tiny parts in the nucleus of a cell determine the hereditary traits of a daughter cell? To what are they attached?

5.5 What term is given to changes in the chromosomes or genes of cell?

5.6 Correlate the items in list A. with those in list B.

List A

a. cell division
b. bisexual cells
c. process of cell change
d. cell damage limited to the individual
e. germ line

List B

differentiation, gametes, somatic cells, hereditary cells, mitosis

5.7 Name two changes to atoms or molecules that will occur when radiation passes through living cells.

5.8 What terms are given to the fragments of molecules that result from radiation?

5.9 In what part of a cell is radiation damage most likely to be serious?

5.10 What term indicates that dose of radiation below which no effects are observed?

5.11 What term is used to indicate the different response of cells to radiation?

5.12 List some of the factors affecting or determining the radiosensitivity of cells.

5.13 Compare genetic damage with somatic damage to cells.

5.14 Define the term "median lethal dose" and indicate its symbol. What level of dose does this term represent in man?

5.15 Explain the term RBE.

5.16 How is LET of the radiation related to the biological damage?

5.17 What are survival curves?

5.18 List some of the qualitative results that apply to high LET and low LET radiation.

5.19 Under what conditions is sublethal damage of a cell repairable?

5.20 Explain the terms

a) CNS death
b) GI death
c) Bone-marrow death

5.21 List the late somatic effects.

5.22 Explain a) stochastic effects
b) nonstochastic effects

5.23 What is the basis for an acceptable level of risk?

5.24 List the three types of blood cells and the fluid that transports them through the body. Indicate the principal function of each.

5.25 Indicate the effects that occur upon loss of

a) white cells,
b) platelets and
c) red cells

5.26 Indicate three functions of the spleen.

5.27 Which part of the digestive tract is highly radiosensitive?

5.28 What is the result of a high radiation dose to the reproductive organs?

5.29 Which organ of the nervous system is the most radiosensitive?

5.30 What radionuclide presents the greatest internal hazard to the thyroid gland?

5.31 What hazard do neutrons and x ray present to the eye?

5.32 From what does the greater radiation hazard to the lungs occur?

5.33 To what does the term "epilation" refer?

5.34 What organ of the body is easily damaged by radiation, but has a tremendous capacity for repair?

5.35 Which part of the bone is the most radiosensitive?

5.36 Explain why the study of hereditary effects resulting from radiation is difficult.

PROBLEMS

5.1 The following table is adapted from the National Safety Council, Accident Facts, 1979. Calculate the risk (the probability of death per person/year) for each profession. How do these compare with cancer risk due to radiation?

PROBABILITY OF ACCIDENTAL DEATH BY TYPE OF OCCUPATION

Occupation	Number of Accidental Deaths for 10,000 Workers for 40 Years
a. Mining	252
b. Construction	228
c. Agriculture	216
d. Transportation and Public Utilities	116
e. Government	44

Answers:
- a. $6.3\text{x}10^{-4}$/person-year
- b. $5.7\text{x}10^{-4}$/person-year
- c. $5.4\text{x}10^{-4}$/person-year
- d. $2.9\text{x}10^{-4}$/person-year
- e. $1.1\text{x}10^{-4}$/person-year

5.2 According to the "absolute-risk projection model" of the 1980 BEIR Report, if 1,000,000 people, representative of the U.S. population, receive a single exposure of 0.1 Gy due to low LET radiation, there would be 766 eventual cancer deaths as a result of the radiation. Calculate the risk estimator. How does this compare with the risk factor suggested by ICRP?

Answer: $7.66\text{x}10^{-3}$ cancer deaths/person-Gy

5.3 A town's population of 60,000 is exposed to radiation due to passage of a radioactive cloud after an accident. The average gamma dose is estimated to be 0.04Gy. What are the predicted cancer deaths

a) by BEIR model above, and
b) by ICRP risk factor?

Answer: a) ~ 18 b)~ 30

SECTION 6 - BACKGROUND RADIATION AND MAN-MADE CONTRIBUTIONS

Apart from the amount of radiation which a worker may receive while he performs his work, he will also be exposed to radiation because of the very nature of his environment. All individuals are subject to some irradiation even though they may not work with radioactive substances. This natural source of exposure is often referred to as "background radiation."

Studies of the nature and origin of this source of exposure to man have revealed three main components: cosmic radiation, terrestrial radiation (which includes the radioactivities of the earth's surface, of air, of water), and, the naturally occurring radionuclides of the human body. One might add that man-made sources influence the contribution from some of these sources. Although the amount which each of these factors contributes varies with the locale, each locale will have some background levels.

The study of these factors throughout the world is of value for a number of reasons. Foremost among these is that the use of such data provides a basis or standard from which allowable exposure limits for radiation workers may be developed. In areas where the levels are much higher because of larger concentrations, knowledge may be gained about human hereditary effects at these increased levels. Such data are also needed in assessing the impact on, or contribution of a nuclear facility to, the existing concentrations in a given area. In the design of buildings and/or shielding for low level work, it is of value to know the radioactive contents of the substances used. Often the levels inside a building are higher than the outside of the building because this factor has been neglected.

Because of these needs, much data about background levels in many areas have been acquired. The rest of this section will be devoted to a discussion of these background factors and the relative contribution of man-made radiation.

A. Cosmic Radiation

Much work has been carried out in the study of cosmic radiation. This factor in background levels was discovered during attempts to reduce background. Though detection devices showed a response even in the absence of any known sources, it was assumed this background was due entirely to traces of radioactive substances in the air and ground. Thus, if a detector were elevated to a greater height above the earth's surface, the background should be greatly reduced. The use of balloons carrying ion chambers to great heights yielded data which showed the effect increased, rather than decreased. These and other data showed that radiation was really coming from outer space. The name "cosmic rays" was given to this high energy, extraterrestrial radiation.

Further study has shown that cosmic radiation consists of two parts: primary and secondary. The primary component may be further divided into galactic, geomagnetically trapped radiation, and solar.

The galactic cosmic rays come from outside the solar system and are composed mostly of positively charged particles. Studies have shown[1,2] that outside the earth's atmosphere, cosmic rays consist of 87% protons, of 11% alpha particles, and about 1% each of other heavier nuclei and electrons at latitudes above 55°. These particles may have energies in the range of about 1 GeV to at least 100 EeV (10^{20} eV).

As a charged particle approaches the earth, it is acted upon by the earth's magnetic field. In order to pass on through to the earth, the particle must have a certain momentum. Otherwise, it may be trapped by the earth's magnetic field. This gives rise to the second type of primary cosmic rays, the geomagnetically trapped radiation.

Solar cosmic rays are produced following severe solar flares on the surface of the sun. These rays consist of protons. The events are classed as high energy or low energy. The high energy events can be observed by ground-level neutron devices. The low energy events are more frequent but must be detected at high altitude. Since these events produce radiation throughout the solar system, they are of great importance in shielding design for manned space missions.

Secondary cosmic rays result from interactions which occur when the primary rays reach the earth's atmosphere. When the high energy particles collide with atoms of the atmosphere, many products result: pions, muons, electrons, photons, protons, and neutrons. These, in turn, produce other secondaries as they collide with elements or decay on the way toward the earth's surface. Thus, a multiplication or shower occurs in which as many as 10^8 secondaries may result from a single primary.

Most of the primary rays are absorbed in the upper 1/10 of the atmosphere. At about 20 km and below, cosmic rays are almost wholly secondary in nature. The total intensity of cosmic rays shows an increase from the top of the atmosphere down to a height of about 20 km. Although the primary intensity decreases, the total effect increases because of the rapid rise in the number of the secondaries. Below 20 km, the total intensity shows a decrease with height because of attenuation of the secondaries without further increase in their number due to primaries. At $<$ 6 km of altitude, the highly penetrating muons, and the electrons they produce, are the dominant components.[2]

At the earth's surface, the secondary cosmic rays consist of muons mainly (hard component), electrons and photons (soft components), and neutrons and protons (nucleonic component). At sea level, about 3/4 of the cosmic ray intensity is due to the hard component.

Because of the earth's magnetic field, cosmic ray intensity also varies with latitude. The energy which is needed for a charged particle to reach the earth's atmosphere at the geomagnetic equator is larger than that needed at other latitudes. The effect is greatest for latitudes between 15 and 50°. Above 50°, the intensity remains almost constant. Thus, the lowest value of the intensity occurs at the geomagnetic equator, and the effect is expressed as the percentage increase at 55° over that at the equator. At sea level, the effect is small for the ionizing component (10%) but is larger for the neutron component.

The dose rate produced by this source of background may be divided into two parts. The portion caused by the ionizing component is estimated from ion chamber readings. The portion caused by the neutron component is hard to measure because the dose rate depends so much on the energy spectrum of the neutrons. For the neutron dose estimates one must rely on calculations.

At sea level and high latitudes, the ionization rate I, is about 2.1×10^6 ion pairs/m^3 s (2.1 ion pairs/ccs), giving a dose rate of about 270 μGy per year (27 mrad) for soft tissue (based on 1 ion pair/ m^3 s = 4.17 pGy/s). Using the neutron calculations of O'Brien[3], the sea level dose would be increased by about 5%. Taking into account the dose variation with altitude, and the population distribution with altitude, the average yearly dose equivalent rate to the U.S. population from cosmic radiation is estimated to be 310 μSv (31 mrem).[4] This dose equivalent rate would be expected to decrease slightly with latitude and increase with altitude. For example at Denver, the yearly dose would be about 500 μSv (50 mrem). To allow for shielding from structures, a 10 percent reduction factor may be applied, given a final estimate of approximately 280 μSv (28 mrem), see Table 6.1.

B. Radioactivity of the Earth

As was pointed out before, the presence of certain small amounts of radioactivity in the soil adds to the background levels to which man is exposed. The amount of radioactive materials found in soil and rocks varies widely with the locale. The main contribution to the background is the γ ray dose from radioactive elements of the uranium and thorium series and from radioactive ^{40}K and ^{87}Rb.

In certain areas of the world, such as Brazil, France, India, Niue Island, and the United Arab Republic, much higher amounts of these substances have been found in soil and rock. In some of these areas, the average yearly γ dose exceeds 10 mGy (1 rad).

The amount of exposure one is subjected to depends upon the concentration in the soil and the type of soil. In the U.S., three broad areas have been found. These are: the coastal region along the Atlantic Ocean and the Gulf of Mexico, the Colorado Plateau region, and the remainder of the country. The yearly whole body dose equivalent rates in these areas range from 150-350 μSv, 750-1400 μSv, and 350-750 μSv, respectively. When absorbed dose rate measurements are weighted by population, and averaged over the entire U.S., the yearly average is estimated as 400 μGy (40 mrad).[5] If one applies a correction factor 0.8 for indoor reduction of the outdoor rate, and 0.8 for a screening factor for bone marrow irradiation, the result is 260 μSv (26 mrem). This result assumes Q=1.

C. Radioactivity of Air

The background which is found in air is due mainly to the presence of radon and thoron gas, formed as daughter products of elements of the uranium and thorium series. The decay of ^{238}U proceeds to ^{226}Ra. When ^{226}Ra emits an α as it decays, the gas ^{222}Rn is formed. In the thorium chain, the decay of ^{224}Ra results in the gaseous product ^{220}Rn, which is called thoron.

Since uranium and thorium are present to some extent throughout the crust of the earth, these products are being formed all the time. Since they are gases, they tend to diffuse up through the earth's surface to become airborne. In turn, the decay products of these gases attach themselves to dust in the air.

The amount of these gases in the air depends upon the uranium and thorium content of a certain area. In any given area, the weather conditions will greatly affect the concentrations of these gases. It is also common to find that the levels indoors are higher than those outdoors. This is a function of the material of the building and the ventilation rate. In mines and other underground caverns, the concentrations have been found to be quite high.

Some homes in Grand Junction and Durango, Colorado, have been found to have high radon levels. This was traced to the use of uranium mill tailings, residues rich in radium, as backfill.[2] This discovery has led to radon measurements in homes in other areas of the country. Some homes in Pennsylvania are situated on land with naturally elevated radium concentrations, giving rise to increased indoor radon levels. Rundo and Toohey[6] have made investigations of radon levels in homes in the Chicago area. Their results indicated that 6% of the homes studied had radon concentrations comparable to those found at Grand Junction. Because of the studies showing these levels of radon, and the implications of the potential population dose from this source, much more work on defining this potential problem will be carried out.

The major source of exposure from radon in air occurs when the daughter products attach themselves to aerosols and are inhaled. This leads to an internal dose to the lungs (see 6.E). As for external exposure, the external γ dose rate from ^{222}Rn and ^{220}Rn is estimated to be less than 5% of the total external terrestrial dose rate.

Among other radioactive products which are found in air in measurable amounts are ^{14}C, ^{3}H, ^{22}Na, and ^{7}Be. These are called cosmogenic radionuclides, since they are produced in the atmosphere by cosmic rays. None of these products add a significant amount to the background dose rate.

Estimates of the mean air concentrations of relevant atmospheric radionuclides can be found in Reference 7.

D. Radioactivity of Water

Depending upon the type of water supply one is talking about, a number of products may turn up. For example, sea water contains a large amount of ^{40}K. On the other hand, many natural springs show amounts of uranium, thorium, and radium. Almost all water should be expected to contain certain amounts of radioactivity. Since rain water will pick up radioactive substances from air, and ground water will pick up activity present in rocks or soil, one would expect to find some radioactivity in water throughout the world.

The chief source of dose rate from this background factor occurs as the result of uptake of these waters by ingestion. This leads to an internal exposure. Any estimate of the dose rate from this source is thus included in the estimate of the dose rate from radioactivity in the human body. The transfer of radioactive substances to the body seems to be mainly by food intake except in cases of very high water concentrations. Information concerning the concentration of radionuclides in water can be found in References 8 and 9.

E. Radioactivity of the Human Body

Since small amounts of radioactive substances are found throughout the world in soil and water, some of this activity is transferred to man by way of the food chain cycle. A number of studies have been made to try to find a correlation between the amounts in soil and that in man. Results have not shown a clear-cut relationship as yet.

In the human body, ^{40}K, ^{87}Rb, ^{226}Ra, ^{238}U, ^{210}Po, and ^{14}C are the main radionuclides of concern. Of these, ^{40}K is the most abundant substance in man. The amount in food varies greatly, so that intake is quite dependent on diet. However, variations in diet seem to have little effect on the body content.[2] The content in the body organs of man varies widely. Based on an average content of 0.2% by weight in soft tissue, 0.05% in bone, the yearly dose equivalent rate to the gonads is estimated to be 190 μSv (19 mrem); 150 μSv (15 mrem) to bone surfaces; and 150 μSv (15 mrem) to bone marrow. ^{87}Rb contributes only a few percent of these values.

Most of the ^{226}Ra which is taken into the body will be found in the skeleton. Much data has been gathered on the concentration in humans, and the present assumed average skeletal concentration is taken as about .29 Bq/kg (7.8 pCi/kg). The skeletal content of ^{228}Ra is taken as .14 Bq/kg (3.8 pCi/kg). The yearly dose rate produced by these components is estimated to be 5 μSv (.5 mrem) to the gonads, 146 μSv (14.6 mrem) to bone surfaces and 22 μSv (2.2 mrem) to bone marrow.

Based upon an average concentration of ^{238}U of 0.26 Bq/kg (approximately 7 pCi/g) in bone, the estimated annual doses in man are

48 μSv (4.8 mrem) to bone surfaces and 9 μSv (0.9 mrem) to the marrow. From the estimated content in the gonads, the annual dose equivalent is estimated to be about 10 μSv (1 mrem).

Similarly, the ^{210}Po chain contribution is taken as 2.22 Bq/kg (60 pCi/kg), yielding annual dose equivalents of 240 μSv (24 mrem) to bone surfaces and 49 μSv (4.9 mrem) to bone marrow. The soft tissue concentration is taken as 0.111 Bq/kg (3 pCi/g), but is about twice that in the gonads.[2] This gives an annual gonad dose equivalent of 60 μSv (6 mrem).

The average whole body content of carbon is taken as 23%. However, ^{14}C is present in normal carbon only to a very small extent ($^{14}C/^{12}C \sim 10^{-12}$), so that only a small amount of ^{14}C is present. The annual average dose equivalent turns out to be about 10 μSv (1 mrem) total body. In soft tissue, the annual dose is 7 μSv (0.7 mrem). The annual dose to the bone surfaces is 8 μSv (0.8 mrem), and to the bone marrow, 7 μSv (0.7 mrem).

Table 6.1 gives a summary of the present estimates for the dose equivalent rates from all background sources. The data for this table was adapted from NCRP 45 (Reference 2). The values in parentheses are corrected values supplied by J. H. Harley (Reference 10) to account for the increase of Q from 10 to 20 for α particles and indoor exposure to radon.

F. Fallout

The term "fallout" has been applied to debris which settles to the earth as the result of a nuclear blast. This debris is radioactive and thus a source of potential radiation exposure to man. Since fallout is a man-made source of irradiation, it does not properly come under the title of natural background. However, since fallout can alter the contribution to other background sources, it will be treated in this section.

Table 6.1

Summary of Average Annual Dose Equivalents in μSv*
from Natural Background Radiation in the U.S.

(Adapted from NCRP Report No. 45 - Natural Background Radiation in the United States)

	Gonads	Bone Surfaces	Bronchial Epithelium
Cosmic Radiation	280	280	280
Cosmogenic Radiation	10	10	10
External Terrestrial	260	260	260
Inhaled Radionuclides			4500 (30000)
Radionuclides in the Body	270 (350)	60 (105)	240 (400)
Rounded Totals	800 (900)	120 (160)	5000 (30000)

*To obtain mrem, divide μSv by 10.

1. Production of Fallout

Because of the intense heat produced in a nuclear explosion during a very short time, matter which is in the vicinity of the bomb is quickly vaporized. This includes fission products formed in the fission process, unused bomb fuel, the bomb casing and parts, and, in short, any and all substances which happen to be around. These are caught in the fireball which expands and rises very quickly. As the fireball cools and condensation occurs, a mushroom shaped cloud is formed, containing small solid particles of debris as well as small drops of water. The cloud continues to rise to a height which is a function of the bomb yield and the meteorological factors of the area. For yields in the megaton range (1 megaton equals an energy release equivalent to 10^6 tons of TNT), the cloud top may reach a height of 40 km.

The fallout which occurs may be described as local or worldwide. The portion of debris which becomes local fallout varies from none (in the case of a high altitude air burst) to about half (in the case of a contact surface burst). The height at which the bomb goes off is thus quite important in the case of local fallout. If the fireball touches the surface of the earth, it will carry aloft large amounts of surface matter. Also, because of the suction effect created by the rapid rise of the fireball, other matter may be taken up into the rising fireball. This leads to the formation of larger particles in the cloud that tend to settle out quickly. If the wind is not too great, the fallout pattern will be roughly a circle about ground zero. Ground zero is the point on the surface directly under, at, or above the burst.

Other bits of matter will fall out at various stages. The distance from ground zero at which they strike the surface and the time it takes depend upon the height from which they fall, their size, and the wind patterns at all altitudes. This results in a cigar-shaped pattern downwind of the burst point. Local fallout usually occurs within the first 24 h after the blast.

If the height of the burst is such that the fireball does not touch the surface, the debris is carried aloft and dispersed into the atmosphere. This matter then descends to earth at a later time and is called world-wide fallout.

The residence time of this debris is a function of the bomb yield. For yields in the kiloton range, the debris is not projected into the stratosphere. It is limited to a region called the troposphere, between about 9,000 to 17,000 m. In this region, there is quite a bit of turbulence as well as precipitation. The debris is removed rather quickly from about one day to one month.

If the burst is in the megaton range, the debris is carried into the stratosphere. In this region little mixing will occur, and the absence of rain or snow prevents this matter from being washed down. The time that it takes for this debris to return to the troposphere and be washed down varies. It is a function of both the height in the stratosphere to which the debris is lifted and the locale at which the burst occurs. It may take up to 5 years or more for this debris to return to earth. On the other hand, for bursts in the northern hemisphere in which the debris is confined to only the lower part of the stratosphere, the half-residence time is thought to be less than one year. The concept half-residence time is the time for one-half of the debris to be removed from the stratosphere.

In all, there are more than 200 fission products which result from a nuclear blast. The half life of each of these products covers the range from a fraction of a second to millions of years. Local fallout will contain most of these products. Because of the time delay in the appearance of world-wide fallout, only a few of these products are important from that standpoint. Since local fallout is confined to a relatively small area, its effect on the human population can be negated by proper choice of test sites, weather conditions, and type of burst. The fallout of interest from the standpoint of possible effects on man due to testing is the world-wide fallout.

2. Effects of World-Wide Fallout

A number of factors must be considered when one attempts to assess the hazard from world-wide fallout. Because of the associated time delay before world-wide fallout shows up, many fission products and activation products decay out in transit. Others, because they are produced in such small amounts, are diluted so that they do not produce much of an effect. Also, once the fallout does arrive, to be of importance internally, there must be a transfer to the body and absorption into the body organs. All these factors combine to limit the number of fission products which may have an effect on man. The main contribution comes from ^{90}Sr, ^{137}Cs, ^{131}I, ^{14}C, ^{3}H with minor contributions from ^{85}Kr, ^{55}Fe and ^{239}Pu. Although the U.S. ceased atmospheric testing in 1962, the inventory of fission products from previous bursts has committed man to future doses. The concept dose commitment refers to the integral dose over a period of time which results from a certain practice. In the case of fallout, Table 6.2 shows the projected U.S. population dose from this source, through the year 2000.

Table 6.2 Projected Annual WB Dose Equivalent to U.S. Population From Fallout*

Year	Per Capita H, μSv**
1963	130
1965	69
1969	40
1980	44
1990	46
2000	49

*Adapted from Reference 4.
**Divide value by 10 to find mrem.

G. Nuclear Facilities

By April 1979, 70 nuclear power plants were licensed in the U.S. with the number expected to continue to increase.[4] In addition, over 300 other reactors, classed as non-power reactors, are being operated. In order to provide fuel for these reactors, mining and milling of uranium ore is carried out and fuel fabrication plants are operating. There are several hundred mines, 20 uranium mills and 21 fuel fabrication facilities. Discharge of radionuclides from these facilities is controlled by federal regulations.

Three radionuclides released during routine operations, which contribute to the population dose, are ^{3}H, ^{14}C and ^{85}Kr. Current estimates of the yearly average dose equivalent in the U.S. from environmental releases is < 10 μSv (1 mrem). Projections of the annual dose rate from this source to the year 2000 can be found in Reference 4.

H. Consumer Products and Miscellaneous Sources

In NCRP Report 56[10], a number of consumer products and miscellaneous sources of radiation exposure to the U.S. population are discussed. In general, two groups of sources have been found:

1. Those in which the dose equivalent is relatively large and many people are exposed,

2. Those in which the dose equivalent is small but many people are exposed or the dose equivalent is large but only a few people are exposed.

Such products as television sets, luminous-dial watches, smoke detectors, static eliminators, tobacco products, airport luggage inspection systems, building materials and many other sources have been studied.[12] Table 15 of Reference 11 contains estimated exposures from the individual sources studied. The estimated annual average whole body dose equivalent to the U.S. population from consumer products is < 50 μSv (5 mrem). The major portion of this exposure (approximately 70%) is due to radioactivity in building materials.

Another source of radiation exposure to the public arises from the wide use of coal. Coal contains ^{14}C, ^{40}K, uranium and thorium and when burned, the resulting flyash released to the atmosphere carries some of this radioactivity with it. This leads to inhalation of airborne flyash producing lung exposure. The dose equivalent rate in the vicinity of one of these plants has been estimated to be in the range 2.5-40 μSv/y (.25-4 mrem/y).[11]

I. Medical and Dental Exposures

The exposure to the U.S. population from x rays used in medical and dental procedures is the largest source of man-made radiation. It is estimated that more than 300,000 x ray units are in use in the U.S., and that about 2/3 of the U.S. population is exposed. In 1970, the estimated annual average bone marrow dose equivalent from dental and medical and dental x rays to the U.S. population was about 780 μSv (78 mrem).[4]

In addition to the exposure from x rays, nuclear medicine programs use radiopharmaceuticals for diagnostic purposes. Radiologists also use radionuclides for therapy treatments. It has been estimated that more than 10 million doses are administered each year. Of this, about 81% involve the use of ^{99m}Tc, and 15% involve ^{131}I, ^{133}Xe, ^{67}Ga and ^{123}I. The estimated annual average bone marrow dose equivalent from this source is about 140 μSv (14 mrem).[4]

A summary of the estimated dose rates from significant sources of radiation exposure in the U.S. can be found in Table III-23 of Reference 4.

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QUESTIONS

6.1 What term is given to the radiation one receives from his environment?

6.2 What are the three main components or sources of natural radiation to which man is exposed?

6.3 What artificial component or source of environmental radiation affects man throughout the world?

6.4 Why may the levels of radiation exposure be greater inside of some buildings than in open spaces?

6.5 What are the three sources of the primary component of cosmic rays and what do they consist of?

6.6 Explain the origin of secondary cosmic rays.

6.7 Why does the cosmic ray intensity vary

a. with altitude and
b. with latitude?

6.8 What portion of the cosmic ray background is difficult to assess from the standpoint of dose rate?

6.9 Identify the principal radioactive chains or materials found in rocks and soils which are the main contributors to the earth's background radiation.

6.10 Identify by name and origin the principal radioactive gases found in the atmosphere. Of what chemical element are they isotopes of?

6.11 What are cosmogenic radionuclides found in air in measurable quantities?

6.12 What are the radionuclides of concern found in the human body? Which of these is most abundant?

6.13 What term is used to identify the debris which settles to earth as the result of a nuclear blast?

6.14 Indicate some of the factors which affect

a. the amount of fallout and
b. the distribution of fallout.

6.15 Why does fallout remain longer in the stratosphere than in the troposphere?

6.16 What does the time for one-half of the fallout debris from a nuclear blast to return to earth define?

6.17 Name the fission products which, as fallout, will probably have the greatest effect on man. Which will be of concern for only a short time?

6.18 What is required before the fission products from world-wide fallout can have any material effect on the body?

6.19 What nuclides are released during routine operations of nuclear power plants?

6.20 List some of the consumer products and miscellaneous sources of radiation exposure to the population.

SECTION 7 - RADIATION PROTECTION STANDARDS

A. History of Protective Standards

The task of setting exposure limits is both a vital and yet a very difficult undertaking. It is vital because workers must be protected from the harmful effects of ionizing radiation. It is difficult because of the many factors which enter into the effects which radiation produces (see Section 5). Even though a vast amount of data has been gathered and studied, there are still many areas where much work is needed before firm conclusions can be drawn. Nevertheless, in order to advance in the field of nuclear energy, men must work with radiation. Thus, certain levels must be set which will protect workers from undue exposure.

Because there are still several unknowns which must be evaluated, the setting of limits involves judgments which cannot be wholly based upon the present body of scientific knowledge. For this reason, the concept of an "acceptable risk"[1] is used. In other words, the benefits are weighed against the potential damage and then limits are set at some level at which the most benefit to mankind will accrue. However, since all exposure is assumed to involve risk to the individual, exposures should always be kept as low as practicable. This implies that efforts be continually directed toward improving performance, techniques and safety designs to reduce exposures.

From time to time, these limits will be revised as new knowledge is gained. When some of the assumptions can be replaced by facts, then it becomes prudent to review the limits and perhaps make firmer recommendations. The whole history of the development of exposure limits points out this feature of re-evaluation in the light of current knowledge.

With the discovery of radioactivity and a consequent intensive investigation of the phenomenon, many people were subjected to very high dose rates, and it did not take long for deleterious effects to become manifest. As early as 1897, cases of skin damage began to appear.

1. Erythema Dose

Early efforts at control were hampered by a lack of quantitative methods. There were no units by which one could assess the amount of radiation. No one even knew what was how much, let alone, too much radiation! As a result of the use of radiation by doctors in treating patients, a unit called the erythema dose came into use. This was a highly qualitative unit; defined in terms of the amount of radiation which would produce a well-defined reddening of the skin. It soon became apparent that this dose unit was not at all satisfactory. It varied not only with the type of radiation and the dose rate, but also with the response of different parts of the body. Thus, two people could receive the same supposed fraction of an erythema dose, yet one might show skin effects and the other none. This lack of a certain value for this unit made protection work more or less of a trial-and-error process.

Around 1914, radiation began to be used in industry. The radium dial-painting process came into being, and x rays were found useful for showing up flaws in materials. Larger numbers of people were now being exposed. No longer could the vague notion of erythema dose serve the purpose of a protection standard. Yet progress toward better standards still lagged because of lack of knowledge of the many complex factors which enter into radiation effects.

2. ICRU, ICRP, and NCRP

In 1925, at the First International Congress of Radiology, the International Commission on Radiological Units and Measurements (ICRU) was formed, mainly because of the lack of a suitable dosage unit of international acceptance.[2] In 1928, this group adopted the definition of an international unit, the Roentgen. For the first time measurements throughout the world could be made in terms of the same unit. Over the years the ICRU has been the main force in defining and adopting units for use on an international basis (see Section 4).

At the Second International Congress of Radiology in 1928, the first international body concerned with protection standards was formed.[3] At first known as the International X ray and Radium Protection Commission, this group is now called the International Commission on Radiological Protection (ICRP). This group discusses and reviews basic protection principles, and these recommendations then serve as a guide from which regulations can be drawn up by each country to suit its needs. Although this group acts only as an advisory board, it has had a tremendous impact on the field of radiation protection.

In 1934, the ICRP made its first recommendation of a tolerance level of exposure: 0.2 R/day. This limit remained in force until 1950. However, because of World War II, the ICRP did not meet between 1937 and 1950. This left much of the study of protection standards during this time to the national committees.

In this regard, one cannot help but mention the work done by the National Committee on Radiation Protection and Measurements (NCRP). This group was formed in the United States in 1929.[4] The work of this body was coordinated by the National Bureau of Standards. The early recommendations of the Committee appeared in the National Bureau of Standards Handbooks. The NCRP recommendations as outlined in Handbooks 20 and 23, which have been superseded by later reports, served as the basis for protection practices during the days of the Manhattan project. This was the name given to the project developing the atomic bomb. Many members of the NCRP were engaged in this program and were helpful in seeing that protection standards prevailed.

From the standpoint of protection problems, it is hard to believe the dramatic impact that the war years produced. Of course, most of this effect can be traced to the development of the atomic bomb. Before the war, most of the problems concerned rather low energy x rays. Now, not only were there these to treat, but also other types of radiation with a wide range of energies. Added to this was the large increase of workers in the radiation field. Also, many new techniques and operations became a topic of real concern. New units would be needed to define the dose contributed by radiation other than x rays. Large amounts of waste were now produced and methods of disposal would have to be worked out. With reactors in use, not only the workers, but also others not connected with the work, would have to be considered. The scope of the radiation field had enlarged to an undreamed of extent.

The NCRP met in 1946 to reorganize. At this time a number of subcommittees were formed to deal with the new problems more effectively. This resulted in the publication of a number of handbooks after the war which represented changes and additions to the old recommendations. The Committee was replaced by a non-profit corporation chartered by Congress in 1964 and is now known as the National Council on Radiation Protection and Measurements.[5] The Council is the successor to the Committee and was formed to carry on the work begun by the Committee.

The Council is made up of the members and the participants who serve on a number of committees. These committees develop proposed recommendations on various aspects of radiation protection and radiation measurements, which when approved by the Council, are published as NCRP Reports. The initial report issued by the Council was NCRP Report No. 32.[6] The current NCRP recommendations on limits of exposure to ionizing radiation can be found in Reference 5.

The three organizations, ICRU, ICRP and NCRP, have figured prominently in the development of present day radiation protection practices. Although these bodies act as advisory boards only, much of the radiation protection philosophy which has evolved and which has been adopted by various regulatory agencies throughout the world, had its origins in the recommendations of these organizations.

B. Radiation Exposure Concerns

Over the years, the development of standards for radiation protection has evolved through several phases. Initially, the concern was for the protection of patients and medical personnel from external radiation from the use of x rays for diagnosis and therapy. World War II produced a shift in emphasis due to the increase in the number, type and uses of radioactive materials. This introduced considerations about internal exposure and the dose to the general public. Finally, a concern over the potential genetic effects of radiation and the impact of long-term exposure at low dose rates emerged.

Data from biological studies seemed to indicate that one could not assume that all effects had a threshold dose. Also, in the case of gene damage, effects could be expected at very low doses. This implied that any dose, no matter how low, carried a certain risk of deleterious effects.

Efforts have been directed toward quantifying the risk associated with a certain level of exposure. If one assumes a non-threshold relationship, then any dose carries some risk of producing damage (see Section 5.H). Under this assumption, all exposure should be kept at the lowest practical levels. However, several factors need to be considered. Namely, the information available for the quantification of risks is imperfect so that our knowledge of the absolute value of the risks involved is not that complete. In addition, the assumptions of a risk by an individual, in general, presumes that he is willing to chance the risk in exchange for some resultant benefit, which, ideally, exceeds the risk. Then, the resultant benefit which accrues, in a sense, justifies the risk. However, the resultant benefits in the case of radiation exposure are also poorly known. Therefore, the balancing of risk versus benefit in order to obtain a net benefit is not easily accomplished. For this reason, the prudent approach, adopted by both the ICRP[7] and the NCRP[5] is to keep exposures as low as reasonably achievable (ALARA).

Down through the years since the discovery of radiation, one can see the care and concern with which the problem of radiation protection has been approached. Back in the early days, the main problem was the gross somatic effects. Now, the main concern has switched from these blatant effects to the more subtle effects of radiation. As knowledge has been gained, it has become quite evident that more knowledge is needed.

In any case, the quest for knowledge in this field has not suffered and more and more groups have joined in the search. In addition to the work of the ICRP, NCRP, and ICRU, the National Academy of Sciences-National Research Council has undertaken the study of biological effects.[8] This group consists of a large number of scientists throughout the country. The reports issued by this body are in summary form and the group functions as an advisory body. Its purpose is to supply technical information as a basis from which regulations can be developed. On a world-wide scale, the United Nations has established a Scientific Committee. Their reports[9,10] on the effects of atomic radiation has helped to supply much needed background information.

The results of continuing reviews of biological data have revealed two types of radiation effects. Those for which a practical threshold dose for occurrence can be demonstrated and those for which there is apparently no threshold.[1,8] As pointed out in Section 5.H, these are referred to as nonstochastic and stochastic effects, respectively. Nonstochastic effects can be prevented by limiting the dose to the individual to a value below the threshold dose for occurrence of the effects. Since stochastic effects presume that there is no threshold level, and that the probability of the effect occurring increases with dose, any dose represents some probability of producing that effect. For stochastic effects, one can only limit the probability of occurrence to some level (deemed acceptable) by limiting the radiation exposure. The ICRP has based its recommendations for a system of dose limitation on the features discussed above.

C. ICRP Basic Recommendations

From time-to-time, the ICRP has altered and updated its basic recommendations. In its current reports,[11,13] the ICRP recommends a basic system of dose limitation which includes these three interrelated aspects:

(1) No practice shall be adopted unless its introduction produces a positive net benefit.

(2) All exposures shall be kept ALARA, economic and social factors being taken into account.

(3) The dose equivalent to individuals shall not exceed the recommended limits.

Much of the following discussion with respect to the above aspects is based upon material contain in References 11 and 13.

1. Justification

In order for a given practice involving radiation exposure to be acceptable, one should weigh the various cost-benefit factors. To be justified, a net benefit should result. This is referred to as the justification of the practice. However, as stated previously, both the benefits which may accrue and the costs which the exposure incurs are not always that well known. So, in practical situations, one must attempt to analyze all the known merits and harms associated with a practice, and also consider alternatives in those cases in which it is necessary to decide on the issue of justification. Obviously, only some of the relevant factors which influence cost-benefit considerations are known in a given situation. In addition, not all of the factors which should enter into the decision-making process, involve radiation protection concerns exclusively. As in the previous recommendations, all unnecessary exposure is unjustifiable and should be avoided.

2. Optimization

The basic recommendation that all justifiable exposures be kept as low as reasonable achievable (ALARA) is referred to as the optimization of radiation protection. As indicated above, the Commission identifies two specific considerations which are to be taken into account-economic and social. So, the recommendation requires an effort be made to reduce even justifiable exposures to a level such that any further reduction in risk (due to lowered dose) would not be considered reasonable based upon the economic and social considerations. Implicit in optimization is the reduction initially of all exposure which does not require considerations of economic or social cost. Then, one must look at the increase in the degree of protection relative to the risks of a given decreased level of exposure. At some point, the expense of supplying the necessary added protection to further reduce the radiation exposure may outweigh the further reduction in risk. One is then concerned with the incremental benefit in performing an operation at one level of exposure rather than another. The question is then whether the cost and effort are reasonable.

The comparison of the costs to health from exposure to radiation with certain economic and social factors involves some highly subjective value judgments.[7] Since neither the costs nor the benefits can be quantified with the desired precision, the optimization of radiation protection is a complex process which sometimes results in only crude approximations. In its report on optimization[14], the ICRP presents procedures and techniques for estimating the optimization of radiation protection for a variety of applications. Some of the techniques are drawn from operational research, some from economics and some from engineering. In all cases, the use of a given technique implies value judgments about the objectives of optimization. Reference 14 discusses in some detail the techniques based on cost-benefit analysis in the optimization of radiation protection.

a. Application to Protection Measures

Optimization of radiation protection can be applied to all situations wherein radiation exposures can be controlled by protection measures. These situations include design of protection systems, performance of operations, improving radiation measurements, training personnel, establishing radiation control programs, and providing support personnel, equipment and facilities.[15-18] In design considerations, the techniques tend to be more quantitative, since one may reasonably estimate such things as necessary shielding, degree of containment, efficiency of air cleaning, etc. The costs associated with such protection systems can be readily compared to the degree of dose reduction which results. Techniques used for optimization during operations may be less quantitative. Such things as decisions regarding the number and qualifications of personnel, type and extent of personnel protective equipment, the approach or procedures in carrying out work and the level of monitoring to provide, may not lend themselves to quantitative analysis. However, the overall effect of a number of such decisions regarding the above factors may be a reduction in the dose to personnel. Sometimes, one finds that increased protection in one area merely shifts exposures from one group to another. In order to qualify as ALARA in such practices, the collective dose (the sum of the products of the number of individuals receiving a dose and that dose) must decrease. For example, suppose a job is to be done in a high radiation field for which one worker would be exposed to 60 mSv (6 rem). If one uses a group of 6 workers to perform the job, each receiving a 10 mSv exposure, then the collective dose would be (6 men)(10 mSv) = .06 man-Sv (6 man-rem). Since this collective dose would be the same as that received by the one worker, you have not reduced the exposure in terms of the risk which the operation represents.[19] However, in the case of the third factor mentioned above, one must also limit the dose the individual receives. Since the limit is 50 mSv in a year (5 rem per year), one also needs to limit the exposure received by an individual in this one operation. Now, suppose that a crew of workers is called in to decontaminate the equipment and reduce the high radiation field. In performing the decontamination, two workers receive 10 mSv (1 rem) a piece. Then, the group of 6 workers performs the job, each workers receiving 4 mSv (.4 rem). In this case, the collective dose is (2 men)(10 mSv) + (6 men)

(4 mSv) = .044 man-Sv (4.4 man-rem). By this technique, the ALARA philosophy has been demonstrated. This example has been simplified since in many situations decisions must be made without prior knowledge of dose reductions that are attainable. Often, the decision involves complex components: the radiation dose, cost and components of the system needed for a given protection level, availability of suitable trained workers and the impact on the working procedures. Generally, any effort to increase the level of radiation protection implies a cost for society in terms of work (manpower, energy, other resources), time, materials and even other risks.

b. ALARA Program

The concept of optimization of radiation protection has been embraced by regulatory agencies (USDOE, USEPA, and USNRC among others) and these agencies have provided guidance for achieving ALARA in the operation of facilities.[15-17] Each of these guidance documents calls for the establishment of an ALARA program at a given facility and emphasizes the importance of management commitment to the ALARA philosophy if the program is to be effective. A number of the features of a good ALARA program are discussed below, see References 15-17 for a more complete discussion.

1) Education and Training

Training, both classroom and on-the-job, must be provided for radiation workers in order to increase their level of skill. As they become more proficient in handling radionuclides, and become increasingly familiar with their operations, the radiation exposures from specific tasks will tend to decrease. The concept of reducing exposure (ALARA) must be continually promoted. As processes or operations change, and new equipment is introduced, periodic retraining is also called for.

In addition to the workers at the facility, the radiation protection personnel must also be trained and the ALARA philosophy instilled in their protection approach. As in the case of radiation workers, training must be both on-the-job and in the classroom. To keep personnel current with operational changes, they should be retrained with a frequency of 1-2 years. Continuing education opportunities need to be provided for the radiation protection staff.

To promote training activities among its contractors, the Department of Energy supports the activities of the Training Resources and Data Exchange (TRADE) Executive Committee. Trade activities are designed to increase communication and exchange of ideas, information and resources in the field of human resource development. Among publications available from TRADE are inventories which contain descriptions of training courses. These courses involve technical and nontechnical training, computer-based training, health and safety, hazard communication and radiation protection training (workers, radiation protection technicians and other technical personnel). More can be learned about the TRADE activities and training materials through Reference 20. Of particular relevance in the area of health physics training for DOE facility personnel are two recent TRADE publications: Radiation Protection Training Resources Guide[21] and Training Materials for Draft DOE Order 5480.11 Survey Report.[22] The former contains information on available training resources in the DOE system, while the latter deals specifically with material useful for meeting the requirements of DOE Order 5480.11.

2) Design Considerations

Both in the planning of new facilities and modifications of existing facilities, the design goals should be to provide protection so that resultant exposures are well below the radiation limits. The DOE[18] requires that new designs limit exposure to a design goal of 1/5 of the limits. With respect to design to minimize or avoid internal exposures, containment, ventilation control systems and air cleaning should be designed to assure that any release will produce concentrations well below the DAC (the air concentration given in DOE Order 5480.11). In the past, large numbers of exposed personnel received exposures during maintenance operations. Design should address ease and convenience of maintenance.

Part of the health physics responsibility is to review designs with specific attention to ALARA considerations. If radiation control approaches, or control equipment can be included in the design early, money can be saved and ALARA goals can be achieved. The recognition of potential design flaws, which could develop into significant hazards, is also a critical part of the design review. Consideration should also be given to the location of equipment and the implications with respect to potential exposure to maintenance personnel. The proper storage and handling of radioactive wastes must also be reviewed.

3) Operational Health Physics Program

There are several aspects of an operational health physics program which can be utilized with respect to ALARA. One of these aspects is design review, which has been discussed above. Another is the training function-the presentation of orientations to all employees and the provision of training for radiation workers, general employees and health physics personnel.

The routine measurements taken during a radiation control program can be the basis for assessing progress with respect to ALARA principles. By analyzing survey results one can focus on the particular operations for which additional dose reduction procedures are indicated. A program of exposure control can be audited to determine what improvements have been made. It may turn out that little improvement has been made and that other efforts will be needed to pinpoint the source of major exposure. In this case, one may proceed with further investigations into the problem.

One of the most important ways in which the HP program can bring about ALARA principles is through the operation review or review of radiation work permits. In this review, the health physicist meets with the individuals responsible for the routine operations or experiments, to discuss the operation and the health physics concerns. This preplanning can greatly influence the subsequent exposures received during an operation. It tends to nullify the repetition of mistakes, since one can draw on past experience to point out errors and correct them. In addition, the exchange of ideas and discussion of concerns can lead to improvement in techniques and a subsequent saving in the exposures which occur during the actual experiment. The revision of a course of action, pointed out as necessary in the operation review, can lead to significant exposure reduction. Sometimes, just the existence of survey results from previous measurements can be helpful in planning reduced exposures for a new experiment or setup.

In addition, the analysis of survey results with respect to a given operation can lead to conclusions regarding the source of radiation exposure and may give a clue as how to correct the problem. In short, all documentation generated by radiation protection surveys may provide insight into the achievement of ALARA.

Finally, the documentation produced by the surveys, investigations and discussions of operations may serve to suggest certain improvements in the program as well as suggest possible ALARA goals to be pursued.

3. Dose Limits

The third factor in the ICRP dose limitation system is referred to as the dose limits for individuals. The ICRP believes that, for stochastic effects, the dose equivalent limit may be based on the total risk of all irradiated tissues. So, the system sets a single limit for uniform irradiation of the whole body and a weighting system to ensure that the total risk from partial body irradiation does not exceed the risk from uniform whole body irradiation.[11] In addition, no single tissue should receive more than the dose limit to prevent nonstochastic damage.

The ICRP recommendations are intended to limit somatic effects in the individual, hereditary effects in the individual's immediate offspring, and somatic and hereditary effects in the population as a whole. For any organ, the dose limitation refers to the sum of the annual dose equivalents from external sources and the committed dose equivalents from internal sources during that year.

a. Occupational Limits

The Commission believes that nonstochastic effects will be prevented by applying a dose equivalent limit of 0.5 Sv (50 rem) in a year to all tissues except the lens of the eye, for which the recommendation is 0.15 Sv (15 rem) in a year.[23] In addition, the limits apply whether the tissues are exposed singly or in combination with other organs.

For stochastic effects, the annual dose equivalent limit for uniform irradiation of the whole body is 50 mSv (5 rem). If the irradiation is non-uniform, the system is based upon the premise that the risk should not exceed that for uniform whole body irradiation. This obtains if

$$H_{WB,L} \geq \Sigma_T w_T H_T \qquad 7.1$$

in which $H_{WB,L}$ is the annual whole body limit (50 mSv), H_T is the annual dose equivalent in a given tissue T, and w_T is a weighting factor. This weighting factor expresses the ratio of the stochastic risk in tissue T to that in the total body. Values of w_T are given in Table 7.1. The summation on the right side of equation 7.1 is called the effective dose equivalent, H_E.[13]

Table 7.1 - Weighting Factors for Stochastic Risk[11]

(Reprinted with permission from Annals of the ICRP 1, No. 3, ICRP Publication 26, Recommendations of the ICRP, Copyright 1979, Pergamon Press, Ltd.)

Tissue	w_T
Gonads	0.25
Breast	0.15
Red bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bone surfaces	0.03
Remainder[1]	0.30

[1] w_T = 0.06 is applied to each of the five organs of the remainder receiving the highest dose equivalents, and the remaining tissues can be neglected.

The Commission does not intend that the hands and forearms, the feet and ankles, the skin and the lens of the eye be included in the determination of the remainder in Table 7.1.

b. Members of the Public

The recommendation for individual members of the public is an annual effective dose equivalent limit of 5 mSv (0.5 rem). This is applicable to short-term irradiation. For the case of long-term exposure, the recommendation is to restrict the dose equivalent limit to 1 mSv per year (0.1 rem per year).

The recommended annual dose limit in the case of non-stochastic effects is 50 mSv (5 rem).

Although the ICRP functions only as an advisory body, their recommendations have generally been adopted and applied as the basis for the radiation protection standards in use throughout the world. The NCRP has also endorsed these recommendations.[5]

D. Federal Policy on Radiation Matters

Because of the scope of the nuclear energy field in this country, the Federal Radiation Council (FRC) was formed in 1959 (Public Law 86-373). This body advised the President concerning radiation matters and provided guidance for all Federal agencies in setting standards and in working with the States.[24] While in existence, the Council issued a number of staff reports (8), two of which are cited (References 25 and 26). The FRC was abolished by Reorganization Plan No. 3 in 1970, and its responsibilities were transferred to the newly formed U.S. Environmental Protection Agency. The Office of Radiation Programs (ORP) of the USEPA took over the activities of the FRC.

While in existence, the FRC provided the basic general philosophy of the Federal policy on radiation matters. This guidance was contained in their first two reports.[25,26] Each Federal agency had the responsibility to determine specific regulations in its area of jurisdiction. In some cases, the guides could be exceeded but "...only after the Federal agency having jurisdiction over the matter has carefully considered the reason for doing so in light of the recommendations in this staff report."[25,26]

The recommendations of the FRC were approved in 1960 and formed the basis of the Federal radiation protection guidance. In 1981, the EPA drafted proposed revised recommendations in the Federal Register[27] regarding occupational exposure, and solicited comments. Following review of the comments, and discussions during an interagency review, the conclusion was reached to revise the previous Federal guidance. The EPA believes that it is appropriate to adopt the general features of the ICRP approach in radiation protection guidance for use by Federal agencies for occupational exposure. The revised EPA guidance was approved and issued in January 1987.[28] The recommendations replaced portions of the previous guidance which applied to workers exposed to ionizing radiation but did not change the previous guidance for exposure of the general public.

With respect to the Federal policy concerning radiation protection for diagnostic x rays, the Bureau of Radiological Health of the U.S. Department of Health and Human Services has developed a set of recommendations which serve as the radiation protection guidance.[31]

E. Regulating Agencies

So far, our attention has been directed to those groups which supply recommendations for exposure levels and safe practices. The rest of this section will be concerned with the organizations which are charged with developing regulations. Of prime interest will be those groups which regulate radiation matters in this country.

Under the Atomic Energy Act of 1954, the United States Atomic Energy Commission (AEC) was given the responsibility of regulating the atomic-energy industry. The Act authorized the AEC to set up a licensing program to be augmented by whatever rules or regulations are deemed appropriate. The bases for these rules are: to protect the public health and safety, and provide for national defense and security. Under this mandate, the AEC was concerned with the development of regulatory safety standards.

The Energy Reorganization Act of 1974 abolished the AEC and established two agencies to perform the functions of the AEC. The U.S. Nuclear Regulatory Commission (NRC) has taken over the licensing and regulatory functions. Licensed material under the control of the NRC includes source material (uranium and thorium or ores containing .05% of these materials), special nuclear material (plutonium, U-233, U enriched in U-233 or U-235), and by-product material (radioactive material resulting from producing or utilizing special nuclear material). The regulations of the NRC are set forth in the Code of Federal Regulations, Title 10. Part 20, Standards for Protection Against Radiation,[29] deals specifically with the regulations for control of radiation hazards by the licensee. Other parts of Title 10 deal with licensing and regulatory requirements associated with the use of source, special nuclear material and by-product material.

As part of its duties, the NRC is charged with the task of seeing that these measures prevail. This aspect requires inspection and review in order to assure this. This function is carried out by NRC personnel (inspectors) at regular intervals. Their job is to make the inspections and report their findings. In the event that a failure to comply is noted, the licensee is required to correct this.

Many of the states have taken up the task of setting up their own safety standards. The NRC has been directed to assist the states to assure that the state and Commission programs are compatible. These states are referred to as Agreement States.

The U.S. Department of Energy (DOE) has taken over the remaining functions of the AEC. These activities related to energy research and development and involved activities carried out by the Commission or by its contractors. The DOE has issued regulations which pertain to its own activities as well as to those of its contractors, not subject to licensing. These regulations appear in the DOE Orders, which replaced the Manual Chapters of the AEC. The standards which apply specifically to radiation protection are contained in DOE Order 5480.11.[18] These standards are based upon the recommendations of the ICRP, NCRP and the guidance of the EPA.[28] Similar to the NRC, the DOE is charged with the inspection of its contractors to see that they are in compliance with the DOE Orders.

Safety in the shipment of radioactive substances is principally the responsibility of the U.S. Department of Transportation (DOT). Title 49-Transportation, of the Code of Federal Regulations,[30] deals with hazardous shipments including radioactive materials. See Section 16.K for a discussion of these regulations.

From time-to-time, changes need to be made in various regulations. The Code of Federal Regulations is revised through submission of changes proposed by an agency and the Federal Government to other governmental and private agencies and to the general public. Hearings are held, if needed, to discuss amending the proposals. Subsequently, the proposals as amended are published in the Federal Register. If no adverse action is taken, the changes or additions become part of the Code of Federal Regulations and have the effect of law. Other agencies of the Federal Government having an interest in the regulatons for the shipment of radioactive substances are: Interstate Commerce Commission, Coast Guard, Federal Aviation Agency, Postal Service, DOE and the NRC. The Department of Transportation has made an effort to make its labeling system conform with the regulations of the International Atomic Energy Agency (see Section 16.K).

This concludes the brief outline of the main groups whose function it is to regulate. From the dynamic nature of the field of atomic energy, one can expect that many new problems will arise. For this reason, no attempt has been made to discuss any of the present regulations in detail. As new problems arise, new rules must be worked out. Thus, as in the case of exposure limits, changes will occur. To keep up on current changes, it is necessary to periodically review the Federal Register.

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QUESTIONS

7.1 What term is used to indicate a judgment made on the basis of benefits weighed against potential damage in the setting of radiation exposure limits?

7.2 What term is given to the dose of radiation which just causes reddening of the skin?

7.3 Why did progress lag in the setting of standards for radiation protection?

7.4 What term was given to the dose below which, it was presumed, the body would not suffer unacceptable injury?

7.5 What term succeeded the tolerance dose?

7.6 What term has been recommended to succeed "maximum permissible dose"?

7.7 What term is used to indicate the sum of the radiation doses received by an individual?

7.8 Identify: a) ICRU b) ICRP
c) NCRP d) FRC

7.9 Identify a) "the radiation dose which should not be exceeded without careful consideration of the reasons for doing so; every effort should be made to encourage the maintenance of radiation doses below this guide as practicable."

7.10 Explain the basis of the ICRP dose limits for individuals are set?

7.11 What organs need not be included in the weight factors for stochastic risk?

7.12 Which of the individual organs is weighted highest for stochastic risk? Why?

7.13 For long-term exposure, what is the dose rate limit recommended by the ICRP?

7.14 When designing for a new facility, what is the DOE recommended design goal for exposure?

7.15 Where does the responsibility of the Federal government lie with respect to radiation matters?

7.16 How does the Department of Energy establish its regulations pertaining to radiation safety that they may have the force of law?

7.17 What agencies of the Federal government regulate safety in regard to the shipment of radioactive substances?

7.18 What private transportation agency works with the Federal government agencies in regard to the shipment of radioactive substances?

SECTION 8 - EXTERNAL PROTECTION FACTORS-TIME, DISTANCE, SHIELDING

A. Factors Affecting Exposure in a Radiation Field

The philosophy inherent in any program of radiation safety is to reduce exposure, whether internal or external, to a minimum. If it is impossible or impractical to remove a source of radiation, other means must be considered for purposes of personnel protection. This brings about a certain reliance on judgment, in that one must make decisions regarding the level of exposure which is to be tolerated. Since expense and time consuming factors may be involved, the goal of radiation protection is embodied in the acronym "ALARA", which stands for As Low As Reasonably (Readily) Achievable.[1] That is, one should make whatever reasonable efforts one can to reduce exposure to the lowest levels, taking into account economic and practical considerations.[2]

Three factors which determine the total exposure one receives in a given external radiation field are:

(1) time of exposure;
(2) distance from the source;
(3) amount of shielding present.

1. Time

The time factor simply means that the longer one remains in a radiation field, the greater will be the exposure received. At times, especially during emergencies, work must be performed in a high radiation field. In all cases, the work activities should be carefully planned outside the work area so that a minimum amount of time is used to complete the job. If the time required for one man to complete the job would result in an exposure beyond prescribed limits, then a team of workers should be employed. This would mean a small exposure for several people instead of a large exposure for one individual. Whenever working in a radiation field, one should attempt to complete the work quickly. One should not linger in radiation fields discussing the work. If discussions are necessary, leave the radiation field to conduct the discussions and minimize exposure time.

For time consuming jobs in high radiation fields, the crew should rehearse, or conduct dry runs, without using radioactive materials. Rehearsing will allow one to discover where potential "bottlenecks" in the procedures might occur before someone gets exposed. By correcting the procedures, the time to complete the job will be reduced.

In some cases, one may be able to limit the amount of radioactive material being used at any one time, or one may be able to distribute the exposure over a longer time frame by limiting handling time.

Assume that work must be done in a field of $2.6\text{x}10^{-4}$ C/kgh (approximately 1 R/h). If we are allowed $5.2\text{x}10^{-6}$ C/kg (approximately 20 mR) each working day, how long can a person work in this area so as not to exceed this limit?

$$X = \dot{X}t$$

$$= t(h) \text{ x } 2.6\text{x}10^{-4} \text{ (C/kgh)} = 5.2\text{x}10^{-6} \text{ C/kg}$$

$$\text{Then } t = \frac{5.2\text{x}10^{-6}}{2.6\text{x}10^{-4}} \text{ h}$$

$$t = 0.02 \text{ h (1.2 min)}$$

This means that an individual would receive a daily maximum allowable exposure in 1.2 min while working in this radiation field. Suppose the job needed to be completed, and the estimated time for completion was 15 min. How many men would be needed so that no one exceeded a daily limit?

$$N = \frac{15 \text{ min}}{1.2 \text{ min/man}} = 12.5 \text{ men required}$$

2. Distance

The intensity of a radiation field decreases with distance from the source. If we consider a point source of penetrating radiation, the decrease in intensity will be inversely proportional to the square of the distance between any two radial points from the source. This "inverse square law" can be utilized only when the dimensions of source and measuring device are small compared with the distance between them and the distances are measured in air or vacuum (see Section 3.D).

For other than point sources, the intensity will decrease inversely with distance, but not necessarily as the square of the distance. Nevertheless, one should store sources far from any work area to take advantage of the reduced radiation field.

Increasing the distance from a source is often the most effective way of decreasing radiation exposure. However, one is often faced with a space problem, or a job in which the worker has to be in close proximity to the source. This essentially places a restriction on the distance factor. In these cases, one must still avoid hand contact. To reduce exposure to the body, work at arm's length. Use tongs, long-handled tools, and, if the radiation field is very high, remote control devices to handle radioactive materials. Also, one may need to employ shielding.

3. Shielding

We can reduce the radiation hazard by placing a suitable attenuating material, or combination of materials, between the source and worker. The choice and/or thickness of attenuating material (shield) will depend upon the type, and energy, of the radiation.

A shield is a medium of some thickness which will stop or effectively attenuate radiation to nonhazardous levels. The effectiveness of the shield is determined by the interactions between the incident radiation and the atoms of the absorbing medium. The interactions which take place depend mainly upon the type of radiation (α, β, x-γ

photons, neutrons, etc.), the energy of the radiation, and the atomic number of the absorbing medium. An effective shield will cause a large energy loss in a small penetration distance without emission of more hazardous radiation.

In choosing a shielding material, our first consideration must be personnel protection. However, other factors may influence our choice of material-such as; is it economical; is it too heavy; how much space are we allowed for the shield; does it have proper structural strength? Also, will it create a toxicity or contamination problem due to radiation damage; to what levels must the radiation be reduced in order to obtain accurate measurements with various instruments in the area, etc.?

When massive shields are required, cost is a very critical factor, as well may be the type of material and space limits. In this case, the transport of the radiation must be studied in more detail to arrive at optimum design features. For some applications, the shielding requirements are less severe and can be adequately estimated by simpler methods.

The time, distance, and shielding factors are used individually or in combination to minimize external radiation hazards. For more discussion about protection against external radiation, one may refer to References 3-7.

Of the three basic principles of external radiation protection (time, distance, shielding), shielding presents the most complex problem in many applications. There are so many factors influencing the design of shields that a large number of such calculations must be performed on computers. For this reason, it is hard to treat the subject from the health physics standpoint in that, the most interesting aspects of shielding are usually not the concern of the radiation protection section. However, it is of value to examine some of the quantitative relationships because they, on occasion, do become useful in practical health physics work. Certainly, the qualitative approach to shielding should be studied and understood.

B. Alpha Radiation

Alpha particles, which lose energy rapidly in any medium because of their relatively high ionization loss, can be stopped by very thin absorbing materials. A few sheets of paper or thin (less than 400 μm) aluminum foil will absorb alphas from α-emitting sources. The most energetic alphas will travel only a few tens of mm in air. The outer layer of skin, approximately 0.07 kg/m^2 (7 mg/cm^2) in thickness, will absorb α particles up to approximately 7.5 MeV. Since this is dead tissue, no harmful effect is produced upon the body. Therefore, α radiation is not considered an external exposure problem.

Alphas accelerated in machines attain much higher energies but are of little consequence in regard to bremsstrahlung. These alphas bombard target matter which then emit radiation products (neutrons, gammas) which require more shielding than that needed to stop the α particles. The combination of α sources with certain light elements such as Li, Be, B, O, N, F, will result in neutron emission which adds to the shielding considerations.

C. Beta Radiation

The processes by which β particles lose energy in absorbers are similar to those for alphas. However, an additional process must be considered in dealing with β absorption. This is the process whereby electromagnetic radiation, called bremsstrahlung, is produced.

A β particle has a very small mass and one-half the value of the magnitude of charge on an α particle. So for a given energy, a β particle will have a much greater velocity than an α particle. As a result of these and other factors, the β particle has a lower specific energy loss, which means that its penetration in any absorber will be much greater than that of an α particle. However, the β radiation from most natural sources have ranges < 20 kg/m^2 (2 g/cm^2).

A 1 MeV β particle will travel about 3.5 m in air. In order to penetrate the dead layer of skin, a β particle must have an energy of about 70 keV. Beta radiation is considered a slight external hazard, but mainly a skin exposure hazard.

The thickness and choice of material for shielding from β radiation depends upon: (1) stopping the most energetic β, and (2) shielding any bremsstrahlung.

The shielding thickness which is necessary to stop β particles of a given energy will decrease with increasing density. For example, approximately 280 μm of aluminum (ρ=2.7x10^3 kg/m^3) will stop 1.5 MeV betas, whereas only approximately 66 μm of lead (ρ=1.134x 10^4 kg/m^3) is needed for the same purpose.

However, the production of bremsstrahlung increases with increasing atomic number of absorber. A β source with E_{max} = 1 MeV will lose about 3% of its energy as bremsstrahlung when lead (Z=82) is the absorber. If aluminum (Z=13) is the absorber, the fraction is about 0.4%. Therefore, low-Z absorbers such as aluminum, plastics (lucite), or even glass are effective β shields.

If cost and/or weight must be considered, a combination shield may be used. In this case, a low-Z absorber is used to stop the betas, followed by a high-Z absorber to attenuate the bremsstrahlung. This type of shield (as a shipping container) is further discussed in Section 8.D.3.

1. Electron Accelerator Shielding

Whereas the shielding of β sources in general is a simple matter, the picture changes with respect to electron beams. The interactions of the electrons in the beam are the same as those of betas. However, because electrons may attain much higher energy (> 1 GeV), the secondaries which are produced become the item of greatest concern. At electron energy > 10 MeV, photons produced in bremsstrahlung begin to initiate the release of fast neutrons (see Figure 8.1). At higher electron energies, higher energy bremsstrahlung spectra occur which lead to even

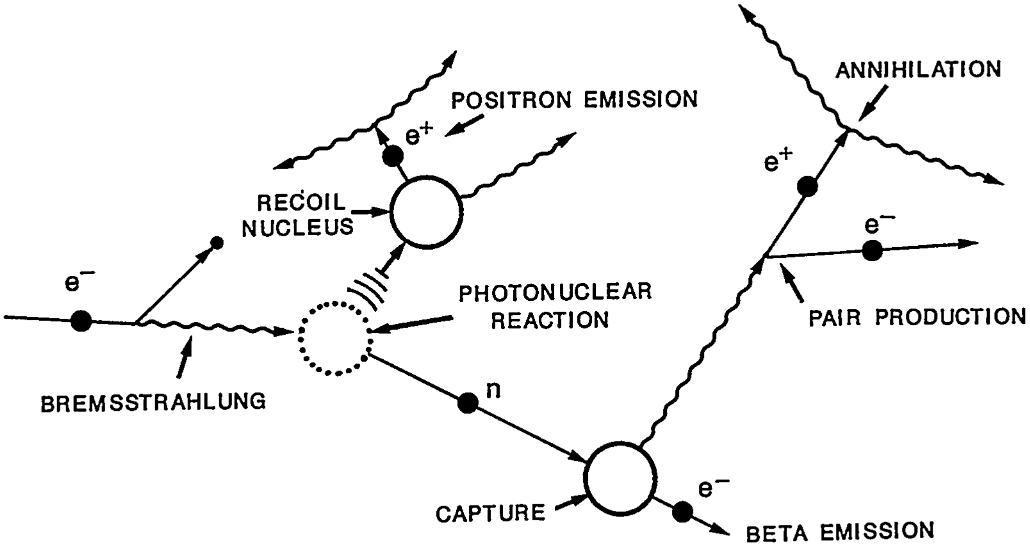


Figure 8.1 High energy electron interaction in matter - for E_{el} high enough, the process becomes multiplicative.

higher energy neutrons, pions, photofission, as well as electron-photon showers. In the GeV range, production of nucleonic cascades (see Section 8.E.1) will occur. Much of the shielding will then be dependent upon neutron attenuation. A further problem is the residual radioactivity due to the released neutrons and those which are then captured farther on in the shield, or by nearby matter.

The problem of shielding electron beams is one of shielding for the secondary radiations produced in electron interactions. As mentioned above, the main products are bremsstrahlung (x rays), γ rays, neutrons, and residual radioactivity (β^+, β^-, and γ rays) for electron beams up to about 100 MeV.[8] The approach to shielding in many respects is similar to the approach used for reactors. However, in the accelerator case, the neutron spectrum may have higher energy components than the reactor spectrum (for an electron accelerator of E_e = 100 MeV, the main problem is neutrons of E_n > 40 MeV). A second point is that electron and x ray beams are highly directional (mostly forward) so that the output end of the accelerator becomes the object of shielding concern.

Here, the interactions produce neutrons which can be treated as fairly isotropic, so much of the shielding design concerns these products. The neutrons of concern for lower energy accelerators (< 30 MeV) are referred to as "giant resonance neutrons," and have an effective energy of approximately a few MeV. As the accelerator energy increases, eventually the high energy component of neutrons (E_n > approximately 100 MeV) will dominate the shielding considerations.

References 8 and 9 contain information on the shielding of electron accelerators of E up to 100 MeV. Reference 9 also contains information on the shielding of electron accelerators of E > 100 MeV. In addition, Reference 10 discusses approaches to neutron shielding at high energy electron accelerators.

D. Gamma Radiation

Gamma rays do not lose energy continuously, as do α or β particles, in traversing a medium. As a result, γ rays are much more penetrating than α or β radiation. Gamma radiation from most radionuclides is attenuated in matter mainly by: (1) the photoelectric effect, (2) Compton effect, and (3) pair production.

Until the photon interacts, no energy is lost to matter. When the photon does interact, the result is the release of energetic electrons. The interactions discussed above are directly proportional to the atomic number (Z), generally to some power between 1-5. By increasing the number of available electrons, one enhances the probability of photon interactions. So, materials of higher Z and higher density, such as Pb, U, Th, Au, and W, are best suited for γ shields. The use of some of these metals is limited by their high cost and/or weight. Therefore, metals such as Fe, Pb, Cr, and Ni are used for γ shielding. In addition, because concrete is an effective γ and n shield, concrete is often used for γ when neutron fields are also of concern, as in the case of reactors and accelerators.

1. Calculations of Shield Thickness

Gamma-ray absorption is an exponential process. Theoretically, this means that γ rays are never completely absorbed no matter how thick the absorber. However, practically, we can choose a shield thickness which will reduce the intensity to nondetectable or nonhazardous levels.

Three relationships (see Figure 8.2) can be used for calculating shield thickness for monoenergetic photons:

$I = I_o e^{-\mu x}$; μ=Linear Attenuation Coefficient eq. 3.39 (A)

$I = I_o e^{-(\mu_{en})x}$ μ_{en}=Linear Energy Absorption Coefficient eq. 8.1 (B)

$I = I_o b e^{-\mu x}$ b = buildup factor eq. 3.51 (C)

In radiation protection work we prefer equation (B), since this equation incorporates a safety factor in our calculations (see Figure 8.2). The attenuation coefficient μ_{en} assumes that all scattered gammas reach our point of interest. This means that we will overestimate the intensity I when radiation I_o moves through a shield thickness x. As a result, the calculated thickness will be greater than is necessary to reduce the intensity to prescribed levels.

Equation (A) is used for narrow (collimated) beam conditions. Here, we assume that all γ photons which deviate (scatter) from the beam never reach our detector. If this equation is used in treating a wide beam of radiation, we would underestimate the required shield thickness. This occurs because some scattered radiation would arrive at the detector. In addition, some contribution may be made by fluorescent radiation, bremsstrahlung, and e^+ annihilation photons.

The third equation (C), is used when cost, weight, or space are factors which must be considered. If we know the buildup factor b, we calculate the true γ intensity I; hence, we get a more accurate estimate for the needed shield thickness.

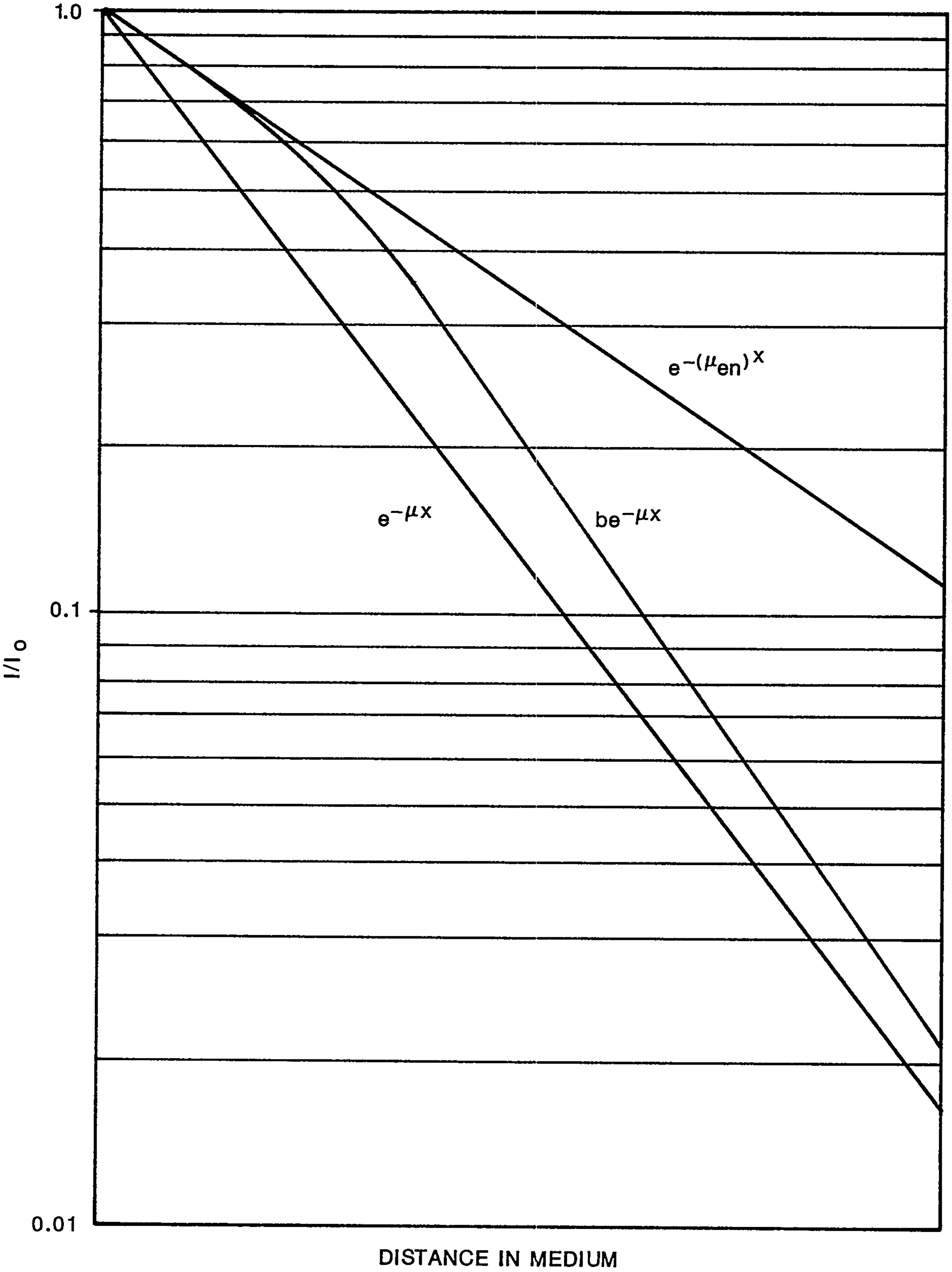


Figure 8.2 Relationships for calculating shield thickness. The curve $be^{-\mu x}$ gives the correct attenuation for distance x; $e^{-\mu x}$ predicts too much attenuation, $e^{-\mu_{en} x}$ predicts too little.

For quick shielding estimates, we can use multiples of tenth value or half value layers. A half value layer ($x_{1/2}$) is defined as that thickness of a material which reduces the radiation intensity to one half its initial value. A tenth value layer, or any other thickness which reduces the intensity to some desired fraction of the initial intensity, is similarly defined (see Section 3.E.6).

A half value layer is determined as follows:

$$I = I_o e^{-(\mu_{en})x} \qquad 8.2a$$

Then: $$I/I_o = 1/2 = e^{-(\mu_{en})x_{1/2}} \qquad 8.2b$$

or $$\ln\frac{1}{2} = -0.693 = (\mu_{en})x_{1/2} \qquad 8.2c$$

From this:

$$x_{1/2} = 0.693/\mu_{en}. \qquad 8.3$$

The above expression can be used to determine the number of half value layers needed to reduce the initial intensity to any other desired level. In Section 3.E.6, $x_{1/2}$ was defined in terms of the linear attenuation coefficient μ, which would be the HVL for narrow beam conditions.

If we substitute $0.693/x_{1/2}$ for μ_{en} in the exponential expression, we obtain

$$I/I_o = e^{-0.693(x/x_{1/2})}. \qquad 8.4a$$

From this, taking the reciprocal of each side, and then the logarithm,

$$\ln(I_o/I) = \ln 2(x/x_{1/2}), \qquad 8.4b$$

where $x/x_{1/2}$ is the number of half value layers. If we let $n = x/x_{1/2}$, then $I_o/I=2^n$, which is the same result as equation 3.46.

EXAMPLE 1. It is desired to reduce a beam of γ rays to 1/16 of its initial intensity. The gammas have an energy of 1 MeV and lead will be used as the shielding material. How many half value layers are required? How many m of lead are required?

(a) $I_o/I = 16 = 2^n$

Now

$$\ln 16 = n \ln 2$$

or

$$n = \frac{\ln 16}{\ln 2} = \frac{2.773}{0.693} = 4$$

Therefore, 4 half value layers are required.

(b) The value of $(\frac{\mu_{en}}{\rho})_{Pb}$ is obtained from the energy absorption coefficient versus energy curve (Appendix E),

$$(\frac{\mu_{en}}{\rho})_{Pb} = 0.0038\frac{m^2}{kg}; \quad (0.0038 \frac{m^2}{kg})(1.134x10^4 \frac{kg}{m^3})=43.1 \ m^{-1}(.43\ cm^{-1}) = (\mu_{en})_{Pb}.$$

Then

$$1 \text{ half value layer} = 0.693/(\mu_{en})_{Pb} = 0.693/43.1 \ m^{-1}$$

and

$$4 \text{ half value layers} = 4x(0.693/43.1) = 64.3 \text{ mm of lead } (6.43 \text{ cm}) = 0.064 \text{ m.}$$

EXAMPLE 2. A radium source will be contained at the center of a wooden box. The γ reading through the surface of this box is 7x 10^{-8} C/kgs (approximately 1 R/h). What thickness of lead will be required, as a lining inside the box, to reduce the reading at the surface to $1.4\text{x}10^{-10}$ C/kgs (approximately 2 mR/h). Assume an effective energy of 0.8 MeV. Using equation 3.46,

$$I_o/I = 7\text{x}10^{-8}/1.4\text{x}10^{-10} = 500 = 2^n$$

Then, taking the logarithm of each side, and dividing by ln 2 gives

$$n = \frac{\ln 500}{\ln 2} = \frac{6.215}{0.693} \sim 9.$$

Therefore, 9 half value layers are required. To find the Pb thickness,

$\frac{\mu_{en}}{\rho}$ is found from Appendix E.

$$(\mu_{en}/\rho)_{Pb} = 0.0047\ \frac{m^2}{kg};0.0047\ \frac{m^2}{kg}\ (1.134\text{x}10^4\frac{kg}{m^3})$$

$$= 53.3\ m^{-1}(.533\ cm^{-1}) = (\mu en)_{Pb}$$

Hence,

$$9\text{x}0.693/53.3\ m^{-1} = .117\ m\ (4\text{-}5/8\ in)\ \text{of lead}$$

2. Buildup Factor

In calculations which include the buildup factor b, we take into account the scattered photons resulting from the Comptom effect and the "uncollided gamma flux" which reach our detector. The value of b varies with radiation energy, shield material, source geometry, and depth of shield penetration.

Values for b in a number of relevant materials can be found in the literature (11-13) and values for lead are shown in Figure 8.3. Frequently, the values of b are plotted on semilog paper against values of the mean free path for the initial radiation. The mean free path (mfp) is that thickness of absorber which will result in a reduction of 1/e in the <u>initial</u> beam intensity I_o, i.e., $I=0.368\ I_o$. For radiation of attenuation coefficient μ_o in the medium, the number of mean free paths can be found from

$$N_{mfp} = \mu_o x, \qquad 8.5$$

in which x is the thickness of the shield.

We use the narrow beam relationship, $I = I_o e^{-\mu x}$, to determine the value of b. The number of mean free paths is given by $\ln(I_o/I)$ to a first approximation. One may iterate to obtain a better approximation of b and the required shielding.

<u>EXAMPLE 3.</u> A 1 MeV γ source is surrounded by a lead shield which reduces the intensity at a point outside the shield by a factor of 20 from what it would be without the shielding. What is the value of b to a first approximation? Using equation 3.39

$$I_o/I = e^{\mu x}$$

or

$$\ln I_o/I = \mu x = (\mu_o x)_1,$$

Then

$$\ln 20 = 2.99573 = 3 \text{ mfp}.$$

Now consult the 1 MeV curve for a point isotropic source in lead (Figure 8.3). From this we see that b = 1.93 for $\mu_o x = 3$ and E = 1 MeV. The value of b_1 is the first approximation to the actual buildup. To iterate, we use the product $b_1\ I_o/I$, to arrive at a new estimate of the penetration $(\mu_o x)_2$,

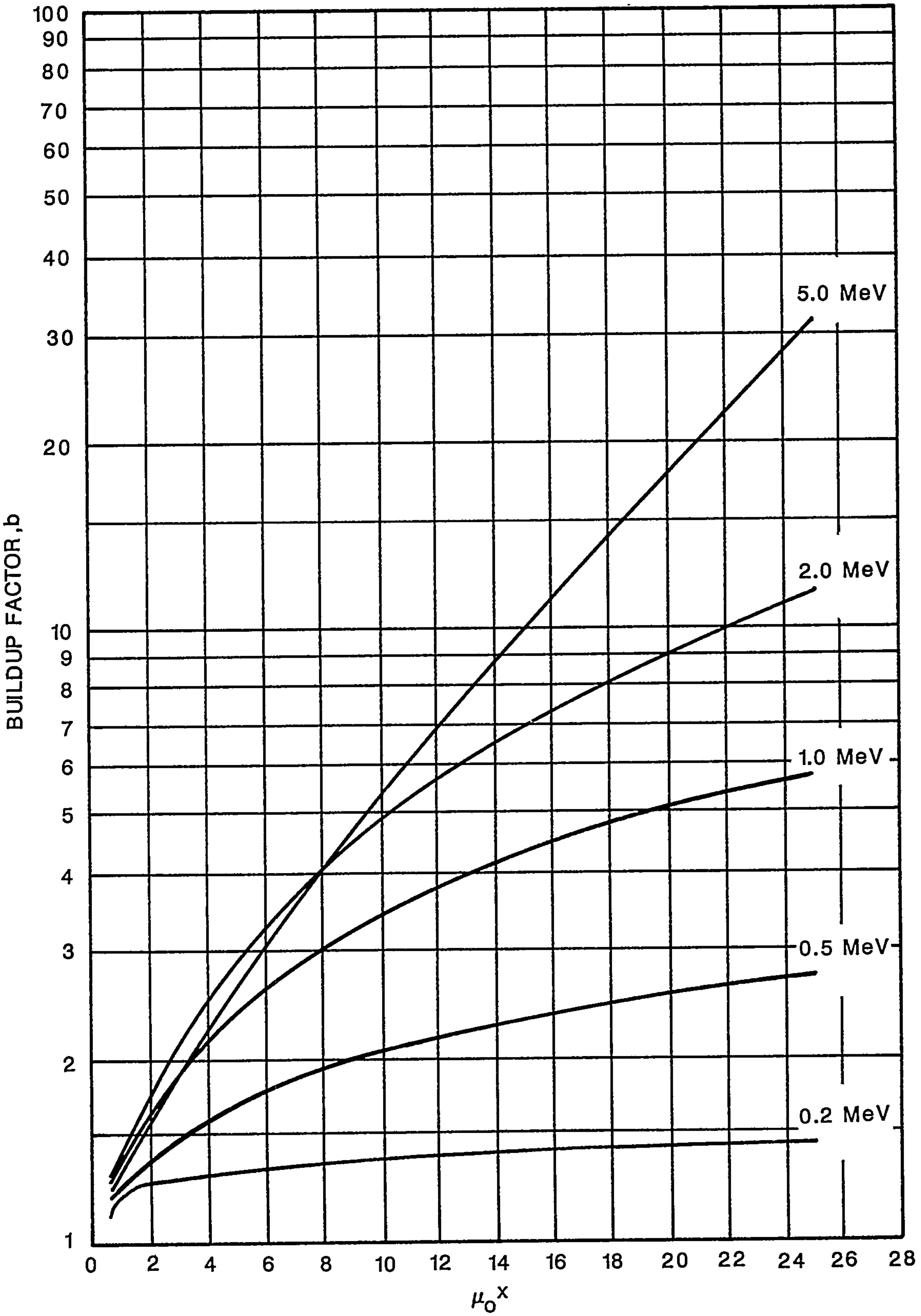


Figure 8.3 Exposure buildup factor in Pb - Point isotropic source (adapted from reference 13).

$1.93\ I_o/I = e^{\mu x}$

$\ln 1.93\ I_o/I = \ln 1.93(20) = (\mu_o x)_2$

$\ln 38.6 = 3.65325 \sim 3.65$ mfp.

The new value b_2, for $(\mu_o x)_2 = 3.65$, is ~ 2.08. Repeating the iteration,

$b_2\ (Io/_I) = e^{\mu x}$

$2.08(20) = e^{\mu x}$

$\ln 41.6 = 3.72810 = (\mu_o x)_3$ mfp.

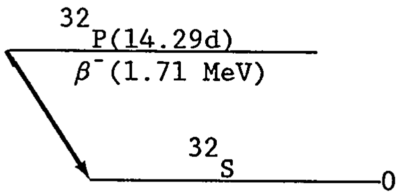
The value of b_3 is very nearly equal to b_2, so we stop the iteration process at this point and estimate the final b as 2.1. The value of the iteration approach is that many times one is asked to compute the thickness of a shield required to obtain a given attenuation. In cases where the scattering component of radiation may be important, one could significantly underestimate the required shield thickness by simply using equation 3.39.

To illustrate the use of both buildup, b, and calculations involving bremsstrahlung, we have the following example.

EXAMPLE 4. A 3.7 TBq (100 Ci) ^{32}P source will be shipped in a cylindrical container, $7.6\text{x}10^{-2}$ m (3 in) in diameter and .1016 m (4 inches) deep. The shipping regulations require a reading of $<2.6\text{x}10^{-6}$ C/kgh (10 mR/h) at 1 meter. Calculate the container wall thickness using equations (A), (B), and (C) in Section 8.D.1.

We first consider an all lead container; we will then work out the problem for a combination wall material of Lucite and lead.

The decay scheme[14] for ^{32}P is as follows



The fraction of β energy converted to bremsstrahlung is given by equation 3.16, $F = 3.33\text{x}10^{-4}$ ZE. For the all lead container:

$$F = 3.33\text{x}10^{-4}(82)(1.71) = 0.047.$$

Now

$3.7\text{x}10^{12}\ (\frac{\text{dis}}{\text{s}})(\frac{1\ \beta}{\text{dis}})(0.047)=1.739\text{x}10^{11}$ ph/s of bremsstrahlung

The C/kgh at 1 meter, unshielded, is $3.74 \text{ x } 10^{-15}\ n_\gamma$ $CE_\gamma = 3.74\text{x}10^{-15}\ (1.739\text{x}10^{11})(1.71) = 1.112\text{x}10^{-3}$ C/kgh (4.31 R/h). For E/3=0.57, as suggested in Section 3.B.3, $(\mu/\rho)_{Pb}$ = 0.0133 (m^2/kg) from Appendix E. $(\mu_{en}/\rho)_{Pb}$=0.0078 (m^2/kg), and b is found from the curve in Figure 8.3 by interpolation. Number of mean free paths=$\ln(I_o/I)=\ln(1.112\text{x}10^{-3}/2.6\text{x}10^{-6})=\ln\ 427.7 = 6.05482$; therefore b ~ 3.1. The density of lead is $1.134 \text{ x } 10^4$ (kg/m^3),

$I = I_o e^{-\mu x}; I_o/I = e^{\mu x}; \ln 427.7 = 0.0133(1.134\text{x}10^4)x;$

$\ln 427.7 = 150.8x;\ x=6.058/150.8 = 4.02\text{x}10^{-2}$ m(1.6 in) (A)

$I = I_o e^{-(\mu_{en})x};\ I_o/I = e^{(\mu_{en})x};$

$\ln 427.7 = 0.0078(1.134\text{x}10^4)x;$

$\ln\ 427.7 = 88.45x;\ x=6.058/88.45 = 6.85\text{x}10^{-2}$ m (2.7 in). (B)

$I = I_o b e^{-\mu x};\ I_o b/I = e^{\mu x};\ \ln 427.7(3.1)=0.0133(1.134\text{x}$
$10^4)x;$

$\ln\ 1325.9 = 150.8x;\ x=7.190/150.8 = 4.77\text{x}10^{-2}$ m (1.9 in). (C)

Using the above value in (C) for the wall thickness, the weight of the container would be about 46 kg (101 lbs). In using μ_{en} to calculate the amount of lead required, we get a container with a weight of 90 kg (approximately 200 lbs). This is about twice the amount of lead that is actually needed. Note that the value obtained in (A) would represent insufficient shielding.

3. Effective Atomic Number

We will now consider the problem if lucite is used to shield the betas and lead to shield the bremsstrahlung.

According to Evans (Reference 15), the effective atomic number Z_{eff} for a compound, in reference to the bremsstrahlung formula, is given by

$$Z_{eff} \simeq \frac{N_1Z_1^2 + N_2Z_2^2 + N_3Z_3^2+\ldots}{N_1Z_1 + N_2Z_2 + N_3Z_3+\ldots},$$

where N_i is the number of i^{th} atoms, of atomic number Z_i, per cc. For Lucite ($C_5H_8O_2$), $Z_{eff} = 5.85$.

The maximum range of ^{32}P betas in Lucite is approximately 0.008 m (approximately 5/16 in). The fraction of β energy converted to bremsstrahlung is

$$F = 3.33 \times 10^{-4}(5.85)(1.71) = 0.0033.$$

Since the source activity is 3.7×10^{12} Bq, the amount of bremsstrahlung is $3.7 \times 10^{12}(\frac{dis}{s})(1\frac{B}{dis})(0.0033) = 1.221 \times 10^{10} \frac{ph}{s}$ of bremsstrahlung.

It is difficult to derive a function which describes the energy spectrum of the photons emitted. However, one can assume that the bremsstrahlung consists of monoenergetic x rays equal to the maximum β energy. This will result in a safe estimate for the exposure rate, since only a small fraction of the photons will have E=Emax (see Figure 3.10a.).

In this case, the exposure rate is:

$$X \text{ at 1 meter} = 3.74\text{x}10^{-15}\ n_\gamma C E_\gamma$$
$$=3.74\text{x}10^{-15}(1.221\text{x}10^{10})(1.71)$$
$$= 7.81\text{x}10^{-5}\ \text{C/kgh}\ (.303\ \text{R/h}).$$

Once again b is found from the curve (Figure 8.3):

$\ln(7.81\text{x}10^{-5}/2.6\text{x}10^{-6})=\ln 30.04 = 3.403 \sim 3.4$ mfp

From the curve b~ 2.2. Recalculating, as before,

$$I=I_o e^{-\mu x}; I_o/I=e^{\mu x};\ 30.04=e^{0.0133(1.134\text{x}10^{4})x};$$

$$\ln 30.04 = 150.8x;\ x = 3.403/150.8 = 2.26\text{x}10^{-2}\ \text{m}\ (0.9\ \text{in}). \quad (A)$$

$$I = I_o e^{-(\mu_{en})x};\ I_o/I = e^{(\mu_{en})x};$$
$$30.04 = e^{0.0078(1.134\text{x}10^{4}/x};$$

$$\ln 30.04 = 88.45x;\ x = 3.403/88.45 = 3.85\text{x}10^{-2}\text{m}(1.5\ \text{in}). \quad (B)$$

$$I = I_o b e^{-\mu x};\ I_o b/I = e^{\mu x};$$
$$30.04(2.2) = e^{0.0133(1.134\text{x}10^{4})x};$$

$$\ln 66.088 = 150.8x;\ x = 4.191/150.8 = 2.78\text{x}10^{-2}\text{m}(1.1\ \text{in}). \quad (C)$$

Hence, a small amount of lucite will appreciably reduce the lead requirement. This container needs about 40% as much lead as is necessary for the "all lead" container. Again, if μ_{en} is used in the calculations, the amount of lead is almost 1.7 times as much as is actually needed.

It should be remembered that the use of the buildup factor is important where economy, space, and weight must be considered, such as in the design of a reactor. If these considerations are unimportant, the simpler expression involving μ_{en} can be used. For calculations in air, μ_{en} can always be used. The buildup factor is important when absorbers are used in which the scattering can be quite extensive.

4. Point Kernel

In the previous examples, a combination of exponential attenuation and inverse square law drop off was used to obtain the total attenuation. This combination for a point source is referred to as the point kernel. Expressed in terms of the fluence rate, ϕ, the point kernel may be written

$$\phi = \frac{N_\gamma \, b \, e^{-\mu x}}{4\pi r^2} \quad \frac{\text{ph}}{\text{m}^2\text{s}} \qquad 8.6$$

In equation 8.6, x is the thickness (m) of the shielding medium of attenuation coefficient, $\mu(\text{m}^{-1})$, and buildup factor, b, r is the distance (m) from the point source to the exposure point (see Figure 4.1), and N_γ is the photon source strength (ph/s). If the source and exposure point are within the same medium, then x=r. When the medium is air, and r=x is only a few meters, $e^{-\mu x} \sim 1$, b=1, and the expression reduces to equation 4.6. Suitable conversion factors may be included, on the right side of equation 8.6, to arrive at the exposure, kerma, or absorbed dose rate.

The point kernel is often used to compute the total contribution to an exposure point from an extended source. In this application, the point kernel is used to represent the contribution to the fluence rate at an exposure point, from each small element of the extended source. The total contribution is found by summing all the contributions by integrating over the source dimensions.[13]

One may express the broad beam attenuation in terms of a transmission (or attenuation) factor. This approach is used to estimate shielding requirements for medical equipment.[16] If we rewrite equation 8.6, and express our quantity as exposure rate, $\dot{X}$, then

$$B = \frac{\dot{X}r^2}{\dot{X}_o} \qquad 8.7$$

in which $\dot{X}_o$ is the exposure rate at 1 m, r(m) is the distance to the point of interest, and B is the transmission factor, defined as the ratio of the radiation field with the shield present to that with the shield removed. In this case, the factor B includes both attenuation and buildup. Transmission factors can then be measured with depth of penetration, and plotted as shown in Figure 8.4. One simply reads the factor B from the curve to find the effectiveness of an iron shield of a stated thickness. Conversely, given a desired exposure rate $\dot{X}$, and knowing the rate at 1 m, $\dot{X}_o$, one may compute the necessary value of the transmission B to obtain $\dot{X}$ at a certain distance r. Having computed B, one may use Figure 8.4 for the relevant radionuclide to find the corresponding thickness of iron needed. For example, a 37 GBq (1 Ci) source of ^{60}Co has an exposure rate, $\dot{X}$, of approximately 9.27×10^{-8} C/kgs (approximately 1.3 R/h) at 1 m. How much iron is needed to reduce the field to 1.8×10^{-10} C/kgs (approximately 2.5 mR/h) at 4 m? Using equation 8.7,

$$\dot{X} = \frac{\dot{X}_o}{r^2}B,$$

$$\text{and} \quad B = \frac{r^2\dot{X}}{\dot{X}_o} = \frac{(4)^2 1.8\times10^{-10}}{9.27\times10^{-8}} = 3.11\times10^{-2}$$

From Figure 8.4, when $B=3.11\times10^{-2}$, the thickness of iron needed would be approximately .119 m (4.7 in). Transmission curves for a number of radionuclides in several different absorbers can be found in Reference 16.

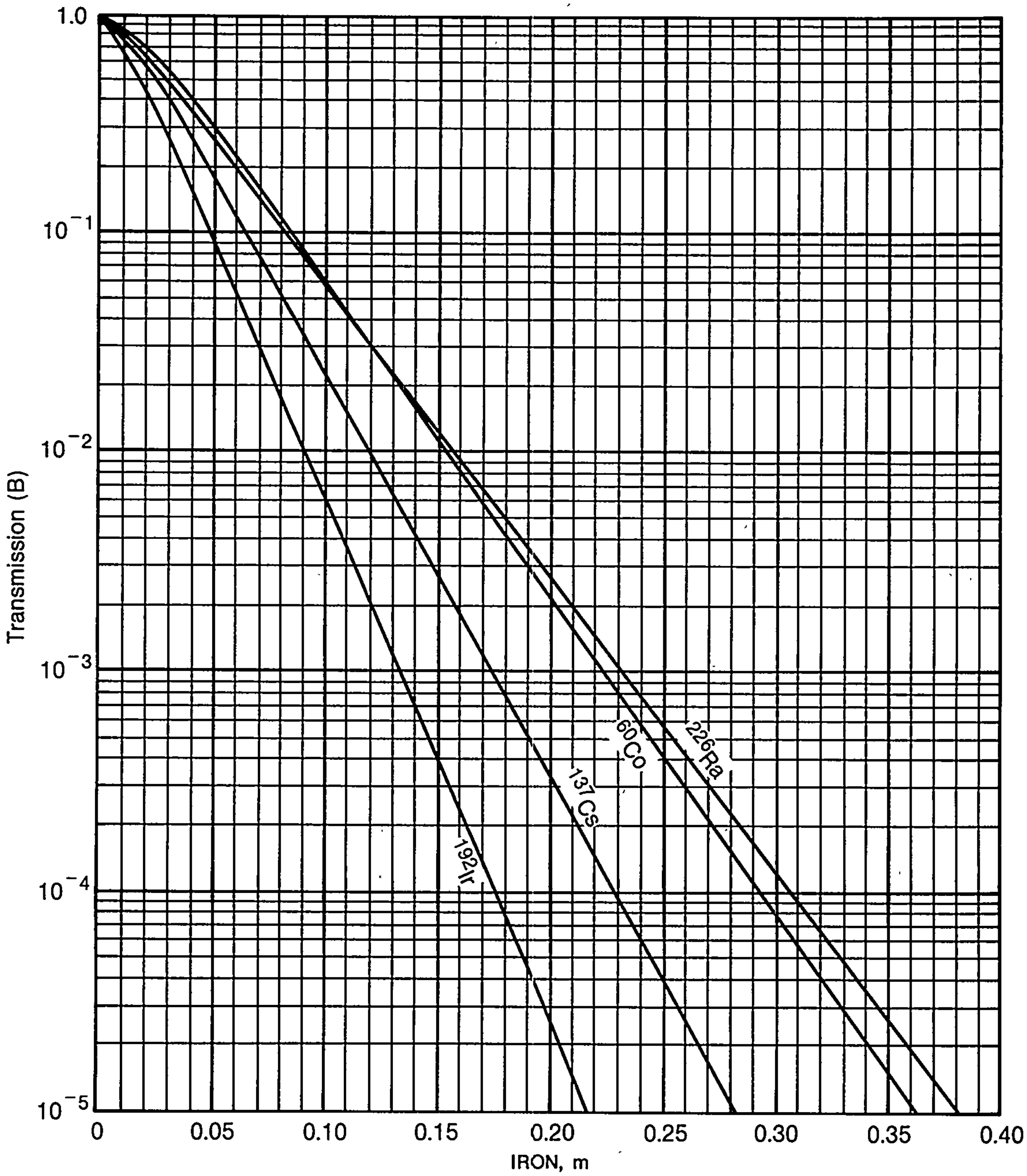


Figure 8.4 Transmission of gamma rays through iron. (Adapted from Reference 17).

With respect to some x ray equipment, the exposure is expressed in a different manner. The term weekly workload, W, is used to designate the degree of use and is given by the product of the target current (mA) and total machine running time (min), for a certain kilovoltage machine. In addition, two other factors are used to modify the value of the workload. The use factor, U, corrects for the fraction of the total time that the radiation is directed at a particular barrier. For the direct (primary) beam, the use factor is generally taken as 1 for floors and ¼ for walls.[16] The occupancy factor, T, corrects for the fraction of time that the area is occupied while the beam is on. The value of T varies from 1 (full occupancy), to ¼ (partial occupancy), to 1/16 (occasional occupancy).[16] With these adjustments, we may rewrite equation 8.7, in terms of exposure X, as

$$X = \frac{WUT}{r^2} B. \qquad 8.8$$

In Reference 16, Appendix D, a number of curves are shown giving the attenuation (transmission) factor in lead and concrete for several values of machine kilovoltage. As an example, let W = $1.2\text{x}10^6$ mAs ($2\text{x}10^4$ mAmin), U=1, T=1, for a 250 kV x ray machine. Find the thickness of lead necessary to reduce the weekly exposure to $5.16\text{x}\ 10^{-6}$ C/kg (20 mR) at 4 m. Using equation 8.8 to find B,

$$B = \frac{Xr^2}{WUT} = \frac{5.16\text{x}10^{-6}(4)^2}{1.2\text{x}10^6(1)(1)} = 6.88\text{x}10^{-11}\ \frac{C}{\text{kgmAs}}\ \left(1.6\text{x}10^{-5}\ \frac{R}{\text{mAmin}}\right).$$

Using Figure 2, Appendix D of Reference 16, we find the thickness of lead for $B=1.6\text{x}10^{-5}$ (since the units of the curve are R/mAmin). The required Pb thickness is approximately 10 mm. Notice the use of $5.16\text{x}10^{-6}$ C/kg in the shield design. This is 1/5 of the maximum allowable weekly exposure, and represents the application of ALARA principles. That is, designing the shield so that the exposure will be less than the maximum allowable weekly exposures. When this is done <u>before</u> shielding is installed, the cost is often reduced.

In addition to primary beam attenuation of x ray machines, scattered radiation also needs to be treated. Reference 16 deals with the aspects of shielding the secondary, or stray, radiation.

E. Neutrons

Neutrons, like gammas, are a highly penetrating form of radiation. They possess no charge and, therefore, are unaffected by the electric fields of atoms in the traversed medium.

Neutron attenuation is accomplished mainly through elastic and inelastic collisions which reduce the energy of neutrons until they are absorbed in the medium. In many cases, penetrating γ rays are produced as a result of neutron absorption. In addition, other secondary radiations, including more neutrons may be produced. The probability of scattering of neutrons is much higher than the absorption probability in many materials. The cross sections for neutron interactions tend to show more variation with energy than those exhibited for photon interactions. For these reasons, buildup factors for neutrons are a strong function of the material composition, the incident E_n spectrum, and the geometry of the setup.[13]

The problem of neutron shielding is further complicated by the fact that substances which effectively attenuate neutrons are generally poor γ shields. Since photons usually accompany neutrons, one may shield neutrons adequately, but not γ rays. Neutrons are more hazardous to a biological system than gammas; therefore, the neutron fluence emerging from a shield should be smaller than the γ fluence. However, for deep penetrations, the neutrons may be effectively removed and then a γ problem may be encountered. Fast neutrons are more difficult to attenuate and therefore are the main concern in the shielding problem. Almost all neutrons are fast upon release from a source.

A further consideration is that neutrons are generally emitted with a spectrum of energies. Thus, the initial attenuation characteristics are determined by a number of components. Each component is then modified as it moves through the medium, since the neutrons lose energy. The result is that as the neutrons move through the material, the lower energy neutrons are removed more easily than are the higher energy neutrons. So, as the penetration in the medium increases, the surviving neutrons are those which undergo the least interactions. This results in what is called "hardening" of the spectrum. This means that the average energy of the components shifts from a lower value to a higher value as penetration increases.

Elements of low mass number (A) are ideal neutron moderators. Therefore, hydrogen (in the form of water, plastic, or paraffin), beryllium, carbon (in graphite), and LiH are popular shield constituents. A shield composed entirely of a hydrogenous material, such as water, is satisfactory for neutron attenuation; however, intermediate or heavy elements alone are not suitable neutron shields. This is due to the fact that in a nonhydrogenous material, a larger number of collisions are necessary before absorption can take place. As a result, neutrons will penetrate to greater depths in such a shield. If iron is a shield component, additional precautions must be taken because thermal neutron capture in iron produces high-energy gammas (approximately 7.5 MeV). Also, high energy neutrons are effectively slowed down in iron until the energy is about 250 keV, in which case streaming occurs. That is, inelastic scattering is rare, and elastic scattering does not reduce the neutron energy very efficiently. So, neutrons tend to just "stream" through the medium. If the neutron energy is reduced enough so that capture is more probable, a 2.2 MeV γ is produced upon neutron capture by hydrogen. This photon must also be considered in the shielding approach.

Concrete is an effective neutron shielding material because of its water content. In addition, it is cheap, strong, and can be formed in almost any desired shape and size. Also, concrete is a better γ attenuator than water because of its higher-density additives. However, it is possible for the water content of concrete to change with time and this will affect the attenuation.

For most cases, a single shielding material will not provide adequate protection. Therefore, shields are generally composed of both neutron-attenuating material and γ absorbers.

1. Shielding Approaches

The attenuation of a narrow beam of monoenergetic neutrons passing through a substance is determined from the relationship in equation 3.58,

$$\phi = \phi_o e^{-\Sigma_t x},$$

where ϕ_o is the incident fluence rate, Σ_t is the total macroscopic cross section, and x is the absorber thickness. In using Σ_t, we assume all scattered neutrons are removed from the beam.

For a wide beam, some of the neutrons which are scattered away from the point of interest are replaced by neutrons scattered toward the point of interest. Therefore, the calculated value for ϕ will be smaller than the measured value; hence, we overestimate the effectiveness of the attenuating substance.

In <u>thick</u> shields which contain sufficient hydrogen (60 kg/m²), the removal cross section μ_R is used in the equation instead of the total cross section. It may also be used for a shield composed of one substance followed by a sufficiently thick hydrogenous shield. In these cases, the removal cross section itself accounts for buildup.

When the shield is thin, even one which contains hydrogen, the scattered neutrons may contribute significantly to the fluence rate at a point of interest. Since the removal cross section is based on measurements through thick hydrogen shields, the concept does not adequately apply to thin shields.

For the case of thin hydrogenous shields, one may use an empirical expression from Casper to account for the buildup of small-angle scattered neutrons.[13] For a point ^{235}U fission source in water of source strength, 1 n/s, the absorbed dose rate of fast neutrons is:

$$\dot{D}(Gy/h) = \frac{9.68x10^{-11}}{4\pi\ x^2} x^{.349}\ e^{-[10.503x^{.698} - 3.08x]} \qquad 8.9$$

in which x is the distance (m) for a point fission source of neutrons. This may be extended to <u>any</u> hydrogenous substance, if we drop the part of the exponential term due to oxygen in water (-3.08 x), and rewrite as

$$\dot{D}\left(\frac{Gy}{h}\right)=\frac{9.68\times10^{-11}}{4\pi\, x^{2}}(kx)^{.349}e^{-10.503(kx)^{.698}} \qquad 8.10$$

in which k is the ratio of the atomic density of H in the substance to that in water. Although the above expressions are applicable to thin shields, there must still be sufficient hydrogen (approximately 7% by weight for low mass number material). The above expressions apply primarily to a fission neutron spectrum, but can be used for neutron sources which have an average energy in the range 1-2 MeV. With the proper choice of x in the above expressions, one may estimate the necessary neutron shielding for a fast neutron source with a spectrum of energies.

Reference 18 contains curves of the attenuation factors of neutrons in concrete, Nevada Test Site soil, water and polyethylene for monoenergetic neutron beam sources. In addition, this reference also has curves of the neutron tissue kerma for normally incident monoenergetic neutron beams on concrete. These curves can be useful if one has some spectrum information, since then one can compute an average transmission factor for a given neutron spectrum. That is, the average transmission factor, B, is

$$\bar{B}=\frac{\sum_{i=1}^{n} F_i(E)\overline{AF_i}(E).}{\sum_{i=1}^{n} F_i(E)} \qquad 8.11$$

in which $F_i(E)$ is the fraction of the neutron spectrum in energy group E, and $AF_i(E)$ is the average attenuation factor for the energy group. Most neutron sources one encounters consist of an energy spectrum of neutrons being emitted. To estimate the shielding and assess the dose from a neutron source are difficult tasks.

For very high energy neutrons, a multiplicity of energetic secondaries are produced in the first few mean free paths of the penetration of thick shields. Since these energetic secondaries may also produce a further number of secondaries at these energies ($E > 20$ MeV), a buildup of radiation occurs during this phase. The term nucleonic cascade is used to describe this buildup in particles during the early stages of penetration (see Figure 8.5). The region in which the buildup rises to a maximum and then decreases to the value at the start of penetration is known as the transition zone. The penetration distance of this transition zone is typically 2-4 mfp of the incident particle. Following this region, the attenuation of the beam is approximately exponential, with an attenuation length of λ(kg/m^2). This leads to a general expression for the attenuation factor, AF,

$$AF = B\, e^{-\frac{x}{\lambda}} \qquad 8.12$$

References 10 and 13 discuss the value of the buildup factor B and the attenuation length for a number of shielding substances. For a definitive treatment of the shielding aspects of high energy proton accelerators, as well as other aspects of particle accelerator health physics, the book by Patterson and Thomas[19] is suggested.

The previous discussion of neutron shielding dealt with fast neutrons. For thick shields, intermediate neutrons and thermal neutrons also must be considered. In particular, the intermediate neutrons may contribute appreciably to the transmitted dose. Capture of thermal neutrons may lead to a high capture γ dose rate. Reference 13 discusses methods of dealing with these shielding considerations.

F. Shielding Materials

Water, concrete, steel, and lead are the more common materials used for shielding purposes.

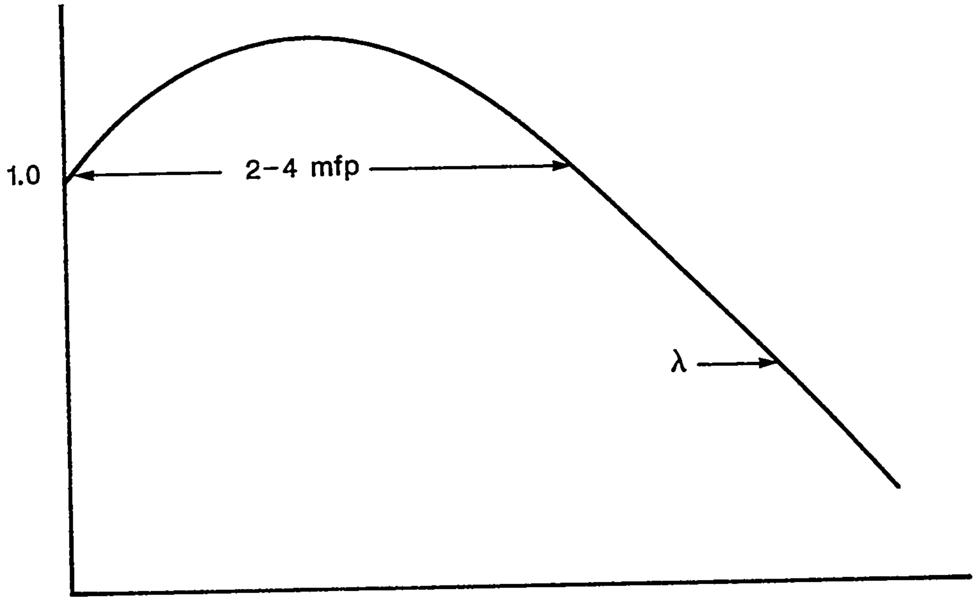


Figure 8.5 Transition zone - Radiation increases to a maximum and eventually undergoes exponential attenuation.

Some of the things which must be considered when choosing materials for a neutron shield are: (1) will it effectively decrease the neutron energy; (2) does it have a high capture cross section for thermal neutrons; and (3) will it effectively attenuate the accompanying γ radiation? In addition, radiation absorption in the shield will release energy in the form of heat. Therefore, to insure shield integrity, it may be necessary to know the temperature distribution in the shield.

Water, because of its high hydrogen content per unit volume and low cost, is widely used as a neutron shield; however, it is a relatively poor gamma absorber. In cases where a liquid is undesirable, plastics, wood, or paraffin can be used, although some of these materials are flammable. Polyethylene is probably the most commonly used plastic.

Concrete is an adequate neutron shield material if the water content is at least 7% by weight. High-density concretes are recommended where space considerations are important. A certain amount of both neutron and γ attenuation can be obtained by use of a thinner shield of high-density concrete than is possible with ordinary concrete.

Boral, a mixture of aluminum and boron carbide (B_4C), is used for neutron shielding. Boron carbide, containing 80% boron, is a good thermal neutron absorber, and the capture reaction in boron yields a relatively low-energy (0.5 MeV) γ. However, this photon contribution may be significant in some situations.

High-density materials are necessary for shielding the γ radiation. Lead is a valuable γ shield; however, uranium, tin, and iron (in steel) are also used. For γ rays of about 2 MeV, a given mass of lead will achieve approximately the same attenuation as an equal mass of iron. Above or below this energy, lead is superior to iron. However, lead has a low melting point and temperatures must be considered if this element is used for shielding. Lead also presents a toxicity problem if improperly handled.

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QUESTIONS

8.1 What are the three factors which determine the total exposure one receives in a given radiation field?

8.2 Explain how a job can be completed within safe limits when the time required for one man to complete the job would result in an excessive exposure.

8.3 What geometrical limitations are placed on the use of the "inverse square law" for penetrating radiation?

8.4 What is the first consideration in choosing a shielding material and what other considerations cannot be overlooked?

8.5 Explain why α radiation is not considered an external exposure problem.

8.6 Why do α sources need shielding?

8.7 What is the threshold energy above which β particles must be considered by shielding?

8.8 What secondary radiation is produced when β particles are stopped by shielding?

8.9 List the radiation products that result from electrons when accelerated to high energies.

8.10 When do high energy neutrons dominate in shielding consideration for electron accelerators?

8.11 In what materials is the secondary radiation produced by β shielding the greatest?

8.12 Why are high Z materials good as γ shields?

8.13 Explain the difference between μ and μ_{en} used as exponents in shield thickness formulae.

8.14 What term is used to identify the thickness of a material which reduces the intensity of a radiation by one-half?

8.15 What term is given to the variable ratio involving radiation energy, shield material, and source geometry that is used to obtain a more accurate prediction of the probable effectiveness of radiation shielding?

8.16 What term is given to that thickness of absorber which will reduce the initial beam by a factor of 1/e?

8.17 What value must be determined when a shielding material is either a compound or a mixture before using the bremsstrahlung formula in β shielding calculations?

8.18 Explain the reasons for using the buildup factor in γ shielding.

8.19 What is a point-kernel?

8.20 What is the transmission factor?

8.21 Explain the term weekly workload in reference to x ray machines.

8.22 By what processes does a neutron lose its energy in a given medium?

8.23 Explain why a single shielding material is usually unsatisfactory for neutron shielding.

8.24 What is meant by "hardening" of the neutron spectrum?

8.25 Compare iron, water and concrete as neutron shielding materials. Which is the most effective and why?

8.26 What are the two principal processes included in the total macroscopic cross section for neutrons? For the total microscopic neutron cross section?

8.27 What factor replaces the macroscopic cross section in the neutron shielding formula in the case of a thick shield containing hydrogen?

8.28 Why should the neutron flux emerging from a shield be smaller than the gamma flux?

8.29 For what reason other than being a good neutron absorber is boron used in neutron shielding?

8.30 What are some of the drawbacks of lead as shielding material?

PROBLEMS

8.1 How long can a person work in a radiation field of 7×10^{-5} C/kgh if he is allowed 5.2×10^{-6} C/kg per day? If the estimated time for the job is 15 minutes, how many persons are needed for the job?

Answers: 4 minutes 17 seconds, 4 persons

8.2 The original intensity of a 2 MeV γ beam from a point source is 7.74×10^{-2} C/kgh. Find the intensity of the beam after it emerges from a 0.06 m thick piece of lead using the formulas:

a) $I=I_o e^{-\mu x}$
b) $I=I_o e^{-\mu_{en} x}$
c) $I=I_o b e^{-\mu x}$

Assume the following for lead as a shield for 2 MeV photons: $\rho = 1.134\times10^{4}$ kg/m³, $b = 2.1$, $\mu/\rho = 4.5 \times 10^{-3}$ m²/kg, and $\mu_{en}/\rho = 2.35 \times 10^{-3}$ m²/kg.

Answers: a) 3.62×10^{-3} C/kgh
b) 1.56×10^{-2} C/kgh
c) 7.61×10^{-3} C/kgh

8.3 In problem 8.2, it is required to reduce the exposure outside the lead shield by a factor of 10,000. Use all the three formulas to estimate the thickness of lead needed.

Answers: a) 0.18 m
b) 0.35 m
c) 0.21 m

8.4 Find the approximate buildup factor using iteration and Figure 8.3, for a 2 MeV γ beam from a point source, when the original intensity is $1.3x10^{-1}$ C/kgh, and the intensity upon passing through a lead shield is $1.0x10^{-3}$ C/kgh.

Answer: 3.3

8.5 Find the effective atomic number for carbon dioxide, CO_2. Assume for the atomic numbers: Z: C=6, O=8. Note: Do not confuse the effective atomic number with the effective atomic mass of a substance.

Answer: 7.45

8.6 One of the organic scintillators used in neutron spectroscopy, stilbene, has a chemical formula $C_{14}H_{14}N_2$. Find the effective atomic number of stilbene. The atomic numbers of hydrogen, carbon and nitrogen are 1, 6 and 7, respectively.

Answer: $Z_{eff} = 5.5$

8.7 Cesium-134, $^{134}_{55}Cs$, emits β^- particles of which

a) 13% are 0.683 MeV,
b) 50% are 0.655 MeV,
c) 5% are 0.31 MeV,
d) 32% are 0.083 MeV.

Compute the bremsstrahlung produced by 10^8 Bq when the β^- particles are completely stopped in lead.

Answers: a) $2.42x10^5$ photons/s
b) $8.94x10^5$ photons/s
c) $4.23x10^4$ photons/s
d) $7.25x10^4$ photons/s

8.8 A 250 kVp x-ray therapy unit operates with a tube current of 20 mA for an average of 20 h/week. How thick should a concrete protective barrier be such that the weekly exposure in the controlled area 3 m from the tube is less than 5.16×10^{-6} C/kg? Use a use factor of 1/2 and occupancy factor of 1. What would be the thickness, if the wall were to be made of lead sheet? Note: use Figure 2 and Figure 3 of Appendix D, NCRP Report No. 49.

Answer: Concrete thickness - 0.45 m
Lead thickness - 0.01 m

8.9 Compute the remaining fluence rate of a narrow beam of 1 MeV neutrons passing through a foil of cadmium 1×10^{-3} m thick. Data for cadmium: $\sigma_t=6.2$ barns, $\rho=8.65\times10^{3}$ kg/m^3, A = 112.40

Answer: 97.2%

8.10 Using the data in the preceding problem except that the neutrons are thermal for which cadmium presents a total cross section of 2500 barns.

Answer: ~ 9.3×10^{-6} or 0.

8.11 The atomic density of H in concrete is 1.375×10^{28} atoms/m^3, and in water, 6.69×10^{28} atoms/m^3. Compute the absorbed dose rate from a 5×10^{5} n/s point fission source, located behind a 0.4 m thick concrete wall. Assuming Q=20, what is the dose equivalent rate?

Answer: 1.6 μGy/h
32 μSv/h

SECTION 9 - INTERNAL DOSIMETRY CONSIDERATIONS

A. Main Factors Affecting Dose Calculations

Much effort has been directed toward the problem of calculating the dose which a person receives as a result of internally deposited radionuclides.[1-4] Among the many factors of main concern that enter into such a calculation are: the shape of the organ, the type of radiation, the amount of the deposit, and the distribution of the deposit.

When the organ is in the shape of some standard geometric form, such as a sphere, the problem is somewhat lessened. If one can assume a uniform distribution of the deposit, the problem is further reduced. Then, if the range of the radiation is small compared with the size of the organ, the calculation of the absorbed dose becomes rather simple. One can then say that the energy emitted per unit time is equal to the energy absorbed per unit time. A knowledge of the concentration (Bq/kg), decay scheme of the radionuclide, and the energy of the emitted radiation is all that is needed to complete the calculation.

It is rare when one is able to apply this ideal model to an actual case. For this reason, one cannot make a truly precise calculation of the organ dose. The most vital factor which affects the accuracy of the result is the nonuniform distribution of the deposit. This is clearly seen in the case of alpha particles, whose range in tissue is so very short. In this respect, local "hot" spots may be the item of chief concern, rather than the average dose over the tissue mass.

Of further interest is the problem of stating the dose in terms of a common scale. One can make a reasonable estimate of the absorbed dose in an organ. However, this merely specifies the energy absorbed and does not supply information about the effect of this dose on the organ. The absorbed dose which will give the same biologically equivalent dose is not the same for all types of radiation. Also, the lack of knowledge about the effects from low dose rates over long exposure times adds to the complex nature of the problem.[5]

For purposes of protection work, the present method of dose determination, proposed by the ICRP in 1979,[1] provides an adequate estimate. One must keep in mind that as knowledge is gained, more changes may be warranted. But the changes which may occur will only serve to reflect an attempt to obtain more precise results.

Given an amount of a radionuclide in an organ of the body, a reasonable estimate of the organ dose can be made. A much greater problem is to estimate the amount of a radionuclide in the organ which results from intake into the body. This involves a number of biological factors, many of which are not quantitatively known. Much work is needed in this area to improve the body of knowledge which now exists. At present, many assumptions must be made in order to relate an environmental concentration to which a person is exposed to the amount of the radionuclide which he receives as a result. In view of this fact, the ICRP has always taken the stand that review and modification are needed on a continuing basis. The present recommendations[1] define a quantity called the Annual Limit on Intake (ALI). The ALI is the activity of a radionuclide which taken alone would irradiate a person to the limit set by the ICRP for each year of occupational exposure.

The method of calculation suggested in ICRP 30 is still not very precise. It can only yield a good estimate in cases where there are sufficient data to back up the assumptions made.[3,4] But the value of these calculations lies in the fact that they provide a guide for evaluation of a given environmental concentration. For the purposes of protection work, there is a very special need for these guidelines. for this reason, the methods of estimating the derived air concentration (DAC) values are useful. The DAC is found by dividing the ALI for a given radionuclide by the volume of air inhaled in a working year (2400 m^3).[1] On the other hand, one must be careful to note the conditions under which these values apply. In many cases, the use of these values without due regard to the assumptions which underlie them can lead to many pitfalls.

Let us focus on the nature of the problem. Assume an intake of a radionuclide which deposits in one organ and we wish to determine the dose. A knowledge of the mass, size, and shape of that organ is needed. For a certain radionuclide of known decay scheme and energy of emissions, one can calculate an initial organ burden. This will be the amount of the radionuclide in the organ which delivers the stated dose equivalent limit (see Section 7.C.3.a) for that organ. This limit may be based on the risk of stochastic effects or nonstochastic effects.[6]

There are many ways in which the radionuclide may reach this organ: inhalation, ingestion, diffusion through skin, and absorption through wounds or punctures. For most cases, only one mechanism is of concern for occupational exposure: that is, inhalation. Ingestion may be important in the considerations of the internal dose to the general public.[4] The problem then is to relate a concentration in air (Bq/m^3) to the amount of the radionuclide which ends up in the organ. Many factors affect the amount of the radionuclide which is deposited in a given organ.

From the standpoint of absorption into the body, the transportability (solubility) of the radioactive substance in body fluids plays a major role. Whether a given substance will be readily transported in body fluids or not is not easy to tell. However, as a mode of entry, ingestion is of concern only for transportable (soluble) radionuclides. Also, in the case of elements not required by the body, absorption by ingestion is poor, leading to rapid elimination of most of the material.

In those cases where absorption does occur, one is then concerned with the transfer of the substance to the organ, that is, only a fraction of the ingested material will go into the body fluids (blood stream). Of this, only a fraction will go into the organ of concern. These fractions are often poorly known. To add to the problem, a portion of the substance may end up in a large number of organs. In this case, more than just a single organ will have to be considered. Then, an organ will be irradiated not only by the radioactivity in that organ, but may also be irradiated by the radioactive emissions in nearby organs.

In regard to inhalation, both transportable and non-transportable matter must be treated. Other factors to consider are density and particle size. The movement and retention of matter in the respiratory tract are functions of the particle size and dissolution rate.[7] Because of the complex relationships which exist, estimates of the transfer of radioactive matter by this mode of entry are poor. Of that released from the lung, a fraction will get into the body fluids (transfer compartment) and then be transported to a given organ.

Until recently, the ICRP used a general model for use in the calculations.[8] In that model, deposition in the lung was based upon classification of materials as soluble or insoluble, with little regard for particle size and chemical form effects on clearance rates. A more sophisticated model[7] utilizing dust deposition data and clearance fractions for portions of the respiratory tract has been used in the ICRP 30 methodology (see 9.D.3).

One can see that many factors are already present in an attempt to state intake parameters. But one must also treat the matter of elimination; although a substance is deposited in an organ, the time of stay in that organ is not indefinite. A fraction of this substance will be eliminated from the organ as time passes. Because the substance is radioactive, a certain fraction will decay per unit time, but also some of the material will be lost from the organ because of biological processes. Therefore, one must have knowledge of the net retention of a material in a given organ in order to proceed with the calculation. This means that one must have an elimination model for substances which are taken up by a given organ. The ICRP assumes that each organ can be treated as either a separate compartment or a number of compartments in which the elimination of a substance from each compartment proceeds at a constant rate. This exponential model for elimination means that each organ compartment has a half life for biological elimination of a given substance. This is only very roughly true and in some cases a power series or some other mathematical representation may be a better model. However, it provides a simple means of obtaining a rough estimate of the retention in a certain organ.

Given the yearly intake rate for air (2400 m^3 occupational), one can then solve for the DAC value. The calculation consists of finding the daily intake, over the working year, which would not result in an accumulation greater than the ALI. Then, the ICRP dose limit will not be exceeded. For the purposes of this methodology, it is immaterial whether the ALI is spread out over the year or acquired in a single incident within the year,[4] except in the case of occupational exposure of women of reproductive capacity and pregnant women.[6]

B. Reference Man

Throughout the above outline of the calculation of the DAC value, and its basis, many factors which concern biological processes in man were mentioned. Because one individual varies from the next, these factors differ from one person to the next. In order to obtain agreement throughout the world in regard to the calculations of internal dose, certain values have been agreed upon for these factors. The ICRP achieves this goal by the use of the "Reference Man" concept.[1] The Reference Man represents a set of agreed-upon values for the many characteristics of man which are needed for internal dose calculations. These values can be found in ICRP Report 23.[9] Since the values are intended to represent an average adult, no account is taken of the differences which occur among individuals. The published DAC values[1,10-12] have been calculated using the Reference Man data, dosimetric data for the relevant radionuclides contained in References 13-16, and the metabolic data for these radionuclides found in References 1, 10 and 11.

C. Internal Exposure - ICRP 2 Model

Previous estimates of internal dose equivalent were based upon the ICRP Publication 2 model,[8,17] in which it was assumed that the organ retention could be represented by a single exponential term, that a specific organ could be considered to be the critical organ, that physical characteristics of the model (e.g., intake parameters, transfer fractions, tissue size and weight) could be represented by the "Standard Man" data, that spherical geometry could be assumed for organ shape, and that scattered radiation could be ignored. This basic model was used to generate inhalation dose factors that are contained in USNRC Regulatory Guide 1.109.[18] For the basic dosimetry model, it was assumed that uniform deposition of the radionuclide in the organ occurs and that the energy emitted is equal to the energy absorbed, modified by a correction factor for the escape of photon energy from an organ of small dimensions. Integration of the dose-rate equation over a suitable time interval yielded the dose equivalent, H, delivered by the radionuclide deposited in the organ for the stated period. The system of dose limitation required that the specified annual dose equivalent to the critical organ not be exceeded.

The intake of radionuclides was then limited by establishing "Maximum Permissible Concentration" (MPC) values in air and water that would ensure that the dose-equivalent rate in the critical organ would not exceed that allowed by the dose-rate limitation value over a 50-year intake period. Values of the parameters needed in these calculations for the various radionuclides are contained in ICRP Publication 2, along with the MPC values. For purposes of dose limitations, the ratio of a given concentration to the MPC value for a specific radionuclide was considered as approximately equal to the ratio of the respective dose rates. This was generally true for radionuclides with relatively short retention times in the body and increasingly conservative for those radionuclides with longer retention times.

The dosimetric model incorporated the assumption that the radionuclide is uniformly deposited in an organ of spherical shape. For a large enough radius of the organ, one could assume that the energy emitted is equal to the energy absorbed. For an organ of small dimensions, the previous assumption must be modified by a correction which accounted for the fraction of photon energy that escapes. For this model, the dose equivalent rate $\dot{H}(t)$ was given by:

$$\dot{H}\ (t) = \frac{51.2\ \Sigma EF(RBE)n}{m}\ q(t)\quad \left(\frac{rem}{d}\right) \qquad 9.1$$

where q(t) is the activity (μCi) of the radionuclide in the organ at some time t, as determined by a single exponential retention function, ΣEF(RBE)n is the effective absorbed energy per transformation in the organ (MeV/dis), including the correction for escaping radiation, and m is the organ weight (g). Values of the parameters needed for the various radionuclides are contained in ICRP Publication 2.

Equation 9.1 indicates that the dose equivalent rate, $\dot{H}(t)$, is directly proportional to the activity of the radionuclide present at any time in the organ. To limit the organ dose rate, the buildup of activity must be limited. The supply rate was limited such that the uptake rate of the radionuclide by the organ balances the elimination rate at the end of the 50-year continuous intake period. At this point, the activity in the critical organ delivers the allowable organ dose equivalent rate.

For radionuclides with short retention times (rapid elimination rates), the activity builds up quickly in the organ and rapidly approaches an equilibrium value during continuous exposure. For these cases, the activity in the organ will be approximately proportional to the MPC for the radionuclide, and thus the MPC will be approximately proportional to the permissible organ dose equivalent rate. The ratio of the dose equivalent rate for some other concentration of the given radionuclide to the permissible organ dose equivalent rate will be the same as the ratio of that concentration to the MPC. This was the basis for using the published MPC values as an index of the organ dose equivalent rate delivered by the radionuclide.

Some present standards are based upon the ICRP-2 model and application of this methodology will be required until the regulatory agencies officially adopt the new ICRP-30 models. However, the ICRP-30 model represents a revision and updating of the older ICRP-2 model, particularly with respect to the metabolic data, and its use as a model for internal exposure can be expected to be adopted. The NCRP[4] has recommended adoption of this new methodology, and proposed revisions of USDOE, USEPA, and USNRC regulations have included the basic methodology of ICRP 26 and 30.

D. Internal Exposure - ICRP 30 Model

The ICRP has changed both its basic recommendations and revised the system of dose limitation. These revisions, as contained in ICRP Publication 26,[6] reflect the availability of sufficient data about the effects of radiation for the Commission to estimate the risk per unit dose equivalent (H) with respect to fatal cancer in exposed people and to serious disease in the offspring of exposed people. Cancer and hereditary effects are referred to as stochastic effects (see Section 5.H) and the risks are assumed to be directly related to the dose equivalent, without threshold. So the probability of the effect occurring, rather than its severity, is a function of H. Other effects, called nonstochastic, are those in which the severity of the effect varies with H.

For the revised dosimetry model used in ICRP Publication 30, Part 1,[1] it is assumed that organ retention can be represented by one or more exponentials, that the critical organ concept no longer applies and that one must account for contributions to the dose in one organ due to photons which are emitted from other body organs containing the given radionuclide, and that the physical characteristics and other parameters of the model can be represented by the "Reference Man" data.[9] Spherical and other geometrical shapes are assumed for appropriate representations of body organs, and the scattered radiation contribution is accounted for by Monte Carlo calculations of the fraction of absorbed photon energy. More recent radiation transformation data for radionuclides[16] and metabolic data for elements and their compounds have been utilized. Revisions to the ICRP 2 models for the dosimetry of the respiratory tract, digestive system and bone are also used.

In this model, it is assumed that deposition in an organ is uniform. The total dose equivalent averaged throughout the tissue mass over 50 years after an intake is computed; then, an annual limit of intake (ALI) is determined for the particular radionuclide, whether for inhalation or ingestion. Values of the ALI for inhalation and ingestion can be found in the ICRP 30 Reports, Parts 1, 2 and 3, as well as the Supplements to these reports.[1,10-15]

If intake is only by inhalation or only by ingestion, the ICRP recommendations for dose limitation will be satisfied if the intake by either mode is less than the ALI for that mode. If both modes are involved, as well as several radionuclides contributing to each mode, the dose limitation will be met if:

$$\sum_i \left(\frac{I_i}{ALI_i}\right)_{ING} + \sum_j \left(\frac{I_j}{ALI_j}\right)_{INH} \leq 1 \qquad 9.2$$

where I_i is the total annual intake by ingestion of radionuclide i and I_j is the total annual intake by inhalation of radionuclide j. The terms $(ALI_i)_{ING}$ and $(ALI_j)_{INH}$ represent the respective ALI values for ingestion and inhalation.

In the use of the above inequality, one should be cautioned that the values of ALI for ingestion in the ICRP reports are for the <u>total</u> contribution from all ingestion pathways, such as food, drinking water, etc. To satisfy the Commission's limits, one must determine the contribution from each pathway and sum the results, which then can be compared to the ALI for ingestion of that radionuclide. Reference Man data[9] can be used to estimate the consumption rates for the various ingestion pathways, and the intake ($I_{n,i}$) for a given pathway n is the product of the radionuclide concentration (Bq/kg) for that pathway and the yearly consumption (kg), and

$$I_i = \sum_{n} I_{n,i} \qquad 9.3$$

For example, assume an individual has an annual intake of 10^5 Bq of a radionuclide whose ALI for inhalation is 10^6 Bq. In addition, the individual drinks water (730 ℓ/y) containing an average concentration of the same radionuclide of 5 kBq/ℓ, and consumes foodstuffs (520 kg/y) with a concentration of 100 Bq/kg. The ALI for ingestion is 7x 10^6 Bq. Will the ICRP dose limit be satisfied?

Using equation 9.3,

$$I_i = \sum_{n=1}^{2} I_{n,i} = 5\text{x}10^3\left(\frac{\text{Bq}}{\text{l}}\right)730\left(\frac{\text{l}}{\text{y}}\right) + 100\left(\frac{\text{Bq}}{\text{kg}}\right) 520\left(\frac{\text{kg}}{\text{y}}\right)$$

$$= 3.702\text{x}10^6 \text{ Bq.}$$

This is the amount ingested. Then, using equation 9.2,

$$\frac{3.702\text{x}10^6}{7\text{x}10^6} + \frac{10^5}{10^6} = 0.63 \le 1,$$

so the dose limit is satisfied.

An obvious extension of the method used above can be made for the case involving external exposure as well as inhalation and ingestion. The inequality would then be stated

$$\sum \left(\frac{H_{I,k}}{H_{WB,L}}\right)_{EXT} + \sum_i \sum_n \left(\frac{I_{n,i}}{ALI_i}\right)_{ING} + \sum_j \left(\frac{I_j}{ALI_j}\right)_{INH} \le 1 \qquad 9.4$$

$H_{I,k}$ is the deep dose-equivalent index (defined as the maximum value of H that would occur in a 30 cm diameter tissue sphere [see Section 12.A.3.d)] for the kth contributor to the external exposure, and $H_{WB,L}$ is the annual dose equivalent limit for uniform whole body irradiation, 0.05 Sv (5 rem) - see Section 7.C.3.a.

1. Dose Limits

As discussed in Section 7.C.3, the dose equivalent limits recommended by the ICRP are intended to prevent nonstochastic effects and limit the occurrence of stochastic effects to an acceptable level. To this end, an annual limit for occupational exposure of 0.5 Sv (50 rem) to all tissues [except 0.15 Sv (15 rem) to the lenses of the eyes][10] is deemed sufficient to prevent nonstochastic effects. This limit applies irrespective of whether the tissues are exposed singly or together with other organs. For stochastic effects, the limit on risk should be equal whether uniform whole body irradiation or non-uniform irradiation of several organs occurs. For non-uniform irradiation, the relationship expressed by equation 7.1 should be met:

$$\sum_{T} w_T H_T \le H_{WB,L},$$

where w_T is a weighting factor given by the ratio of the stochastic risk in tissue (T) to the total risk for uniform whole body irradiation (see Table 7.1), and H_T is the annual dose equivalent received by tissue (T). In ICRP Publication 30, Part 1, the limits for intake of radionuclides are set by satisfying both of the conditions:

$$_{T}\; w_T H_{50,T} \le 0.05 \text{ Sv} \qquad 9.5$$

and

$$H_{50,T} \le 0.5 \text{ Sv}, \qquad 9.6$$

where $w_T H_{50,T}$ is called the weighted committed dose equivalent (committed effective dose equivalent) and $H_{50,T}$ is the committed dose equivalent, which is the total dose equivalent averaged through a tissue (T) in the 50 years following the intake of material in the given year.[1]

EXAMPLE Assume that the lung, bone surfaces, red marrow and kidney all receive committed dose equivalents of 0.15 Sv in a year. Are the basic limits met?

By inspection of equation 9.6, we see that the committed dose equivalent is not exceeded. Utilizing information from Table 7.1, in equation 9.6, we have

$$\Sigma\, w_T\, H_{50,T} = .12(.15)+.03(.15)+.12(.15)+.06(.15)$$

$$= 0.0495$$

which is < 0.05 Sv and the limit is met.

It should be noted that in DOE Order 5480.11, the DOE proposes a modification of the ICRP system to arrive at an annual dose equivalent. The dose equivalent received in a year by a given organ from internal exposure is multiplied by the organ weighting factor w_T, and this product is added to the external effective dose equivalent for that year to arrive at an annual effective dose equivalent.

This procedure follows the NCRP recommendation that the committed effective dose equivalent not be used for the evaluation of the consequences of radiation exposure in individuals.[4] One may use the anticipated or prospective committed effective dose equivalent estimated for some planned practice for the purpose of design and control of the workplace. However, to demonstrate compliance with the exposure limits, the actual or retrospective assessed organ dose equivalent received by the individual during the year is used.

2. Dosimetric Model

The basic dosimetric model assumes uniform deposition in an organ. The total dose equivalent averaged throughout the tissue mass over 50 years after the intake is computed from:[1]

$$H_{50} = \sum_i \frac{\int^M D_{50,i}\, Q_i \, dm}{\int^M mdm} = \Sigma\, Q_i\, \bar{D}_{50,i}, \qquad 9.7$$

in which $D_{50,i}$ is the total absorbed dose during 50 years following intake averaged throughout the tissue mass, M, for each radiation type i. The quality factor, Q_i, for radiation of type i has one of three values in the ICRP model.[1] These are:

Q=1 for β particles, electrons and all electromagnetic radiation including γ, x rays and bremsstrahlung.

Q=10 for fission neutrons emitted in spontaneous fission and protons (Note that the ICRP has recommended Q=20 for fast neutrons).[19]

Q=20 for α particles from nuclear transformations, for heavy recoil particles and for fission fragments.

When contributions for each radiation of type i from radionuclide j are summed, and if a number j of radionuclides, such as in a mixture, are contributing and need to be summed, and if several other source organs (S) are irradiating the given organ (T), the general expression becomes:

$$H_{50,T} = 1.6\text{x}10^{-10} \sum_S \sum_i \left[U_s \;{}_i\; \text{SEE } (T \leftarrow S)_i \right]_j \text{ Sv}, \qquad 9.8$$

in which U_S is the number of transformations of j in a source organ S over 50 years following unit intake as obtained by integration of the retention function and SEE (MeV/g-trans) is the <u>specific effective energy</u> term for radiation of type i, absorbed in T for each transformation in S, modified by the quality factor. Values of $w_T H_{50,T}$, $H_{50,T}$, U_S and SEE (T←S) have been tabulated for radionuclides of 94 elements in the

Supplements to the various parts of ICRP Publication 30.[13-15] These values are given for both inhalation and ingestion.

The limit on intake of a particular radionuclide is established by solving the following inequalities:

$$I \sum_{T} w_T \, (H_{50,T} \text{ per unit intake}) \leq 0.05 \text{ Sv} \qquad 9.9$$

and

$$I \, (H_{50,T} \text{ per unit intake}) \leq 0.5 \text{ Sv}, \qquad 9.10$$

whether for inhalation or ingestion. The annual limit on intake (ALI) is the greatest value of I which satisfies both of the above inequalities. The ALI[1] is defined as the activity (Bq) of a radionuclide which taken alone would irradiate Reference Man to the limit set by the ICRP for each year of occupational exposure. Values for the ALI for inhalation and ingestion can also be found in the Supplements to ICRP 30.[13-15]

The values of ALI developed in ICRP Publication 30 are based upon an occupationally exposed adult. The data and models described in the report are not recommended by the ICRP for estimating the dose equivalent (H) to members of a population based solely on the differences in mass of organs or magnitude of intake. At the end of Chapter 9 of ICRP-30, Part-1 (Reference 1), a bibliography has been included concerning the methodology of estimating H for different age groups.

With respect to members of the public, the ICRP recommends the application of the appropriate dose equivalent limit to the weighted mean whole-body dose equivalent of the critical group (that group expected to receive the highest dose equivalent), see paragraph 85 of ICRP Publication 26.[6] In the calculation of the dose equivalent from intake of radionuclides, the metabolic differences between children and adults need to be taken into account.

In the data given in the ICRP 30 report for $H_{50,T}$, ALI and DAC, allowance is made for the committed dose equivalent contributed by daughter buildup in the body from parent decay. The metabolic behavior of all the radioactive daughter products is assumed to be the same as that of the parent. In the dosimetric data for any radionuclide, values are given for U_s for the parent and for the daughters which have built up in the body for the 50 years following intake.

3. Respiratory Model

For inhalation, ALI values are listed for three classes of material based upon their relative retention in the pulmonary or deep section of the lung-D, W or Y. The classification is based upon a range of half times: $D < 10$ d, 10 d $< W < 100$ d and $Y > 100$ d. These categories represent increasing retention with respect to half times, and the metabolic data, found in ICRP Publication 30, Parts 1, 2 and 3,[1,10,11] classifies certain chemical forms of the material in terms of these categories. When the chemical form of the material is known, the proper choice of category--D, W or Y--can be made; otherwise, a conservative choice should be made.

The mathematical model used in the methodology is shown in Figure 9.1, and is taken from ICRP 30, Part 1.[1] The model originally was developed by the Task Group on Lung Dynamics[7] but has undergone changes in the values of some of the parameters over the years. Three major deposition regions are defined: the nasal-pharyngeal, tracheo-bronchial and the pulmonary region. The fractions initially deposited in these sections are D_{N-P}, D_{T-B} and D_P, respectively, based upon an aerosol particle size of 1 μm. Deposition estimates for other sizes can be made using information in Reference 1.

As seen from the drawing in Figure 9.1, each of the three sections is divided into compartments. These compartments are associated with certain clearance pathways and have a defined half time T for clearance with an associated fraction F of the material removed by that clearance pathway. Either T, or F, or both may have a different value as the retention category changes from D to Y. Also, for some retention categories, some of the compartments are not applicable. Compartments a, c, e

		Class					
		D		W		Y	
Region	Compartment	T day	F	T day	F	T day	F
N-P (D_{N-P} = 0.30)	a	0.01	0.5	0.01	0.1	0.01	0.01
	b	0.01	0.5	0.40	0.9	0.40	0.99
T-B (D_{T-B} = 0.08)	c	0.01	0.95	0.01	0.5	0.01	0.01
	d	0.2	0.05	0.2	0.5	0.2	0.99
P (D_P = 0.25)	e	0.5	0.8	50	0.15	500	0.05
	f	n.a.	n.a.	1.0	0.4	1.0	0.4
	g	n.a.	n.a.	50	0.4	500	0.4
	h	0.5	0.2	50	0.05	500	0.15
L	i	0.5	1.0	50	1.0	1000	0.9
	j	n.a.	n.a.	n.a.	n.a.		0.1

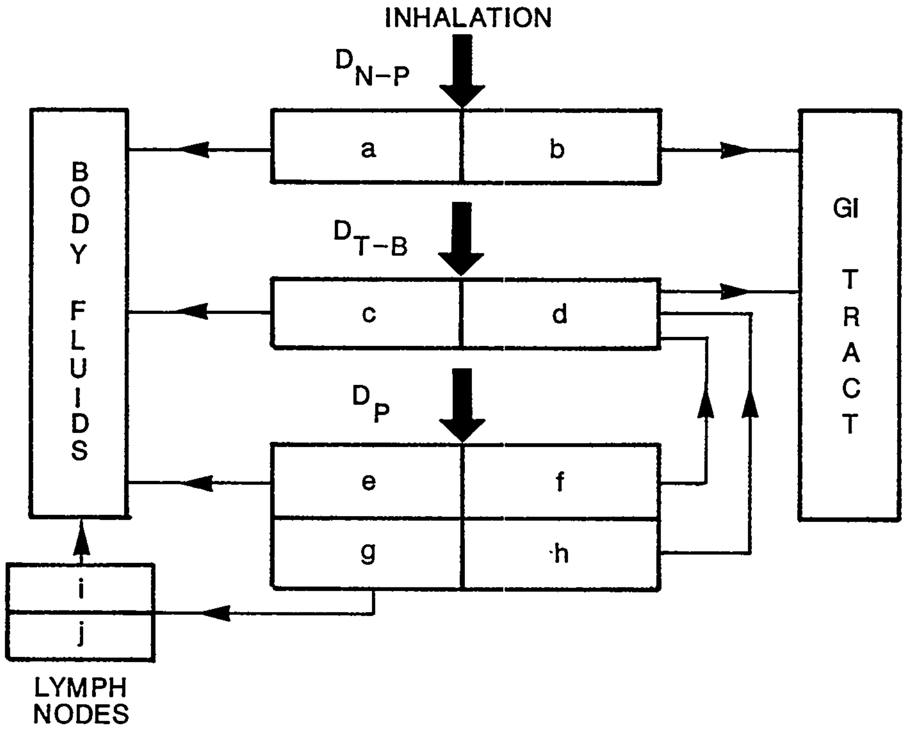

Figure 9.1 Mathematical model used to describe clearance from the respiratory system. The values for the removal half times $T_{\alpha-i}$ and compartmental fractions, $F_{\alpha-i}$ are given in the tabular portion of the figure for each of the three classes of returned material. (Reprinted with permission from Annals of the ICRP, Vol. 2, No. 314, ICRP Publication 30, Part 1, Limits for Intakes of radionuclides by Workers, Copyright 1979, Pergamon Press, Ltd.)

represent direct transfer to the body fluids (transfer compartment) by which some material may be transported to systemic organs and some excreted. Compartment h represents an indirect transfer to the body fluids through the lymph nodes. Also, for a Class Y aerosol, only a portion of the material (namely that from compartment i) is transferred by this pathway. The remainder is retained indefinitely in compartment j. The remaining compartments: b, d, f and g, transfer material to the gastrointestinal tract (GI tract). The metabolism of a radionuclide once it reaches either body fluids or the gastrointestinal tract is then governed by the metabolic model.[1,10,11] Some material which goes through the GI tract may be routed back to the transfer compartment and then on to the systemic organs.

4. Gastrointestinal Tract Model

For ingestion, the ALI is based upon the fraction of material which is transferred from the GI tract to the systemic system (f_1). These values are also listed in the metabolic data and associated with the above categories. Again, in the absence of specific knowledge, one should choose a conservative value.

The model for the GI tract is based upon a biological model originally put forth by Eve.[20] The GI tract is assumed to consist of 4 sections, each section being a single compartment with a defined half time for transfer of material. This model is shown in Figure 9.2, which is adapted from Reference 1. Included in the figure are the data from ICRP 23[9] for the mass of the walls and contents of the various components of the system.

During ingestion, material passes through the four sections of the GI tract: stomach, small intestine, upper large intestine and lower large intestine. Each section is considered as a single compartment, so the entire system may be treated as a linear chain of first order differential equations. A portion of the material is assumed to be absorbed from the small intestine to the body fluids, shown by the transfer rate constant λ_B in Figure 9.2. The particular values of

Section of GI tract	Mass of walls (kg)	Mass of contents (kg)	Mean residence time (day)	day^{-1}
Stomach(ST)	0.15	0.25	1/24	24
Small Intestine(SI)	0.64	0.40	4/24	6
Upper Large Intestine(ULI)	0.21	0.22	13/24	1.8
Lower Large Intestine(LLI)	0.16	0.135	24/24	1

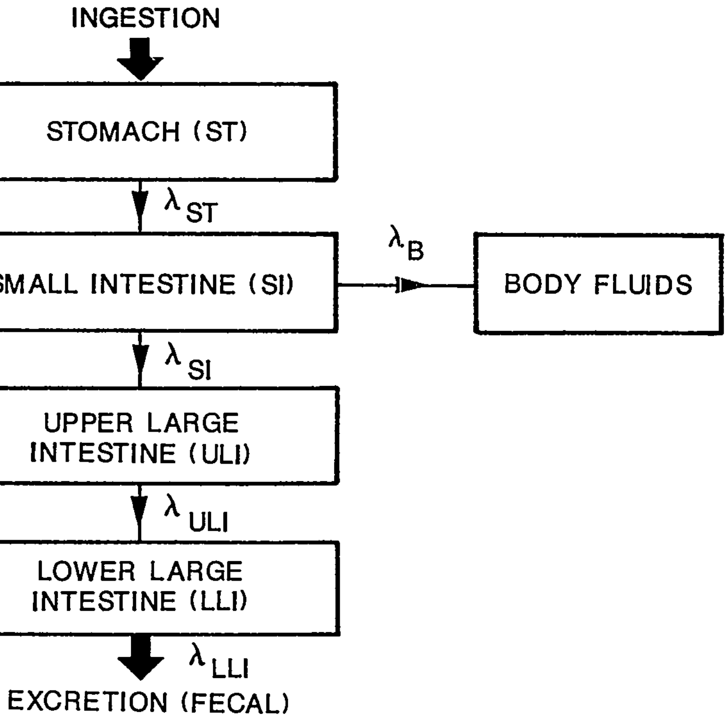

Figure 9.2 Mathematical model used to describe the kinetics of radionuclides in the gastrointestinal tract. (Reprinted with permission from Annals of the ICRP, Vol. 2, No. 3/4, ICRP Publication 30, Part 1, Limits for Intakes of Radionuclides by Workers, Copyright 1979, Pergamon Press Ltd.)

λ_B, are estimated from the metabolic parameter f_1, which is the fraction of the stable element which reaches the body fluids following ingestion. The relationship used is

$$\lambda_B = \frac{f_1 \lambda_{SI}}{1-f_1}, \qquad 9.11$$

in which $\lambda_{SI} = 6d^{-1}$ from the tabular data in Figure 9.2, and f_1

is obtained from the metabolic data for the particular compound of the radionuclide. Of the material in the transfer compartment (body fluids),some is translocated to organs in the systemic system, as designated by the metabolic model, and the remainder is excreted. Material which is not cleared from the GI tract is excreted in the feces.

5. Systemic Model

When an inhaled or ingested radionuclide has been transferred to the body fluids (transfer compartment), its subsequent transfer to the compartments of the various tissues and organs of the systemic system is indicated schematically in Figure 9.3, taken from Reference 1. Whatever time is taken to transfer from a deposition site (lungs, GI tract) to a tissue by means of the body fluids, is represented by transfer compartment a. Unless otherwise stated in the metabolic model, the clearance half time for this compartment is 0.25 day. Each tissue which receives the radionuclide will consist of at least one, and maybe more, compartments.

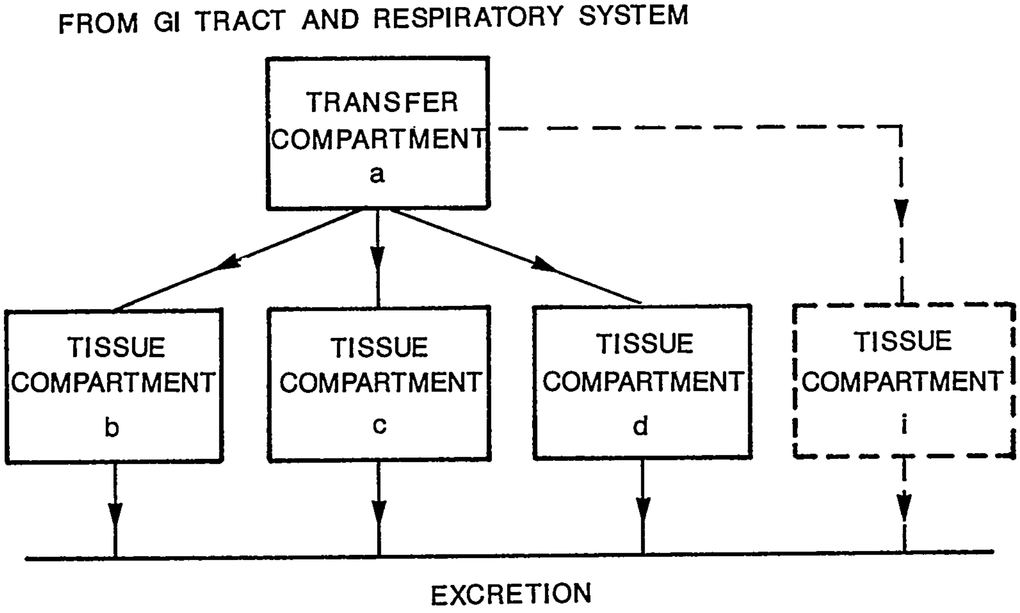

Figure 9.3 Mathematical model usually used to describe the kinetics of radionuclides in the body. (Reprinted with permission from Annals of the ICRP,Vol. 2, No. 3/4, ICRP Publication 30, Part 1, Limits for Intakes of Radionuclides by Workers, Copyright 1979, Pergamon Press Ltd.)

Each compartment, in turn, has an associated elimination rate for excretion of the radionuclide. In this model, it is usually assumed that there is no feedback, or recycling, to the transfer compartment, either from tissue compartments or excretion routes. So, the metabolic models, with the exception of the alkaline earths, may be expressed as systems of first order differential equations with constant coefficients.

The final compartment model for a given radionuclide may be quite complex from the standpoint of the total number of compartments involved. Figure 9.4 shows the ICRP-30 Pu model for a Class Y aerosol, schematically. In this model, material entering the transfer compartment is distributed mainly to liver and bone ($F_L = F_B = .45$), with the fraction .10 going to all other tissues and early excreta. The fraction going to gonads is taken as $3\text{x}10^{-4}$ and $1\text{x}10^{-4}$ for males and females, respectively.

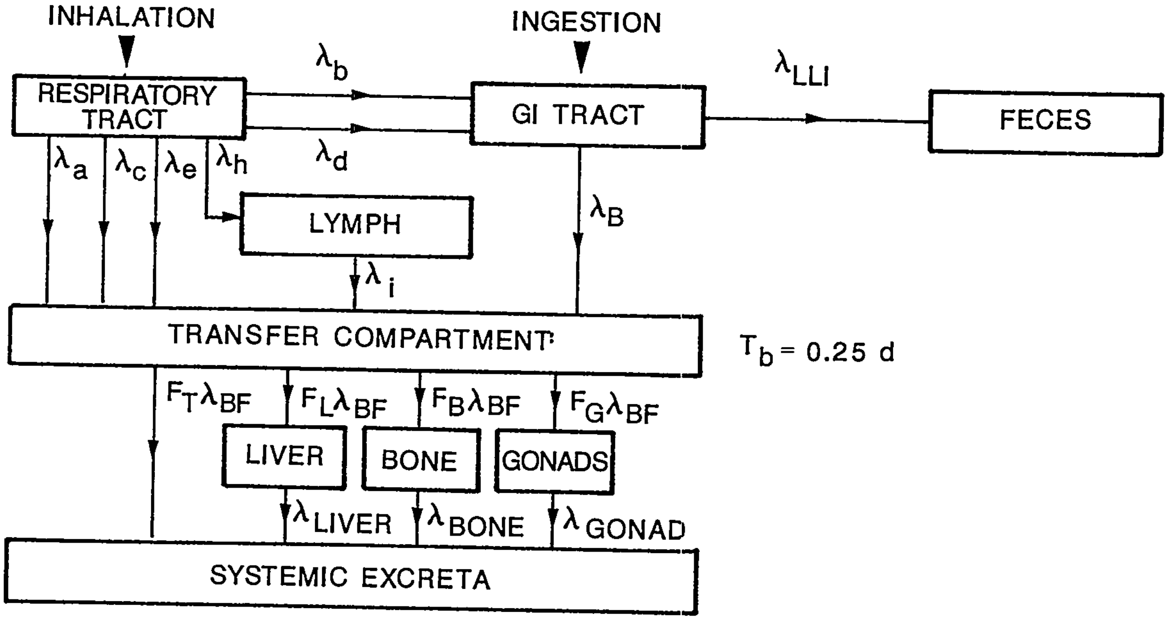

Figure 9.4 ICRP-30 Pu metabolic model, Class Y aerosol for inhalation or Class Y for ingestion. (Reprinted with permission from Annals of the ICRP, Vol. 2, No. 3/4, ICRP Publication 30, Part 1, Limits for Intakes of Radionuclides by Workers, Copyright 1979, Pergamon Press Ltd.)

With respect to the model in Figure 9.4, let us look at the case of inhalation. For a single intake of unit activity (1 Bq), the initial amount of material in lung compartment x is

$$q_x\ (0) = D_x F_x, \qquad 9.12$$

in which D_x corresponds to $D_{N\text{-}P}$, $D_{T\text{-}B}$ or D_P, depending upon the location of compartment x. F_x is the fraction of the inhaled material of the appropriate aerosol (Class D, W or Y) which deposits in compartment x. The differential amount retained $q_x(t)$ is given by

$$\frac{dq_x(t)}{dt} = -\ (\lambda_x + \lambda_r)\ q_x(t). \qquad 9.13$$

in which λ_x is the biological removal rate constant for compartment x and λ_r is the radioactive decay constant for the particular radionuclide. Solution of the above, for $q_x(0) = D_x F_x$ gives

$$q_x(t) = D_x F_x\ e^{-(\lambda_x + \lambda_r)t}. \qquad 9.14$$

The contribution from the N-P region is not included in the ICRP respiratory dosimetry model, but the model does include the lymph compartment contributions. The retention function for the lung then, is the sum of the retention functions for compartments c through j (see Figure 9.1), for dosimetry purposes. This leads to a retention function containing 6 exponential terms for a Class Y material.

The material in the lung serves as a continuous supply source for the body fluids, lymph nodes and GI tract. The contribution from a given lung compartment x, to the body fluids (BF) or transfer compartment may be expressed as

$$\frac{dq_{BF}(t)}{dt} = \lambda_x q_x(t)\ -\ (\lambda_{BF} + \lambda_r)q_{BF}(t), \qquad 9.15$$

which yields the solution

$$q_{BF}(t) = \frac{\lambda_x D_x F_x}{\lambda_{BF}-\lambda_x} \; e^{-(\lambda_x + \lambda_r)t} - e^{-(\lambda_{BF} + \lambda_r)t} \qquad 9.16$$

when $q_{BF}(0) = 0$. Referring to Figure 9.4, additional sources of supply are from the lymph and GI tract. Each of these sources yields a contribution similar to that expressed in equation 9.16. Following combination of similar terms, the retention function contains 9 exponential terms for a Class Y material.

In turn, the body fluids act as a continuous supply source for the organs of the systemic system. The contribution from the body fluids to a particular organ will yield a differential equation similar in form to 9.15. Due to the chain nature of the model, a recursive relationship may be applied to successive differential equations in the chain to aid in generating the solutions.[21]

The retention function for an individual organ; say, bone in the case of ^{239}Pu and Class Y material, consists of the sum of ten exponential terms. Although expressions of this size become somewhat cumbersome for hand calculations, computer programs can perform the necessary computations swiftly and easily.

The final form of the retention function for a given organ relates the time course of the material in the organ back to a unit intake of the material. Integration of the retention function, over a 50-year time period following the intake, yields the total disintegrations, U_s, of the radionuclide in that organ (see 9.D.2).

6. Bone Dosimetry Model

The ICRP 30 model for bone deposition, distinguishes two types of bone depositors-surface and volume. For those radionuclides considered to be uniformly distributed over bone surfaces, the deposition is assumed to be equally distributed in cortical and trabecular bone. For bone volume depositors, 80% of the deposition is assumed to be in cortical bone, 20% in trabecular bone. In the bone dosimetry model, cortical and trabecular bone are considered to be the source organs. The target sites are the bone surfaces (BS) and red marrow (RM), which are the radiosensitive tissues of concern in bone.

Bone dosimetry utilizes estimates of U_S in the source organs, cortical and trabecular bone, and computed values of the fraction of the emitted energy from these source organs which is absorbed in the target organs, bone surfaces and red marrow. Fractions are estimated for α, β, and photon radiations emitted by the source. This dosimetric information is contained in the quantity $SEE(T \leftarrow S)$ of equation 9.8, and is given in the dosimetric information.[13-15] The specific effective energy SEE is discussed in Section 9.F.

7. Submersion Exposure

The problem of setting DAC values for inert gases involves primarily external exposure rather than internal exposure. These gases are poorly absorbed by the body, so that only a small portion of the gas in a cloud of large volume is held in the body. The ICRP has considered three sources of exposure: the external dose due to submersion, the internal exposure due to absorbed gas, and the lung dose due to contained gas. They conclude[1] that the external exposure received in the cloud will be of most importance, and one may ignore the other contributions.

One is then concerned with the exposure of an individual essentially surrounded by a semispherical cloud of radioactive gas. For radioactive noble gases, other than radon and thoron which are treated in Reference 12, the exposure is limited by the DAC (submersion) which is based upon 2000 exposure hours in a working year. Details of the methods used to derive the DAC values are discussed in Chapter 8 of Reference 1.

E. Absorbed Dose Computations

1. MIRD Method

The method of calculation used to determine absorbed doses to tissues in the ICRP methodology has been adapted from the MIRD (Medical Internal Radiation Dosimetry Committee) method.[22-24] This model was developed for application to medical dosimetry, in which it is often desirable to know the dose to one organ from radioactivity source in another nearby organ.[3] Because of the quantities of radionuclides used in some clinical studies, one needs to be concerned with irradiation of nearby organs from the radionuclide being used in the study of a particular organ. A review of this methodology can be found in ICRU Report 32,[2] along with a discussion of clinical applications.

In the MIRD methodology, one distinguishes between target regions and source regions. Then, the concept of the absorbed fraction may be introduced. The absorbed fraction $\phi(T \leftarrow S)$ is the fraction of the energy emitted in a source region S that is absorbed in a target region T. Target and source regions may be completely separated; they may overlap, or they may coincide. One may define the specific absorbed fraction $\Phi(T \leftarrow S)$ as the absorbed fraction divided by the mass of the target region.

The absorbed dose in a target region depends upon a number of physical and biological parameters: activity, radioactive half life, decay scheme data, the location, mass and shape of the target region, the absorbed fraction and the temporal and spatial distribution of activity in the source region.[2] The specific absorbed fraction depends upon the activity distribution in S. The activity is assumed to be uniformly distributed in the source organ. The specific absorbed fractions for Reference Man were computed by the Monte Carlo method for a representative phantom. Values can be found in Appendix I of Reference 9.

Assume a radionuclide that decays with the emission of one type of radiation only. The activity of this radionuclide is uniformly distributed in some source region, and the emissions from the source irradiate the target region. The mean absorbed dose, $\bar{D}$, in the target region, is given by the product of five terms:[2]

- (1) the number of transformations (disintegrations) in the source during the time of interest, $\tilde{A}$;
- (2) the mean number of ionizing particles per nuclear decay, n;
- (3) the mean energy per particle, E;
- (4) the absorbed fraction, $\phi(T \leftarrow S)$; and
- (5) the reciprocal of the target mass, m^{-1}.

This may be written

$$\bar{D} = \tilde{A}\ n\ E\ \phi(T \leftarrow S) m^{-1} \qquad 9.17a$$

when all quantities are expressed in SI units. The product nE is usually written as Δ, the mean energy emitted per disintegration and ϕ divided by m is the specific absorbed fraction, so

$$\bar{D} = \tilde{A}\ \Delta\Phi\ (T \leftarrow S) \qquad 9.17b$$

Now, if we extend the concept to a source with a number of different emissions, each of which has an associated specific absorbed fraction, the expression becomes

$$\bar{D} = \tilde{A}\sum_i \Delta_i \Phi_i (T \leftarrow S). \qquad 9.17c$$

In the equation, $\tilde{A}$, called the cumulated activity, is a function of time and is obtained by the integration of the retention function over the time period of interest. The other quantities under the summation are generally grouped together and tabulated as the "S" value.[25] These are given in units of rad/μCih in Reference 26 and can be converted to nGy/dis by dividing S by 13.32.

If one substitutes the activity value A into the above equation, the dose rate, $\dot{D}$, is obtained

$$\dot{D} = A \sum_i \Delta_i \Phi_i(T \leftarrow S). \qquad 9.18$$

EXAMPLE A sample of 4×10^4 Bq (dis/s) of ^{131}I is uniformly distributed in the thyroid. Compute the initial absorbed dose rate, $\dot{D}$, to the thyroid. From Reference 25, p. 185, S for $^{131}I = 2.2 \times 10^{-2}$ rad/μCih with thyroid as target and source. So,

$$\frac{nGy}{dis} = \frac{S}{13.32} = \frac{2.2 \times 10^{-2}}{13.32} = 1.65 \times 10^{-3}$$

and $\dot{D} = 4 \times 10^4 \frac{dis}{s} (1.65 \times 10^{-3} \frac{nGy}{dis}) = 66 \frac{nGy}{s} = 6.6 \times 10^{-8} \frac{Gy}{s}$ (23.8 mrad/h).

If A is expressed in Bq, Δ_i in MeV/dis and $\Phi_i(T \leftarrow S)$ in kg^{-1}, the absorbed dose rate in Gy/h will be

$$\dot{D} = 5.76 \times 10^{-10} A \sum_i \Delta_i \Phi_i(T \leftarrow S) \quad Gy/h, \qquad 9.18a$$

in which the constant includes the factors to convert MeV to J and s to h. Again, the S value from Reference 25, when multiplied by 469, can be substituted for $\sum_i \Delta_i \Phi_i(T \leftarrow S)$ in equation 9.18a. That is,

$$\dot{D}=5.76\times10^{-10}(4\times10^{4})\ 469(2.2\times10^{-2}) = \frac{2.377\times10^{-4}\ \mathrm{Gy/h}}{3.6\times10^{3}\ \mathrm{s/h}}$$

$$= 6.6\times10^{-8}\frac{\mathrm{Gy}}{\mathrm{s}}\ (23.8\ \mathrm{mrad/h}).$$

Since there may be a number of source organs involved because a radionuclide will distribute in several tissues, the total dose, or dose rate, is obtained by a summation of the individual contributions.

For the case of a uniformly distributed source in a large volume, one can assume that the energy emitted per unit mass will be equal to the energy absorbed per unit mass.[2] Then, the absorbed dose rate is

$$\dot{D} = C\sum_{i}\Delta_{i}, \qquad 9.19$$

in which C is the activity per unit mass (Bq/kg) and $\Sigma\Delta_i$ is the total mean energy emitted per disintegration. Since all energy is absorbed, the absorbed fraction will equal 1. Values of Δ_i for a number of radionuclides, many of which are of medical interest, can be found in Appendix A of Reference 2.

a. Penetrating and Nonpenetrating Radiations

In some cases, the fraction of energy lost outside of a given tissue volume by the emitted radiation will be small enough to ignore. This is generally the case for α and β radiation. For photons or neutrons though, a large fraction of the emitted energy may be lost outside of a given tissue volume. The term penetrating radiation is used for those radiations which may lose a significant fraction of their energy outside of an organ. The term nonpenetrating is used for α, most β and those photons of $E<15$ keV. In the case of nonpenetrating radiation, the absorbed fraction is taken as 1 if the source and target regions are the same. The value of $\phi(T\leftarrow S)$ is 0 if the source and target regions are separate.[2]

b. Cumulated Activity

The activity as a function of time in a tissue is expressed by the retention function for that radionuclide in the particular organ. For the general case, the activity as a function of time can be written

$$A(t) = e^{-\lambda t} \sum_i k_i \, e^{-\lambda_i t}, \qquad 9.20$$

in which $e^{-\lambda t}$ is the term expressing radioactive decay and is common to all the compartments, k_i is the initial value of the activity for each compartment i, and $e^{-\lambda_i t}$ expresses the biological elimination from the organ. Note that k_i may also be expressed as a fraction of unit activity which is deposited in the ith compartment. The cumulated activity, $\tilde{A}$, is obtained by integration of the above expression, and gives

$$\tilde{A}(t) = \sum_i \frac{k_i}{\lambda+\lambda_i}\left(1 - e^{-(\lambda+\lambda_i)t}\right), \qquad 9.21a$$

which for $(\lambda+\lambda_i)t$ large enough, becomes

$$\tilde{A}(t) = \sum_i \frac{k_i}{\lambda + \lambda_i} \qquad 9.21b$$

For the case in which the retention function is expressed by only a single exponential,

$$A(t) = A_o e^{-(\lambda+\lambda_1)t} = A_o e^{-\lambda_{eff_1} t} = A_o e^{-\frac{\ln 2}{T_{eff_1}} t} \qquad 9.22$$

and

$$\tilde{A}(t) = \frac{A_o}{\lambda+\lambda_1} = \frac{A_o}{\lambda_{eff_1}} = \frac{A_o T_{eff_1}}{\ln 2} = 1.443\, A_o T_{eff_1} \qquad 9.23$$

In the above expression, T_{eff}, is the effective half life in the particular compartment 1 and is found from

$$T_{eff_1} = \frac{\ln 2}{\lambda + \lambda_1} = \frac{\ln 2}{\lambda_{eff_1}} = \frac{T\ T_1}{T + T_1} \qquad 9.24$$

<u>EXAMPLE</u> The activity deposited in a certain organ, with a single exponential retention function of biological half life 6d, is 10 MBq (10^7 dis/s) of half life 3d. Find the cumulated activity to complete decay.

$$A(t) = e^{-\lambda t} \Sigma\, k_i\, e^{-\lambda_i t} = e^{-\frac{\ln 2}{3}t}(10^7 e^{-\frac{\ln 2}{6}t})$$

$$= 10^7\ e^{-(\frac{\ln 2}{3} + \frac{\ln 2}{6})t}$$

is the retention equation and

$$\tilde{A}(t\sim\infty) = \frac{10^7}{\frac{\ln 2}{3} + \frac{\ln 2}{6}} = \frac{10^7}{0.5\ \ln 2} = 2.885\text{x}10^7 \text{ dis.}$$

Note from equation 9.24, $T_{eff} = \frac{3.6}{3+6} = 2\text{d}$ and $\tilde{A}(t\sim\infty) = 1.443\ (10^7)2$

gives the same answer.

Examples of absorbed dose calculations using the MIRD method can be found in Reference 2, Appendix C; Reference 24, Chapter 10 and Reference 26, Chapter 6.

The modification of the MIRD system to incorporate the quality factor and to obtain the specific absorbed fraction by transport methods using Monte Carlo calculations is attributed to Snyder and his collaborators. In addition to the tabulation of "S" in MIRD Pamphlet 11, Snyder, et al.[27] also published a tabulation of dose equivalent per μCi-day for various radionuclides. This formed the basis of the internal dosimetry model which is used in the new ICRP methodology

F. Specific Effective Energy (SEE)

The ICRP has utilized the Snyder modifications to the model for absorbed dose calculations based upon the MIRD method in its new internal dosimetry methodology. In the supplements[13-15] to ICRP 30 Parts 1, 2 and 3, data is given for the Specific Effective Energy, defined by

$$\mathrm{SEE}(T \leftarrow S)_j = \sum_i \frac{Y_i E_i \mathrm{AF}\ (T \leftarrow S)_i Q_i}{M_T},\ \frac{\mathrm{MeV}}{\mathrm{g\ trans}} \qquad 9.25$$

in which $\mathrm{SEE}(T \leftarrow S)_j$ is the specific effective energy absorbed in target T from radionuclide j in source organ S. The summation on the right side of the equation is taken over all radiation types i emitted in each disintegration of radionuclide j. The individual terms are:[1]

Y_i	is the yield of radiations of type i per disintegrations,
E_i(MeV)	is the average or unique energy of radiation type i as appropriate;
$\frac{\mathrm{AF}(T \leftarrow S)i}{M_T}$	is the specific absorbed fraction as defined in the MIRD method, and can be found in ICRP 23[9] for photons, but is expressed in units of $g^{-1}(10^{-3}\ \mathrm{kg})^{-1}$;
Q_i	is the quality factor and has the values discussed in 9.D.2

Values for Y_i and E_i can be found in ICRP 38.[16]

Since there may be a number of target and source organs for a particular radionuclide, the SEE values in the Supplements are arranged in tables with the columns representing source organs and the rows representing target organs. For example, Table 9.1, is an excerpt from Reference 13, showing the general arrangement of the SEE information. As might be expected, the SEE value is largest when the target and source organs are the same. Note that the values shown refer only to the designated radionuclide, and do not include any daughter contribution. The values for any daughter radionuclides are given in a separate table.

Table 9.1 - SEE (MeV per gram per transformation) of Sr-89

TARGETS	SOURCES					
	LUNGS	ULI CONTENT	LLI CONTENT	CORT. BONE	TRAB. BONE	TOTAL BODY
Gonads	4.8E-11	2.3E-09	4.6E-09	2.8E-10	2.8E-10	8.3E-06
R.Marrow	3.4E-10	5.6E-10	8.3E-10	8.6E-10	1.4E-04	8.3E-06
Lungs	5.8E-04	1.1E-10	4.3E-11	2.8E-10	2.8E-10	8.3E-06
Bone Surf.	2.7E-10	1.9E-10	2.8E-10	7.3E-05	1.2E-04	8.3E-06
ULI Wall	1.2E-10	1.3E-03	1.1E-09	2.2E-10	2.2E-10	8.3E-06
LLI Wall	2.6E-11	9.1E-10	2.2E-03	3.0E-10	3.0E-10	8.3E-06

To illustrate the computation of $SEE(T \leftarrow S)_j$, let us assume that the source of ^{89}Sr is located in the lungs and the target is the lungs. From page 195 of ICRP 38, the dosimetric data for ^{89}Sr gives a nonpenetrating component β^- of $Y_iE_i = 5.83x10^{-1}$ and a photon component (E_i =0.9091 MeV) of $Y_iE_i = 8.45x10^{-5}$. The daughter of ^{89}Sr is stable. Utilizing ICRP 23, Appendix I, we estimate the specific absorbed fraction for a 0.9 MeV photon for lung as both the source and the target (page 453 of ICRP 23) as about $4.6x10^{-5}$. The specific absorbed fraction for the beta is $1/M_T = 10^{-3}$ and Q_i is 1 for both β and photons. Utilizing equation 9.25,

$$SEE(Lung \leftarrow Lung) = \sum_{i=1}^{2} \frac{Y_i E_i AF_{(Lung \leftarrow Lung)i} Q_i}{M_T}$$

$$= 5.83x10^{-1} (10^{-3})(1) + 8.45x10^{-5}(4.65x10^{-5})(1)$$

$$= 5.83x10^{-4} = 5.8x10^{-4} \frac{MeV}{g\ trans}.$$

This result gives us one entry in Table 9.1, for lungs as the source and lungs as the target. For the other source organs, specific absorbed fractions for the β component are 0, so the computation only involves the photon component. Using ICRP 23, the specific absorbed fractions for the other source organs can be found and SEE computed. This will fill out the row with lungs as the target. Then, one can choose a different target organ and begin all over again to fill out that row! As the decay scheme becomes more complicated, one has to deal with more radiation types than in this simple example for ^{89}Sr. Again, although hand computations become tedious, a computer can handle this task easily. Moreover, the ICRP Supplements[13-15] already contain this information for most radionuclides of interest to health physicists.

G. Committed Dose Equivalent $H_{50,T}$

The committed dose equivalent, as expressed by equation 9.8, is the product of two factors: U_s, the total number of transformations of the radionuclide in a source organ S over a period of 50 years following intake, and SEE(T←S), the energy absorbed in the target, modified by the quality factor, for each type of radiation emitted per transformation in S. For a radiation type i, the committed dose equivalent is

$$H_{50}(T \leftarrow S)_i = U_s(\text{trans.}) \text{x} 1.6\text{x}10^{-13}\left(\frac{J}{MeV}\right) SEE(T \leftarrow S)_i \left(\frac{MeV}{g\ trans}\right) \left(10^3\left(\frac{g}{kg}\right)\right)$$

$$= 1.6\text{x}10^{-10}\ U_S\ SEE(T \leftarrow S)_i\ Sv \qquad 9.26$$

Multiply by 100 in equation 9.26 to obtain $H_{50}(T \leftarrow S)_i$ in rem. Equation 9.26 is then applied to all types of radiations emitted by the radionuclide. This requires the summation of $SEE(T \leftarrow S)_i$ over all i radiations emitted by the radionuclide. If a mixture of radionuclides is taken in, then a summation over the j radionuclides is required. In addition, there may be several source organs involved so that the contributions will need to be summed over all S, which leads to equation 9.8,

$$H_{50,T} = 1.6\text{x}10^{-10} \sum_S \sum_j \left[U_s \sum_i \text{SEE}(T \leftarrow S)_i \right]_j \text{ Sv}$$

From our example for ^{89}Sr above, i=2 radiations (β and γ) and j=1 radionuclide (^{89}Sr), but S=6 (lungs, ULI content, LLI content, cort. bone, trab. bone and total body) from Table 9.1. In order to find $H_{50,T}$, equation 9.8 reduces to

$$H_{50,T} = 1.6\text{x}10^{-10} \sum_{S=1}^{6} U_S \sum_{i=1}^{2} \text{SEE}(T \leftarrow S)_i .$$

The summation over i has already been done and these are the tabulated values in Table 9.1. What remains is to determine the U_S values which are found on page 82 of Reference 13. Assuming a Class Y aerosol was involved in a unit intake by inhalation, our expression becomes

$$\begin{aligned} H_{50,\text{Lungs}} &= 1.6\text{x}10^{-10}[8.9\text{x}10^{5}(5.8\text{x}10^{-4})+2.3\text{x}10^{4}(1.1\text{x}10^{-10}) \\ &\quad +4\text{x}10^{4}(4.3\text{x}10^{-11})+5.7\text{x}10^{3}(2.8\text{x}10^{-10}) \\ &\quad +4.6\text{x}10^{3}(2.8\text{x}10^{-10})+5.5\text{x}10^{3}(8.3\text{x}10^{-6})] \\ &= 8.3\text{x}10^{-8} \text{ Sv} \end{aligned}$$

So, for each Bq (dis/s) of intake, the $H_{50,\text{Lung}} = 8.3\text{x}10^{-8}$ Sv (8.3x 10^{-6} rem). The weighted committed dose equivalent is then found by multiplying $H_{50,\text{Lungs}}$ by w_T for lungs, which is .12. This gives 1x10^{-8} Sv. It turns out that for this particular situation (Class Y intake by inhalation), only the lung dose is important. Solving for the ALI, using equations 9.9 and 9.10, gives

$$I \leq \frac{0.05}{1\text{x}10^{-8}} = 5\text{x}10^{6} \text{ Bq.}$$

and $$I \leq \frac{0.5}{8.3\text{x}10^{-8}} = 6\text{x}10^{6} \text{ Bq.}$$

In this case, the choice for the ALI would be $5\text{x}10^6$ Bq since this value will satisfy both inequalities. The DAC is then given by the

$$\frac{\text{ALI}}{2400} = \frac{5\text{x}10^6}{2400} = 2.0\text{x}10^3 \text{ Bq/m}^3\text{, rounded to two places.}$$

More examples of the application of the ICRP 30 methodology can be found in Reference 28. References 29 and 30 discuss the use of the methodology for problems of assessing compliance with standards and arriving at cleanup criteria.

H. Internal Intake Assessments

As mentioned earlier in this section, the determination of the amount of a radionuclide which deposits in a specific organ is not an easy task. A multitude of factors affect the deposition of a given radionuclide in a specific organ. Many biological transfer fractions are still poorly known. In terms of the measurement of the amount of a radionuclide which deposits in an organ, two general measurement approaches are used.[31] One utilizes in vitro bioassay samples. In the in vitro bioassay method, the amount of the radionuclide in an organ is estimated by identifying, and measuring, if possible, the quantity of a specific radionuclide in samples that are excreted, secreted or removed from the body and which may include urine, blood, breath, sputum, sweat, saliva, hair, nasal discharges and feces.[32] For some radionuclides, this indirect method allows the identification of the amount of a radionuclide in a specific organ, but to compute the committed dose equivalent requires reliance on a metabolic model.[31]. That is, the determination of the amount of a radionuclide in an organ at any particular time allows one to compute only the initial dose rate in that organ, relative to the time of measurement. The dose commitment and/or the annual dose equivalent may only be calculated if one knows the necessary metabolic data (retention function). The generalized metabolic data which is available in ICRP 30 and the dosimetric data in the Supplements, allows one to compute the dose rate in an organ, and the dose commitment, but then only to the model, which is represented by Reference Man. For any individual, the actual dose commitment will differ from this value depending upon the individual's characteristics. Individual data obtained from excretion analysis should be used to obtain a more realistic estimate of the organ burden and the retention function.[31]

By integration of the retention function for a given organ over a suitable time duration (i.e., one year), the total number of transformations per unit activity during the year can be determined. Multiplying this result by the initial activity in the organ and utilizing the ICRP tables for SEE values, allows one to estimate the annual dose equivalent in that organ by use of an expression similar to equation 9.26.

One further difficulty in bioassay analysis, is that the sample which is collected may represent contributions from a number of organs in which the radionuclide has deposited. One may sometimes relate fractions of the total activity in the in vitro biological sample to a few organs which are excreting the radionuclide. However, this becomes increasingly difficult when the radionuclide is prone to deposit in many organs of the body.

Although bioassay has disadvantages as a quantitative method,[31,35] it is of value as a qualitative indicator of the effectiveness of the control methods. However, to be effective, the bioassay program needs the cooperation of the sampled individual. That is, if one is unwilling to submit the samples, the method will be ineffectual. For samples which are collected, care must be taken to avoid contamination of the sample. More discussion on the aspects and features of a suitable bioassay program can be found in References 33-35, and 37.

Another of the bioassay methods referred to above is in vivo measurements by external whole body counting (WBC). In this direct method, the gamma rays emanating from a radionuclide in a given organ can be counted and their energies analyzed by pulse height analysis. The identification of the nuclide can be accomplished rather easily, but the actual location of the radionuclide material may not be so definite. Since radionuclides may deposit in a number of different organs, the response of the WBC may be a composite of the contributions from a number of organs. This makes the assignment of the amount in a given organ somewhat arbitrary. In addition, the quantification of the amount of a radionuclide in a given organ, even if only one is involved, is not an easy task. Because of the differences in chest wall thickness, and organ shape and weight, the absorption and scattering properties may vary from individual to individual.

Even on the presumption that the radionuclide is located in a single organ, so that identification and quantification are simplified, the computation of the dose commitment will still require the use of the retention functions. Moreover, calibrations must be performed, using phantoms, in order to adequately assess the quantity of a radionuclide which is deposited in a given organ. This is because of the absorption and scattering of the photon radiation as it passes through the body.

A disadvantage of the WBC methodology is that the radionuclide to be identified must emit sufficiently high energy photons so that radiations will reach the detector. In reactor applications, many of the radionuclides of interest emit photons which are energetic enough to meet this criterion. However, radionuclides such as ^{239}Pu, emit only low energy photons (approximately 17 keV) in quantities large enough to be detected. These low energy photons are almost completely absorbed by the chest wall so that the detection of ^{239}Pu in lungs by this method is poor unless the concentration is very high.

The advantage of WBC over in vitro bioassay is the ability to obtain results much more quickly. Bioassay samples require a certain time for collection, followed by an even longer time for analysis, since sample preparation may require a significant amount of time. So, in a suspected incident, one generally cannot get an accurate estimate of the intake within a short time following the incident, if excreta samples are used.[35] On the other hand, whole body counting, following a suspected incident, should be performed as soon as possible after the incident. This will ensure that the majority of material taken in by inhalation will still be residing in the lungs, rather than distributed in many organs. Moreover, for radionuclides which distribute in only a few discrete organs, this method will give a better estimate of the quantity of a radionuclide in a particular organ, even if a count is taken much later.

Care needs to be taken so that personnel who are involved in a suspected incident, and are going to be counted, do not have external contamination, such as on clothing or skin. Not only does this give erroneous results with respect to the whole body count, but it also may result in contaminating the WBC, which is very sensitive to any increased background activity.

Individual organ counts can also be accomplished by this methodology. One of the more notable applications in this respect is thyroid counting for radioiodine. Solid state detectors, notably Cd-Te, have been successfully used for monitoring of cuts and wounds.

One other method used to estimate intake of radionuclides is to use concentration measurements as determined by air samples. The advantage of this method is its simplicity.[36] No special samples or analyses are required. The air sample is typically obtained on a filter paper and counted on relatively simple equipment. If the identity of the radionuclide is not known, the analysis then needs to be performed on more sophisticated equipment to identify the radionuclide in addition to the activity determination. Once the concentration in air has been determined, one can estimate the intake of radionuclides by use of the presumed time that the individual is exposed to the radionuclide concentration. This method of estimating intakes is not recommended by DOE except for unusual circumstances in which bioassay data is unavailable or inadequate.

Although simpler than either of the first two methods, this method has several disadvantages relative to the others. As is discussed in Section 14.D, the placement of the sampler is important. The concentration that the worker breathes can be significantly different from the concentration measured by the sampler. In addition, the worker may be moving around in the relevant area so that he may be exposed to a gradient of concentrations. Moreover, the concentration in the given region may not be constant, at all, during the time of exposure. These factors lead to a significant uncertainty in the estimation of the intake. So, any intake estimates need to be supported by bioassay determinations.

Once the material enters the body, there is more uncertainty with respect to the distribution of the material in the body. That is, material entering by inhalation goes to the lungs, is distributed to the blood and then deposited in the various organs. The fractions involved are also poorly known as are the metabolic parameters.

If the material is not analyzed for particle size, solubility, and chemical properties, more uncertainties are introduced. There is no one set of biological factors which can be universally applied to convert air concentration to the amount deposited in a given organ.[7]

As can be seen from the above discussion, the determination of the organ burden is somewhat difficult. When applicable, whole body counting is the desirable method since it will often yield a better estimate of the quantity of the radionuclide in a particular organ. However, even given the amount in the organ, the initial dose equivalent estimate must generally be determined using the metabolic data from ICRP 30 or some other specific metabolic model. The final estimate can be obtained by using the actual excretion data from the individual bioassay samples.

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QUESTIONS

9.1 Name four factors of main concern that enter into internal radiation dose calculations.

9.2 How do the distributions, uniform and nonuniform, affect the accuracy of the calculation of internal dose. Give an example.

9.3 Explain the difference between the absorbed dose and biologically equivalent dose.

9.4 Define Annual Limit of Intake (ALI).

9.5 What is the Derived Air Concentration (DAC)? How is it useful?

9.6 Name some of the ways in which a radionuclide may reach an organ. What are two important ways of concern?

9.7 What characteristic of a radionuclide determines its absorption in the body?

9.8 What factors determine the movement and retention of matter in the respiratory tract?

9.9 What are the several processes of concern in estimating the net retention of a radioactive material in a given organ of the body?

9.10 Why is it necessary to invoke the concept of "Reference Man"?

9.11 What are some of the major assumptions of ICRP-2 model?

9.12 Explain the concept of "critical organ."

9.13 What is the "MPC"?

9.14 In what circumstances will the activity in the organ be approximately proportional to the MPC?

9.15 In the ICRP-30 model, how is the concept of risk related to dose?

9.16 Explain how dose limitation from inhalation and ingestion by several pathways is to be accounted for in the ICRP-30 model.

9.17 What is deep dose equivalent index? How is it used in the ICRP-30 model dose limitation inequality?

9.18 In the respiratory model, how many classes of materials are listed and what is the basis for this classification?

9.19 What is the basis for the ALI in the case of ingestion?

9.20 What are two types of bone depositors in the ICRP-30 bone dosimetry model?

9.21 What are three sources of exposure considered for inert gases? Which type of exposure is considered most important?

9.22 From which source are ICRP-30 absorbed dose computation methods adapted?

9.23 What is the specific absorbed faction?

9.24 Explain the term "mean absorbed dose" in the target region.

9.25 How are penetrating and non-penetrating radiations classified with respect to the tissue volume?

9.26 What is "effective half-life"? How is it related to the cumulated activity?

9.27 What are the general measurement approaches in assessing the internal intake?

9.28 Mention some of the disadvantages of bioassay method as a tool for internal intake assessment?

9.29 What are some of the advantages and disadvantages of whole body counting?

9.30 What uncertainties are involved in the measurement of concentrations by the air sampling method as a tool to assess internal intake?

PROBLEMS

9.1 An individual has an annual intake of 1000 Bq of Po-210, whose ALI for inhalation is $2\text{x}10^4$ Bq. In addition, the individual drinks water, 700 l/y, containing an average concentration of the same nuclide of 50 Bq/l and consumes food stuffs, 500 kg/y, with a concentration of 100 Bq/kg. The ALI for ingestion is $1\text{x}10^5$ Bq, will the ICRP dose limits be satisfied?

Answer: $0.9 < 1$, YES

9.2 A radiochemist suffers an accidental exposure to ^{131}I. It was found that 740 kBq were deposited in his/her body, of which 220 kBq are in the thyroid gland and the rest distributed in the remainder of the body. Using bioassay and body scanning data, the thyroid dose was estimated to be 370 mGy and the whole body dose, 0.5 mGy. Calculate the chemists' effective dose equivalent. Was this exposure within the ICRP criteria? (Hint: see Table 7.1 for relevant data.)

Answer: 11.6 mSv. Yes

9.3 Calculate the cumulated activity $\tilde{A}$ in organ "L" of the body for a radionuclide, if the initial administered activity is $3.7x10^7$ Bq. Assume 85% of the activity is retained in the organ and the effective half life is equal to the physical half life (6 h).

Answer: $9.8x10^{11}$ Bqs

9.4 In problem 9.3, if 50% of the activity in the organ "L" is eliminated with a half life of 2 h and 50% with a half life of 3 h, what is the cumulated activity in the organ for the same initial activity?

Answer: $2.86x10^{11}$ Bqs

9.5 ^{24}Na decays by β emission to ^{25}Mg, followed by two γ rays of energies 2.75 MeV (100%) and 1.37 MeV (100%). The mean β energy is 0.549 MeV (100%). The physical half life of ^{24}Na is 15.03 h. When ingested, ^{24}Na is found to be uniformly distributed throughout the body, with a biological half life of 11 days, long compared to the physical half life.

Prepare a table of relevant parameters and calculate the mean absorbed dose for 1 MBq of ^{24}Na activity. (The absorbed fractions are 0.314 and 0.274 for the 1.37 MeV γ and 2.75 MeV γ; respectively, and 1 for the β. Assume 70 kg for the mass of the body.)

Answer: A : $7.39x10^{10}$ Bqs, D = $2.93x10^{-4}$ Gy

9.6 Cesium-137 is rapidly and almost completely absorbed from the GI tract. It is also distributed uniformly in the body. The retention of cesium is described by a two-compartment equation.

$$R(t) = 0.1\ e^{-0.693\ t/T_1} + 0.9\ e^{-0.693\ t/T_2}$$

Where T_1 = 2 days and T_2 = 110 days. (Reference ICRP 30, Part 1, Page 91). Calculate the effective clearance rates of the two compartments. $T_{1/2}$ of ^{137}Cs is 30 y.

Answer: 0.347 d^{-1} and 0.00636 d^{-1}

9.7 Calculate the specific effective energy (SEE) for whole body internal exposure to ^{137}Cs using the following data.

Radiation*	n/trans. n_i	Mean Energy Particle E_i, MeV	Absorbed Fraction ϕ_i
β_1	0.935	0.1749	1
β_2	0.065	0.4272	1
γ	0.840	0.6616	0.34
K (ice)	0.0781	0.6242	1
L (ice)	0.0140	0.6560	1
M (ice)	0.0031	0.6605	1
K_1x ray	0.0374	0.0322	0.76
K_2x ray	0.0194	0.0318	0.76
$K\beta_1$ x ray	0.0105	0.0364	0.72
$K\beta_2$ x ray	0.0022	0.0374	0.71

*Neglected K,L,M auger electrons and L-x rays.

Assume the mass of the body to be 70 kg.

Answer: $6.313\text{x}10^{-3}$ MeV/transkg

9.8 Use the SEE value from problem 9.7 and find the initial dose rate for 1 Bq of ^{137}Cs.

Answer: $3.64\text{x}10^{-12}$ Gy/h

9.9 a) Using the information from problems 9.6 and 9.7 and the ICRP stochastic dose limit criterion (0.05 Sv) estimate the allowable limit of intake (ALI) for ingestion. (Hint: Remember that there are two compartments in the tissue with two clearance rates.)

Answer: $4.0\text{x}10^{6}$ Bq.

b) What is the committed dose equivalent for intestion of 1 ALI?

Answer: $1.25\text{x}10^{-8}$ Sv/Bq

9.10 Make use of Figure 9.1 of the text and calculate the initial number of transformations per second, and clearance rates (λ_E), after an inhalation of 1 Bq of ^{137}Cs (cesium is assigned to clearance category D). Also remember that the N-P region does not directly contribute to the lung dose).

Answer:

Compartment	A(o), t/s	λ_E, s^{-1}
(c→BF)	0.076	$8.02\text{x}10^{-4}$
(e→BF)	0.2	$1.60\text{x}10^{-5}$
(h→lymph nodes)	0.05	$1.60\text{x}10^{-5}$
lymph nodes	0.0	$1.60\text{x}10^{-5}$
(d→GI tract)	0.004	$4.01\text{x}10^{-5}$

9.11 From the results of problem 9.10, calculate the total number of ^{137}Cs and ^{137m}Ba transformations in the lung for a time of 50 years (Hint: Since 50 years is long compared to the effective half times in each compartment, use equation 9.23 of the text).

Answer: $1.89\text{x}10^4$ transformations from ^{137}Cs, $1.79\text{x}10^4$ from ^{137m}Ba.

9.12 The specific effective energy SEE(L←L) for ^{137}Cs is $1.9\text{x}10^{-1}$ MeV/kg·trans and for the short-lived ^{137m}Ba it is $9.5\text{x}10^{-2}$ MeV/kg.trans. (ICRP 30, Supplement to Part 1, Page 235). Use the <u>SEE</u>(L←L) and the result from problem 9.11 to calculate the committed dose equivalent to the lung from (^{137}Cs-^{137m}Ba) deposited in the lung.

Answer: $8.47\text{x}10^{-10}$ Sv

9.13 If 30% of ^{137}Cs is deposited in the n-p region (see Figure 9.1), 8% is deposited in the T-B region and 25% in the P-region, what is absorbed into the body for an initial inhalation of 1 Bq? (Note: $f_1 = 1$).

Answer: ~ 0.63 Bq

9.14 Use the information and results in problem 9.6 and the result of problem 9.13, and calculate the total number of ^{137}Cs and ^{137m}Ba transformations in the body from what is deposited in the respiratory tract and transferred to the body.

Answer: $7.7\text{x}10^6$ transformations for ^{137}Cs, $7.3\text{x}10^6$ for ^{137m}Ba

9.15 a) The specific effective energy, SEE(L←Total Body) for (^{137}Cs is 2.7×10^{-3}, and for ^{137m}Ba, 4.0×10^{-3} MeV/kgtrans. (ICRP 30, Supplement to Part 1, Page 235). Use the SEE (L←Total Body), and the results of problem 9.14, to calculate the committed dose equivalent to the lung from ^{137}Cs in the body.

Answer: 8.0×10^{-9} Sv

b) What is the total committed dose equivalent to the lungs from the inhalation of 1 Bq of ^{137}Cs?

Answer: 8.8×10^{-9} Sv/Bq.

9.16 Using the result of problem 9.15b and a weighting factor of 0.12 for lung, calculate the weighted committed dose equivalent in lung from 1 Bq of ^{137}Cs.

Answer: 1.0×10^{-9} Sv/Bq.

SECTION 10 - RADIATION DETECTION PRINCIPLES

A. General

One of the main functions of a health physicist is to protect workers from the harmful effects of radiation. That is, he recommends a certain approach in order to complete a given task safely. This often requires that one have a measure of the radiation field in regard to some standard. Then, the results can be evaluated with respect to the relative hazard in terms of this yardstick. Based upon this analysis, the health physicist is able to suggest a course of action.

The evaluation of the radiation field with respect to the hazard present may be based upon the recommended radiation protection standards (for example ICRP 26[1]) or on regulatory requirements (such as DOE Order 5480.11[2] or 10 CFR 20[3]). These regulations serve as the standards that should not be exceeded without careful thought about the reasons for so doing. In general, administrative limits, well below the maximum value allowed by the standards, will be employed in the exercise of ALARA. To compare the radiation field to these standards or limits, we must have a means of measuring the field in terms of defined units.[4] The study of instrument theory and design deals with this aspect. We may also be concerned with the response of the instrument to radiation. This refers to how well the unit detects both the type and amount of radiation. The sensitivity of an instrument is a measure of how well it detects.

Radiation cannot be detected with the unaided senses. For this reason, we must use some substance that responds to the radiation in some manner (the detector) and a system to measure the extent of the response (the measuring apparatus). Many types of detectors have been used for radiation.[5-11] We may discuss these in regard to the nature of their response to the radiation. A large class of detection systems uses the ionization produced in them. This includes ion chambers, proportional counters, Geiger-Müller counters, cloud chambers, spark chambers, fission chambers, and semiconductor devices.[9-11] Other systems depend upon excitation and even molecular dissociation that occur along with the ionization. These processes are useful in scintillation counters, chemical dosimeters, and devices that depend upon certain optical properties of solids.[8] Other special processes, such as the Cerenkov effect[5,6] or neutron detection through induced activity, have also been used.

This section discusses some of the main features of devices that are of value for health physics applications.

B. Ionization Method

The passage of a photon, neutron, or charged particle through a substance can result in the removal of an electron from a neutral atom or molecule of the substance. If this occurs, an ion pair is formed. The pair is composed of the free electron and the residual atom or molecule, which will have a net positive charge. The process of forming ion pairs in a substance is called ionization. Ion pairs may be formed in a direct or indirect manner. This ionizing power of radiation, whether direct or indirect, is used in many devices to detect radiation.

Charged particles, such as α and β, produce ion pairs by direct action. They may collide with electrons along their path through a substance and remove them from the atom. They may also transfer energy by the interaction of the electric fields when passing close to an electron. If the energy transfer is not enough to remove an electron, the atom is left in a disturbed state. The process by which these states are formed is known as excitation. A large portion (approximately 1/2) of the energy lost in moving through a medium is lost in excitation.

Photons and neutrons produce ion pairs by indirect action. The processes by which they interact with matter produce the charged particles, which in turn form ion pairs.

The number of ion pairs formed in a substance is a function of the energy of the radiation and the nature of the substance. A certain amount of energy is required to remove an electron from a given atom. This is called the ionization potential of the atom. Values for most of the elements are in the range 5-20 eV. In most substances, the energy lost per ion pair formed is larger than the ionization potential. This reflects the fact that some of the energy has been lost in excitation. For this reason, the W value,[4] which is the average energy expended to produce an ion pair, is a more useful concept in dosimetry. A knowledge of the W value of the substance and the energy of the radiation allows one to estimate the number of ion pairs formed. W values are listed in the literature.[6,12,20]

1. Gas-Filled Chambers

One of the oldest devices used to detect radiation is the gas-filled chamber. Consider a two-electrode chamber (shown schematically in Figure 10.1) in the form of a cylinder. We impress a voltage V such that the central wire becomes the positive electrode (anode) and the cylinder the negative electrode (cathode). If we enclose a gas in the chamber, ions form in the chamber wall and in the gas itself. The electric field between the cathode and anode draws the ions toward these electrodes. The speed with which the ions move is a function of the field strength and the nature and pressure of the gas. The positive ions are drawn to the wall of

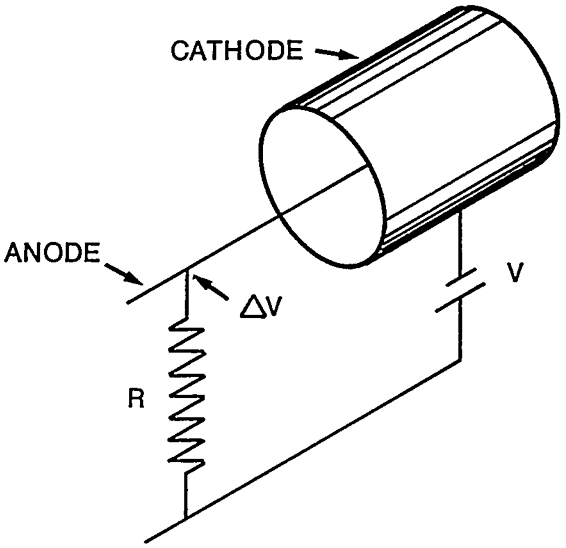

Figure 10.1 Two-electrode, gas filled chamber.

the chamber, and the faster moving electrons are swept toward the central wire. A charge collects on the anode, causing a voltage change in the circuit. This change or drop in voltage is referred to as a pulse. The size of the pulse depends upon the number of electrons collected. The presence of this pulse in the external circuit causes a current to flow. If we include an ammeter in this circuit, a meter reading results. In this way, we can detect the presence of radiation.

a. Pulse Size Considerations

The size of the pulse appearing at the anode is also a function of the applied voltage. This is shown in Figure 10.2, which gives the relative pulse size produced by both α (solid curve) and β (dashed curve) in a given chamber.

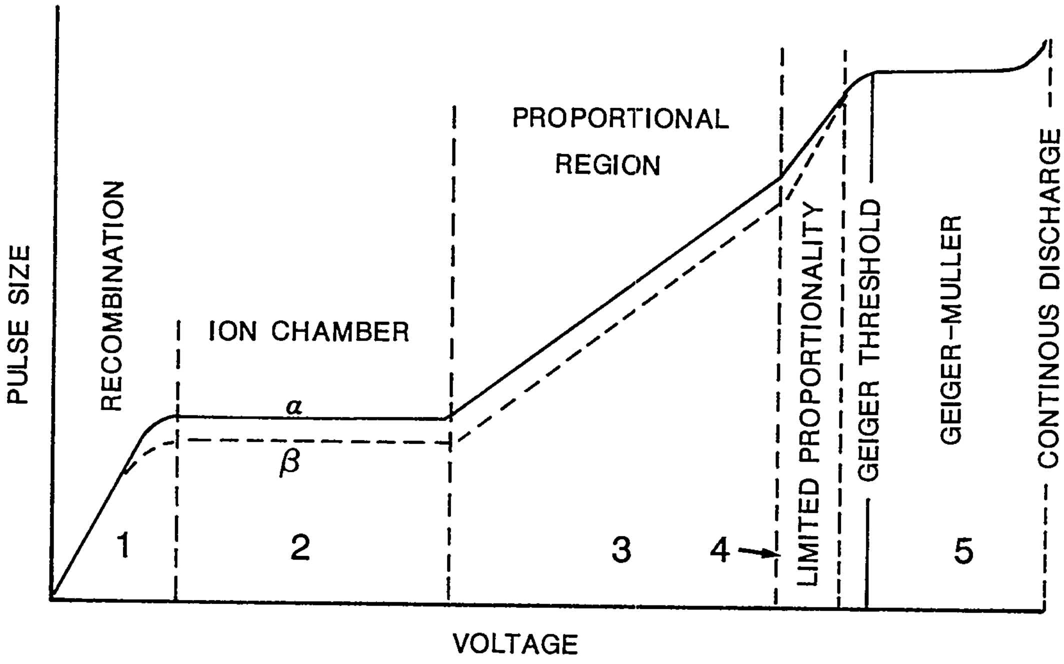

Figure 10.2 Pulse size as a function of voltage in a gas-filled, two-electrode chamber.

When the voltage on the chamber is quite low, the force on the ions that draws them to the electrodes is also low. For this condition, two processes compete for the ions. One of these is ion collection; the other is ion recombination. This means that after an ion has been formed, it may recombine with an unlike ion and become neutral again. The voltage pulse depends upon the outcome between these two processes. This is shown in Region 1, the region of recombination. For low voltage, the time required to collect ions is long enough to permit a large number to be neutralized. As the voltage increases, more ions are collected and the pulse size increases. The time in which to recombine is now shorter, and fewer ions escape collection.

b. Ion Chamber Region

As the voltage is further increased, we enter Region 2, the ion chamber region. The pulse size has now leveled off and no longer increases. The voltage is high enough so that the loss of ions through recombination becomes negligible. Now, almost all the ions formed are collected. Since we collect nearly all the ions, the pulse size can no longer increase with increase in applied voltage. The current flowing in the circuit reaches a maximum value also. This is called the saturation current, and its value depends upon the amount of radiation. To increase the value of the current, we would have to increase the amount of radiation that enters the chamber.

Devices designed to operate at saturation current are known as ion chambers. This type of design works well for survey devices. The current reading is proportional to the amount of radiation to which the chamber is exposed. Since the current is independent of the voltage, there is no need for a highly stable voltage supply. However, we must be careful to supply the proper voltage to ensure saturation. The voltage needed increases as the dose rate increases. For a given voltage, then, there will be a maximum dose rate above which the chamber current will not be saturated.

In the ion chamber region, the ions formed gain energy in the electric field as they drift toward the electrodes. In this voltage range, though, the ions do not gain enough energy to produce more ions as they move toward the electrodes. If enough energy were gained, then we would have the case in which the number of ions would increase. This increase could be defined in terms of a gas-amplification factor. That is, if for each original ion pair, a total A ion pairs result, then the factor would be equal to A. In other words, if 100 ion pairs are formed for each initial pair, the factor would be 100. For the ion-chamber region then, the amplification is 1.

c. Proportional Region

Let us return to Figure 10.2 and discuss Region 3 (the proportional region). As the voltage is further increased, the pulse size again begins to increase. This is the region in which gas amplification occurs. The primary ions now acquire enough energy to produce more ions as they move through the gas. The extent of this increase in a given case is a function of the voltage. That is, as we continue to increase the voltage, the amplification factor increases. This occurs because the voltage becomes high enough so that even the secondary ions gain enough energy to produce more ions. In this way, a cascade or avalanche of ions is formed. The total pulse size that results depends, in this case, upon the initial number of ions produced in the gas. For this reason, devices designed to operate in this region are called proportional counters.

In the proportional region, the size of the cascade may be controlled by (1) use of a cylindrical design for the chamber, and (2) use of a known and stable voltage. For a cylindrical chamber, the field strength at radius r is given by equation 1.12,

$$E = \frac{V}{r \ln (r_2/r_1)},$$

where V is the voltage, r_2 is the inner radius of the cylinder, and r_1

is the radius of the central wire. From this, the field strength is very high near the wire, but drops off quickly as we move away from the wire. At some distance near the wire, the field strength is great enough to initiate the cascade. This means that the cascade always starts at this same distance for a given value of V. Thus, for a given potential in a cylindrical chamber, the size of each avalanche is nearly constant. This differs from the parallel-plate chamber in which the field strength is the same everywhere and the cascade size depends upon where the ions start.

Up to the voltage at which amplification begins to occur, the pulse size V may be expressed as

$$V = \frac{Q}{C} = \frac{Ne}{C} \text{ (volts)}, \qquad 10.1$$

where N is the number of ions of one sign that are collected, e is the electronic charge (1.6×10^{-19} coulombs), and C is the capacity of the system, in farads. The product Ne equals Q, the charge collected by the anode in coulombs. In the proportional region, each of the N ions in turn produces A ions, where A is the gas-amplification factor. The pulse size ΔV_p in Region 3 is then given by

$$\Delta V_p = \frac{AQ}{C} = \frac{ANe}{C} \text{(volts)} \qquad 10.2$$

The value of the amplification factor A in this region depends upon the voltage, the radius of the cylinder and the anode wire, and the nature and pressure of the gas in the chamber. In the proportional region, this factor increases steeply with voltage. With the proper design, values of A in the range > 10^5 can be achieved.[5]

The final pulse size then depends upon the initial number of ions formed. This allows us to use a proportional counter to discriminate between types of radiation on the basis of their ionizing power. For example, suppose the chamber is exposed to both α and β. Let us assume that the gas is air, and that an α and a β each have about 3.5 MeV to expend in the chamber. If the α loses all its energy, it would produce about 10^5 ion pairs (W = 35 eV/ip). For a system capacity of 16 $\mu\mu$F and amplification factor $A = 10^3$, the voltage pulse would be

$$\Delta V_p = \frac{ANe}{C} = \frac{10^3(10^5)(1.6\text{x}10^{-19})}{16\text{x}10^{-12}} = 1.0 \text{ volt}$$

For the same total path length in the gas (approximately .02 m), the beta produces much less ionization. In this case, the value should be around 150 ion pairs. The pulse size would be

$$\Delta V_p = \frac{ANe}{C} = \frac{10^3(150)(1.6\text{x}10^{-19})}{16\text{x}10^{-12}} = 1.5\text{x}10^{-3} \text{ volts}$$

A circuit can be designed that will readily distinguish between these pulses so that we can detect either or both. With the proper circuit design, we can also count each pulse simultaneously.

d. Region of Limited Proportionality

The next region arrived at as the voltage is further increased is the region of limited proportionality (Region 4). In this region, the amplification factor A changes. No longer is this factor independent of the number of ions formed. The value of A becomes larger for the smaller initial pulses than for the larger pulses. We then begin to lose the ability to distinguish between types of radiation. Now, as shown in Figure 10.2, the final pulse size from α as well as β approaches the same value.[5]

e. Geiger-Müller Region

If we continue to increase the voltage, we reach a value at which all pulse sizes become equal. This voltage is known as the Geiger threshold voltage. At this point, the pulse size becomes independent of the number of primary ions formed. This means that even a single ionizing event produces a cascade effect. Also, the voltage is now so high that each ion in the cascade gains enough energy to produce a new cascade. The effect is to produce a discharge along the entire length of the central wire. Because the entire length of the wire is involved, the pulse size is no longer a function of the primary number of ions created. In effect, A and N no longer are variables in that their product will be constant for a given voltage.

If we increase the voltage above the threshold, the counter is operating in the Geiger-Müller region (Region 5) and is called a Geiger-Müller, or G-M counter. The pulses in this region are much larger than those in the proportional region. However, since all pulse sizes are the same, regardless of origin, the device can no longer distinguish between types of radiation. The charge collected in this region is limited by the design features of the chamber and the external circuit.

2. Semiconductor Devices

The term semiconductor describes a class of solids that can be used to detect radiation. In essence, these may be thought of as solid, rather than gas-filled, ion chambers. As such, they can detect charged particles whose range in solids is about 10 mm or less.[5] The collection by means of a strong electric field of the electrons released by ionization along the path of these particles is the basis of using this type of detector.

a. Semiconductors

In terms of the band theory of solids,[13,14] the closely packed atoms in a solid crystal modify the energy states of the individual atoms. This action splits the states into bands composed of a number of closely spaced levels. The number of levels formed is equal to the total number of atoms in the crystal. The structure of these bands in a given crystal substance determines the conducting features of that solid. Figure 10.3 shows a typical band structure for three types of solids (conductors, semiconductors, and insulators). In conductors (mainly metals), there are unfilled levels in the topmost band (often called the conduction band). Enough energy can easily be supplied to an electron to allow it to move into the conduction band.

For semiconductors and insulators, a gap exists between the highest filled band (called the valence band) and the next empty band. This gap, shown as ΔE in the figure, is called the band gap.[14] Now, moving an electron into the empty band is not so easy. The gap may represent a substantial energy difference. We may simply distinguish a semiconductor from an insulator on the basis of the size of the band gap. In an insulator, the gap is large (about 5 eV or more); in a semiconductor, the band gap may be in the range 0.7-2.1 eV.[5] Because of the small ΔE,

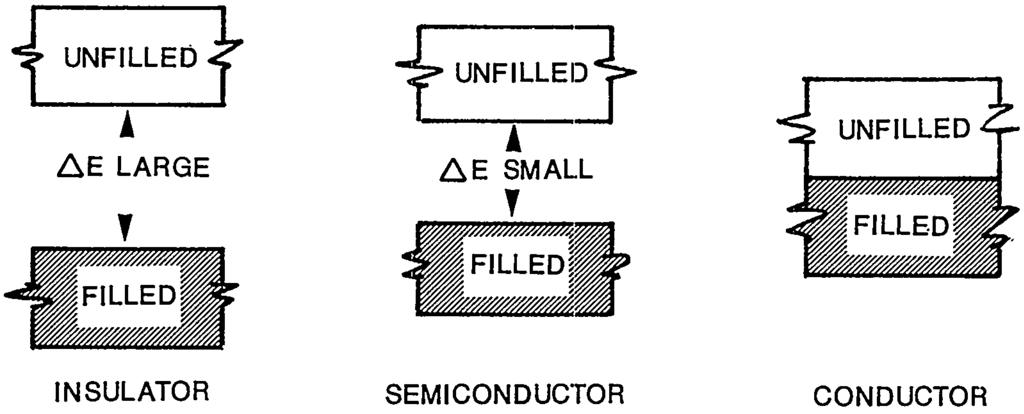

Figure 10.3 Band structure in conductors, semiconductors and insulators. (Adapted from Progress in Nuclear Physics, Vol 9, G.Dearnaley, Semiconductor Counters, Copyright 1964, Pergamon Press, Ltd., Used with permission.)

conduction often occurs in semiconductors at normal temperatures. That is, the thermal energy supplied at room temperature is sometimes enough to raise electrons across the band gap. This effect depends upon the nature of the substance used.

(1) Electron-Hole Pairs

When an electron is raised to the empty band, a vacancy has been created in a filled band. This absence of an electron in the valence band is called a "hole." Ionization of a solid results in electron-hole pairs being formed.

If a strong electric field is applied to the crystal, the electron in the conduction band moves under the action of the applied field. Also, an electron in the valence band moves into the vacancy left by the initial electron. In this sense, then, the hole also moves under the action of the field. That is, the valence band is normally filled. When a hole is created, an electron fills this hole at the expense of leaving a hole in the level it came from in the valence band. In effect, the freed electron moves in one direction, and the hole moves in the opposite direction. For this reason, both the electron and the hole are called carriers. Now, very much like an ion pair formed in a gas, the electrons move faster in the solid than do the holes. When many electrons are freed, the lag in movement allows the growth of a large positive space charge. The presence of this space charge impedes the collection of electrons by the applied field. Under very high bombardment from an ionizing source, the solid may begin to change characteristics. This occurs with substances such as diamond, the silver halides, and cadmium sulphide.[13]

(2) Effect of Impurities; n- and p-type Semiconductors

The presence of very small amounts of foreign substances in the solid can greatly affect the conducting aspects of a given substance. The main features of semiconductors are brought about by the presence of small amounts of impurities. All semiconductors contain these impurities to some degree by their very nature. In this sense, there is no such thing as a "pure" semiconductor as shown in Figure 10.4a. The band structure as pictured is always modified by the presence of the impurity atoms. The nature of these atoms is such as to bring about two main types of effect. The foreign atom may contain an excess outer electron above that needed for valence binding. This electron is very loosely bound and can be easily raised or "donated" to the conduction band. In effect, this decreases the band-gap width to some small value, ΔE_1 (as shown in Figure 10.4b). This type of impurity is referred to as a "donor" substance. We can add such an impurity atom to a semiconductor substance to achieve a desired effect. This process is referred to as "doping." A donor substance, such as phosphorous, added to silicon reduces the band gap to only 0.045 eV,[13] in effect. Since conduction in this instance occurs by the movement of negative charges, the substance is called an n-type semiconductor.

The other effect occurs when the impurity atom does not contain enough outer electrons for valence binding. This means that a vacancy now exists in what was originally a filled band. This creates a hole in the valence band (as shown in Fig. 10.4c). This positive hole can easily "accept" an electron from a nearby atom. This type of impurity is thus called an "acceptor" substance. An acceptor atom, such as boron, added to silicon results in the presence of holes in the valence band. Although electrons move to fill the holes, the conduction that results may be viewed as the movement of positive holes. These substances are referred to as p-type semiconductors.

Even if doping is not applied, a semiconductor most often contains both donor and acceptor impurities. Whether the semiconductor is n-type or p-type depends upon the impurity having the larger concentration. If the number of n-type impurities is equal to the number of p-type impurities, the material is said to be compensated and the substance would be like an intrinsic semiconductor.[5] In this case, the impurities "compensate" for each other. The charge carriers are then due to thermal excitation of electrons across the band gap. Thus, a truly

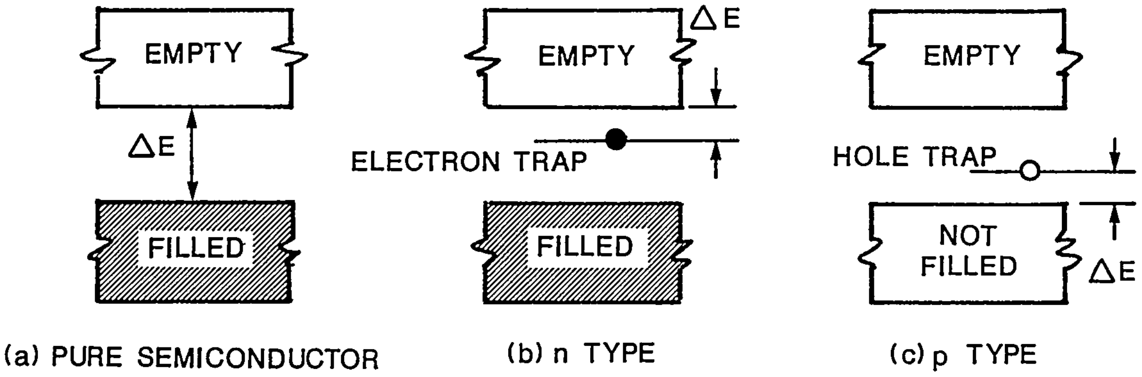

Figure 10.4 Effect of "doping" with a desired carrier type. (Adapted from Progress in Nuclear Physics, Vol. G.Dearnley, Semiconductor Counters, Copyright 1964, Pergamon Press, Ltd., used by permission.)

intrinsic semiconductor amounts to the concept of a pure semiconductor. As might be suspected, the achievement of exact compensation by doping is quite difficult.

(3) Impurities; Traps

The presence of impurity atoms and other type defects in a crystal can be looked upon as presenting centers that tend to trap the charge carriers. That is, each defect upsets the balance of charge in its nearby area. These centers then have a net charge with which to attract other charges. Now, suppose a carrier in a crystal substance is drifting in an applied field toward a collector. At one of these centers, the carrier may be trapped for a while and then released. As it begins to drift, it may then be trapped at some other site, and later released. In this sense, then, the electron or hole is not fully lost in the process. However, it may be delayed long enough in transit so that it does not add to the output. If this occurs, the net effect is the same as if the charge had been lost for good.

(4) Recombination

Another process that can also affect charge collection is recombination. This refers to the true loss of an electron-hole pair before they are collected in the applied field. Thus, the result in this case is similar to the recombination effect in gas-filled chambers. Present thought is that the same types of crystal defects figure both in trapping and recombination action. Our concern is that the recombination and trapping represents loss of charge carriers and affects the carrier lifetime in the crystal. These effects are significant in most semiconductors, so that high purity and reduced imperfections are required to combat the above effects. Based upon practical values for applied electric fields to the crystal, silicon and germanium are more suitable substances at present. These materials can be produced as large crystals without excessive carrier loss.

b. Conduction Counters

The first use of solid counters was the crystal or conduction counter. In these, a small crystal of some insulator (diamond, AgCl, NaCl, or CdS) acts as a solid ion chamber. The electric field is applied between plane electrodes to produce a uniform field. In this instance, the sensitive volume extends throughout the entire crystal.

A charged particle entering the crystal loses energy by creating electron-hole pairs. The energy transferred is generally large compared to that needed to raise an electron to the conduction band. Thus, more pairs may result from the excess energy of these highly excited states. The presence of the field causes the electrons and holes to drift toward the electrodes. This action leads to an induced charge in the external circuit. Thus, the charged particle could be detected.

These early types of counters were erratic in their performance, mainly because of the effects produced by the crystal defects. Recent use of silicon and gallium arsenide have led to improved counters, but at present these counters are not used extensively in health physics work. Moreover, the advent of the semiconductor junction counters has blunted interest in the crystal counter.

c. Semiconductor Junction Counters

The feature of the crystal counter that causes the most trouble is the presence of the impurities in the insulator crystal. Efforts to produce a neutral effect by doping the crystal with an impurity atom met with limited success. However, by means of the junction counter, one can obtain a region that is almost free of carriers. With proper design, this region may extend to greater than 10 mm in depth.

The basic feature of the semiconductor junction counter is the union of an n-type substance with a p-type substance, or vice versa. The new substance may then be referred to as an n-p or p-n junction. These junctions may be prepared in the two forms: the diffused junction and the surface-barrier junction. That is, a diffused p-n junction is obtained by diffusing an acceptor substance to a shallow depth in an n-type substance (Figure 10.5a). This gives rise to the depletion layer, a space-charge region in which a potential difference exists. It is formed by diffusion of electrons from the n-region into the p-region, and the diffusion of holes from the p-region into the n-region. This results in a shallow region of depleted carriers, which behaves as an insulator bounded by conducting electrodes. That is, a net charge on each side of this region impedes the further transfer of charge. This charge is positive in the n-region and negative in the p-region. This barrier can be broken down if we apply an external voltage to the system, with regard to the proper bias. A forward bias is applied when we connect the positive electrode to the p-region. In this case, the barrier breaks down and electrons stream across the junction. However, if we use a reverse bias (negative side connected to the p-region), the barrier height is increased and the depleted region in silicon may extend over as much as several mm. One disadvantage of the diffused junction is that in producing the detector, a somewhat undesirable dead layer of material is formed. As seen in the Figure 10.5a, radiation must travel a certain distance through the material (approximately 1-2 μm) in order to arrive at the depletion region. This means that the incoming energy of the charged particle may be altered significantly. However, these detectors are rugged, reliable and less susceptible to radiation damage.[5]

The surface-barrier type of junction is shown in 10.5b. The junction is often made by the spontaneous oxidation of a chemically etched surface of an n-type substance. The surface states which result, in effect, form a p-layer. Again, by use of the reverse-bias technique, this junction may be used to detect α and other short range radiation.

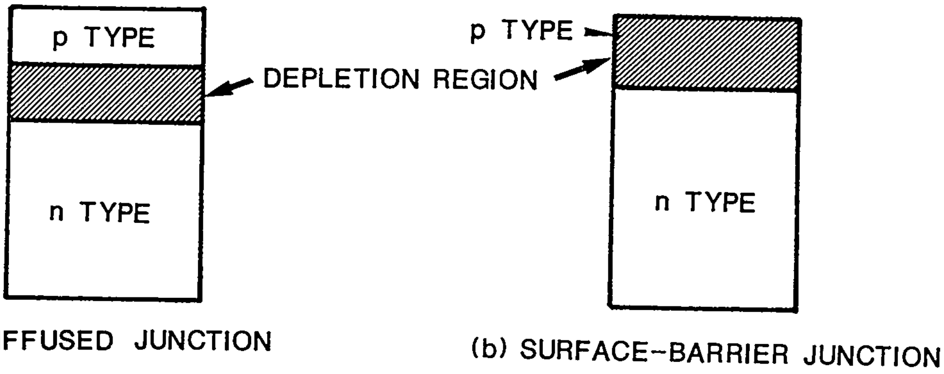

Figure 10.5 Diffused and surface-barrier p-n junctions. (from NUCLEAR RADIATION PROTECTION by W.J.Price, Copyright 1964, second edition, McGraw-Hill Book Co., used by permission)[10]

The resulting very thin dead layer is an improvement over the diffused junction with respect to particle energy degradation. However, the thin entrance windows are optically transparent, so that light photons can cause interference.

One other type of junction counter that has been produced is the p-i-n type. This offers the advantage of an increase in sensitive volume, since the sensitive region may extend to > 10 mm. This type of counter has an almost intrinsic region between p and n surface layers. One device that features this design is the lithium-ion-drifted detector.[15] Lithium, a donor, is diffused into p-type silicon or germanium. The n-p junction that results is put under a reverse bias, and the temperature is raised. This allows lithium ions to drift through the silicon or germanium, and the donor-acceptor level becomes balanced over a wide region. This intrinsic region is preserved in the crystal when the temperature is lowered and serves as the sensitive region of this n-i-p detector. The lithium ion mobility at room temperature in germanium is such that to preserve the drifted state, the (GeLi) crystal must be maintained at liquid nitrogen temperatures. For SiLi crystals, ion mobility is much lower so that these detectors can be stored at room temperature.

Single crystals of Ge can now be produced with enough purity so that lithium drifting is not required.[6] Large volume (approximately 100 cm^3) are now available, and these need not be kept at liquid nitrogen temperature at all times. They are operated at liquid nitrogen temperature to reduce electronic noise, but may be stored at room temperature.

Because of the increased sensitive volume, these detectors are useful for x and γ rays. GeLi or Ge (Z=32) detectors are better for higher energy γ, and SiLi (Z=14) are more efficient for lower energy photons. For a given barrier depletion depth, because of its higher density, Ge will stop a particle with twice the energy that Si can.

Other semiconductors which are being utilized more frequently are cadmium telluride (CdTe) and mercury iodide (HgI_2). CdTe has been used in personal dosimeters and as probes in wound monitors. The high Z number and high density of HgI_2 have been utilized for low energy x and gamma spectrometry.

C. Scintillation Devices

Originally, the scintillation method of counting was used in the experiments of Lord Rutherford. The light pulses registered by scattered alpha particles interacting in a scintillator were counted visually. So, only low counting rates could be handled. When electronic instruments entered the picture, the scintillation method lost favor and was dropped. However, with the advent of the photomultiplier tube in the 1940s, this method has once again achieved wide acceptance.

The scintillation method is very sensitive for almost all forms of nuclear radiation. For this reason, it is one of the more useful ways to detect radiation. Since the energy of the radiation can also be measured, this method may be used for other purposes than simply to detect radiation.

1. Phosphors

In terms of the band-theory view of energy levels in a dense substance, the band gap ΔE (see Figure 10.3) may be pictured as a forbidden region. That is, in the normal state, no electrons exist in those energy states located in the band gap. In terms of quantum mechanics, these are not "allowed" states. We have seen that the presence of any defect in the substance alters the band structure. These defects result in charged centers (traps) in the surrounding areas. In essence, then, the band picture may change so that these traps exist in the forbidden region.

The trapping of electrons at these sites represents stored energy for the time the electrons are held. This excess energy is given up when the electron makes a transition to the ground state. A large part of the excess energy is transferred to nearby atoms in the form of heat. However, some substances have the property of luminescence. That is, some of the excess energy is emitted as light photons. These materials are called phosphors, and the light flashes (or scintillations) form the basis of the scintillation method.

The time that the electron is held in the luminescent trap (activator center) varies and depends on the substance. If the light photon is emitted in about 10^{-8} sec or less, this process is called fluorescence. If metastable states are involved, the lifetime may be much longer. This emitted light is referred to as phosphorescence. In this case, the light photons may be emitted for a span of from microseconds to hours.

2. Types and Properties of Phosphors

The phosphor used depends upon the type of radiation to be detected. All phosphors detect radiation that leads to ionization in the substance. Some of these substances have features that make them more useful for one type of radiation than for others. At present, five classes of phosphors are used in this method: organic crystals, liquid organic solutions, organic solid solutions (plastic scintillators), inorganic crystals, and noble gases.

Some important properties of phosphors for use in this method are: (1) the fraction of energy converted to photons should be large; (2) the phosphor should be transparent to its own emitted light; (3) the decay time for light pulses should be short; (4) the spectral distribution of the light should match the response features of commercial photocathodes; (5) the conversion of energy to light should be linear; and (6) the material should be of good optical quality and able to be produced in large sizes.[5,10]

For equal energy, an α produces less light output than electrons in organic crystals, organic liquid, and organic solid (plastic phosphors).[5] For inorganic phosphors, the alpha light outputs approach those due to electrons depending upon the phosphor. The light pulses from NaI and anthracene ($C_{14}H_{10}$) are about the highest. Zinc sulfide (ZnS) emits a large light pulse, but tends to absorb most of it. This requires that ZnS be used in thin layers.

Organic and noble-gas counters have small decay times, τ. The decay time for the organic phosphor stilbene ($C_{14}H_{12}$) is about 5 ns. Inorganic phosphors such as sodium iodide (τ=0.25 μs) have much longer decay times. The decay time τ is the time required for emission of 63% of the light photons. From this, we see that time resolution for these systems can be quite good.

Most of the present scintillator substances emit a light spectrum that can be well-matched to available photomultipliers. Sometimes one can improve the efficiency of liquid phosphors by adding wavelength shifters. These substances absorb light emitted by the phosphor and re-emit light of a new spectrum. Thus, one can obtain a better match in regard to spectral response.

More information about the features of many of these phosphors and scintillation detectors is contained in the literature.[5,10,16,18]

3. Detection of Pulses

Light flashes, or pulses, from the phosphor can be detected by means of a photomultiplier (PM) tube. The photons strike the cathode (called the photocathode). The surface of the cathode is coated with a substance that emits electrons. In this manner, the photon is converted to a stream of electrons. The efficiency of this entire process is such that between 0.3-2 keV/photoelectron is expended in most of these systems.[6]

4. Photomultiplier Action

The light produced in the crystal must be guided to the photomultiplier tube. The light strikes the photocathode, which has a thin layer of a photosensitive substance, such as bi-alkali (K_2CsSb). This substance then emits electrons. The number of electrons emitted depends upon the intensity and spectrum of the light as well as the photocathode substance. This number may be on the order of one electron per 18 eV of light energy. Near the photocathode is the first of a series of collecting electrodes called dynodes (see Figure 10.6). The unit may contain as many as a dozen of these stages. The first dynode is kept at a given positive potential with respect to the photocathode. Each succeeding dynode is kept at about the same given positive potential with respect to the preceding one. For a 10-stage tube in which the given voltage is 100 Volts, the total voltage above the cathode at the collecting plate would be 1000 volts.

The electrons that are freed at the cathode are attracted by the first dynode. By the time these electrons get to the dynode, they acquire enough energy to free more electrons from the dynode surface. In this way, the number of electrons increases or multiplies. When all these electrons, in turn, reach the next dynode, they again have acquired enough energy to dislodge more electrons. Since the accelerating voltage between any two dynodes is about the same, the number of electrons freed at each stage by an incoming electron should be about the same. After the last stage, the electrons are collected at the anode of the tube. The charge that is

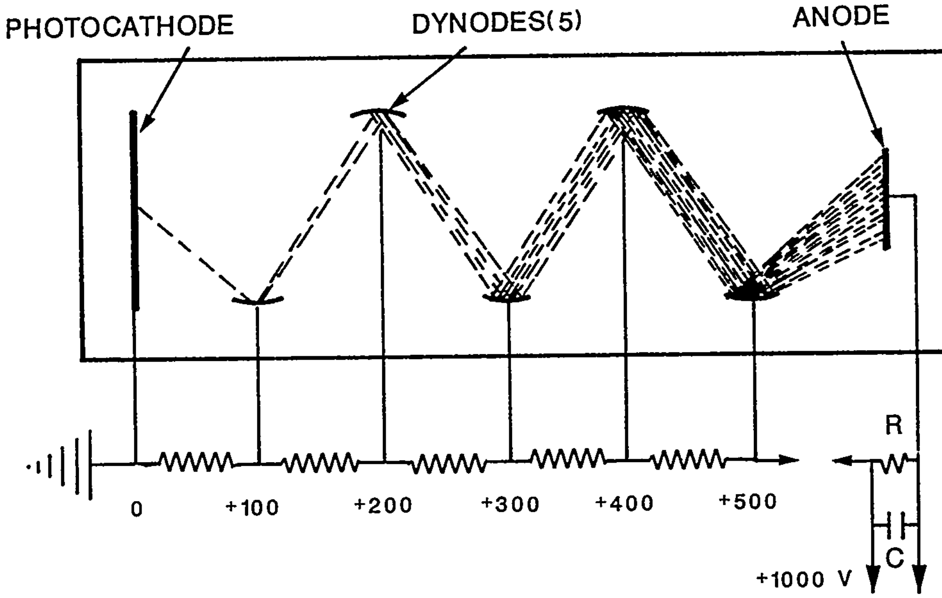

Figure 10.6 Illustrating photomultiplier action. (From ORNL-2808/ORINS-30, Fig. 5, p. 7)[17]

collected may then be applied in the form of a current to the external circuit.

The number of electrons freed at each dynode depends upon the amount of energy acquired by the electrons striking the dynode.The energy acquired depends, in turn, upon the voltage difference between dynodes.For a constant voltage maintained at each stage, a given number of electrons is produced at each dynode per incident electron. For many systems, this number is typically in the range of four to six. The number of electrons freed for each incident electron is called the secondary emission ratio, δ.[5] This ratio is also a function of the dynode surface substance, as well as the energy of electrons that bombard it. Common dynode surface materials are BeO, MgO, and Cs_3Sb.

At each stage, the input pulse is increased by the factor δ. The total multiplication in a system of n stages is then δ^n. In a ten-stage system in which $\delta=5$, the ratio of the output pulse to the primary photopulse is about 10^7. That is, $\delta^n=5^{10} \sim 10^7$. In this way, much larger pulses may result than would be obtained in ion-chamber devices. The overall gain is a function of the applied voltage V and is not exactly a linear function of interdynode voltage. In a ten-stage system then, the overall gain is more likely to vary as V^6 to V^9.[5] The shape of the pulse at the anode depends on the time constant of the anode circuit. That is, the product of the resistance, R, of that circuit (see Figure 10.6) and the equivalent capacitance, C, of that circuit. If RC is large compared to the scintillator decay time, then good pulse size and relative width (resolution) are the major objectives. If RC is small compared to the decay time, faster pulses result but at the expense of pulse size and resolution.

To be useful for energy analysis, the voltage must be very well regulated since small changes in the voltage are greatly multiplied. For example, assume the voltage changes by 0.1%, then for a ten-stage tube, the gain stability will be 1%. Large gain shifts cannot be tolerated if the system is used for pulse height analysis. For the anode circuit pictured in Figure 10.6, the final voltage pulse is given by equation 10.1,

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

in which C is the system capacitance, e is the electronic charge and N is the total number of electrons produced in the PM tube. For a stable system, N will be proportional to the energy of the incoming radiation. So, the voltage pulse ΔV will also be proportional to the energy of the radiation. If instability occurs, N will vary and the relationship will not hold.

5. Scintillation Counters

A device that detects the light flashes from phosphors is called a scintillation counter. The counter consists of a suitable phosphor, which is optically coupled to a photomultiplier tube, which in turn is connected to an electronic circuit. The design of the external circuit depends on the purpose of the device. If the counter is used simply to detect, the output circuit often consists of a battery-operated power supply, an amplifier and pulse shaper, and a rate meter. If the device is used for energy analysis, the output circuit includes a pulse-height analyzer and a scaler. For this type of work, the device is often a single or multichannel analyzer. Several types of counters are discussed in Section 12.

6. Optical Coupling

When light photons are emitted in the phosphor, they move through the crystal in all directions. As much of this light as possible must be directed to the photocathode. In many cases, the phosphor is encased in such a manner that light escapes in only one direction, called the window. That is, photons are internally reflected from all other boundaries until they are directed to the window. When this light reaches the boundary between the phosphor and the photomultiplier, a large portion of the light may simply be reflected back into the phosphor. To overcome this effect, a substance is used that will join the crystal with the photomultiplier so that light passes easily between them. Then, little light is reflected back into the crystal at the junction, and good optical coupling is the result.

In some cases, a substance known as a light pipe is placed between the scintillator and the photocathode. Lucite and quartz have been used in this manner. The light pipe helps to prevent the light from being trapped in the phosphor. Such a device does not eliminate the need for good optical contacts. The choice of the substance to be used to provide good optical contact depends upon the light spectrum emitted by the phosphor. That is, the substance must be highly transparent to the emitted light. Canada balsam has been used with anthracene. Silicone oil has been used with success for NaI crystals.

D. Chemical Devices

Several systems have used the chemical effects brought about by irradiation.[10,19,20-22] These effects result from the ionization produced by the radiation. In many cases, the processes leading to the chemical changes are not understood in detail.

One of the more widely used substances that depend upon these chemical changes is photographic film. Film can be used as a detector and as a dosimeter. Since film devices are important in personnel dosimetry, they are discussed in Section 13.

Chemical devices, both solid and liquid, have been used mainly as dosimeters to measure the absorbed dose. That is, the changes produced by the radiation can be directly related to the energy absorbed in these systems.[22] These products are more stable and somewhat easier to identify in liquid systems than in others. Yield is the term used to relate the changes to the energy absorbed. That is, the energy yield G is the number of changed units produced by an absorbed energy of 100 eV.[21] Systems with G values less than 20 are of most use in the hGy ($> 10^4$ rad) range.

We may class these devices in two broad groups: (1) those using water as a solvent, and (2) other systems. Among the first type are the ferrous sulfate (Fricke dosimeter) and the ceric sulfate dosimeters. The ferrous sulfate system is the only type reliable enough for use as a standard.[21] This unit is based upon the oxidation of ferrous ions to ferric in 0.1-0.8 N sulfuric acid. The ferric ions produced are then measured on a spectrophotometer. If care is taken to keep the amount of oxygen above a certain value in the system, we can measure in the range 5-500 Gy (500-50,000 rad).[19,20]

The ceric sulfate units are prepared by adding ceric sulfate to 0.1-0.8 N sulfuric acid. The reduction of the ceric to cerrous ions may also be measured spectrophotometrically. This system works well in the range 10^3-10^6 Gy (10^5 - 10^8 rad).[19,20]

Other systems in use are often calibrated by comparison with a Fricke dosimeter. Some systems, such as benzoic acid, quinine, trichloroethylene, and chloroform, liberate acid, which will produce color changes in pH-indicator dyes. The chemical changes may be measured in terms of dose by other means. References 19 and 20 describe other systems and give data on the response of these units to different types of radiation.

E. Solid-state Devices

1. Optical Properties

As discussed earlier, the defects found in solids give rise to charge centers that may trap the carriers. By their very nature, all solids contain some of these defects. The presence of these defects, whether they are impurity atoms or crystal structure defects, provide the basis for the use of solid-state devices to detect radiation. That is, many types of effects occur as the result of the type of trap involved. In some substances, the effects that may be studied include changes in the optical properties. Thus, a solid may absorb light in spectral regions where before irradiation the crystal was transparent. It may fluoresce, or it may fail to fluoresce under those conditions in which the normal solid does just the opposite. In a given case, the effect that can be studied depends upon the substance and the impurity type.

In organic systems, the changes that occur in the optical features seem reasonably permanent. In the inorganic systems, the nature of the crystal defects is a vital factor in regard to both the permanence and the magnitude of the response. In the present state of knowledge in solid-state physics, one cannot predict with too much success the effect of adding an impurity to a solid. However, studies[20, 23-25] have shown that certain substances, when added to a host crystal, form centers that lead to fairly permanent optical effects after irradiation. Other substances, called activators, result in luminescent centers in the crystal. These various centers often occur by chance in some crystals, so that doping is not always needed.

2. Trap Depth

The traps, activator, and other stable centers of the crystal are present in the forbidden-band region (see Figure 10.7). We may define the trap depth as the energy needed to raise an electron from the trap to the conduction band. This is shown as ΔE_t in the figure. We may

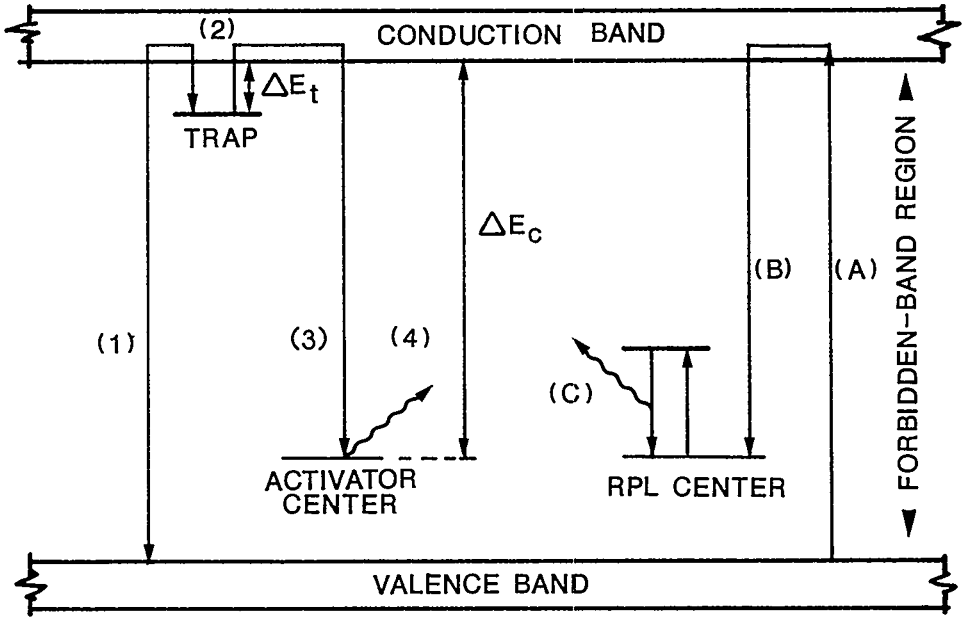

Figure 10.7 Trap effects in solids leading to thermoluminescence and radiophotoluminescence. (Adapted from Personnel Dosimetry Techniques for External Radiation (Symposium) Madrid, OECD/ENEA (1963).)[26]

treat an activator or other stable center as a trap with a very large trap depth, ΔE_c.

If an electron is trapped in the forbidden band, the time it remains trapped may be long or short. For a small trap depth, ΔE_t, the electron may soon be thermally released to the conduction band. For a large trap depth, ΔE_c, the electron may be held for a long time. The trap depth depends upon the type of impurity that formed the trap. The optical methods are most useful if the crystal has deep traps, that is, stable centers.

3. Optical Effects

We may obtain a somewhat rough visual reading from color or luminescence changes in almost all solids. In this sense, then, we can use these devices simply to detect radiation by these changes. However, in the area of dosimetry, the effects can be measured quantitatively. Thus, the readout is performed in most instances on devices such as spectrophotometers, colorimeters, and fluorimeters.

a. Color Changes

In many cases, the color changes that occur in certain glasses and plastics can be related to the absorbed dose. This effect results in a change of optical density, which can be measured when the device is illuminated with light. In this case, the radiation induces stable centers so that the solid now absorbs light to which it was transparent before.

The dose range in which glasses maintain a linear response extends from about 10 Gy to 10 kGy(10^3 - 10^6 rad). Plastics have been used with success in the range 1 kGy-1 MGy (10^5 - 10^8 rad). A poor feature of these devices is the fading of the color with time. The fading depends upon the absorbed dose that the unit is exposed to as well as the wavelength of the measuring light. This problem can be overcome in several ways;[27] therefore, this method is quite useful for high dose applications.

b. Degraded Luminescence

Some organic solids give off a glow when exposed to light of the proper wavelength. For example, anthracene gives off a bright blue glow when exposed to light in the near-ultraviolet region.[27] In this case, the centers formed by radiation result in a decrease in this type of response.

For high-level ranges, this method is better than using inorganic color-change devices. The useful range is from 10^3 to 10^6 Gy (10^5 - 10^8 rad), and the effect is quite stable. Some of these devices have not regained their lost luminescence ability after two years of storage.

c. Radiophotoluminescence

In some glass substances, most notably silver-activated phosphate glass, two types of stable centers are produced. The first type leads to the color change effect discussed above. The other type leads to a different effect when the exposed solid absorbs light. In silver-activated glass, an orange fluorescent light is given off when the dose glass is exposed to ultraviolet light. This effect, called radiophotoluminescence (RPL), is shown schematically on the right side of Figure 10.7. That is, an electron is raised to the conduction band (A) and is then trapped at the RPL center (B). The binding, or trap depth, is such that the illuminating light does not raise the electron back into the conduction band. The electron is raised to the excited state (C), but it quickly drops back to the RPL center. In the return process, orange light is given off. The fluorescence is sustained only for the time the solid is illuminated.

The intensity or brightness of the light can be measured on a fluorimeter and related to the absorbed dose. For this effect, as well as the other two that have been discussed so far, the readout does not destroy the centers. This means that these devices may be reread many times. Over the years, improvements in the response and development of new glasses has allowed the use of photoluminescence devices for personnel monitoring.[24] For this reason, the features of these devices are discussed in Section 13 of this report.

d. Thermoluminescence

Thermoluminescence can occur in substances that contain activator (luminescent) centers. This process is also shown on the left side of Figure 10.7. Again, an electron is freed by the radiation raised to the conduction band, and then trapped (1). When the crystal is heated, these trapped electrons can be raised to the conduction band again (2). The electron may wander about and be trapped by the activator center (3). The trap is then in an excited state and may attract a hole, which can recombine with the electron.[28] The excess energy is then given off as a photon (4) as the center returns to the ground state.

The light photons given off can be measured, and the absorbed dose can be related to the light intensity. As the solid is heated, the brightness increases to a maximum, which occurs when the rate of electron release is greatest. The peak intensity then decreases to zero as the store of traps is depleted. In readout, then, the stored energy in the traps is lost. This means that the device may not be reread. Since these devices are also of interest in personnel monitoring applications, more will be said about them in Section 13.

e. Stimulated Luminescence

In some cases, instead of using heat to remove the electrons from the traps, we may use infrared or visible light. This method, called stimulated luminescence, is thus very similar to thermoluminescence. The response of the two methods in regard to dose range seems to be about the same. The fading that occurs during storage in these devices must be overcome for stimulated luminescence to become a useful method.

f. Exoelectron Emission

In certain materials, notably BeO, trapped electrons may be released from the surface of the material when heated or exposed to light. This effect may also be used for dosimetry. When the material is irradiated, the radiation can cause electrons to move to the conduction band and these are then trapped. When the dosimeter is stimulated, the glow curve consists of these emitted e^-. The response can be detected with the use of windowless detectors or by using electron multipliers to amplify the signal.

F. Activation Detectors for Neutrons

1. General

Many of the devices described may be used to detect all types of radiation. With photons and neutrons, we must use the products released by their interactions in order to take advantage of these detection processes. Thus, many systems used to detect neutrons consist of a substance that enhances a type of neutron interaction leading to charged particles or photons. In this way, we can then detect the products of these processes by one of the previously described methods.[5] One process, activation of a nuclide, permits neutrons to be detected in a way that is not duplicated by other methods.

In some cases, the activation process refers to the ability of neutrons to induce radioactivity in the product nuclide. This activity can be measured and in turn related to the neutron fluence and sometimes to the neutron energy. When the fluence rate is constant, this fluence rate can also be found.[10,20,29] In this method, then, a stable substance is exposed to a neutron fluence rate for a known time span. The substance is then removed, and the induced activity is measured. If the fluence rate remains constant over the time t, then the activity A_t that results in the sample at the end of the exposure time is given by equation 3.64,

$$A_t = \phi\sigma_{act}N(1 - e^{-0.693t/T_{1/2}}), \qquad 10.3$$

in which λ has been replaced by its equivalent, $.693/T_{1/2}$.

In equation 10.3, σ_{act}, N, and $T_{1/2}$ are features of the sample that affect its choice in a given situation. The activation cross section, σ_{act}, most often varies with the energy of the neutron. If the fluence rate ϕ is composed of a spectrum of neutrons, we need to know the value of σ_{act} as a function of energy in the range of the spectrum. The total number of atoms, N, refers to the atoms that can be activated. Many samples contain a number of isotopes, and these may also become radioactive. In this case, many products may be involved. For this reason, we would prefer a substance that results in only one radioactive product. The half life, $T_{\frac{1}{2}}$, is for the product nuclide formed by absorption of a neutron in one of the N atoms. This activation product can have a half life that is too long or too short.[20] If the half life is too long, too much irradiation time may be required to obtain a sample of reasonable activity, or too much counting time may be needed for low activation. If the half life is too short, much of the activity may be lost before one can retrieve the sample and count it.

Most samples used are in the form of foils, pellets or wires. When the activation is done with thermal n, thin foils are needed to avoid flux depression and self-shielding effects (see Section 3.F.7).

2. Activation Detector Applications

Many of the present activation detector applications use the detector substance in the form of thin foils or small diameter wires. This allows ease in handling and presents other favorable features. We can class the applications in three broad categories: thermal-neutron detection, resonance measurements, and threshold detection.

a. Thermal-neutron Detection: Cadmium-difference Method

Thermal-neutron measurements can be made by the cadmium-difference method.[20] In this method, a foil is sandwiched between cadmium sheets. The thickness of the cadmium (approximately .5-1 mm) defines a cutoff energy (usually around 0.4 eV). The activation that results in the foil will be due only to neutrons with energy above the cutoff value. A bare foil of the same substance is exposed to the same neutron field. The thermal fluence can be determined from the difference in the readings, when appropriate corrections are applied.

Assuming both foils are irradiated to saturation, the relationship may be written as:

$$A_{bare} = A_{Th} + A_{res}, \qquad 10.4a$$

and

$$A_{res} = F_{Cd}A_{Cd}, \qquad 10.4b$$

in which A_{Cd} is the activity of the cadmium-covered foil, and F_{Cd} is a correction factor to account for absorption of neutrons in the cadmium cover. Then,

$$A_{bare} = A_{Th} + F_{Cd}A_{Cd}, \qquad 10.4c$$

and

$$A_{Th} = A_{bare} - F_{Cd}A_{Cd}. \qquad 10.4d$$

One may still have to apply perturbation corrections to the foils. The cadmium ratio is given by A_{bare}/A_{Cd}. The cadmium ratio is often used as a relative measure of how well a neutron field is thermalized. The larger this ratio, the greater the degree of thermalization. A number of useful thermal neutron activation detectors can be found in Reference 30. More discussion on the measurement of thermal fluence rate by direct determination of a reaction rate or by absolute counting of a radionuclide can be found in References 20 and 30.

b. <u>Resonance Measurements</u>

In some cases, the fluence of neutrons being slowed down may be assumed to vary as 1/E. The fluence per unit energy interval can then be found by using a foil of known resonance integral. The value of this resonance integral, I_R, has been measured for some substances, and a table of these values is contained in Reference 30. The method consists of exposing a cadmium- or boron-covered foil to the neutron fluence. The induced activity that results can then be related to the total fluence in the energy interval $(E_{max}-E_{min})$. The thickness of the cadmium defines the lower end of this interval, E_{min}. The activation of a resonance neutron spectrum is given by

$$I_R = \int_{E_{min}}^{E_{max}} \sigma_{act}(E)\frac{dE}{E}, \qquad 10.5$$

in which I_R is the resonance integral, in barns, and $\sigma_{act}(E)$ is the cross section for activation as a function of energy. In this case, the activity of the foil is given by

$$A = NI_Rk, \tag{10.6}$$

in which N is the total atoms irradiated and k is the fluence constant (n/m²). For a 1/E spectrum, k, is related to the fluence per unit energy interval by

$$\frac{d\Phi(E)}{dE} = \frac{k}{E}, \tag{10.7}$$

in which dΦ(E)/dE is the fluence per unit energy at energy E. From equation 10.6, the fluence constant k may be found, and

$$\Phi = k \ln \frac{E_{max}}{E_{min}} \left(\frac{n}{m^2}\right) \tag{10.8}$$

in which Φ is the total fluence and E_{max} is the maximum source energy. If the spectrum is not 1/E, a combination of several resonance detectors may be needed.[30]

c. Threshold Detection

The threshold detector concept is based upon the fact that the cross section for production of an induced activity in a given sample varies with the energy of the neutron. It may then be possible to select a substance in which no activation results until the neutron energy is above a certain value.[5] In the ideal case, the cross section is zero below E_T and constant above E_T, the neutron threshold energy. In practice, we rarely find such an ideal case. The true case may turn out to be that either the threshold energy is not so well-defined, or the cross section is not constant above the threshold, or both (see Figure 10.8). In this figure, the actual cross section value is variable and does not have a clearcut threshold. In the use of threshold detectors, the actual distribution is replaced by an idealized one in which E_T represents an artificial threshold and σ_{eff} represents a constant value of the cross section over the relevant energy interval. Expressed as a mathematical equation,

$$\sigma_{\text{eff}} \int_{E_T}^{\infty} \phi(E)\,dE = \int_{o}^{\infty} \sigma(E)\phi(E)\,dE, \tag{10.9a}$$

and

$$\sigma_{\text{eff}} = \frac{\int_{o}^{\infty} \sigma(E)\phi(E)\,dE,}{\int_{E_T}^{\infty} \phi(E)\,dE} \tag{10.10}$$

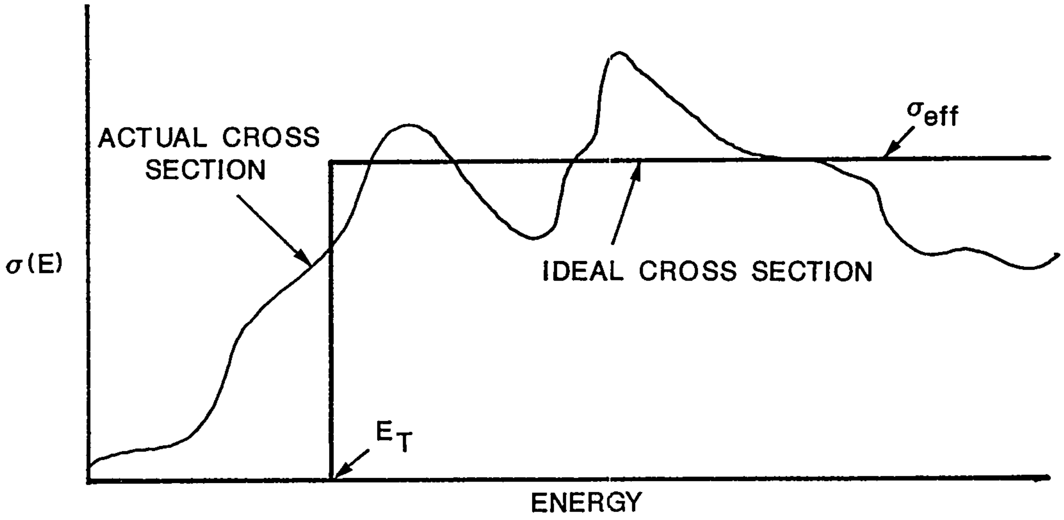

Figure 10.8 Idealized representation of actual cross section variations with energy.

In other words, the area under each of the curves in Figure 10.8 is set equal so that the same total activation occurs using the idealized cross section σ_{eff}. By using a set of threshold detectors, of differing threshold energies, the differences in the activation will allow one to crudely determine the fluence in a number of different energy groups. Tables of threshold activation detectors, with relevant threshold energies and reactions of interest can be found in References 5 and 30.

G. Solid State Track Recorders

A further method of utilizing radiation interactions for detection is the track etch method.[31-34] The devices are often called solid state track recorders (SSTR). In this method, ionizing charged particles, passing through a suitable dielectric material, leave a trail of damaged molecules.[5] In some materials, the small damaged area (approximately 5 nm diameter) can be enlarged by chemical etching, either with a strong acid (HF) or a strong base (LiOH,NaOH,KOH). The etch pits or tracks, can then be counted, either visually, with a microscope for low track density, or automatically.

Originally, this method was used in the 1960s to measure recoiling fission fragments produced in fission foils.[33] The fragments would leave damage tracks in mica or polycarbonate materials (lexan, makrafol, etc.). Following etching, the damage tracks appeared as channels or cylinders of diameter 10-20 μm and length about equal to the particle range (12-16 μm). The number of tracks formed was related to the number of fissions in the foil by

$$\text{tracks} \propto N\ (\text{fissile atoms})\sigma_{Fis}\,\phi t, \qquad 10.11$$

From the above relationship, the fluence $\Phi(=\phi t)$ of neutrons could be found.

This method had these advantages: no track fading, good sensitivity and applicable over a large n energy range. One disadvantage, in personnel monitoring applications, was that radioactive material was on the person of the wearer. This meant that the worker received a small radiation dose from the foil as it was being worn. To circumvent this difficulty, other radiators, such as ^{6}Li and ^{10}B, have been used to produce α tracks in the dielectric material. In other applications, the radiator material is the atoms of the dielectric itself, C, O, H and sometimes N.[35] The detector is usually a thin foil and automatic counting methods are used.

Track etch applications include n personnel dosimetry, indoor and outdoor radon monitoring (using alpha track registration), and other dosimetric applications. Since the registration of tracks requires a certain specific energy loss, the method is insensitive to electrons. The use of allyl diglycol carbonate (CR-39), as a recorder of proton tracks, has led to the development of a new neutron personnel dosimeter (see 13.E). By pre-etching CR-39 with KOH, the neutron track density can be greatly increased over that with no pre-etching. This allows detection of lower neutron doses of ~ 100 μSv (10 mrem).[36] The dosimeter is also insensitive to photon radiation.

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QUESTIONS

10.1 What main function of the health physicist pertains to radiation workers?

10.2 Upon what may the evaluation of the radiation field with respect to the hazard present be based?

10.3 What property of a radiation detection instrument is a measure of how well it detects?

10.4 Why is it necessary to have radiation detection instruments?

10.5 Name some of the basic processes by which radiation detection is accomplished.

10.6 By what two processes does radiation cause ionization?

10.7 What factors determine the number of ion pairs formed in a substance by radiation?

10.8 Why is the center wire of a gas-filled chamber for radiation detection the anode instead of the cathode?

10.9 What determines the size of the voltage pulse in a particular gas-filled chamber?

10.10 Indicate the two processes that compete for the ions formed in a gas-filled chamber.

10.11 What term is used to indicate that the current in an ion-chamber will not increase with an increase in voltage, but only as the amount of radiation increases?

10.12 What is the amplification factor for an ion-chamber operating at saturation current?

10.13 Name the type of gas-filled chamber in which the pulse size depends upon the initial number of ions produced in the gas.

10.14 In which gas-filled chamber are the electrical pulses for all detected radiations the same size?

10.15 What term is given to the voltage value for a gas-filled chamber where all pulses become the same size and the discharge occurs along the entire length of the center wire?

10.16 What is the relative size of Geiger-Müller counter pulses in relation to proportional counter pulses?

10.17 Define a semiconductor.

10.18 In terms of the band theory of solids, what are the three levels for insulators and semiconductors? Which one is essentially missing in a conductor?

10.19 When an electron is removed from a filled band, what is the term given to the vacancy that remains?

10.20 Since both electrons and holes formed by ionization of a semiconductor move under the influence of an electric field, they are both called: ________________________.

10.21 What term is given to the effect produced by the lagging movement of holes compared with the movement of the freed electrons?

10.22 What effect results when a semiconductor is "doped" with selected impurities?

10.23 What are the two general types of impurities used to dope semiconductors and what do they do?

10.24 What term is given to the type of semiconductors which contain both types of impurities?

10.25 What two processes impede or prevent the charge collection in semiconductors?

10.26 Explain the terms:

a) diffused junction
b) depletion layer
c) reverse bias
d) surface-barrier junction

10.27 What term is used for the group of counters which combine n and p substances? What is an n-i-p counter?

10.28 What is the name of the device that detects light flashes that occur in phosphors as the result of radiation?

10.29 Explain the difference between fluorescence and phosphorescence.

10.30 What device is used to detect and multiply the light signal obtained from scintillation phosphors? What are its principal electronic parts?

10.31 What are some of the desirable properties of a good phosphor?

10.32 What is the general term used to indicate the efficiency and the methods by which the light of scintillations in a phosphor is transferred to the phototube detector?

10.33 What are dynodes? Explain the function(s) they serve.

10.34 What is one of the most widely used chemical devices that can be used as a detector and a dosimeter?

10.35 Other than photographic film, what is the principal limitation of chemical devices from the standpoint of human protection against excess radioactivity?

10.36 Upon what do the optical properties of solid-state devices depend?

10.37 What term is used synonymously with the term "center" in discussing optical effects in phosphors?

10.38 What is a major problem in using color changes for dosimetry?

10.39 Explain the difference between degraded luminescence and radiophotoluminescence.

10.40 What is one drawback in the use of thermoluminescent devices for personnel dosimetry?

10.41 Explain a) stimulated luminescence
b) exoelectron emission

10.42 Describe the principle of a neutron activation detector.

10.43 What type of neutron activation detectors are used

(a) for measuring the fluence in a particular energy region and

(b) the fluence above a particular energy level?

PROBLEMS

10.1 The capacitance of a gas-filled chamber is 12 pF. A radiation particle causes the formation of 4×10^4 ion pairs in the gas. What is the size of the pulse in the ion-chamber region?

Answer: 5.33×10^{-4} volts

10.2 The maximum pulse amplitude in an air ion chamber is found to be 1.8×10^{-4} volts. If the system capacitance is 100 pF, determine the number of ion pairs formed. Also, if the average energy dissipated for each ion pair is 35 eV, what is the incident energy of radiation?

Answer: 1.13×10^5 ion pairs, 3.96 MeV.

10.3 The capacitance of a proportional counter is 8 pF. A radiation particle causes the formation of 6×10^5 ion pairs in the gas. The amplification is 150. What is the size of the pulse that is obtained?

Answer: 1.8 volts

10.4 A 20 keV electron is to be detected in a proportional counter and the minimum voltage pulse for detection is 10^{-3} volts. The system capacitance is 20 pF. If 34 eV are needed to produce an ion pair, estimate the amplification factor to be achieved.

Answer: 213

10.5 A gold activation foil weighing 10^{-6} kg was exposed 10 minutes to a constant thermal neutron fluence rate. Assuming the thermal neutron cross section of gold-197 as 98.8 barns and the half life of gold-198 as 64.8 hours, find the fluence rate if the sample activity is 500 Bq. Note: the short formula given in this chapter is suitable for short irradiation periods relative to the half life of the product and to thin foils (targets) provided counting begins immediately upon removal from the flux.

Answer: 9.3×10^{12} n/m^2 s

10.6 The secondary emission ratio of a ten-stage photomultiplier tube is 5. How many electrons reach the anode if 3×10^2 electrons are available from the photocathode?

Answer: 2.93×10^9 electrons

SECTION 11 - INSTRUMENT OPERATING CHARACTERISTICS AND COUNTING STATISTICS

The output signal that is produced in the detector system may be used in several ways. We can measure the average effect due to a large number of interactions in the system. This is called a current mode or mean-level detection system.[1,2] Many survey devices feature this type of design. On the other hand, we can measure the effect of single events in the system. That is, the output consists of several signals resolved in time. This is referred to as a pulse type system, since we measure the individual pulses produced by the radiation in the system.

A. Current Mode System

The current mode system makes no effort to resolve the pulses formed by the radiation. For a given amount of radiation, the output signal in some way reflects a summing of the individual pulses. As we increase the amount of radiation, the output of the detector also increases. In this system, we can relate the amount of radiation to the detector response. We can design these systems to measure either the total amount or the rate of delivery of the radiation. Ion chambers have been designed for both purposes, but other detection devices may also be used.

A measure of the current in an ion chamber gives a reading proportional to the ionization rate. In this method, the voltage drop across a known resistor gives a measure of the current in the external circuit. This induced current varies directly with the rate at which ions are being formed and collected. In turn, the rate at which ions are being formed depends upon the rate at which radiation enters the chamber. The current produced is low, 10^{-6} to 10^{-16} A, so that often electrometers must be employed when current mode is used.[1]

The total amount of radiation can be measured in a condenser type ion chamber. This chamber is such that we can maintain a stored charge on the unit. In this sense, it is very much like a charged condenser. When radiation enters the chamber, the ions formed reduce the stored charge. A measure of the change in charge equals the total ionization that took place. This quantity can then be related to the total amount of radiation that entered the chamber.

B. Pulse Type System

1. General

In the pulse type system, the individual pulses formed during the irradiation are resolved in time so that this system may be used to count individual particle interactions. Thus, the number of pulses at the output of the system is directly related to the number of interacting particles. That is, the number of particles interacting in the system multiplied by the detector efficiency gives the number of output pulses.

2. Pulse Counter

The output pulses from the detector are due to the collection of the charge produced by the radiation. This charge may be in the range 10^{-15} to 10^{-10} coulombs per pulse. The output pulses are fed to an electronic circuit, which often contains a pulse shaping system. This system acts on the pulse signals to produce a series of shaped pulses, which start from a common base line, but retain the size and time relationships of the input signals.[1] The shaped pulses may also be amplified and passed through a discriminator circuit. The system rejects smaller pulses, called noise, which appear in all electronic circuits, and passes only those pulses larger than a given size. The pulses then proceed to the scaling circuit. The scaling circuit allows us to add up the pulses as they arrive from the discriminator. The counts are often displayed by the use of light emitting diodes or liquid crystal display counters. The whole system may simply be referred to as a pulse counter. The average count rate in such a system is found by dividing the total counts obtained by the length of time the counter was operated.

C. Count Rate Meter

The count rate meter registers the count rate of a sample directly. In this device, the pulses that pass the discriminator are used to charge a capacitor, C (see Figure 11.1). Connected to the terminals of the capacitor is a fairly high resistance, R. The pulses are so shaped that as each pulse arrives, it puts a constant charge, q, on the capacitor.

The charging rate is then given by

$$\text{charging rate} = nq, \qquad 11.1$$

where n is the number of pulses arriving per unit time. The presence of the resistor causes some of the charge to begin to leak off as soon as it is put on. However, the next pulse adds more charge, but this will also tend to leak off. The leakage rate depends upon q', the charge present at any time. The magnitude of the leakage is given by

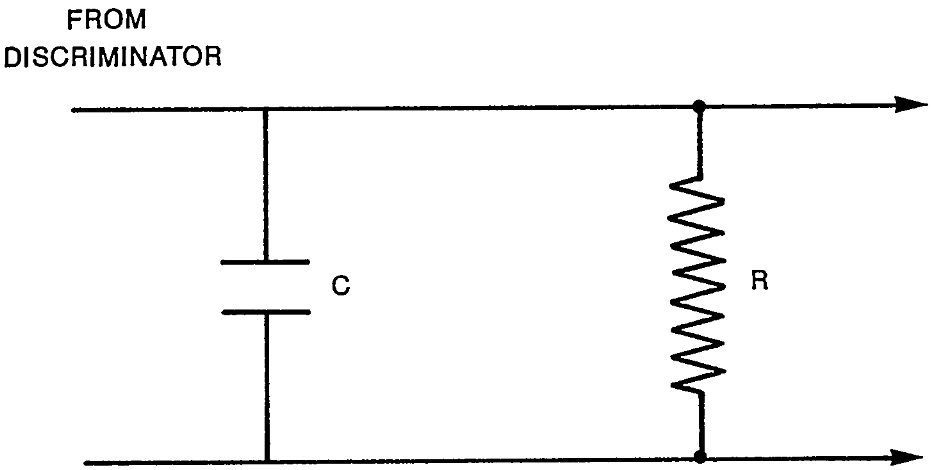


Figure 11.1 Count rate meter circuit.

$$\text{leakage rate} = \frac{q'}{RC}, \qquad 11.2$$

where RC is the time constant of the circuit. At first, the charging rate is faster than the leakage rate. Soon, the charge q′ on the capacitor builds up, and the leakage rate catches up to the charging rate. At this point, equilibrium is reached and q′ no longer increases. The voltage, V, measured across the capacitor is $V = q'/C$. But at equilibrium, charging rate = leakage rate

$$nq = \frac{q'}{RC} = \frac{V}{R}, \qquad 11.3$$

so that $V \propto n$. Thus, a measure of the voltage across the capacitor is a measure of the count rate. This is the principle of the rate meter. Since the value of R is fairly high, if C is also large, the time constant is large. In this case, the approach to equilibrium is slow and the meter reading is fairly steady. If C is small, the approach to equilibrium is rapid but meter fluctuations may be severe.

Some devices allow us to choose a number of different time constants by means of a time constant switch. In most cases, we can compromise between speed and steadiness. In making the choice, we must often take into account the count rate from the source. For low count rates, we may need a long time constant to reduce fluctuations enough to get a meaningful meter reading.

The meter indication in these devices may be either a linear or a logarithmic scale. For wide ranges of count rates, the linear meter is not suitable. Logarithmic meters, which cover more than five decades on a single scale, are available. These eliminate the frequent changes of range that are required in the linear devices. Sometimes, one may obtain an instrument with a digital display of the reading.

D. Counter Plateau

The discriminator circuit in a pulse counter overcomes the effects of the noise in the system. However, the presence of the discriminator requires that the signal pulses be large enough to be counted. We have noted that the pulse size in a detector varies with the applied voltage. For this reason, the number of counts per unit time in such a device also will vary with the applied voltage.

We might expect all pulses to be the same size if all the radiation entering the detector has the same energy. However, the system is not so precise in changing this energy to pulses. For a large number of interaction processes, a spread in pulse sizes results. That is, many pulses are about the same size, but some pulses are larger and some smaller than this size. In this way, we can say that only in a statistical manner is the pulse size proportional to the energy expended. This spread in pulse sizes is often referred to as a pulse size distribution.

Let us again treat the case of a two-electrode chamber, which we wish to operate as a Geiger counter. The problem is to find a means of determining the proper voltage. The circuit is designed to measure pulses that are equal to or greater than a certain size. For a given source, when the count rate reaches a maximum value, the counter should be operating in the Geiger region. The process of finding the voltage at which this occurs is called finding the counter plateau. If we place a $\beta\gamma$ source near our counter and apply the lowest voltage available in the system, no counts are recorded (see Figure 11.2). As we increase the voltage, a small count rate results at a certain voltage, V_1. The count rate increases rapidly for increase in voltage until we reach V_2. At this voltage, called the Geiger threshold, we enter a range in which the count rate remains fairly constant for changes in voltage. With further voltage increase, the count rate again begins to show a rapid increase (V_3).[3] If the voltage is increased still further, the counter may jam.

At the lower voltages ($<V_1$), the largest pulse size is still too small to pass the discriminator and be counted. At V_1, the largest pulses in the pulse-size distribution become large enough to be counted. Between V_1 and V_2, more and more of the pulses in the distribution

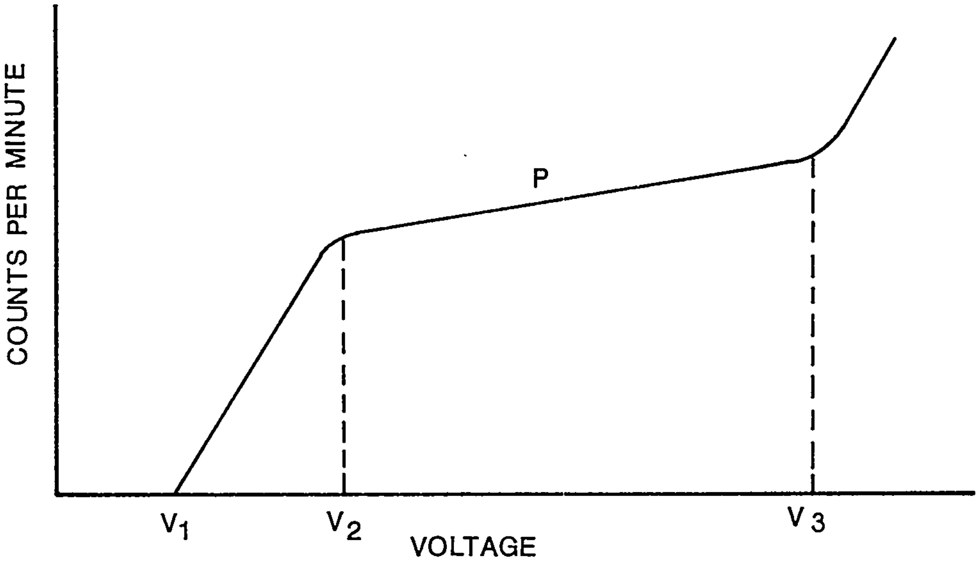


Figure 11.2 Counter plateau in Geiger region.

become large enough to be counted. At V_2, all the pulses are now large enough to be counted. This marks the start of the plateau region P. In a good counter, this plateau may extend for as much as 250 Volts. The plateau has a finite slope, approximately 2-3% per 100 Volts in organic quenched tubes.[1] This slight rise occurs because of an increase in sensitive volume of the counter and the increase in the number of spurious pulses in the counter. When the voltage reaches V_3, it is high enough to cause a breakdown of the quenching system (see Section 11.G), and a continuous discharge may result.

The voltage at which the counter is operated is taken as a point near the center of the plateau voltage range. In this way, voltage shifts do not affect the counting rate.

If the counter were designed to operate in the proportional region, we could find the plateau in much the same manner. In this case, the discriminator is set to pass the larger pulses that result from a higher number of primary ions being formed. As the voltage is raised, we obtain a plateau for the radiation that produces the most initial ions. For example, if the source emitted α, β, and γ the initial plateau region would indicate voltages at which all the α pulses pass the discriminator (see Figure 11.3). An operating voltage in this region would allow us to count only the larger α pulses.

Now, if we continue to increase the voltage, the counting rate once again increases, marking the end of the alpha plateau region. This increase in counting rate begins when some of the βγ pulses become large enough to pass the discriminator. For further voltage increase, a second plateau would result. This second region corresponds to the voltage at which all pulses now pass the discriminator. Because of the β energy spectrum, the statistical variation in the pulses caused by β may vary greatly in size. This leads to the steeper plateau shown for βγ region relative to that for the α plateau.

In the above system, we can obtain the α count and the total count. We then find the count due to βγ pulses by subtracting the two

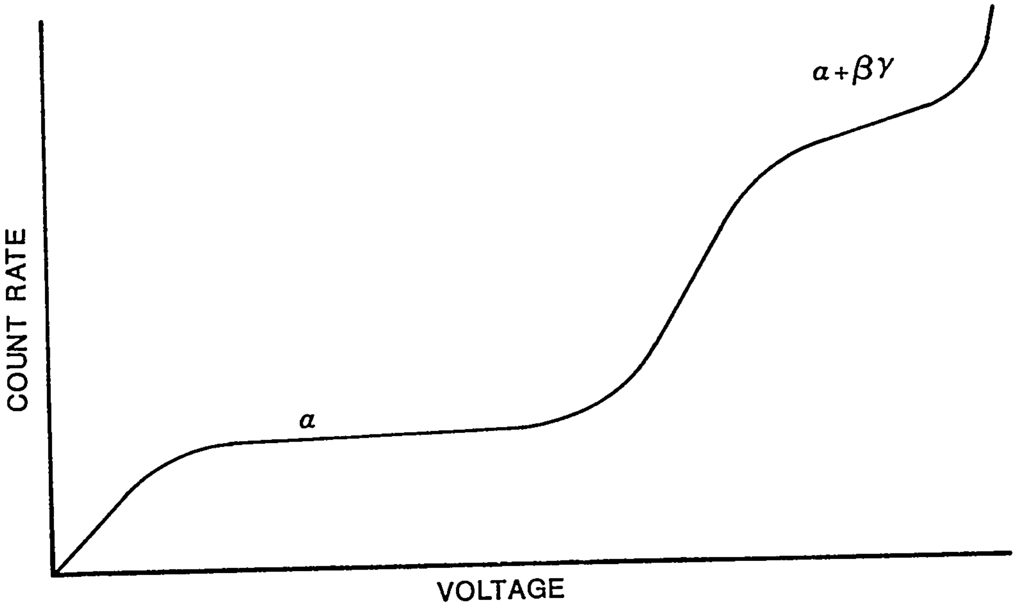


Figure 11.3 Plateau regions in proportional counters.

counts. In some designs, the $\beta\gamma$ as well as the α count can be obtained separately. With the proper detector, fast neutrons alone can be counted in the presence of γ rays.

Many situations require more information about the radiation than we obtain merely from a single count or count rate. We may desire to know which radionuclide is the source of the detected radiation, in addition to how much is there. To obtain this information often requires knowledge of both the half life and energy of the emission. It will also require more than a simple counting system. We can find the half life by following decay of the sample with the counter, if the half life is not too long. This is one way of gaining more information from the count rate alone. We can often determine the energy of the radiation by means of pulse-height analysis. The method selected; however, will depend upon the type and energy of the radiation.

E. Pulse-Height Analysis

Pulse-height analysis can be performed when the output pulse heights are proportional to the energy given up in the detector. In many systems, the energy given up is proportional to the pulse formed. For pulse-height analysis, though, we also need a linear amplifier. The design of this is such that the size of the output pulse is closely proportional to that of the input pulse. In addition, the pulse shaping features of the linear amplifier often dominates the performance of the system.[1] Pulse shaping refers to processing the pulses so that the long decay time (see Figure 11.4) of the pulse, is reduced to only a few μs.

1. Pulse Size

The output circuit of the detector is often designed so that the ions collected are used to charge a capacitor. The voltage appearing across the capacitor is proportional to the number of electrons collected.

The number of electrons collected depends initially on the radiation energy. Thus, the voltage appearing across the capacitor is a measure of the radiation energy.

Not all the charges will be collected at exactly the same time. Because of this, the voltage across the capacitor varies with time (see Figure 11.4). Pulse is the term used to imply that this voltage varies with time. The presence of a resistance in the input circuit allows the charge to leak off. The net effect is that the pulse voltage reaches a maximum value, V_1, and then declines. This maximum value that the pulse obtains is called the pulse height or size (usually measured in volts). The rate of decline is a function of the time constant of the input circuit. For a larger RC value, the decay is slower.

These pulses are fed into the linear amplifier, where the gain is designed to be independent of the pulse height. Each pulse is shaped

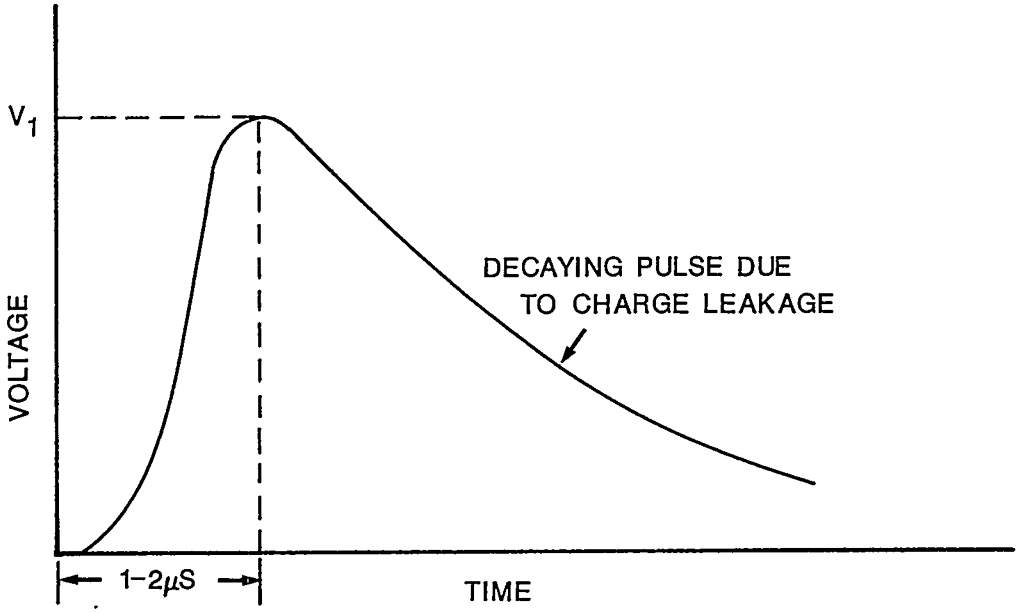


Figure 11.4 Variation of pulse size with time.

and increased in size linearly and moves on to the pulse-height analyzer. To be counted properly, the pulses are then converted to logic pulses. A logic pulse is a standard size and shape pulse which is recorded if it is present, ignored if it is absent. Whether it is recorded or not may be related to its time of appearance.[1]

2. Pulse-Height Analyzer

The basic element in the pulse-height analyzer is the discriminator circuit. The function of such a circuit is simply to accept all pulses above a certain size and produce a logic pulse or reject those below this size. The pulse size that is accepted is controlled by the circuit parameters. If the incoming pulse height is large enough to overcome the bias, the pulse is then accepted. A linear to logic pulse conversion then takes place. If the pulse size is not large enough to overcome the bias, no count results. In a pulse-height analyzer, this system has often been called the window level or lower level discriminator (LLD).

a. Integral Count

The output from such a system can be measured as so many pulses above a given level setting. The range of settings is often expressed in arbitrary units (0-100). We can obtain an integral count of all pulses above a given size as the level setting is varied. Such a count is shown as Curve A in Figure 11.5. As we lower the level setting from the high end (100), no pulses are recorded. This occurs because the size of all pulses in the distribution are less than the maximum level setting.

As the level is lowered, the number of pulses counted shows a somewhat steady increase. When the level reaches the low end (0), all the pulses in the distribution are counted. Of course, some of the circuit noise is also registered at this end of the range.

If the pulses produced by radiation of a given energy were all the same size, then we would obtain Curve B. In this case, the total

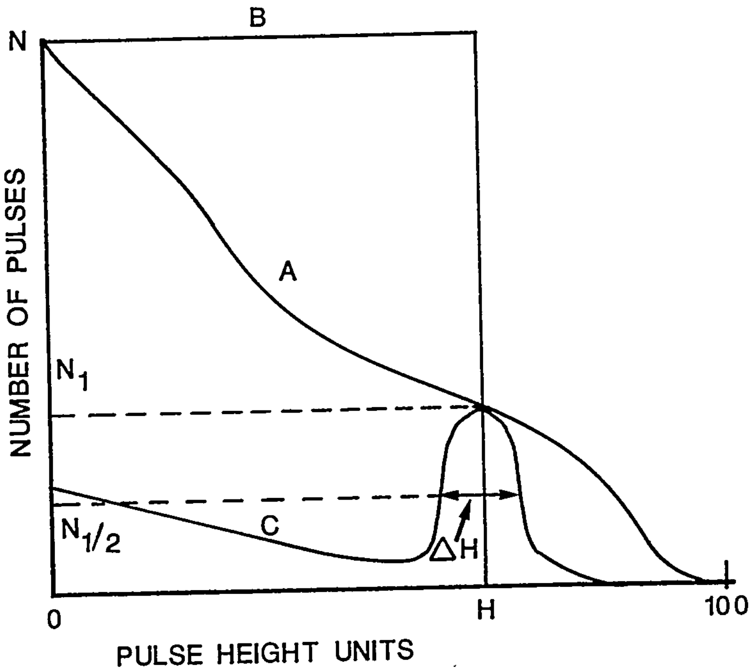


Figure 11.5 Types of pulse height spectra.

number of pulses counted would be constant (N) for all settings below H, and zero above this. Here H would be the pulse size of all the pulses. In most instances, we have a <u>distribution</u> of pulse heights. That is, one pulse size occurs more frequently than others that are larger or smaller than this size. The integral pulse height spectrum would appear as shown in Curve A. It is difficult to tell from Curve A which pulse height value occurs most frequently. For this reason, we would like to be able to count the pulses between two pulse sizes. This can be done by using a "channel" or "window."

b. <u>Window</u>

To provide the system with a window, a second discriminator circuit can be included. Like the first discriminator, this discriminator accepts all pulses above a given size and rejects all others. The second discriminator system is often called the <u>window width</u> system or upper level discriminator (ULD). The pulse size needed to trigger this circuit is greater than that needed to trigger the window level circuit. Now, all pulses above the window level setting and all pulses above the window width setting are accepted. The output pulses from both systems are fed into an anticoincidence or veto circuit. This circuit effectively subtracts the two outputs and passes on the difference to be counted. The counts then are due only to pulses that have sizes between the settings of the window level and window width.

The window width is often designed to be a fixed percentage of the maximum window level setting. Then the width of the window remains fixed even though the window level is varied. For example, suppose the highest window level setting represents a pulse size of 100 volts. Then, a 2% window width would mean that the channel size would be 2 volts. If we now move the window level setting to 60, the window width would still be 2 volts. This means, in effect, that we move the entire window when we adjust the window level. In many designs, we can choose the window width to be a relatively large fraction of the window level. The choice of the width in a given case depends upon the spectrum information desired. For example, one may center on the energy of a given radionuclide, such that one may selectively measure one radionuclide, while excluding others.

c. Differential Count

The pulse height spectrum obtained when we vary the window over the range of pulse heights is shown as Curve C in Figure 11.5. This spectrum type is referred to as a differential count or a differential pulse height spectrum. Now the pulse height that occurs most frequently in the distribution is more easily seen. It is the pulse size (H) at which the bell-shaped portion (often called a peak) of the curve attains it maximum value. Each such peak that occurs in the spectrum can be related to a discrete amount of energy lost by the radiation. Thus, this type of spectrum allows us to determine the energy spectrum of the radiation.

The portion of Curve C at pulse sizes less than that of the peak is due to pulses from radiation degraded in energy. As the number of peaks in the spectrum increases, the magnitude of this portion increases.

For a large number of peaks, the lower-energy peaks may be masked out by this portion of the spectrum as well as by circuit noise.

In large detection system, much of the radiation energy may be converted to pulses. Then, almost all pulses will fall in the peak area. In this case, we can distinguish peaks more easily. The term, resolution, describes the relative peak width. The resolution is often defined as the full width at half maximum intensity. Referring to Figure 11.5, Curve C, at pulse height H, the number of pulses is N_1, at maximum intensity. For half maximum intensity, namely $N_1/2$, the Curve C has a width, ΔH, in pulse units. From this,

$$\text{resolution} = \frac{\Delta H}{H}. \qquad 11.4$$

Often, the resolution of a detector is quoted in percent. Multiply the right side of equation 11.4 by 100 to obtain %.

The topic of pulse height analysis is much too complex to treat in all aspects in this short discussion. However, the subject is discussed in Section 12 with respect to its application in γ spectroscopy. The subject is also discussed in the literature.[1,3-5]

F. Resolving Time

In the previous treatment of counting systems, it was noted that the pulse is not collected instantly (see Figure 11.4). This time delay between the forming of the pulse and its being counted affects the counting accuracy, for if an event occurs that produces a second pulse as the first pulse is being collected, the second pulse may be lost.[6] The chance of losing pulses increases as the count rate from the source increases. At very high counting rates, the pulses may be spaced so closely in time that significant losses occur. That is, after the counter has processed a count, there is some time, called the recovery time, during which it is unable to perform another count.

We can compare counting systems in terms of their ability to handle high count rates. This can be expressed in terms of the resolving times of the systems. The resolving time, τ, is the minimum time between two separate pulses, such that both are counted. The lower this value, the more accurate the system is for high count rates. In some systems, τ is limited either by the nature of the detector or by the system's counting apparatus. In others, the resolving time may depend upon both of these components. Either the system manifests a fixed insensitive time (dead time) or the insensitive time varies depending upon the arrival time of the triggering event.

1. Resolving Time Correction

A knowledge of the resolving time of the system permits us to correct for counting losses at high rates.[1,7] Suppose a source has a true count rate N cps. Then, a system with a fixed τ should give an actual count of n cps. The counts missed in unit time (M) would be given by

$$M = N - n. \qquad 11.5a$$

For each of the n cps that the system records, there is a time, τ, in which the system does not respond. This time is unaffected by events occurring during its duration. The total time during each unit of time that the system is insensitive is $n\tau$. Then, the number of counts missed in unit time is also given by

$$M = Nn\tau. \qquad 11.5b$$

Thus,

$$N - n = Nn\tau, \qquad 11.5c$$

and

$$N = \frac{n}{1 - n\tau} \qquad 11.6$$

Now, suppose that τ is 200 μs, and the observed counting rate n from a source is 60,000 c/min. By use of equation 11.6, the true count rate N can then be found as follows:

$$N = \frac{n}{1 - n\tau},$$

$$n = 6x10^{4} \text{ c/min}$$

$$\tau = 2x10^{-4} \text{ s} = 3.33x10^{-6} \text{ min};$$

therefore,

$$N = \frac{6x10^{4}}{1 - (6x10^{4})(3.33x10^{-6})} = \frac{6x10^{4}}{1 - 0.2} = 7.5x10^{4} \text{ c/min.}$$

In this case, the effect is quite significant so that to ignore the correction involves serious error. We should always be aware of the effect of the resolving time on counting accuracy. In many cases, the counting rates are low enough so that we can ignore counting losses. However, we should know the approximate magnitude of τ for a given system. Then, one can calculate the count rate at which losses become excessive.

At high counting rates, a system may manifest effects which fall somewhere between those for a fixed dead time (nonparalysable system) and a variable dead time (paralysable system).[1,2] If the system performs according to the former, equation 11.6 will then be applicable. If the system performs more like the latter, the length of the non-counting time will not be fixed. In this case, the true count rate N is related to the apparent count rate n by[1,6-7]

$$n = N\, e^{-N\tau}, \qquad 11.7$$

in which τ represents the minimum time following <u>any</u> event, whether counted or not. For small values of $N\tau$, equation 11.7 gives the same result as equation 11.6. However, as N increases, the apparent count rate n reaches a maximum value (Ne^{-1}) and then continually decreases for further increase in N. In general, equation 11.7 must be solved iteratively to find N, knowing n and τ, except in the case of small losses (< 10%). Then, one may use[2]

$$N \sim n[1 - n\tau - \frac{(n\tau)^2}{2}]. \qquad 11.8$$

2. Resolving Time Determination

We can determine the resolving time of a counter by the multiple source method. In this approach, two sources of the same type of radiation of about equal strength can be used. It is also desirable to use two blank or dummy sources of the same size and material as the source mountings but inactive.

In this method, one source and a blank are counted for a certain period of time. The average count rate is found and recorded as n_1. The dummy source may then be removed and replaced by the second source. A count is taken with this setup, and the count rate recorded as n_{12}. We then remove the first source and replace it by the other blank. Another count is taken, and this count rate is noted as n_2. If the second source is replaced by a blank, a background count can be taken. This background count rate can be recorded as b. The resolving time, τ, is then[4,8]

$$\tau \sim \frac{n_1 + n_2 - n_{12} - b}{n_{12}^2 - n_1^2 - n_2^2} \qquad 11.9$$

Now, assume that the above procedure has been carried out. Let the results be: b = 30 c/min, n_1 = 6000 c/min, n_{12} = 12,800 c/min and n_2 = 7000 c/min. Solving for τ gives

$$\tau = \frac{6000 + 7000 - 12,800 - 30}{(12.8x10^3)^2 - (6x10^3)^2 - (7x10^3)^2} = \frac{170}{78.84x10^6}$$

$$= 2.156x10^{-6} \underline{\text{min}},$$

or $\tau \cong 129\ \mu s$.

The above approximate expression for finding τ assumes that the products, $n_1\tau$, $n_2\tau$, and $n_{12}\tau$, are much less than 1.[4] As a matter of fact, $n_1\tau$ and $n_2\tau$ should be no more than about 0.05, so that $n_{12}\tau$ is no more than about 0.1. For an exact expression for τ, using the 2 source method, see Reference 1. It is often of value to use other methods to determine τ.[1,2] However, even the simple determination described above allows us to estimate the range in which counting losses become important. For systems in which τ is rather large, such as the Geiger-Müller counter, high counting rates cannot be tolerated. In some systems, the dead time correction is accounted for by the use of "live time" counting. This feature is usually found on all present day multichannel analyzers (MCA). These systems usually have an internal clock, and the pulses are stored in a memory location (usually channel 0). When the analyzer is dead, these pulses are blocked and do not reach the memory, so that the total pulses in that channel give a measure of the live time. That is, the time the analyzer was actually counting. When the fraction of the dead time is large, errors can occur.[1]

3. Dead Time in Geiger-Müller Counters

The large values of resolving time for Geiger-Müller (G-M) counters are a direct result of the size of the discharge at the center wire. As the discharge grows along the wire, a large positive-ion sheath develops near the wire. This sheath moves toward the cylinder wall to be collected. The electrons are collected quite rapidly (on the order of a microsecond) by the central wire. The time for the positive ions to reach the cylinder wall may be as much as several hundred μs. The presence of this large positive charge in the vicinity of the wire reduces the field at the wire, and the discharge stops. These ions must then be swept away so that the field returns to normal and another discharge can take place. If an ionizing event occurs during the time these ions are being removed it will not produce a pulse. In a sense, then, G-M counters have a certain "dead" time - a period in which no pulses can be formed.[6] As some of the positive ions are collected, the field strength begins to recover. At some instant during this time span, the field will be strong enough for small pulses to be formed. These small pulses may or may not be counted, depending upon the circuit design. For this reason, the resolving time, τ, may be longer than just the dead time in a G-M counter. Sometimes the term recovery time is used to mean that time interval including the dead time and the time to produce the next full pulse.

If all the ions are swept away and become neutral, the field can return to full strength. Sometimes, in the process of ions being neutralized, electrons are released which tend to sustain the discharge. To avoid this multiple pulsing effect, some means of quenching is used.

G. Quenching

Quenching is the term used to indicate the process of ending the discharge in the counter. The quenching may be done by external or internal means. Electronic circuits can be designed that will reduce the voltage across the counter after a count occurs. In this way, the time needed to collect all electrons, even those that result as the ions are neutralized, can be guaranteed. To ensure this, the input circuit time constant must be long, thus leading to long resolving times.

By using a gas mixture in the chamber, we can achieve the quenching effect by internal means. Consider an avalanche or cascade produced in a counter filled with argon and ethyl alcohol. The mixture will typically be about 90% argon and 10% alcohol. The argon functions as the counting gas and the alcohol is the quenching gas.

When the cascade develops, the positive-ion sheath contains both argon and alcohol ions. As these ions move to be collected, collisions with neutral molecules may lead to electron transfer. That is, an argon ion that collides with an alcohol molecule may produce a neutral argon atom and an ionized alcohol molecule. This can occur because the ionization potential in alcohol is less than that in argon. By the time the sheath reaches the counter wall, it will contain almost 100% alcohol ions.[2]

The alcohol ions perform at least two important functions. First, alcohol strongly absorbs any photons that may be produced during the cascade. This decreases the probability of the photoelectric effect, which could result in a continuous discharge. Second, when the alcohol ions reach the wall and become neutral, the excess energy often causes the molecule to break up or dissociate. In the dissociation process, no photons are emitted so that no new discharge occurs.

The above description indicates that a good quenching gas should have these three features: 1) the ionization potential must be lower than that of the main gas; 2) the quenching gas should have broad and intense ultraviolet absorption bands; and 3) the quenching gas must tend to dissociate rather than emit photons when in an excited state. Many polyatomic gases have these necessary features; ethyl alcohol and ethyl formate have been widely used. Counting tubes that contain these gases are called organic quenched. This type of counter has a limited useful life of about 10^9 counts. This is due to the loss of about 10^9 molecules in each pulse because of dissociation. Some counting tubes use chlorine or bromine as the quenching agent. These halogen-quenched tubes have unlimited life since the halogens dissociate but then recombine. This type of tube is preferred for high counting rate applications.

The quenching process is affected by the magnitude of the applied voltage in the counter. As the voltage is raised to higher and higher values, the pulse size in the counter increases. The increase in the pulse size means that more ions must be neutralized in order to quench the discharge. When the pulses are extremely large, there is a great probability of quenching not being achieved. At this point, the counter may go into a continuous discharge, which the quenching gas will be unable to stop.

In tubes in which the quenching gas decomposes, we must exercise care to secure the longest life from the tube. Foremost, the counter should not be operated above the rated voltage. When the counter is not in use, we should not leave the high voltage on, since the unit is still counting background events although not recording them. Also, in the event of a continuous discharge, we should disconnect the high voltage to ensure that the discharge dies out. This is an important point since continuous discharge can ruin a tube in a short time.

H. Sensitivity

In health physics work, many types of samples may need to be counted. These samples may sometimes be measured by use of portable instruments. Such instruments are discussed in Section 12, along with other types of instruments. In other cases, the sample type is such that it can be measured by means of fixed counters. This is particularly true for smear samples and filter-paper samples. Sometimes even liquid samples may be evaporated and counted in these fixed counters.[2]

In general, since knowledge of the radionuclide is not always available, we must have counters sensitive to α, β, and γ. The sensitivity of these counters can be taken as a measure of their ability to detect the given type of radiation. The sensitivity is affected by such factors as the intrinsic efficiency of the detector, the background count rate, absorption factors, and the geometry.

1. Intrinsic Efficiency

The intrinsic efficiency of the detector expresses the probability that a count will be recorded if radiation enters the sensitive volume. This factor varies with the type and thickness of the detector and the nature and energy of the radiation. In gas-filled chambers, the intrinsic efficiency for charged particles is about 100%. In these counters, the value for photons is only about ½-1%.[6] In NaI scintillation devices, the intrinsic efficiency approaches 100% for low-energy photons. The choice of the detector type for a given radiation is governed in many instances by its intrinsic efficiency. We can refer to the intrinsic efficiency as the sensitivity of the detector.

2. Background Count Rate

The background count rate affects the sensitivity because it contributes counts that are not due to the source we wish to measure. We must then correct the counter results to take into account the background contribution. If the source count rate is very low, the background tends to mask the source count rate. We must then count for a very long time to reduce the effects of background. On the other hand, if we reduce the background by some means, the time needed to count this same source for reliable results is much less.[9] In this sense, we have then increased the sensitivity of the counter.

One of the more common methods used to reduce the background is to shield the detector unit. For many $\beta\gamma$ counters, the shield serves as the housing for both the detector probe and the source holder. In this way, we can maintain the distance between the sample and the detector in addition to achieving a reduced background. These shield units, called "pigs" are commonly made of lead or iron. Low background for α proportional counters is not as big a problem as for $\beta\gamma$ counters. Most of the background count rate is eliminated by the discriminator circuit so that only the larger, less frequent background pulses are counted.

When we wish to count samples of very small activity, the background must be further reduced. This reduction can be accomplished with the use of an anticoincidence shield,[10] which would consist of a detector surrounded by a ring of counters. In this, no count is recorded if a cosmic ray passes through the ring and the detector. Only those counts that occur in the detector and not in coincidence with counts in the ring are recorded. Other reductions occur when source radiations scatter in the detector and interact in the ring. Such an arrangement used with a standard shield reduces the background dramatically. Sometimes a reduction factor of 30-50 can be achieved with the use of an anticoincidence shield plus 0.1 m of Pb shielding.

Coincidence circuits have also been used to reduce background for special counting applications.[1] In these, a count is recorded only if pulses from two or more stations occur at the same time. This type setup is of most use when the radiation to be counted is somewhat directional. That is, the background arrives from any direction, but a beam of radiation is more confined. We can then design a telescope system - a number of counters in a line, which record a count only if all devices are triggered. In this way, we can also greatly reduce the amount of unwanted information. Also, for β decay in which a γ immediately follows, the source may be positioned so that the β and γ each interact in a different detector. To register a count will require coincidence, whereas background will generally be produced in one or the other, not both.

3. Absorption Factors

The presence of an absorbing substance in the path of the radiation can greatly affect the sensitivity of the counter. This feature is of most importance for α and β radiation. Charged particles are detected easily if they are allowed to enter the detector. The problem is then to ensure that only a small fraction are lost through absorption on the way to the counter. The scattering of radiation may also produce a loss of particles that would have been counted.

Absorption takes place in the sample, in the medium between the source and the counter, and in the counter window. Scattering occurs in the sample and in the intervening substance. In addition, some particles may be backscattered from the source-holder material. For the absolute determination of the source activity, we can measure these factors for a given counter for ourselves.[10,11] For other cases, some of the information contained in References 2, 4 and 12 may be useful. In many applications, only a relative value of the source strength is needed, and the value of these factors need not be known. That is, a standard source of known activity may be distributed in the same medium as the sample. Then, one need only compare counting rates since absorption should be comparable.

4. Geometry

In most cases, the sample is not surrounded by the counter, so that only a fraction of the radiation emitted reaches the detector. We can sometimes treat the sample as a point source. Then, the solid angle between the source and the detector defines the region in which radiation reaches the detector. The geometry, G, describes the solid angle subtended at the point source by the detector. It is often defined as the fraction of the total solid angle,

$$G = \frac{\Omega}{4\pi} \qquad 11.10$$

where Ω is the solid angle subtended, and 4π is the total solid angle about the point. The geometry expresses the probability of radiation being emitted into the cone of solid angle $4\pi G$. The solid angle Ω is defined as

$$\Omega = \frac{\text{area}}{(\text{radius to point})^2} = \frac{A}{R^2}, \qquad 11.11$$

in which A is the spherical area and R is the radial distance from the point to the spherical area (see Figure 11.6). For the case of a circular aperture detector, the solid angle may be expressed in terms of the parameters in the plane, and hence the geometry also.

The detector has a circular aperture of radius r and the point source, 0, is located on the axis a distance, a, from the front face of the counter. The geometry then depends upon the plane angle, θ. The larger this angle becomes, the larger is the solid angle. The geometry for this setup may be written as

$$G = \frac{\Omega}{4\pi} = \frac{1}{2}(1 - \cos\theta) = \frac{1}{2}\left(1 - \frac{a}{\sqrt{a^2 + r^2}}\right) \qquad 11.12$$

Sometimes the point source is not located on the axis as shown in Figure 11.6. At other times, the source is spread out more, so that it cannot be taken as a point source. Reference 10 treats these cases and

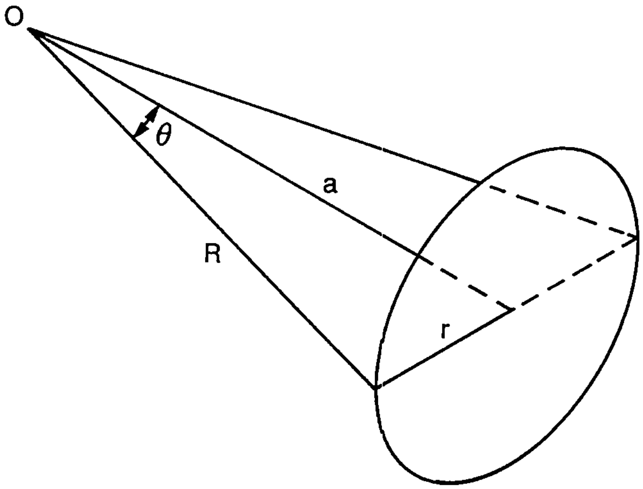


Figure 11.6 Geometry for a circular aperture and a point source.

contains modified forms of equation 11.12 that allow us to calculate the geometry. In addition, a reference is quoted for tables of solid angles subtended by disks and cylinders. Other references for complex shape geometries are listed at the end of Chapter 3 of Reference 1.

In the case of an end window tube, the sensitive volume does not start at the window. Charged particles travel some distance into the tube before a discharge is initiated. In this case, if the tube is positioned in a holder with a defined aperture, the geometry will be determined.

5. Absolute Sensitivity

From what has been stated, the sensitivity of a counter as a unit is influenced by several factors. Suppose we count a beta source, $T_{½} \gg$ the counting time, of unknown activity A (dis/min) with a β counter of intrinsic efficiency, ϵ_I. We would obtain a count rate from which the background rate must be subtracted to give the source count rate n (c/min). Let us assume that the count rate is low enough so that no resolving time losses occur. We may then write

$$n = SA, \qquad 11.13a$$

where S is the absolute sensitivity and is found from

$$S = \epsilon_I G f_b f_w f_s. \qquad 11.13b$$

The correction factors, f_b, f_w, and f_s, stand for backscatter, absorption in the counter window and air, and self-absorption and scattering in the source, respectively. If we have determined all the factors needed to arrive at S, then

$$A = \frac{n}{S} = \frac{n}{\epsilon_I G f_b f_w f_s}. \qquad 11.14$$

We can make an absolute determination of the source activity with a counter of known absolute sensitivity. Here we have assumed that one beta is emitted for each decay. If this were not the case, then the fraction of particles emitted per disintegration would also need to be included in the denominator of equation 11.14.

One further point, if the half life of the emitter is not long compared to the counting time, another correction needs to be made. The quantity n will represent the average count rate over the counting interval Δt. This will be related to the initial counting rate, n_o, at the start of the counting time interval, by

$$n_o = n\left(\frac{\lambda\Delta t}{1 - e^{-\lambda\Delta t}}\right) \qquad 11.15$$

For the case of a short half life emitter in which n_p is the fraction of disintegrations yielding the radiation of interest, equation 11.14 becomes

$$A = \frac{n_o}{n_p \epsilon_I G f_b f_w f_s}. \qquad 11.16$$

6. Yield

The determination of the absolute sensitivity is not easy. For this reason, we often use the relative sensitivity, which is found by means of a standard counting setup with a standard source. In this case, the source emission rate R (β/min) is known, and we can solve for the relative sensitivity. This quantity is sometimes called the counter yield, Y, and is found from

$$Y = \frac{n}{R} = \frac{\text{count rate (c/min)}}{\text{emission rate } (\beta/\text{min})} \qquad 11.17$$

With the aid of the decay scheme, the emission rate, R, is found from the activity, A (dpm). Thus we determine the true number of betas emitted per disintegration. For other radiations, the yield may be determined in much the same way.

The standard source should have a constant strength (^{238}U, ^{239}Pu) or decrease at a known slow rate (RaE). For the same features of source mounting, placement, and counter materials, the factors f_b, f_w, and f_s are about the same for all sources. Since counter function tends to vary with use, the yield should be measured from day-to-day. This often aids in pointing out counter malfunctions.

In the relative method, we compare the response of the counting system for the known source to that of the unknown. We assume that this is valid. However, this comparison is only roughly true for those sources which have about the same energy spectrum. For an emitter with a very different energy spectrum, the response could vary widely. In addition, for the unknown source, corrections may be required to account for a short $T_{1/2}$, high counting rate, and decay scheme! Each of these factors requires more than a single simple activity count to evaluate. From the practical, or operational, health physics standpoint, one is primarily interested initially in <u>detecting</u> the radiation and the yield method is satisfactory. If one is interested in the accuracy of source activity determination, then it is likely that serious errors may result by the use of relative counting.

I. Statistics of Measurements

When we make a number of counts for a given time on the same radioactive sample, the results vary. The fluctuations can be traced to both the counter and the nature of the decay process. We have already treated several factors that concern counter variations. Now, we will discuss the random nature of the disintegration process. By random, we mean there is no set pattern; a decay may occur, or may not occur, during a given time span. Also, the process of decay is independent of what has already occurred, so that each atom acts as a unit. The decay constant, λ, expresses the probability per unit time of decay occurring. We cannot be certain, though, that any given atom will decay in a stated time interval. At best, if the sample is large, we can calculate an average counting rate.[3] This behavior is inherent in the decay process and not a feature of the counting instruments.

There are several probability forms which may be used to describe certain features of randomness. These occur fairly frequently in our experience. The fundamental frequency distribution which governs random events is the binomial distribution.[1,2] This distribution for the probability P, is represented by the expansion of the expression $P = (p+q)^n$, in which p is the probability of success and q is the probability of failure. In this probability distribution both the success rate (p) and the failure rate (q) are fixed. That is, $p+q=1$. To be strictly applicable, this probability distribution applies to discontinuous variables, that is, integers.

The Poisson distribution is obtained for the case of a large number of trials with only a small probability of occurrence.[1,3] That is, the probability of observing the occurrence is small and constant. This distribution is applicable to many counting observations in radiation and nuclear physics. For example, the decay constant of a long-lived radionuclide is small, so that a large sample of atoms is needed to observe a rather steady count rate. This also applies to discontinuous variables; that is, integers.

The exponential distribution, of application mostly in reliability studies, is in effect, derived from the Poisson. It describes the distribution of time intervals with respect to successive events in a random process. An application of this distribution is the expression of failure rate of electronic components.

The normal,[2,3] or Gaussian, distribution is obtained when the output of a process is the sum of a large number of many small, independent factors. That is, the quantities have <u>continuously</u> varying magnitudes. For example, the distribution of stalk heights in a wheat field.

If the radioactive sample is large, one may calculate an <u>expected</u> number of counts in a certain time. The actual time needed to observe some number of counts is not at all the same every time. So that, even for a fixed counting time, repeated for a number of trials, the observed count rate varies. The expected count, for a number of k trials, will be closest to the true average count if we use the arithmetic mean, that is

$$\bar{N} = \frac{1}{k} \sum_{i=1}^{n} N_i. \tag{11.18}$$

1. Poisson Distribution and Gaussian Error Curve

For many counts taken on a <u>small</u> radioactive sample, the statistical fluctuations are quite dramatic. It is found that N, the number of counts in a given time, obeys a Poisson distribution, in which P_n is the probability of obtaining N cts:

$$P_N = \frac{N^{-N} e^{-N}}{N!}, \tag{11.19}$$

in which $\bar{N}$ is the average number of counts in the given interval. This gives a curve that is nonsymmetric with respect to the maximum value (see Figure 11.7a). The maximum value of the curve may be taken as the most probable number of counts, N_p, occurring in the given time. For a small source, then the value of N_p is less than the average number of counts, $\bar{N}$.

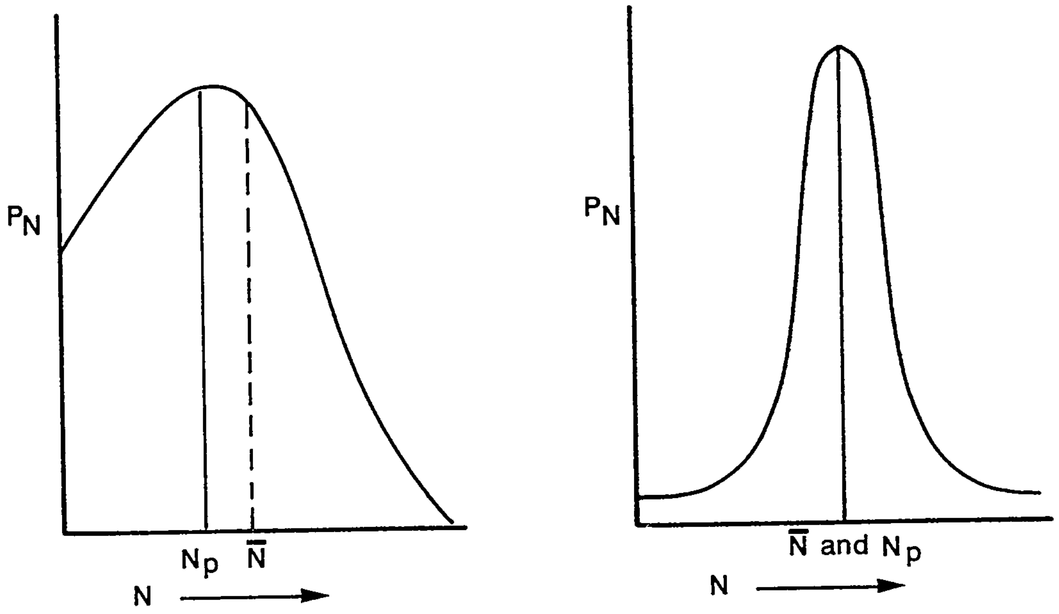
As an example, compute the probability of obtaining three counts in a given time interval, if the average number of counts in the interval is 4.

$$P_3 = \frac{\bar{N}^3 e^{-\bar{N}}}{3!} = \frac{(4)^3 e^{-4}}{3!} = 0.1954$$

If we obtain a <u>larger</u> activity sample, we expect more decays to occur, on the average, in the same given time as before. In this case, $\bar{N}$ would be much larger. For this condition, the distribution approaches

$$P_N = \frac{1}{\sqrt{2\pi\bar{N}}} e^{-(N-\bar{N})^2/2\bar{N}} \tag{11.20}$$

which gives a Gaussian error curve (Figure 11.7b). This curve is more symmetrical with respect to the maximum so that the most probable value,



(a) POISSON DISTRIBUTION

(b) GAUSSIAN ERROR CURVE
OR NORMAL DISTRIBUTION

Figure 11.7 Probability distributions for low and high values of N.

N_p, is the same as $\bar{N}$. Also, the ratio of height to width of the maximum increases. This means that as we go to larger activity samples, the variations in the counts become smaller. Therefore, for a given count time, we get a more <u>precise</u> count for the larger samples.

The Gaussian error curve is also called the normal distribution curve since this kind of distribution occurs in many other physical measurements. In our discussion of pulse height analysis (Section 11.E.2.c) the distribution of pulse heights turned out to be such a bell-shaped curve.

The width of the error curve gives the range of values over which the count varies from $\bar{N}$. Since $\bar{N}$ is also the most probable value, it is usually taken as the source count. In many cases, the value N from a single count is close enough to $\bar{N}$ to be adequate. Along with the value for the source count, we would also like a measure of the expected deviation from this count. This is supplied by the standard deviation of the count.

2. Standard Deviation of a Count

The standard deviation, σ, is related to the minimum value of the sum of the squares of the deviations from an average value.[1] If an average $\bar{N}$ counts are observed in a given time, then the standard deviation of the count is

$$\sigma = \sqrt{\bar{N}}. \qquad 11.21$$

The standard deviation expresses the range of values about the average count in which about two-thirds of all similar counts should fall. Since about one-third of the counts are outside this range, we have a measure of the <u>reproducibility of the type of count taken</u>. If the source is large enough, a single count, N, can be used to calculate σ. Suppose we count a source for 20 minutes and obtain 20,000 counts; then,

$$\sigma = \sqrt{N} = \sqrt{2x10^4} = 141 \text{ counts.}$$

We would then express the result as 20,000 ± 141 counts for this source counted for 20 minutes. Now, suppose we make a large number of counts on this source, all for 20 minutes. Then, on the average, two-thirds of the number of counts would be within the range 20,000 ± 141, and one-third would be outside this range.

As was mentioned above, when the source is relatively large, the distribution of counts goes from a Poisson toward a normal distribution. To attain this requires a large number of counts in a given interval. If one limits the counting time to a fixed, small value, then one limits the accuracy with which N may be determined. On the other hand, since $\sigma \sim \sqrt{N}$ in a single count, the accuracy will improve as the total number of events counted increases. For example, if we count long enough to observe 2500 counts, $\sigma = \sqrt{2500} = 50$ and the relative error = 50/2500=.02 or 2%. If we now count long enough to get 10,000 cts, $\sigma = \sqrt{10^4} = 100$ or 1%.

3. Error Designation

We can express the error in our count in several ways. The four most common errors are: the probable error (pe), the standard error (σ), the 9/10 error or 90% confidence level, and the 95/100 error or the 95% confidence level. The standard error or standard deviation is the one we have just treated. The other errors are related to the standard deviation through

$$\text{Error} = K\sigma = K\sqrt{N}. \qquad 11.22$$

The value of K is a constant for each of the stated errors (see Table 11.1). As in the case of σ, each error expresses a range of values about the average value. The size of the range depends upon the value of K. For the standard error, K = 1, so if K is larger than unity, the range is extended. We would then expect to find more than two-thirds of a number of counts falling in this range. As K gets larger, the range is such that the count is more reproducible. For K<1, the range is smaller than for the standard error, and the count is then less reproducible.

The probable error is the error for which the probability is equal that a count will be within the range or outside of it. In a large number of counts, half the counts would fall within N ± pe and half outside this range.

The 90% confidence level expresses the probability that 9 out of 10 counts fall in the range. Similarly, the 95% confidence level means 95 out of 100 counts fall in the range. Table 11.1 lists the constant, K, for these errors as well as the probable fraction of counts inside and outside the ranges.

TABLE 11.1

Types of Counting Errors

Name of Error	K	Probable Fraction within Range	Probable Fraction outside Range
Probable Error	0.6745	0.5000	0.5000
Standard Error	1.0000	0.6827	0.3173
90% Confidence Level (9/10 Error)	1.6449	0.9000	0.1000
95% Confidence Level (95/100 Error)	1.9600	0.9500	0.0500

The results of any count may be expressed in terms of any of the above errors. However, when quoting the results of a count, one should always specify which error is being used. For example, a sample is counted for 10 minutes, and the result is 10,000 counts. We wish to express the source count and its error. In this case, we will find the 9/10 error using equation 11.22,

$$\text{Error} = K\sigma = K\sqrt{N};$$

$$9/10 \text{ Error} = 1.6449\ \sqrt{10^4} = 164.49 \text{ counts.}$$

Our result is then 10,000 ± 164 counts at the 90% confidence level. The more commonly used designations of counting errors are the 9/10 error (90% confidence) and the 95/100 error (95% confidence level). The latter is sometimes called the 2σ confidence level, since K is so close to 2.0.

4. Standard Deviation of a Count Rate

In general, our interest is in the standard error in the count rate, n. In this case,

$$n = \frac{N}{t} \qquad 11.23$$

where N is the total number of counts obtained in the time t. Assume that any error in t may be neglected. Then, the standard deviation of the count rate, σ_n, is given by

$$\sigma_n = \frac{\sqrt{N}}{t} = \sqrt{\frac{N}{t^2}} = \sqrt{\frac{n}{t}}. \qquad 11.24$$

Let us express the source count rate and its error for the above example:

$$n = \frac{10{,}000 \text{ counts}}{10 \text{ minutes}} = 1000 \text{ cpm};$$

$$9/10 \text{ Error} = 1.6449\ \sigma_n = 1.6449 \sqrt{\frac{n}{t}} = 1.6449 \sqrt{\frac{1000}{10}} = 16.45$$

Our result is then 1000 ± 16 c/min at the 90% confidence level.

5. Background Influence on Standard Deviation

So far we have only treated the case in which one source was being counted. The presence of activity due to background radiation really means that we have two sources to contend with. In this case, the accuracy of reproducing the source count depends upon the error in both counts - source and background. When two counts are involved, σ is then

$$\sigma = \sqrt{\sigma_1^2 + \sigma_2^2}, \qquad 11.25$$

where σ_1 and σ_2 are the standard errors of the two counts. From before,

$$\sigma_1 = \sqrt{N_1} \text{ and } \sigma_2 = \sqrt{N_2}$$

so that

$$\sigma = \sqrt{N_1 + N_2}. \qquad 11.26$$

In this, N_1 is the total count of source plus background and N_2 is the background count. Suppose the background count is 300 counts for 10 minutes. The source plus background count turns out to be 6000 counts for 20 minutes, so that

$$\sigma = \sqrt{N_1 + N_2} = \sqrt{6000 + 300} = 79.4 \text{ counts.}$$

Once again, in terms of the <u>count rate</u>, the standard deviation is

$$\sigma_n = \sqrt{\frac{n_1}{t_1} + \frac{n_2}{t_2}} \qquad 11.27$$

where t_1 and t_2 are the counting times for sample plus background and for background alone, respectively.

6. Relative Standard Deviation

Let us define a relative error as the fractional deviation or error. In this case, the relative standard deviation, σ_r, equals the standard deviation divided by the source net count

$$\sigma_r = \frac{\sqrt{N_1 + N_2}}{N_1 - N_2} = \frac{\sqrt{n_1 t_1 + n_2 t_2}}{n_1 t_1 - n_2 t_2} \qquad 11.28$$

and

$$(\sigma_r)_n = \frac{\sqrt{\frac{n_1}{t_1} + \frac{n_2}{t_2}}}{n_1 - n_2} \qquad 11.29$$

for the relative deviation of a count rate.

By using the K values in Table 11.1, we can extend the concept to include the other errors. That is,

$$\text{Relative Error} = K\sigma_r. \qquad 11.30$$

The concept of the relative error enables us to make more precise measurements. Up to now, we have only talked about how reliable the count itself is. We have been able to state that a count under given circumstances may be reproduced with a certain confidence level. The condition we have been assuming is that our count is taken within a given time. To get more precise results, many counts must be observed. Thus, if we have low count rates, the counting time must be increased to obtain many counts.

To obtain this precision for a source with a low count rate, the total counting time depends upon both the source and background count rates. When the source strength approaches the background activity, long counting times are needed. The <u>total</u> counting time is a minimum for a given precision level if[1,2,9]

$$\frac{t_1}{t_2} = \sqrt{\frac{n_1}{n_2}} \qquad 11.31$$

We can illustrate the point with an example. Let a source give a counting rate of about 260 c/min in an area where the background is 92 c/min. The question is: How long must we count the sample to obtain a relative 95/100 error of 10%? That is, we would like to be within 10% of the source activity in 95 out of 100 times that we perform a similar count for the same total time.

In the first place, we assume a preliminary short count was made to arrive at the above figures. This is done to get an idea of the relative magnitudes of the count rates. Now, the minimum counting time, which includes the background and the source-count times, is

$$\frac{t_1}{t_2} = \sqrt{\frac{n_1}{n_2}} = \sqrt{\frac{260}{92}} = 1.663\cdot$$

In this case, we can see that to get precise results the source and background counts must be taken for comparable times. As the source rate gets even lower, we approach the case where we must devote equal time to the background and source counts.

Using the result from above, we substitute for either t_1 or t_2 in the expression for the relative error (re) of a <u>count rate</u>, using equation 11.29 and 11.30.

$$re_{95/100} = 1.96\sigma_r = \frac{1.96\sqrt{\frac{n_1}{t_1} + \frac{n_2}{t_2}}}{n_1 - n_2}. \qquad 11.32$$

Suppose we choose to substitute for t_2, that is, $t_2 = t_1/1.663$ so that

$$0.10 = \frac{1.96\sqrt{\frac{260}{t_1} + \frac{92(1.663)}{t_1}}}{260-92}$$

$$= \frac{1.96\sqrt{\frac{413}{t_1}}}{168}$$

then, $$8.571 = \sqrt{\frac{413}{t_1}}$$

$$(8.571)^2 = \frac{413}{t_1}$$

$$t_1 = \frac{413}{73.462} = 5.62 \text{ min,}$$

and

$$t_2 = \frac{t_1}{1.663} = \frac{5.62}{1.663} = 3.38 \text{ min,}$$

so that $t = t_1 + t_2 = 9.0$ min.

This is the total time we must count the sample and background to obtain a relative 95/100 error of 10%. The time we would count to achieve the same accuracy increases rapidly as the source strength approaches the background rate. Let our source rate equal the background rate; then $n_1 = 184$ cpm, and

$$\frac{t_1}{t_2} = \sqrt{\frac{n_1}{n_2}} = \sqrt{\frac{184}{92}} = 1.414.$$

Then, $t_2 = t_1/1.414$, which gives

$$0.10 = \frac{1.96\sqrt{\frac{184}{t_1} + \frac{92(1.414)}{t_1}}}{184-92}$$

$$= \frac{1.96\sqrt{\frac{314}{t_1}}}{92}$$

$$(4.694)^2) = \frac{314}{t_1}$$

$$t_1 = \frac{314}{22.03} = 14.25 \text{ min},$$

$$t_2 = \frac{t_1}{1.414} = \frac{14.25}{1.414} = 10.08 \text{ min},$$

and $$t = t_1 + t_2 = 24.33 \text{ min}.$$

In this case, the total counting time is about 2 3/4 times what it was before. If the source activity is less than the background rate, then even longer times would be required. In the above, if the source rate was half the background rate, we would have to count for about 83 minutes to obtain the same relative error. Reference 12 contains graphs and nomograms that can be used to find the ratio t_1/t_2 as well as the error in low counting rates.

In order to determine the total counting time, T, necessary to obtain a given relative error, we may make use of an expression adapted from Reference 9,

$$T = \frac{\left(\sqrt{n_1} + \sqrt{n_2}\right)^2}{\left(\frac{\sigma_r}{K}\right)^2 (n_1 - n_2)^2}. \qquad 11.33$$

The above expression, used in conjunction with equation 11.31 to determine the optimum division of the counting time, allows one to plan for the efficient counting of a large number of low activity samples.

J. Counter Reliability

The previous discussions have pointed out the factors for finding the error in the count. Through all this, we have assumed that the counter has functioned correctly. We may at times get data that perhaps would be "too consistent." One means of checking counter performance is provided by the "chi-squared test" of Pearson.[7] The test evaluates the probability that a certain data collection follows a Gaussian (normal) distribution. The method requires the obtaining of a number N(~20), of determinations. The quantity, χ^2, in which χ is the Greek letter chi, is then given by

$$\chi^2 = \frac{\sum_{i=1}^{k} (n_i - \bar{n})^2}{\bar{n}}, \qquad 11.34$$

in which $\bar{n}$ is the average value of the k determinations, and n_i is the value obtained for the ith evaluation.

To illustrate the method, we assume that 5 counts are taken on a given source (note that approximately 20 cts should be taken for better statistics). The results are 2000, 1980, 2040, 2030 and 1950. The average count is then 2000. To compute χ^2, we form the differences:

n_i	$n_i - \bar{n}$	$(n_i - \bar{n})^2$
2000	0	0
1980	-20	400
2040	40	1600
2030	30	900
1980	-50	2500

$$\sum_{i=1}^{N} (n_i - \bar{n})^2 = 5400$$

and, from equation 11.34,

$$\chi^2 = \frac{5400}{2000} = 2.7.$$

Tables[3,13] are available for the probability P of obtaining a value of χ^2 for a given degree of freedom (expressed as N-1). In general, the value of P for a perfect counter would be 0.5. Because of fluctuations and other operational factors, one may assume that a value of P such that

$0.1<P<0.9$ indicates proper counter operation. If, on the other hand, $0.98<P<0.02$, than the data is suspect. In our example, using Table 6-2 in Reference 3, the value of P would be between 0.5 and 0.1, indicating that the counter is operating correctly.

At times we may wish to know what type of counter to use for low counting rates. In this case, we can use the criterion that the best counter is that for which one obtains the smallest value of

$$\frac{\sqrt{n_2}}{Y}, \qquad 11.35$$

As before, n_2 is the background counting rate, and Y is the yield of the counter. The above value is as a "figure of merit" to compare different counters with respect to low counting rates.

Other topics concerning the statistics of nuclear detection are contained in Chapter 4 of Reference 1.

K. Minimum Detectable Activity

A difficult question to answer is the meaning of the term minimum detectable activity (MDA). This term has been used in many different contexts and depends very much on a number of parameters associated with the counting system. On the basis of counting statistics alone, two situations may occur.[2] From the data we may suppose we have a count above background but actually we do not. This is an error of the first kind-a false detection. An error of the second kind occurs when we assume there is no count above the background, but there actually is. These errors are often referred to as type a (first kind) and type b (second kind) errors, respectively. For a particular application, one must settle on the size of each error that can be tolerated. On the basis of $a=b=0.05$, one would expect 5% of false detections and 5% of false non-detections in a large number of determinations. Then, one may define a decision limit, L_C, and a detection limit, L_D.[2] These are

$$L_C = 2.32\sqrt{N_2} \qquad 11.36$$

and

$$L_D = 2.71 + 4.65\sqrt{N_2}, \qquad 11.37$$

in which N_2 is the background count for the given application. In equation 11.36, L_C is the minimum net sample count for which detection is assumed. In equation 11.37, L_D is the minimum net sample count which is likely to be detected.

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QUESTIONS

11.1 Identify the type of radiation detection instruments in which no effort is made to resolve the pulses.

11.2 Identify the type of radiation detection instruments in which each pulse detected is subject to being counted individually by a scaling system.

11.3 Identify the type of radiation detection instruments in which pulses are individually detected, but the readout is a count rate in terms of counts per unit of time.

11.4 What two rates are in equilibrium when the count rate of a count rate meter is constant?

11.5 What are the advantages and disadvantages of a long time constant for count rate meters?

11.6 Explain why the middle of the counter plateau for either Geiger-Müller or proportional counters is preferred for reliable counting.

11.7 What term is used to indicate the collection and recording of pulses from a radiation detector on the basis of size or energy?

11.8 Identify the pulse height spectra when

a) each part of the spectral curve is the sum of all the pulses or pulse energies above a particular discriminator threshold, and

b) when each part of the spectral curve represents the pulses having a specific restricted energy range.

11.9 In order to count pulses between two pulse sizes or energies, it is necessary to have an upper and lower discriminator to form a ______________.

11.10 What is the term used to indicate the minimum time between two pulses such that both are counted?

11.11 Explain the difference between the resolving time and the dead time.

11.12 What term is used to indicate the process of ending the discharge in a counter? How is this accomplished?

11.13 List some of the features of a good quenching gas.

11.14 What does the measure of the ability of counters to detect a given type of radiation define?

11.15 What does the probability that a count will be recorded if radiation enters the detector's sensitive volume define? On what variable does this depend?

11.16 When is the background count rate most likely to cause serious counting errors? What is the simplest method of reducing the background count?

11.17 How may radiation be "lost" causing a lower count rate?

11.18 What term is used to express the relationship of sample size and position to the size and position of the sensitive volume of the detector?

11.19 What is the result obtained by multiplying the intrinsic sensitivity of a counter by the factors of geometry, backscatter, absorption by the counter window and air, and self-absorption of the sample being counted?

11.20 What does the count rate divided by the emission rate from a sample define? What corrections should be made under what conditions to this quotient?

11.21 When does the Poisson distribution approach the Gaussian distribution in value?

11.22 What is the alternate term for the Gaussian distribution?

11.23 What term is used synonymously with the term standard deviation?

11.24 What term indicates that the probability is equal that a count will be within or outside its range?

11.25 Will 1000 counts per 10 minutes be equal to 1000 counts per 1 minute in counting error computations or is there a difference? Explain.

11.26 Upon what two factors does the relative error of a count depend?

11.27 Name the test recommended for determining counter reliability.

11.28 What fraction is used as a criterion for determining the best counter to use for low counting rates?

11.29 Explain the term minimum detectable activity.

11.30 What are type a and type b errors? How do decision limit and detection limit differ from each other?

PROBLEMS

11.1 a) If 3000 pulses/minute of $7\mathrm{x}10^{-9}$ coulomb arrive from the discriminator and the capacitance and resistance of the count rate meter are 5 μF and 10^6 ohms respectively, what is the voltage across the resistor at equilibrium?

b) What is the charge on the capacitor?

c) What is the time constant? Note: the time constant is that time in seconds for a capacitor to achieve 1/e of its total charge for a particular voltage.

Answer:
a) 0.35 volts
b) $1.75\mathrm{x}10^{-6}$ coulomb
c) 5 s.

11.2 A Geiger plateau runs from 950 to 1250 volts. Using the same radioactive source and geometry, the count rate at 950 volts was 15,000/min and 16,000/min at 1250 volts. What is the slope of the plateau in percent?

Answer: 2.22%/100 volts

11.3 a. Plot the following integral curve:

counts	PH Setting	counts	PH Setting
230	0		
230	10	90	50
220	20	40	60
160	30	10	70
130	40	0	80

b. Plot the differential curve using the preceding data.

11.4 Determine the resolving time from the following data: first source 11,000 cpm, second source 13,000 cpm, both sources together 23,000 cpm, and b=40 cpm.

Answer: $2.41\mathrm{x}10^{-4}$ s

11.5 The observed counting rate from a source is 42,000 cpm and the resolving time of the detector is 300 μs. Find the true count rate.

Answer: 53,165 cpm

11.6 Two counters have dead times 30 and 100 μs, respectively. At what true count rate will the dead time losses in the second counter be twice as great as those for counter one?

Answer: 1.333x10^{4} cps

11.7 A counter with negligible background gives 10,000 counts per second, with a standard source. An identical source is placed beside the first, and the counter now records 19,000 counts per second. With the first source removed, the count is also 10,000 counts per second. What is the dead time of the counter?

Answer: 6.21x10^{-6} s

11.8 What is the geometry, G, for a detector that has a 0.03 m diameter window and is situated 0.06 m immediately above a point source?

Answer: 0.0149

11.9 The α count rate of a 7.4x10^{4} Bq, Rn-222 source is 2870 cpm. The background rate is 9 cpm. Assuming that the α emission is 100% per disintegration, find the yield.

Answer: 6.44x10^{-4}

11.10 The yield of a counter is 0.052, the count rate 1650 cpm, and the background 20 cpm, find the activity of the sample.

Answer: 31,346 cpm or 522 Bq

11.11 A single count corrected for background and resolving time is 31,500 counts. Find:

a) the standard deviation
b) the standard error
c) the probable error
d) the 90% confidence level, and
e) the 95% confidence level

Answer: a) 177.5 b) 177.5
c) 119.7 d) 292.0
e) 347.9

11.12 How long (total time) must a sample be counted to obtain a relative probable error of 5% when the counting rate is 170 cpm and the background is 60 cpm? Assume the minimum total time conditions.

Answer: 6.50 min

11.13 A technician has the choice of two counters to detect α particles. Counter A has a background of 7 cpm and a yield of 0.02. Counter B has a background of 3 cpm and a yield of 0.016. Which counter should he use for low-level work?

Answer: Counter B

11.14 The following set of 30 observations were recorded under identical detector conditions and counting times (1-minute intervals). Apply the χ^2 test to see whether the observed fluctuations are consistent with Poisson statistics.

No. of Counts in 1-Minute Interval					
32	34	30	38	27	31
31	31	36	34	25	24
28	28	21	30	31	24
27	24	24	29	20	27
30	25	26	34	15	33

SECTION 12 - HEALTH PHYSICS INSTRUMENTS

Up to now, much has been said about how we detect radiation. In many instances, all that is required is a counter that records the events occurring in the detector. That is, suppose we count a source in order to measure the activity. Then, we can relate the counter response to the source activity by means of the counter yield. In other cases, the energy spectrum may be needed and pulse height analysis will be called for. A special need, which arises in health physics, is to measure the relative effect of radiation on human and/or other systems. To do this requires devices that do more than merely detect the radiation. In this case, we are concerned about the energy transfer to the system and the effect this produces. For this reason, the absorbed dose, which is the energy imparted to matter, is often chosen as the quantity to be measured. The field of study devoted to this aspect of the measurement problem is called radiation dosimetry.

A. Radiation Dosimetry

Of the many detection principles covered in Section 10, almost all have been adapted to measure the absorbed dose. Some of these methods are still in the process of development and will require further study. More information can be found in References 1-5 which contain discussions of possible methods. The discussion here will include some of the common aspects of radiation dosimetry.

1. Ionization Method

Perhaps the most common method used to find the absorbed dose has been to measure the ionization.[1] In this method, a gas-filled chamber operated as an ion chamber has been used in many applications. As the radiation enters the chamber, ions are produced as the result of interactions in both the chamber wall and the gas. For alpha and beta, the wall should be thin since it only acts to absorb energy that could produce ions in the gas. For this reason, many instruments contain a window (a thin section of wall to allow charged particles to enter).

For photons and neutrons, most of the interactions occur in the chamber walls. The secondaries that result produce the ionization. The problem is then to relate the ions collected in the gas to the energy imparted to a unit mass of the wall material. In this way, we can then obtain a measure of the absorbed dose in the given wall material. With the choice of a proper wall substance, we can measure the absorbed dose in some desired medium. For our purposes, tissue- and air-wall materials are of most interest.

a. Bragg-Gray Principle

The ions collected in the gas may be related to the energy imparted to the wall by the Bragg-Gray principle.[2,4,6,7] Let E_m be the energy imparted per unit mass of the wall substance. Now, suppose we have a small cavity of gas, which is surrounded by this substance. Let J_g be the ionization per unit mass of the gas. The Bragg-Gray relation is then given by

$$E_m = \left(\frac{S}{\rho}\right)_m WJ_g, \qquad 12.1$$

where W is the average energy needed to create an ion pair in the gas and $(S/\rho)_m$ is the relative mass stopping power of the medium with respect to the gas. If the wall and gas are of the same composition, then $(S/\rho)_m=1$, and

$$E_m = WJ_g \qquad 12.1a$$

and the cavity need not be so small. This simple expression will be valid if a number of conditions[6,7] are met. Some of these may be satisfied by proper chamber design. Others are features of the radiation interactions and may limit the usefulness of a given chamber design.

b. Air- and Tissue-Equivalent Material

The choice of the wall material for a given chamber design is governed by the measurement to be made. A chamber designed to measure the absorbed dose in air should have walls of air. Such a chamber could also be used to measure the exposure due to photons. A chamber designed to measure the absorbed dose in tissue must have walls of tissue. Also, we would like the gas to be the same material as the wall so that the stopping-power ratio would be unity.

In the free-air ionization chamber, or standard ion chamber,[4,6] the walls are really air. This device has been used as a primary standard in the past to measure the exposure in roentgens (2.58×10^{-4} C/kg). In most devices used for survey work, though, we cannot have true walls of air. Likewise, "tissue" walls are also difficult to obtain. But, for a chamber designed to measure the absorbed dose in either of these media, we must attempt to match the wall and gas to the particular medium. These chambers are then referred to as tissue-equivalent or air-equivalent, as the case may be.

For one material to match another, both must exhibit the same absorption for the primary radiation and the same stopping power for the secondary particles.[2] For a photon beam, an air-equivalent substance attenuates the beam just as air does. Also, the electrons, formed as the photons interact, are stopped in this substance just as they are in air. Since an air-equivalent substance is not truly air, it does not match the air response at all energies. For this reason, substances will be air-equivalent only in a limited energy range. For example, aluminum is an air-equivalent substance for photons, but only in the range about 0.3 to 3 MeV.

c. Energy Dependence

When a chamber is used to measure photons, the ionization in the gas is due largely to secondaries produced in the wall. The response of the chamber, therefore, depends upon the wall substance used. For an air-equivalent wall, the photon energy range is such that the attenuation coefficient of the wall substance matches that of air. Outside this range, the values depart from each other. When this occurs, the instrument response no longer follows the air response. To measure the absorbed dose in air, the response of the chamber should match the air response, regardless of photon energy. In photon-energy regions where this does not occur, the response is said to be energy-dependent. The energy dependence of a chamber results in a reading that is in error outside the energy range in which the wall substance is an equivalent material. The magnitude of the error in the reading depends upon design features of the chamber. The effect is most pronounced in the photon energy region 50-250 keV. A typical correction-factor curve for some instruments is contained in Reference 8. This figure indicates that the reading may be as much as a factor of 3 high or 7 low depending upon the instrument, condition of use, and the photon effective energy.

d. Charged Particle Equilibrium

For a beam of photons or neutrons that strikes a chamber wall, some of the secondaries ejected move toward the chamber interior. As the wall thickness, t, is increased, the number of secondaries at any depth in the wall increases (see figure 12.1, solid curve). The curve goes through a broad maximum, reaches a peak value at thickness t', and then slowly decreases with further increase in t. If there were no attenuation of the primary radiation in the medium, the curve would continue to rise as shown by the dotted portion (marked no attenuation) in Figure 12.1, eventually reaching a constant value throughout the medium. The dashed curve marked K in the figure, indicates the decrease in the primary intensity with thickness caused by attenuation in the medium.

The thickness, t', is called the equilibrium thickness and is equal to the maximum range of the secondaries produced. At this thickness, an approximate state of charged particle equilibrium (CPE) exists.

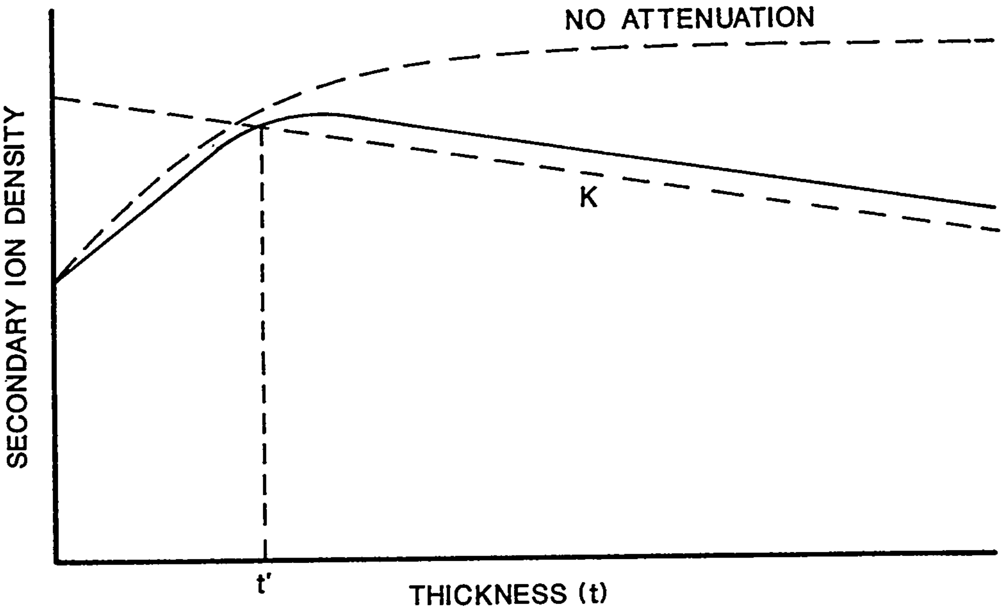


Figure 12.1 Equilibrium thickness, t', for charged-particle equilibrium (adapted from Reference 2).

That is, for every secondary that leaves a small volume about a point at this thickness, one with about equal energy would enter.[2] Then, the energy given up in the volume is the same as if all secondaries that originate there lose all their energy in that volume.

There will be a slight decrease in the beam intensity due to absorption of photons or neutrons as the thickness of the wall increases. For thickness > t', the secondary concentration falls at the same rate as the beam is absorbed. This reflects the fact that more secondaries are absorbed than produced. For this part of the curve, the ratio of primaries to secondaries attains a constant value so that a transient equilibrium results.[4]

In a given case, CPE would exist at a point if two criteria are fulfilled. First, the radiation field in a region that surrounds the given point should be constant. This region should extend for a distance equal to the maximum range of the secondaries produced. Second, in this same region, the primary attenuation coefficient and the secondary stopping power should be constant. For photons below about 3 MeV and fast neutrons, the above can be met with fair success. In some cases though, CPE will not exist. Some typical instances are: (1) near a source where the field changes rapidly with distance; (2) at a boundary between two different substances; and (3) for high-energy photons or neutrons.[4]

e. Kerma

As pointed out in Section 4.A.3, the kerma is the energy per unit mass transferred to secondaries (charged particles) by the <u>uncharged particles</u> (photons and neutrons) passing through a substance. In Figure 12.1, the dashed line marked K indicates the kerma in the medium. When CPE exists at a point in the substance, the kerma is equal to the absorbed dose. This assumes the bremsstrahlung losses are small. At thickness t′ in Figure 12.1, the absorbed dose (which is depicted by the solid line) equals the kerma at this point, and for t>t′, for transient equilibrium cases, the kerma is just slightly less than the absorbed dose.[6] The absorbed dose may be obtained by calculating kerma[2] as long as the mean free path of the radiation is large compared to the size of the medium. When the size of the medium is large, the absorbed dose is greater than that given by calculating the kerma for the medium in air. In this case, the dose at the point is no longer due to just the primary beam, as in the case of a small mass of tissue. Scatter, as well as multiple collisions, cause the dose to depend upon the size and shape of the object.[9] For a nondirectional field (not a beam), the true absorbed dose in a large tissue mass may even be less than the kerma. For a high energy photon beam, the secondaries which are released are high energy electrons. This leads to increased bremsstrahlung formation and more energy may be transferred to the medium than is absorbed in the medium. When this occurs, the absorbed dose could be less than the kerma in the medium.[10]

f. Ion Chamber Dose Measurements

We can use an ion chamber to measure the kerma under conditions of charged particle equilibrium. The kerma will then be equal to the absorbed dose at the point in the absence of scatter. That is, the chamber records the dose that a small mass of tissue receives. For a man at the same point, the absorbed dose that he receives is greater than that given by the kerma. Depth-dose data give a buildup on the order of 1.6 for fast neutrons and moderate-energy photons.[11] This factor may be much greater for lower-energy neutrons and high-energy photons.

An air-equivalent ion chamber, designed to measure the kerma in air, may also be used to find the absorbed dose in air. This follows, since, for a state of CPE, the exposure describes the kerma in air. The absorbed dose is then

$$D_{air} = K_{air} \qquad 12.2$$

where K is the kerma in air measured in Gy. Note that some presently used instruments measure the exposure in roentgens (2.58×10^{-4} C/kg). To convert readings of exposure in R to absorbed dose in air, in Gy, multiply roentgens by 8.7×10^{-3}.

On the other hand, a tissue-equivalent ion chamber enables us to measure the absorbed dose in tissue. Since the chamber responds to almost all radiation, we would read the total absorbed dose. We can also use more than one chamber of proper design to measure each dose component of the field.[4]

For charged particles, the absorbed dose can be measured by thin-walled ion chambers. Such chambers may also be used for uncharged particles when CPE does not exist. In each of these cases, the relative mass stopping power for the spectrum of secondaries must be evaluated at the dose point.

g. Other Ionization Dose Measurements

At present, the use of semiconductors to measure dose is not widespread. The main application has been to detect charged particles.

However, n-i-p silicon junction devices have shown promise as fast-neutron dose indicators.[12] Diffused-junction units may also be of value for thermal-neutron dose measurements.

The dose due to fast neutrons may be measured by a proportional counter in which the pulse heights are summed.[13] The chamber can be designed so that only pulses due to neutrons in a neutron and gamma field are counted. On the other hand, we can design a proportional counter to measure only gamma pulses in the presence of neutron pulses.

2. Other Dosimetry Methods

Since much of the energy lost by radiation in matter appears in its final form as heat, the absorbed dose may be measured by calorimetry.[4] In this case, the rise in temperature of the object can be used as a measure of the absorbed dose. This method presents many difficulties, which preclude routine use in survey work at this time.

As pointed out in Section 10, chemical effects have been employed to measure the absorbed dose. In practice, the dose range covered by these systems limits their usefulness for health-physics work. One form of chemical dosimetry, namely film dosimetry, is quite useful to health physicists (see Section 13).

The optical properties of solid-state devices have also been used to measure dose. The utilization of these devices for health physics purposes is at present focused in the area of personal monitoring. Section 13 treats devices that have been of use as personnel dosimeters.

Scintillation devices have been used with success to measure the absorbed dose. With x and gamma rays, the main problem has been the energy dependence.[2] Neutron dose may be measured with plastics loaded with scintillating substances. The large light pulses due to proton recoils formed in the plastic by neutrons can be counted, but the smaller gamma pulses are rejected.

Recently, a neutron detector has been developed at the Chalk River Nuclear Laboratories by Ing and co-workers. The detector utilizes an elastic polymer in which superheated liquid droplets are dispersed. Neutron interactions with the droplets cause the formation of fixed gas bubbles which can be visually counted and related to the neutron dose. The detector covers a wide energy range and is insensitive to gamma rays.

3. Present Dosimetry Techniques

When a field to be measured consists of only one type of radiation, the choice of instrument and method may be clear-cut. A problem arises, though, when we must deal with a field of mixed radiation. The problem has two aspects. First, many devices respond to more than one type of radiation. This gives rise to the problem of choosing the proper instrument. Should each component of the dose be measured, or is only the total absorbed dose important? Second, an equal absorbed dose of two types of radiation can result in differing degrees of biological effects for each type. In this case, the dose-equivalent concept must considered. This allows us to scale up the absorbed doses in order to better compare the effect of different types of radiation on human systems.

Over the years, many attempts have been made to solve this problem. Some have been discarded, others have been improved, and newer approaches have replaced older ones. In any event, as progress in dosimetry is made, we should expect the techniques to reflect these advances. At present, three popular approaches are being used to arrive at the dose equivalent. First, we can measure the absorbed dose for each component of the field.[2] These doses are then weighted by the proper quality factor and summed to obtain the dose equivalent. Second, we can measure the total absorbed dose with a tissue-equivalent ion chamber. Then, if the dose delivered at each level of LET (linear energy transfer) is measured by some other method,[14] the dose equivalent can be found. Third, we can measure the fluence rate and energy spectra of the components of a mixed field, then we can compute the absorbed dose.[9] Again, the dose equivalent is obtained by using the proper quality factors. The last technique has been useful for neutron-dose determinations by threshold and other spectral detectors.

a. Linear Energy Transfer and Quality Factor Relationship

The second technique mentioned above referred to a measure of absorbed dose at each level of LET. The ICRU[15] defines this concept as the average energy that is locally imparted to a substance by a charged particle of given energy. In this sense, the term applies to energy given up in a limited volume. The concept is of value since it allows us to focus on the manner in which energy is absorbed. That is, given two particles with the same total energy loss, the amount of energy lost along portions of their paths may vary greatly. It has been found that these variations in the local energy loss can account for the differences in biological effectiveness. This implies that we can relate the RBE and Q to the LET of the radiation.[16,17] For photons and neutrons, the LET is that of the secondaries produced.

In practice, it is not easy to determine which losses are local. That is, we must define a range of energy loss or a distance from the main path, which is taken as local. For this reason, the ICRP recommends that the quality factor Q be related to LET_∞, the stopping power in water.[15] The stopping power, L_∞, gives the energy loss per unit distance regardless of where this energy is absorbed. A table of the L_∞-Q relationship, taken from Reference 18, is shown in Section 4 (Table 4.1).

b. Distribution in LET of Dose

The dose equivalent may be found then by measuring the absorbed dose at each LET, multiplied by the Q for that value of LET, and summing these products. The <u>distribution in LET of dose</u> is the function that expresses the absorbed dose at each LET. This function gives the fraction of the absorbed dose given up in each LET interval. The method devised to measure this distribution employs a tissue-equivalent, spherical proportional counter.[2] The method requires no knowledge of the radiation field and, thus, is useful for finding this distribution function near high-energy accelerators. A drawback of this approach is that both the equipment and the method are quite complex. These features preclude routine use of this method as a survey measure. However, measurements made at locations in which the radiation field can be assumed to be relatively constant can characterize a useful average quality factor.

c. Problems of Neutron Dosimetry

There are several problem areas associated with present day neutron monitoring. These aspects involve personnel monitoring devices as well as survey and dosimetry instrumentation. Some of these problem areas are the following:

(1) Energy dependence of the response of dosimeters and instruments to neutrons,

(2) Lack of detailed neutron spectrum information in most monitoring applications,

(3) The over-response of existing neutron rem meters for a spectrum with a large intermediate component,

(4) The interference in mixed-field measurements due to the presence of photons, and

(5) No one instrument or device adequately covers the entire range of neutron energy of interest in dosimetry.

With respect to the response of personnel dosimeters to neutrons, all exhibit some energy response problem. NTA film does not respond at all to neutrons below 0.5 MeV. In the range 0.5-0.8 MeV, the tracks are difficult to count and may be missed. For neutrons in the range 0.8-20 MeV, the system is adequate but less sensitive than is desirable.

Albedo dosimeters do not have a threshold response energy but exhibit a severe energy dependence, with sensitivity decreasing rapidly as neutron energy increases. In order to obtain a proper calibration factor for the albedo dosimeter, the neutron energy spectrum must be known and remain fairly constant in the location of interest. The use of CR-39 track recorders shows promise as a suitable detector over the energy range of interest.

Tissue-equivalent ion chamber devices can be used to measure fast neutron absorbed dose rates, but are not usable for energies below about 200 keV. No information is obtained concerning the quality factor, however. To determine a mean quality factor requires a knowledge of the neutron spectrum or a different measurement.

Instruments used to measure the dose equivalent generally make use of a fortuitous response which follows the ICRP curve of fluence per unit maximum dose equivalent (MADE) fairly closely for energy greater than a few hundred keV. These instruments over-respond by a maximum factor of 1.6-4 times in the region 5-10 keV. The error can be significant if a major fraction of the neutron spectrum lies in the intermediate range (0.5-200 keV).

Instruments whose response is relatively independent of energy, such as the Long Counter or the Eberline PNC (see 12.B.4.d) measure neutron fluence rates over the energy range of interest, but the conversion of the response to the neutron dose equivalent is uncertain unless the neutron spectrum is known. Since, in practice, a quality factor suitable for fast neutrons is used, the dose equivalent will be over-estimated if the measured field contains a large component of intermediate energy neutrons.

The Bonner spheres (see 12.B.5.d) used to determine neutron spectra are based partly on calculated response functions and partly on experimental determinations. High resolution spectrum information is not obtainable with these devices and the computed average energy and inferred quality factors are dependent upon the computer analysis, that is, on the iteration and smoothing procedures used to unfold the experimental data obtained with the various-sized spheres. The 9" and 3" sphere technique used to obtain an effective energy is a practical method used to characterize the neutron distribution, but this and the Bonner system yield calibration factors which overestimate the dose equivalent for highly moderated spectra. An additional problem of neutron spectrometers is the inability to measure the stray radiation fields around certain facilities. The small size of some leakage radiation fields, coupled with the much larger size of the spectrometer units, precludes any meaningful spectroscopy since the instruments require a relatively uniform field over the dimensions of the device.

As was mentioned, rem meters utilize adjusted moderators which produce a response which approximates the ICRP fluence/MADE relationship. By adjusting the moderator thickness, the response to incident fast neutrons can be made similar to that of tissue. However, this optimizes the device for only a limited energy range and the response may not be correct in a different energy range. The result is an inherent over-response when used to measure a moderated fast neutron spectrum. In many applications, we are attempting to measure just such a field, so our results should be conservative. We can afford this luxury because the present limits allow us the room to do this. A significant reduction in the dose limits would require a better estimate in the intermediate energy region. For example, the change in neutron Q will force some reevaluations.

Another difficulty occurs because a neutron field is almost always accompanied by a photon field. To adequately describe the mixed-field requires that both components be evaluated. The photon interference causes problems because of the relative response of neutron devices to photons, and vice versa. Corrections must be applied to account for these effects in absorbed dose measurements and calibrations. The difficulty is somewhat mitigated by using two instruments to measure the components of the field, one with about equal sensitivity for both types of radiation (a tissue equivalent chamber) and one with a lower sensitivity to neutrons than to photons (a CO_2-graphite chamber).

Finally, no one instrument or device has a response which adequately covers the entire energy range of interest for neutrons and their interactions in tissue. The problem is related to the concept of the dose equivalent, the product of a physical quantity (the mean energy absorbed per gram of tissue) and an assigned quality correction factor (Q). One may be able to measure the physical quantity in a tissue equivalent material, but the ratio of energy imparted to the mass of the object does not solely determine the biological effect of the irradiation. The microscopic distribution of the energy absorption with respect to amount and spatial distribution is also important. Since neutrons release primarily densely ionizing nuclei and particles of different energies, the fluctuations in the energy imparted are considerable. To attempt to characterize the radiation quality correction by a single mean value Q for this situation is often inadequate.

d. Depth Dose Equivalents

The ICRP[15] defines the dose equivalent index, H_I, as the maximum dose equivalent within a soft tissue equivalent ($\rho = 1000$ kg/m^3) sphere of diameter 0.3 m, which is centered on the dose point. In most cases, the maximum dose does not occur at the sphere center. Also, a small tissue mass irradiated in air does not receive nearly the dose that a large tissue mass does. So, the ICRP has divided the reference sphere into three regions. The outermost shell is the dead layer of tissue (assumed thickness of 70 μm) and is ignored with respect to dose determinations. The second region, from 70 μm to 10 mm, constitutes the maximum shallow dose equivalent index ($H_{I,S}$) region. From 10 mm to the core represents the deep dose equivalent index ($H_{I,d}$) region. The need to determine the shallow dose index has introduced an added impetus for beta dosimetry improvement.

The Department of Energy has adopted the ICRP system for specifications of dose equivalent index stated above, in its revised Order on Radiation Protection for Occupational Workers.

e. Summary

The ICRP favors the use of the dose equivalent concept to express its recommendations. For this reason, much effort is being devoted to the problems of dosimetry in mixed fields. As more is learned, new trends may spring up and better methods may then be proposed. For the present, any method that does not underestimate the true dose equivalent is acceptable from the protection standpoint.

B. Survey Instruments

Over the years, many devices have been used in health physics. These include instruments that only detect, as well as those that also measure the exposure or the absorbed dose.[1,20-22] Some of these devices are large immobile units so that we must bring the sample to be measured to these units. Others are designed to give a continuous reading in a given area and are referred to as "fixed" monitors. These include such devices as stack monitors and continuous air monitors (see Section 14). Many assay samples, such as smears, liquids, and filter papers, are handled by means of fixed counters. A large class of measurements calls for the health physicist to either detect or measure the field in a given area. These "surveys" require the use of portable devices that can be transported to the locale to give a quick measure of the field or activity. This type of device is often referred to as a survey instrument. Many factors affect the performance of these devices and a number of these are discussed in Reference 23, which reports on the results of a testing program. In addition, References 24 and 25 discuss performance testing of these instruments to assure their proper operation.

Many survey instruments have been developed and used over the years. Because of the dynamic nature of this field, we are faced with extreme difficulty in any attempt to present a current review of these devices. For this reason, only <u>some</u> of the many useful instruments are discussed here. As time goes by, some of these may be replaced by newer approaches.

1. Ion Chamber Dosimeters

In its simplest form, the ion chamber was the first instrument used to measure radiation. We merely need a two electrode system, which defines a collecting volume of gas. Then, the ions produced in the gas may be collected by means of an applied voltage to the electrodes. The charge collected can then be related to the quantity of radiation. We have seen that with the proper choice of wall and gas, the ion chamber can also be used to measure dose (Section 12.A.1.6). In a strict sense, dosimeters are not really survey instruments, but certain types are most useful in health physics work.

a. Electroscope

In its original design, the ion chamber was used in the form of an electroscope.[20] This device measured the total charge collected (see Figure 12.2). The chamber walls form one electrode and define the collecting volume of air. The central electrode is insulated from the rest of the chamber. A moveable fiber is attached to this electrode. Now, when a charge is placed on the electrode and the fiber, they repel each other. The distance the fiber moves away from the electrode depends upon the value of the applied voltage. If ions are produced in the chamber gas, they move to the wall and to the fiber. The collected charge reduces the voltage, and the fiber moves closer to the central electrode. The displacement of the fiber can then be related to the amount of charge being collected in the chamber. From this, we can determine the total dose delivered by the radiation.

The motion of the fiber in these devices is viewed by means of a microscope, which is part of the device. A scale in the eyepiece allows us to obtain a reading in dose units. The fiber can be charged with a battery voltage to a scale reading of zero, which represents fully charged. When looking through the eyepiece, we see the image of the fiber moving across the scale.

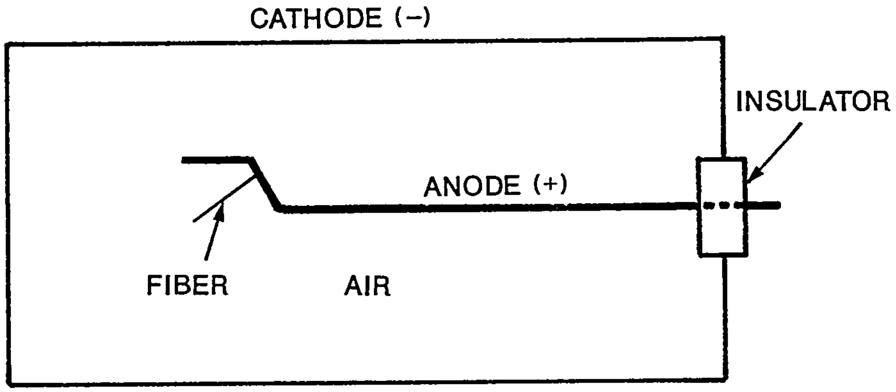


Figure 12.2 Quartz-fiber type electroscope.

At present, the electroscope principle is used in both the pocket chamber and the pocket dosimeter (see Section 13). The pocket chamber is similar to the pocket dosimeter except that it is not a self-reading device. This type chamber is used in conjunction with a minometer, which measures the charge.

b. Condenser Chambers

The condenser chamber is an air-equivalent ion chamber, which measures the total charge collected and, thus, may be used to measure the exposure or the air kerma. The device is often very simple. It may consist of a Bakelite shell, coated on the inner surface with graphite to make it conducting, and a well-insulated central electrode. When a charge is placed on the central electrode, the device becomes a charged condenser. Since we can measure charge in terms of the voltage change of a condenser, these chambers are used with an electrometer.[1] In this use, the chamber has a stem, which fits into a bayonet-type socket to make contact with the electrometer and the charging system. Again, we can observe the charge being placed on the device by noting the movement of a fiber image on the scale of the electrometer microscope. The system is fully charged when the fiber is at zero on the scale. When the chamber is removed, it can be capped to guard against leakage.

The chamber is then ready to be exposed to x or gamma at some location. The exposure can be read when the device is reinserted into the electrometer. The charge collected while in the radiation field is shown on the scale as a displacement of the fiber image. The scale on these devices is often marked in roentgens in many presently available chambers.

Chambers vary in size as well as in wall substance and thickness. The choice of wall substance and thickness offsets the energy dependence of the chambers. The total exposure a chamber measures decreases as the chamber size increases. Victoreen condenser chambers are available which cover the exposure range from 0 to 0.645 C/kg (0-2500 R). These devices can achieve a reading accurate to within a few percent in the design range.[26] These chambers, referred to as field instruments, are used for the accurate measurement of radiation fields, such as from calibration sources or teletherapy machines, and not as routine survey devices.

c. Tissue-Equivalent Dosimeter

Earlier in this section, we discussed the use of a tissue-equivalent (TE) ion chamber to measure the absorbed dose of any ionizing radiation. Such a device was developed by Rossi and Failla.[27] Further studies in this field have led to other plastics, whose response to radiation is close to that of tissue.[18] The composition of tissue may be taken as $(C_5H_{40}O_{18}N)_n$.[28] It has been found[18] that a mixture of methane, carbon dioxide, and nitrogen in the proper proportions, produce a TE gas, although TE gas has been produced with a propane base instead of methane. TE plastics have been made, which use polyethylene as the base substance and contain certain additives in the proper amounts. In this respect, we can add or replace a substance in the exact composition and still maintain adequate tissue-like response. This is necessary for the proper measurement of neutron fields of different spectral distributions.[18]

The chamber size and shape may vary with the purpose of the measurement. The TE plastic walls of these chambers are usually about 6 mm (approximately ¼-inch) thick. This thickness precludes the entry of low-energy charged particles, but is necessary for reliable chamber function.[27] The central collecting electrode is also of TE plastic, and the TE gas-fill is kept at less than 1 atm. As before, chambers of large volume are needed to attain the sensitivity needed for protection work. Chambers in the form of spheres with inside diameters of 0.2 m have been used for this purpose. Small-volume devices (thimble chambers) are used to measure very high doses.[26]

Charged particles with enough energy to pass through the chamber wall produce ions in the gas. Photons and neutrons interact with the chamber walls, and the secondaries produce the ionization in the gas. In this case, the W value is about the same in a given gas for all these products. This means that the response to each type of radiation is about the same. The charge collected then results from any ionizing radiation that enters the chamber. Because of the almost equal response, we can then obtain a reading of the total absorbed dose without knowing the components of the radiation field.

The TE dosimeter is used in conjunction with some type of electrometer circuit. The sensitivity can be increased if the chamber is combined with a vibrating-reed electrometer. Then the charges that arrive from the chamber as a function of time are added up to give a total reading. In this sense, we measure the total charge collected while the dosimeter is in a radiation field.

In this, as well as many other dosimeters, we can obtain dose-rate readings by noting the drift rate. Since the charge is not collected all at once, the meter needle moves to some scale value only after some time has elapsed. We can measure the time it takes the needle to move between two selected scale values. This can then be expressed as the drift rate in the given radiation field. We find that this drift rate varies directly with the dose rate. Hence, we can use this means to measure the dose rate in a radiation field.

2. Ion Chamber Survey Meters

If we include a suitable external circuit with the ion chamber, the collection of charge causes a small current to flow in the circuit. The current can then be read with an ammeter and related to the radiation intensity. Thus, we have a measure of the dose rate. Ion chamber survey meters are usually designed to give a measure of the exposure rate or the dose rate. In these, the circuit may include an ion chamber, an electrometer tube, and an ammeter. Batteries supply the voltage needed to ensure saturation. Since the entire unit weighs approximately 2.3 kg (approximately 5 lb.) it can be easily carried.

In a typical circuit, a variable resistance (zero control) is used to bring the external circuit into balance. That is, no current flows through the ammeter and the electrometer tube circuit. The grid of this tube is connected to the resistor in the output circuit of the chamber (see Section 10, Figure 10.1). When ion pairs are formed in the chamber, they are collected and cause a pulse. Any pulse in the chamber output is applied to the electrometer grid. This causes an imbalance in the tube circuit, and the current flows through the meter. The meter reading can be calibrated to read directly in the desired units.

Many ion chamber survey meters are designed along the lines discussed above. The resistance value in the chamber circuit may be chosen from three or four resistors by means of a switching arrangement. This allows the device to be used over a number of ranges of intensity. The higher-value resistors give the greater sensitivites (or the lower ranges). To protect the ammeter, always approach an unknown field with the range selector switch on the highest range. In this way, we can avoid surges of current, which can damage the meter.

Ion-chamber designs vary widely with the type of radiation to be measured. Thin windows are required for alpha and low-energy beta. When photons are involved, the wall substance becomes an important factor. Some chambers, which are designed to measure x and gamma exposure rates, include windows so that the device indicates alpha and beta also. In these cases, the scale reading is only qualitative since the device is not designed to measure dose rates for alpha and beta sources. Also, for any reading to be of value, the radiation field must be uniform over the dimensions of the chamber. With a large chamber, for example, we will be in error if we read exposure rates near a point source. The reading will be an underestimate and can be quite significant. For a beta source, the error can be a factor of 10 or more depending upon the geometry and the beta energy.

Ion-chamber troubles most often arise from degradation of the insulation.[21] Small amounts of dust can lead to insulation breakdown during temperature and humidity changes. Also, the collecting system is sensitive to stray pickup and must be shielded. For all survey devices, performance is affected by the useful life of the batteries.

The following is a brief description of typical survey instruments, although other ion chamber instruments are available and in use at many facilities.

a. Victoreen Panoramic Survey Meter

The Victoreen Panoramic survey meter, Model 470A (see Figure 12.3) is a portable instrument used to measure the exposure rate of photons (> 10 keV). It can also be used to detect alpha (> 8 MeV) and beta (> 120 keV). It utilizes an unsealed air ionization chamber composed of expanded polystyrene (275 cc volume and 0.17 kg/m^2 [17 mg/cm^2] wall thickness). The instrument is supplied with a Cycolac plastic equilibrium sleeve and removable end cap (5 kg/m^2 = 500 mg/cm^2). Low energy photon radiation can be measured with the sleeve removed. In the energy region between 10-300 keV, the energy dependent response is stated to be within 15% of the true reading. With the sleeve in place, the quoted energy

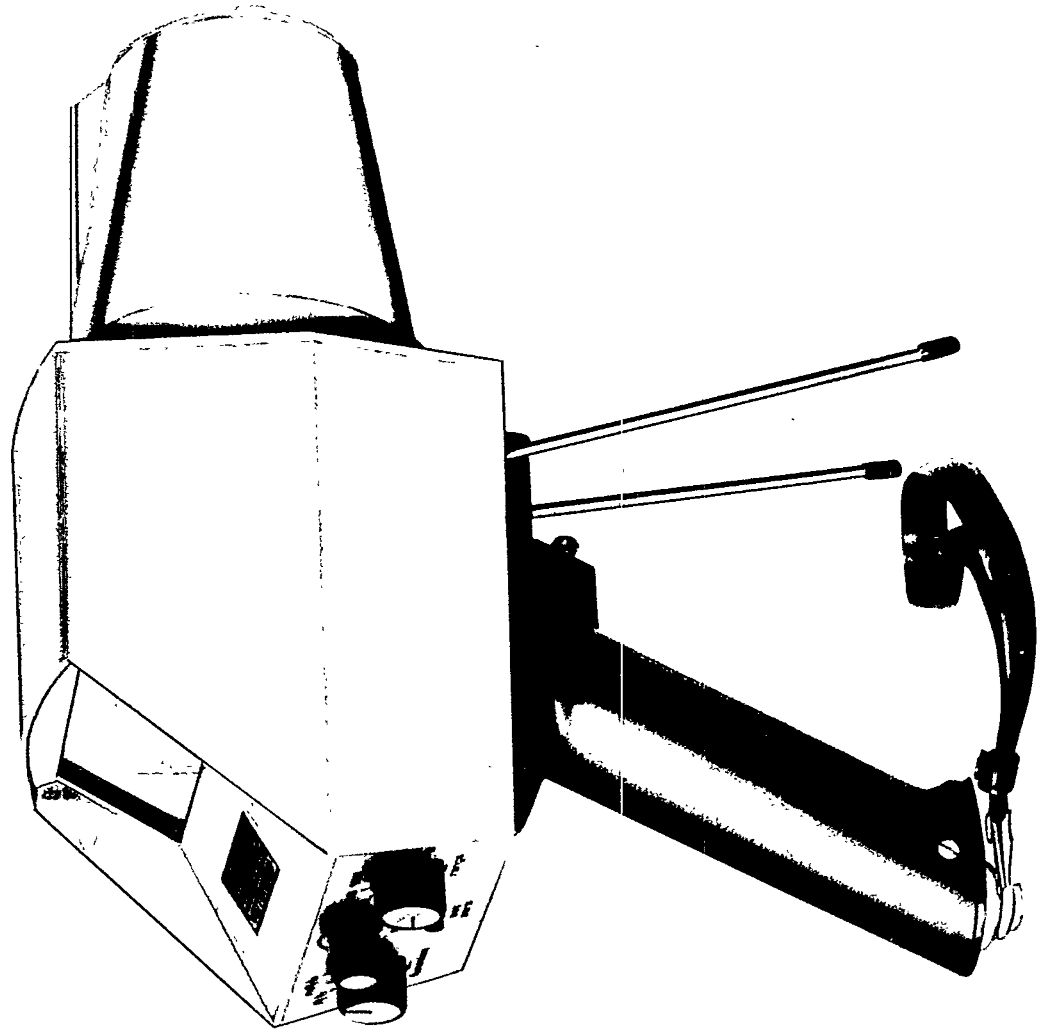


Figure 12.3 Victoreen ion chamber. (Courtesy of Victoreen, Inc.)

dependence is within 10% in the range 40-2000 keV. The reported accuracy of the instrument, independent of the energy dependence, is within 10% of full scale.

The instrument responds to photon radiation over 2 π solid angle. With the end cap removed, the instrument has a directional capability for some radiations. A beta check source (2.4 Bq of ^{238}U) is mounted to the outside of the end cap. This check source produces a field of about 2.58×10^{-7} C/kgh (1 mR/h) when placed on top of the bare chamber.

The center of the chamber volume is about 60 mm from the chamber front face. With the equilibrium sleeve on, the distance from the end cap to the chamber center is about 70 mm. Because of the chamber size, readings up close to a point source do not measure the contact radiation field but an average over the rapid variation with distance of the radiation field.

The instrument is supplied with 12 linear ranges in the exposure rate mode covering 0-0.258 C/kgh (0-1000 R/h), and 6 ranges in the integrate mode covering $0\text{-}2.58 \times 10^{-4}$ C/kg (0-1000 mR). The weight of the instrument including batteries is about 1.8 kg (approximately 4 lbs.).

The Panoramic is designed to measure the exposure or exposure rate and is originally calibrated in a uniform photon field from ^{137}Cs (661.6 keV). It is not calibrated for alpha or beta, so for these radiations the readings are only relative. In addition, using the instrument with the equilibrium sleeve on, for low energy x rays, will also give an erroneous response. Since the chamber and sleeve are made of plastic, the instrument will exhibit response to a neutron field.

b. Bicron Ion Chamber Survey Instrument

The Bicron RSO-500 ion chamber survey instrument (shown in Figure 12.4) is designed to measure the exposure rate of photons. It is also able to detect beta radiation. It features an air-filled chamber which is vented to the atmosphere through a silica gel desiccant. The

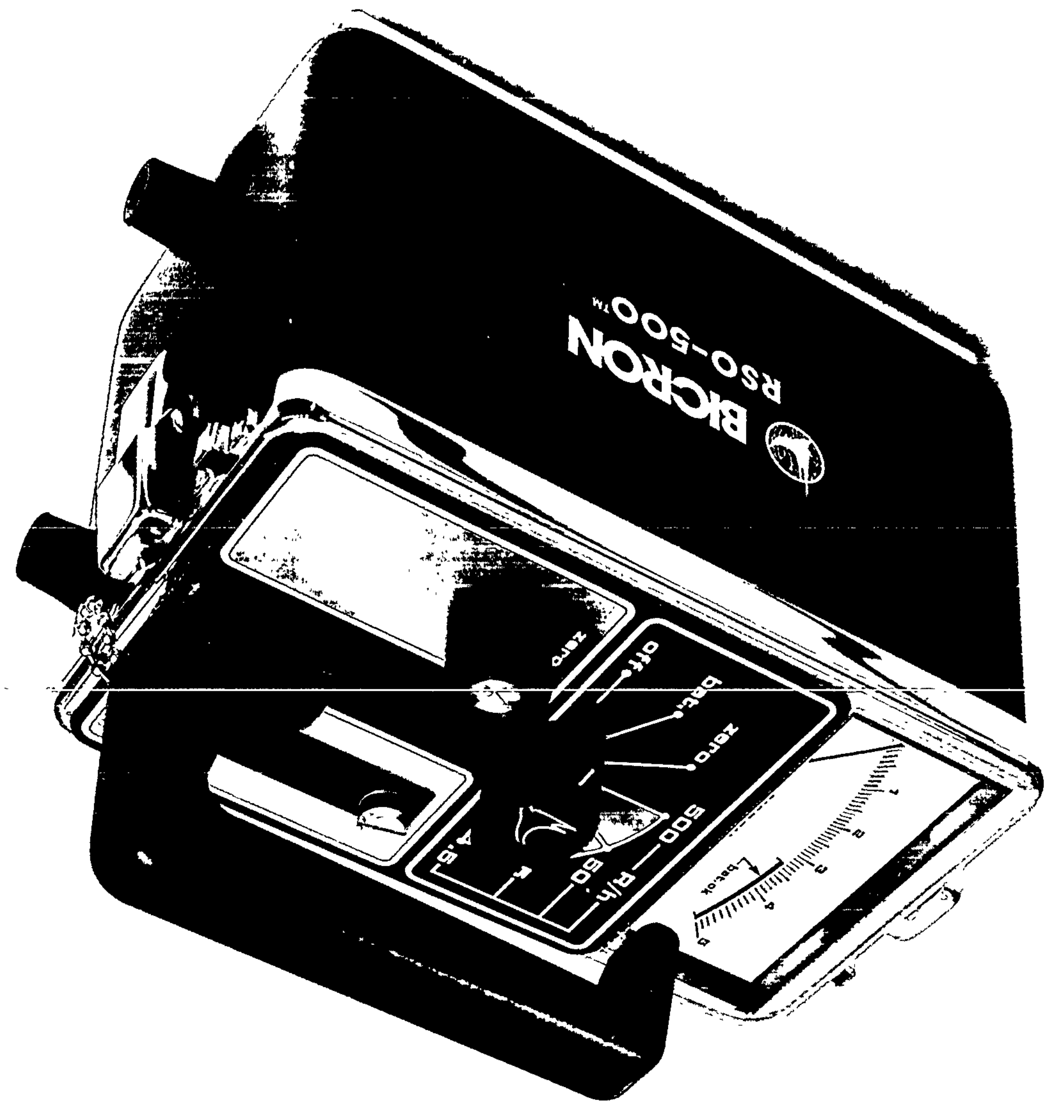


Figure 12.4 Bicron portable ion chamber. (Courtesy of Bicron Corp.)

cylindrical chamber of volume 200 cc is constructed of phenolic with walls of 2 kg/m^2 (200 mg/cm^2) thickness. One end of the chamber has an aluminized mylar window wall, which is coated on the inside to make it a conducting surface. This window is aligned with another external aluminized window, located in the bottom of the case, giving a total window thickness of 0.07 kg/m^2 (7 mg/cm^2).

A sliding beta shield (4 kg/m^2 = 400 mg/cm^2 of phenolic) allows discrimination in the detection of beta in the presence of gamma. With the bottom window open, the stated photon energy dependent response is within 20% from about 15-7000 keV. With the window closed, the quoted response is within 20% from about 30-7000 keV.

The reported accuracy of the reading is within 5% of full scale. The instrument covers the exposure rate range from 0-0.129 C/kgh (0-500 R/h) in 4 linear ranges. The weight of the instrument is 1.6 kg (approximately 3.5 lbs). Photon radiation incident on the front, sides and bottom of the case will be detected. For beta or low energy x rays, the sliding shield must be open and the radiation incident on the exposed window. The scale indication for beta should <u>not</u> be interpreted as exposure rate, it is only useful as a relative indicator of the beta field.

Ion chamber instruments in which the chamber is at the end of a long cable or attached to a telescoping pole are available for remote applications. A probe type ion chamber with a long cable is often located inside a hot cell, while the electronic package is on the outside of the cell. This type of device has been found useful for determining the radiation field from high level waste prior to removal from the cell.

Special design ion chambers for low energy x rays are also available. These are useful for surveys of x ray producing equipment. Other ion chambers have been found useful for beta monitoring under certain conditions.[29]

In addition to the ion chamber instruments discussed above, a number of other companies, such as Keithley, Eberline, Ludlum, Health Physics Instruments and others make ion chamber survey instruments which are presently being used at many facilities.

3. Geiger-Müeller Survey Meters

The Geiger-Müeller survey meter employs a G-M tube and count rate meter to obtain a simple and sensitive detecting device. The electronics for the probe are generally an all purpose package to which a number of different type of probes (G-M, scintillator, proportional, other) may be connected. Geiger tubes are available for the detection of alpha, beta and gamma radiation. Some tubes are contained in a sliding shield which allows discrimination between penetrating and non-penetrating radiation. In these, the entire unit, tube and shield, is the probe. Other probes have thin end windows, made out of mica (.014 kg/m^2 = 1.4 mg/cm^2) which allow detection of alpha (>3.5-4 MeV), beta (>35-70 keV) and gamma (>6 keV). A circular G-M tube, called a "pancake" probe, is a thin window tube of area about $2x10^{-3}$ m^2, generally used for surface or personnel surveys. This can be obtained as a shielded or unshielded probe.

a. Design Features

The G-M tubes in survey devices generally have a cylindrical cathode, which is also part of the counter wall. The cathode may consist of metal, metal sheath, or carbon-coated glass a few centimeters in diameter. The length may vary from 2 to 10 times the diameter. Most wall material is aluminum or stainless steel. The anode is an axial wire (tungsten) of about 0.1 mm in diameter.[30] These tubes are usually operated at reduced gas pressure of about 70 to 200 mm Hg.[20] The gas-fill varies, but some type of self-quenching gas (see Section 11.G) is used for survey-meter tubes. Depending upon the gas used and the tube dimensions, the operating voltage may range from 250 to 1500 volts. Many common types in present use require about 900 volts.

Pulses formed in the counter probe are fed to the count-rate meter circuit. The meter reading then gives the average pulse rate in the G-M tube. With these circuits, the device needs no zero control or warm-up period. In the past, some counters would "saturate" in a high radiation field-approximately 2.58 mC/kgh (~ 10 R/h). That is, the pulse rate becomes so high that the count-rate circuit fails to function properly. As a result, the meter reads near zero rather than off-scale. The condition of saturation can also quickly ruin the tube because of the rapid loss of some types of quenching gas. Most newer counters will peg off scale when saturated in high fields.

The G-M survey meter may be used to detect alpha, beta and gamma, when an end window probe is used. Figure 12.5 shows a typical portable G-M survey meter with an end window probe. The tube wall thickness is about 0.3 kg/m^2 (30 mg/cm^2). This is much too thick to allow alpha penetration, but the device detects beta of $E > 0.2$ MeV and photons of $E > 12$ keV incident on the tube wall. In an end window tube, mica is quite often used as the window substance. Since the window is very fragile (.01-.02 kg/m^2 = 1 to 2 mg/cm^2), care must be taken in using these devices. Sometimes, the least contact with an object will "pop" the window, even though they are supplied with a protective screen.

End window counters are used extensively to detect ^{14}C, which gives off a low-energy beta. We must always be aware of the highly directional beta response for both the end window type and other G-M survey meters. The G-M survey meter only detects beta if the open area in the sliding shield is facing the source. The sliding shield (approximately 15 kg/m^2 = 1500 mg/cm^2) rejects all beta from normal sources.

b. Energy Dependence

The G-M survey meter is not an accurate instrument for exposure-rate measurements. The response of this device is not directly proportional to the energy absorbed in the sensitive volume.[30] The energy absorbed per unit mass in any medium is a function of the photon energy fluence (Ψ) and the mass energy absorption coefficient (μ_{en}/ρ), (see Section 4.4). This is true if the secondaries do not have too high an energy. The count rate for a G-M survey meter depends

Figure 12.5 Ludlum end window Geiger counter. (Courtesy of Ludlum Measurements, Inc.)

upon the efficiency of the counter (counts per incident photon). This, in turn, is a function of the cathode material.[30] Regardless of the cathode substance used, the count rate for a given exposure rate is not constant as photon energy varies. Thus, the response of these instruments in terms of exposure rate is a function of the photon energy.[1] To overcome this energy dependence, the cathode material must have a response nearly proportional to the photon energy. Then, we could calibrate a G-M survey meter to correctly read the exposure rate regardless of photon energy.

No substance has been found that has the proper response for all photon energy. Copper seems to be one of the better cathode substances. It gives a flat response within ± 15% over the range of 0.3-3 MeV.[31] Below about 0.1 MeV, the response of any cathode material is highly energy-dependent, and none of the GM survey meters are suitable for exposure-rate readings.

Shields placed around the GM tube improve the energy response. However, the compensation is affected by the angular response of the detector. Such shielded tubes will be geometry dependent when used close to a source.

Wagner and Hurst[32] reported the use of a halogen-filled G-M tube, which can be used to measure gamma exposure rates. This requires the use of a shield over the tube to correct for the energy dependence. However, the use of energy compensated G-M tubes in survey instruments has not gained wide acceptance as shown by the recent test results in Reference 23. As a consequence, many commercially available G-M counters are still highly energy dependent. Energy-compensated halogen quenched tubes can be obtained from some commercial suppliers.

c. Application

At present, the G-M survey meter is best used as a device to detect radiation. The G-M counter serves as a useful, low-level survey device. Such counters are available from a number of commercial companies.

Usually, these devices are supplied with a nominal scale range of 0-50 μC/kgh (0-approximately 200 mR/h), although higher ranges are available. As stated, these counters are not suitable for radiation field measurements so that scale units should be read in counts per minute, rather than mR/h.

For aid in detection, these devices also provide for the use of headset earphones or have an audio attachment. These should always be used when surveying since their response is much faster than the meter indication. Also, the audible response does not fail even if the device saturates and the meter indicates zero.

4. Proportional Counter Survey Meters

Proportional counters have many design features that differ from G-M counters. The counter gas can be a simple mixture because a quenching agent is not needed.[30] The counters may be operated at atmospheric pressure or slightly above. In many cases, the gas is allowed to flow through the chamber.[20] Methane, ethylene and other hydrocarbons are popular gases for use in flow-type counters. The counter may take the form of a cylindrical tube, a flat plate, or even a sphere. The device may contain a window of nylon, rubber hydrochloride, or aluminized mylar for alpha detection. These counters can be used to detect neutrons if the proper wall substance and gas are used.

The design factors for portable devices are more critical than those for stationary units. Portable counters require special discriminator circuits, very stable high-voltage supplies, and, in some cases, very sensitive amplifiers.

Trouble sometimes arises in counters that use air at atmospheric pressure as the fill gas. Changes in atmospheric pressure can induce variations in the pulse size or calibration of the device. These counters often suffer from humidity effects, which can cause erroneous discharges.

The main advantage from the use of these devices is the ability to discriminate between radiation types. In a mixed field, the pulses formed in the counter gas have many sizes. The size of any pulse depends upon the number of initial ions formed. The circuit can be designed to count only pulses of a certain size or greater. In this way, the proportional counter can measure a type of radiation based on its specific energy loss. This then requires that we operate the counter within a given voltage range to ensure that the right pulses are counted.

a. Air Proportional Alpha Counter

The Eberline PAC (proportional alpha counter) is used to measure alpha activity. The chamber is in the form of a flat, plate-type probe with air as the counter gas. Aluminized mylar 0.009 kg/m^2 (0.9 mg/cm^2) thick, forms the chamber window, which has an area of either .006 or .01 m^2. The probe is connected by cable to the small, lightweight, approximately 1.9 kg (approximately 4¼ lbs) case that houses the count-rate meter and voltage supply. A jack enables earphones to be used or the instrument can be supplied with a built-in audible response or a speaker attachment.

The meter has four scale ranges, allowing measurements up to 5x10^5 cts/min based upon approximately 50% geometry. The device measures <u>only</u> alpha, even in the presence of beta-gamma. Both the discriminator and high-voltage circuits can be adjusted to allow detection of low-energy beta.

In the past, the use of air as the counting gas caused the probe to be sensitive to pressure and humidity, which subjected the scale reading to much fluctuation. Redesign and utilization of modern electronic circuits has apparently overcome these problems.

b. Gas Proportional Alpha Counter

The Eberline Gas PAC employs essentially the same circuits as the Eberline PAC. In the Gas PAC shown in Figure 12.6, the chamber gas

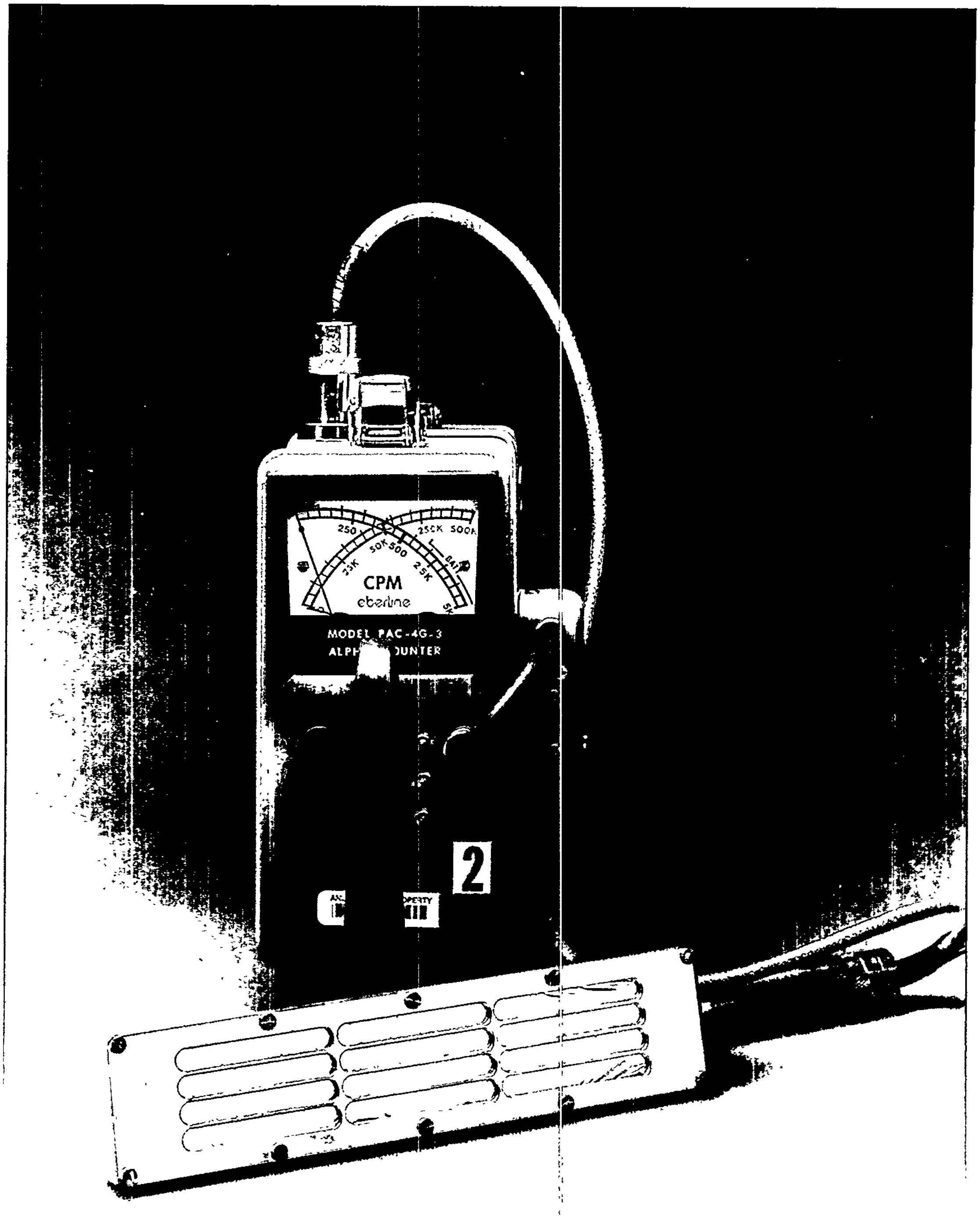


Figure 12.6 Eberline alpha counter with 0.005 m^2 probe.
(Courtesy of Eberline Corp.)

is propane, and the device is used as a gas-flow counter. In this way, any humidity and pressure effects can be overcome, and the probe gives a higher counting efficiency than that for the air probe. Since propane contains hydrogen, the Gas PAC also responds to a fast neutron field because of the proton-recoil pulses.

Propane, in the liquid state, is supplied in bottles, which are housed in a separate case. The case attaches to the bottom of the meter case to make a compact unit weighing about 4.1 kg (approximately 9 lb.) The gas bottle holds about a 24 h supply of gas at normal flow. A three-position gas flow control is provided.

To use this instrument, the flow control is pulled to the flush position. The chamber should be flushed for about 2 minutes. The control is then moved to the operating position, and the device is ready for use. After use, the control is returned to its off position.

The PAC-4G-3 can also be supplied with a .01 m^2 probe (100 cm^2). The instrument has a high voltage switching mechanism so that one can monitor alpha only, or switch to a higher voltage and monitor for alpha, beta and gamma. The counter should be used with earphones while performing surface surveys. The instrument can also be obtained with an internal audible response or a speaker attachment. The linear ranges have been replaced by a lin-log meter which allows readings from 0-$5x10^5$ cts/min, without scale switching. A check source should be used to determine proper operation before the instrument is used for measuring.

c. BF_3 Proportional Neutron Counter

Boron-lined or BF_3 gas proportional counters can be used to measure thermal-neutron fluence rates. These instruments can be designed to be insensitive to high gamma fields (approximately 10^4 photons/n).[33] The capture cross section of thermal neutrons in boron is very high. The capture reaction can be traced mainly to the ^{10}B isotope which is about 20% abundant in natural boron. The reaction is

$$^{10}B + {}^{1}n \rightarrow {}^{7}Li + {}^{4}He$$

and has a cross section of about 3840 b for thermal neutrons. This cross section varies with energy in such a manner that the count rate obtained is proportional to the thermal-neutron fluence rate. The alpha and recoil-lithium nucleus have enough energy to produce about 10^5 ion pairs in the gas. The device is then operated in the proportional region to count only the large pulses produced by the alpha and the lithium.

The boron may be used as a thin lining on the counter wall or as the fill gas, boron trifluoride (BF_3). The efficiency of these devices depends upon the size of the sensitive volume, the number of ^{10}B atoms in the gas or wall, and the capture cross section. The boron-lined counters allow the use of other counting gases, but do exhibit poorer plateau behavior.[1] These devices are often less efficient than the BF_3 counters because of absorption of alpha and lithium in the counter wall.

The counters are most often of cylindrical design in which the diameter, active length, and gas pressure are available in several sizes. The signal from the tube may be fed to a pulse counter or to a count-rate meter. The sensitivity is often expressed in terms of cts/min/unit fluence rate when the device is calibrated in a known thermal ϕ.

d. Long Counter

BF_3 detectors have been used not only to measure thermal neutron fluence rate but also to detect fast neutrons after they have been slowed down by a suitable moderator. Such a device, first developed by Hanson and McKibben,[34] is referred to as a "long counter." In this design (see Figure 12.7), a BF_3 tube is embedded in a cylindrical paraffin holder, called a tub. An incident beam of fast neutrons with a spectrum of energies is slowed down to thermal energy. High-energy neutrons penetrate deeper than the low-energy neutrons before they become thermalized. Since the BF_3 tube is parallel to the incident beam, the

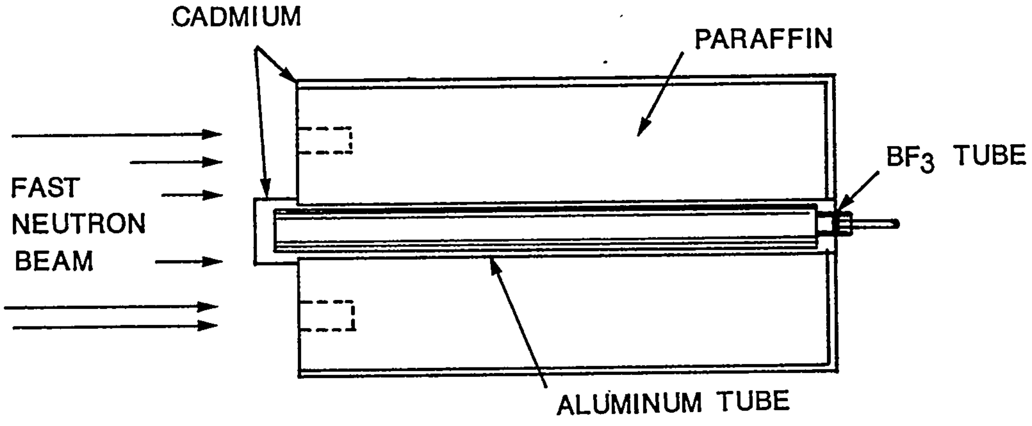


Figure 12.7 Typical setup for long counter applications.

neutron energy should affect only the depth in the counter at which they are detected.[33] In this way, we can detect fast neutrons with a device that is efficient only for thermal neutrons.

The special features of the long counter are found in the paraffin tub. The tub may be a cylinder about 0.3 m long and 0.2 m in diameter. An aluminum tube along the axis of the tub provides electrical shielding for the BF_3 tube. This tube protrudes slightly past the front face of the paraffin and is covered by a cadmium cap. A set of concentric holes (shown as dotted lines in Figure 12.7) increases low-energy neutron response. That is, the paraffin reflects the lower-energy neutrons as they impinge on the face of the tub. By removing some of the paraffin, we decrease the probability of reflection for these neutrons.

The outside wall of the tub is enclosed in a cadmium jacket. This jacket absorbs all neutrons with $E<0.5$ eV to remove all unwanted thermal neutrons. Finally, a brass jacket, which provides mechanical strength, is used as a cover for the cadmium jacket.

DePangher modified the design features of this device to improve the performance. Some details about this device are contained in

Reference 2. The basic principle of the long counter has also been used in a smaller, more portable device, which is commercially available.[35] This instrument is useful for making relative measurements of the neutron field in order to assess potential hazards. Another design, using a cylindrical BF_3 counter surrounded by polyethylene, first suggested in Reference 36, is used as a neutron dose equivalent meter.[36] This instrument has been commercially marketed under the name "Snoopy."

The long counter derives its name from the fact that the response is relatively flat over a wide neutron-energy range (see Figure 12.8). In this curve, the ordinate can be taken as a measure of the relative sensitivity. At thermal energy, the relative sensitivity is about 0.85 of that at 1 MeV.[37] For this reason, the long counter can be used for fluence rate measurements in neutron beams of wide spectrum variations.

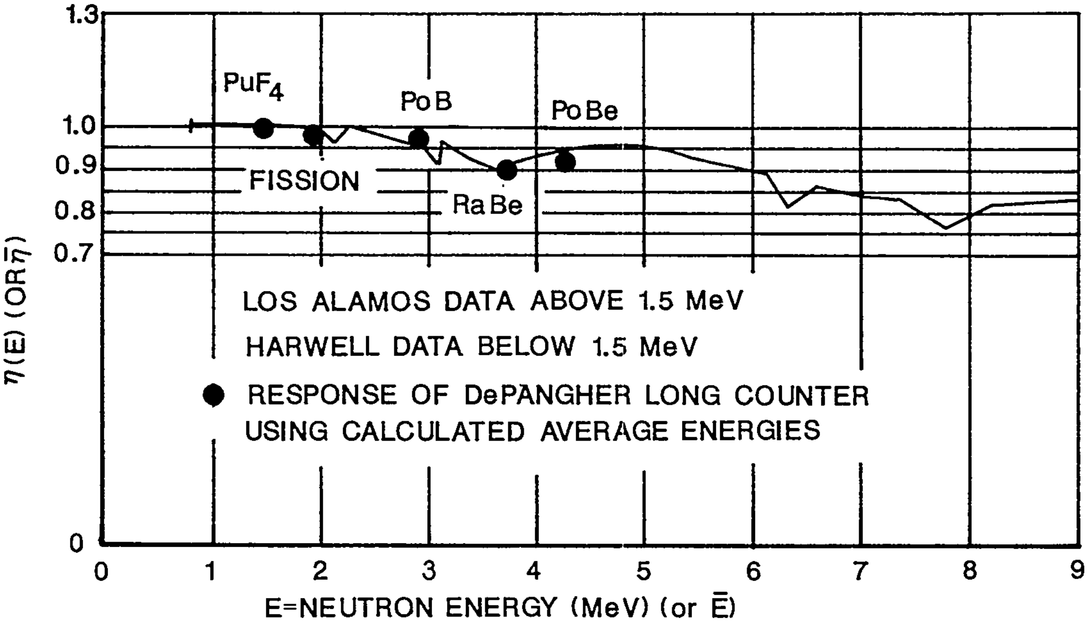


Figure 12.8 Long counter response (from Reference 2).

The long counter must be calibrated for both fast and thermal fluence rate. For thermal neutrons, the BF_3 tube is removed from the paraffin tub and calibrated in a uniform known field. The device is calibrated for fast neutrons with a source facing the front of the tub on the axis of the cylinder. The effective center of the counter depends upon neutron energy. For this reason, the response at a given distance from a source varies with the source energy. We can find the effective center for a given source by observing count rate versus distance.[21] The response is given as cts/min/unit fluence rate when the long counter is calibrated with a known fast neutron source.

Since the response is not isotropic, the long counter must be used with the front face of the tub facing the source. Both air and wall scattering can greatly affect the long counter readings obtained. Although designed to measure only neutrons, the device will respond to high gamma fields (approximately 0.1 Gy/h = 10 rad/h).

The long counter measures only the fluence rate and does so without regard to neutron energy. For this reason, we cannot use the instrument to measure the absorbed dose. To evaluate the dose properly, we would need to know the energy spectrum of the neutron field. If we can estimate the spectrum, then the fluence rate factors in Table 4.3 may be used to estimate the dose equivalent rates. This approach should lead to conservative results.

e. Other Proportional Neutron Survey Meters

Other proportional neutron survey meters have been designed that approximate the absorbed dose[9,11] or dose equivalent.[36,38] The device mentioned in Reference 38 is available commercially, see Figure 12.9, and was adapted from another device using a scintillator in a polyethylene sphere (see Section 12.B.4.c). This device features a 0.23 m (9-inch) diameter polyethylene sphere with a 16 mm by 25 mm BF_3 tube at the sphere center, surrounded by a thin Cd shield. The response closely follows the shape of the theoretical neutron dose equivalent curve from

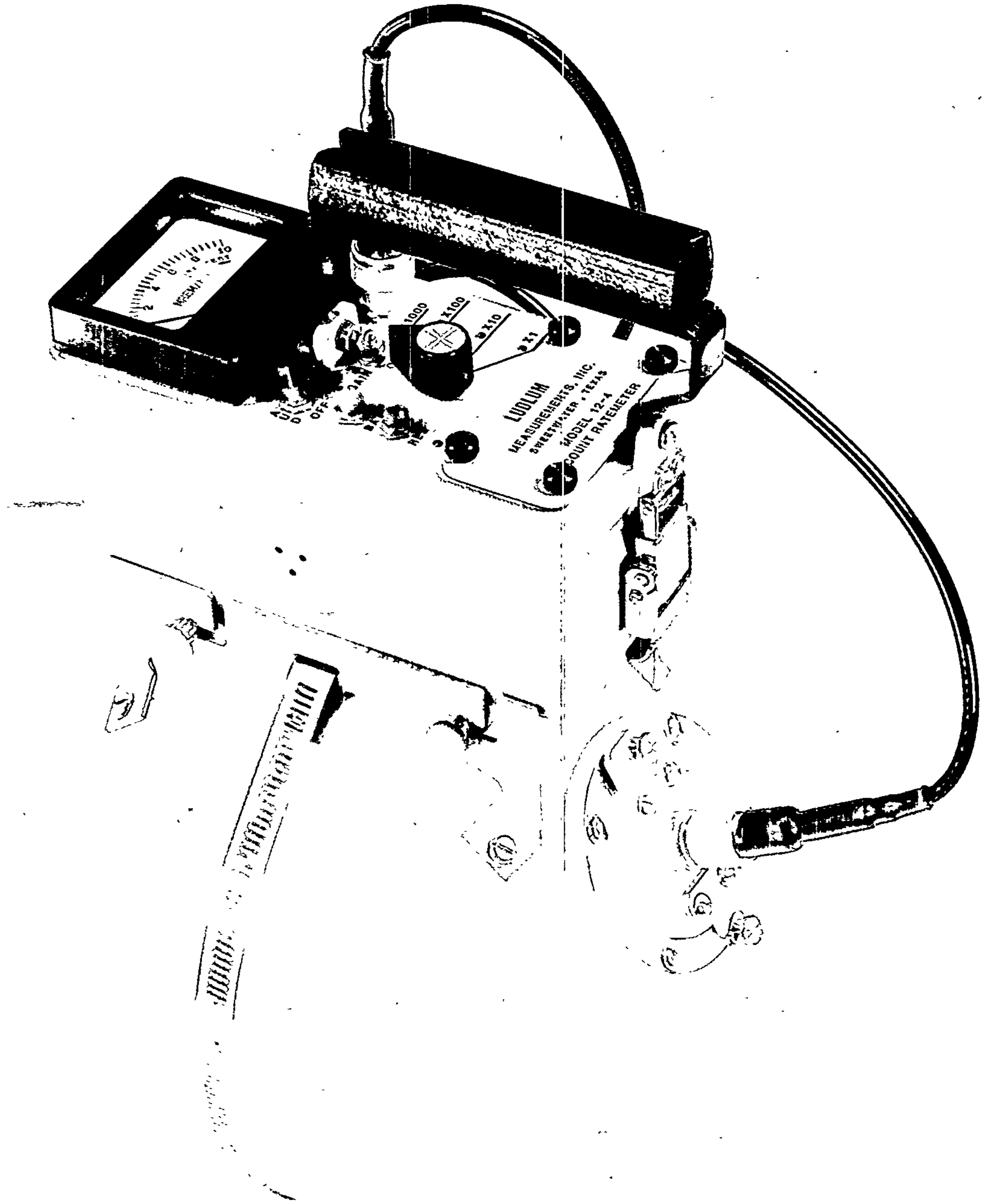


Figure 12.9 Ludlum neutron counter. (Courtesy of Ludlum Measurements, Inc.)

thermal to about 10 MeV. The BF_3 tube also provides excellent gamma discrimination. The instrument is supplied with 4 counting ranges reportedly covering from 0 to 0.1 Sv/h (0-10 rem/h).

5. Scintillation Survey Meters

Scintillation counters are used extensively in many phases of radiation measurements.[20,21] Many devices have been developed to fill a special need in regard to a certain application. Our concern at this time is with the devices that can be used as survey meters. The use of the scintillation method for energy analysis is taken up later in this section (see Section 12.D.2.C).

a. General Features

Although the scintillation method can be used for all types of radiation, portable scintillation counters are most useful for alpha, neutrons, and low-level gamma. These devices will contain a suitable phosphor for the given type of radiation. In most cases, this will be ZnS for alpha and NaI(Tl) for gamma. Several substances[20,39] can be used for neutrons, depending upon the energy spectrum of the radiation field.

The phosphor is optically coupled to a photomultiplier tube, and the entire system can be mounted as a probe unit. In this case, the rest of the circuit is mounted in the instrument case. The circuit consists of an amplifier, a count-rate meter, and a battery supply.

b. Gamma Scintillation Counters

For gamma devices, the instrument is best used simply to detect radiation. Typically, two types of design are used. One, called a micro R (μR) meter, utilizes a 25 mm diameter by 25 mm high NaI crystal contained in the instrument case. The response of these counters to gamma is such that maximum readings are limited to about 1.29 μC/kgh (5 mR/h). The energy dependence of these devices precludes their general use as exposure rate meters. Because of their sensitivity, these units have found widespread use. Applications such as trash surveys, ambient background determinations and elevated environmental levels all lend themselves to determinations by micro-R meters. At the levels for which these instruments are most useful, the absolute accuracy of the scale reading is generally irrelevant.

The second type utilizes a very thin (approximately 1-2 mm thick) NaI crystal, of diameter varying from 25 to 180 mm in a probe unit. Because of the thin crystal, the efficiency for high energy photons is poor. So, this survey instrument is useful in situations requiring a sensitive instrument for low levels of low energy photons. The Eberline PRM-5-3 is one of the commercially available instruments of this type. A pulse height analysis capability is included to respond to only certain calibrated energies of photon emitters. Because of its sensitivity, this instrument is also a valuable tool in locating contamination and discovering activity levels.

c. Alpha Scintillation Counters

In the alpha scintillation counters, the ZnS can be found as a coating on the photomultiplier tube itself or on Lucite that serves as a light pipe. To protect the ZnS coating from light, the device is supplied with an alpha window. This may be a mylar film, which has been aluminized on both sides. The thickness may be as much as 0.015 kg/m^2. The active probe area of these instruments (approximately .006 m^2) tends to be non-uniform in response near the edges. This may be due to poor light transfer by the light pipe. Also, one must protect the light-tight screen from damage, since even a pinhole admits enough light to give an erroneous reading.

The meter gives a reading in cts/min and may provide a choice of three or four ranges (up to 2x10^6 cts/min). Some of these devices tend to saturate in a high alpha field. This occurs when the photomultiplier can no longer resolve the pulses. The instrument should include an earphone jack to provide an aural indication, or an internal audible response or external speakers.

d. Neutron Scintillation Counter

Many substances have been employed to measure neutron fields with scintillation counters. Some of these have been modified so that the response to fast neutrons approximates the tissue-absorbed dose. These methods fall roughly into two main types for fast neutrons: (1) The scintillator is used in conjunction with an hydrogenous substance and detects the proton recoils;[11] or (2) the neutrons are slowed down and then detected with a thermal neutron scintillator. The latter method has been quite useful when the intermediate-energy neutrons may be important.[40] The former method has been used mostly for fast neutron measurements.

One example of this type of device grew out of studies directed toward neutron-spectrum measurements.[41] The phosphor was a small ^{6}LiI(Eu) crystal, which makes use of the ^{6}Li (n,α) ^{3}H reaction. This process has a high capture cross section for thermal neutrons (approximately 1000 b), and the alpha and the recoil triton lead to a large light pulse. To use this crystal to detect fast neutrons, a moderator must be employed.

As a beam of fast neutrons moves into the moderator substance, the elastic collisions that take place reduce the energy of the neutrons. If enough elastic interactions occur, or in other words, if enough H is present, elastic collisions result in a large ΔE, and the energy of the neutrons are quickly reduced, leading to a distribution of thermal neutrons shown as E_1 in Figure 12.10. The location of the maximum of the distribution will be dependent upon the energy of the incident neutron. If one places a phosphor in the moderator at some depth which corresponds to the maximum thermal neutron distribution from E_1, the

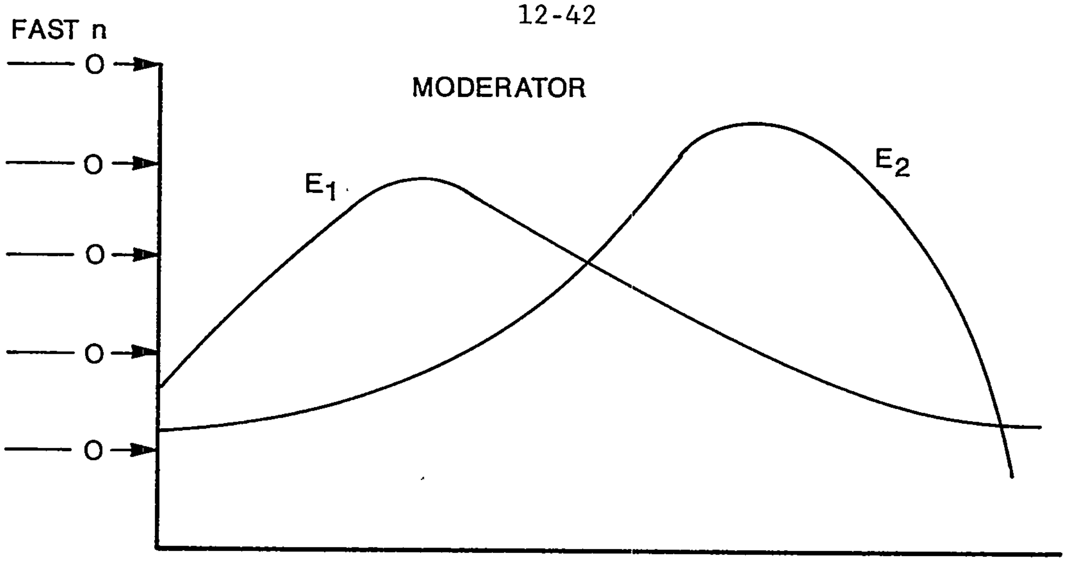


Figure 12.10 Variation in detector response for different initial neutron energy.

response of the phosphor will be less for the case of initial neutron energy, E_2. Also, the response will vary with the location of the phosphor in the moderator. If one uses spherical geometry, the response should be relatively non-directional.

In the original study, the crystal was mounted at the center of several cadmium covered polyethylene spheres. The instrument was similar in appearance to the one shown in Figure 12.9. The light produced in the phosphor was then guided by a lucite light pipe to the photomultiplier tube. The output of the tube was recorded on a scaler.

The spheres varied in diameter from 0.05-0.3 m (2-12 inches). The response of a given diameter sphere was studied as the neutron energy was varied. The response curve for the 0.3 m-diameter sphere had about the same shape as the dose equivalent curve (see Figure 12.12 later). This aspect was investigated for a 0.25 m-diameter sphere. The response of this device approximated the dose-equivalent curve in the energy range from thermal to 7 MeV.[42] In this device, the scaler was replaced by a count-rate meter. The method has seen adoption in the use of several different diameter spheres (called Bonner spheres) to utilize the different response of any one sphere to an incident neutron energy.[1] By using several spheres(5-7), of varying diameter, a crude neutron spectrum may be obtained, as well as fluence and dose equivalent information.

As was mentioned in 12.B.4.e above, this method has been adapted to take advantage of the relative response of the 0.23 m diameter sphere. The response of this device approximates the dose equivalent curve over a wide energy range, but over-responds in the region of intermediate neutrons.

In addition, it is possible to use the ratio of the response of two different diameter-spheres to gain some information about the neutron spectrum in a given location.[43]

6. Activation Unit Survey Meters

At present, only a few devices that use the activation principle may be classed strictly as survey instruments. In this type, the substance to be activated is usually combined with some means to detect the activation products. Then, we can obtain a reading of the field based upon the response of the entire unit. To be of value, these applications require a high fluence rate. The more common type uses the substance to detect the radiation field, and then the activation products are counted by some other means. In this way, the activation detectors can be used for pulsed sources as well as for steady-state fields. In this method, the device departs from the concept of the survey meter.

Two common approaches use the latter method: (1) Threshold detectors can be used in conjunction with some other substance or as a combined unit.[11,28,31] We can then measure the total fluence above the threshold energy or the fluence between any two thresholds. (2) Thermal-activation detectors can be used to detect thermal fluence or placed in a moderator to measure thermalized fast fluence.[28]

a. High Energy Neutron Monitor

The threshold detector technique has been used in a device to monitor high energy neutrons and protons.[44] In this method, a cylinder of Pilot-B plastic scintillator of 0.13 m diameter and 0.13 m height is used. The $^{12}C(n,2n)^{11}C$ and $^{12}C(p,pn)^{11}C$ reactions in the carbon of the plastic lead to the 1 MeV positron emitter. Carbon-11 has a half life of 20.5 minutes. The absorption of the positron and most of the annihilation gamma energy leads to light pulses. These light pulses can then be counted on a scintillation spectrometer. This method is useful in the energy range $20 < E < 400$ MeV.

The scintillator sample is exposed in the radiation area for a known time and is then brought back to the counting system. Such a system is generally a multichannel analyzer. In this case, we obtain a measure of the fluence above the threshold of 20 MeV. The device can be used with other total-fluence devices to give an estimate of the fractional fluence above 20 MeV in certain energy ranges.

b. Spherical Neutron Foil Monitor

A device that uses thermal-activation foils in a spherical paraffin moderator has been developed at ANL.[45] This spherical neutron foil monitor is shown in Figure 12.11. The diameter of the sphere is 0.3 m and a symmetrical array of three sensors is placed at the center. Six other sensors (0.05 m in diameter) are located on the rectangular coordinate axes at the same distance from the center. In this design, the foils are 0.03 m below the moderator surface and 0.13 m from the sphere

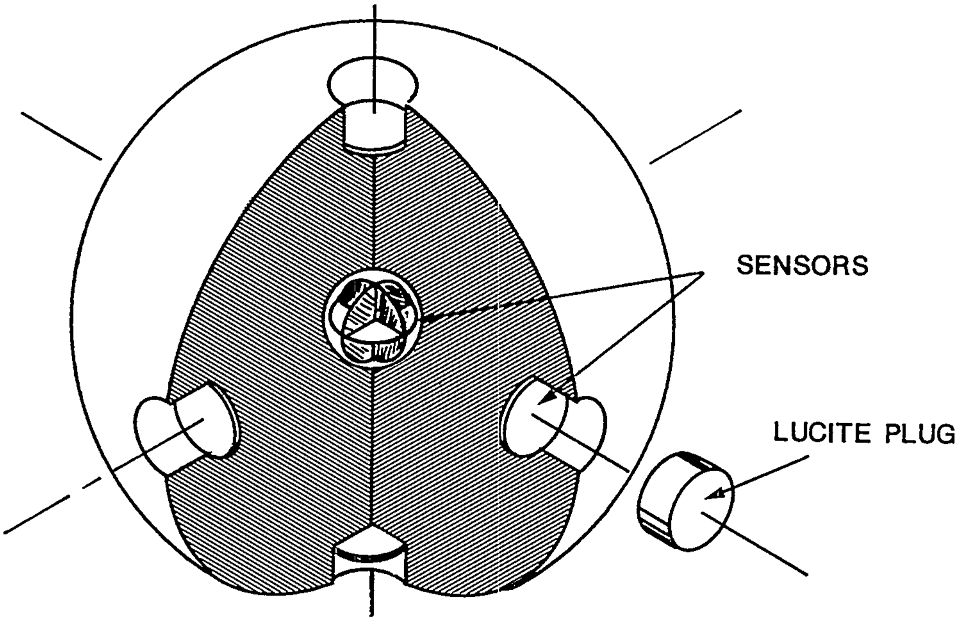


Figure 12.11 Sensor geometry for the Spherical Neutron Foil Monitor (from ANL-7085[45]).

center. The actual detector is composed of two 0.3 m diameter, hemispherical aluminum shells, which are filled with paraffin. This allows easy access to the central foil array. The other six foils are placed in lucite holders located at the proper positions in the sphere. Lucite plugs hold the foils in place.

For a large enough sphere, the response of the symmetrically placed outside sensors is fairly independent of the energy of the incident neutrons.[45] The outside array can thus be used as a flat-response counter to measure fluence. If we use the sum of the foil counts to measure the activation, then the response should also be relatively nondirectional. In actual tests, the response was flat within 10% and nondirectional within 5%.

The central sensor provides a response that approximates the shape of the dose-equivalent curve. In this way, it somewhat resembles the device described in Section 12.B.5.D. In the foil monitor, the foil array replaces the ^{6}LiI(Eu) scintillator as the thermal fluence detector. The response of the ANL device is compared to the dose-equivalent rate curve in Figure 12.12. Also shown are the response curves for the other spherical detectors of Bramblett et al[41] and Hankins.[42]

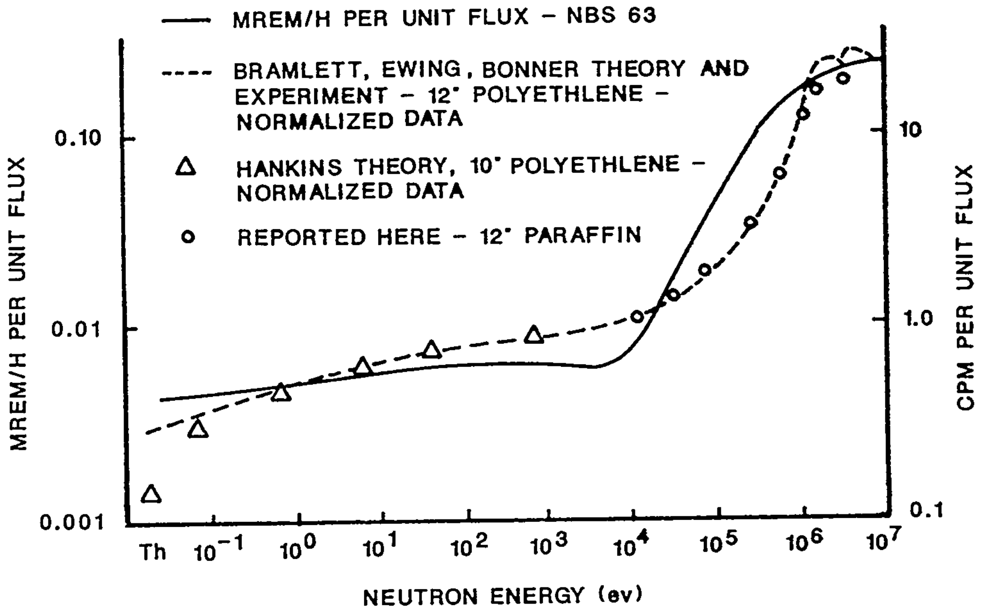


Figure 12.12 Response curves for three spherical neutron detectors compared to the dose equivalent rate function (from ANL-7085[45]).

The central sensor should also be nondirectional in response, but a 10% directional effect was noted.[45] The central array can still be used to measure an approximate dose equivalent.

Now, since we have two detectors, whose energy response differs in a given field, we can find an effective energy. We can plot the ratio of the summed activity for the outer foils to that for the inner foils as a function of the known neutron energy. The resulting curve can then be used to find the effective energy when the foil monitor is used in an unknown field.

This device is used to obtain a simultaneous measure of the fluence, dose-equivalent, and effective neutron energy. The unit may be used to measure fields as low as a few tens of μSv/h (a few mrem/h). However, since the foils must be removed and counted, the results of a survey are delayed. The technique has been used in remote monitors by using small, silver-wrapped G-M counters in place of the foils.

C. Microprocessor Applications

With the development of microprocessors, the capabilities of the readout instrumentation for portable instruments have been greatly increased without significant increase in size and cost (see Figure 12.13). Microcomputers can be incorporated into the electronics package and these allow several pre-programmed functions to be selected from a menu. Selection is performed from the seven button keypad.

The Eberline unit in the figure is designed to operate with several different probes - G-M, scintillation and proportional - so it can be used to detect alpha,beta, x, gamma and neutron. The readout can be either as a ratemeter or a scaler. Supplied with an 8 kilobyte memory, about 500 data points can be logged in the memory for either transfer to a personal computer or to a printer to produce a hard copy. In this way, the survey results for an entire large room area can be recorded and maintained during the survey. An additional feature, called the "peak trap" allows one to store only the maximum reading in a series of readings taken in a

Figure 12.13 Eberline microprocessor readout. (Courtesy of Eberline Instrument Corp.)

given area. Readings are corrected for coincidence loss in the case of high count rates. A single channel pulse height analysis option is also available.

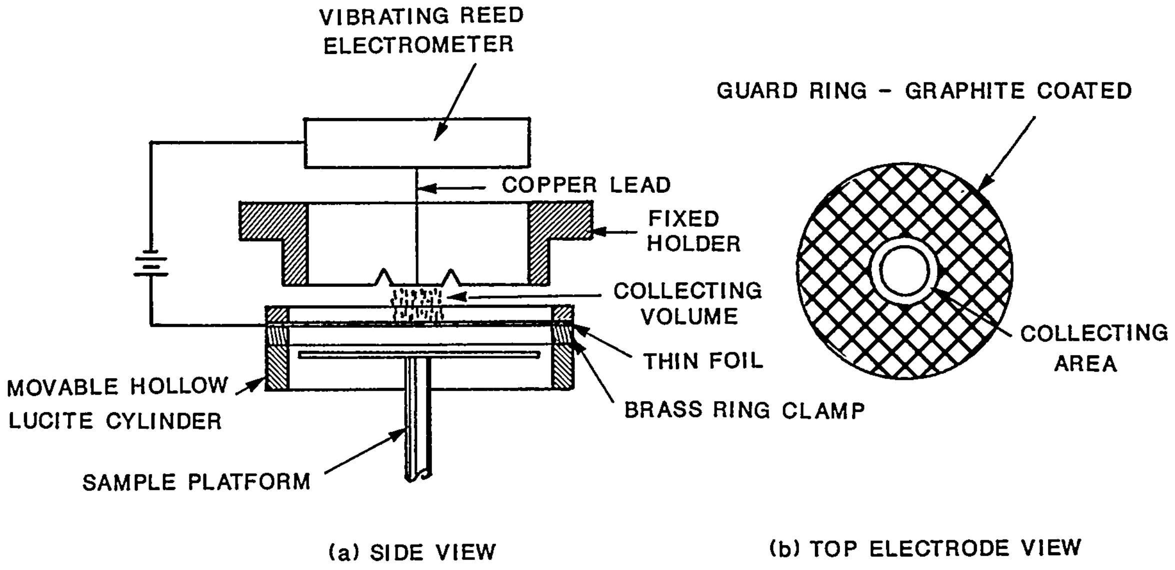
The entire package size is .13 m by .13 m by .26 m (5-in. by 5-in by 10¼-in.), and weighs 1.75 kg (3.8 lbs). The instrument is supplied with an internal speaker which can be set to alarm at a certain level. An audio headset can also be used.

D. Special Purpose Instruments

Along with survey instruments, some special devices are also useful in a radiation safety program. These units provide a complement to the survey devices and in some cases may supply information that survey instruments cannot supply. We will treat only a few of these special-purpose units, which are of use at the present time.

1. Extrapolation Chamber

The extrapolation chamber was devised by Failla[46] to study the effect of cavity size in Bragg-Gray chambers. In essence, the device is a parallel-plate ion chamber in which the plate spacing may be varied (see Figure 12.14). A thin foil (doubly aluminized mylar) serves as the accelerating electrode of the chamber. This foil is clamped between two brass rings, which are attached to a hollow lucite cylinder. We can move this entire unit up or down. The other electrode is formed by coating the surface of a cylindrical lucite block with Aquadag. A thin copper lead passes through the lucite and makes contact with the graphite surface. This electrode is divided into a small collecting area and a large guard-ring area (see Figure 12.14b). A source is placed on a platform (see Figure 12.14a) and moved up to the foil. The entire unit may then be moved up close to the other electrode. We measure the current produced in the small collecting volume (see Figure 12.14a) as a function of the plate spacing. The currents are very small (approximately 10^{-15} A) so that the current output is read on a vibrating-reed electrometer.



(a) SIDE VIEW (b) TOP ELECTRODE VIEW

Figure 12.14 Extrapolation chamber for surface dose rate measurements.

The current may then be plotted as a function of the plate spacing (see Figure 12.15). When the plate spacing is very small, the curve approaches a straight line. The current that is collected is proportional to the rate of energy absorption in the chamber volume. Thus, the slope of the straight line in Figure 12.15 is proportional to the absorbed dose rate. The value of the proportionality constant depends upon the design features of the device.

The chamber has been most useful for measuring surface dose rates of beta and soft x rays (< 30 keV). For beta, lucite or acrylic provides a good tissue-like scattering pattern. In this way, we can measure an approximate skin dose rate. Surface dose rate can be measured if the source dimensions are larger than the collecting area of the plate.

The unit described above is a stationary unit. The PTW extrapolation chamber is a portable unit which can be used for field measurements. The chamber is shaped like a flat cylinder. One plate is the entrance foil, a graphite coated film (.026 kg/m^2 thick), while the other, made of acrylic, is part of a micrometer mechanism which allows adjustment of the volume. The chamber housing is .14 m in diameter, while the diameter of the air chamber is .06 m, and the spacing of the plates is variable from 0 to 10 mm.

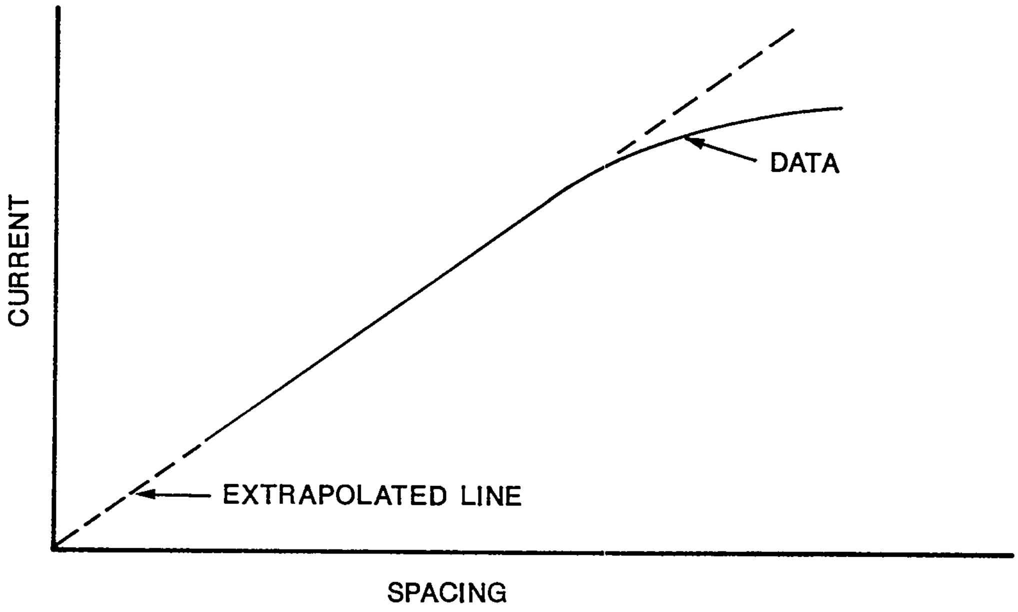


Figure 12.15 Plot of extrapolation chamber data.

Portable units can be used to investigate situations which are representative of the workplace.[29] This allows testing of instrument response to different beta spectra in order to evaluate the instrument performance for beta dosimetry.

More information concerning extrapolation chamber principles can be found in Reference 47.

2. Spectrometry Units

In most cases, we can identify a radionuclide if given both the half life and the energy of the emissions. For this reason, the role of the spectrometer is important in health physics work. Such a device can incorporate the means to measure both the half life and spectrum, if the conditions are right. Also, quantitative results can be obtained for the activity of the sources in certain cases.

We can use any number of methods to detect the radiation. As long as the output pulse is proportional to the energy lost in the detector, the method is useful for spectrum analysis. The energy analysis is brought about by use of a pulse height analyzer (see Section 11.C.2).

We may have either a single channel or multichannel unit. In either unit, the pulses are sorted with regard to size, and only those pulses whose sizes fall in the window or channel are recorded. In the single channel (SCA) device, we must adjust the channel level over the range of pulse sizes to record the entire pulse size distribution. This is accomplished by varying the window level (Lower Level Discriminator) over the range of the potentiometer. On the other hand, in a multichannel analyzer (MCA), we can record counts simultaneously in all the channels. In this way, we obtain the entire spectrum much faster. In either case, a scaler, a count rate meter, and a display device (cathode ray tube or liquid crystal display) may be included.

For low counting rates, the statistics of a single channel unit are poor unless one uses longer counting times for each interval. For this reason, single channel analyzers are of most value when one is looking for a certain radionuclide. By setting up the analyzer for the energy of that radionuclide, one can decrease the amount of counting time needed for proper identification. However, one may miss other radionuclides which do not give off radiations of that energy. In addition, the interactions in the detector of a higher energy radiation may produce pulses in the channel of the radionuclide of interest. If enough of these pulses are present, one may incorrectly conclude the presence of the radionuclide of interest. One method of avoiding the latter error is to supply 2 channels. One for the radionuclide of interest and one with the same channel width but set to respond at a higher energy. One can then use a suitable higher energy radionuclide to produce counts in each of these channels. The ratio of the counts in these two channels will then give a correction factor, which can be used to adjust the counts in the channel of interest. For example, suppose 1000 cts are obtained in the higher energy channel and only 500 in the channel of interest. The ratio would be 500/1000 = 0.5. For an actual count then, we would correct the count in the channel of interest by subtracting 0.5 times the count in the higher energy channel from the count in the channel of interest.

The multichannel analyzer allows display of the entire spectrum as it is being collected, so that many radionuclides can be seen at the same time. Originally very bulky because of the individual components, present multichannel units are all contained in compact units which utilize modern electronic circuits. General characteristics of MCAs are discussed in Reference 1.

A spectrometer may consist of the detecting system with its power supply, a linear amplifier and pulse shaper, a single or multichannel pulse height analyzer, and the display system, all contained in a relatively small package. The extent of the information we can obtain from these units depends upon the design features of the device. Common sizes for MCA units are 4096 or 8192 channels.

The system has three functional segments:[48] an analog-to-digital conversion, which converts pulse amplitudes to a digital number, a memory with a number of registers to add up the number of pulses in each channel, and an input/output section, which allows display or printout of the data.

The choice of a certain detector system determines the type of radiation to be handled. That is, we can choose a system that responds only to neutrons, or only to beta, etc. In this respect, then, we can construct any number of spectrometers. The following paragraphs describe some of these systems and their outputs.

a. <u>LET Spectrometer</u>

The LET spectrometer employs a .05 m or 0.2 m-diameter tissue equivalent chamber filled with tissue equivalent gas. The output is fed to a multichannel pulse height analyzer. This device is based upon the unit described by Rossi and Rosenzweig.[49] The counter is operated in the proportional region at low gas pressure. The entire system consists of two mobile units: one a gas-pumping station, the other the electronic system.

Since the energy lost by the particles depends on the LET and track length,[19] a distribution of pulse heights occurs. The approach assumes that the particles travel along straight paths through the gas. This will be valid for those charged particles with Q>1 (mainly positively charged particles).[2] The gas pressure in the chamber is kept low (10-30 mm of Hg) since the method assumes the particles lose only a small amount of energy in crossing the sphere. That is, the LET of the particles that traverse the cavity must remain about the same. A typical spectrum is shown in Figure 12.16. Assume a given beam of fast neutrons impinges on the chamber wall (approximately 6 mm thick); the result is a spectrum of recoil protons formed in the wall. As the protons traverse the cavity, some must travel a greater distance than others. This leads to different energy losses in the gas and thus to the spectrum of pulse heights.

To convert the pulse height curve above into the distribution in LET of dose (see Section 12.A.3.b) requires a mathematical analysis that accounts for the difference in path lengths.[19] This

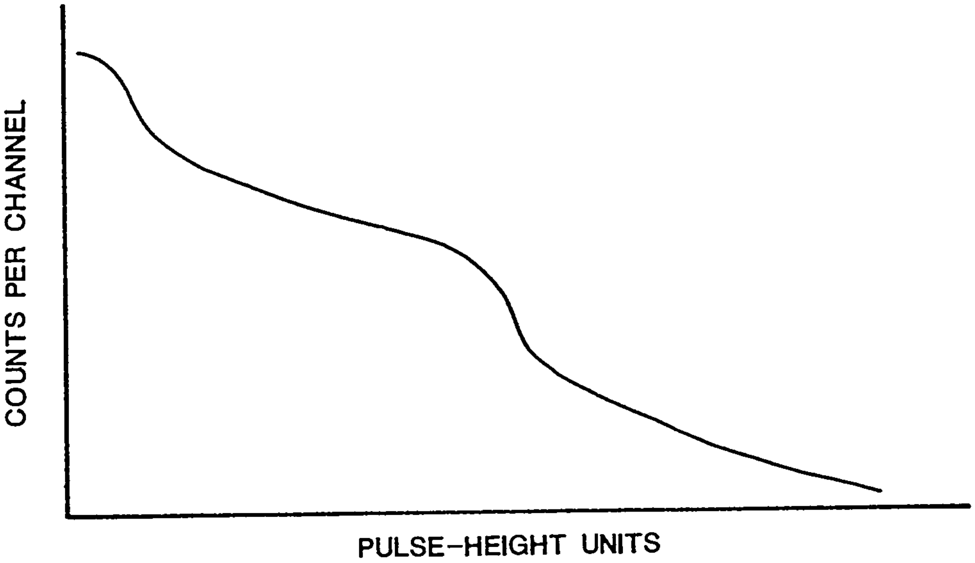


Figure 12.16 LET spectrometer pulse height distribution.

analysis can be set up as a computer program, which performs the needed data reduction.

b. Surface-Barrier Alpha Spectrometer

Alpha spectrometry is generally accomplished with the use of silicon surface-barrier detectors (p-n junctions). Surface barrier detectors, which required vacuum operation and could not be cleaned without damaging the detector, are giving way to silicon detectors which can be operated outside of a vacuum. Silicon has become the primary diode used for particle spectrometry.[1] These detectors are available generally in the range from 100-900 mm^2 of active area, although one can obtain up to 2000 m^2.[1,48] Depletion depths of ≤ 1 mm are common, although up to 5 mm are obtainable.

For the units requiring vacuum operation, the detector is mounted in a metal box, into which the sample is also placed. The box is closed, and the unit is operated under high vacuum. The pulse signals from the detector are amplified and sent to a multichannel analyzer. The vacuum system improves the shape of the pulse height curve. However, the sample must be plated on a thin planchet or deposited on a thin film to decrease energy absorption.

The pulse height spectrum obtained when a thin alpha source is measured in a vacuum is shown as the solid line in Figure 12.17. We obtain very good resolution of the pulse sizes since little energy is lost by absorption in the source or along the path. The <u>dotted</u> portion of the figure is obtained if some of the alphas become degraded in energy. In this case, the pulse sizes are smeared out because most of the alphas no longer have about the same energy. This type of spectrum results when the alpha emitter is embedded in the sample or if absorber material is in the intervening space. The degradation which occurs when the vacuum is inadequate can lead to complete loss of the full energy peak.

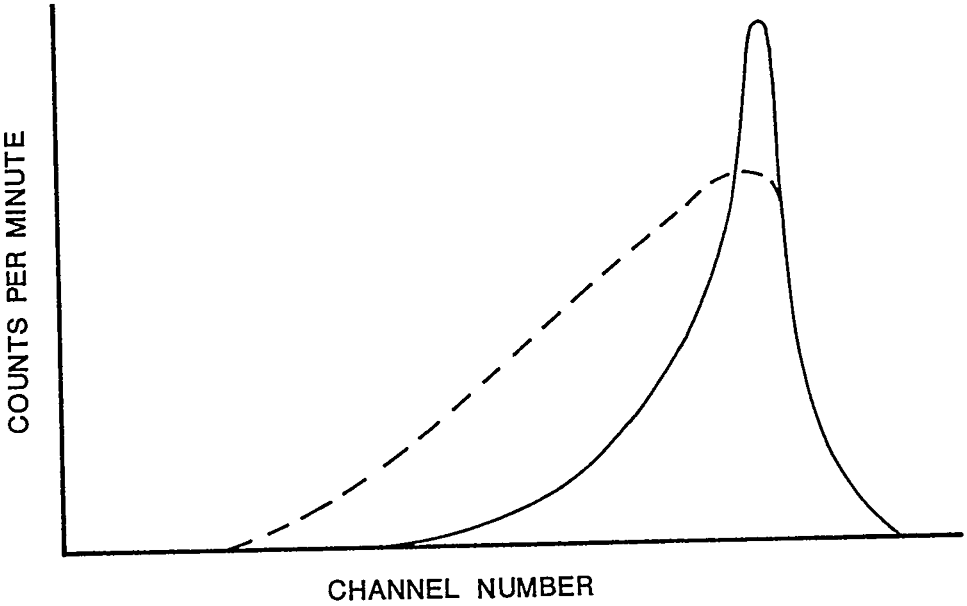


Figure 12.17 Alpha pulse height spectrum in a silicon surface barrier detector.

c. Gamma Ray Spectrometers

Two methods are popular for gamma spectroscopy. These are the scintillation method and the semiconductor method. In the former method, the systems are less costly but the resolution of the spectrum is poor. In the latter, high resolution is obtained but the systems are relatively expensive.

(1) Scintillation Method

The most common phosphor used in this method is a NaI crystal, with a small (approximately 0.1%) thallium impurity added. This substances gives off large light pulses and has a high density (ρ = $3.67\text{x}10^3$ kg/m^3), and large crystals can be grown. Since the iodine has a high atomic number (Z=53), the phosphor absorbs gamma readily. The NaI crystals, most often in cylindrical form, come in many sizes. The size is often stated in terms of the diameter and height of the crystal. Thus, a 0.01x0.02 m crystal has a diameter of 0.01 m and height of 0.02 m. The phosphors are hygroscopic (pick up water vapor from the air) and must be sealed in a tight container (aluminum). The crystal and the photomultiplier tube are then supplied as one unit already optically coupled. The high-voltage supply must give a stable output in these units since small voltage changes can result in large changes in photomultiplier gain.

The output from the photomultiplier tube is sent through a linear amplifier to the pulse height analyzer, either a single channel or multichannel device. The rest of the unit then contains the readout system, which may consist of a scaler, count rate meter, and display system.

In order to increase the detector sensitivity, the crystal is usually mounted inside of a lead (0.1 m thick) cave. This helps to reduce the interference from background.

(a) Interactions in the Crystal - As photons pass through a crystal, they may interact with the substance in essentially three ways: the photoelectric effect, Compton effect, and/or pair production (see Section 3.E.2, 3 and 4). When the photoelectric effect occurs, the gamma energy is completely absorbed. The electron ejected from the atom has a certain energy, equal to the gamma energy minus the binding energy that held the electron in the atom. Since the electron leaves the atom, a vacancy results in one of the atom shells. The outer electrons fill this vacancy, but the characteristic x rays and Auger electrons are then emitted. Electrons, and the x rays, are both easily absorbed in the crystal, so that this effect usually results in the absorption of all the gamma energy.

In the Compton effect, the photon strikes one of the more loosely bound electrons. In this, a gamma is scattered at some angle while the electron recoils at some other angle. The electron receives some energy from the photon in this process. Since a gamma is scattered, though, the initial photon loses only part of its energy. For this process, the electron is absorbed by the crystal. The scattered gamma may just move through the phosphor without interacting again. It may also lose the rest of its energy in the crystal as a result of more interactions taking place.

For pair production to occur, the photon must have at least 1.022 MeV. In this process, the photon interacts in the vicinity of the nucleus, losing all its energy. An electron-positron pair is created in the process, and these share the excess gamma energy. That is, any energy above the 1.022 MeV needed to form the pair appears as kinetic energy of the pair. Both members are quickly stopped in the crystal, and the positron combines with an electron to form two 0.511 MeV gamma rays. These gammas then go off in opposite directions and may or may not interact in the phosphor. In this case, the original gamma is absorbed, but when the annihilation process occurs, some of the gamma energy may escape from the crystal.

In an NaI(Tl) crystal, the photoelectric effect is more likely to occur for photon energy less than about 250 keV.[51] The Compton effect is predominant from about 300 keV to 5 MeV. Above 5 MeV, the pair-production effect is the most important process. For a given photon source and NaI(Tl) crystal, we can obtain light pulses that result from a number of possible sequences. Thus, some photons are completely absorbed, and some transfer only part of their energy. Also, some photons pass through without interacting at all. The spectrum which results is produced by interactions which result in energy which is absorbed in the crystal.

(b) Spectrum Peaks - In a dense substance, the electrons are stopped quickly so that few escape from a NaI(Tl) crystal. Since x rays and Auger electrons are low in energy, they also are quickly absorbed. The photoelectric effect generally leads to the total photon energy exciting the crystal. Thus, the largest light pulse and the largest voltage pulse result. The pulse height peak that results from the largest light pulse is called the photopeak.[50,51] The pulse-height value of the photopeak can then be related to the gamma energy, E. Sometimes, a K_x ray from iodine may escape from the crystal. If this occurs, we obtain an escape peak at a pulse height value given by $E_\gamma - K_x = 28$ keV for iodine. This effect is not too likely unless the gamma energy is low and the process occurs near the crystal surface.

When the Compton effect occurs, the energy that excites the crystal varies from zero to some maximum value.[51] This value is called the Compton edge, E_C, and depends upon the energy of the incoming photon, E_γ. We can compute E_C from the relationship (equation 3.37),

$$E_C = \frac{E_\gamma^2}{E_\gamma + 0.256}$$

where E_γ is in MeV. For most of the common gamma sources, the most important interaction will be Compton scattering. This leads to a broad distribution of pulse sizes from zero up to E_C, which appears, more or less, as a plateau region rather than peak. The spectrum at E_C usually appears as a gradual decrease in the distribution rather than a sharp cut-off. As the crystal size is increased, the probability of capturing the Compton photons leading to total absorption is enhanced. This results in more full-energy pulses (photopeaks) and a reduced Compton distribution.[48]

The kinetic energy of the electron-positron pair formed in the pair-production process excites the crystal. When annihilation occurs, both of the gammas formed may lose all their energy in the crystal also. This gives rise to a pulse that contributes to the photopeak. Since one or both of the photons may escape from the phosphor, other peaks are also possible. A single-escape peak occurs at a pulse size given by E_γ - 0.511 MeV. This is due to the escape of one gamma but not the other. Also, a double-escape peak may occur at a pulse size given by E_γ - 1.022 MeV.

Several other peaks may also show up in the spectrum. The Compton scattering in materials other than the detector (often a backing substance) may give rise to a backscatter peak. This peak occurs at a pulse size that corresponds to the energy, E_b, given by

$$E_b = E_\gamma - E_C. \qquad 12.3$$

A coincidence sum peak may appear when a source emits two or more gamma rays in cascade.[51] The time delay between the emission of the two products is so short that the crystal "sees" them as only one. Thus, the pulse size that results will be found at $E_1 + E_2$. This type of peak may also occur when the source decays by more than one mode. This may be internal conversion and gamma emission, for example. A random sum spectrum occurs as a continuous distribution of pulses above a photopeak to a size about twice the photopeak value.[48,51] This effect occurs when random decay events in the source occur too close together in time to be resolved by the system.

Extra pulses may be produced in the Compton region because of scatter from the source surroundings or from x rays emitted during absorption of source photons by nearby substances. At very low energies in the spectrum, we begin to get a continuous distribution due to bremsstrahlung. This occurs when electrons produced by photon interactions, are stopped in an absorbing substance.

(c) Pulse Height Spectrum Examples - The width of the peaks that occur in gamma spectrum analysis can be traced to statistical fluctuations in the system. So, the peak consists of a distribution of pulse sizes. That is, a large number of pulses are about the same size (the photopeak value), but some pulses are larger and some smaller than this size. The resolution of the peak is defined as the ratio of the full width at half maximum to the energy value at the photopeak.[1] The resolution improves as one goes to higher gamma energy, but the energy resolution of scintillators is the poorest of commonly used detectors. For a good detector, the resolution for the ^{137}Cs gamma ray will be about 6%. For a given peak, the resolution R is given by

$$R = \frac{\text{full width at half maximum (FWHM)}}{\text{pulse height of photopeak}}. \qquad 12.4$$

The pulse height and the FWHM must be in the same units, channels or MeV.

The spectrum from a given source depends upon the interactions taking place in the crystal. The interactions in turn are a function of the photon energy from the source. If the source gives off many gammas that vary in energy, the pulse-height spectrum is very complex.[50,51] Many of the peaks may be masked out by the Compton spectrum from higher peaks. If the source has only a few gammas, the spectrum may be simple with only a few peaks. For any given source, the pulse height spectrum reflects <u>only</u> the energy given up in the crystal. Figure 12.18 shows a typical pulse height spectrum obtained for ^{137}Cs gamma rays in a NaI(Tl) crystal. The large bell-shaped peak is the photopeak from the highest energy (661.6 keV) gamma of ^{137}Cs.

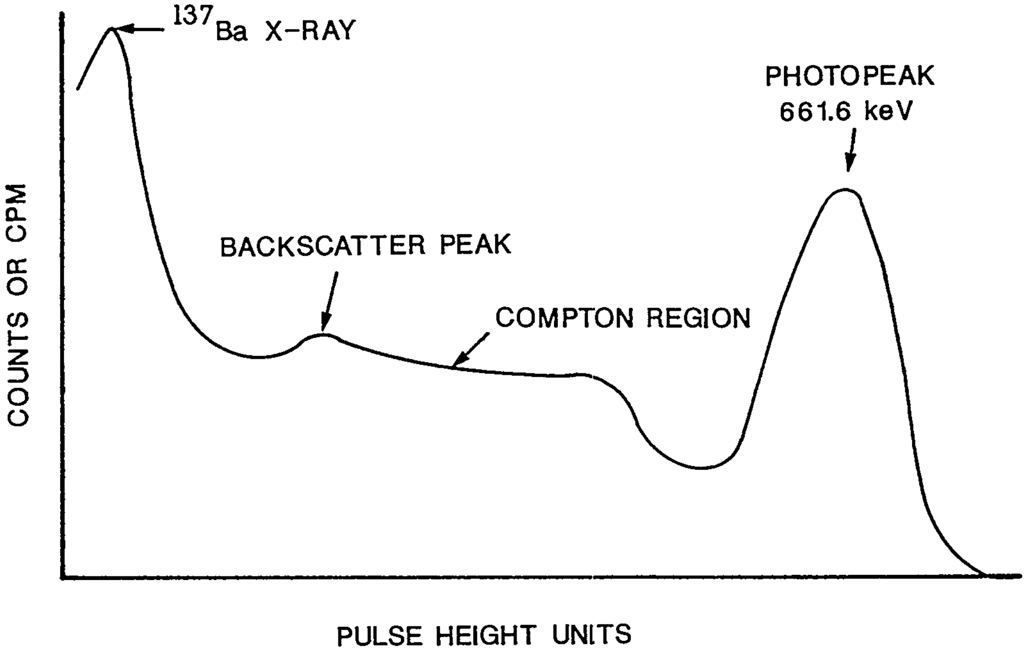


Figure 12.18 ^{137}Cs pulse height spectrum from a NaI(Tl) crystal.

The Compton region starts at a pulse height equivalent to an energy of

$$E_C = \frac{E_\gamma^2}{E_\gamma + 0.256} = \frac{(0.6616)^2}{0.6616 + 0.256} \sim 0.48 \text{ MeV}$$

This region extends from the energy E_C down toward zero. The pulse heights are due only to the energy transferred to recoil electrons in the crystal. This energy is, at most, 0.48 MeV; hence, all the pulse heights fall at this value or less.

The backscatter peak occurs at a pulse height that corresponds to $E_b = E_\gamma - E_C = 0.6616 - 0.48 = 0.18$ MeV. the peak is produced when a gamma that starts out of the source in a direction opposite to the crystal undergoes a 180° scattering reaction. The photon then proceeds into the crystal, where it loses its energy. In a 180° scattering event, the gamma gives up the maximum amount of its energy to the electron. We have seen above that this is E_C.

The final peak at the low end of the pulse height scale is the K x ray from ^{137}Ba. This 32 keV x ray follows internal conversion, an alternate decay mode. That is, ^{137}Cs emits a beta and goes to ^{137m}Ba. This is an excited state from which the 0.6616 MeV gamma usually comes. Instead of the gamma, the excited nucleus may eject a K-shell electron to rid itself of excess energy. Then, this internal conversion results in a shell vacancy, which gives rise to K x rays when it is filled.

Our last spectrum (Figure 12.19) is that for ^{24}Na. ^{24}Na emits a beta particle, which is followed by a 2.75 MeV gamma and a 1.37 MeV gamma in cascade. Here, the 2.75 MeV peak appears at the highest position on the pulse height scale. This photon has enough energy to interact by pair-production. The peak at 2.24 MeV is then the single-escape peak. This shows up if only one of the 0.511 MeV annihilation gammas escapes from the crystal. The peak at 1.73 MeV is the double-escape peak. This will show up only if both gammas escape. The

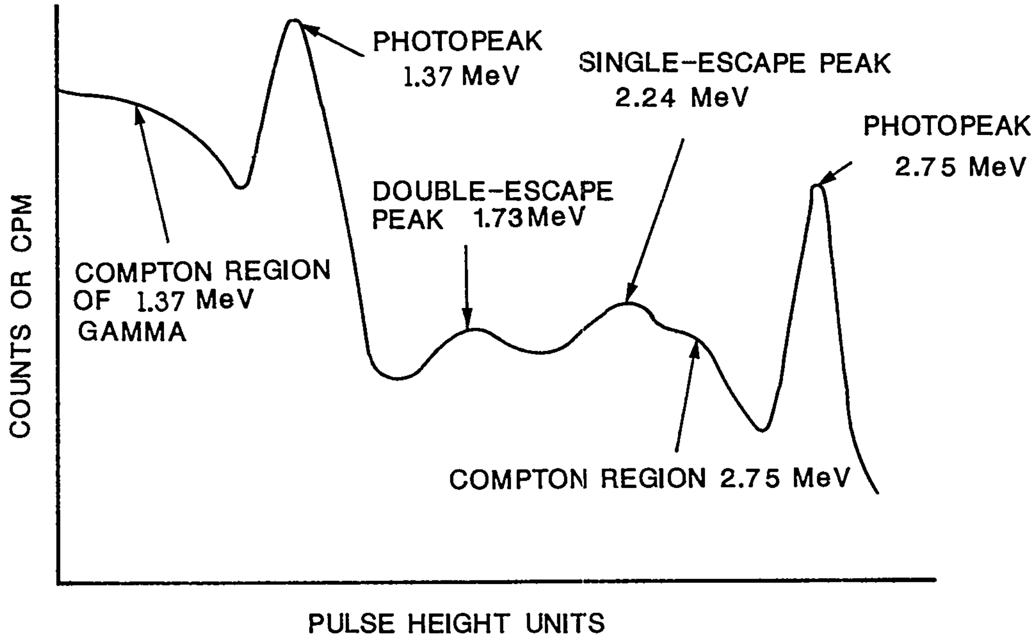


Figure 12.19 ^{24}Na pulse height spectrum from a NaI(Tl) crystal.

shape of these peaks is affected by the presence of the Compton region for the 2.75 MeV gamma. That is, $E_C = 2.52$ MeV.

The next peak shown is the photopeak for the 1.37 MeV gamma. This peak structure is also affected by the Compton distribution from the higher peaks. In this case, though, the photopeak count is much greater, because of higher crystal efficiency at lower energy, so that the peak is still bell-shaped. The Compton region from this peak starts at a pulse height equivalent to $E_C = 1.15$ MeV. Again, if we go to the lower end of the spectrum, the effects are somewhat the same as in the case of ^{137}Cs.

More examples have been published elsewhere.[48,52] These spectra have accrued over the years because of the many varied applications of the gamma scintillation-spectrometry method. The scintillation method itself has been useful in many health physics programs. Information on the proper setup and calibration of NaI systems can be found in Reference 53.

(2) Semiconductor Method

In this method, the detector may be lithium-drifted germanium (GeLi), high purity germanium (HPGe), or for low energy applications, lithium-drifted silicon (SiLi). The latter is used primarily for regions below 60 keV. GeLi detectors must be kept cooled at all times by liquid nitrogen or the detector and the FET (field effect transistor) in the preamplifier will suffer permanent damage.[54] HPGe and SiLi detectors require cooling when the high voltage is applied to protect the preamplifier, but the detectors can be allowed to warm up to room temperature. The detector is supplied with a Dewar (approximately 30 l), see Figure 12.20, which can be filled with liquid N_2. The detector is mounted on the end of a long metal rod (called a cold finger), which conducts heat away from the detector to the liquid nitrogen.[48]

The detector is generally located in a lead cave. The rest of the counting system is similar to that employed in the case of scintillators. Semiconductor detectors are generally smaller in volume than NaI crystals and so are less efficient. However, their resolution is far superior (approximately ¼% for ^{137}Cs) to that of NaI. Although the same interactions occur in these detectors, the peaks in the spectrum are much narrower than those in a NaI system. Consequently, the enhanced resolution makes it possible to more easily identify individual peaks in the spectrum. In addition, the contributions to the Compton scattering region are reduced since more scattered radiation escapes from the smaller volume detectors. This increases the ratio of the counts in the peak to the average counts in the Compton region which also improves detection.[54]

In expressing the relative efficiency of a germanium system, the comparison standard has been the absolute efficiency of a 76 mm by 76 mm (3-in by 3-in) NaI(Tl) detector. For a ^{60}Co source, and a source to detector distance of 0.25 m, the absolute efficiency for the NaI detector is 1.2×10^{-3} for the 1332.5 keV line of ^{60}Co.[51] The relative efficiency for a semiconductor detector is expressed as a percentage of

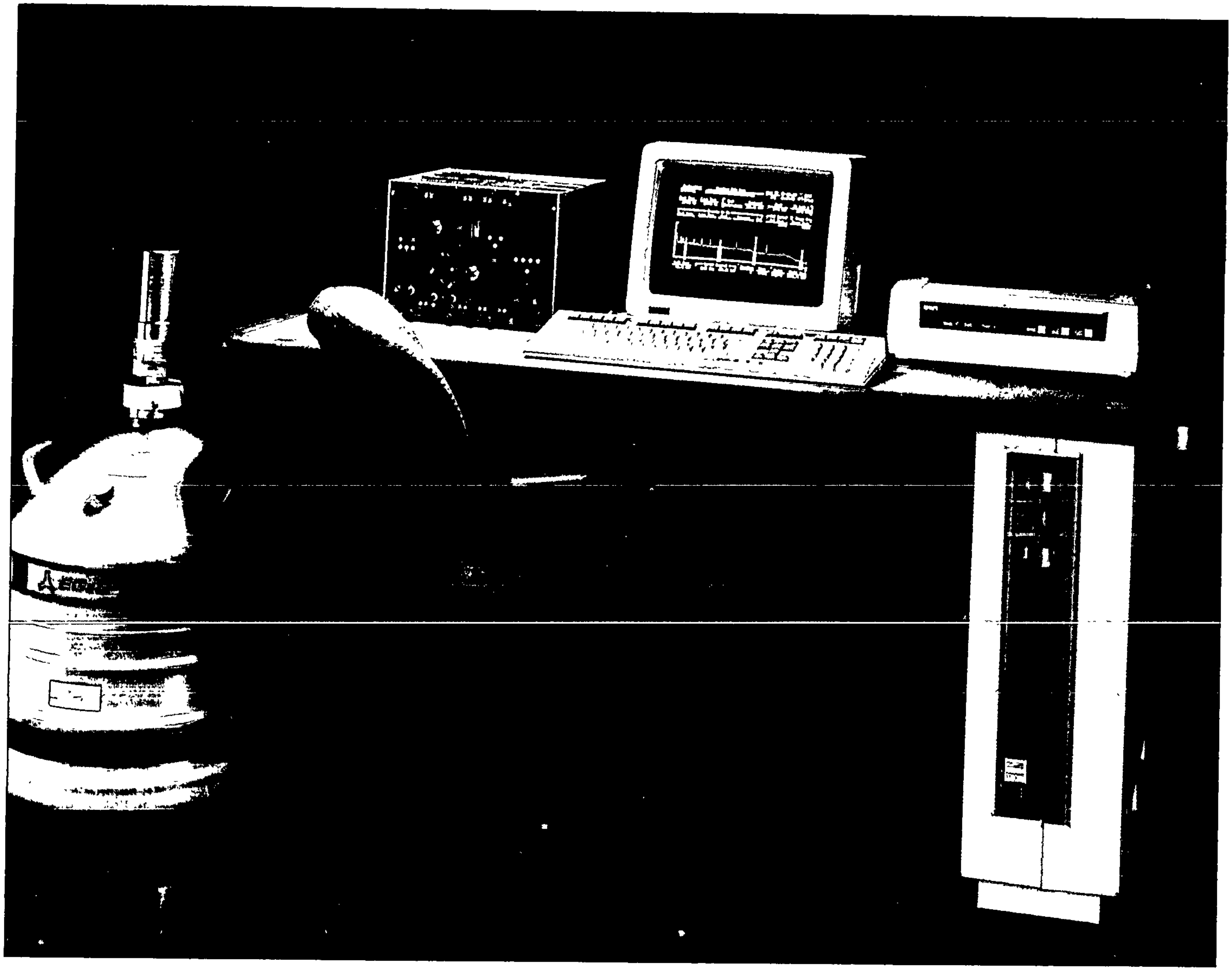


Figure 12.20 A gamma spectroscopy system utilizing a desk-top personal computer for data analysis. (Courtesy of EG&G ORTEC.)

the above efficiency for the NaI detector. Thus, a germanium detector with a relative efficiency of 20% means that when the detector is at .25 m from a ^{60}Co source, the efficiency for the 1332.5 keV line will be $(.2)(1.2x10^{-3}) = 2.4x10^{-4}$.

Modern analysis systems have incorporated features to quantify the peaks found in a spectrum. The EG&G Ortec ADCAM system, shown in Figure 12.20, is composed of a data collection system, used in conjunction with a desk-top personal computer, dual disk drive and printer. Data is collected in a multichannel buffer, and transferred to the computer system for analysis, using available software packages. The analysis provides the identity and the activity of the radionuclides in a given sample. The spectrum can be stored on diskette in order to retain a permanent record, if desired. The system allows the development of individual software for specialized applications.

Details concerning the analysis of spectra by methods which may be performed by computers can be found in References 1, 48, 50 and 54. These discussions include methods of spectral smoothing, peak location and peak search, peak shape determination, and activity determinations. With the development of these approaches, the use of gamma spectrometry has evolved into both a quantitative, as well as a qualitative, method for the determination of radionuclides. In these computer aided analyses, a number of different calibrations can be run and the results stored in the computer or on a diskette. Then, when a certain type of sample needs to be counted in a given sample-detector geometry, the information can be retrieved and utilized to identify and quantify the activity in a given sample. A number of geometry and absorption corrections, corresponding to certain sample counting situations, may be stored in the computer.

The spectrometry system in Figure 12.20 is a stationary system which requires that samples be brought to the system for analysis. However, advances in electronics allow multichannel analyzers to be packaged in a portable, lightweight, battery-operated package (see Figure 12.21). The entire 4096 channel MCA is contained in a .18 m by .28 m by .23 m (7-in by 11-in by 9-in) completely enclosed case which

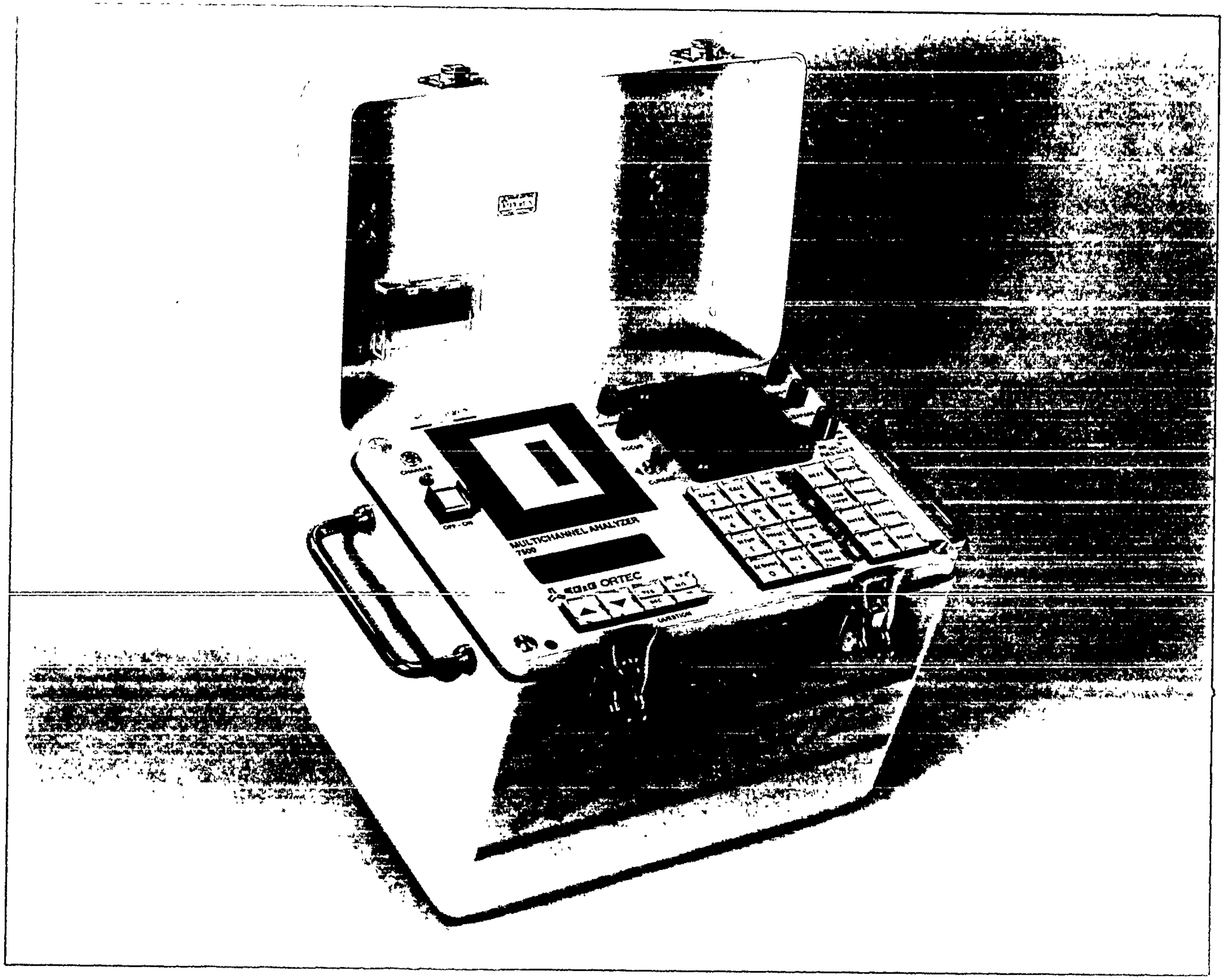


Figure 12.21 Portable MCA which can be used with a portable germanium or NaI detector. (Courtesy of EG&G ORTEC.)

weighs 7.2 kg (16 lb). Used in conjunction with a portable HPGe detector or a NaI crystal, this unit allows analysis in the field at the location of interest. The spectrum is displayed on a cathode ray tube during acquisition, and can be stored on a digital cassette recorder which is also part of the unit. The stored spectrum can be returned to the counting laboratory where it may be used in conjunction with the ADCAM system for full analysis.

Information on the proper setup and calibration procedures for germanium systems can be found in References 54 and 55.

(3) Spectrum Analysis Principles

As was stated above, in addition to the energy spectrum of a photon source of radiation, it is also possible to determine the activity. It is possible, but difficult, to compute the efficiency of a gamma spectrometry system. If one knows the attenuation coefficient of the radiation in the detector material and the geometry of the setup, the efficiency may be calculated.[48] More often, efficiencies are measured by using a number of known sources, well spaced over the energy region of interest, and measuring the response to these sources.[53,55]

For a detector in the form of a cylinder of radius r_o (see Figure 12.22), and a point source of radiation located on the axis of the cylinder, a distance Z away, the geometry depends only upon the radius of the detector face and the distance between the source and detector, Z. From equation 11.12, the geometry is given by:

$$G = \frac{1}{2}(1 - \cos\theta) = \frac{1}{2}\left(1 - \frac{Z}{\sqrt{Z^2 + r_o^2}}\right).$$

So, G may be calculated if one knows the radius of the active area of the detector.

The intrinsic efficiency,[1] ϵ_E, may be defined as the fraction of the photons incident on the crystal face which interact in the crystal. This fraction will depend upon the energy of the photon,

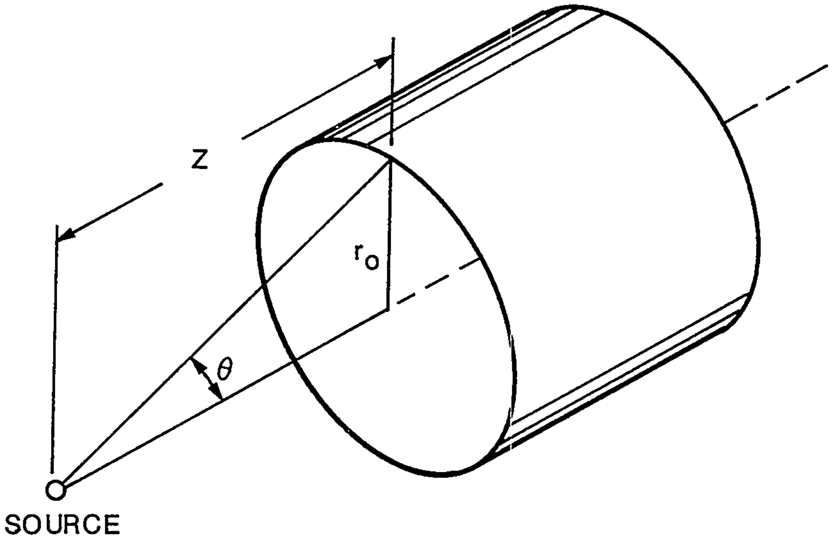


Figure 12.22 Geometry for a cylindrical detector and a point source.

and the active area and the volume of the crystal. For a beam of photons incident on the detector face, the fraction interacting, F_I, will be

$$F_I = 1 - e^{-\mu x}. \qquad 12.5$$

For a point source, the radiation strikes the crystal at different angles, which will be influenced by the geometry of the situation, so that, in general,

$$\epsilon_E = \frac{1}{G}\int_{\text{Vol.}} (1 - e^{-\mu x})\,d\Omega, \qquad 12.6$$

requires an integration over the volume of the detector, with respect to the solid angle, Ω. In general, the intrinsic efficiency decreases as E increases, and decreases faster in small crystals compared to large as E increases. At this point, one may say that the probability of a photon being intercepted by the detector and interacting in the detector is $\epsilon_E G$. If the photon then interacts, the result will be energy lost in the detector, which results in a certain pulse height being produced. But, this pulse may not be in the photopeak. The ratio of the counts in the photopeak to the counts in the rest of the spectrum is called the peak-to-total ratio, R_E, and may be expressed as

$$R_E = \frac{\text{area under photopeak}}{\text{area in total spectrum}}, \qquad 12.7$$

in which R_E is generally experimentally determined. For the absolute determination of a source activity A, the final expression will be

$$A\ (\mathrm{Bq}) = \frac{N}{R_E G \epsilon_E n_\gamma t}, \qquad 12.8$$

in which N is the total counts in the area under the photopeak, n_γ is the fraction of disintegrations in which the γ of interest is emitted, t is the counting time in s, which is expressed as the live time on MCA units, and $R_E G \epsilon_E$, the photopeak efficiency, has been explained above. The quantities G and ϵ_E can be computed adequately, but R_E usually requires experimental determination.

Since the determination of the absolute efficiency is difficult, relative efficiencies are usually determined. In this approach, a number of point isotropic photon sources, covering the desired energy range are used to determine the efficiencies.[54] The yield, Y_P is given by

$$Y_P = \frac{\text{area under the photopeak (counts)}}{\gamma \text{ emitted from the known source}}. \qquad 12.9$$

By plotting the Y_P values versus gamma energy for a number of gamma energies, a curve can be drawn through the points to obtain the yield for other gamma energies in the range covered.

In equation 12.9, the numerator contains an expression for the total counts (area) under the photopeak. In most situations in which a sufficient number of counts from the source is acquired, a bell-shaped curve will result in the spectrum. Then the peak may be mathematically treated as a Gaussian error curve or normal distribution (see Section 11.I.1). That is, the number of counts between a pulse size x and an increment Δx (generally in channel width), is given by

$$Y = Nf(x)\Delta x, \tag{12.10}$$

in which f(x), see Figure 12.23, is the function which describes a normal distribution, and N = the total counts in the photopeak (the area under the photo peak). The function f(x) is expressed by

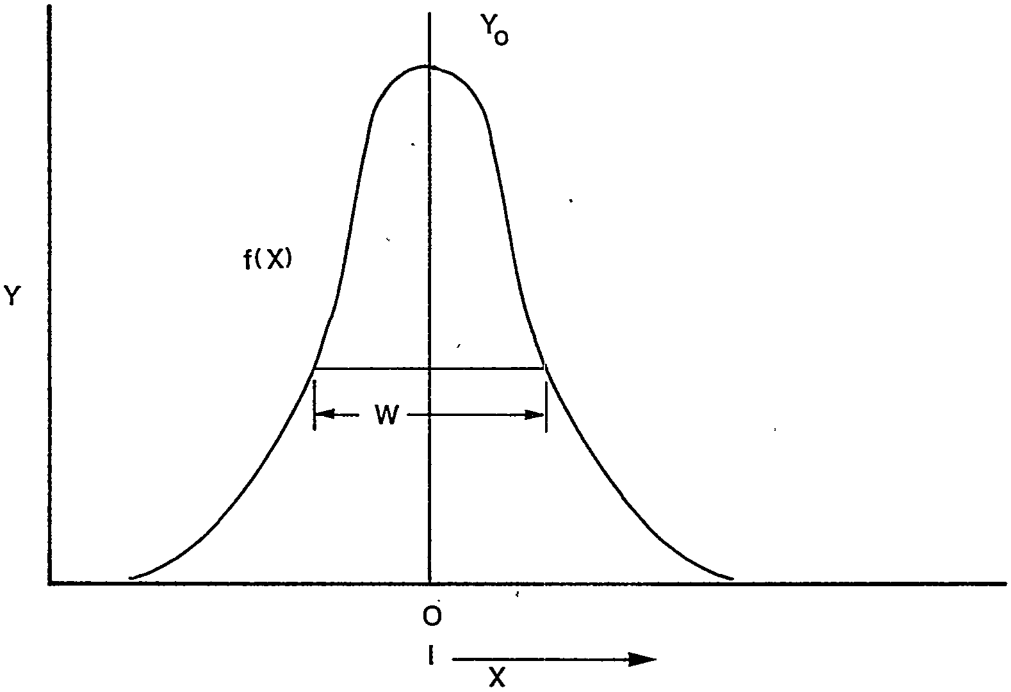


Figure 12.23 Peak parameters of a Normal (Gaussian) distribution.

$$f(x) = \frac{1}{\sqrt{2\pi}\,\sigma}\, e^{-\frac{(x-\bar{x})^2}{2\sigma^2}} \tag{12.11}$$

in which σ is the standard deviation of the distribution (Section 11.I.2), in the same units as x, and $\bar{x}$ is the mean value of the distribution, or peak channel in this case. Substituting this expression into 12.10 gives

$$Y = \frac{N\,\Delta x}{\sqrt{2\pi}\,\sigma}\, e^{-\frac{(x-\bar{x})^2}{2\,\sigma^2}} \tag{12.12}$$

which then describes a typical peak shape in terms of the total counts in individual channels, x.

When the peak is at the maximum, $x = \bar{x}$, and $Y = Yo$, which from equation 12.12 gives

$$Y_o = \frac{N\,\Delta x}{\sqrt{2\pi}\,\sigma}. \tag{12.13}$$

At half maximum intensity, $Y = Y_o/2$ and from equations 12.12 and 12.13,

$$\frac{Yo}{2} = Yo\, e^{-\frac{(x-\bar{x})^2}{2\,\sigma^2}}$$

$$\frac{1}{2} = e^{-\frac{(x-\bar{x})^2}{2\,\sigma^2}}$$

and $$2 = e^{\frac{(x-\bar{x})^2}{2\,\sigma^2}}$$

so $$\ln 2 = \frac{(x-\bar{x})^2}{2\,\sigma^2},$$

but $x-\bar{x}$ equals the half width. The full width, W, at half maximum intensity, is then

$$W = 2(x-\bar{x}) = 2\sqrt{2\ln 2\sigma^2} \qquad 12.14a$$

$$= 2\sigma\sqrt{2\ln 2} = 2.3548\ \sigma \qquad 12.14b$$

and

$$\sigma = \frac{W}{2.3548} \qquad 12.15$$

The total counts in the peak N can then be estimated from equation 12.13,

$$Y_o = \frac{N\ \Delta x}{\sqrt{2\pi}\ \sigma}$$

and

$$N = \frac{\sqrt{2\pi}\ \sigma Y_o}{\Delta x} = \frac{\sqrt{2\pi}\ Yo}{\Delta x}\ \frac{W}{2.3548} = 1.0645\ \frac{W}{\Delta x}\ Yo \qquad 12.16$$

so the total counts N in the peak distribution can be determined from the counts in the peak channel, Y_o, and the full width at half maximum (FWHM), W, expressed in units of Δx. The count rate may then be found by dividing the total counts in the peak by the counting time.

d. Fast Neutron Spectrometry

The most common method used for fast neutron spectrometry utilizes proton recoils produced in elastic scattering interactions.[1] Because the cross section for (n,p) reactions is well known from thermal energy to 20 MeV in hydrogen, this material is usually chosen as the target nucleus. In addition, the neutron can transfer a larger fraction of its energy (up to its entire energy) in elastic collisions with protons than with heavy nuclei. The average energy transferred by an incoming neutron in collisions with protons is about ½ of E_n. This allows detection of the larger pulses formed by recoil protons while excluding pulses from gamma rays or other unwanted pulses. This ability to discriminate becomes difficult below about 200 keV, unless special techniques are employed.[1]

The theoretical response function for hydrogen predicts a recoil proton energy distribution as shown by the solid line in Figure 12.24. However, the response function of a detector based upon simple hydrogen scattering does not have this shape. Other factors act to complicate the shape,[1] so that the response may actually look more like the dotted line in Figure 12.24. The analysis of the data then, requires a computer program to unfold the spectrum data.

For neutrons in the range approximately 10 keV-1.5 MeV, neutron spectrometry is best performed with the use of gas proportional counters utilizing an hydrogenous gas. In the range from 1.5-20 MeV, scintillators containing hydrogen, in which the recoil protons produce light are used.[56] NE-213 is one of the more common scintillators used for this purpose. The non-linear response of the scintillator material, as

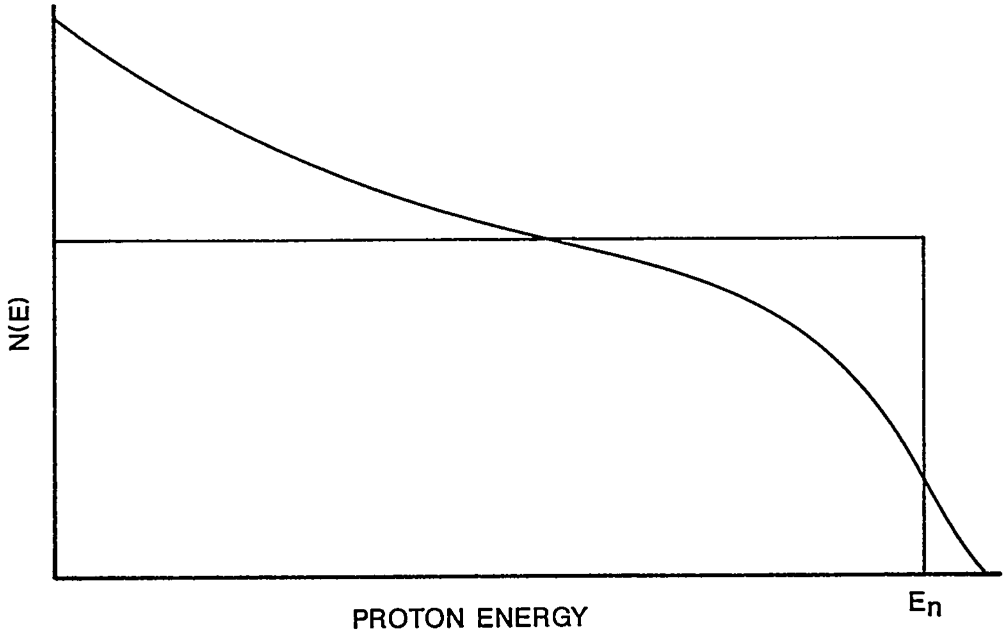


Figure 12.24 Energy distributions of recoil protons from neutron elastic scattering in a detector.

far as the light output is concerned, is one of the factors which contributes to the low energy portion of the dotted line response shown in Figure 12.24.[1] One of the problems in the scintillation method is that gamma also produces a response (^{60}Co gamma rays produce as much light as neutrons of several MeV). A fortuitous circumstances is that the decay of the light from gamma produced scintillations occurs faster than that from neutrons, so that the electronics can be designed to separate the neutron and gamma signals. In effect, whatever scintillation detector is used, one must provide a means of discriminating between n- or gamma-produced pulses, unless the radiation field is gamma free.

3. Liquid Scintillation Counting

The main application of liquid scintillation counting is for identification of low energy beta emissions - particularly ^{3}H and ^{14}C.[1] The method may also be applied to the identification of alpha emitters.[48]

The approach which is generally used is to incorporate the sample directly in the liquid scintillator material. One of the problems which may be encountered is the reduction in the light output when a sample is added to the liquid scintillator.[1] This aspect is referred to as quenching and since it alters the expected response, it can introduce significant error. With respect to practical applications, it limits the amount of sample material which can be introduced into the liquid scintillator.

A common procedure consists of preparing a standard amount of liquid scintillator in a vial, to which is added a suitable quantity of the sample. For example, polystyrene smears for ^{3}H may be added to the appropriate liquid scintillator and the polystyrene will be dissolved, giving a suitable sample. The vial is then placed into a light tight enclosure, and the sample is viewed by one or more photomultiplier tubes. The output is recorded as the total counts seen in a given energy range.

These systems are prone to electronic noise pulses of one type or another. Since the light pulses produced by radiation; namely, ^{3}H or ^{14}C, are small, care must be taken to increase the signal (pulses from radiation) to noise (background pulses) ratio. One approach in commercially available counters is to have two photomultiplier tubes, one on each side of the vial, which operate in the coincidence mode.[1] In this way, light output pulses can be counted while the background is substantially reduced.

E. Calibration

An important aspect in the use of instruments to measure radiation fields is the accuracy of the reading. The accuracy is the degree of agreement between the observed value and the true value of a quantity being measured.[24] Many of the instruments and devices used to measure radiation are not absolute devices, meaning that the response of the device must be related to known radiation values in order to adequately measure the parameter of interest. In some cases, one is interested in the exposure X, or the exposure rate $\dot{X}$. In other cases, the interest is in the absorbed dose D, or the absorbed dose rate $\dot{D}$. A further interest may be to characterize levels of surface contamination with respect to alpha or beta radiation. For neutron fields, due to the variation in the quality factor Q, one may need to measure some parameter which gives an approximation to the dose equivalent H, or the dose equivalent rate $\dot{H}$. Our previous discussions in Section 12 have focussed on describing the many types of available instruments used to perform some of these measurements. In order to properly interpret the response of these instruments, calibrations need to be performed. To calibrate means to determine the response or reading of an instrument relative to a series of known radiation values covering the range of the instrument, or to determine the strength of a radiation source relative to a standard.[24] In the latter case, this is sometimes referred to as a measure of the source output.[57] Much of the material found in the following discussion is based upon material in

References 21, 24 and 57. ANSI N323-1978 (Reference 24) is a required standard for test and calibration of instruments at DOE facilities.

The objectives of a calibration program include assuring that the instrument works properly, determining whatever error in reading may be occurring and, if possible, adjusting the instrument response to correct this error, investigating the reliability of the type of instrument, and providing guidance to the users on the relative quality of the instrument.[57]

Many nonradiological characteristics which may affect the performance of the instrument should have been evaluated by the manufacturer. Included are such things as effects of magnetic fields, rf fields, etc.[24] Radiological characteristics, such as energy dependence, range, sensitivity and others should also have been evaluated. Since with use, the components of an instrument tend to wear out, certain of these radiological characteristics need to be periodically evaluated.[21] In addition, each new instrument should be given an initial checkout.

1. Approach

In general, two approaches are used in the calibration of portable instruments.[57] One is to use a radiation field whose properties are well known. This may be a standard source (national standard) calibrated by the National Bureau of Standards (NBS) or a measurement of the field with a secondary standard instrument. A secondary (derived) standard instrument is an instrument which has been calibrated by comparison with a national standard. Such an instrument cannot have an uncertainty greater than ±10% and must be recalibrated at intervals no greater than three years. The second approach is the substitution method. A certain type of instrument, which has been calibrated against a national standard, is placed in the radiation field and the response noted. The same type of instrument which is to be calibrated is then substituted for the original instrument. Calibration is based upon comparison of the two readings. Cavity chambers (thimble chambers) are frequently used as the secondary standard instrument for this method.

Instruments are calibrated with either national standards or derived standard sources. Radionuclide sources used for calibration should have a relatively long half life (at least several years), sufficient output to cover the full ranges of those instruments to be calibrated, small physical size, and emit radiation of energy similar to that which the instrument is expected to measure. For photon energies below about 300 keV, x ray machines with heavy filtering are used as the calibrating radiation field. Instrument response at several photon energies over the energy range covered by the instrument should be evaluated if the instrument will be used to measure fields of vastly differing energies.[24] Exposure rates for photon sources or the fluence rate for neutron sources must be known within ±10%.

A reproducible source detector geometry is required for all calibration test measurements. Positioning errors of either instruments or radiation sources must not affect the radiation field values by more than ±2%.[24] The source-to-detector distance is measured between the effective center of the source and the effective center of the detector. If this distance is greater than seven times the maximum dimension of either source or detector, the source-detector combination can be considered a point source and point detector system. The distance from the source or the instrument to scattering objects should be at least twice the source-detector distance.

2. Checkout for Calibration

ANSI N323 requires that a number of checkout conditions shall be accomplished prior to an actual instrument calibration. This will ensure that the device is operating properly. The first thing to do is check the instrument for contamination. Batteries are to be checked and replaced if necessary. The high voltage is then set to the instrument requirements and the instrument warmed up and set to zero or a reading specified by the manufacturer. For digital readout instruments, the performance of any internal sampling time base is verified as within the manufacturer's specifications. Any geotropic effect noted in the three mutually perpendicular planes should be corrected for in the subsequent calibration.

3. Calibration Methodology

In the actual calibration process, the reproducibility of the instrument must be established before any adjustments are made. To check this, the device is exposed to a radiation field for three or more times. Readings should fall with ±10% of the mean reading for a properly operating instrument.[24]

The response of the instrument may be affected by several environmental factors as well as measurement conditions.[21,24] So, the calibration should be performed under conditions specified by the manufacturer. Such parameters as temperature, pressure and humidity need to be recorded since they may be required in arriving at correction factors.

Each linear scale control should be adjusted in the radiation field according to the manufacturer's specification or at the scale midpoint. The calibration is then checked at about 20% and 80% full scale on that range. Readings within ±10% of the known radiation field are acceptable. If only one control is supplied, adjustment is made at the manufacturer's specified point, or near the middle scale midpoint, or near the midpoint of the scale most likely to be used at the facility.[24]

For a logarithmic readout, one may encounter a single readout scale covering a number of decades with several adjustment controls. Adjustments for each decade should follow the manufacturer's specifications. Calibration is checked at a minimum of one point near the midpoint of each decade. Readings within 10% of the known value are acceptable.

If one prepares a calibration chart which is supplied with the instrument, readings within 20% are acceptable for either a linear or logarithmic readout.

To assure continued proper operation of a calibrated instrument, check sources should be used to verify instrument performance. For instruments in continual use, checks should be made several times a day. For less frequent usage, the check is made prior to <u>each</u> use of the instrument.

A primary calibration is required for survey instruments at least annually. When subjected to unusual operating conditions, hard usage, or corrosive environments, more frequent calibrations are indicated. Some instruments require calibration every six months and regulatory requirements for radiographers specify quarterly calibrations.

4. Calibration Sources

a. Photon Sources

No one source will cover the full photon energy region from a few keV to several MeV. Above 300 keV, a number of radionuclide sources are generally used. The most common have been ^{137}Cs, ^{60}Co and ^{226}Ra. Below 300 keV, heavy-filtered x rays or fluorescence x rays are employed.[57] These x rays may be calibrated by use of a derived standard instrument. An ideal radionuclide calibration source would emit photons with an energy spectrum similar to that which is to be measured and also have a long half life. The ideal can seldom be achieved. However, listings of sources useful for photon field calibrations can be found in References 21, 24 and 57.

b. Beta Sources

Beta response of an instrument can be determined with a source which has been calibrated with an extrapolation chamber (see 12.D.1).[24] This allows determination of the surface dose rate. To obtain a uniform field, the source must be large in area and flat, as well as uniform in activity distribution. Calibration of a beta point source is difficult because of the spectrum of beta energy involved, and the extensive scattering and attenuation which takes place in the intervening air. Many applications of beta surveying require only a measurement of the surface activity, such as in contamination surveys. In these instances, the activity of the source, not the dose rate, needs to be known for calibration purposes. Beta sources suitable for instrument calibration can be found in References 24 and 57.

c. Alpha Sources

Standard alpha sources may be purchased from NBS. These sources are usually electroplated metallic alpha radiation emitters on a stainless steel disk.[21] The source is assayed in a 2π standard counter (50% geometry). A table of suitable alpha sources can be found in References 24 and 57.

d. Neutron Sources

The types of sources used for neutron instrument calibration include radionuclide sources [(α,n), (γ,n), (fis,n)], particle accelerators and reactors. Radionuclide sources in common use include ^{226}Ra-Be, ^{239}Pu-Be, ^{238}Pu-Be, ^{241}Am-Be and ^{252}Cf. Radionuclide sources can be sent to NBS for calibration and then become secondary (derived) standard sources. Standardization of neutron fields from accelerators or reactors must be accomplished with standard instruments or techniques. In addition, continuous monitoring of the neutron output during calibration procedures is required since changes may occur during the calibration.

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QUESTIONS

12.1 What term identifies the field of study devoted to the measurement of energy imparted to matter by radiation?

12.2 What method is most used for the detection of α, β, and γ radiation?

12.3 Explain or state the Bragg-Gray principle as it applies to radiation detectors.

12.4 What term is used to identify materials that have the same absorption for primary radiation and the same stopping power for secondary particles as

a) air and
b) tissue?

12.5 A radiation detector which does not give uniform response for all energies of the radiation for which it is designed is said to be:

12.6 What condition is said to exist when for every secondary particle leaving a small volume of a material at a point of a certain thickness, one with equal energy will enter?

12.7 When the ratio of primary particles or photons to the secondary radiations in a material remain constant as the thickness increases that condition is said to be:

12.8 Under what circumstances does charged particle equilibrium not obtain?

12.9 What conditions are to be fulfilled for charged particle equilibrium to exist in a given case?

12.10 What term is given to the energy per unit mass transferred to charged particles by the uncharged particles?

12.11 When, and under what conditions, is kerma less than the absorbed dose in the medium?

12.12 Why are the chemical effects caused by radiation of limited value for personnel dosimetry?

12.13 What is the principal reason for having to use different instruments or different techniques with the same instrument for personnel dosimetry?

12.14 Identify and explain the meaning of LET.

12.15 Why is LET important in personnel dosimetry?

12.16 What term is given to the broad class of devices and instruments that readout in terms of dose?

12.17 List some of the problems of neutron dosimetry.

12.18 Do instruments which measure dose equivalent, underestimate or over estimate the neutron dose, if the neutron spectrum energy is in the intermediate region?

12.19 What radiation(s) usually interfere in a neutron field? How can one mitigate the interference in measurement?

12.20 Explain the depth dose equivalent indexes.

12.21 Explain the difference between a pocket chamber, a minometer, and a pocket dosimeter.

12.22 In what basic terms is the readout for most survey meters?

12.23 Although dosimeters measure dose, not dose rate, the dose rate may be obtained by observing the dosimeter's:

12.24 From what do most of the troubles with ion chambers arise?

12.25 Of the instruments described in this section, which would be most suited to the circumstances given below?

a) A survey meter which can discriminate between alpha, beta, and gamma radiations.

b) An instrument suited principally to alpha surveys, but which can identify any beta-gamma contribution.

c) A detecting device sensitive to the individual pulses produced by beta particles and gamma photons. Especially suited to low fields and locating individual spots of contamination.

d) An instrument with a wide range of response for gamma rays and high energy photons which may be used remotely.

e) An instrument suited to the monitoring of an individual who will be working a low-level gamma field over extended periods.

f) Two instruments which are capable of discriminating electronically between alpha particles and a beta-gamma field. Which instrument could pose a special hazard to air transportation? Why is it preferred over the other instrument?

g) An instrument that will detect neutrons over a very wide range of neutron energies.

h) A proportional counter which will detect thermal neutrons, but which is almost insensitive to high energy neutrons.

12.26 What is a "long counter" and why is it called a "long counter"?

12.27 What radiation is detected by activation unit survey meters?

12.28 What neutron detector can indicate the fluence, dose equivalent, and effective neutron energy?

12.29 For what purpose is an extrapolation chamber most useful?

12.30 What is the name of the group of spectrometers which measure pulse size? What are the two classes of this type of instrument?

12.31 What determines the type of radiation which can be measured and recorded by a spectrometer?

12.32 Explain the advantage of the LET spectrometer. Why is the proportional counter gas kept at a low pressure?

12.33 Explain the value of the surface barrier alpha spectrometer.

12.34 What three types of interaction by gamma rays entering a scintillation crystal are expected and can be shown through the use of a gamma spectrometer?

12.35 Identify the part of a gamma ray spectrum as shown by a spectrometer that corresponds with the following:

a) The pulse-height peak that results from the largest (single) light pulse.

b) The maximum energy of the Compton effect electron.

c) The peak that results when two photons from a cascade occur so closely in time that the spectrometer "sees" them as one.

d) A peak that occurs in the spectrum 0.51 MeV less than the photopeak.

e) A peak that occurs in the spectrum 1.02 MeV less than the photopeak.

f) The continuous distribution of the spectrum beyond the photopeak.

g) A peak 28 keV (0.028 MeV) less than the photo peak when a NaI(Tl) detector is used.

12.36 What term is used to express the sharpness of a photo peak in terms of its width.

12.37 What are the two semiconductor detectors that are widely used for photon spectrometry.

12.38 How is the relative efficiency of a semiconductor detector expressed?

12.39 What is the intrinsic efficiency? On what factors does it depend for a given semiconductor detector?

12.40 What is the most common method used for fast neutron spectrometry? Name the most commonly used scintillator for this purpose.

PROBLEMS

12.1 What is the air absorbed dose in SI units for an exposure of 35 mR?

Answer: 3.045×10^{-4} Gy.

12.2 Compute the energy absorbed per kilogram in a large tissue-equivalent chamber if a charge of 2.58×10^{-4} C/kg is produced in a small air-filled cavity in the chamber. The relative mass stopping power of tissue is 1.11. Use $W_e = 33.85$ eV/ip.

Answer: 9.69×10^{-3} J/kg

12.3 What is the chance that a single thermal neutron will interact with 0.014 m depth of BF_3 when its density is 3 kg/m^3? Atomic weights of boron-10 and fluorine-19 are 10.013 and 18.998 respectively. The n,α cross-section for thermal neutrons in boron-10 is 3840 b.

Answer: 13.49%

12.4 An ionization chamber instrument has a lowest range of 5 alphas per second full scale. What current flows in the chamber for full scale deflection? Assume the alpha particle energy to be 4 MeV. Use $W_\alpha = 35$ eV/ip.

Answer: 9.14×10^{-14} amp

12.5 Complete the nuclear equation ${}^{10}B + {}^{1}n \rightarrow {}^{7}Li + {}^{4}He$ by indicating the Z numbers and the 2.5 MeV reaction energy.

12.6 Calculate the Compton edge for the following:

a) ${}^{65}_{30}Zn$, 1.11 MeV γ

b) ${}^{131}_{53}I$, 0.364 MeV γ

Answers: a) 0.902 MeV
b) 0.214 MeV

12.7 Find the backscatter peaks for the γ emission of the radionuclides given in the preceding example.

Answers: a) 0.208 MeV b) 0.15 MeV

12.8 Find the sum peak for $^{60}_{27}$Co having principal photo peaks of 1.17 and 1.33 MeV.

Answer: 2.50 MeV

12.9 What is the maximum intrinsic efficiency of a 2.54×10^{-2} m-thick NaI(Tl) detector for ^{60}Co gammas normally incident on its surface? The total attenuation coefficient for NaI at 1.25 MeV is 1.9×10^{1} m^{-1}.

Answer: 38%

12.10 Find the resolution of an instrument for a 1.61 MeV γ when the peak width at half maximum energy is 15,000 eV.

Answer: 0.932%

12.11 If the energy resolution of a given NaI(Tl) scintillation detector is 6.5% for ^{137}Cs gamma rays (0.662 MeV), estimate its energy resolution for the 1.33 MeV gamma rays from ^{60}Co. (Hint: the square of resolution is inversely proportional to the energy of the photopeak, in the first order approximation.)

Answer: 4.59%

12.12 A germanium detector has a photopeak efficiency of 15% relative to a standard NaI(Tl) scintillation detector. Estimate the count rate for the 1.33 MeV peak of a 0.2MBq ^{60}Co point source at a distance of 0.50 m.

Answer: 9 counts/s

SECTION 13 - PERSONNEL MONITORING DEVICES

A. Introduction

The preceding sections discussed many of the instruments and detection principles used in health physics. These are mostly electronic and measure a radiation field at a point and for an instant of time. The rate meters are very useful in survey work. We need the information they provide as an aid toward the evaluation of a potential hazard. We further need devices or methods that will indicate the total amount of radiation to which a person has been exposed. These devices are called dosimeters.

From the start of the Manhattan Project, a need for dosimeters was felt. Because of the large number of people who would come in contact with quantities of radiation, a small, rugged, and inexpensive dosimeter was needed. Studies revealed several features an ideal dosimeter must have. Some of these features are: (1) The response of the unit to equal exposures should be independent of the radiation energy, (2) the dosimeter should cover the range of exposures from less than 260 μC/kg (1 R) to greater than 0.13 C/kg (approximately 500 R), (3) the response should not be produced by agents other than the radiation measured, (4) the device should measure all ionizing radiation, and (5) the unit should be small, light, easy to handle, comfortable to wear, and low in cost. Up to the present, no known dosimeter possesses all these features. Over the years, though, many devices and methods have been studied in an attempt to improve the performance of personnel monitors.

From the outset, one type of device, the film dosimeter, has been used with satisfactory results.[1] Film dosimeters were used in almost all cases in which large numbers of people are involved. Development of solid state and other types of dosimeters has made alternatives to film dosimetry more attractive.[2-5] However, the film dosimeter is still one of the current devices in use for personnel monitoring, although TLD dosimeters have found widespread use (see Section 13.D.1).

B. Photographic Film Dosimetry

It would seem fitting that the very means by which radiation was first detected should also result in a means of measurement. Photographic film is one of the simplest means of <u>detecting</u> radiation. On the other hand, using film as a dosimeter requires careful attention to many details. Among these are: the type of emulsion, the response of the film, the processing techniques, and the interpretation of the exposure. Despite these factors, the film badge is still currently used. The fact that it is small, rugged, and low in cost enhances its value as a dosimeter. Many treatments in the literature[6-8] discuss the features of the film technique for dosimetry.

Since the effect of x rays on film was first noted, attempts have been made to relate the response of the film to the amount of exposure. We are, therefore, concerned with such aspects as the nature of the photographic process, the action of the radiation on the film, and the nature of the film response.

1. Emulsion Properties

Photographic film most often appears in the form of thin, even layers of the emulsion, spread on a thick support base (see Figure 13.1).

The base substance may be paper, glass, or cellulose. A typical x ray film would have a base about 200 μm thick and two emulsion layers, each about 12 μm thick.[7] The emulsion consists of small silver halide crystals, or grains, embedded in a gelatin matrix. The size of the crystals (most often silver bromide) and their content by weight differ for x ray films and neutron films. The average grain size, which greatly affects the film response, is about 1-2 μm in diameter in x ray film and about one-fifth that size in neutron film.[7] The amount of silver bromide in the emulsion ranges from 40 weight percent (w/o) for some common films to 80 w/o for neutron films.[6] Neutron films also differ from x ray films in that the neutron film emulsion is usually much thicker and often only in a single layer.

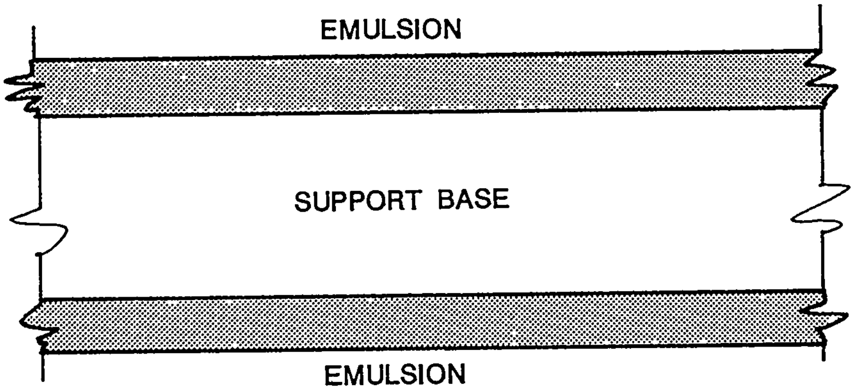

Figure 13.1 Cross section through a typical x-ray (photon) film. (Adapted from PERSONNEL DOSIMETRY TECHNIQUES FOR EXTERNAL RADIATION (Symposium), Madrid, OECD/ENEA (1963).[7])

2. Photographic Effect

In essence, the theory describing the effect of an exposing agent on a film was first advanced by Gurney and Mott.[9] More recent treatments[1,6,8] discuss the modern aspects of this theory.

a. Latent-Image Formation

In the Gurney-Mott theory, the energy transferred by the exposing agent causes electrons to be raised into the conduction band of the silver halide crystals. This requires about 5.8 eV per electron.[7] These electrons are trapped at defects in the crystal lattice. The trapping action creates a space charge, which then attracts a silver ion. This silver center traps further electrons and silver ions. When these clusters become large enough, the grain can be developed. These larger silver clumps are referred to as latent-image centers.

The formation of the latent image is enhanced as the number of electrons set free in the grain increases. For charged particles, the probability of a latent image being formed depends upon AgBr grain size and the specific energy loss of the particle. For photons and neutrons, the grain size and the energy transferred to the secondaries are key factors.

For grains of a given size, the response depends upon how stable the resulting silver clusters are. The stable centers can be developed. During this process, the latent-image centers are reduced to silver by the chemical action of the developer. The silver plates out as a black deposit on the film. The silver forms only at points where a stable latent-image center has been produced. The unchanged silver halide is dissolved from the film by use of sodium thiosulfate (fixer or hypo.)[6]

b. Film Density

The final effect is that the film becomes darkened. We can measure the degree of the darkening and relate it to the amount of radiation. If a beam of light (intensity I_0) falls on the darkened film, some of this light is transmitted (intensity I) and some is absorbed. The opacity O is then

$$O = I_0/I. \tag{13.1}$$

The film density D is defined by

$$D = \log_{10} O = \log_{10}(I_0/I). \tag{13.2}$$

With x ray film, the density produced on the film is related to the exposure. For fast neutrons, track counting is used to measure the response (Section 13.B.8.a).

3. Properties of Film Response

Assume that an x ray film is exposed to a photon beam. The photon action on that film produces a darkening. The degree of response is a function of the quantity of radiation. If film is exposed to more radiation, the film response is greater darkening. This can be seen on a curve that relates the film response to the exposure. A common method to present this is to plot the density versus log exposure. This type curve is often referred to as a <u>characteristic curve</u> (see Figure 13.2).

a. Characteristic Curve

The characteristic curve typical of many film types is shown by the solid line in Figure 13.2. It consists of three typical portions: the toe (A), a linear portion (B), and a shoulder (C).

In the toe portion of the curve, the film responds very slowly. That is, the density does not change very much with small changes in the exposure. This portion is of limited value in personnel monitoring since it is not easy to obtain accurate exposure readings. In region (B),

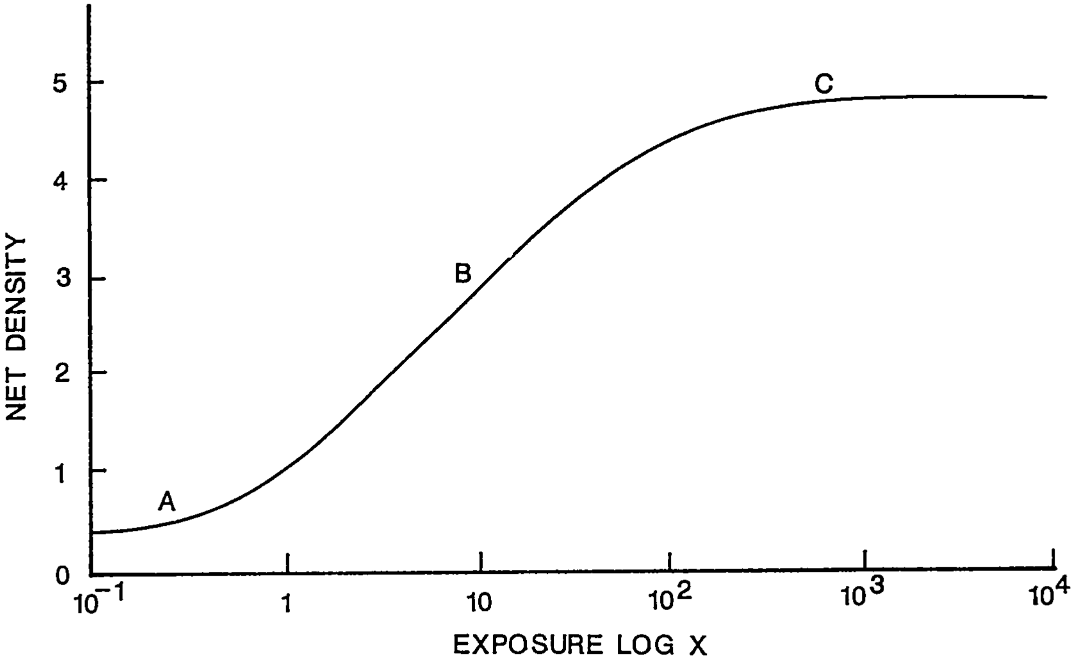

Figure 13.2 Characteristic curve for photon film. (Adapted from PERSONNEL DOSIMETRY TECHNIQUES FOR EXTERNAL RADIATION (Symposium) Madrid, OECD/ENEA (1963).[7])

the density change is very rapid for small changes in exposure. This is the region in which the film should be used to measure exposure. The accuracy is best in this region since two nearly equal exposures may produce densities that are well-separated. We are then able to pinpoint exposures better.

The useful range of a film in terms of exposure roughly corresponds to the extent of the linear portion. The <u>contrast</u> of the film is often taken to mean the slope of this straight portion. In general, we cannot have both a wide useful range and high contrast in a single film.

The shoulder portion (C) shows that a saturation density is reached. That is, large changes in the exposure produce little, if any, change in the density. Clearly, this is the least useful region in which to measure exposure.

b. General Features of the Characteristic Curve

The actual characteristic curve obtained is affected by the techniques of development. However, for a given process, the shape of the curve does not depend upon the energy of the photons. Also, if there is no latent-image fading, the shape of the curve is independent of the rate at which exposure is received. This aspect of film response is known as the reciprocity law.[6] This law fails if we are near the maximum density for a given film.[10]

Although the shape will be the same, the location of a curve along the log X axis is a function of the energy of the radiation. For a given film, the curves produced by two different energies may be displaced from each other. The shape of the curve produced by beta is the same as that for x or gamma rays,[11] but the location depends upon beta energy.

c. Latent-Image Fading

The fact that latent-image centers tend to fade with time, limits the time a badge may be worn. Under normal conditions, such as room temperature or below, and low relative humidity, the fading is slight for x ray films. Neutron films, because of their fine grains, continue to fade unless protected from humidity. Even with photon films, the humidity seems to be the vital factor in latent-image fading.[12]

d. Sensitivity (Response)

We can measure the sensitivity of a photon film in terms of the reciprocal of the exposure needed to produce a given density. Films are available in a wide range of sensitivities from a number of companies.[6]

For most film types, the sensitivity, or response, below 100 keV may be > 30 times that at 600 keV. This is due to increased absorption of photons of low energy in the film over that in air.[1] For the region above 600 keV, the absorption in air becomes proportional to the absorption in the emulsion so that the response remains fairly constant. This aspect of film response is referred to as <u>energy dependence</u>.

e. Energy Dependence for Unfiltered Film

Suppose several films of the same type are given an equal exposure, but the photon energy is varied. If we plot the densities obtained from the films against the photon energy, the solid curve in Figure 13.3 results. This type of curve is obtained for unfiltered film. It shows that for the same exposure, the density produced by photons below 0.3 MeV is much higher, reaches a peak value for a certain energy and then decreases. For the region below 0.3 MeV, the density depends on the energy of the photon. This energy dependence of the film hinders proper interpretation of the film response. In most cases, the exposure is exaggerated so that readings are on the safe side. That is, we read a higher density as a greater exposure. For increased exposure, the density values increase proportionately, but the peak remains at the same value of photon energy. This peak occurs because of the silver halide in the emulsion.

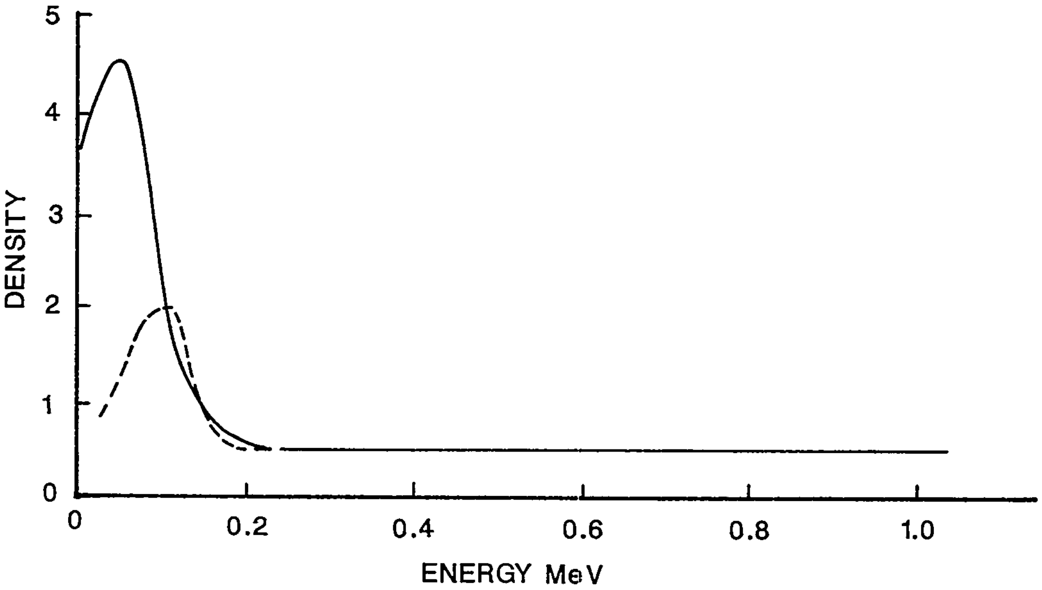

Figure 13.3 Energy dependence in typical photon film, with and without a filter. (Adapted from NBS Handbook 57)

That is, the response curve reflects the absorption of energy in the silver bromide. However, the exposure X reflects the energy absorbed from photons in air. The peak should occur roughly at the energy for which

$$\frac{\left(\frac{\mu_{en}}{\rho}\right)_{AgBr}}{\left(\frac{\mu_{en}}{\rho}\right)_{air}} = \text{maximum value.}$$

Although the K-absorption edge of silver is at 22 keV, the peak in the response curve occurs at about 45 keV because of the above ratio.[7] The film absorbs a larger fraction of the photons at this energy than at any other which results in the increased darkening.

In some instances, the photon energy is above 0.3 MeV and well known. This presents little, if any, problem in using films. In other cases, we must use film to monitor a wide energy range, which extends below 0.3 MeV. When this is the case, we need a means to correct for the effects below 0.3 MeV.

f. Energy Dependence for Filtered Film

When we place a metal or plastic filter in front of the film, the filter absorbs some of the photons before they reach the film. These filters are thin enough to not greatly affect the photons in the region above 0.3 MeV. They are thick enough, though, to affect those below 0.3 MeV. Filters tend to flatten the response in the region below 0.3 MeV. This is shown by the dotted portion in Figure 13.3. General features produced by the use of a filter are: (1) The peak response is reduced, or flattened; (2) peak location may shift; and (3) the uniform response may extend down to 200 keV, or even below.

An ideal filter would make the response the same no matter what the photon energy. For most combinations of filter and film, we cannot readily obtain uniform response below 30 keV. However, above 200 keV, this is obtainable.

The response of filtered film depends upon: (1) photon energy; (2) filter thickness; (3) the atomic number of the filter; (4) the filter K-absorption edge; and (5) filter area. For these reasons, different filter materials are used. Some common substances used include Cd, Sn, W, Pb, Ag, Cu, and Al. Each produces its own preferential absorption of the softer radiation. The substance used, and its thickness, depend upon the information we are after. For simple applications, a single filter may be all that we desire. When we would like to know something about the energy spectrum, multishield systems are employed. In this case, the density ratios behind filters are compared. Because of the differences in Z number and/or thickness of the filters, the ratios can provide a crude indication of the energy.[1,6]

One of the more common filter substances in use is cadmium. A 1 mm cadmium filter-film pair allows us to measure the exposure to within about ± 20% for energies $\geq$ 100 keV.[13] A single shield composed of many elements has also been reported.[14]

4. Reading and Interpretation

If a film is processed without being exposed, it will show a slight density. This density is known as the base fog. The value of the base fog is a function of the type of emulsion, film age, the manufacturing processes, and the film storage conditions and development time.

To correct for the base-fog effect, control films are processed with the monitoring films. The control films should be from the same batch as the other films and stored under the same conditions. Then, if all are processed in the same way, the base fog on the control films spotlights extraneous effects as well as normal fog. The base-fog values can then be subtracted from the density readings of the monitoring films to give corrected values.

a. Calibrations Films

Once the film density is obtained, it must still be related to the exposure. That is, the density must be calibrated against known exposures, since films are not absolute devices. A simple means of attaining this end is to prepare and process a set of calibration films with each batch of film. This accounts for: (1) the variation of response because of energy, and (2) the variation in density because of processing factors.

Films may be calibrated for beta by the use of a thick piece of uranium metal. This acts as a diffuse source of beta incident on the film. The surface dose rate from uranium can be measured with an extrapolation chamber.

b. Interpretation

The interpretation of the readings in terms of the exposure often depends upon the number of filters used in the badge. In some units, this number may be as much as five. Besides the filters, most units have an "open-window" section. That is, part of the film is not covered by a filter. The window section records the response due to a mixture of both hard and soft radiation. The softer radiation affects the window section more than the hard radiation does.

If only hard photons (> 200 keV) are involved, there is no problem, since only the bare film would be needed. For a pure photon field, without regard to spectrum, we can use just a single filter. This filter should be designed to match the response profile of the film used.[7] If the radiation field consists of a mixture of beta and gamma of unknown energy spectra, then we need a multishield device.

For a multifilter badge, the densities behind the filters reflect somewhat the character of the transmitted radiation. This enables us to gain some knowledge of the energies involved from density differences beneath pairs of filters. By proper use of calibration charts, we can then interpret the readings in terms of the indicated spectrum.

The window sections presents more of a problem. Since the response is not at all uniform, detailed spectrum information is needed. However, this is impossible in most cases; therefore, we must use whatever knowledge we can glean from the filtered response. By use of this knowledge, we can partially evaluate the reading in the open-window section. As yet, there is no really satisfactory way of assessing exposure from the window section of the badge. All present methods suffer in that they fail to properly identify the response due to β.[15]

5. Filter Holders

The film holder is often in the form of a clip-on badge, in which the shields are fastened to the inside surfaces. That is, a shield is both in front of and behind the film when it is inserted into the badge. The number and type of shields used vary greatly throughout the country.

Some badges have been designed to be "tamperproof" by the use of a locking device, which is opened by magnets. Others serve as identification badges as well as film holders. Film dosimeters in the form of wrist badges and/or rings have also been used. Badges with a number of devices to detect a nuclear criticality are also in use. Typically, these contain a number of foils or other detector materials to detect both fast and thermal neutrons. A high dose γ detector is also included. Not all badges are as elaborate as described. For facilities at which a criticality incident might occur, nearly all have at least an indium foil. This allows for quick sorting of exposed persons following such an incident.

6. Film Packets

Many films used for monitoring are similar to medical x-ray film and are contained in dental-size packets. Usually no one film covers the range of exposure needed for personnel monitoring; therefore, most packets contain at least two films. For example, the Kodak type 3 packet contains a double-coated film (high sensitivity) and a single-coated film (low sensitivity). These allow coverage over the range 4 μC/kg (approximately 15 mR) to 0.46 C/kg (approximately 1800 R). Another design, Kodak Type 2, is a single film with a fast emulsion on one side and slow on the other and covering about the same exposure range.

The packet material is often an opaque wrapper of about 0.2-0.3 kg/m^2 (20-30 mg/cm^2) in thickness. This thickness is enough to stop all beta below about 150 keV. For this reason, low-energy beta monitoring is impossible by this method.[7]

7. Electronic Equilibrium Conditions

The film and its wrapper constitute an equilibrium thickness up to about 0.3 MeV. For photons above 0.3 MeV, the range of secondaries becomes greater than the thickness of film plus wrapper. This results in a loss of electronic equilibrium since we cannot assume that for each electron leaving a small volume, one of about equal energy enters. To maintain electronic equilibrium, the packet must be surrounded by a substance thicker than the range of the secondaries. Many badges employ plastic as the equilibrium layer.

The thickness of the layer needed is a function of the highest photon energy expected. It must be thicker than the range of the highest-energy secondary produced by the photons. This will insure the electron density measured by the film response is proportional to the photon intensity at some point in the substance. But it also must not be so thick that it greatly attenuates the photon beam, for then the film does not record the maximum exposure, since this occurs at a depth in the layer about equal to the average range of the secondaries. Since the range depends upon this layer substance, the thickness needed is also a function of the material used. A graph of equilibrium thickness versus photon energy is contained in Reference 16.

The preceding pages have pointed out some of the vital factors we must treat in order to use film as a dosimeter. Despite the presence of these pitfalls, a detailed study[17] points out that we can obtain good results with proper care. On the other hand, dosimetry methods other than film have offered a number of attractive features.[18]

Much of what has been already said has dealt with photon films. The next few pages discuss neutron monitoring by film techniques.

8. Neutron Dosimetry

As in the case of instrument design, neutron monitoring by film presents more problems than monitoring for other radiations. The response of film to neutrons depends upon the neutron interactions with the atoms of the emulsion. These interactions may result in charged particles and/or photons being emitted. This occurs when elastic scattering and/or capture take place. As in the case of photon film, one must be concerned with assuring that measurements are made under charged particle equilibrium conditions.

The capture of thermal neutrons in silver and bromine leads to the release of beta and gamma so that neutron response can be read as film density.[19] Also, elastic collisions between fast neutrons and hydrogen in the emulsion and surrounding lead to recoil protons. These protons in turn lose their energy to the grains, which result in the film density.

Neutrons interactions that lead to film density permit quick and easy evaluation of the response. Because photons also produce the same effect, it is hard to determine only the neutron effect when photons are present.[6] For this reason, fast-neutron monitoring by film is often done by track counting methods.

a. Fast Neutrons

(1) Track Counting - Track counting of fast neutron response depends upon the hydrogen in the emulsion and in the surrounding substances. Elastic scattering of fast neutrons with hydrogen produces recoil protons. These protons lose energy to the grains, which are much finer than in x ray films. The emulsion contains more of these AgBr grains than do photon films. The latent images formed consist of a number of individual grains along the path of the proton. Then, when the film is developed, the track of the proton can be seen by means of a microscope.

In this country, the most preferred track emulsion film is the Kodak Personal Neutron Monitoring, Type A. This type of emulsion is usable in the energy range 0.5 to 14 MeV, in the proper kind of packet.[20] This is the range in which the film response is proportional to the neutron dose. The lower limit of this range results from the nature of the effect. That is, to count as a track, at least three grains must be developed. This requires a proton with energy of about 0.25 MeV.[1] Since, on the average, a neutron transfers about half its energy to a proton in an elastic collision, this sets the lower limit. The upper limit is imposed by the emulsion thickness. That is, the practical limit of the emulsion thickness sets the high-energy end of the proton energy loss that can be detected.

(2) Sensitivity - The sensitivity of the Type A film varies, being on the order of one track per 1500 to 3000 incident neutrons. This seems to be a function of the neutron spectrum.[21] The use of these emulsions is also limited by the response to gamma. The response is a fogging effect, which makes it quite difficult to count tracks. Thus, a gamma field that produces an exposure of about 5 mC/kg (approximately 20 roentgens) renders this type of film unreadable for neutrons.[6]

Since thermal neutrons also produce recoil protons when capture occurs in nitrogen, the total response may contain tracks produced by these protons. However, the sensitivity to thermal neutrons is about 25 to 30 times lower than that of fast neutrons. Under normal circumstances, the thermal-neutron effect is small. At any rate, the tracks counted from this source lead to a safety factor in the evaluation of the fast-neutron response. We can also correct for this effect in multishield badges that contain cadmium filters. A 40-mil-cadmium filter (approximately 1 mm) absorbs the thermal flux so that proton-recoil tracks behind this shield should be due only to fast neutrons.

(3) Latent-image Fading - For photon films, the fading is slight and occurs mostly in a short period of time. For track film, the fading is extreme and seems to progress with time.[22] We can protect the film and reduce this effect by sealing the film in humidity-proof envelopes.

(4) Reading and Interpretation - The tracks are often counted by means of a dark-field microscope. In this device, the background of the field of view appears dark, so the tracks are seen as a series of white dots. The method consists of simply counting the number of tracks seen in a field of view. The field in this case is only a small portion of the film.

The number of tracks in any one field is small and varies from field to field. The fields to be viewed during counting should be chosen at random. In this way, the person counting the tracks is less likely to influence the result. The number of fields that must be counted to obtain consistent results is a function of field size. In this country, most counting programs require from 25 to 40 fields for good statistics. For this reason, the counting of a large number of films may be quite tedious.

As for photon film, the use of calibration films that are processed along with the monitoring film allows us to interpret the unknown response. That is, film exposed to a known fluence of fast neutrons allows us to measure the expected number of tracks in a given number of counted fields. Thus, we can relate the response of a monitoring film to that of the calibration film. The fluence derived can then be evaluated in terms of a dose equivalent based on the curves of Snyder.[23]

For the range of neutron energy between 0.5 and 14 MeV, the number of proton-recoil tracks remains independent of the neutron energy. This assumes charged particle equilibrium conditions. In view of this, we can evaluate the fast-neutron dose equivalent fairly well, since the quality factor is almost constant in this range. However, for an intermediate spectrum (between thermal and 0.5 MeV), this method fails to give any response. Thus, film dosimetry is not able to cover the entire range of neutron energy desired for health physics operations.

b. Thermal Neutrons

In the discussion of the fast neutron methods, we mentioned that thermal neutrons may also be monitored by density and/or track-counting methods. That is, a shield of cadmium, silver, indium, or rhodium over a photon film may be used for density readings. The capture of thermal neutrons leads to beta and gamma being emitted, which produces the density on film.

For a dense photon field, it is difficult to determine the portion of density due to thermal neutrons. Now, if we have both a high-capture cross-section substance (cadmium) and a low one (tin), it allows estimation of the thermal-neutron response.[19] The density under the cadmium is caused by both capture gammas and the other photons present. The density under the other substance results mainly from the photons not due to capture.

Although track counting may be used for thermal neutrons, the sensitivity is low for recoil protons from nitrogen. The response may be increased by loading the emulsion with a good thermal-neutron absorber.[13] Lithium (tritium recoils) and boron (alpha recoils) are two of the common loading substances. These substances may increase the sensitivity by more than an order of magnitude.

In badges that contain either cadmium or rhodium, we can use differential track counting to measure the thermal response. Thus, recoil tracks under these shields are a measure of the fast neutron response. The tracks in an open-window section are a measure of thermal plus fast neutron response. Then, the difference allows us to obtain the thermal response alone.

C. Pocket Dosimeters

The pocket dosimeter, often used in conjunction with a film badge or other primary monitor, is a small electroscope, about the size and shape of a fountain pen. This device can be either self-reading or not. In the self-reading type, a small compound microscope allows us to observe the response. In the other type, the response is read on a separate device.

The detection system usually consists of a small quartz-fiber electroscope, which forms a part of the collecting electrode. The instrument case, which is insulated from the fiber system, serves as the other electrode. The collecting volume is small (approximately $2\text{x}10^{-6}$ m^3) and most often contains air at ambient pressure.

1. Principle of Operation

A positive charge may be placed on the electrode and fiber by means of a separate charger. The leakage current must be kept low; for this reason, a magnetic switch is often built into the system. That is, contact between the electrode and the charger cannot be made unless a magnetic field is used.

The fiber is repelled by the electrode since they both acquire a like charge. The image of the fiber can be viewed by means of the microscope. A scale in the eyepiece of the microscope can be calibrated, so that exposure may be measured in terms of image movement. The light for viewing enters through a window either on the side of, or at the end of, the device. By adjusting the voltage on the charger, we can bring the image to zero-scale reading.

When ions are produced in the volume of the chamber, they move to the fiber and electrode. This neutralizes some of the charge on the system, and the fiber moves closer to the electrode. The amount of charge lost depends upon the number of ions formed in the volume. If the agent causing the ionization is radiation, then the movement of the fiber image is a measure of the amount of radiation. For photons, the scale divisions can then be related to the exposure.

2. Exposure Measurement

The system may be viewed as a capacitor of C farads charged to a voltage V. Then, the charge on the system is given by equation 1.16

$$Q = CV, \text{ in coulombs.}$$

The value of the charge put on the device is a function of the design. A typical value for the capacitance is around 3 pF, and the charging voltage may be 180 volts. In this case, the total charge that would result is then $Q = (3\text{x}10^{-12})(180) = 5.4\text{x}10^{-10}$ coulombs.

The ions produced in the chamber volume are then collected by the electrode, and the system loses some of its charge. This results in a drop in the voltage of the system. This voltage change causes the fiber to deflect, and the fiber image moves along the scale in the eyepiece. The exposure can be measured in terms of this voltage change in the system.

The exposure, X, is defined in terms of the charge collected in a volume of air of known mass; that is, adapting equation 4.3,

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

in which Q is the coulombs, and m is in kg. Therefore, for the pocket dosimeter,

$$X = \frac{Q}{m} = \frac{C\Delta V}{\rho \text{Vol}}(\text{C/kg}); \qquad 13.3$$

where ΔV is the drop in voltage, ρ is the density of air in kg/m^3, and the volume, Vol, of the chamber is in m^3. When we replace ρ by its value of 1.293 kg/m^3, the result is

$$X = \frac{0.773C\Delta V}{\rho Vol}(C/kg) \qquad 13.4$$

Since the density of air is a function of both temperature and pressure, a correction term is often needed to account for this variation.

The range of the device is then a function of the parameters in equation 13.4. Suppose we charge a dosimeter of sensitive volume 2x 10^{-6} m^3 and capacitance 3 pF to 180 volts. What will be the voltage change if the device is expected to read an exposure of 26 μC/kg (approximately 0.1 R) full scale? The require voltage change would be

$$X = \frac{0.773\ C\Delta V}{Vol}\ (C/kg)$$

$$2.6x10^{-5}\ C/kg = \frac{0.773\ (3x10^{-12})\Delta V}{2x10^{-6}},$$

and

$$\Delta V = \frac{5.20x10^{-11}}{2.32x10^{-12}} = 22.4\ volts$$

The scale of the device can then be designed so that, for the above voltage change, the fiber image moves from 0 to 26 μC/kg (0.1 R). The scale is so arranged that when the fiber is fully charged its image is on zero.

3. Characteristics

Pocket dosimeters come in many ranges, such as 26 and 52 μC/kg (200 mR), as well as .26 - 12.9 mC/kg (1-50 R) full scale. The response of a dosimeter is seldom linear, except in the region of the calibrated scale. For this reason, we should not attempt to estimate readings if the device shows more than a full-scale reading. Also, when reading a dosimeter, we should keep the fiber image vertical. This reduces the geotropic effect, that is, the tendency to give a reading that depends on the orientation of the device.

These devices are subject to discharge when dropped or bumped against an object. Most of these units are put through a "drop" test before they are put into routine use. Since any leakage of charge produces a reading, good insulation of the electrode is needed. When exposed to high-humidity areas, dosimeters may suffer a breakdown of the insulation. The normal leakage rate of a good dosimeter should be less than 3% in a 48 h period. More details concerning the desirable performance specifications for dosimeters may be found in Reference 24.

4. Energy Dependence

Most pocket dosimeters have walls of aluminum, Bakelite, or a plastic substance. The inner surface may then be coated with Aquadag to insure a conducting surface. These devices show a high degree of energy dependence for photons below 300 keV. Above this value, the response can be accurate to within ± 10% of the true exposure. Below 300 keV, the error sometimes may be as high as a factor of 2 to 3.

Because of the case thickness used in most of these devices, the β response is often poor. Most of these units are highly directional in their β response and do not respond to β below 1 MeV.

Some of these dosimeters may have an inside coating of ^{10}B so that they can be used as thermal-neutron dosimeters.[25]

In many cases, the purpose of the pocket dosimeter is merely to supplement the film badge, not to replace it. For this reason, the device is widely used, although it has several disadvantages. The main feature in its favor is the direct reading capability. Thus, we can check the unit at frequent intervals and get at least a rough idea of the exposure received. This will, at times, allow us to better ration our work time in the radiation field. In addition, when one needs to distribute the exposure in a given operation, the direct reading capability can be invaluable. Also, in some applications in which radiation pulses are of importance, direct reading capability allows one to get a relative indication of the variation in radiation fields.

D. Solid State Dosimeters

As a result of irradiation, some solid substances undergo changes in some of their physical properties. These changes amount to a storage of energy in some way, and this forms the basis of using these solids for dosimetry. These main features have been studied: optical density changes, radiophotoluminescence, thermoluminescence, and conductivity changes. Not all of these methods are suitable for personnel dosimetry.[2,5,26]

In studies of optical-density changes, glasses and plastics are most often used. Radiation induces color changes in these substances, and the solid can then absorb light in spectral regions that were transparent to the normal substance. When light of a given wavelength is passed through the solid, the optical density can then be measured on a spectrophotometer. In glasses, the dose range for linear response extends from about 10 to 10^4 Gy (10^3 - 10^6 rad). This range for plastics is between about 10^4 and 10^7 Gy (10^6 - 10^9 rad) for most substances.[5] The high ranges in this method preclude its use for personnel dosimetry.

Before irradiation, some substances fluoresce when light is passed through them. When these solids are exposed, the result may be a loss of fluorescence.[27] Anthracene, naphthalene, and other organic solids are among the substances used in this method. This method seems to be better than the color change effect for high-level dosimetry. Since the range for this method is from about 10^3 to 10^6 Gy (10^5 - 10^8 rad), it is also not suited for personnel dosimetry.

1. Thermoluminescence

Phosphors that were originally used the most in studies of thermoluminescence include manganese-activated calcium fluoride (CaF_2:Mn) and lithium fluoride (LiF).[28] Other substances, such as

$CaSO_4$:D_4, LiB_4O_7:Mn,Si, CaF_2:Dy, and BeO, etc., have also been used.[5,28-30] In these substances, electrons are moved from their normal places when the solid is irradiated. They migrate about until "trapped" by lattice defects in the solid. At normal temperature, the electrons remain there for quite some time, but are released from the traps by heating. The luminescence appears when the electrons return to their normal positions. This light (blue-green) for LiF, CaF_2:Mn can be measured and related to the absorbed dose in the phosphor. This method requires a means of heating the phosphor as well as a system to measure the light output.

a. Lattice Defects or Traps

Solids that have a crystal structure contain many kinds of lattice defects (activator centers, see Figure 10.7). Some of these defects may be: missing atoms or ions, dislocations in the crystal plane structure, interstitials (atoms or ions at interlattice positions), or even the presence of foreign substances (impurities).[27] The regions about these defects are not always able to maintain a state of neutral charge. The presence of any charge at the activator center attracts unlike charges. If electrons move into these regions and a positive center is there, the electrons are attracted to this charge. Also, positive ions are attracted to those regions in which there is a negative center about the defect. When the charge centers are strong enough, they can even bind the ions that are attracted to them. These stronger centers are referred to as "traps," and the strength of the binding is called the "trap depth." The trap depth depends upon the type of defect (trap); a solid may contain many kinds of traps, each with its own trap depth.

To be of use as a dosimeter, the thermoluminescent (TL) material must be able to retain the trapped electrons at room temperature and must be capable of a strong light output when heated. If the traps are near the edge of the band gap, the number of carriers which can be trapped will be large. This will mean that the material will have good detection sensitivity, but, on the other hand, may exhibit fading.[31] $CaSO_4$:Mn is such a material. Its response goes down to 0.2 μGy (20 μrad) but shows fading. LiF and CaF_2:Mn have deeper traps so that fading is less but these are not as sensitive.

Another problem of TL material used for environmental monitoring applications is the presence of radioactive impurities in the solid. This contamination from trace materials can be significant.[32]

b. Glow Curve

When a solid is exposed to radiation, both electrons and positive ions ("holes") are produced. When some of these are trapped at the defects, they remain bound there as stored energy until they are freed by some means. The energy needed to free these charges is supplied by heating the solid. When the charges are released from the traps, they recombine with unlike charges in other parts of the solid, and light is given off. The light output may be measured by means of a photomultiplier tube.[27] A plot of the brightness (relative intensity) versus temperature or time as the solid is heated is called a glow curve (see Figure 13.4). If the heating rate is uniform, both curves are similar. However, the total light emitted as a function of time (area under the glow curve) can be related to the total thermoluminescence.[28] The area under the curve as a function of temperature does not have a simple interpretation. As the solid is warmed, the electrons begin to escape from the more shallow traps. When the store of these traps is depleted, the brightness decreases, and this gives rise to the first peak in Figure 13.4. If there are no other type traps in the solid, this would be the only peak. More often, more than one peak shows up. The number of such peaks is a function of the number, as well as the concentration, of different types of traps. The area under a glow peak, called the light sum, is constant for the same solid exposed to a given dose. However, many factors can affect the shape of the glow curve.[28] Changes in heating rate produce dramatic effects. For very fast heating, the peak height increases but the time in which light is given off decreases, and the temperature at which the peak occurs shifts to a higher value. For TL dosimetry, one requires a reproducible heating rate, but it need not be uniform.

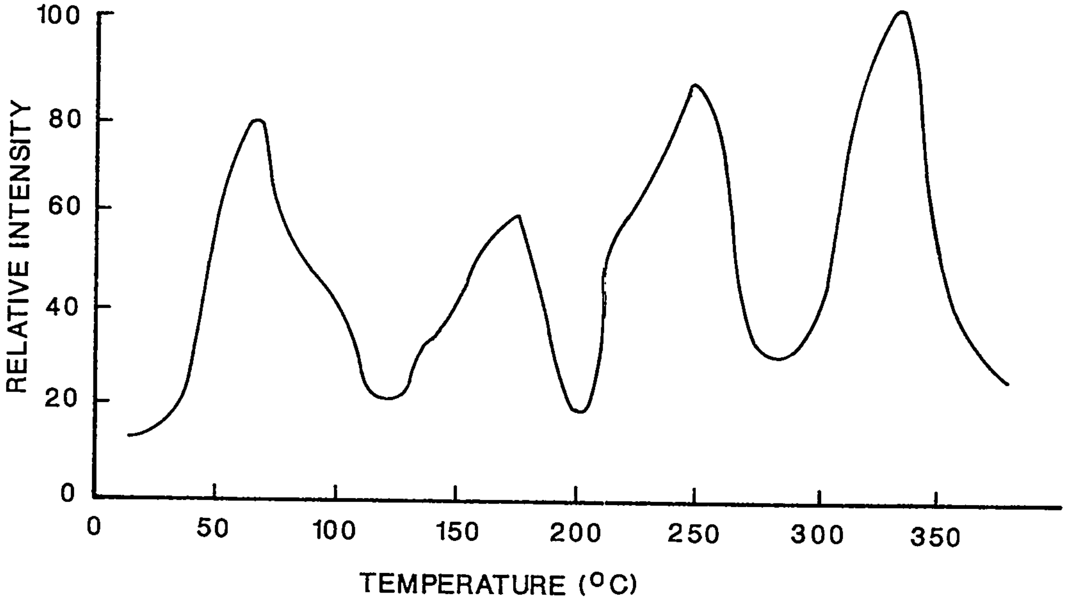

Figure 13.4 Thermoluminescent glow curve. (Adapted from PERSONNEL DOSIMETRY TECHNIQUES FOR EXTERNAL RADIATION (Symposium) Madrid, OECD/ENEA(1963).[27])

Following conversion of the light output into current, the readout may be related to dose. Two approaches are used: the peak height is used or the integrated area under the glow curve is used.[28] The peak height method is more dependent on the heating rate and should not be used for loose powder. The area under the glow curve, during the heating time, is directly proportional to the integrated current from the photomultiplier. This latter method is less dependent on the heating rate.

Phosphors which have glow peaks at too low a temperature tend to exhibit fading (loss of the stored energy) with time. If the glow peaks have too high a temperature, infrared radiation will be produced which interferes with the reading. If the glow peaks are ill-defined, then integration becomes difficult with respect to choice of end point. An ideal glow curve would contain a single TL peak at a temperature of about 200°C.[28]

c. Characteristics

The response to γ is linear in the exposure range from about 1 μC/kg (approximately 4 mR) to greater than 0.26 C/kg (approximately 10^3 mR); therefore, these phosphors can be used for personnel dosimetry. Above this energy range, some TL materials exhibit a non-linear response. This is called "supralinearity," but the effect can still be accounted for. The response to fast neutrons is low in most of these substances. The thermal neutron response of CaF_2:Mn units is about one-fourth that of its γ response. The neutron response of LiF is better because of the presence of ^{6}Li. That is, natural Li contains about 7.5% ^{6}Li. Because of the presence of ^{6}Li in LiF, the thermal neutron response is about 1/7 that of the γ response on an energy absorbed basis. If we use "almost" pure ^{7}Li in the unit, the response to thermal neutrons is negligible. Thus, we can use a pair of Tl chips, one with almost all ^{7}Li (called TLD-700 with 99.99% ^{7}Li) and one with mostly ^{6}Li (called TLD-600 with 95.6% ^{6}Li), to arrive at both γ and the thermal-neutron doses.[28] The TLD-700 does not respond to neutrons, so that one can obtain the γ contribution from this chip. The TLD-600 responds to both γ and thermal neutrons, so the difference in readings between TLD-700 and TLD-600 can be used to estimate the thermal neutron contribution. Normal LiF (available as TLD-100)[28] is used extensively for photon dosimetry applications. However, this phosphor may also be used in conjunction with TLD-700 to estimate the thermal contribution.

Characteristics of other TL phosphors are discussed in the literature.[5,28,30]

(1) Energy Dependence - The energy dependence is related to the type of phosphor used. Cameron[28] reports that LiF is energy-independent in the range 30 keV to 1.33 MeV, within 25%. CaF_2:Mn is quite energy-dependent. Because of its high effective atomic number (Z), the response per unit exposure is approximately 13 times greater at 40 keV than at ^{60}Co energy. To combat this effect, filters have been used.[27] However, the response to soft x rays is then altered because of the attenuation in the filter.

On the other hand, lithium borate with a manganese activator, is relatively energy independent and has an effective Z which is about equal to air or tissue. Also, BeO, which has an effective Z close to tissue, has a sensitivity comparable to LiF. However, BeO has not found wide acceptance as a TL material.[33]

(2) Fading - These phosphors suffer from fading.[2] If we use the deeper traps in the solid, the fading is less severe.[28] When a substance is treated, or doped, with the proper impurity, we can obtain a phosphor with only one kind of trap. For those substances with glow peaks around about 200°C, the fading becomes much less severe. If we measure the main peak in LiF, we find that the stored signal does not fade severely at room temperature.

The LiF units are not affected by atmosphere as are the CaF_2:Mn substances, which require careful packaging. However, LiF seems to be more prone to damage effects than CaF_2:Mn.[34] Both substances can be made into sufficiently small units for personnel dosimetry purposes.

(3) Charged Particle Equilibrium - As was the case for photographic detectors, the dosimetric information which is obtained must be under conditions of charged particle equilibrium. This requires that the phosphor be covered by a sufficient thickness of material. This thickness is often referred to as the buildup region. Values of the required thickness for photons of various energies can be found in Reference 28. Without this material fronting the TL material, the dose will be somewhat underestimated. Plastics such as Lucite, Teflon or Nylon are often used to supply the buildup layer.

(4) Applications - Thermoluminescence was first applied to dosimetry in 1953.[35] At that time, it was not fully known if the method could be extended to personnel monitoring. Since then, results have shown that the method can be used for this purpose; yet progress has not been spectacular. Nevertheless, TL has been used for β, γ and neutron dosimetry.[28] Because of the small size of TLD chips, they have been useful in determinations of extremity doses, utilizing finger ring

badges for this purpose. One drawback in the past has been the practice in the U.S. to hold film badges indefinitely for potential recheck. As we have seen, to measure the dose in a TL phosphor, we must remove the electrons from the stored-energy centers. That is, we deplete the traps of the trapped electrons and holes, so that the dosimetric information is then gone. This precludes the option of a recheck.[2] Nevertheless, this is no longer seen as a drawback and increased use of thermoluminescent dosimeters (TLD) is occurring. This has been accelerated in recent years by the availability of automated reading systems.

2. Albedo Dosimeters

An extension of the use of TL absorbers has been in the use of these for neutron dosimetry in albedo dosimeters. The concept albedo refers to reflection from a surface. These devices are designed to detect the thermal neutrons reflected from the body surface when an incident fast neutron fluence interacts with the body. One of the early applications of this method for dosimetry was described by Hoy.[36]

The detection of thermal neutrons is generally based on the reaction

$$^{6}_{3}\mathrm{Li} + ^{1}_{0}\mathrm{n} \rightarrow ^{7}_{3}\mathrm{Li}^{*} \rightarrow ^{4}_{2}\mathrm{He} + ^{3}_{1}\mathrm{H}.$$

The energetic alpha (4.8 MeV) and the recoil triton lose their energy in the TL material, which is generally LiF. In a typical application, TLD-700 and TLD-600 are used although TLD-100 may be used in place of TLD-600. The γ response of these phosphors are similar. If one uses the pair, say TLD-700 and TLD-600, the response of the TLD-700 will represent only the γ contribution. The TLD-600, with increased ^{6}Li content, gives the total γ plus neutron response. By subtracting, one can estimate the contribution due to neutrons. To account for any thermal neutrons which are incident on the body, cadmium (approximately .76 mm thick) or boron-loaded plastic are placed over the phosphor.[37] That is, the desired response is from the fast neutrons which are incident on the body. When these interact, are thermalized, and reflected from the body, this represents the contribution we wish to measure. If the incident fluence contains thermal neutrons to begin with, their response in the phosphor would introduce an erroneous contribution. The cadmium, or boron-loaded plastic, absorbs the incident thermal neutron fluence so that it does not produce an undesirable response.

a. Energy Dependence

The major disadvantage of albedo dosimeters is their severe energy dependence. For this reason, the reading can only be evaluated correctly if the proper calibration factor has been obtained. Two methods of determining these factors are in use.[37] The relevant dose equivalent rate can be determined at the particular work place. This can be related to the response of the albedo dosimeter, placed on a phantom (usually a jug of water), at the same location for a measured time interval. By dividing the dosimeter response by the computed dose equivalent, the calibration factor may be determined. The other method was developed by Hankins.[38] It consists of using a 0.23 m diameter spherical rem meter and a 0.076 m diameter sphere (covered with .25 mm of Cd). Measurements with each sphere are taken at the relevant location, and the ratio of the readings is determined. The dose equivalent at the location can then be related to the ratio that was obtained with the spheres. Then, for a given measured ratio, the calibration factor can be determined. A plot of these ratios versus the computed calibration factor yields a straight line. So, in order to correctly interpret the albedo response, one must also have evaluated the neutron field at the specified location.

b. Dosimetric Features

Albedo dosimeters are quite sensitive and are attractive since the processing may be automated. They overrespond to moderated neutrons and have a poor response to fast neutrons. Because of this, some facilities employ albedo dosimeters to measure the moderated neutron component and NTA film to read the fast component. Although the response may be difficult to interpret on occasion, a virtue of albedo dosimeters is that a reading will always be obtained when the device is exposed to a significant neutron dose.[37] These systems are capable of detecting about .5 mSv (50 mrem) of unmoderated fission neutrons. Albedo response is influenced by high γ fields. Since each phosphor responds comparably to the photon field, one may end up subtracting two large numbers to estimate the neutron contribution. Since there is an error in the measurement of each phosphor, the difference between two large numbers will give a small number with a relatively large error. The response of the albedo dosimeter to neutrons may be improved by adding polyethylene, which moderates the fast neutrons and thereby improves the sensitivity of the device to high-energy neutrons.

3. Radiophotoluminescence

a. General

As a result of the type of center used in the measurement, thermoluminescence results in the loss of the stored information during readout. The radiophotoluminescent (RPL) method is not subject to this feature; therefore, RPL units may be reread. This method has been used with success for silver-activated phosphate glass.[39]

Studies[5] with this type glass have revealed that at least two types of stable centers are induced by photon irradiation. One type gives rise to the color effect used in the optical density method. The other type also absorbs light, but will emit an orange fluorescent light when the dose glass is illuminated by ultraviolet light.[27] The intensity of the fluorescence gives a measure of the absorbed dose for the type of radiation used. The light output is measured by a fluorimeter.

b. Characteristics of Silver-activated Glass

In the type glass first used by Schulman, the linear dose range extends from about 0.1 Gy to greater than 10 Gy (10 - 10^3 rad). This Schulman glass was often referred to as high-Z glass[2] because the effective atomic number is about 28. The glass was designed for use mainly as a casualty-type dosimeter.[4] In that state of limited sensitivity, the glass was not of use for routine personnel monitoring.

Yokota[40] devised a new low-Z glass as well as a new fluorimeter. The effective Z of this glass was about 16 and was obtained by replacing the KPO_3 in the Schulman glass with $LiPO_3$. The linear response was retained, and the sensitivity to lower photon doses was greatly increased. The reported[40] γ sensitivity of less than 0.5 mGy (50 mrad) made this type glass of interest in personnel monitoring. Becker and Cheka[41] developed a silver-activated glass utilizing lithium borate, instead of metaphosphate, which had an energy dependence comparable to that of LiF, but which required excessive annealing.

The fast-neutron response is small in RPL devices, being perhaps a few % of the γ response.[4] The thermal-neutron response is greater for the low-Z than the high-Z glass. The magnitude of the response in the glass depends upon the presence of thermal-neutron absorbers such as ^6Li and ^{10}B. We can alter the thermal-neutron sensitivity for the low-Z glass by using pure ^7Li.

One of the main drawbacks of RPL devices is the severe energy dependence of the glass. Because of the presence of the silver, the response peaks near 50 keV, just as it does for film. For the high-Z glass, the response at the peak may be 20-30 times the response at ^{60}Co energy. The peak response may be flattened by the use of metal filters, as is done with film. Silver, tin, aluminum, gold, and cadmium have been used as filters with these glasses. Better response is achieved if these filters are perforated. Yokota[40] describes a number of filter arrangements for decreasing energy dependence.

The glass of either high- or low-Z type is most often used in the form of small cylindrical rods or small blocks. The glass rods are about 1 mm in OD by 6 mm long. The blocks, such as the Yokota type, are about 8x8x4.7 mm. Fluorimeters, although rather expensive,[2] are commercially available for reading these devices.

The advances in design, which have improved the performance of this type device, have shown promise of its further use in personnel dosimetry. At present, RPL devices have not received wide acceptance for this purpose in the U.S. However, these are employed as routine dosimeters in Europe and Japan and have achieved good accuracy of dose measurement, stability, and energy and directional independence.[4,43]

4. Conductivity Methods

Little has been done with the class of solids known as semiconductors in regard to personnel dosimetry. One reason for this is the low sensitivity which precludes dose measurement below about 0.1 Gy (10 rad). However, CdS and Si crystals have been used in clinical dosimetry and other applications.[16,44]

More frequently, silicon junction detectors have been used for electron and photon dosimetry.[44] In these applications, either a temporary or permanent change in conductivity is observed.

Silicon p-n junction counters have been applied to fast neutron dosimetry. This method uses the change in forward resistance of the diode due to the damage caused by fast neutrons. Fast neutrons displace silicon atoms from the lattice and thereby change the silicon diode resistance. The change in resistance can then be related to the fast-neutron dose. The unit can be made quite small and is insensitive to γ rays. It is not yet suitable for low-level use since its sensitivity is limited to about 0.1 Gy (10 rad)[16]. For neutron energy > 350 keV, the diode is relatively energy independent (within ± 20% up to 15 MeV).[16,37] Again, because of the inability to read low doses and the relatively high cost, this method has not been applied to personnel dosimetry. It has been used successfully as a fast neutron dosimeter in nuclear accident applications.

5. Thermally Stimulated Exoelectron Emission (TSEE)

The use of thermally stimulated exoelectron emission has not been extensive in the past. Various early studies of the use of this method for dosimetry have not led to wide acceptance. Recent interest in this method has come about because of the ICRP and ICRU recommendations concerning the deep (penetrating) and shallow (superficial) dose equivalents.[45,46] Many existing personnel dosimeters have filter/detector thicknesses which are unable to adequately measure the superficial dose.[15] However, a recent investigation[47] has indicated that the use of thin films of BeO and utilizing TSEE measurements shows promise.

E. Track Etch Dosimeters

The original track recorders which used fissile material had the disadvantage that monitored personnel received a radiation dose from the dosimeter itself.[3,48] These dosimeters employed a foil of fissionable material as a radiator of recoil particles (fission fragments). A suitable glass (quartz, flint) or dielectric material (mica, Lexan, cellulose nitrate, etc.) was used as the track recorder. Later improvements in the method led to the use of the recoil tracks produced in the material of the recorder itself. Since the recoils consisted mainly of C, O, H and N atoms from the polycarbonate material, the specific energy loss along the track is high. This energy loss could not be supplied by electrons or photons, so the recorders are inherently insensitive to electron and photon radiation.[31] This is an advantage for neutron monitoring, as is the low cost, negligible track fading, and ease of processing.[37] By employing the electrochemical etching technique, the recoil tracks in polycarbonate are increased in size and are made more visible. This enhances the counting of tracks.

Polycarbonate foils do not register tracks from proton recoils, so these devices are relatively insensitive to neutrons below 1 MeV. The response of these dosimeters is approximately proportional to the neutron dose equivalent from about 1 - 20 MeV.[49] Some polymers are unsuitable because cracks or other defects show up as high background track density.

Another plastic recorder, CR-39, has been found to be more sensitive than polycarbonate.[50] Whereas polycarbonate does not appear to detect recoil protons, CR-39 does. This allows detection of neutrons by CR-39 down to an energy of ~ 0.1 MeV.

The system makes use of a two-step electrochemical etch process at an elevated temperature.[51] The foils are put into an etch chamber and placed in an oven which is maintained at a temperature of 60°C overnight. 60°C KOH (6.5 N) is added to the chamber and etching is carried out for 5 hours at 3000 V and 60 Hz.[52] In the 23-minute second step, the voltage is kept at 3000 V but the frequency is changed to 2 kHz. This second step makes the tracks much larger and nearly uniform in size. The enlarged tracks are more easily distinguished from the defects normally found on the foil. This enhances the counting of tracks and also results in an improved track density relative to that obtained without etching. The foils are scanned with an optical bacteria counter and approximately 50% of the available etched fields are counted.

The sensitivity is about 7500 tracks/m^2 μSv (7.5 tracks/cm^2 mrem)[52] and the typical background on a foil is around 80 μSv (8 mrem). The detection limit for a single foil is ~ 100 μSv (10 mrem).

There is no fading problem but the foils must be protected from light and high temperatures. Foils are usually stored in the dark and, when in use, must not be exposed to ambient light.

The foils respond to neutrons in the energy range of about 0.1 -18 MeV. The neutron energy response is relatively flat from 0.08 - 3.5 MeV, but is down by a factor of 3 in the region 13-16 MeV.[52] This energy dependence is less severe than that for albedo dosimeters. The response of CR-39 should be correct for slightly moderated neutrons coming through shielding as well as for unmoderated neutrons. The linearity of the response extends up to 0.015 Sv (1.5 rem). One drawback of the CR-39 foil system is the highly directional dependence of the response. The response may vary by nearly a factor of five depending upon the angle of incidence of the neutrons.[52]

The CR-39 dosimetry system is an improvement over albedo dosimeters because of the reduced energy dependence and offers less reading problems and fading concerns than NTA film. In addition, since polycarbonate foils do not respond below about 1 MeV, the CR-39 dosimeter offers more capability for monitoring moderated neutron fields. The Department of Energy is supporting the development of the CR-39 dosimetry system for use as a personal neutron dosimeter.

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QUESTIONS

13.1 What term is given to devices that indicate the total amount of radiation to which a person has been exposed?

13.2 List some of the features desired in a personnel dosimeter.

13.3 What is the most frequently used personnel dosimeter?

13.4 Explain why photographic film is more difficult to use in dosimetry than in radiation detection.

13.5 What is the active material in photographic film?

13.6 What term is given to the underdeveloped image or effect produced by radiation in photographic film?

13.7 Name the chemical used to "fix" the developed photographic film so that it will not change further when exposed to light.

13.8 Explain the difference between film opacity and film density.

13.9 What name is given to the plot of film density versus the logarithm of exposure?

13.10 Why is the linear region of the plot referred to in the previous questions important for measuring exposure?

13.11 The ability of a film to distinguish between two nearly equal exposures is a measure of the film's:

13.12 The effect in which increased exposure of a photographic film causes a decrease in the film's density upon development is known as:

13.13 What law of science indicates that film response for a particular amount of radiation is independent of the rate that the film is exposed?

13.14 Describe the relationship of shape and position of the plot of a characteristic curve in relation to the energy of photons and beta particles.

13.15 List some factors that enhance latent image fading.

13.16 What does the reciprocal of the exposure to produce a given film density define?

13.17 What term is used to indicate that the sensitivity of a film varies with photon energy? What principal technique is used to reduce this effect?

13.18 List the factors which affect the final density of a film.

13.19 What term indicates film density that occurs in unexposed film? How can this effect be corrected.

13.20 What instrument is used to determine film density?

13.21 How is film reading standardized and interpreted?

13.22 List some other purposes that may be served by a film holder other than a device for containing the film.

13.23 Why do film packets for personnel monitoring usually contain two or more films?

13.24 What should determine the equilibrium thickness of the film wrapper and the film holder or film badge?

13.25 What techniques are used in film dosimetry to discriminate between fast neutrons and photons in a mixed field? Between fast and thermal neutrons?

13.26 What term needs correction for a pocket dosimeter reading on

a) a very humid day?
b) on a very hot day?

13.27 How can a pocket dosimeter be constructed to measure thermal neutrons?

13.28 Why should a pocket dosimeter be read soon after use?

13.29 What are the energy dependent characteristics of a pocket dosimeter?

13.30 List advantages and disadvantages of the following for use in personnel dosimetry:

a) thermoluminescence,
b) infrared stimulations
c) radiophotoluminescence,
d) conductivity methods.

13.31 What is the nuclear reaction that is the basis for the detection of thermal neutrons in albedo dosimeters?

13.32 What are the commonly used materials for track etch detectors?

PROBLEMS

13.1 The intensity of a beam of light is reduced to one-third of its original intensity upon passing through a piece of photographic film. Find:

a) the opacity and
b) the density of the film.

Answers: a) 3
b) 0.477

13.2 If the net density of a film is 2 for $1.29\text{x}10^{-4}$ C/kg, what will be the exposure when the net density is 3.5? Assume that both densities are on the linear portion of the characteristic curve. Assume a slope of 1 per unit log exposure.

Answer: $4.08\text{x}10^{-3}$ C/kg

13.3 A pocket dosimeter has a volume of $2.5\text{x}10^{-6}$ m^3, a capacitance of 150 pF and is charged by a 200 volt potential. If full scale is $1.3\text{x}10^{-3}$ C/kg, what will be the voltage change ΔV when the instrument is discharged to read $7.7\text{x}10^{-4}$ C/kg?

Answer: 16.60 volts

13.4 Determine the practicality of a dosimeter identical to the one in the preceding problem except that its capacitance is reduced to 1.5 pF.

13.5 An air equivalent pocket chamber having a capacitance of 75 pF is initially charged to a voltage of 25 V. If the active volume contains $5\text{x}10^{-5}$ m^3 of air at STP (pressure: 760 mm Hg, temperature 273 K), what photon exposure will drop the chamber voltage to 20 V?

Answer: 5.8 μC/kg.

13.6 The exposure measured by a pocket dosimeter is to be corrected for variation of density with pressure and temperature (see equations 13.3 and 13.4). The density at temperature, T, can be written as

$\rho_T = \rho_o \frac{PTo}{P_oT}$, where ρ_o (1.293 kg/m^3) is

the density of air at 760 mm Hg of pressure (P_o), and at temperature T_o=273 K. Prepare a table of the density correction terms for a range of reasonable pressures and temperatures. Apply these values to the worked example in the text. What conclusions can you draw?

SECTION 14 - AIR SAMPLING

A. Introduction

Air sampling is performed to determine the cleanliness of the air environment in the work area, or in the air exhausted to the outside in the case of stack sampling. It is also done to check the effectiveness of laboratory design and/or work procedures as applied to contamination control. Radioactive air sampling falls within this general framework and is distinguished chiefly because the contaminant generally owes its noxious character to the fact that it is a source of ionizing radiation. This radiation is usually the key to the analysis performed on the collected air sample. The collection, or sampling method, is influenced by the physical characteristics of the contaminant.

In principle, regulations governing radiation dose equivalent to individuals apply to internally deposited emitters as well as to external sources of radiation. Permissible levels for external radiation dose equivalent are to be reduced by the dose equivalent obtained from internal emitters. Therefore, it is common practice to keep environmental concentrations of radionuclides well below the allowable level. To confirm that this is being done, air samples are collected.

The objectives of air sampling are to measure the concentration of the contaminant in the air (detection and analysis), to identify the type and characteristics of the contaminant to help evaluate the hazard potential, and to appraise the performance of control equipment or procedures. The primary emphasis of the radioactive air sampling program is to determine that the level of protection afforded the radiation worker is sufficient. A common practice is to use allowable concentration values as an index of the degree of control achieved. Then, measured concentrations well below the allowable limit imply satisfactory control. To document that control is being maintained, a routine air sampling program is carried out.

Stack sampling is performed to determine the levels of radioactivity being released under normal conditions, to indicate whether release controls are adequate, and to signal when accidental releases occur. In general, releases to the environment are regulated. One is required to report both normal levels of release as well as those in accident situations.

B. Nature of the Contaminant

Airborne contaminants are generally divided into two groups:[1,2]

1. Gases and vapors
2. Particulates.

For our purposes, gases are defined as substances that, under normal conditions of temperature and pressure, exist in the gaseous phase. The distinction between gases and vapors is somewhat vague, but a vapor is generally considered to be the gaseous phase of a substance that is primarily liquid or solid under normal conditions of temperature and pressure.[2]

Particulate matter includes solid and liquid particles ranging upward from molecular size (approximately 10^{-3} μm). Solids are further subdivided into dusts, fumes, and smokes, which are distinguished mainly by their mode of generation. A dispersion of liquid particulates in air is called a mist or fog. The term aerosol is used collectively for air suspensions of liquid and solid particulates.[3]

Frequently the atmosphere being sampled contains a combination of types of contaminants - for example, gaseous and particulate. Some caution must be exercised in taking and interpreting samples from such atmospheres.

Particulate matter is more readily retained in the lungs than are gases. But, particulate matter is seldom of only one size. In general, particulate matter is found in a distribution of sizes. The retention in the lungs varies with the size of the particle, so each different distribution will exhibit a different retention. The type of distribution often encountered is known as a log normal distribution. In this, the median size can be used to describe the behavior of the distribution.[4]

The parameter used is the median aerodynamic diameter. The aerodynamic diameter is the diameter of a unit density sphere which has the same settling velocity as the particle in question.

On this basis, particles of diameter > 10 μm are trapped in the nasal region and rapidly cleared.[5] Those of diameter 5-10 μm are stopped in the upper respiratory tract, removed by ciliary action and swallowed. Below this range, the chance of particles reaching the deep portion of the lungs increases as particle size decreases. Generally, the respirable range is taken as 0.01-10 μm, but the region of practical interest is 0.1-10 μm.

Since deposition is a strong function of the particle size, the proper evaluation of an air sample will be dependent upon the size distribution. This suggests that air samples should be analyzed for particle size distribution. One cannot accomplish this on a routine basis in most practical cases. However, sometimes initial studies may be used to characterize a distribution, which does not vary much over the long term. In accident situations, it is very important to attempt to obtain particle size distributions, since these may differ greatly from the normal situation.

C. Natural Airborne Radioactivity

Airborne radioactivity has always been a part of man's environment. It is largely due to the presence of uranium and thorium deposits in the soil. The isotopes ^{238}U and ^{232}Th are naturally radioactive, and each gives rise to a long chain of radionuclides (as shown in Figure 14.1). One of the daughter elements in each chain is a noble gas - radon (^{222}Rn) in the U-238 chain, and thoron (^{220}Rn) in the Th-232 chain. Chemically inert, they can migrate from the soil to the air. There, with their radioactive daughters, they comprise the major portion of the natural airborne radioactivity.

Their activity varies over a rather wide range with geography and with time in a given location. Geographical variations occur, presumably because radium-bearing soils are not uniformly distributed, and because they do not all liberate radon to the same extent.

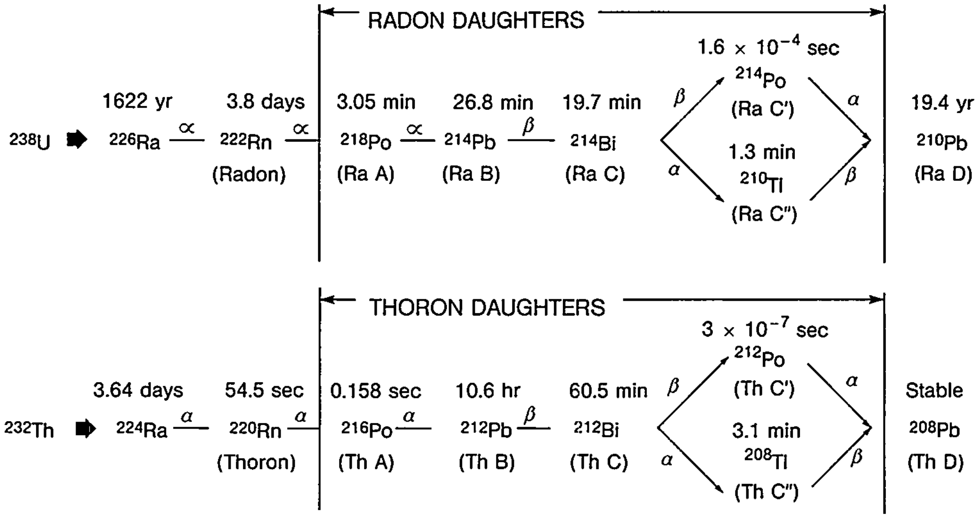


Figure 14.1 Uranium and thorium decay chains.

The activity in a given location varies with meteorological variables such as wind direction and velocity, temperature, humidity, and lapse rate. Generally, concentrations are greatest in the morning, dropping off in late afternoon before beginning an early morning rise.[6] The rise is attributed to a general decrease in wind velocity and increasing stability of surface conditions. The decrease in concentrations is associated with vertical transport from the earth's surface as it is warmed. Seasonal as well as diurnal variations are observed.[8,9]

The daughter products of radon and thoron are particulates and attach themselves to dust in the air. Mainly they become associated with the smaller atmospheric dusts. These are captured on filters, and their activity may overwhelm that of the isotope of interest if the filter is counted shortly after sampling.

This "noise," which is variable and unpredictable, hinders the interpretation of an air sample. Schemes for dealing with this problem are discussed in 14.I below, <u>Analysis of a Filter Sample.</u>

D. Obtaining a Representative Sample; Choice of Sampler Location

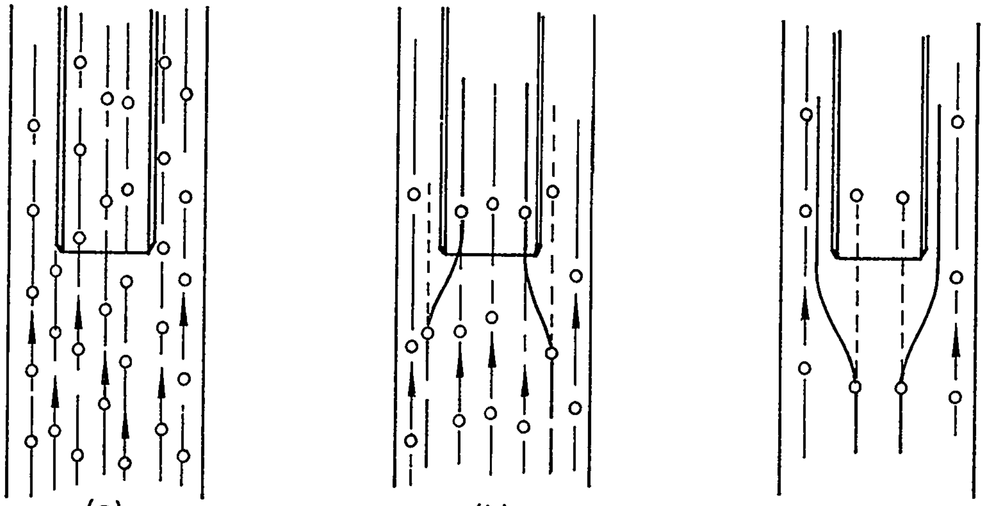
These are two related considerations in obtaining a representative sample. First, the concentrations entering the sampling line must be representative of the concentrations in the air near the sampling head. For example, in sampling in a stack for particulates, care must be taken to be sure that the sampling is reasonably isokinetic.[2,10] That is, the linear flow rate into the sampling orifice is the same as the air velocity in the stack at the same location. If not, a discrimination is made against small or large particles, depending on whether the flow velocity is lower or higher than the stack-flow velocity (see Figure 14.2).

The reason for the discrimination is that the inertia of the more massive particles prevents them from following an airstream that makes an abrupt direction change. The need for isokinetic sampling diminishes as the particle size in the aerosol decreases. Particles $< 3\ \mu m$ usually follow the airstream lines so losses are not too large.[10]

If the gas being sampled contains chemically reactive material, one must guard against losing part of the sample during collection. For example, the material may be adsorbed onto surfaces of the sampling lines before being collected. Or condensation on surfaces may trap particles which would otherwise be collected.

Sampling lines should be as short and as straight as possible to avoid plating out along the walls of the tube. Vertical sampling lines should be used, horizontal lines should be avoided. The sampling port should be sufficiently far removed (at least five stack diameters) from the last bend, opening or obstruction to avoid the effects of excessive turbulence in the airstream.[2] Restrictions and bends in the sampling line itself lead to loss of particles by inertial impaction. The use of nonconducting materials (such as rubber tubing) in the sampling line will also lead to increased particle losses due to electrical forces. Reactive vapors or gases may be absorbed on or react with certain materials used in sampling lines, such as iodine with rubber, copper and some plastics.[2]

Secondly, the concentration at the sampling head must be representative of the concentration at the point of concern. Since this is usually



(a)
$V_{SAMPLE} = V_{STACK}$
ISOKINETIC

(b)
$V_{SAMPLE} > V_{STACK}$
DISCRIMINATES AGAINST
LARGER PARTICLES

(c)
$V_{SAMPLE} < V_{STACK}$
DISCRIMINATES AGAINST
SMALLER PARTICLES

Figure 14.2 Isokinetic sampling.

the nose of an individual, the sample must be of a concentration approximating that in the breathing zone of the individual. Sherwood and Greenhalgh[11] have reported variations between exposures indicated by a personal air sampler and those indicated by a fixed-station sampler in the same room,

On the other hand, a fixed-station sampler may be relevant for people working in a location well away from possible significant sources of air contamination, such as hoods or glove boxes. Care should be used in the selection of both the number and placement of these general work area samplers. Samplers placed near ventilation exhaust ducts will tend to give an integrated average. However, this may not be representative of what the worker breathes. Differences of a factor of 10 have been observed between personnel and fixed air sampler results in a given area.[12]

E. Sampling Program

Factors which need to be considered in setting up an air sampling program are frequency of sampling, collection time, sampler type and volume flow rate.[2] The choices that one has will often be influenced by the form, nature and containment of the radionuclide being sampled as well as the type of operation being performed. In general, for a given radionuclide, one should sample a large enough volume so that 1/10 of the allowable concentration can be measured. However, the method of analysis with respect to its sensitivity and accuracy will affect the volume needed.[2] The time that one samples will be affected by the interfering background activities, as well as the nature of the radioactivity being sampled. That is, the natural radioactivity in air may be able to swamp the activity from a concentration of a long-lived emitter if the sampling time is not long enough. On the other hand, one loses a significant fraction of the collected activity of a short-lived contaminant if the sampling time is too long. In those instances where the identity of the potential radionuclide is known, one will be able to make intelligent decisions with regard to proper sampling. The frequency of sampling is usually dictated by the experience one has with the given operation. New or unusual operations will require an initial frequency of sampling which may later be relaxed when the degree of control has been established. However, for high hazard material or for processes with a high potential for release, high frequency air sampling, in conjunction with frequent monitoring by health physics personnel, may be required. For other considerations concerning applied health physics air sampling, see References 13 and 14.

F. Choosing a Sampler

Basically two types of samples are collected:

1. A volume or grab sample in which part of the universe with the contaminant in its original concentration is isolated. This sample gives conditions at a point in time and space. Many such samples are needed to adequately describe average conditions in a large area.

2. An integrated sample, in which the contaminant is concentrated in the collecting medium. This sample gives a kind of average of the collection time and does not reveal any "fine structure" in the air concentration.

Samples of either type may be taken for both gases and particulates.

Choosing a sampler involves choosing a method of analysis as well. They depend upon each other, as well as upon the characteristics of the contaminant and the allowable levels of concentration in the atmosphere. A less sensitive method of analysis requires a larger or more concentrated sample - for example, a larger volume of grab sample, or a higher flow rate for an integrated sample. The most prominent characteristic of a radioactive nuclide is usually its radioactivity; it will normally be the basis of any analysis made.

Finally, the choice of a sampler depends on the type of information desired. If immediate readout is needed, collection and analysis are done at the same time. If not, analysis is often done at some central location removed from the sampling site; the sampler is more economical and less bulky. The analysis is more flexible and generally better. The use of a central location for analysis works well when a large number of the same type of air samples are generated. In these cases, automated counting equipment may be employed.

G. Sampling Methods and Devices

Methods and devices employed for radioactive air sampling are briefly discussed in this section.

1. Filter Samplers

Filtration is the most common method of sampling for radioactive particulates.[15] It is simple and efficient. The mechanics of filtration are discussed in 14.H below.

A wide choice of filters is available, and there is one suitable for almost any purpose. Among the characteristics that influence a choice of filter are collection efficiency, flow resistance, and mechanical strength. Clearly, no one filter is best in all of these. A compromise is required. Properties of a number of air sampling filter media are given in Reference 2. In recent years, the cellulose-asbestos filters have replaced the asbestos with other media.

Analysis is commonly done on filter paper samples by gross counting of alpha and beta activities. This may be accomplished in stacked proportional counters (piggyback probes). Solid-state detectors, and/or plastic scintillators, such as Pilot B (diphenyl stilbene in clear plastic), have also been employed to count these samples. Semi-automatic, microprocessor (minicomputer) based systems have been used at facilities with a large number of routine samples to analyze.[16] For alpha counting, the self-absorption in the paper must be taken into account.[17]

Membrane filters offer advantages for radioactive work. The pore sizes are closely controlled in manufacturing and are well known. A number of these are available and very high efficiencies can be obtained by proper selection. The filters can be made transparent for light microscopy and are readily dissolved in organic solvents. More importantly, they are basically surface collectors so that alpha-particle spectroscopy can be done with good resolution.[18] However, they are somewhat brittle and must be well supported during collection and carefully handled. Moreover, they exhibit a rather high pressure drop to accompany their high efficiency so that high flow rates are difficult to achieve.

The filter sampling technique is to draw air at a known flow rate through a filter for a known length of time. Figure 14.3 depicts a typical system.

A sampler in common use consists of a Filter Queen vacuum cleaner, modified by mounting it on a wheeled base and attaching a rotameter and a filter-paper holder. It is used with LB-5211 (made by Hollingsworth & Vose) paper (100x225 mm) at a flow rate of 15-20 m^3/h. Collection time is usually limited to less than 8 h.

Portable, high-volume air samplers (Hurricane, Staplex, and Unico) are used for short collection times requiring large volumes. With 125x200 mm LB-5211 papers, these units sample at rates up to about 60 m^3/h.

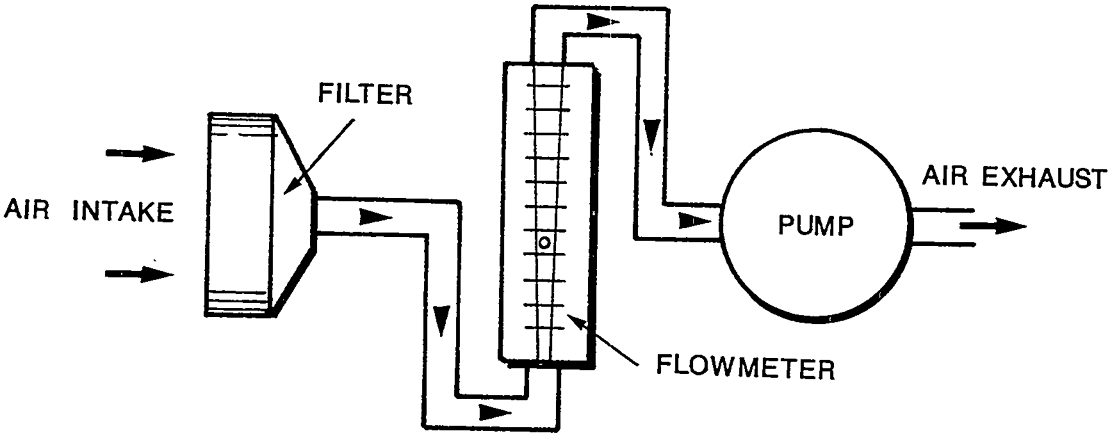


Figure 14.3 Typical filter sampler arrangement diagram.

Membrane or cellulose fiber filters in small (25 to 50 mm diameter) holders are sometimes used to obtain breathing-zone samples because of the relative ease with which they can be positioned. The holder can be clipped to a lapel and connected by Tygon tubing to a small portable air pump.

Commercial lapel samplers, operated on rechargeable batteries, are also in use and offer an opportunity to sample the breathing zone of an individual moving about.

Once the sample is collected, the activity on the filter can be determined in several ways. Gross α or β counts can be taken, and from these the filter activity can be calculated using a previously determined counter yield. Individual counting can also be performed for specific energies of a suspected contaminant.

Given the total filter activity, the portion due to the contaminant of interest must be found; this is complicated by the presence of the natural airborne activity. This problem is discussed in 14.I below.

Gamma spectroscopy can be done if the contaminant has a suitable x or γ emission, but interference from natural airborne activity can be a problem here. Alpha spectroscopy is possible if collection is made on the surface of the filter.

2. Grab Samplers

A partially evacuated container can be used to sample for airborne gases or particulates. The container is connected to a vacuum pump, evacuated, isolated, and then removed to the sampling location. There it is opened admitting the atmosphere to be sampled, then isolated once more, and removed for analysis.

The volume of sample admitted to a partially evacuated sampler is given by

$$V = V_f \frac{p - p_r}{p} \qquad 14.1$$

where V is the volume of sample taken,

V_f is the volume of the sampler flask,

p is the atmospheric pressure,

p_r is the residual partial pressure in the container after evacuation, and

$p - p_r = p_g$ is the pressure measured by a vacuum gauge.

That is, the gauge measures the difference in pressure between the ambient pressure outside and inside the flask. Thus,

$$V = V_f \frac{p_g}{p}. \qquad 14.2$$

The gauge is usually calibrated in inches of mercury and can be converted to millimeters of mercury if the barometric pressure is measured in those units.

The sample volume can be converted to a standard volume (0°) and 760 mm Hg) to give

$$V_s = V\frac{P\,T_s}{P_s T} = V\frac{p}{760}\left(\frac{273}{273 + t°\ C}\right) = V_f\frac{p_g}{760}\left(\frac{273}{273 + t°\ C}\right) \qquad 14.3$$

This allows for comparison of different samples on a standardized basis. Generally, this correction for sample volume is ignored in practice since the error is usually only 5-10%.

The grab sampler (see Figure 14.4) is a 3.5 liter brass cylinder with a thin-walled G-M tube mounted along its central axis. For analysis, the G-M tube is connected to a scaler, and a gross count is made. This count can be converted to activity in the sampler volume if the system has previously been calibrated with a known activity of the isotope of interest. Such a calibration must be made since the counting efficiency of the tube depends strongly upon the energy of the emitted β particle.

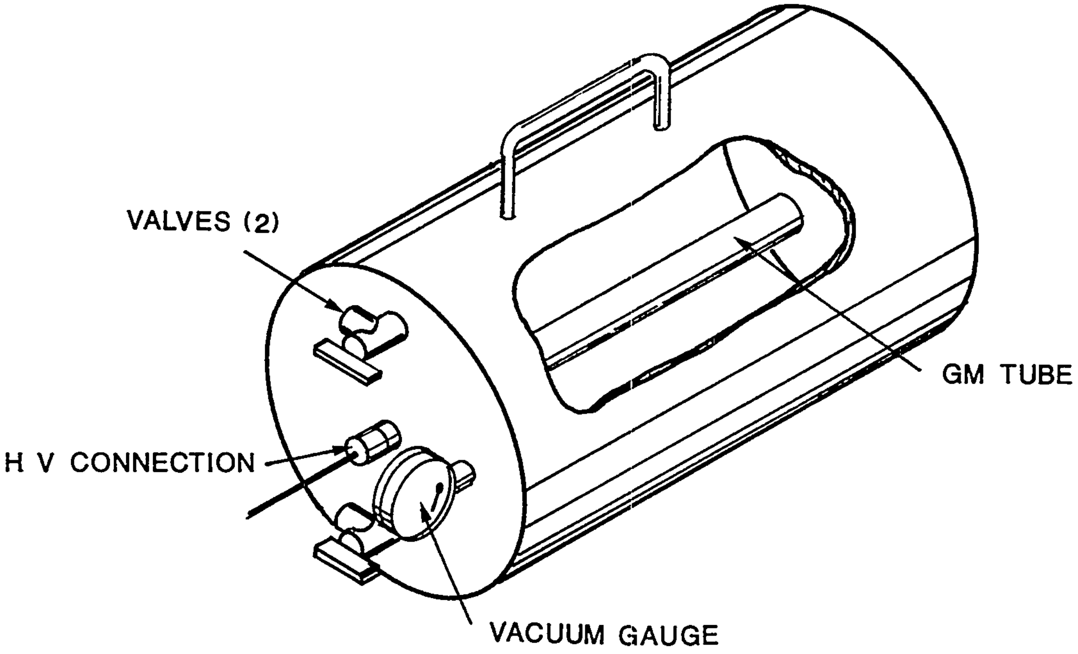


Figure 14.4 3.5 liter grab sampler.

The weakness of this scheme is its low efficiency. The 3.5 liter sampler, using a 0.3 kg/m^2 tube 76 mm long, has a yield of about 1 % for ^{41}Ar. To assure a more representative concentration, one should take a number of grab samples and average the results.

Grab samples taken in glass bulbs or bottles can be analyzed using γ pulse-height analysis. This has the added advantage that one is able to identify the radionuclide. With the proper setup, the system may be calibrated to determine the activity concentration.

3. Impingers and Impactors

In the impinger, particles are removed by inertial precipitation from a sharply deflected airstream. After impinging against a surface at right angles, the airstream continues. Because of their inertia, particles above a certain effective size cannot follow and are collected.

Particles are collected on the deflecting surface, for example, a glass microscope slide, which has been greased to enhance retention. More commonly, the impinging surface is immersed in a liquid such as distilled water or alcohol, which serves as the collecting medium. An aliquot of the liquid is then counted.

Instruments of this type are the Greenburg-Smith impinger (Figure 14.5) and its smaller analog, the midget impinger. The sampling rates for these instruments are 20-40 l/min and approximately 3 l/min, respectively.[19] Their collection efficiency becomes generally poor for particles smaller than about 0.7 μm. For this and other reasons, impingers are not extensively used in radioactive sampling. However, they are useful in dusty atmospheres where there is no need to determine the particle size distribution. They are usually inadequate for sampling clean ambient air.[19]

The cascade impactor (Figure 14.6) consists of several impingement stages in series followed by a final filter. Particle sizes are selected by reducing the size and impingement distance of each succeeding orifice, thereby increasing flow velocity and causing smaller particles to be deflected from the airstream. A crude particle-size spectrum is obtainable this way. However, distortion may be introduced because of the breakup of the larger agglomerates undergoing impingement and retention failure (particles bouncing from an upper stage to a lower stage). This is a problem with all impingement devices. If collection were ideal, then the

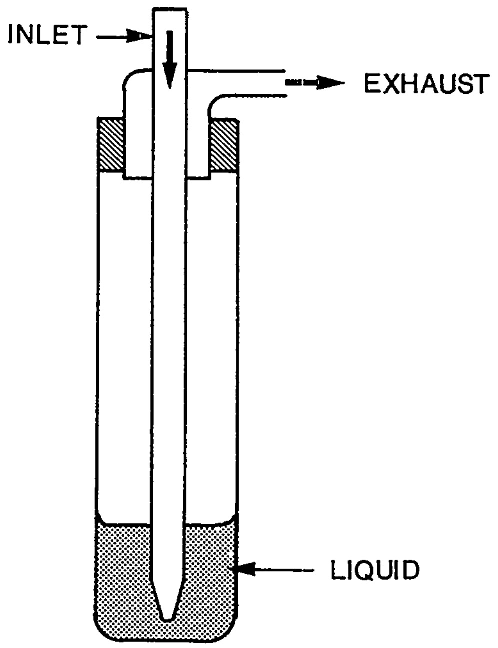


Figure 14.5 Greenburg - Smith impinger. Wet impingement is the collection principle employed in this device.

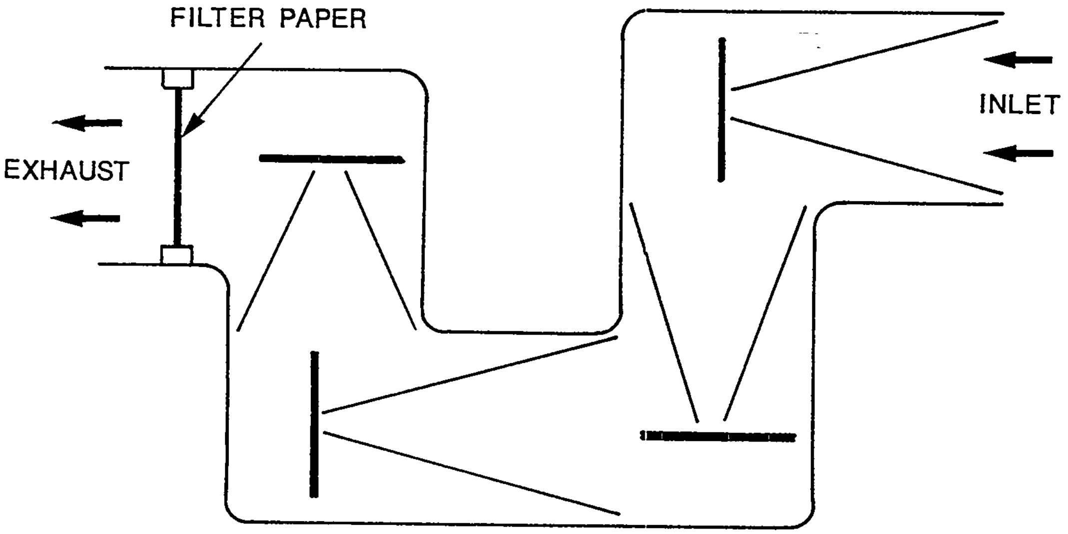


Figure 14.6 Cascade impactor.

particles collected on any stage would have diameters between two fixed values.[19] Overlap in sizes does occur, so each stage must be calibrated. Impactors have been found useful for sampling in the presence of natural radon-daughter activity, which is usually associated with particles smaller than 0.3 μm in diameter.[20,21]

4. Condensation Devices

A cold trap immersed in a bath of dry ice and alcohol, liquid nitrogen, or other suitable coolant can be used for sampling non-reactive or insoluble gases. Dry ice and alcohol are used as a coolant to freeze out tritiated water vapor. A liquid scintillation count of the collected water gives the tritiated water-vapor activity per unit mass of water vapor. Relative humidity and temperature measurements, made at sampling time with a psychrometer, give the concentration of water vapor per unit volume of air, when corrected for the barometric pressure. The product of these two numbers, gives the tritiated water concentration per unit volume of air at the sampling time.

The condensation method is not used extensively for other gases, since the water vapor in the air hinders the collection process. Often, it is necessary to remove the water for efficient collection.[2]

5. Adsorbers

Adsorbers such as activated charcoal, silica gel, or metal gauzes (Ag, Cu) are used for collecting organic vapors and non-reactive gases and vapors such as xenon, krypton, and argon. The technique involves bringing the sampled atmosphere into intimate contact with a finely divided or porous adsorber. The efficiency of collection is generally enhanced by maintaining the adsorber at a reduced temperature.[2]

Activated carbon cartridges of the type described by Sill and Flygare[22] have been used to sample for radioiodine. Sampling is done at room temperatures, and efficiencies are generally greater than 95% for elemental iodine. Commercially available organic-vapor cartridges are also used for such sampling and with similar efficiency. Problems may be encountered when the iodine is not in a gaseous and elemental form.[15] Silver zeolite cartridges have been used when retention of noble gases may interfere with iodine analysis.

Analysis of the cartridges used for iodine is accomplished by γ spectroscopy.

6. Continuous Air Monitors

Continuous air monitors are used to maintain a watch on the level of air activity in an area such as the inside of a reactor containment shell. Such devices, either facility specific or commercially available, are capable of monitoring for several radiation components.

There are many varieties. The collecting devices and detectors they use depend on the nuclides and levels of interest. We can incorporate several systems in parallel; a flow-through ionization chamber may be used to monitor gases, while an end-window G-M tube looks at the particulates collected on a filter. Other designs use surface barrier detectors for α, plastic scintillators for β, and NaI crystals for γ.

One instrument, the continuous air monitor CAM-5,[23] was originally designed at ANL. Several later versions, incorporating improvements, have followed. The device uses the equilibrium counting ratio of radon daughters by simultaneously counting the β and α activity built up on a fixed filter. The activities are each counted separately in two stacked gas proportional counters (piggyback probe) and read out on charts. The unique feature is a third channel, which reads out the difference between the alpha counting rate and an arbitrarily selected fraction of the β counting rate. Although the individual counting rates may vary, their ratios remain somewhat steady under conditions of equilibrium. The fraction chosen is a function of the β/α ratio of the background activity and the respective efficiencies of the two counters. It is chosen to give a null reading on a null chart. The subsequent introduction of an α- or β-emitting contaminant causes the reading to deviate from the null in one direction or the other. With alarm circuitry, this instrument gives a very sensitive indication of the presence of a contaminant.

Another instrument which has been designed to monitor for plutonium in the presence of radon daughter products[35] is discussed in Section 16.B.5a(2).

For additional discussions concerning sampling devices, methods and detection of particular airborne radionuclides, consult References 1, 2, 15, 24-27 and 35.

H. Filters

Several mechanisms are at work in the filtering process, which is not merely a sieve action. Particles larger than the filter pore size are strained, but high efficiencies are also attained for small particles.[18,28,29]

Relatively large particles impinge on the filter fibers because of their inability to follow a sharply deflected airstream. The size of the particle effectively collected by this mechanism decreases with increasing velocity.

Other large particles settle onto the fibers in a manner governed by Stokes' Law. Still others may be directly intercepted by a fiber. Electrostatic forces due to charges on the fiber and/or particles aid collection and retention on some filters. Humidity may also play a role. Very small particles are brought to the fibers because of their Brownian motion.

Overall, there will usually be an optimum airflow for a given filter and a given aerosol. Theoretically, we would expect a minimum collection efficiency (maximum penetration) for particles of some intermediate size at a given airflow because of particles too small for inertial effects or settling to be effective and yet too large for efficient Brownian diffusion.

Penetration maxima have been reported for many filters, but not for all - at least not in the range of velocities and particle sizes tested.[30] When found, the maximum often occurs in the range 0.06-0.2 μm.[2] Minimum deposition in the deepest part of the lung is observed in this region.[5]

In sum, collection efficiency cannot be tied to a single variable. It is a complicated and unpredictable function of many variables, including particle size and density, flow velocity, filter pore size, fiber diameter, filter loading, and the electrostatic charge and chemical characteristics of the aerosol. Collection efficiency should be determined for each sampling situation.

Knowledge of the absolute collection efficiency is not as important for air sampling as it is for air cleaning. An actual collection efficiency of 99% in a sampler rather than an assumed 99.9% does not introduce a serious error into the analysis. However, the same error for an air-cleaning filter could have serious consequences.

For example, suppose that a 20 m³ sample is taken for ^{226}Ra, 99.9% efficiency is assumed, and a net counting rate of 220 c/min is obtained on a counter system having a 50% yield (Y) for ^{226}Ra. Then,

$$A(^{226}\text{Ra}) = \frac{220 \text{ c/min}}{\text{Y c/min/dis}} = 440 \text{ dis/min} = 7.33 \text{ Bq (198 pCi)}.$$

$$C(^{226}\text{Ra}) = \frac{7.33 \text{ Bq}}{20 \text{ m}^3(0.999)} = 3.67\text{x}10^{-1} \frac{\text{Bq}}{\text{m}^3}(9.92 \text{ pCi/cc}).$$

If the penetration were actually 1% rather than 0.1%, the concentration, C, would really be

$$C = \frac{7.33 \text{ Bq}}{20 \text{ m}^3(0.99)} = 3.70\text{x}10^{-1} \frac{\text{Bq}}{\text{m}^3}(10 \text{ pCi/cc}).$$

Truly, this is a negligible error.

Now, assume that an air-cleaning filter had a concentration of 370 Bq/m^3 on its upstream side. An assumed efficiency of 99.9% would give a concentration of 0.37 Bq/m^3 on its downstream side, whereas an actual efficiency of 99.0% would mean a downstream concentration of 3.7 Bq/m^3.

For the nuclide chosen above, this means a difference of a factor of ten in the concentrations released to an uncontrolled area.

I. Analysis of a Filter Sample

1. General

Assume a filter sample is collected from air having a single nuclide at a constant concentration C_c. The sampling takes place for a time, t_s, after which the filter is removed. Figure 14.7 shows the buildup of activity on the filter and its decay after sampling ends.

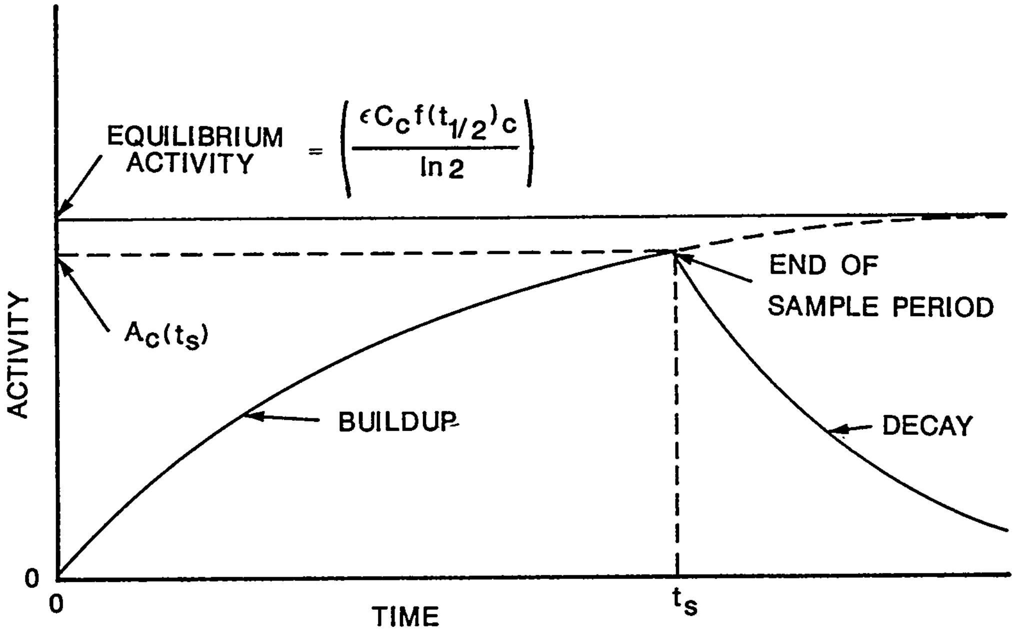


Figure 14.7 Activity buildup - decay of a single contaminant on a filter.

The net activity on the filter due to the contaminant at any time during sampling is given by[31]

$$A_c(t_s) = \frac{\epsilon C_c f (T_{\frac{1}{2}})_c}{\ln 2}\left(1 - e^{-\frac{\ln 2\, t_s}{(T_{\frac{1}{2}})_c}}\right) \qquad 14.4$$

where

ϵ is the efficiency of collection of the filter (often assumed to be approximately 1),

C_c is the average concentration of contaminant in the air,

f is the flow rate of air through the sampler, and

$(T_{\frac{1}{2}})_c$ is the half life of the radionuclide.

If there is a delay, $t_1 - t_s$, before counting, the contaminant undergoes decay on the filter, and the activity determined at the counting time is related to the activity at the end of sampling by

$$A_c(t_1) = A_c(t_s) e^{-\frac{\ln 2\, (t_1 - t_s)}{(T_{\frac{1}{2}})_c}}. \qquad 14.5$$

The activity is obtained from the sample counting rate, $R(t_1)$, on the filter by

$$A_c(t_1) = \frac{R(t_1)}{Y} = \frac{R_{tot}(t_1) - B}{Y}, \qquad 14.6$$

where

Y is the yield of the counting system for the radionuclide of interest,

$R_{tot}(t_1)$ is the total counting rate on the filter, and

B is the background counting rate of the counter.

Finally, the concentration in air (assumed constant over the sampling period) can be related to the activity on the filter by

$$C_c = \frac{A_c(t_1)}{\frac{\epsilon f (T_{\frac{1}{2}})_c}{\ln 2}\left(1 - e^{-\frac{\ln 2\, t_s}{(T_{\frac{1}{2}})_c}}\right) e^{-\frac{\ln 2}{(T_{\frac{1}{2}})_c}(t_1 - t_s)}} \qquad 14.7$$

For long-lived emitters, $T_{1/2} \gg t_1$, and

$$C_c = \frac{A_c(t_1)}{\epsilon\, f t_s}. \qquad 14.8$$

The above analysis applies to a condition in which a single radionuclide is collected on the filter and the background air activity is either zero or known. Sampling in a duct downstream of an absolute filter approximates such a condition.

Unfortunately, most air sampling takes place where the natural airborne contamination (background air activity) is neither zero nor predictable. The total activity on the filter at any time is then the sum of the activities from the contaminant of interest plus the activities from the natural background, i.e., the radon- and thoron-daughter products:

$$A = A_{Rn}(t) + A_{Th}(t) + A_c(t), \qquad 14.9$$

where

A_{Rn} is the activity on the filter due to radon daughters,

and

A_{Th} is the activity on the filter due to thoron daughters.

The problem is to determine A_c so that the concentration of the nuclide of interest, C_c, can be determined using the above equations. A_c can be determined with reasonable accuracy if the contaminant is sufficiently long-lived with respect to the thoron daughters. However, this will take several hours, by which time the information may be only of academic interest. What is often needed is a scheme for quickly deciding whether a hazardous conditions exists.

2. First-count Factor

A quick indication of a possible airborne hazard can often be based on the ratio of the activities of β emitters to the activities of α emitters on the filter, determined as quickly as possible after sampling:

$$\Gamma = \frac{\text{dis/min m}^3(\beta\gamma)}{\text{dis/min m}^3(\alpha)}. \qquad 14.10$$

This "first-count factor," Γ, is remarkably constant when only natural (i.e., radon-thoron) daughters are present. If the ratio changes significantly upward, the release of a β emitter is signaled and conversely, if the ratio changes significantly downward, the release of an α emitter is signaled.

The actual measured value of the first-count factor depends upon the calibration of the particular counting system and the values assigned to parameters such as the α self-absorption factor. It will likely vary from system to system and is not terribly important in itself. What is important is that the range of values for a given counting system be well known for the condition when only naturally-occurring air contamination is present. $\overline{\Gamma}$ is then obtained from the air sampling "history" of the area. That is, a series of samples are taken and the value of Γ is determined for each sample. From these, $\overline{\Gamma}$ is the average value for the series of air samples.

Because the natural background varies so widely, the first-count factor is a much more sensitive indicator of the presence of a contaminant than is the actual β or α activity. It also allows the concentration to be estimated using the known mean value of the first-count factor, $\overline{\Gamma}$, for a clean area. In most practical applications though, the first count factor is used as a flag, or index, to point toward a potential release. For example, assume a series of samples are taken and it is found that Γ ranges from about 1.8-3.0 in a given area. For a continuing air sample program in that area then, samples with Γ values in this range would indicate no release of airborne activity. Samples outside this range would signal the need for some follow-up action. In particular, values > 3.0 indicate a potential β release; values < 1.8 indicate a potential α release.

In the case of a suspected release, the activity on the filter due to the concentration can be estimated from

$$A^\circ_{\alpha c} = A^\circ_{\beta} \left(\frac{1}{\Gamma} - \frac{1}{\bar{\Gamma}}\right), \text{ for an indicated } \alpha \text{ release} \qquad 14.11$$

and

$$A^\circ_{\beta C} = A^\circ_{\alpha} (\Gamma - \bar{\Gamma}), \text{ for an indicated } \beta \text{ release,} \qquad 14.12$$

in which $A^\circ_{\alpha c}$ or $A^\circ_{\beta c}$ is the initial activity on the filter due to a contaminant, A°_{α} or A°_{β} is the initial activity due to natural background, $\bar{\Gamma}$ is the known average first count factor, and Γ is the first count factor determined for the sample in question.

The use of the first count factor for quickly estimating concentrations of around a few times the allowable concentration guide for the most hazardous α emitters is unreliable. The sensitivity improves with increased sampling time and with decreased background, but for other conditions, quantitative information cannot be obtained by a quick estimate. In addition, when both α and $\beta\gamma$ counts are greatly elevated over normal values, the first count factor is not useful.

To illustrate in the case of ^{239}Pu, assume that $\bar{\Gamma}$ for a given area is 2.1±0.4. One then obtains an air sample for which A°_{β}=4200 dis/min-m^3 and A°_{α} = 2168 dis/min-m^3. In this case we are looking for a hazardous α emitter, so the first count factor should be unreliable. If we compute Γ, we get

$$\Gamma = \frac{A^\circ_{\beta}}{A^\circ_{\alpha}} = \frac{4200}{2168} = 1.94 \text{ which is within } 2.1 \pm 0.4.$$

Since the composite half life of radon daughter products is about 35 minutes, a recount taken about 35 minutes after the initial count should give more information about the sample. If the activity drops to about ½ of the initial value A_0, then the indication is that the sample may contain only radon daughters. If the activity A_{35} at that time is larger than $A_0/2$, then an estimate of the long-lived activity on the filter is

$$A_{LL} = 2\ A_{35} - A_0. \qquad 14.13$$

The estimate made from this count generally gives only a crude overestimate of the activity and further counting is needed to obtain a more accurate determination. Returning to our example, suppose a recount is made 35 minutes after the first count and the α activity is now 1100 dis/min m³. Then, from equation 14.13,

$$A_{LL} = 2(1100) - 2168 = 32 \text{ dis/min m}^3,$$

which is above the DAC value of 4.4 dis/min-m³ (Class W). This example shows the difficulty in determining high hazard, long-lived α emitter concentrations by means of filter paper samples alone, when the natural background interference is significant. This is one reason why surface barrier detectors, which have the ability to measure the energy of the alpha particles, are generally used for detecting these airborne α emitters.[32]

3. Concentration of Long-Lived Contaminants

We return now to the problem of calculating the concentration in air, C, of a long-lived emitter that has been collected on a filter along with an unknown activity of radon and thoron daughters.

If the filter is counted about 6 hours post sampling, radon daughter activity is reduced by a factor of > 1000 and can, therefore, be ignored. If we call this time t_1, the activity on the filter is given by

$$A_1 = A_{Th}(t_1) + A_c(t_1). \qquad 14.14$$

Counting again at a later time, t_2, we get

$$A_2 = A_{Th}(t_2) + A_c(t_2). \qquad 14.15$$

Now,

$$A_{Th}(t_2) = A_{Th}(t_1)e^{-\frac{\ln 2}{(T_{1/2})_{Th}}(t_2-t_1)} \qquad 14.16$$

and

$$A_c(t_2) = A_c(t_1)e^{-\frac{\ln 2}{(T_{1/2})_c}(t_2 - t_1)} \qquad 14.17$$

We can solve for the unknown to obtain

$$A_c(t_1) = \frac{A_2 - A_1 e^{-\frac{\ln 2}{(T_{1/2})_{Th}}\Delta t}}{e^{-\frac{\ln 2}{(T_{1/2})_c}\Delta t} - e^{-\frac{\ln 2}{(T_{1/2})_{Th}}\Delta t}} \qquad 14.18$$

where

$$\Delta t = t_2 - t_1.$$

From here we can use equation 14.7 to solve for C_c. In practice, Δt should be long enough so that there is a substantial difference between A_1 and A_2. It is usually taken to be about 10 h, which is about the composite half life (10.64 h) of the thoron daughters. The half life of the contaminant should be considerably longer than 10.64 hours so that the denominator in equation 14.18 does not approach zero.

In the usual case, where the contaminant half life is much longer than t_2 (measured from the beginning of sampling),

$$A_c(t_1) = \frac{A_2 - A_1 e^{-\frac{\ln 2}{(T_{1/2})_{Th}}\Delta t}}{1 - e^{-\frac{\ln 2}{(T_{1/2})_{Th}}\Delta t}} = \frac{A_2 - A_1 e^{-.0652\Delta t}}{1 - e^{-.0652\Delta t}} \qquad 14.19$$

and

$$A_c(t_1) = A_c(t_s) = \epsilon C_c f t_s. \qquad 14.20$$

That is, the contaminant does not undergo significant decay on the filter during sampling or analysis, a condition that applies to almost all the nuclides of real concern for usual sampling times (less than 8 hours).

Since the activity ratio for thoron daughters can be predicted for the times t_1 and t_2, the fraction of activity at t_2 that is due to the contaminant can be stated as a function of the count ratio:[33]

$$\frac{A_A(t_2)}{A_2} = \frac{R - A_1/A_2}{R-1} \qquad 14.21$$

where

$$R = \frac{A_{Th}(t_1)}{A_{Th}(t_2)} = e^{-\frac{\ln 2}{(T_{1/2})_{Th}}\Delta t} = e^{-.0652\Delta t} \qquad 14.22$$

In Figure 14.8, $A_c(t_2)/A_2$ has been plotted against A_1/A_2 for various values of Δt. The figure can be used to calculate the activity on a filter at t_2 due to a long lived contaminant.

Example: A filter sample is taken for 4 hours at 20 m³/h; ^{239}Pu is the suspected contaminant. The following disintegrations rates (Bq) were obtained:

A	t		
	t_s = 4 h	t_1 = 10 h	t_2 = 20 h
$A_ß$	170	17	9
A_α	90	20	16.70

Estimate the concentration of ^{239}Pu. Assume that $\bar{\Gamma}$ = 2.1±0.3 represents the normal situation in the area.

From the first counts, we get $\Gamma = 1.88$. Since this is within the expected range for $\bar{\Gamma}$, no real information is obtained. However, from the later counts, we get the ratio

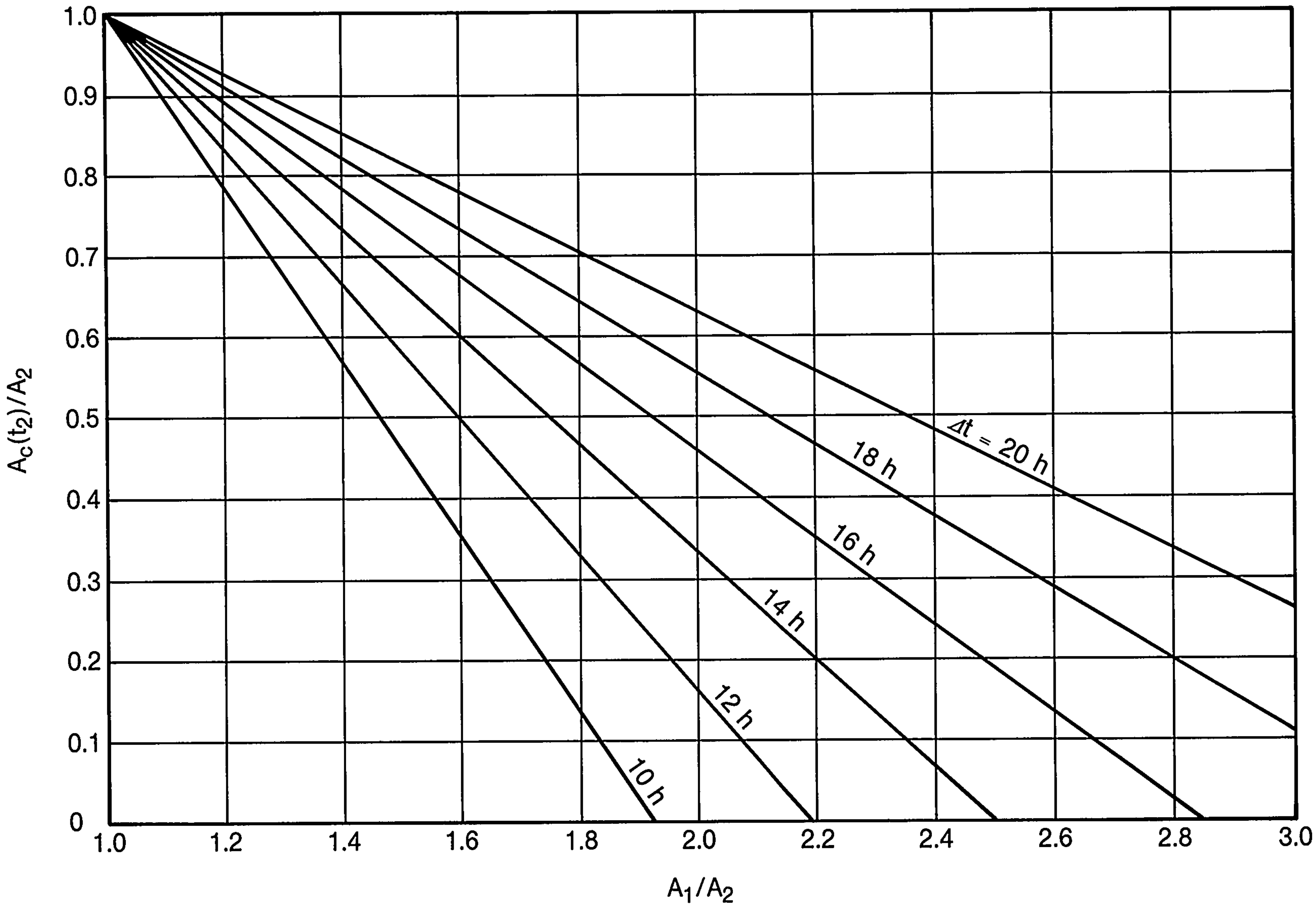


Figure 14.8 Contaminant activity fraction versus activity ratio of two seperated counts.

$$A_{1\alpha}/A_{2\alpha} = 1.2,$$

and from Figure 14.8 we get, for $A_1/A_2 = 1.2$ and $t = 10$ h,

$$\frac{A_c(t_2)}{A_2} = 0.78;$$

therefore,

$$A_c(t_2) = (0.78)(16.7) = 13 \text{ Bq}$$

Thus, we estimate the average concentration during sampling assuming $\epsilon = 1$, as

$$C_{Pu} = \frac{A_{Pu}}{\epsilon f t_s} = \frac{13 \text{ Bq}}{(20 \text{m}^3/\text{h} \times 4 \text{ h})} \sim 0.17 \text{ Bq/m}^3 (10 \text{ dis/min-m}^3).$$

4. Concentration of Short-Lived Emitters

As the half life of the contaminant approaches that of the thoron daughters (10.64 h), calculations using equation 14.18 become increasingly unreliable because the difference

$$e^{-\frac{\ln 2}{(T_{\frac{1}{2}})_c}\Delta t} - e^{-\frac{\ln 2}{(T_{\frac{1}{2}})_{Th}}\Delta t}$$

becomes too small.

Fortunately, short-lived radionuclides have allowable concentration values that are quite large so that the levels of activity of real interest overwhelm the radon-thoron daughter activity on the paper. The latter can be ignored, and concentrations may be estimated directly from equation 14.7. Decay during sampling must be accounted for, since the concentration might otherwise be seriously underestimated.

In many practical sampling situations, the concentration is initially figured from the expression equation 14.8,

$$C_c = \frac{A_c(t_1)}{\epsilon f t_s},$$

and decay is not normally accounted for, so that for short-lived radionuclides, the actual concentration is significantly underestimated. Let C_c in the above be the apparent concentration, then the true concentration C_t will be

$$C_t = C_{app} \frac{t_s \lambda_c}{(1-e^{-\lambda_c t_s})} = \frac{A_c(t_1)\,\lambda_c}{f\,(1 - e^{-\lambda_c t_s})} \qquad 14.23$$

assuming that one may ignore decay between collection of the sample and counting. The above may be rewritten as

$$C_t = C_{app}\, Y(T_{\frac{1}{2}}/t_s), \qquad 14.24$$

in which

$$Y(T_{\frac{1}{2}}/t_s) = \frac{\ln 2\, t_s}{T_{\frac{1}{2}}\left(1-e^{-\frac{\ln 2\, t_s}{T_{\frac{1}{2}}}}\right)} \qquad 14.25$$

is a correction factor. A graph of $Y\,(T_{\frac{1}{2}}/t_s)$ is shown in Figure 14.9.

For very short-lived isotopes ($T_{\frac{1}{2}}$<35 min), a high volume sampler should be used and the duration of sampling should be as short as possible consistent with obtaining a large enough sample. This is because these short-lived isotopes build up to saturation very quickly, and sampling for times longer than about 3 half lives would only allow the radon-daughter activity fraction on the filter to increase.

J. Stack Sampling

Stack sampling is basically air sampling in that air-sampling techniques are used. The effluent being sampled is the "off gas," usually a mixture of air and other gases and particulates from a facility such as a cave, reactor, or isotope laboratory.

It is important to know, or at least to have a reasonable estimate of, the activity released to the environment. Once released, the activity

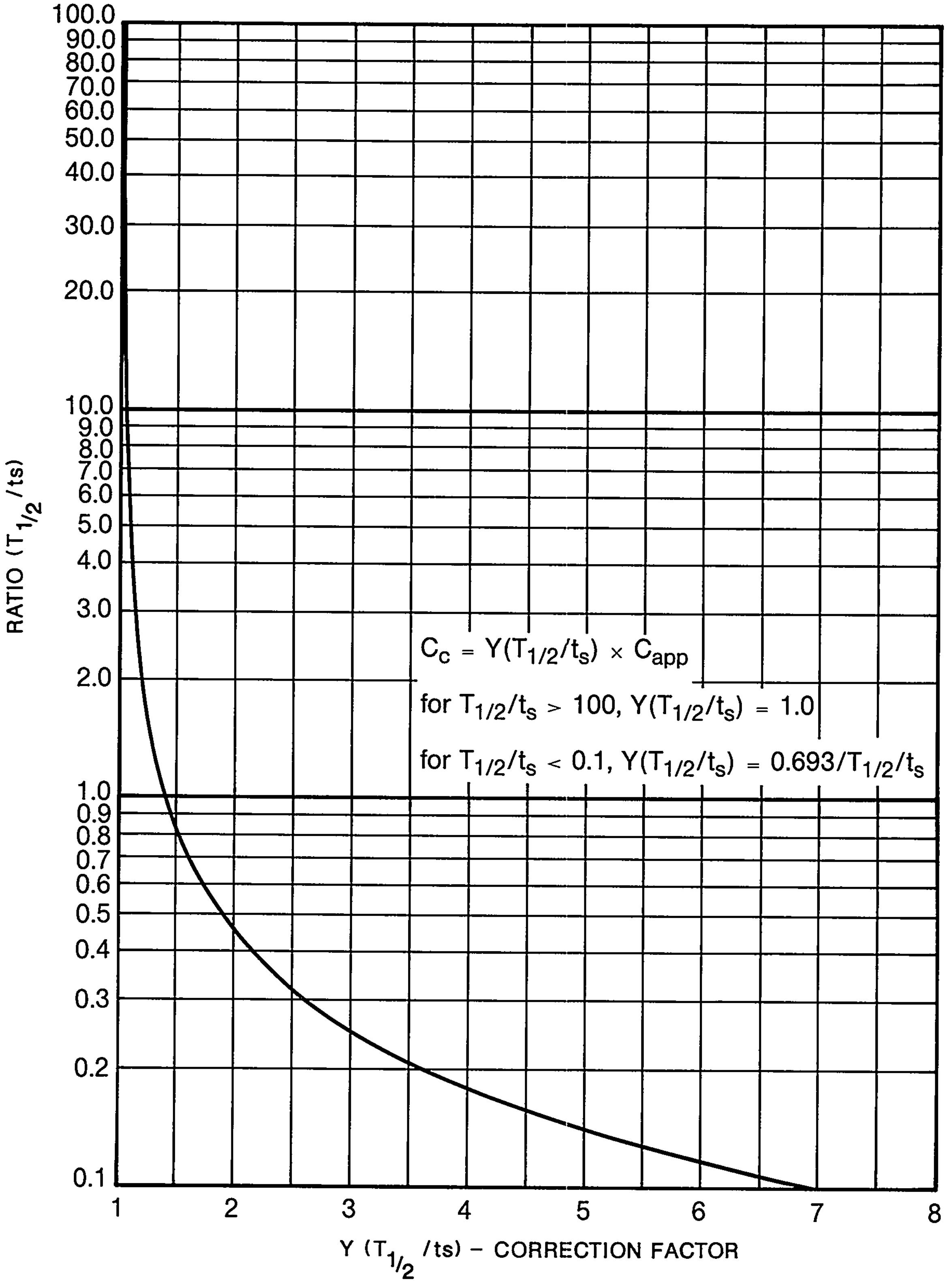


Figure 14.9 Correction factor for short-lived emitters.

is beyond the control of the producer. Moreover, federal regulations require that the radioactive components of the effluent be known. Otherwise, the applicable concentration value is the smallest of the values listed in the regulations. In addition, the regulations require reporting of released quantities.

Stack sampling needs to be performed whenever there is a known, or suspected, source of radiation pollution. A number of objectives need to be satisfied by a given stack sampling program. Among these are:[10]

1. Identify Sources - determine the processes, and radionuclides which are released in the process,
2. Determine compliance - since release of radionuclides are regulated, it is necessary to demonstrate that all releases are within prescribed limits,
3. Monitor performance - establish whether installed controls are performing as designed,
4. Improve control - study data for potential performance improvement by additional controls,
5. Evaluate performance changes - monitor for loss of performance or accidental releases, and
6. Document adequate control - to record satisfactory performance of control mechanisms.

In addition to routine releases, accidental releases also must be reported. Depending upon the magnitude of the release, reporting requirements range from immediate notification, to one-day notification to normal routine reporting.

The Department of Energy requirements for DOE facilities regarding offsite concentration limits can be found in Chapter III of DOE Order 5400.5.[34] This Order contains the revised requirements for environmental protection.

To be able to comply with these rules, we should be able to sample the stack continuously and over a rather large range of concentrations. Identification of nuclides is necessary so that the highest applicable concentration value may be applied.

One big problem in sampling of this kind is the large dynamic range of instrument response required. To cover a range from normal operations below the allowed level to about 10^4 times this level requires a range of about 10^5, and this would be minimal. To achieve this, two detectors having quite different sensitivities or placed in different geometries may be needed.

The simplest monitor which has been employed is a G-M tube in the stack downstream of the filters, reading out to a recorder and set to alarm at some preset level. During normal operations, the recorded level should be more or less constant for a given process - such as reactor operation at constant power. The chart could then be calibrated in terms of Bq/m^3 in the stack effluent if the radionuclides present are known. An abrupt rise in counting rate would suggest a release of some kind, but the nature and extent would not be known without further analysis.

Further analysis can be done if a grab sample can be taken of the effluent at a proper time relative to the release. For example, a sample could be continuously drawn through a filter to trap out particulates. It could then be looked at by a pulse height analyzer for γ emitters. Absorption curves in aluminum and comparison with appropriate standards would help identify β emitters. A charcoal filter could be looked at for iodines or noble gases. Alpha emitters could be monitored with solid state detectors.

A grab sample could be taken, or the flow through a collecting bottle could be isolated at an appropriate time. To ensure that the sample is taken at a meaningful time, it should be taken automatically; a high-level indication on the simple G-M tube monitor could serve as the trigger. Better, a series of such samples should be taken at specified intervals.

Stack monitors, like continous air monitors, are sometimes facility-specific instruments assembled for a particular job. They may be as elaborate or as simple as desired.

With the advent of many different types of detectors, the ability to discriminate among different types of radiation has been enhanced. With these new capabilities, commercially available stack monitors, with the ability to simultaneously measure several different radiation components-α particulate, β particulate, volatiles (iodine, xenon, etc.) and gases, have become available. These systems are contained in compact housings and feature mini-computer analysis of the data, as well as computer controlled readout. Many of these currently available commercial systems are now being used.

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QUESTIONS

14.1 Basically speaking, for what two purposes is air sampling done?

14.2 What are the objectives of air sampling?

14.3 What is the goal in stack sampling?

14.4 Into what two groups are airborne contaminants generally divided?

14.5 What term is used collectively for air suspensions of liquid and solid particulates?

14.6 What type of distribution is observed for the size of particulate matter?

14.7 What is the aerodynamic diameter?

14.8 What particulate sizes are of practical interest? Why?

14.9 Name two radionuclides that comprise the major portion of natural atmospheric activity and identify the radioactive chains to which they belong.

14.10 With what do radon and thoron particulates become associated making them filterable?

14.11 What problem is presented by natural radon and thoron when testing for a particular contaminant?

14.12 Explain the term isokinetic and explain why isokinetic stack sampling is desirable.

14.13 What factors are to be considered in setting up an air sampling program?

14.14 Explain the two basic types of air samplers.

14.15 How do size of sample and concentration of sample relate to sensitivity of the analysis method?

14.16 What is the most common method of sampling for radioactive particulates?

14.17 What corrections must be made in taking a grab sample of the atmosphere?

14.18 Explain the advantage of grab sampling relative to filter sampling and the principal weakness of grab sampling for radioactive contaminants when using a GM tube.

14.19 Explain the principle of operation and the advantages and disadvantages of impingement devices including the cascade impactor.

14.20 What purpose do condensation devices serve?

14.21 Explain the reason for using adsorbers in air sampling?

14.22 Why is a knowledge of absolute collection efficiency of filters important for air cleaning?

14.23 List some of the variables on which the collection efficiency of a filter depend.

14.24 What is the first-count factor, Γ, and why is it used?

14.25 Under what circumstances is the first count factor analysis:

a) not useful and
b) not reliable?

14.26 Why can the radon-thoron activity be ignored from a practical standpoint when the hazards of short-lived isotopes are considered?

14.27 What term is used to indicate the stack effluent that is being sampled?

14.28 What are the objectives of a stack sampling program?

14.29 What big problem exists in selecting stack sampling radiation detection equipment?

14.30 What is the simplest monitor employed in a stack downstream of the filters?

PROBLEMS

14.1 The volume of a grab sampler is 10 liters. The ambient temperature of the atmosphere to be tested is 45°C. The partial pressure of the sampler is 25 mm of mercury and the local barometric pressure is 750 mm of mercury. Find:

a) the volume of the sample at local temperature and pressure, and

b) the volume at standard temperature and pressure:

Answers: a) 9.67 liters b) 8.19 liters

14.2 An air sampler runs 3 hours and 20 minutes at an average flow of 4.25 m^3/min. How many liters of air have passed through the filter?

Answer: 8.5×10^5 liters

14.3 A filter has a collection efficiency of 98.6%. Upon drawing 55 m^3 of air through the filter, a net counting rate of 860 cpm is observed by use of an instrument having a yield of 37%. Find the amount of contamination per m^3, assuming the contaminant has a very long half life.

Answer: 0.714 Bq/m^3

14.4 Freshly extracted Uranium-235, ^{235}U, catches fire in a machine shop. An air filter sample is obtained by a 10-minute run with a 0.085 m^3/min sampler. The filter is counted 24 hours later and an α count of 6650 cpm is obtained. The α background is 5 cpm. The yield of the detector is 28% and the efficiency of the filter is 99.2%. Find the concentration of the contaminant per m^3. Half life of uranium-235 is 7.1×10^8 y.

Answer: 469 Bq/m^3

14.5 An aqueous solution of gold chloride $^{198}AuCl_2$, escapes as a vapor when a glass still explodes. An air filter sampler is obtained by a 1-hour run with a 0.15 m^3/min sampler. The filter is counted 6 hours after the run, the β background is 500 cpm, and the total activity is 30,000 cpm. The yield of the detector is 31% and the efficiency of the filter is 98.6%. Find the concentration of the contaminant per m^3. Half life of ^{198}Au is 64.8 hours. Use the complete formula.

Answer: 192 Bq/m^3

14.6 Use the following table to find:

a) the first-count factor and

b) the α activity (Bq/m^3) resulting from the release of Thorium-232, ^{232}Th, to the atmosphere. The filter sample was taken for 3 hours at 15 m^3/h. Activity values in the table are in Bq.

A	t		
	t_s = 3 h	t_1 = 12 h	t_2 = 22 h
A_β	420	37	17
A_α	170	50	36

Answers: a) 2.47 b) 0.46 Bq/m^3

14.7 Starting from the definition of "first-count factor" derive equations (14.11) and (14.12).

SECTION 15 - REACTORS AND NUCLEAR SAFETY

In many instances, the health physicist uses the instruments discussed in the previous sections. In some cases, the situations are routine. In others, the events may be new or complex and thus may require special methods or devices. The approach chosen is often based on knowledge of the nature of the operation. That is, we may become aware of certain hazards that are present in a given process through analysis of the process itself. If we then know the source of a radiation hazard, we can take steps to measure and control the radiation.

For this reason, our intent in the remaining sections is to first discuss the nature of systems that may lead to radiation hazards. We can then discuss the potential hazards that certain aspects of these processes can produce. This section is concerned with reactors.

A. History of Development

The discovery of the neutron by Chadwick in 1932 led to many studies in which neutrons were used to bombard other substances.[1,2] Among the target elements, Fermi tried uranium. In most other substances, the process resulted in artificially radioactive products, which could be identified. In Fermi's study, slow-neutron capture by uranium produced beta emitters, which were difficult to identify. The new products were first thought to be transuranium elements. That is, neutron capture in ^{238}U, followed by β^- decay, would lead to an unknown element (Z-93). Further β^- decay would then lead to another unknown element (Z=94).

The attention of chemists was directed to separating these products. In 1938, Curie and Savitch obtained a radioactive substance with the chemical properties of lanthanum. This seemed to upset the view that the products were transuranic. Then, Hahn and Strassman showed that the activity was due to barium. At this point, Meitner and Frisch concluded that when uranium captures a neutron, the atom splits into two nuclei of roughly equal size. They referred to this process as <u>fission</u>.

Late in January 1939, Bohr gave an oral report about this new concept at a physics meeting in the United States. This attracted much interest, and studies were begun in this country. Later in the year, Bohr and Wheeler predicted that the fission process was due to slow-neutron capture in the ^{235}U isotope. Work at Columbia University led to an estimate of the cross section for the process. Also, many investigators noted that several neutrons were emitted. This suggested that the process could be self-sustaining.[3]

Since a large amount of energy was released in each fission event, this new process showed promise for military use. For this reason, research work after the summer of 1940 was kept secret. The groundwork leading to military usage was to obtain a self-sustained chain of controlled events. Then, the details could be worked out to result in a fission chain in a very short time. This uncontrolled reaction would result in a tremendous release of energy.

By 1941, Fermi and his group at Columbia had constructed a graphite-uranium cubical lattice. This device was used to study the multiplication of neutrons. Later in 1941, A. H. Compton was placed in charge of this project, which was moved to the University of Chicago. This move occurred as the result of the atomic bomb project authorized by the United States Government. By the summery of 1942, Fermi's work had progressed far enough so that the self-sustained reaction was within sight. The design of a test assembly to achieve the first self-sustained reaction was begun. The device was called a "pile" and consisted of graphite-uranium oxide blocks. The first self-sustaining chain reaction was produced in this pile on December 2, 1942.[3] This day marked the birth of reactors.

B. Fission

The fission process that occurs in some substances forms the basis for reactor operation. That is, once the process is started by a neutron, the result is the emission of more neutrons. Thus, the reaction can be made self-sustaining if fuel is available. Although a reactor is more complex than simply an array of fuel, the heart of the operation is the fission process in the fuel. Consider this process in the fuel substance ^{235}U (see Figure 15.1). A thermal neutron, n_{th}, is captured by the

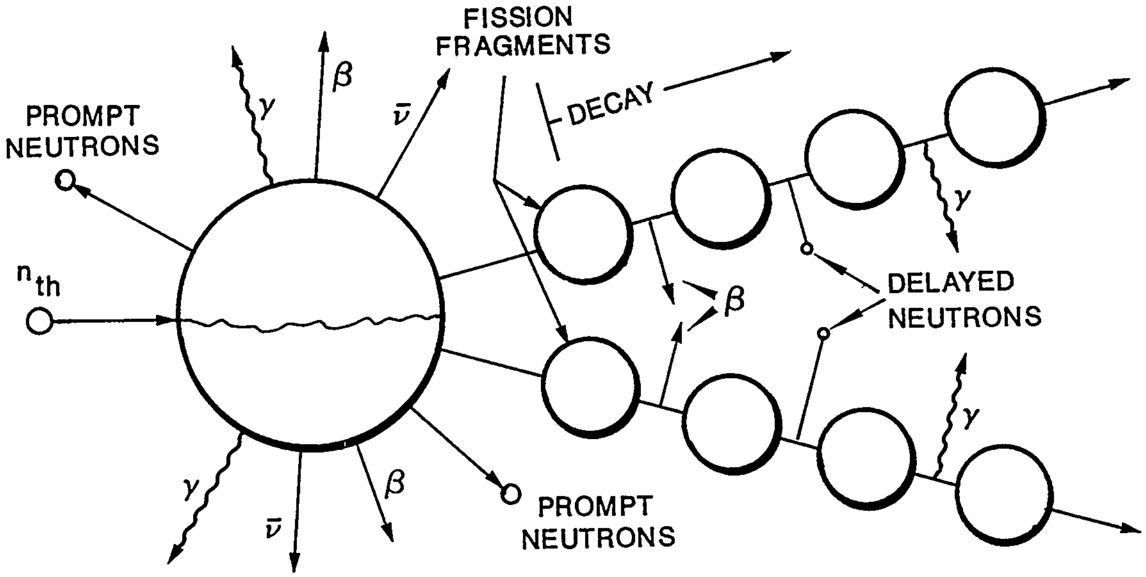


Figure-15.1 Fission process in ^{235}U initiated by a thermal neutron.

^{235}U nucleus. The uranium may just become a ^{236}U atom. This non-fission type capture occurs about 18% of the time.

The other result is that the compound nucleus fissions and the two fission fragments carry off most of the kinetic energy given up in the process. At the time of fission, γ, as well as β and antineutrinos $(\overline{\nu})$ are also emitted. These are referred to as the prompt β and γ. About 9 MeV/fission of prompt betas and about 7 MeV/fission of prompt γ are emitted in ^{235}U fission. Fission fragments come off with velocities of about 10^7 m/s and have ranges up to 30 mm in air.[1] These fragments have too many neutrons to be stable, so they eject one or more neutrons. Most of the neutrons (> 99%) are emitted at fission or slightly after the fission occurs; these are known as prompt neutrons. These are emitted almost isotropically with a spectrum of energies from 0 to 17 MeV. The average energy is about 2 MeV. Some of the neutrons (called delayed neutrons) are released over a period of minutes. These are classified into six groups, which are identified by an assigned half life.[4,5] Delayed groups include neutrons emitted at times $> 10^{-3}$ s after the atom splits.

During the decay phase of the process, more β and γ are given off by the fragments as they decay to stable systems. The entire fission process results in a release of about 200 MeV/atom. Of this fission fragments carry off about 170 MeV. Note that this is the energy released in only <u>one</u> fission. This implies that a tremendous amount of energy can be released from even small amounts of fissioning uranium since so many atoms are available. For example, find the energy release in the complete fission of 1 kg of ^{235}U:

$$\text{Number of } ^{235}\text{U atoms} = \frac{m}{A}(6.022\text{x}10^{23})$$

$$= \frac{1}{0.235}(6.022\text{x}10^{23}) = 2.56\text{x}10^{24}.$$

The total energy released, E_t, will be the product of the number of atoms times 200 MeV/atom:

$$E_t = 2.56\text{x}10^{24} \text{ atoms } (200 \text{ Mev/atom}) = 5.12\text{x}10^{26} \text{ MeV}$$

$$= 5.12\text{x}10^{26} \text{ MeV}(3.83\text{x}10^{-14} \text{ cal/MeV}) = 1.96\text{x}10^{13} \text{ cal}.$$

A mass of one kg of high explosive (TNT) releases about 10^6 cal of energy.[2] A ton of TNT would release about $9.1\text{x}10^8$ cal. Then, the energy released in the fission of 1 kg of ^{235}U would be equivalent to

$$\frac{1.96 \text{ x } 10^{13} \text{ cal}}{9.1\text{x}10^8 \text{ cal/ton TNT}} \sim 21{,}500 \text{ tons of TNT}$$

When a large amount of uranium fissions in a very short time, the energy release produces a terrific blast. This is demonstrated in the "atomic" bomb in which the release occurs in a fraction of a microsecond!

1. <u>Fission Type</u>

The fission process is not restricted solely to slow neutron captures in ^{235}U. Other substances may also be fissioned by slow neutrons. Some may be fissioned by fast neutrons or charged particles as well. Even γ rays produce photofission. Some substances, such as ^{238}U, are fissioned by fast neutrons (threshold ~ 0.65 MeV),[5] but not by thermal neutrons. Many heavy elements, including uranium, undergo spontaneous fission. That is, they break apart in a random process very much like radioactive decay. Thus, we can assign a half life for spontaneous fission to these substances.[6]

We can distinguish between substances that can be fissioned and those in which a chain reaction can also occur. That is, fission has been produced in many substances with $Z \geq 73$. In many of these materials, the fission process cannot be made self-sustaining. For thermal fission, only ^{233}U, ^{235}U, and ^{239}Pu, in which the chain reaction can be maintained, are of value for thermal reactor use.[5]

2. Fission Yield

The term fission yield expresses the percent of fission events that result in a given fission fragment. Since a fission results in two fragments, the total fission yield adds up to 200%. Studies have shown that thermal fission tends to be asymmetric. That is, the two fragments have unequal mass, usually in about the ratio 3/2. A fission-yield curve for thermal fission of ^{235}U is shown as the solid line in Figure 15.2. Note the two maxima in the curve, showing a light and heavy group of fission fragments. These broad peaks occur at about mass numbers 95 and 139 with a yield of around 6.4%.[4] Mass numbers are used since the products in these groups are unstable and emit β. There are about three β decays, on the average, for each fragment, so the atomic number varies greatly, but the mass number remains comparatively constant. When two unequal masses, m_L and m_H, are released in fission, the energy carried off by each is different. From the principles of conservation of energy and momentum, we obtain

$$\frac{E_H}{E_L} = \frac{m_L}{m_H} \qquad 15.1$$

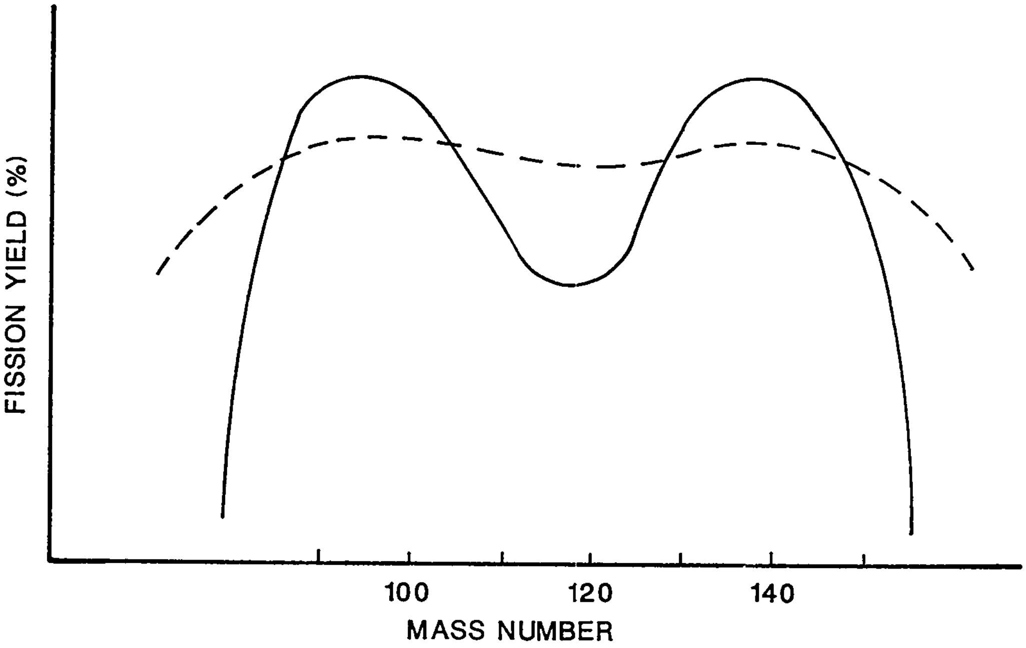


Figure 15.2 Fission yield in ^{235}U. (adapted from ANL-5800.[4])

where the subscripts L and H refer to the light and heavy mass number products, respectively.

Symmetric fission occurs very rarely in thermal fission. However, for fast fission (14 MeV neutrons), symmetric fission occurs much more frequently. This is shown as the dotted curve in Figure 15.2. At even higher neutron energies, one obtains a single peak.

3. Fission Rate

The rate at which fissions occur depends upon many factors, including the fuel properties and the neutron characteristics. Let us define the fission cross section, σ_f, as the probability per atom of fission occurring. Then, σ_f depends upon both the neutron energy and the nature of the fuel substance. In a uniform neutron beam of fluence rate $\phi(n/m^2s)$, the fission rate R (fission/s), in a given fuel substance is

$$R_f = \phi\sigma_f N, \qquad 15.2$$

where N is the total number of atoms of the fuel exposed to the neutrons. Suppose a $2\text{x}10^{-3}$ kg sample of ^{235}U is irradiated by a constant thermal neutron-fluence rate of 10^{17} n/m^2s. Then, if σ_f = 577 barns,[5] what is the fission rate in the ^{235}U sample?

$$R_f = \phi\sigma_f N = \frac{\phi\sigma_f m N_a}{A}$$

$$= 10^{17}(577\text{x}10^{-28})\frac{2\text{x}10^{-3}}{0.235}(6.022\text{x}10^{23}) = 2.957\text{x}10^{13}\ \frac{\text{fissions}}{\text{s}}$$

The total number of fissions in one day would then be:

$$\frac{\text{fissions}}{\text{day}} = 2.957\text{x}10^{13}\ \frac{\text{fis}}{\text{s}}(8.64\text{x}10^{4}\ \frac{\text{s}}{\text{d}}) = 2.55\text{x}10^{18}.$$

If we are dealing with a region in which the neutron fluence rate is uniform, then the fission rate is

$$R_f = \phi\Sigma_f V, \qquad 15.3$$

where Σ_f is the macroscopic fission cross section $(= \sigma_f N_a/A)$, in m^{-1}, and V is the volume of the region in m^3.

C. Nuclear Reactors

As a simple description, consider a reactor as a system in which a fission chain reaction can be safely maintained in a controlled manner. From fission itself, neutrons are available in each event. If we can ensure that at least one neutron causes another fission event, then the chain reaction is possible. On the other hand, we must control the system or the process could result in too great an energy release. The complex form a reactor may take reflects the desire to achieve both these ends.

1. Reactor Components

Many designs have been used in the hundreds of reactors already built. Though individual differences exist, most reactors contain the following components:[1,5,7,8]

(1) Core - The core contains the fuel that fissions to produce the neutrons. The fuel design and array vary greatly depending upon the purpose of the reactor. The core may also contain a substance (called the moderator) used to slow down the neutrons.

(2) Moderator - The moderator slows down neutrons but should not absorb them. Light elements make good moderators since the neutron loses a larger fraction of its energy in elastic collisions with light elements than with heavy elements. Water, D_2O (heavy water), beryllium, and carbon have been used with success in reactor systems. Note that in a fast reactor, a moderator is not needed, since the fissions are caused by fast neutrons, not thermal.

(3) Coolant - The coolant flows through the core to remove the heat produced by the fission process. As fission fragments lose their kinetic energy, this energy generally appears as heat. For reactors that operate at powers greater than about 1 Watt, cooling is required. Many systems have used such substances as water, liquid metal (NaK), and gases as coolants. Cooling is also needed after shutdown in reactors which have been operating for a significant time. The power level will be about 7% of the operating level due to fission product decay, immediately after shutdown.[5,8]

(4) Reflector - The reflector surrounds the core and reduces the leakage of neutrons from the system. The reflector substance must scatter neutrons rather than absorb them. For this reason, materials such as beryllium and graphite, which are good moderators, are also good reflectors. As neutrons leave the core, they encounter the reflector substance, which scatters or reflects them back into the core.

(5) Control System - The control system is needed to start the reactor, to keep it running safely, and to allow shutdown. The system includes the control rods, which contain neutron absorbing substances and thus can be used to regulate the fission rate.

(6) Shield - The shield absorbs the intense radiation emitted both during and after the fission process. Since the main components are neutron and γ, massive shielding may sometimes be required, because of the penetration of these radiations.

2. Critical Size

To maintain the chain reaction, we must conserve the neutrons produced in fission. When thermal fission occurs, neutrons are emitted, usually two or three (2.43 on the average for ^{235}U fission[5]). Emitted as fast neutrons, they move through the core and many events can occur. The neutron may escape, it may be scattered about, or it may be absorbed by some substance in the core. The reaction rate for any of these events depends upon the neutron fluence rate and the cross section for that type of event. Since other processes compete with the fission for neutrons, we must supply a large number of neutrons to keep the chain going. That is, a certain amount of a fissionable substance must be present before a self-sustained chain reaction can take place. This amount is called the critical size. The critical size is reached when the number of neutrons lost is balanced by the number produced. The value of the critical size in a given system depends upon several factors.

Suppose we have an assembly of uranium fuel, moderator, and reflector as shown in Figure 15.3. In this case, the use of a moderator implies that the device is designed for thermal fission. Since ^{238}U is not fissioned by slow neutrons, the fuel must contain ^{235}U. Natural uranium contains about 0.72% of the isotope ^{235}U, but quite often the fuel material is enriched uranium. That is, the abundance of the ^{235}U is increased above that found in normal uranium. Now, assume a stray neutron is captured by a ^{235}U atom. This can be a non-fission capture,

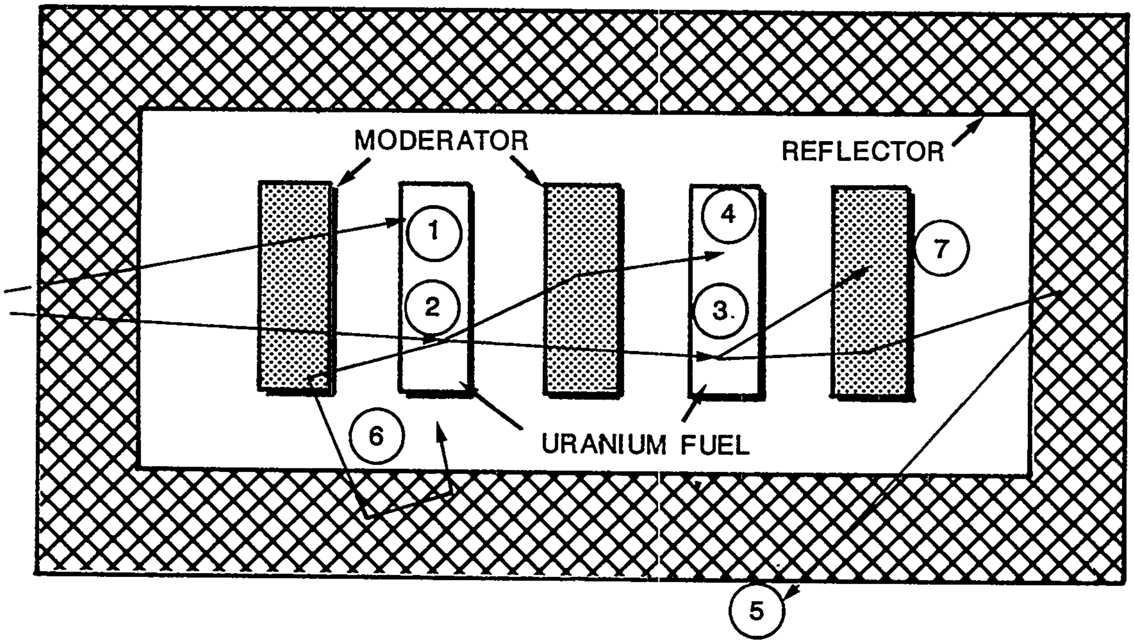


Figure 15.3 Nuclear assembly for thermal fission.

which leads to ^{236}U ($T_{½}$ = $2.34x10^7$ y; 4.49 MeV α) and no release of neutrons (1). It may result in fission, which leads to the release of two or three neutrons (2). These fast neutrons may engage in many processes, such as absorption in ^{238}U. Fast neutron absorption in ^{238}U may also produce fission if the neutron energy is above 0.6 MeV. This results in more neutrons being released (3). The resonance capture of neutrons in ^{238}U results in the formation of plutonium by the following mechanism:

$$^{238}U + {}^1n \rightarrow {}^{239}U \rightarrow {}^{239}Np + \beta^- \rightarrow {}^{239}Pu + \beta^-.$$

If the process of resonance capture occurs, no neutrons are released (4).

The neutrons may collide with atoms of uranium, of the moderator, and of the reflector substance. These collisions slow down the neutron and change its direction, but the neutron may still escape from the system (5). On the other hand, it may be slowed down and reflected back toward the fuel to be exposed once again to the above processes (6).

During slowing down, the neutron may be absorbed by some impurity atom (7). Substances such as boron, cadmium, or rare earth impurities may have high capture cross sections for neutrons. This type of capture removes neutrons without releasing any more.

To sustain a chain reaction in such a system, we must make the output from processes (2) and (3) equal the loss due to processes (1), (4), (5), and (7). The critical size will be a function of the type and purity of the fuel, the neutron energy used for fission, and the type, nature, and placement of the moderator and reflector. If one adds a coolant and a control system, these features would also affect the critical size of the unit. For a given choice of parameters, we have to supply a certain amount of fuel to arrive at a balanced state. A system that is below this critical size is called subcritical; one that passes the critical size is called supercritical.

3. Four-factor Formula

We can approach reactor design from the standpoint of the history of a typical neutron in the system.[5,7] Suppose a thermal neutron is captured in the ^{235}U fuel and produces fission. On the average, the number of neutrons released will be 2.43. However, a neutron could have been captured without producing fission. We define the factor η as the average number of fast neutrons produced in the fuel per thermal neutron absorbed. Since some of the neutrons absorbed in ^{235}U do not produce fission, $\eta < 2.43$. On the average, we expect η fast neutrons to be produced for each thermal neutron captured in ^{235}U.

As the fast neutrons move through the system, they are slowed down. If they are captured as fast neutrons in ^{238}U, a fast fission effect can result. The fast-fission factor, ϵ, gives the increase in the number of neutrons due to fast fission. In some systems, this factor has a value of about 1.03. The effect is small since the threshold is about 0.6 MeV and most neutrons are quickly reduced to energies below this in just a few collisions. At this point, we expect to have $\eta\epsilon$ fast neutrons.

Elastic and inelastic collisions in the system reduce the energy of the fast neutrons. At lower energies the elastic collisions predominate, and at $E < 1$ keV,[1] resonance absorption in ^{238}U can occur. As the neutrons are further reduced in energy, some may be lost in capture. Those not lost are reduced in energy until they become thermal neutrons. The resonance escape probability, p, is the probability that a neutron escapes capture to reach thermal energy. This factor depends upon the amount and placement of both the fuel and the moderator and the type of moderator substance. Usually, some neutrons are captured so that p is less than unity. In this case, we expect to end up with $\eta\epsilon p$ thermal neutrons.

The thermal neutrons drift through the volume of the system. As these neutrons diffuse through the system, some are absorbed in the fuel and others are absorbed by impurities. The thermal utilization factor, f, gives the fraction of thermal neutrons absorbed by the fuel, compared to all the thermal neutrons absorbed in the system. Thus, the number of thermal neutrons absorbed in the fuel is given by $\eta\epsilon pf$.

The value of the product $\eta\epsilon pf$ in a given design indicates whether multiplication of the neutrons in the system occurs. The multiplication factor, k_∞, may then be defined by the four-factor formula:

$$k_\infty = \eta\epsilon pf. \qquad 15.4$$

This factor, k_∞, gives the ratio of the number of neutrons in succeeding generations. For the number of neutrons to increase during each cycle will require that $k_\infty > 1$.

4. Effective Multiplication Factor

In arriving at k_∞, we neglected the neutrons that may escape from the system. That is, k_∞ is the multiplication in a system of infinite extent since no neutrons are lost from the system. In any practical design, some neutrons diffuse out from the surface of the reactor and are lost. This "leakage" reduces the actual multiplication by a factor of L. This non-leakage factor is the fraction of neutrons that do not escape. We can define an effective multiplication factor, $k_{eff} = k_{\infty}L$, in which $L < 1$. The value of the non-leakage factor, L, is a function of the reactor size. For a small system, the leakage is large because the surface-to-volume ratio is high. For larger systems, this ratio decreases and the leakage is smaller.

We can describe the operation of the system in terms of the value of k_{eff}. For $k_{eff} = 1$, the reactor is just critical. Neutrons are produced at the same rate as they are lost, and the system is balanced. If $k_{eff} < 1$, the assembly is subcritical, and more neutrons are consumed than are produced. In this case, the chain reaction is not self-sustaining. For $k_{eff} > 1$, the system is said to be supercritical, and the number of neutrons increases with each generation.

5. Reactor Control

Reactor control can be approached from the standpoint of the change in the neutron population in the system. Assume that k_{eff} can be made greater than one so that a power output can be obtained. To maintain a chain reaction at least one neutron per fission is required. This means that the number of neutrons increases in each cycle by the fraction $k_{eff}-1$. This quantity, $k_{eff}-1$, often written as Δk, is called the excess reactivity.

If we assume a number N_o neutrons at time t=0, and a neutron generation time τ_o, the number of neutrons at time t will be

$$N = N_o e^{\Delta k/\tau_o}. \qquad 15.5$$

In a thermal reactor, τ_o may be around 10^{-4} s for the prompt neutrons.[5] Then, if $k_{eff} = 1.002$, $\Delta k = .002$ and if $N_o = 1$ and $t = 1$ s,

$$N=(1)e^{.002(1)/.0001}=e^{20}=4.85\times10^{8}$$ neutrons for each initial neutron for each second.

This rapid rate of increase is much too dangerous for safe control.

If we consider only the prompt neutrons, control of the reactor is difficult. The presence of the delayed neutrons from the fission process, less than 1% of the total neutrons, allows for better reactor control. There is a time lag before the delayed neutrons are emitted, which increases the generation time of all fission neutrons. The mean lifetime for the delayed groups in ^{235}U fission is 0.085 s.[5] Thus, the generation time for all fission neutrons becomes:

$$\tau = 0.0001 + 0.085 = 0.085 \text{ s}$$

In the same example as before

$$N = N_0 e^{\Delta k t/\tau} = e^{0.002(1)/0.085} = e^{0.024} = 1.27 \text{ neutrons.}$$

In this case, the increase is not so very rapid, and we have time in which to achieve control. However, this means that the amount of excess reactivity must be limited.

a. Reactor Period

The reactor period, T, is useful for expressing the variation in neutron population with time. The period T is the time required for the neutron-fluence rate to increase by the factor e. From equation 15.5, T is given by

$$T = \frac{\tau}{\Delta k} \text{ (s)}. \qquad 15.6$$

From the standpoint of reactor control, the period must not be too short. If we use our example of $\Delta k=0.002$ and $\tau=0.085$ s, then T = 0.085/0.002 = 42.5 s. This period allows ample time for reactor control. Now, suppose the prompt neutron generation time τ_0 = 10^{-4} s is used; then T = 0.0001/0.002 = 0.05 s. In this case, the increase is so rapid that safe control of the reactor would be in jeopardy. Any control system would have to operate almost instantly. We could not afford any delay, such as the reaction time of the person operating the system. Thus, one would need automatic shutdown systems which must activate within a few ms.

b. Reactivity

In the steady-state condition, the neutron population is composed of both prompt and delayed neutrons. Any change in the system tends to affect the neutron multiplication. A change affects the prompt neutron population immediately but does not affect the delayed group immediately. The effect upon the neutron multiplication is described by the reactivity, ρ,

$$\rho = \frac{\Delta k}{k_{eff}}, \qquad 15.7$$

where Δk expresses the effect on the multiplication factor brought about by the system change, and k_{eff} is the value before the change occurred. For most systems, k_{eff} is close to unity, so that $\rho \sim \Delta k$. Up to this point, we have implied that a reactor can be controlled because of the presence of the delayed neutrons. However, the effect of the delayed neutrons may be negated under the right circumstances. Suppose the fraction of delayed neutrons is β. The values of β for ^{233}U, ^{235}U and ^{239}Pu are 0.026, 0.0065 and 0.0021, respectively.[5] Then for any $\Delta k < \beta$, the delayed neutrons control the rate of rise in the neutron population.[1] This is known as the delayed critical condition. When $\Delta k = \beta$, the chain reaction is maintained by the prompt neutrons alone (<u>prompt critical condition</u>). For $\Delta k > \beta$ the multiplication of the neutron fluence rate occurs quite rapidly; therefore, this state should be avoided.

c. Reactor Control Systems

Several aspects must be considered when we view control systems for reactors. That is, not only mechanical systems, such as the control rods, are effective, but also certain processes and features of the design itself may be important.[1,2,5,7]

Two of the foremost built-in features that are used to advantage in many thermal reactors are the temperature effect and the void effect. That is, as the reactor power increases, the core temperature increases. This raises the energy of the thermal neutrons and decreases the reactivity of the system. Also, the expansion of the moderator substance as it is heated results in a density decrease, which tends to reduce reactivity. In a water moderator, bubbles (voids) may appear which also decrease the reactivity. Thus, when reactivity is increased, the above inherent features tend to counteract the increase. For fast reactors, however, loss of coolant may lead to increased reactivity.[5]

The fact that the fission process uses up fuel atoms is a prime reason why reactors should be supplied with excess reactivity. To allow for burnup, the fuel loading is more than the minimum needed for the chain reaction. A neutron absorber may then be included to balance the excess reactivity of the fuel. With use, both the fuel and the absorber are "burned-up" so that the balance is somewhat maintained.

However, as the fuel is depleted, some of the fission products formed tend to strongly absorb neutrons. This loss of neutrons decreases reactivity, since it lowers the value of f. These products are called <u>poisons</u>, and ^{135}Xe is the most troublesome.[5] Of the others, ^{149}Sm is next in importance.[8]

The product ^{135}Xe, formed both by the fission process and the decay of ^{135}Te, has a thermal neutron absorption cross section of 2.7×10^6 barns.[2] During steady operation, the ^{135}Xe concentration reaches an equilibrium value that depends upon the neutron fluence rate in the system. When the system is shutdown, the decay chain of ^{135}Te leads to an increase in ^{135}Xe, which reaches a peak value in about 12 hours.[2,8] Unless the system has been designed with sufficient reactivity, this effect hinders reactor start-up shortly after a shutdown.

In many instances, the control system refers to the mechanical systems. For thermal reactors, the control is accomplished by use of rods containing neutron-absorbing elements (B_4C, Ag-In-Cd, Cd, B, Hf). For fast reactors, the system may contain control rods, movable fuel rods, or a movable reflector. Control rods can be grouped into three types: shim, regulating, and safety rods. Shim rods are used for coarse control; that is, they bring the reactor to about the desired power level. Regulating rods are then used as a fine control to keep the reactor at the given power level. Safety rods aid in quickly shutting down the reactor (called a "scram"). In terms of reactivity, each of these rods has a certain value. Thus, by, insertion or withdrawal, we can affect the neutron multiplication.

In addition to the normal control system, we may have emergency systems, which allow for a quick scram.[5] In some systems, the moderator can be quickly dumped from a core vessel to aid in shutdowns. Sometimes a gas purge is used to introduce voids in the system. In other cases, gases and/or liquid poisons can achieve the needed effect. For more discussion on control systems see References 5 and 8.

6. Reactor Materials

Many problems in reactor design center about the choice of materials. This can be quite a task since core components must meet stringent requirements. Not only must the substance have a low neutron-absorption cross section, but it must have good high-temperature properties. It should transfer heat quickly and easily so that materials do not expand or buckle out of shape. Also, the substance should resist corrosive agents and radiation damage.

The features needed for substances used outside the core are not as stringent as those for core materials. However, the task of arriving at a final design for an entire reactor involves contributions from many sources. For this reason, the subject is much too complex to treat in a short outline. The subject of shielding design alone requires extensive and complex calculations as well as attention to the material properties.[6,8-11] Information concerning the features and engineering aspects of reactor design is contained in References 5, 12, 13 and Chapter 15 of Reference 1.

7. Power Level

Relating the neutron fluence rate to the power level is not easy, except for certain special designs.[1] However, we can directly relate the power level and the fission rate by means of the energy released per fission. The watt (W) is a unit of power that equals 1 J/s (10^7 ergs/s). Of the 200 MeV released per fission, about 10 MeV is carried away by anti-neutrinos and does not appear as heat. The energy, in ergs, per fission which does contribute to the thermal power is

$$190\ (\frac{\text{MeV}}{\text{fis}})1.6\text{x}10^{-6}(\frac{\text{ergs}}{\text{MeV}}) = 3.04\text{x}10^{-4}(\frac{\text{ergs}}{\text{fis}}).$$

For a power level of 1 W, the fission rate is

$$\text{fission rate} = \frac{10^7\ (\frac{\text{ergs}}{\text{sW}})}{3.04\text{x}10^{-4}(\frac{\text{ergs}}{\text{fis}})} = 3.3\text{x}10^{10}(\frac{\text{fis}}{\text{sW}}). \qquad 15.8$$

Suppose we calculate the total atoms per day that are fissioned in a 1-MW (10^6W) reactor operating for a full 24 h:

$$\text{1-MW} = 10^6\text{W} = 3.3\text{x}10^{10}(\frac{\text{fis}}{\text{sW}})10^6\text{W} = 3.3\text{x}10^{16}(\frac{\text{fis}}{\text{s}}). \qquad 15.9$$

Then,

$$3.3\text{x}10^{16}(\frac{\text{atoms}}{\text{s}})8.64\text{x}10^4(\frac{\text{s}}{\text{d}}) = 2.85\text{x}10^{21}(\text{atoms/d}).$$

From Section 15.B, we found that 1 kg of ^{235}U consists of 2.56x 10^{24} atoms. Therefore,

$$\frac{2.85\text{x}10^{21}\ \text{atoms/d}}{2.56\text{x}10^{24}\ \text{atoms/kg}} \sim 1.1\text{x}10^{-3}\ \text{kg/d of}\ ^{235}\text{U}(1.1\ \text{g})$$

Thus, a 1 MW reactor burns about 1 g of fuel per day.

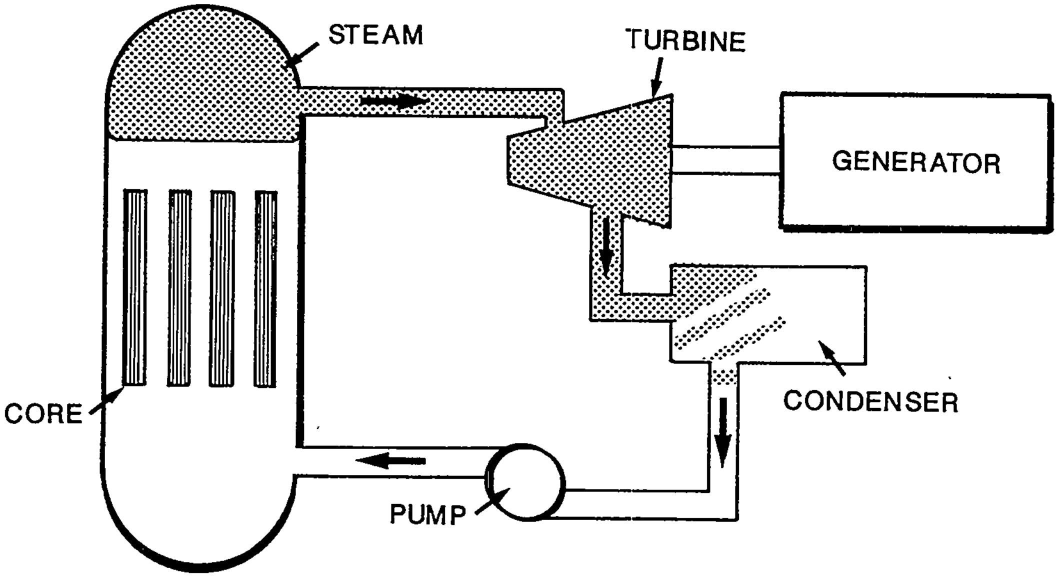
8. Reactor Types

We can class most common reactors into four main categories: research, breeder, power, and isotope production.[1,5,7,8] Research reactors may also be used as a system for training. The features of research reactors are discussed in a general way in Reference 1. The principal purpose of a breeder reactor is to produce ^{239}Pu or ^{233}U. In these systems, fertile materials (^{238}U and ^{232}Th) capture neutrons, and the subsequent decay leads to fissionable material.[5] Power reactors convert the fission energy to electrical energy, which is then used to supply power.[1,5] Most commercial power reactors in the U.S. are pressurized water reactors (PWR) with about 1/3 being boiling water reactors (BWR).[5] In the BWR (see Figure 15.4a), the water enters the core and absorbs heat from the fissioning fuel. Steam is formed and is directed to the turbogenerator to produce the electricity. In this design, the water in the reactor vessel is both the coolant and the source of the steam. In order to have enough energy to drive the turbine, the steam must be produced at a high temperature. To accomplish this, the BWR is operated at approximately 1000 psi, which raises the boiling point of the water. Then, the operating steam temperature is about 285°C (545°F).[5] Not shown in the figure, is the necessary containment structure designed to prevent release of steam and radioactivity in an accident.

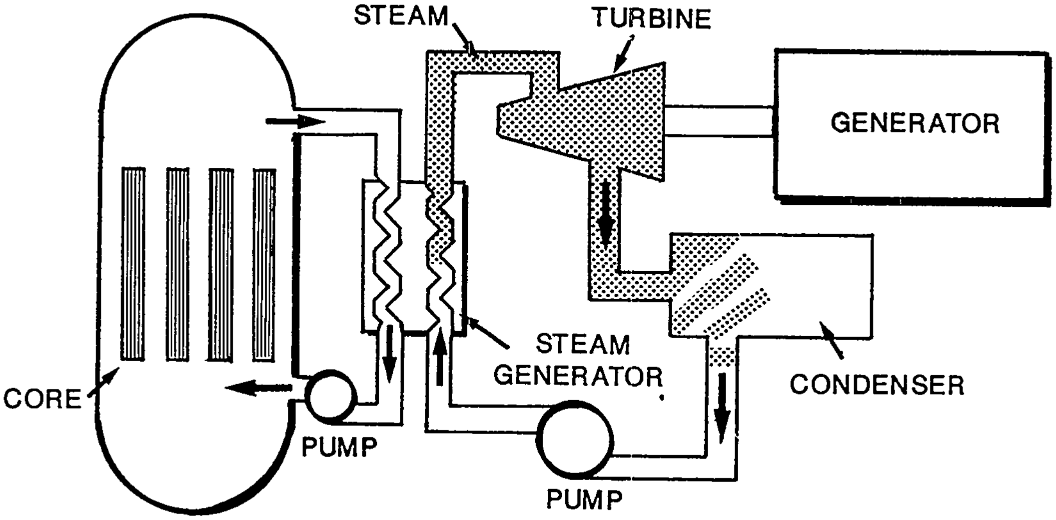
In the PWR (see figure 15.4b), the water going through the reactor vessel does not boil. The pressure is kept at about twice what it is in a BWR and water can be heated to > 320°C (608°F) without boiling. The heated water is directed to a steam generator, where the heat is transferred to another water system at a lower pressure, causing the water to boil and form steam. This steam, at around 260°C (500°F), is then used to generate the electricity. In the PWR, two separate water systems, which do not mix, form the steam supply. A PWR also requires a substantial containment structure to prevent release of radioactivity and steam in an accident, such as loss of coolant.

D. Fusion Power

In order to realize fusion, two light nuclei must overcome the electrical repulsive force between them.[5,14] As the kinetic energy of each component increases, the chance of fusion occurring also increases. This is because the particles may acquire enough kinetic energy to overcome the repulsive force. To improve the probability of obtaining fusion, one should choose nuclei with the lowest possible nuclear charge.[1] This will minimize the energy needed to overcome the repulsive force. For this reason, nuclei of hydrogen show the most promise.



(a) BOILING WATER REACTOR



(b) PRESSURIZED WATER REACTOR

Figure 15.4 Power reactor designs.

As discussed in Section 1.G.3, energy is given up when two light nuclei unite to form a heavier nuclide. To use this to advantage; however, requires that a significant number of atoms undergo this reaction. In an accelerator, particles acquire enough energy for these reactions, but most of the particle energy is dissipated as heat. If one could contain this heat in a small, enclosed volume without loss, then fusion would be possible.

An alternate approach is to heat the atoms to an elevated temperature, this will increase the energy of the atoms. However, this requires heating the atoms to many millions of degrees. Unfortunately, at these temperatures, the atoms also do not want to stand still so that confinement remains a problem. If one could ensure that the heated atoms remained in a confined space for a specific time, then repeated collisions among the high-energy atoms would lead to fusion. So, one must be able to sustain a certain density of nuclei.

The necessary confinement time is that in which the recovered energy (usable energy) from the fusion reactions equals the energy input and energy losses in the process. This is the so-called "break even" time. The relationship between the nuclei density, confinement time and the break even time was developed by Lawson.[1] For n (nuclei/m^3) and t (confinement time in s), the Lawson criterion is

$$nt > 10^{20} \qquad 15.10$$

The above relationship implies that for a reasonable density range of 10^{20} to 10^{22} nuclei/m^3, the necessary confinement time is between 1-.01 s.[14]

In the process of heating atoms to a very high temperature (approximately 100 million degrees), a plasma is formed. A plasma is an ionized gas composed of equal numbers of positive and negative ions. Since the numbers of ions are equal, the plasma as a whole is electrically neutral.[14] However, because the negative ions (electrons) move within the plasma, they interact with nuclei of atoms (positive ions), resulting in bremsstrahlung. This represents emission of photons which reduces the energy content in the plasma. This energy loss mechanism must be overcome for the fusion process to become self-sustaining. As a consequence of this and other loss mechanisms, the estimated practical temperature necessary for production of self-sustaining fusion is around 100 million degrees (minimum) for the most plausible fusion reaction, deuteron-tritium (D-T) reactions. From Section 3.F.1.f, this reaction is

$$^{2}_{1}\mathrm{H(D)} + {}^{3}_{1}\mathrm{H(T)} \rightarrow {}^{4}_{2}\mathrm{He} + {}^{1}_{0}\mathrm{n}(14.1\ \mathrm{MeV}).$$

For the deuteron-deuteron reaction also discussed in 3.F.1.f, the required temperature would be about an order of magnitude higher. On this basis, the first generation fusion reactor will probably utilize the (D-T) reaction shown above.[14,15]

Besides the bremsstrahlung from the plasma, loss of energy occurs if nuclei strike the vessel wall. When high energy nuclei collide with the wall, cooling of the plasma occurs as well as removal of wall atoms. The latter process tends to increase the pool of nuclei available for interaction with electrons.

In order to minimize the energy losses occurring in these processes, the plasma must be confined. Two methods are being studied to obtain this confinement: inertial and magnetic confinement.[16] In the former method, one approach uses D-T contained in a small pellet which is rapidly heated to fusion temperatures by lasers. The heating occurs so quickly that about 20% of the fuel may react before disassembly occurs.[14] In magnetic confinement, the pressure exerted by the plasma particles is balanced by a large magnetic field (several tesla, at least). The particles are thereby confined to a controlled orbit. Several other approaches to obtaining the necessary confinement are discussed in Reference 14.

On the basis of a D-T reaction, a fusion reactor will require a large inventory of tritium. Deuterium is readily available since it is found naturally in hydrogen (0.015%), but tritium is rare. However, tritium can be produced by these reactions:

$$^{6}_{3}\mathrm{Li} + {}^{1}_{0}\mathrm{n}_{\mathrm{thermal}} \rightarrow {}^{3}_{1}\mathrm{H} + {}^{4}_{2}\mathrm{He} + 4.8\ \mathrm{MeV},$$

and

$$^{7}_{3}\mathrm{Li} + {}^{1}_{0}\mathrm{n}_{\mathrm{fast}} \rightarrow {}^{3}_{1}\mathrm{H} + {}^{4}_{2}\mathrm{He} + {}^{1}_{0}\mathrm{n}_{\mathrm{thermal}} - 2.5\ \mathrm{MeV}$$

In the latter reaction, energy is absorbed but the thermal neutron produced can produce more tritium in ^{6}Li.

The first generation fusion reactor is likely to be in the form of a toroidal chamber, called a tokamak (from the Russian words for toroidal, chamber and magnetic, respectively).[14] Such a chamber would be surrounded by a lithium-containing blanket and a radial shield (see Figure 15.5). Not indicated in the figure is the placement of the confinement system. As a result of fusion in the toroidal chamber, neutrons will be emitted from the plasma and will react in the blanket. In addition to producing tritium, these neutrons will heat up the blanket. The heat is removed by gas or liquid coolant from the blanket and used to produce electricity. The tritium is recovered from the blanket region and is used as fuel with new deuterium. The shield acts to stop bremsstrahlung emitted from the plasma and the intense neutrons produced in fusion, which are slowed down but not absorbed in the blanket.

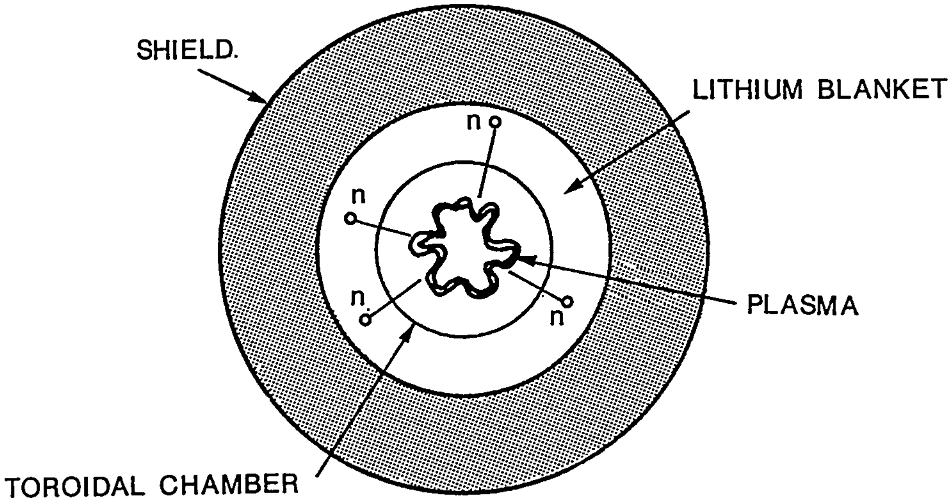


Figure 15.5 Cross - sectional view of a toroidal fusion reactor.

Although the technology is difficult to achieve, the attainment of a practical fusion device is under development. Nuclear fusion is a very attractive source of energy because of its inherent safety, inexpensive and abundant fuel, and ease of control.[14]

E. Radiation From Reactors

A number of processes give rise to potential radiation hazards in reactors. Not all these sources are found in each system, but some will be common to all systems. For any given reactor, the specific problems depend somewhat on the design features. In this discussion, the factors that lead to radiation problems will be pointed out without regard to any special nuclear system. For a discussion of radiation releases associated with power reactors, see Reference 17.

1. Leakage Radiation

The fission process gives rise to a number of products: fission fragments, β particles, γ rays, and fast neutrons.[11,17,18] The decay of the fragments results in the fission-product chains, which produce more β and γ as well as the delayed neutrons. The β and the fission fragments are quickly stopped in the fuel so that they do not contribute to any external radiation field. Most of the energy from these products appears as heat, except for a small amount of bremsstrahlung.

The neutrons in the system are slowed by scattering processes in the fuel, moderator, and reflector. If inelastic scattering occurs, γ rays (1-5 MeV) may be emitted. Elastic scattering reduces the neutron energy and may lead to capture of the neutron or its escape from the system. Capture in the fuel (assume ^{235}U) may or may not lead to fission (see Section 15.B). Capture in some other substance may result in capture γ rays (0.5-10 MeV) or activation of the substance. The latter process may then lead to the emission of γ rays (0.1-3.0 MeV) in the decay process. The neutrons that escape from the system are exposed to scattering and absorption processes in the shield. If they escape the shield, they become part of the external radiation field. Capture of neutrons in the shield substance may result in the emission of γ rays. In all neutron interactions that lead to γ rays, the photons may escape from the system and become part of the external field or leakage radiation.

The photons, either prompt or delayed, that result from the fission process may vary in energy from about 0.05 to 8.0 MeV.[9,19,20] These photons may interact with the light elements of the moderator to produce photoneutrons ($E < 2$ MeV). They may also be attenuated by the core materials and the shield. Some of these photons may escape from the shield and add to the external field.

Capture photons which result from absorption of neutrons occur when the fast neutrons are reduced in energy below about 2.5 keV. This can be a significant source of radiation. The capture of the low energy neutron results in the emission of a high energy γ (up to approximately 10 MeV). This may be the most important source of leakage radiation in a reactor. That is, one may effectively attenuate the neutrons down to intermediate energy, but then create a photon problem.

In addition to the above mentioned photons, bremsstrahlung produced by stopping β particles and annihilation photons are also produced. Activation of certain materials by neutrons leads to radioactive products which also emit γ as they decay.

About equal numbers of photons and neutrons are emitted in the fission process. For this reason, both photons and neutrons are important from the shielding standpoint. The shield must be designed to reduce the fluence rates of these products to the desired levels. For most reactor systems, the shielding provided is sufficient to ensure a negligible leakage-radiation field for normal situations. In many systems, though, beam holes and irradiation ports or trays are included so that the integrity of the shield is altered because of these penetrations. These arrangements may result in increased leakage radiation due to streaming of the radiation. The presence of holes in the shield increases the chance of a significant void in the shield structure. There is also the danger of radiation beams if a shielding plug has not been replaced correctly.

2. Sources of Contained Activity

Once a reactor has been in use for some time, it becomes a source of radiation even when it is shut down. That is, the fission products build up to some level while the system is in operation; on shutdown, each product decays with its own half life.[8,17] Since many fission-product chains are involved, the composite decay lasts for quite some time. In this case, the decay cannot be given by a single half life for the entire group. The approximate decay rate for the group is proportional to t^{-n}, where t is some time after fission and $n = 1.2$. If the system has been operating for a long time, the value $n = 0.2$ should be used. The presence of this γ radiation precludes access to the core for systems that run at high power for a long time. For low-power runs of short duration, access to the core may be permitted after a short delay in some cases. This will often involve only a limited exposure time.

For extended reactor operation time T(s), one may estimate the fission product activity A, within a factor of two or better, from[8]

$$A \sim 5\times10^{11} P[\tau - T)^{-0.2} - \tau^{-0.2}]\mathrm{Bq} \qquad 15.11$$

$$(14P[(\tau - T)^{-0.2} - \tau^{-0.2}]\mathrm{Ci}),$$

in which P is the thermal power in watts, and $\tau - T$ is the time after shutdown (s). In the case of short operation, or an excursion, the fission product activity A will be proportional to $t^{-1.2}$

A Rule of Thumb for calculating the fission product activity for a reactor, operated for an extended time in a steady state (such as a power reactor), and shutdown for a time T(y) is [21]

$$A = \frac{3.7\times10^{12} E}{T}\ \mathrm{Bq}\ \left(\frac{100\ E}{T}\ \mathrm{Ci}\right),\ T > 100\ \mathrm{d}, \qquad 15.12$$

in which E is the total thermal energy produced during operation (MWd). For example, assume a 1 MW thermal power reactor operates for 1.5 y, estimate the fission product activity present after 146 d of shutdown.

$$A = \frac{3.7\mathrm{x}10^{12}(1\ \mathrm{MW})(547.5\ \mathrm{d})}{146/365} = 5.064\mathrm{x}10^{15}\ \mathrm{Bq}(1.37\mathrm{x}10^{5}\ \mathrm{Ci}).$$

When samples or other components are inserted into a reactor, they become activated. When they must be removed, they constitute a source of potential external exposure. If fuel must be removed or changed or reactor maintenance is required, these procedures also present radiation-exposure situations. Corrosion products, such as ^{58}Co, ^{60}Co, ^{54}Mn, ^{55}Fe, ^{59}Fe and ^{51}Cr, are formed from irradiated steel components.

For systems in which the coolant is part of a closed cycle, we may be faced with a problem of neutron-activation products. The magnitude of the effect depends upon a number of factors, including the type of coolant and the reactor power. For any coolant, foreign substances in the coolant act as poisons in the system. Once these substances absorb neutrons and become radioactive, they may deposit on the inner surface of the cooling system. When the system lines run outside the shield, extra shielding may be required. The coolant passing through the system may also acquire activated scale, which can also then be deposited at points in the cooling system. Of course, any defects in the cladding of the fuel can also result in fission products in the coolant.

Even if the coolant were free of impurities, the passage through the high neutron fluence in the core would result in coolant activation. With liquid metal coolants, sodium and potassium both become quite activated. For water coolants, the oxygen absorbs neutrons and becomes radioactive nitrogen and oxygen. The most important products are ^{16}N and ^{17}N, which are short-lived (a few seconds). Since these are gases, they tend to escape from the system, but any delay results in significant decay, which reduces the problem.

When Na is used as the liquid coolant, this results in the production of ^{24}Na and ^{22}Na during operation. Although production of ^{24}Na is greater than that of ^{22}Na during operation, the ^{22}Na ($T_{½}$ = 2.64 y) becomes a long-term problem. In the case of power reactors, the Li contained as a contaminant in Na can lead to tritium production via $^{6}Li(n,^{3}H)\alpha$. In systems using NaK, absorption of neutrons in potassium leads to ^{39}Ar and ^{41}Ar. An additional activity, ^{23}Ne is also produced in Na by an $^{23}Na(n,p)^{23}Ne$ reaction. However, the half life of ^{23}Ne is only 38 s so this does not represent a major radiation component.

In a closed system that contains a heat exchanger, we often find ion-exchange columns or other systems that are used to filter out activation products. Many reactors employ purifying systems before the liquid is put into the reactor. Despite these efforts, activation products are always found to some extent. For this reason, the filters or columns may accumulate enough radioactive substances to require shielding.

For reactors using graphite moderators, there is enough lithium impurities in the material that the production of ^{35}S and ^{3}H needs to be considered.

3. Source of Airborne Activity

The activation and fission products produced in the system are not all solids. About one-fourth of the fissions result in gaseous fission products. These include noble gases (mainly Kr, Xe) and volatiles (I,Br). In an accidental release, ^{131}I would be the most serious immediate problem, as determined in the Windscale incident.[17] However, in power reactors, the release of ^{129}I, with its much longer half life, is of more concern with respect to long-term considerations (see Table 3.1 of Reference 5).

Other radioactive gases are produced by neutron activation of substances contained in or near the core. Thus, activation of the coolant, moderator, structural features, and air surrounding the system may produce radioactive gases.

For an air-cooled system, the products are radionuclides of nitrogen, oxygen, argon, carbon oxides, and other trace components. Many of these products are short-lived so that a modest holdup time effectively cancels out their influence. The main problem arises from neutron activation of ^{40}Ar, which comprises about 1% of normal air. The product ^{41}Ar emits a 1.2 MeV beta and a 1.3 MeV γ with a $T_{\frac{1}{2}}$ = 108 min. Since argon is an inert gas, the greater concern is not the internal dose, but the external dose an individual receives when immersed in a cloud of the gas. Since the exposure is an external dose, the time one spends in the radiation field determines the total dose received.

Besides the gaseous products in the air, particulate matter may also be found in the exhaust air. A good part of the dust in the air can be removed by filters, but the remainder becomes activated as it passes through the core. On the other hand, activated debris may be picked up in the reactor and carried out by the air flow. In most cases, a good deal of this activity is removed by exhaust filters before the air is released. In any event, this activity should be minor compared to the ^{41}Ar. Of course, if a fuel-element cladding develops a defect or rupture, large amounts of fission products may be found in the exhaust air. Other products can also become airborne from sample burnup or rupture. For long periods of operation, such as in power reactors, the probability of defects occurring increases.

Another source of activity in the system cam be attributed to what is called "tramp" fuel. In preparation, small amounts of fuel may be present on the outer surfaces of the fuel cladding. When the fuel element is irradiated, this fuel also fissions. This leads to the release of small amounts of fission products in the exhaust. The natural conclusion is that there is a potential fuel defect, but this would not be the case.

When water is the coolant-moderator, the products include radioactive nitrogen, oxygen, fluorine, and tritium. The presence of any impurities in the water can also be a minor source of radioactive gas products. Since many systems have a gas blanket above the core, activation may be important in this volume. In most systems, the nitrogen and oxygen are the main sources to be dealt with, as mentioned earlier.

If heavy water is the moderator, the production of tritium becomes a significant problem.[22] Tritium, which has a $T_{½}$ = 12.3 y and emits an 18 keV β, is formed in the reaction $^2H(n,\gamma)^3H$. In addition, ternary fission occurs on the order of $1x10^{-4}$ (thermal reactors) to $2x10^{-4}$ (fast reactors). In this process tritium is produced in the fuel of the reactor. Additional tritium can be formed in the control rods, burnable poisons, and coolant of a light water reactor (LWR). If boron and lithium are present in the system, tritium will be formed. Neutron capture in boron results in the formation of ^7Li, which in turn captures a neutron to form an α particle and tritium. As discussed in Section 15.D, tritium is formed in both ^6Li and ^7Li. The gas itself does not add greatly to the amount of airborne activity at an LWR since much of the tritium is retained in the water.

The buildup of ^3H_2O in heavy water is a function of the reactor power and operating time. In the form of a gas, tritium is poorly absorbed (approximately 0.1%) by the body, but as an oxide (H_2O vapor), almost all of that taken in remains in the body and is rapidly distributed in body water (within approximately 2 h). Thus, if tritium becomes airborne as water vapor, it may present a serious internal hazard. Any leaks in the system then can lead to the presence of tritiated water vapor in the air. In addition, tritium enters the body by breathing and penetration though skin, in about equal amounts.[23] It also adheres to and eventually penetrates most protective clothing after a certain time. Because of the weak β emitted by this radionuclide, the monitoring of tritium presents a further problem.[24-26]

In organic-cooled reactors, the hydrogen and carbon present result in activated gases in which ^3H and ^{14}C are the main radionuclides. The source of radioactive gas products for liquid-metal coolants is not the coolant itself, but the gas blanket above the liquid-metal. These products include ^{39}Ar, ^{41}Ar and ^{23}Ne.

Even though the fuel is clad, the fission gases still have some freedom of motion in the fuel element. If the cladding maintains its integrity, the products are trapped in the fuel element. Any weakness in the cladding allows the gases to diffuse out and partly dissolve in the coolant. Some of the products may also diffuse through the coolant and become airborne while the reactor is operating. In some research systems, the water moderator may be dumped from the core vessel at shutdown. This allows the gases to diffuse more easily into the atmosphere. The amount of these products that becomes airborne depends upon several factors. Among the more important aspects are the nature of the cladding defect, the type of fuel, and the operating temperature.

The products that actually become airborne also depend upon a number of factors. Some characteristics that have been noted are: (1) high fission yield, (2) ability to diffuse through semipermeable solids, (3) $T_{1/2}$ in the 10- to 30-minute range, and (4) solubility to some extent in the coolant.[27]

Among the fission products of highest yield are the bromine-krypton-rubidium and the iodine-xenon-cesium chains. In these two groups, the noble gases krypton and xenon are the ones most likely to become airborne. Since these gases decay to form particulate matter, we would expect to find rubidium and cesium radioisotopes (^{133}Cs and ^{88}Rb mainly) airborne also. The products krypton, xenon, cesium, and rubidium are all $\beta\gamma$ emitters.

The above products are most likely to be found for the less serious defects. In a major incident, such as a fuel-element rupture or a core meltdown, a larger release would be expected. In this case, iodine, bromine, xenon, and krypton would be the major products released.[5] In the major reactor incidents before 1986, in which releases have occurred, the containment of the radioactive material has prevented catastrophic releases. A discussion of these can be found in Reference 17. In the recent Russian reactor incident which resulted in over 30 deaths, the Chernobyl meltdown (1986), large quantities of potentially lethal fission products were released from the core, necessitating the extensive evacuation of an area of essentially 19 miles radius around the reactor.

This incident occurred in a reactor with no containment. A number of Eastern European countries were contaminated by the fission products released in the incident.[28]

4. Sources of Contamination

In addition to the above sources, contamination problems may also be present. That is, when substances are in the reactor and become activated, some of this activity can be loose materials, such as dust or powder. When the sample is then removed, this loose activity may deposit on some surface. Since a sample is often in a container, the container itself may pick up loose activity. This can be transferred to storage or transfer pots and the contamination tracked about eventually.

When the reactor shielding must be opened for maintenance or other work, this also presents an occasion for contamination. Here again, loose material that has been exposed to a neutron-fluence becomes activated. For this reason, any part of the system may contain this loose activity. When the reactor is opened up, the loose substance may become airborne for a brief time and then deposit on nearby surfaces.

For maintenance purposes, approach any part of a reactor cooling system as a potential contamination hazard. Since the water or coolant may contain activated scale or other impurities, any spill or leak can result in contamination. Repair of pumps, piping and heat exchangers present situations in which care must be exercised. Activity tends to deposit at low points in these systems.

With a large fission-product release, we can also expect some of these contaminants to deposit out on surfaces. Such a release could be brought about by burnup or rupture of a ^{235}U sample, which would result in deposition of contamination on nearby surfaces.

The nature of the contaminants is varied in many cases. In others, the presence of certain fission products may be detected. For this reason, we cannot list any specific radionuclides that may be found. For a given system, though, certain products occur more frequently. In general, identification of the products is not easy. To pinpoint the source of the contamination may require a knowledge of the history of the particular area, previous review of certain proposed experiments, and γ spectrometry equipment. In a given contamination incident, the exact cause may never be found, regardless of the time, energy and manpower expended. Sometimes, the best that can be done is to contain the spread of contamination, effect cleanup, and return the area to operational status in a short period of time.

F. Radiation Surveys

In principle, before startup of any new reactor, a survey should be made to establish background levels. This should include radiation in air, water, soil, vegetation and small animals. There should be continuous monitoring of effluents following startup.

During startup of a reactor, a detailed study should be made to check the integrity of the shield design.[18] This survey should be performed at low power (approximately 5% of maximum power) in order to detect the minor weak spots. At high power, these tend to be masked out by the larger leakage points. Since neutrons and γ rays are the more important radiations, the survey should be designed to obtain the levels of both neutrons and γ rays. At most reactors, the fast-neutron hazard greatly outweighs the thermal-neutron hazard. In some cases, the intermediate neutrons may also be important. Thus, the survey should attempt to include the levels from each of these neutron components. In the survey, attempt to find leaks due to voids, streaming through poorly fitted plugs, scattering around the shielding, and diffusion along structures which penetrate the shield.

During shutdown conditions, most of the required surveys are performed. This is because during this time these activities take place:

(1) Fuel Changes - When fuel is removed, changed or new fuel inserted, surveys are needed to determine the radiation field.

(2) Samples - When samples are inserted into the reactor or when they are removed, surveys are required.

(3) Shielding Adjustments - This requires surveying since the change may possibly affect both the direct and scattered radiation not only in the immediate vicinity but in adjacent areas. This includes temporary removal of shielding, such as reactor top shielding plugs or other shielding plugs.

(4) Equipment Changes - Additions or removal of any equipment from the reactor, especially involving experiments in which beams are brought out of the reactor will require surveys. One should check to insure that there is no unshielded line of sight to a radiation source along beams. After changes, surveys should be made when the reactor is operating to look for changes in direct and scattered radiation levels. In addition, decontamination operations during these procedures will require health physics surveys.

Surveys should also be performed when the reactor is operated at new or unusual power levels. Some limited access areas could show dramatic increases in radiation levels. Following changes, scattered radiation patterns may change and represent the cause of the increase.

Usually the most difficult activity to survey properly is maintenance work. That is partly due to extraneous reasons - people are in a hurry, a job needs to be completed in a short time to meet a startup schedule, contamination occurs to slow down the operation, and one tends to lose sight of keeping exposures ALARA. Maintenance operations usually require a preplanning meeting in which health physics personnel and operational personnel must sit down and discuss the steps to be taken in order to perform the job safely and successfully. The other factor is that radiation levels in different regions tend to be quite different. One cannot simply assume that a maintenance job in a new area of the system will have the same radiation field as that in another part. In general, many maintenance activities will require maximum use of the principles time, distance and shielding, as well as internal hazard control. This is sometimes dictated by the lack of foresight in design which leads to inadequate access space and lack of manueverability.

Air samples should be taken on a routine basis as well as in special instances. In many cases, stack monitors are part of the exhaust monitoring system. Over and above this, both particulate and gas samples should be taken to be certain that normal conditions prevail. Depending upon the particular reactor, one may have knowledge of, or an "educated" suspicion of, the likely contaminant.

Smear papers and liquid samples need to be taken at times to detect contamination. In many cases, induced activity and loose contamination are both present. In these instances, the loose contamination is included in the instrument reading. A separate method, such as a smear, must be used to find the loose activity. In heavy water moderated reactors, polystyrene smears may need to be taken to determine ^3H contamination by analysis in a liquid scintillation counter. Cold-trapping of air produces liquid samples of the water vapor in air, which also can be counted on liquid scintillation equipment to reveal the presence of ^3H.

G. Reactor Survey Instruments

The instruments needed to monitor a reactor should include devices that can measure the neutron and γ fields. Many of these specific portable devices have already been described in Section 12. Portable devices such as G-M survey meters and ion chambers can be used for surveying gamma rays.

It should be remembered that G-M survey meters are primarily detection instruments which are very sensitive with respect to locating the presence of radiation. However, in order to measure dose rate adequately, one must use an ion chamber instrument. Or, some detector whose response has been calibrated for the spectrum of the radiation being measured, may be employed.

Neutron fluence rate can be measured with BF_3 proportional counters. These instruments may be conventional long counters or variations of this design (PNC). Fast neutron measurements are made with the detector tube in the moderator. Thermal neutrons are measured with the bare probe. For dose equivalent estimates of the neutron levels, one may use a number of instruments whose response approximates the dose equivalent. These include either the 228 mm or 254 mm diameter polyethylene sphere devices with ^{6}LiI(Eu) crystals (see Section 12).

Although these can be overly conservative for neutrons which have energies in the intermediate region, this has not been a severe problem up to the present. However, the increase in the Q of neutrons by a factor of two will force reevaluation of the use of these devices. Alternatively, one may use the series of Bonner spheres to evaluate the dose equivalent using the variable response of a number of spheres.[29,30] The merit of this approach is that one may obtain a crude estimate of the neutron spectrum which allows the computation of an average quality factor, $\overline{Q}$, for the particular radiation field. An additional method, utilizing a Rossi tissue equivalent proportional counter is discussed in Reference 29. Other instruments which may be used have also been disucssed in Section 12.

As mentioned above, air samples must also be taken. Devices and methods for taking air samples have been discussed in Section 14. LB5211 or Millipore filter samples have been generally used to obtain particulate samples. For gas samples, the "grab" sampler can be used (see Section 14). For heavy-water-moderated systems, sampling for tritiated water vapor may be accomplished by use of cold traps. This method seems to be about the most satisfactory for the difficult task of monitoring tritium around reactors, although Kanne chambers have also been used with success. For suspected iodine release, charcoal filters have been the most successful means of detection.[31]

Smear samples are taken with either LB5211 or Millipore filter papers. Almost any substance may be used for $\beta\gamma$ activity. For α contamination, we would like to know the self-absorption features of the smear-sample substance. Some means of obtaining and analyzing water samples should be provided for water coolant-moderator systems. This is needed since this water may have to be dumped or pumped out at some time. A common method is to obtain a liquid sample that approximates an average concentration for the system. A measured portion of this sample is then placed in a planchet. The sample can then be exposed to infrared light to evaporate the liquid. The residue can then be counted to obtain the activity for the known volume. From this information and the known allowable levels, we can decide if the liquid can be dumped or must be treated. Other analysis methods, such as liquid scintillation counting or radiochemical preparation of samples for counting, may be required.

Smear samples can be counted for activity levels and also subjected to high resolution spectrometry. The latter method allows for identification of radionuclides, provided the number of activities present is not too large. If a large number of contaminants are present, the γ spectrum may be complicated. In some cases, α spectrometry can be of value, though α is seldom a problem around reactors. Beta spectrometry using plastic scintillators is also available.

H. Fixed Monitors

Reactor systems often contain a number of devices that guard against abnormal function. These may be part of the safety features of the design itself. In addition, other fixed monitors can be used to warn when radiation levels are on the rise. Stack monitors are used to provide data on the radiation concentration released through the exhaust system. These monitors are required since one must estimate the quantity of radioactive substances discharged in the effluent.

The devices can be equipped with an alarm system that sounds the alarm when levels exceed a preset value. Gamma alarm devices may be used to monitor radiation fields in the reactor room. These devices can also initiate shutdown mechanisms when the alarm level is exceeded. Continuous air monitors (see Section 14) can be used to continuously record airborne-activity levels. These devices can also be set up to provide an audible signal in the event of a release.

In addition to the above mentioned measurements, an environmental measurements program is generally required for reactor systems. This will require the planning of stations at which to monitor the environmental effect produced by the reactor in a given area. More discussion on this aspect can be found in Reference 17.

I. Nuclear Safety

A runaway reactor does not approach anything like an atom bomb even under the most adverse malfunctions.[5,8] In many cases, the inherent safety features prevent further rapid power rise and enhance shutdown. The power rise can be rapid enough, though, to result in a chemical explosion or a steam explosion, either of which may eject parts of the core. In these instances, severe damage to the reactor vessel occurs, but the damage should be limited to the reactor area. Chemical reactions involving U, Th, Al, Zr and H_2O can be violent enough to result in an explosion, if the fuel temperature goes high enough. If a core meltdown occurs, large amounts of fission products will be released from the fuel material. At Three Mile Island, most of the radioactivity released in the accident was contained;[17] at Chernobyl, the absence of containment resulted in a large release of fission product activity to the environment.[28]

1. Characteristics of Unwanted Nuclear Criticalities

Over and above these excursions, the presence of fissionable material in the core or in storage, and their handling can sometimes lead to an unwanted criticality. These nuclear accidents are less severe in that no blast occurs. However, these effects can result: (1) high radiation levels, both n and γ, which can be lethal for a considerable distance; (2) the release of enough heat to melt metals in some parts of a contained system; (3) contamination of the nearby area severe enough to present a hazard to personnel for perhaps months; and (4) high $\beta\gamma$ residual radiation levels from fission products.[32]

Despite care in handling and safety in design, an unwanted criticality may still be possible, though unlikely. The incident occurs with no warning, so that there is no time to invoke safety measures. That is, the system is subcritical one instant and then, suddenly, critical. When one of these accidents begins, nothing stops the process until the system returns to the subcritical state. In many cases, a single short burst (spike) of radiation occurs in a fraction of a second followed by a power plateau of about one second duration.[33] In other cases, a sustained self-controlled reaction for a rather indefinite time period may result. Both types have occurred on occasion, but the more likely event is a burst type.

The spike occurs because the rapid release of energy in the supercritical system provides a shutdown mechanism. This may be thermal expansion, boiling, or other effects which reduce the system reactivity. If there is no available restoring mechanism, the system would then remain subcritical. If the event takes places in a vessel, the initial spike may be followed by a number of spikes, if the vessel maintains its integrity or sufficient material is not ejected or evaporated.

A burst-type event may occur in an unshielded system of low initial radioactive content. It may also occur in highly radioactive systems such as reactors and processing plants. In the latter case, the presence of large amounts of shielding would greatly reduce the initial radiation exposure. However, the release of airborne radioactive material would result in extensive contamination. In the case of the former, the contamination would be high only in the local area of the event. The minimum value for the burst size in a system may be estimated as 10^{15} fissions for alarm purposes; 10^{16} fissions for dosimetry applications.[33,34] For a water-moderated assembly, one may expect 10^{17}-10^{18} in a short-burst incident. For metal assemblies, accidental bursts of 10^{17}-10^{19} fissions are assumed possible. Note that in a liquid moderated system, a series of events may result in up to 10^{20} total fissions.[35]

2. Prompt and Residual Radiation Dose Estimates

From the data on the mockup experiment of the Y-12 incident[36] it is possible to estimate the prompt neutron and γ doses as a function of distance from a burst of 10^{18} fissions. These are shown in Figure 15.6. From this figure, one can obtain the total dose at some distance and scale this up or down for an assumed number of fissions. No shielding effects have been taken into account nor changes in the n/γ ratio for other types of critical assemblies. The data may be used to

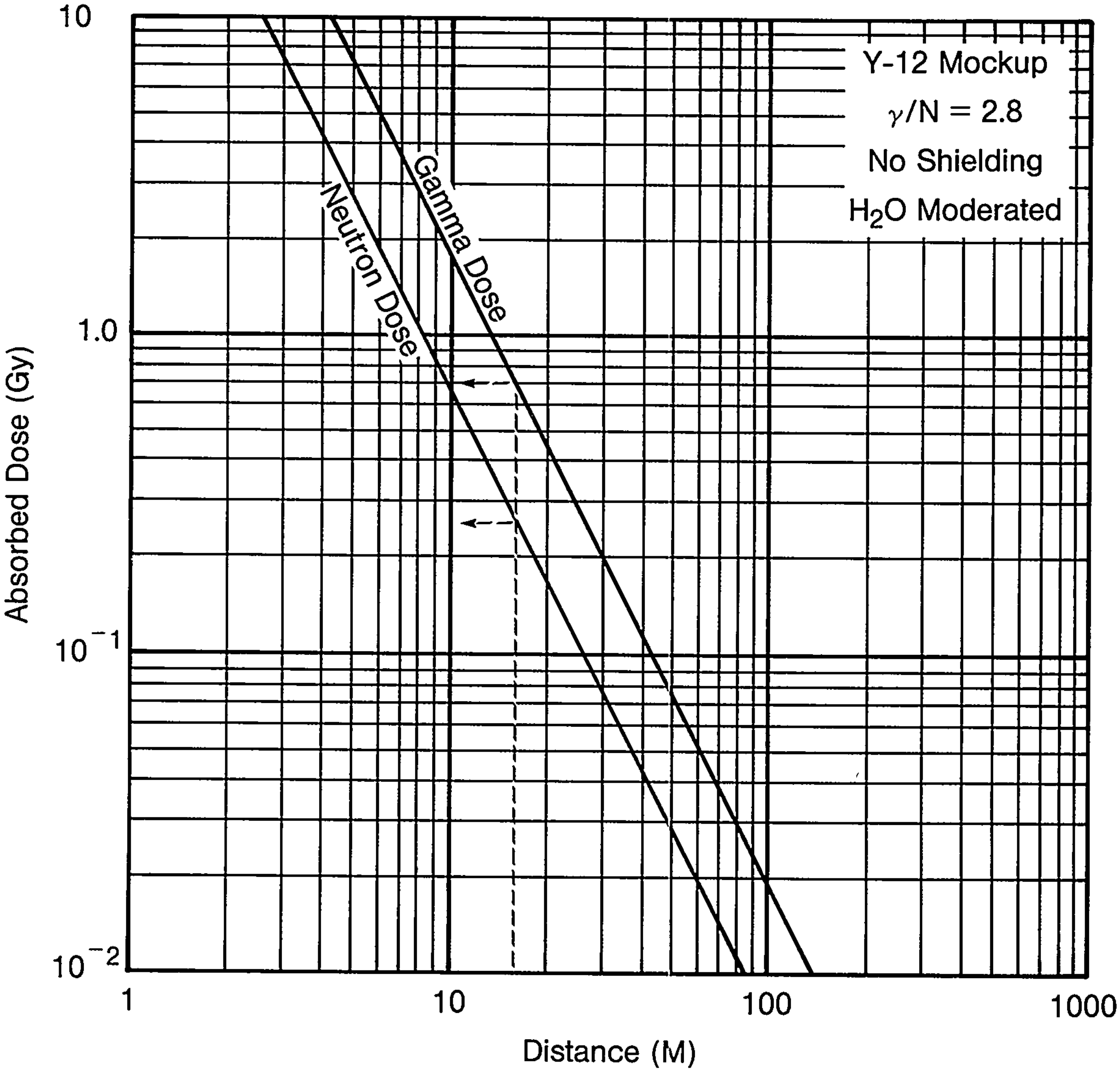


Figure 15.6 Prompt radiation dose from a burst of 10^{18} fissions. Total absorbed dose is ≥ 6 Gy (600 rad) up to 6m; ≥ 1 Gy (100 rad) up to about 15m.

obtain a rough estimate of the doses possible in an incident involving a moderated system.

To estimate potential doses from an incident in an unmoderated system, the data from Reference 33 may be used. The n/γ ratio is taken to be 10. A burst of 3×10^{15} fissions in a metallic, partially reflected ^{239}Pu assembly produced 0.51 Gy (51 rad) of neutron and 0.051 Gy (5.1 rad) of γ at 1.8 m. Assuming no shielding, this should yield total doses of $\geq$ 6 Gy (600 rad) up to about 10 m; and $\geq$ 1 Gy (100 rad) up to around 25 m.

Note that these estimates are tied to certain n/γ ratios. These ratios may vary from system to system, as well as in a series of bursts using the same system.[37] In general, for a moderated system, one would expect the γ dose to be the higher; in a metal system, however, the neutron dose would be expected to be the higher one.

For a burst > 10^{18} fissions, dispersion of the fissile substance and the fission products would follow. Heavy local contamination would result, giving a very high residual dose rate. This dose rate decreases very rapidly shortly after the burst, but may still be on the order of hundreds of mGy/h (> 10 rad/h) one hour after the event. A rough estimate of the dose delivered by the residual radiation may be obtained from an expression given by Way and Wigner[38]

$$\Gamma (t) = 1.26\ t^{-1.2}\ \frac{\text{MeV}}{\text{s.fis}} \qquad t > 10\ \text{s} \qquad 15.13$$

in which t is the time, in s, following the criticality event. For the nominal assumptions: average γ energy of 0.7 MeV for residual γ radiation, 10^{18} fissions occurring, correcting for air attenuation and buildup, and using the conversion factor of 1 Gy/h (100 rad/h)=10^{11} MeV/m^{2}s, one may crudely estimate the residual dose rate. This is shown in Figure 15.7, in which the residual radiation dose rate at four selected distances is plotted versus time following the criticality event.

Notice that even at 30.5 m (100 ft) from the event, the dose rate shortly after the burst is > 10 Gy/h (> 1000 rad/h). Even an hour

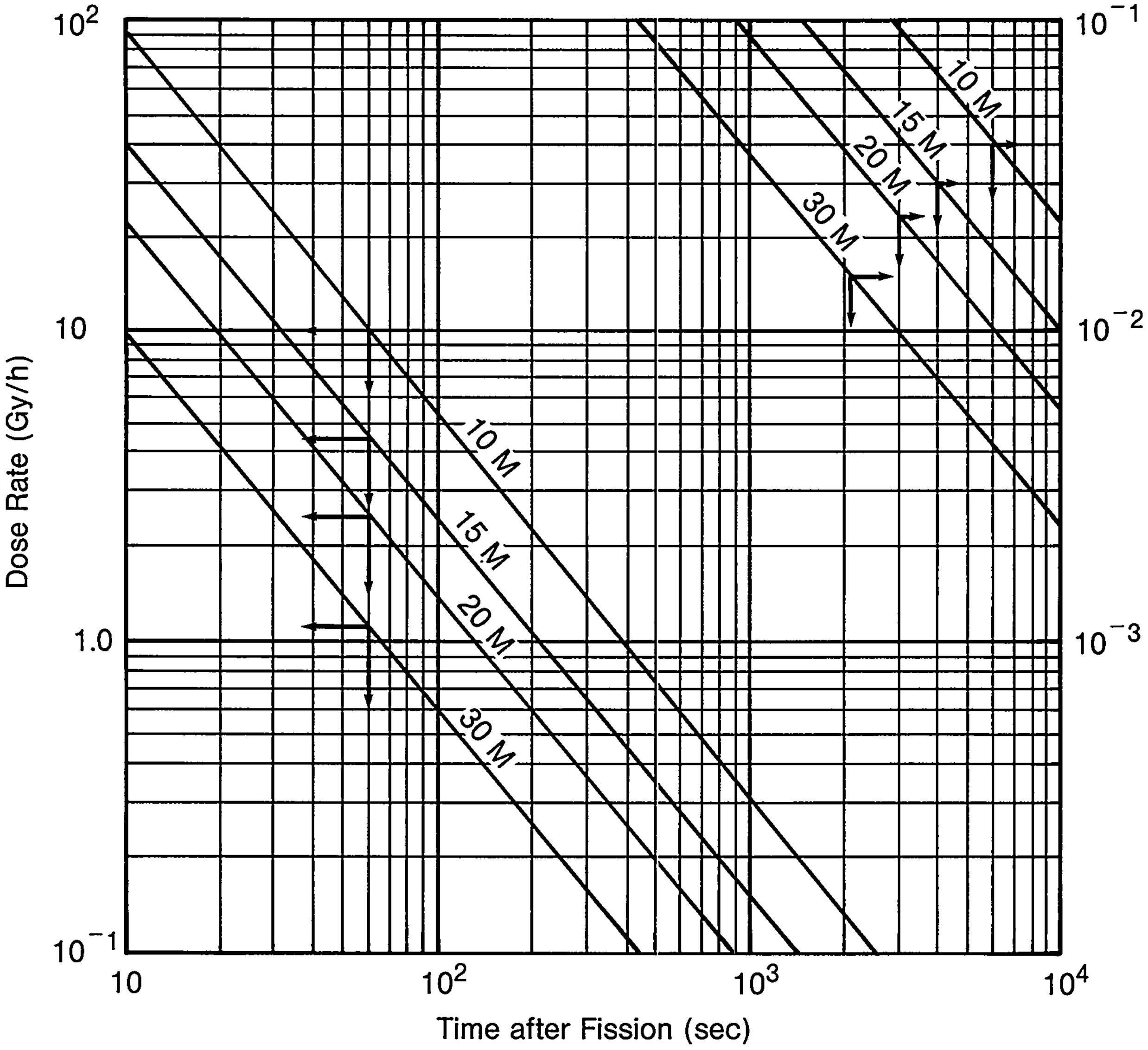


Figure 15.7 Dose rate due to residual radiation.

after the assumed event, the dose rate at 9.1 m (approximately 30 ft) is still greater than 0.1 Gy/h (> 10 rad/h). This is the basis for instructing personnel to leave immediately when a criticality alarm sounds. Indeed, one should run not walk! Seconds are important immediately following the criticality, especially if one is relatively near the event. Added to the prompt dose one may receive, will be the dose one receives from the residual radiation if he loiters in the area.

3. Safety Approach

Since the hazards of an unwanted criticality only exist if the event occurs, the emphasis is placed upon prevention of these incidents. The mere presence of fissile material is not the only requirement. There must be a critical mass and the proper set of conditions. Earlier in this section we have seen that the critical mass is a function of many variables of the material (nuclide, density, enrichment), form (chemical and physical), shape, surroundings (moderators, reflectors, neutron absorbers) and the neutron interactions (absorptions, scattering, non-fission capture).

Nuclear safety may be defined as the prevention of uncontrolled critical-mass reactions. For our purposes, this means preventing a criticality in any system meant to be subcritical. This task involves solving problems of a scientific, engineering, and administrative nature. Scientific problems involve the setting of values for the critical mass of the substance. Engineering problems involve process design to avoid conditions that favor a chain reaction. Administrative problems involve setting up practices, rules, and controls that attempt to preclude such occurrences due to errors in handling and processing.[39]

Using the data from critical-mass studies, the process and equipment can be analyzed to arrive at safe mass limits throughout the plant. Then, detailed rules to achieve nuclear safety can be drawn up, and an administrative system of checks and balances can be set up to ensure that these rules are followed.

4. Control Methods

The goal of nuclear safety is to minimize the chance that a criticality or excursion occurs. This goal is attained if a sufficiently large fraction of the neutrons released in fission are captured (nonfissile) or escape from the system (leakage). Then, fewer fissions occur in one generation than in the preceding one. A number of factors influence the neutron balance of the system so that several types of controls may be used.[39-41] When these factors are applied, the principle used is that nuclear safety should depend as little as possible upon actions or decisions by personnel.

For solutions, safe geometry or "always safe shape" is the preferred method of control.[41,42] In this method, vessel dimensions are such that enough neutron leakage occurs to prevent any excursion (see Figure 15.8). The general approach is to limit one of the dimensions (such

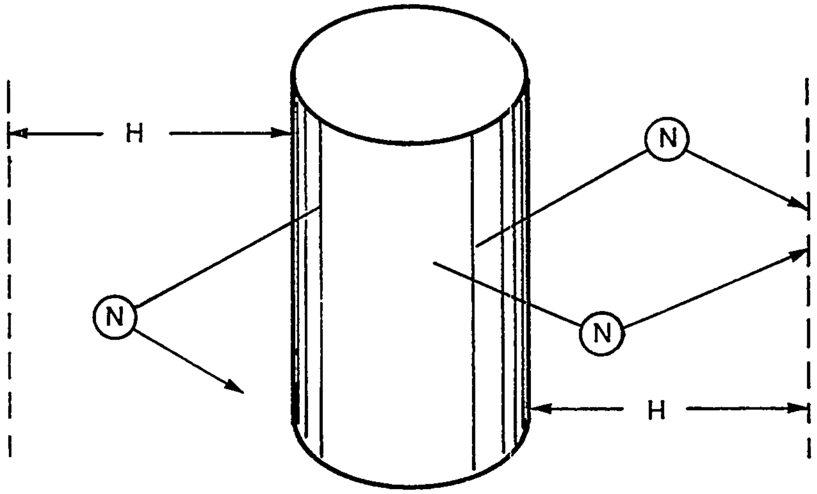


Figure 15.8 Example of safe geometry for pipes and cylinders.

as the diameter of the cylinder in the figure) to ensure that enough neutrons will be permanently lost. Then, a chain reaction cannot be achieved in the isolated system, regardless of the amount of fissile material. That is, the cylinder in the figure could be of infinite length and contain any concentration of fissile matter in the solution without going critical. Remember that we are discussing an isolated system! The term "geometrically favorable" is sometimes used since these containers may not be "safe" if more than one is present.[41]

In many processes, the absence of substances that may moderate and/or reflect neutrons cannot be guaranteed. For example, the human body contains a great deal of hydrogen. For this reason, the values assume that a thick hydrogenous reflector surrounds the material (H in Figure 15.8). The thickness, diameter, or volume of vessels can be designed so that criticality cannot occur. This method may also be used for metals in the form of slabs, cylinders, and spheres.

Mass control is a method used for metals if geometrically favorable shapes cannot be used. The mass limits are given in ANS-8.1-1983,[42] and are called subcritical limits. The subcritical mass limit insures that a system will be subcritical under the specified conditions. These subcritical limits allow for uncertainties in calculations and experimental data, but do not allow for contingencies. In some cases, composite limits may apply; however, these are only valid if all conditions are met on the controlled parameters. In the past, mass limits were based on a quantity called the minimum critical mass. This mass value was < ½ of the experimentally determined critical mass for a given set of conditions. This presumably allowed for the contingency of one "double batching" error. Note that this is not the case with the subcritical mass limit. However, the Standard requires that adequate safety margins be applied to the subcritical limits to insure an added degree of safety. A number of single-parameter and multi-parameter subcritical limits can be found in Reference 42. Mass control has often been used for both glovebox and hot cell nuclear safety controls.[40] The posting of the mass limit on the enclosure is a common practice that aids proper control practices. Administrative procedures requiring the logging in, and out, of fissile material help to avoid mistakes.

Other methods of control, such as concentration limits, use of fixed and soluble poisons, and Pu/U ratio are discussed in the literature.[41,42] In all cases, the limits used apply to isolated systems.

In the use and storage of fissile substances, the spacing must be taken into account to nullify any interaction between two systems.[40,41] In Figure 15.9, container A is separated from container B by enough distance so that much of the neutron leakage from A is not intercepted by B, or vice versa. In this case, both containers A and B could be geometrically safe, as isolated systems. In the setup pictured, those systems would also be safe by virtue of the sufficient separation. However, the probability of interaction occurring is proportional to the solid angle, which increases with the decrease in the distance of separation. Thus, if container B were at position C in the figure, more of the n leakage would be intercepted. This could produce a situation in

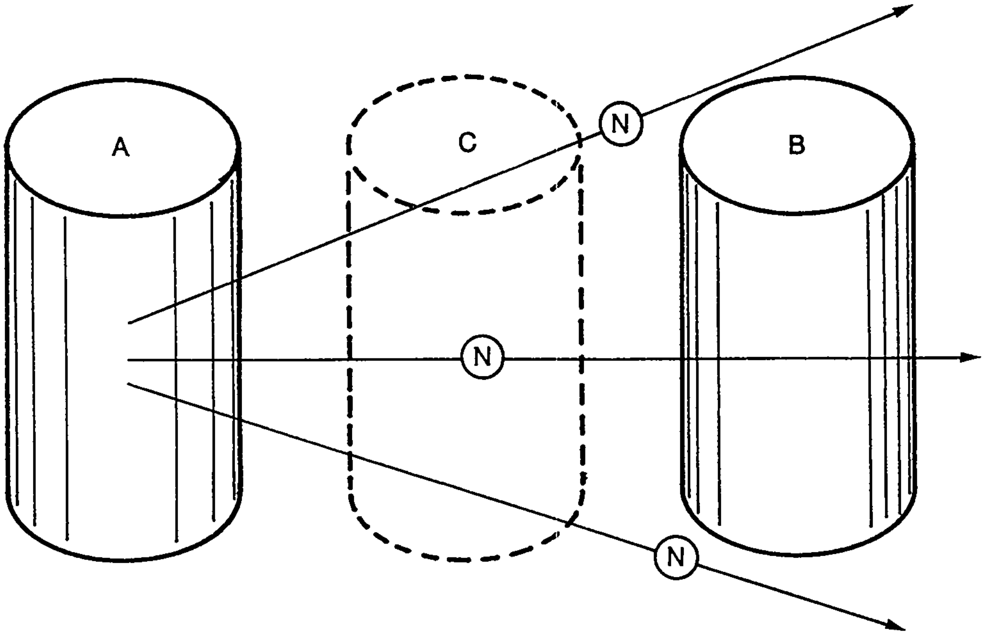


Figure 15.9 Interaction effects due to adjacent containers.

which enough neutrons would not be permanently lost from the system of two containers. That is, enough neutron leakage from the entire system to avoid an excursion may not be obtained if the spacing of fissile matter is too close together. These neutrons may cause fission in a nearby system, and neutrons from that system may do likewise. In this manner, two subcritical systems may be made critical if care is not taken. Storage in arrays or cubicles at fixed separations which also have posted mass limits, is one means of obtaining the proper separation.

5. Administrative Controls

Administrative controls may require that a written nuclear safety statement or manual be prepared for review by competent individuals if more than safe amounts are to be handled routinely. These statements should describe the processes, facilities, and equipment to be used as well as the makeup, masses, and configurations of the fissile substance. Credible accidents must be treated as well as safe procedures for double-batching or the occurrence of two simultaneous, unlikely and independent events. The latter is referred to as the double-contingency principle.[41,42] There should be written specifications which address these important features: accounting of the movement of material, labeling or identification of material at all times, posting or recording of material in enclosures, communication of information regarding any process changes, and training of personnel. With respect to training, this should cover aspects of the operating procedures, potential hazards which may occur, and especially the consequences of deviations from established procedures.

In addition to facility imposed controls, there are also regulatory controls imposed by the United States Department of Energy, Nuclear Regulatory Commission, Department of Defense and other agencies. These must also be addressed in the nuclear safety statement.

For each approved statement, one individual should be responsible for seeing that only approved procedures are followed.[42] Because of the importance of nuclear safety, the responsibility for nuclear safety must be clearly defined within the facility.[39] Detailed controls that are set up must be arrived at by each facility through analysis of their unique functions.

6. Monitoring and Dosimetry Practices

Although the risk of an excursion is very low, it cannot be entirely eliminated.[41] For this reason, one should provide (1) a means of alerting workers to the threat of high radiation fields and (2) a plan for their quick evacuation. The exposures in such an event can be quite severe, so a delay of even a few seconds in leaving the area can be crucial, depending upon one's location with respect to the incident. This is indicated graphically in Figures 15.6 and 15.7. These figures indicate that for an individual close to the source of the criticality event, both the prompt dose (Figure 15.6) and the delayed dose (Figure 15.7) can be significant, even dangerous, in the sense of life threatening.

The radiation alarm system should have a very rapid response and good reliability, give little or no false alarms, and emit a distinct, audible signal.[33] This system should sound an immediate local alarm so that an area can be evacuated. It should also indicate the location and radiation level of the event at some central station. The sensor, or radiation detector, may respond to either γ or neutron fields. This is usually placed near the point where an excursion is a possibility. The alarm system, which may consist of a number of horns, may be distributed throughout the rest of the area. Personnel should be instructed so that they are aware of the alarm sound, the proper evacuation route, and the vital need for <u>immediate</u> evacuation. Drills and periodic training should be carried out to ensure that this information is not forgotten.

If a criticality occurs, a system is needed to obtain data on the absorbed dose to which a worker has been exposed. To obtain this information, we need a measure of both the neutron fluence and spectrum and the γ absorbed dose. Such information can be obtained by the use of one of a number of nuclear-accident dosimeters.[43] Since placement of these fixed units may differ from the location of the exposed worker, a measure of the ^{24}Na activity induced in the person's body is also needed. With the ^{24}Na data, the dose at the dosimeter station can be related to the dose the worker receives.

One nuclear-accident dosimetry system in use utilizes a modified Hurst dosimeter package.[44] The dosimeter is made up of a neutron section and a γ section, which are physically tied to each other. The neutron section contains a bare gold foil, a cadmium-clad gold foil, a sulfur pellet, a cadmium-clad copper foil, and a ^{237}Np, ^{235}U and ^{238}U fission-foil system in a cadmium and ^{10}B shield. This unit measures neutron fluence in the thermal, intermediate, and fast regions to provide a rough spectrum estimate. This information allows the neutron dose to be estimated. The γ dosimeter has chemical components and a TL dosimeter for measuring γ dose.

In addition to the Hurst threshold detector units (TDU), which are the primary dosimeters, a system of secondary dosimeters are also employed.[45] These are used to provide corrections for those cases in which spectrum differences between the primary and secondary dosimeters are expected.

A number of detectors have been used for nuclear accident dosimetry. Reference 45 discusses various systems and their comparative response to bursts from an unmoderated, unreflected reactor (HPRR).

At DOE Facilities in which there is a possibility of excessive exposure of personnel to radiation from a nuclear accident, both area dosimeters and personnel nuclear accident dosimeters (see below) are required.

To aid in dose determination, a "quick-sort" method is needed to roughly divide those exposed from the unexposed.[47] A common method used is to provide the person with an indium foil. When exposed to thermal neutrons, the foil gives off $\beta\gamma$ radiation. The foil can be easily mounted on a dosimeter or film badge. The induced 54-minute $\beta\gamma$ activity can be read with a $\beta\gamma$ survey instrument or ion chamber, if the check is made soon after the incident. This is used primarily as a screening mechanism to determine which individuals may need more analysis. Note that if the dosimeter is discharged, one may be dealing with a potentially exposed individual.

An alternative method is to use a γ survey meter at the body midsection to measure induced radioactivity in the sodium in blood. This method is less sensitive, so one cannot use an ion chamber instrument, however, the activation is not dependent on orientation as is the previously mentioned method. A GM probe is placed against the abdomen of the individual and he is asked to bend forward to enclose the tube. Very roughly, the dose is related to body weight and meter reading by

$$D(Gy) \sim \frac{80\ (\text{instrument reading-mR/h})}{\text{Body Weight (lbs)}} . \qquad 15.14$$

The important point is that the sensitivity of this method is low, so even a small reading may indicate a significant exposure. On the other hand, one must exercise care in the application of this method. As an alternate, the probe may be placed under the armpit, with the open window facing the chest.

In addition to the fixed dosimeters, the worker is provided with neutron-measuring components that are part of his film-badge unit. One such system includes sulfur powder, an indium and two gold foils in a cadmium box, and a copper foil taped to the outside of this box. In addition to spectrum information, some information in regard to orientation in the neutron field can be obtained.

In principle, nuclear accident dosimetry methods are simple and easy enough to apply. In practice, an accident may be a complex affair in which many unforeseen obstacles can appear.[48] These factors may delay or impede the dose determination in any given event. For example, a dosimeter may be destroyed or badly contaminated in the incident. Also, the radiation field may be so high that the unit cannot be readily retrieved. In these cases, other activated substances may be found that can be used to measure neutron activation.[47]

In any event, the priority actions are: to rescue personnel, prevent further incidents or exposure, and quickly determine those who have been seriously exposed. For individuals receiving severe exposures, the clinical symptoms of acute radiation injury which manifest themselves may provide enough information for treatment.

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QUESTIONS

15.1 Name the process by which an atom bombarded by a neutron becomes another isotope of the same element.

15.2 Name the process by which an atom bombarded by a neutron is split into two nuclei of roughly equal size.

15.3 What two things made the pursuit of nuclear energy through the splitting of the atom by neutrons worthy of further study.

15.4 By whom and when was the first self-sustaining nuclear reactor put in operation?

15.5 What alternate term is commonly used for slow neutrons?

15.6 What product may result when uranium-235, ^{235}U, absorbs a neutron, but fails to fission?

15.7 Explain the difference between prompt and delayed neutrons.

15.8 What is the approximate amount of energy given up by the fission of a single atom?

15.9 What nuclides are of value for use as reactor fuels?

15.10 Explain the term fission yield.

15.11 What term expresses the probability per atom of fission occurring?

15.12 Upon what two factors does the probability that a neutron will fission an atom depend?

15.13 Name the components of a nuclear reactor. What is found in thermal, but not in fast reactors? What might be omitted in reactors of very low power?

15.14 For a particular reactor, what is the term that indicates the minimal amount of fissionable material by which it can sustain a chain reaction?

15.15 What term is used to indicate that the percent of ^{235}U in uranium fuel has been increased above the natural level?

15.16 What product is desired when ^{235}U is intentionally bombarded by neutrons having less than 1 MeV energy?

15.17 What are the factors in the four-factor formula? What term is the result of the four-factor formula?

15.18 How does the effective multiplication factor, k_{eff}, differ from the multiplication factor k_∞?

15.19 Relate the effective multiplication factor to the terms subcritical, critical, and supercritical.

15.20 Give the term and the expression that indicates the fractional difference represented by $k_{eff}-1$.

15.21 Give the term that indicates the time rate that a reactor gains in neutron fluence rate by a power of e.

15.22 Explain the meaning of prompt critical.

15.23 What does the change in the multiplication factor caused by a system change divided by the multiplication factor define? When the change in the multiplication factor exceeds the fraction of delayed neutrons, what is the condition of the reactor?

15.24 Explain the two foremost built-in features that provide a degree of reactor safety in preventing a reactor runaway.

15.25 What term is used for fission products that strongly absorb neutrons thereby causing a loss of reactivity?

15.26 What term is used to indicate an emergency shutdown of a reactor?

15.27 To what is the power level of a reactor directly related? Why may the neutron flux density not be directly related to the power level?

15.28 What are the four main categories of nuclear reactors on the basis of purpose for which they were designed?

15.29 Name two basic requirements for fusion of two nuclides to occur.

15.30 What is "break even time"?

15.31 What is a plasma?

15.32 What are the most plausible nuclides for fusion reaction? Why?

15.33 Name two methods that are being studied for plasma confinement.

15.34 Why is nuclear fusion considered an attractive source of energy?

15.35 Identify some direct sources of radiation resulting from reactor operation.

15.36 Identify some sources of radioactive contamination that may result from reactor operation.

15.37 Name some of the radioactive nuclides found as corrosion products. What is the origin of these nuclides?

15.38 Name some of the radioactive contaminants found in the coolants.

15.39 What airborne nuclide is of major short-term concern in an accident?

15.40 Name some of the reactions that are likely to produce tritium in a reactor moderator.

15.41 Name some of the control methods to minimize the chance of an "excursion."

15.42 When should radiation surveys be made for reactor operations?

15.43 What type of test will usually determine whether contamination is loose or induced?

15.44 What are the two principal radiations that will leak directly from a reactor?

15.45 What are the two radiations most likely to be present as the result of radioactive contamination?

15.46 What fixed monitors should be used in reactor operations?

15.47 Give the term which indicates that a nuclear reactor has experienced a runaway situation.

15.48 What means are used to determine the neutron fluence that results from a criticality accident?

15.49 What measure is used to determine the neutron dose received by an individual in a criticality accident?

15.50 Summarize the priority actions in case of an accident.

PROBLEMS

15.1 a) Two of the stable end products of fission chains from thermal fission of ^{235}U are ^{91}Sr and ^{139}La. What are the kinetic energies carried off by these fragments, for a complete fission of 0.001 kg of ^{235}U?

Answer: 67.3 MeV/atom of ^{235}U for ^{139}La
102.7 MeV/atom of ^{235}U for ^{91}Sr

b) Assuming a fission yield of 6.0% for each of these fragments, find the weights of each one of these two nuclides, after the complete fission of 0.001 kg of ^{235}U.

Answer: $3.55\text{x}10^{-5}$ kg of ^{139}La
$2.32\text{x}10^{-5}$ kg of ^{91}Sr.

15.2 Find the fission rate of 1.5 kg of plutonium-239, ^{239}Pu, in a reactor in which the average fluence rate is $5\text{x}10^{15}$ n/m^2-s. The fission cross section for $^{239}_{94}Pu$ is 740 b.

Answer: $1.40\text{x}10^{15}$ fissions/s

15.3 Thorium-232 can be converted to an isotope of uranium by an initial bombardment with neutrons. Show the scheme for this conversion.

15.4 Using the four-factor formula, find the multiplication using the following data: η=2.7, ϵ=1.01, p=0.81, f=0.46.

Answer: 1.016

15.5 Find the effective multiplication factor, k_{eff}, for the preceding problem when 0.1% of the neutrons escape by leakage.

Answer: 1.015

15.6 Find the excess reactivity, Δk for the preceding problem.

Answer: 0.015

15.7 Find the reactor period when the excess reactivity is 0.006 and the generation time for all neutrons is 0.083 s.

Answer: 13.83 s.

15.8 Find the power level of a small research reactor in which 5×10^{12} atoms are fissioned per second.

Answer: 152 watts.

15.9 How much uranium-235, ^{235}U, will the reactor in the preceding problem use in one man-year (50 weeks of 40 hours each)?

Answer: 1.4×10^{-5} kg.

15.10 a) A reactor produces heat at the rate of 3030 MW. How many atoms of ^{235}U suffer fission per second?

Answer: ~ 1×10^{20} fissions/second.

b) If the fuel in the core contains 3000 kg of ^{235}U, what percentage of the fissile material has been used up after 30 days of continuous operation?

Answer: ~ 3.3%.

SECTION 16 - HEALTH PHYSICS ASPECTS OF ENCLOSURES AND CONTAMINATION CONTROL

A. Design Philosophy

Through the years, the safety-design philosophy for hazardous radionuclides has evolved into the central theme "containment and concentration." Contain the process to ensure a barrier exists between the worker and the toxic substance. Then, if the barrier breaks down, concentrate or confine the released material to a limited area. This philosophy implies the need for multiple barriers. The containment protects the worker from the hazards of the material. A source enclosed in a capsule would be a simple means of containing or enclosing the material. Sometimes, both the substance and the process need to be contained, and enclosures such as gloved-hoods, gloveboxes, hot cells, and other structures are used.[1-6]

In Figure 16.1, a train of glovebox systems is shown. Transfer of hazardous material between modules is carried out through transfer ports, so that the worker remains in a different environment from the hazardous radionuclide. In the case of pyrophoric (spontaneous ignition) materials, such as finely dispersed plutonium or uranium, the glovebox atmosphere is an inert gas such as nitrogen. The gloves are usually made of neoprene or butyl rubber, which may have special coatings (hypalon) for acid protection or may contain lead for shielding purposes.[6]

Depending upon the degree and type of hazard, one may require a completely enclosed compartment as shown in Figure 16.1, or, for low hazard applications (i.e.-low radiotoxicity, non-pyrophoric and low external radiation), an open hood with no shielding.[2,3] When the radiation hazard from the material becomes great enough, gloveboxes, even when shielded, may not be adequate. In this case, recourse is made to large, heavily shielded enclosures, called hot cells or caves, which are often supplied with thick concrete walls and lead-glass shielded viewing windows. Operations are performed with the use of manipulators, see Figure 16.2, which allow the operator to perform the necessary operations on the highly radioactive substance. In some cases, in-cell gloveboxes may be used,[3] for highly radioactive α material such as occurring in trans-plutonium element research.

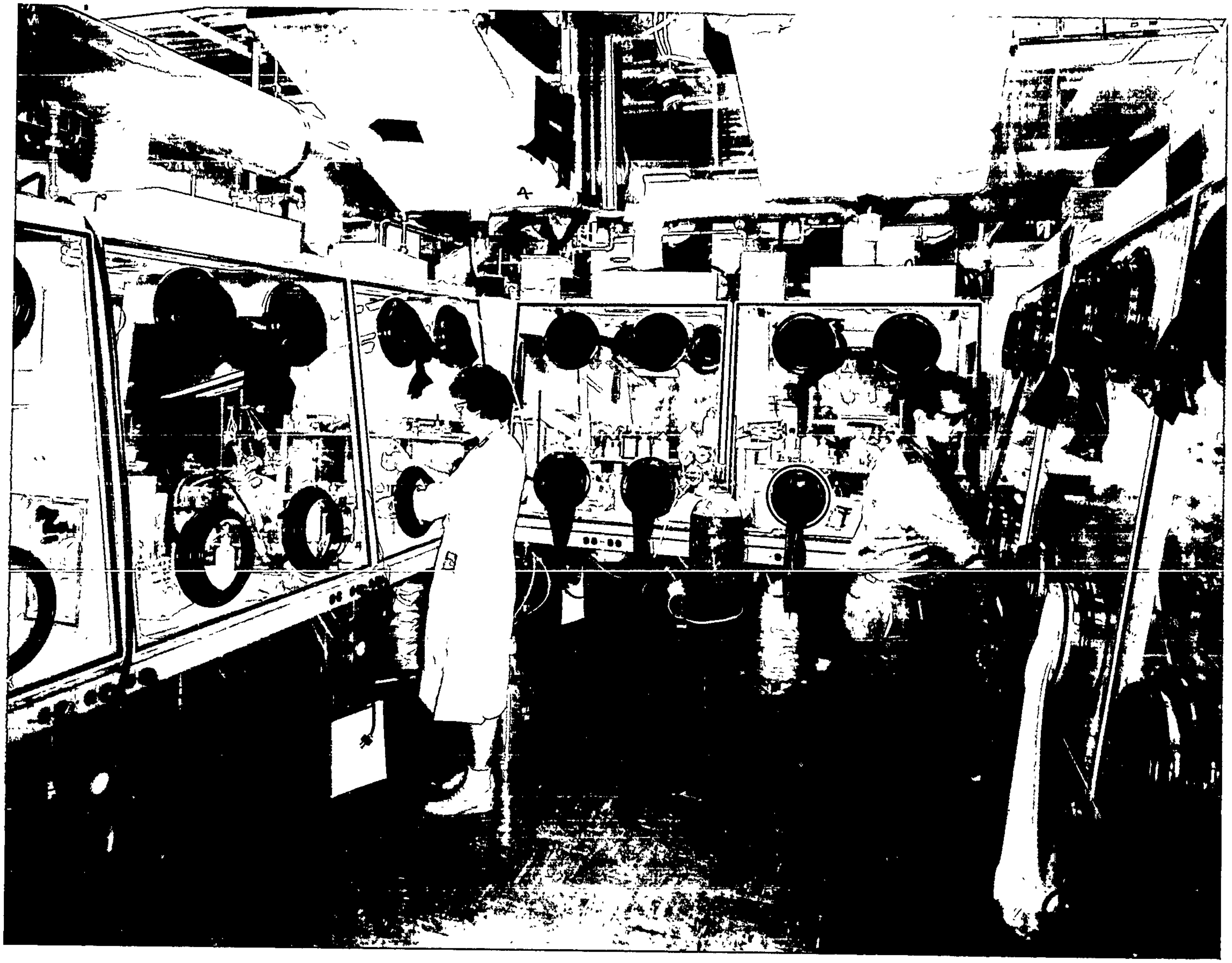

Figure 16.1 Typical glovebox train.

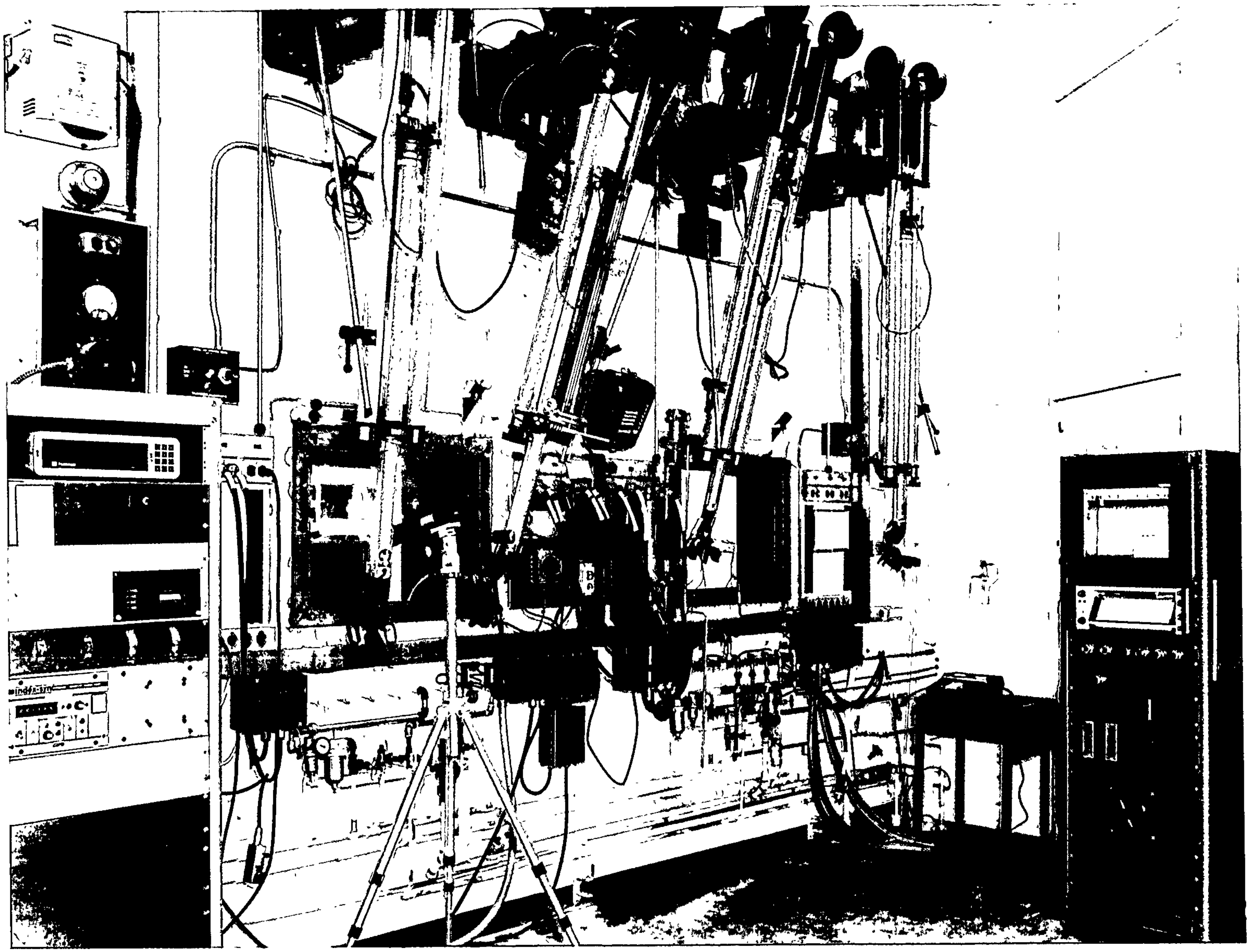

Figure 16.2 Manipulators on a hot cell.

Containment refers to the barriers for preventing the release of the radioactive substance. However, a release should not result in the discharge of a hazardous amount of the contained substance from the facility. This requires the presence of other systems to concentrate any release of contained material when an accident occurs. Proper design of airflow patterns and tandem exhaust high efficiency filtering provide a multiple barrier to releases. Airflow patterns are directed from clean areas toward those in which hazards may develop, as shown in Figure 16.3. Air in these areas flows from offices through corridors into the laboratory to pass through the enclosed system. The enclosure (hood, glovebox, etc.) atmosphere is discharged through the filter system. The flow pattern is always into the enclosure, which reduces the chance of material escaping. The room, the airflow, and the filter system provide the concentration or confinement of the released material.

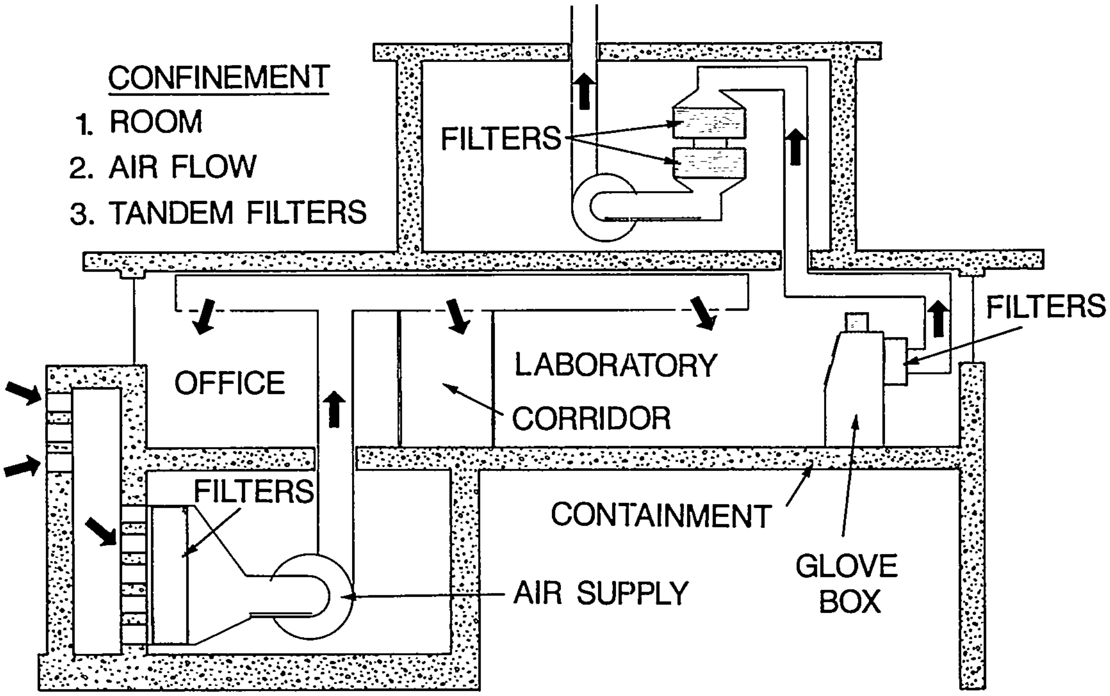

Figure 16.3 Concentration (Confinement) and Containment for a Plutonium Laboratory.

For the more hazardous radionuclide materials, such as plutonium or the trans-plutonium elements, at least two lines of defense are required. The primary line of defense is the containment. This is the enclosure, shown as a glovebox in Figure 16.3. Other lines of defense are provided by the confinement or concentration systems. These are the airflow pattern, the room or laboratory itself and the tandem high efficiency filters (HEPA)[3] used.

Various designs of safety systems have been employed over the years. The purpose of the safety design in all cases is to achieve these goals:[7]

(1) Ensure that personnel radiation exposure is kept as low as reasonably achievable (ALARA) under normal operations.

(2) Control effluent release to ensure no increase in present environmental background levels occurs. Ensure that any radiation exposure to nearby population groups is kept as low as reasonably achievable. Design systems to prevent the release of hazardous levels of material in the event of accidents.

(3) Include features that will prevent or reduce the severity of incidents. The impact of the event in regard to personnel injury and threat to life should be the prime concern. Damage to property and program delays should also receive some thought.

The special features of plutonium and certain other radionuclides that increase their utility require that these substances be handled most carefully. This need for safety arises from four main features of the materials. First, they are unstable substances, whose decay results in the release of radiation. This radiation may present a hazard, even if one is distant from the source. Second, these elements are toxic. Intake into the body may result in long-term deposition in bone and other organs with severe local damage at these sites. Third, the substances are fissile. When enough of such material is present in the proper setting, a fission chain reaction can occur. Such an event would release large amounts of energy and radiation (see 15.I.1.2). Fourth, some forms of these substances are pyrophoric; that is, they ignite spontaneously. Once started, a fire could spread rapidly and result in great damage. To reduce the hazard potential and to achieve safety in handling these substances, control of these features is required.

Because of the special hazards that are present in work with certain radionuclides, many design features must be treated. The type of functions to be carried out will affect the degree of safety needed. Basic studies of the site, plant layout, ventilation patterns, waste-handling needs, fire control approach, radiation safety problems, and nuclear-safety aspects will dictate many design goals.[4-6, 8-12] Structure design, process flow, confinement, and containment through gloveboxes or other devices will also call for a certain level of engineered safety.[1,13-15] The form, amount, and composition of the radionuclide handled are also factors. These aspects modify the extent of hazards expected from the radiative, biological, nuclear, and ignition properties of the radionuclide.

The need for enclosures then, is basically tied to these conditions: the use of large amounts of long half life, highly radiotoxic substances which require frequent processing in operations exhibiting a high release potential.[2]

If one uses small amounts of short half life, low hazard radionuclides in infrequent processes of low release potential, then one may allow the worker and material to be in the same environment.

B. Control Procedures - Gloveboxes

The type and extent of control measures used in radioactive work will vary. Many factors enter in, and only some of these were briefly mentioned above. For example, in some cases, plutonium is handled in open-faced hoods, but in most instances, it is handled in semiremote units (glove boxes) or remote enclosures (hot cells).[4-6, 9, 12, 13] The features of a particular handling operation may dictate special control measures, as does the state of the plutonium being used. Low levels (< 37 kBq - approximately 1 μCi) of nondusty plutonium samples do not require the degree of control that higher levels do. Sealed or clad plutonium, regardless of amount, may be handled in hoods rather than gloveboxes for certain steps in a process. Several reports and articles deal with both the hazards and control measures of certain programs.[6, 9, 16-19,21]

The laboratory handling of plutonium often differs from that found in a plant or production facility.[4] In general, up to 0.2 GBq (~ 5 mCi) of plutonium may be handled in open hoods; gloveboxes are used for larger amounts. In plants in which gram or kilogram amounts are used, gloveboxes are used almost exclusively. For dusty operations or for plutonium in other pyrophoric forms, these call for gloveboxes with inert atmospheres. In processes involving irradiated plutonium samples that have not been separated from fission products, hot cells are used to provide shielding. In the future, glovebox handling of plutonium may be precluded because of the high radiation level produced by Pu isotopes in recycled plutonium. The following discussion will treat aspects of control procedures for enclosures (primarily glovebox systems).

1. Worker Procedures

Transfer of items into and out of the enclosure must be done so as to preserve the barrier integrity. For a closed system, such as a glove box, the design often includes plastic-bag ports or sphincter valves which allow such transfers.[1] Procedures for changing gloves, as well as windows, while maintaining barrier integrity, must also be developed.

The probability of accidents such as spills, unexpected chemical reactions, small fires, or explosions should be considered.[9,12] Safety design will attempt to overcome or minimize the effects of such incidents. However, since these events may still occur, it is extremely important that personnel be trained in methods of combating such episodes. Needless to say, plans must be formed that treat the problems arising out of such accidents.

Regardless of the design excellence, the worker must also follow certain approved, albeit regimented, practices, which should be spelled out in written work instructions. These measures should include such things as:

(1) Remove all unnecessary equipment from the enclosure before starting a new project.

(2) Use absorbent matter or strippable paint on the surface to avoid contaminating the floor of the enclosure.

(3) Limit or avoid the use of pointed tools, wire, or sharp objects, and quickly report any punctures, wounds, or scratches obtained while performing work with radionuclides.

(4) Follow good housekeeping practices: Clean interior surfaces often, clean spilled matter at once, remove solid waste promptly, limit amount of solvents and other combustible material, and segregate pyrophoric matter.[1] Do not let dust pile up!

(5) Inspect gloves often for signs of rupture, wear, pin holes or deterioration. Be aware that acids and other mixtures attack the gloves. Also, gloves fail rapidly in work with oxides of high specific α activity.[22] Change gloves frequently to avoid mishaps (preventive measure).

(6) Remember that stored vessels that are not vented can result in pressure buildup leading to rupture and spillage. Also, the storage of high specific-activity α emitters in solution may result in heat evaporation of the liquid. This process may occur in conjunction with breakdown of the liquid (radiolysis) in the solution, causing gas buildup. For a very high specific activity, such as for ^{238}Pu, gas bubbles rising to the surface may burst and form a very fine oxide dust, which can grossly contaminate the enclosed system.[22]

(7) Avoid splashing of liquids or spraying them during enclosure operations and keep liquid containers in a secondary container.

(8) Use latex or surgeon's gloves in addition to those that are part of the enclosure. This ensures a second line of defense if enclosure gloves fail.

(9) Do not perform radioactive work with open wounds or scratches unless you have obtained prior approval from medical authority.

(10) Know what your responsibilities are in regard to work performance and safety procedures.

The above listing is not meant to be complete, nor should it be taken as such. It is provided as a guide to point out some areas of safe practice and perhaps serve as a stimulant to enable the reader to reflect upon other pertinent safety practices. For each facility, other suitable practices may also be recommended.

The toxic nature of α emitters requires that these substances be kept out of the body. For some of these α emitting substances, exposure to air tends to form a finely divided, loosely held oxide. There is a danger of this material being dispersed into air and remaining airborne.[1] The presence of such oxide in air creates an inhalation hazard. To prevent this, the substance is contained. However, the enclosure then becomes highly contaminated with the loose matter, so that a negative pressure must exist in the enclosure to avoid contamination leaks. Then, the airflow in the area will pass through the enclosed system and out through the filters in the exhaust system. This arrangement prevents the spread of contamination. However, since no system is perfect, leaks may still occur.

Since one must have access to the α emitter in gloveboxes, rubber or neoprene gloves, which are attached and sealed to glove ports on the box, are provided. These gloves protect the worker from contamination, but are subject to deterioration. Thus, surgical gloves are also worn to provide another means of protecting the individual from contamination. In the event of a slight breakdown of the rubber gloves, the individual may not know this merely from appearance. Some other means must then be used to detect the spread of contamination, such as surveying each time hands are withdrawn from the glovebox.

Besides the two examples given above, loss of control may also occur as the result of operator error. Then again, an incident may be due to some unforeseen reason or unknown cause, but loss of control will still be the result.

In hood work, surgical gloves are worn to protect the worker's hands from contamination. If at all possible, one should not handle objects within the hood with the gloves directly. Something else, such as tissues, should be used. Then, the tissue becomes contaminated first, and the gloves act as a second line of defense. Hands should not be pulled out of the hood unless gloves are removed, since this may cause a loss of control. Also, if the gloves pick up significant contamination, they should be changed and disposed of as waste. The use of highly contaminated gloves only disperses the material throughout the hood and increases the chance of hand contamination if the glove fails.

2. Facility Layout

The goal of any contamination-control program is to limit the spread of contamination to as small an area as possible. This aids in reducing the exposure potential of the incident, eases the task of controlling the extent of the event (spill), and enhances cleanup of the area. Prudent facility layout can help to achieve these goals. One common approach is shown in Figure 16.4:

(1) Set up zones or areas to isolate the more hazardous processes. Allow work with radioactive materials only in the laboratory area.

(2) Control access. Set up a security guard (G) at the entrance, or require a special badge to actuate the door lock.

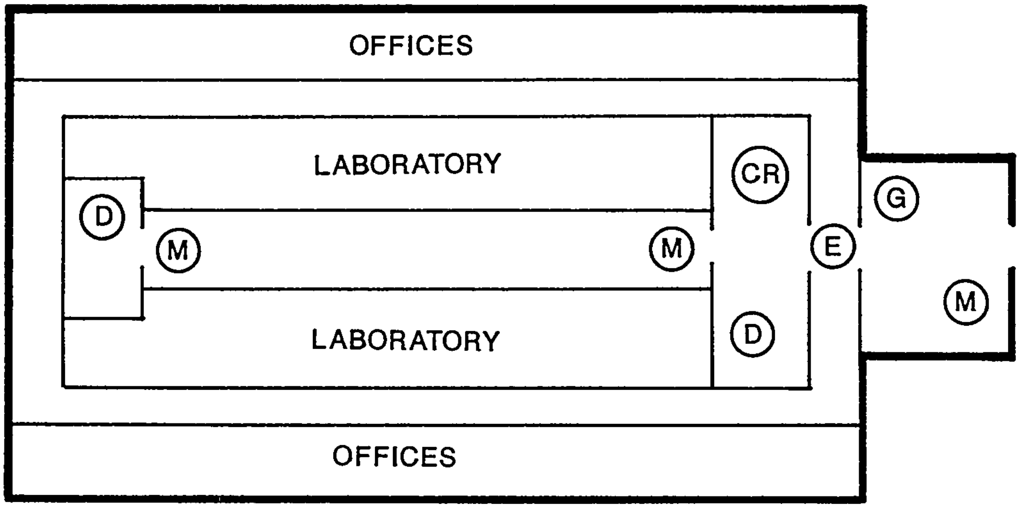

Figure 16.4 Facility design for contamination control.

(3) Direct traffic patterns. Permit access to the laboratory area only through the door at the end of the area (E) and past monitoring stations (M).

(4) Set up monitoring (M) and/or decontamination stations (D) at zone interfaces. Contamination monitors should also be located at selected locations along glovebox lines.

(5) Use protective clothing that is limited to the zone. Leave clothing worn in the laboratory area in the change room (CR) when leaving the laboratory area.

(6) Insist that anything leaving the hazardous zone be surveyed for radiation.

The above scheme provides a means to detect and limit contamination that might otherwise be tracked or transferred to other areas. The entrance may also be marked with instructions for use of monitoring devices and special protective clothing and equipment while in the radiation zone. Such devices or equipment may be provided there or in the change room. In some cases, decontamination stations located at the zone boundary may be desirable. All these features may be set up, but depend upon the good will of the workers if they are to succeed. That is, unless the workers, and the visitors to the area, realize the necessity for checking everything that leaves the hazardous area, the contamination-control concept will fail. The scheme to contain any contamination within the laboratory area in Figure 16.4 works well only if the total cooperation of all personnel involved is obtained. When cooperation breaks down and exceptions to the control procedures begin to increase in frequency, then the probability of losing control is greatly increased. On the other hand, with strict adherence to the control procedures, the extent of any contamination can be limited to the hazardous zone alone. In addition, if frequent surveys of the laboratory area are performed, contamination incidents are picked up more quickly. This also aids in reducing the size of the area involved in a contamination incident.

3. Area Contamination Control Practices

In addition to the practices discussed in 16.B.1 above, personnel should observe other control measures. These measures seek to reduce the hazard potential at the area level as well as prevent intake of contamination. The use of shoe covers, laboratory coats, and surgical gloves that are not worn out of the area will help contain the spread of contamination. Survey of surgical gloves for radiation each time the hands are removed from the glovebox permits early detection of loss of control. The forbidding of smoking, eating, or drinking in the radiation area will reduce chances of radioactive matter being inhaled or ingested. Hands should always be washed following work with radioactive materials. When leaving a radiation area, one should monitor his hands, shoes, and any equipment being removed to further prevent the chance of spreading any contamination. Wounds or punctures should always be monitored for radiation and promptly treated by a doctor.

Fissile material storage practice involves nuclear safety aspects that require special attention.[23] For this reason, storage must be strictly controlled. Large amounts of fissile material are often stored in areas or vaults of special design, and under the control of a responsible individual. This individual must keep records of the amount on hand and be responsible for proper storage and movement of fissile material. For safety purposes, access to these areas should be restricted.

For smaller amounts, the user should submit a material-handling plan that should include storage plans. He is responsible for the movement and storage of this material in his area. This plan should be reviewed for approval in regard to nuclear safety. For amounts below an established minimum the threat of a criticality is no longer a concern. However, depending upon the physical form of the fissile material, special storage precautions may need to be taken. Such factors as the proper type of container or storage in an inert atmosphere will warrant some thought.

4. Control Problems

The previous discussion was concerned with the general design features and administrative aspects of contamination control. However, the problems in the degree of control are specific to the material which is involved. For example, refer to Table 16.1. This table groups together a number of radionuclides which are presently in use. Included in the Table are the ICRP 30[24] values for the most restrictive DAC and the ALI, in both Bq and μg, for each of the radionuclides. If one compares only DAC values, the relative control problem appears to be roughly the same

Table 16.1 Occupational DAC and ALI for Selected Radionuclides[24]

Nuclide	DAC Bq/m^3(μCi/cc)	ALI Bq(μCi)	 μg
^{239}Pu	0.08($2x10^{-12}$)	200 ($5.4x10^{-3}$)	$8.7x10^{-2}$
^{238}Pu	0.09($2x10^{-12}$)	200 ($5.4x10^{-3}$)	$3.2x10^{-4}$
^{252}Cf	0.4 ($1x10^{-11}$)	1000 ($2.7x10^{-2}$)	$5x10^{-5}$
^{210}Po	10 ($3x10^{-10}$)	$2x10^4$($5.4x10^{-1}$)	$1.2x10^{-4}$
Nor. U	0.7($2x10^{-11}$)	2000 ($5.4x10^{-2}$)	$7.8x10^4$

(within a factor approximately 9) for all radionuclides, except ^{210}Po. If we compare ALI values, in Bq, the result is roughly the same (within a factor of 10, except for ^{210}Po). On the basis of both the DAC and ALI (in Bq) comparison, ^{210}Po appears to be less hazardous than ^{239}Pu by a factor of 100-125, whereas for the other radionuclides, the relative hazard is roughly equal. From a control standpoint; however, the degree of control required will be based upon consideration of mass loss, not activity. It is a certain mass of material which will escape the enclosure in a loss-of-control incident. If we look at the ALI expressed in μg, the relative hazard of a release for a given degree of control will differ greatly depending upon which of the radionuclides in the table is involved.

For example, assume that a system is 99.9% efficient in controlling releases. One would then expect 0.1% to be released. If the sample were 87 μg of ^{239}Pu, the release would be 8.7×10^{-2} μG or 1 ALI. The same system employed for 87 μg of normal uranium would result in a release of $8.7 \times 10^{-2}/7.8 \times 10^{4} = 1.1 \times 10^{-6}$ ALI, too small to be even detected. However, if the incident involved 87 μg of ^{252}Cf, then the release would represent $8.7 \times 10^{-2}/5 \times 10^{-5} = 1740$ ALI! Clearly, the degree of control necessary in each case is vastly different. With respect to control design then, one must be aware of the additional containment which may be needed on the basis of mass considerations.

5. Health Physics Control Measures

To supplement the efforts of the worker, as well as maintain the control achieved by the safety-design features, requires a program that can evaluate the effectiveness of the control mechanisms. This latter need is supplied by the health physics program. Even though highly sophisticated control devices are used, and techniques are refined to the utmost detail, events that result in the loss of control will still occur. To mitigate the impact of these events by limiting the extent of the loss of control quickly is a function also supplied by the health physics program.

The features of such a program in regard to a high hazard material, such as plutonium, will be discussed. Note that practices at different facilities will not be identical and will depend upon a host of factors.[25-27] In addition, this discussion is not intended to be all inclusive, since with time, procedures, equipment, processes and even the type and form of the radionuclides used will change thus requiring adjustments in control procedures.

Health Physics control measures may be grouped into two classes: (1) provide fixed monitors for detection of hand, shoe, clothing, or air contamination; and (2) provide portable instrument surveys and monitoring service on a routine basis to maintain contamination control. A survey is the process of monitoring for ionizing radiation, analyzing the results in regard to established protection standards, and defining, or evaluating, any radiation hazard which may exist.

a. Fixed Monitors

A fixed monitor may be used to trigger an alarm when the measured radiation field exceeds a preset level. In this sense, the device indicates that the loss of control is significant. A fixed monitor may also be used to check the contamination status of an item. In this application, the device may indicate either no contamination or some level of contamination. Both types of devices may be used for certain operations.

(1) Personnel Monitor

A personnel monitor is an ac-operated device used to survey hands, shoes, and portable objects (see Section 12). One type features a chamber in the form of a flat plate-type probe of large area through which the counting gas passes. Another type uses a non-flow air probe. Alpha particles enter the chamber through a thin Mylar window and ionize the chamber gas. The chamber output, which is proportional to the α activity, is displayed on a count-rate meter and indicated by audible clicking from an attached speaker. The same device may be used for β monitoring if the operating voltage is changed.

This type of monitor can be placed both at the enclosure and at exits to the work areas. When a hand is withdrawn from the enclosed system, it can be surveyed immediately without the risk of spreading any contamination. Upon leaving the work area, a worker is also able to survey his hands, clothing, and shoes. By making frequent surveys, he will quickly detect the presence of contamination. This action will limit the spread of any contamination.

Another type of personnel monitor, the hand and shoe counter is also used at the entrance to radiation areas. Such units may be designed to monitor both hands and shoes during the same counting period without requiring the use of a probe.[28] At the end of a fixed counting time, an indication of contamination or no contamination is given. When contamination is indicated, the level can then be measured with portable health physics instruments. These monitors are useful when a large number of people require hand and shoe surveys.

(2) Continuous Air Monitors

Continuous air monitors are used to maintain a watch on the level of air activity in the work area. These instruments consist of some air-collecting device and a suitable detector for the radiation. For gases, a flow-through chamber may be used to collect and detect the presence of the radioactive gas. For particulates or dust, filters are often used to collect the material, and the type of radiation emitted will dictate a choice of detector. The device is often provided with an alarm, since the presence of an airborne toxic substance may represent a severe inhalation hazard. Commercially designed units are also available and can provide monitoring of several radiation characteristics.

The presence of natural airborne radioactivity affects the ability of an air monitor to detect airborne plutonium. These radioactive products give rise to short-lived α and β activities that tend to mask out any activity due to plutonium in an air sample. In the past, the detection limit of air monitors has suffered because of this feature. This can somewhat be overcome by the use of an air monitor which contains a silicon surface barrier detector. This type can be used in conjunction with a single channel analyzer to count the α particles of a particular energy emitted by the specific radionuclide. In the case of ^{239}Pu, the analyzer is set to measure the 5.15 MeV α particles. Air is drawn through a millipore filter which is "seen" by the surface barrier detector. Counts will be recorded for alpha particles in the energy window, which is centered about 5.15 MeV and generally ±.5 MeV to either side. Some interference is still noted because radon daughters produce some α particles of energy greater than this range. Counts occur in the window when these alphas are degraded in energy before interacting in the detector. However, this background is greatly reduced in magnitude and can be approximately accounted for.

An improved alpha spectrometry method has been developed which utilizes alpha counting in a vacuum.[20] In this device, air is drawn through a filter which is viewed in-line by a surface barrier detector. If a release occurs, the detector senses the increased activity and actuates an alarm. The filter may then be advanced to an off-line vacuum chamber to be counted under vacuum. The increased resolution obtained for the vacuum count results in decreased interference from radon daughter products in the plutonium channel.

(3) Stack Monitors

This type of fixed monitor is used to estimate levels of radioactive substances which are released to the environs. In case of an accident, it also serves to denote if control by the filter system has been breached and what the extent of the release may be. These types of monitors are often devices that are designed and assembled for specific jobs, based upon the characteristics of the radiation and the process.

b. Monitoring Services

The monitoring services provided by an active health physics program include routine instrument and smear surveys for surface contamination, personnel and item contamination surveys, and continuous monitoring.[29] Air sampling of the general work area, as well as special samples during certain operations, are also carried out.

(1) Surface-contamination Surveys

Routine surface surveys should include both instrument and smear surveys. The frequency of these surveys will depend upon the form of the radionuclide and the type of operation carried out. In most cases, because of the degree of containment used for radionuclides such as plutonium, a positive survey result is cause for concern. That is, even a rather low level of contamination should warrant both cleanup and further investigation. For this reason, surface-contamination guides or levels are not readily found for plutonium.[30] At any rate, the toxic nature of plutonium is such that any loss of control should mandate followup surveys. In some cases, certain contamination levels may be tolerated on the basis that proven control techniques are able to maintain contamination at or below these levels.

A number of portable, or survey, instruments are used to perform routine surface surveys (see Section 12). Normally, alpha survey instruments are used for surface surveys when plutonium is involved. If the sample contains fission products, a $\beta\gamma$ survey instrument may be useful. Gas-flow proportional probes are used for all types of α monitoring. When humidity may vary greatly, these are preferred over air probes. However, α air probes have been used successfully in other climates. The counting rate displayed on the instrument meter is proportional to the activity of the α source. Because of the short range of α particles, the probe window must be used very near to the surface being surveyed (< 6 mm). Earphones should be used with the counter to increase detection sensitivity. The lower limit of detection is about 100 dis/min in the area (0.01 m^2) under the probe.

This type of counter may be modified so that low-energy beta particles can be detected. This latter capability is useful in monitoring for some of the transplutonic radionuclides. Alpha scintillation counters using ZnS crystals are also used for α surveying. Gas-flow and scintillation counters both respond to neutron fields, so that extra care is needed when surveying for α surface contamination.

The smear survey consists of wiping a filter medium or other suitable material across a surface that is suspected of being contaminated. Sometimes, wet smears are obtained by using alcohol or acetone on the smear. The smear can be surveyed for activity with a portable instrument or counted for activity in a laboratory counter. This technique is easy to apply and allows quick spot-checking of areas for loose contamination. In many cases, one is concerned more about the presence of the contamination rather than its level. The qualitative results obtained in smear surveys are most useful in these instances. The method is also used to check the integrity of clad or sealed sources to ensure that no leakage has occurred.

(2) Personnel and Item Surveys

In addition to the fixed monitors used for surveys in the work area, many other surveys of both personnel and items are made utilizing portable survey instruments. These surveys are required when transfers into and out of enclosures are made. Surveys are also required for such things as skin punctures or wounds, glove changes on enclosures, decontamination of personnel or items, and repair work on potentially contaminated objects. Surveys are always needed by personnel at the end of their work period. These surveys should be complete surveys of the personnel's hands, shoes and clothing, even though surveys of his surgical gloves show no contamination. Cases have occurred in which pinhole leaks have led to clothing contamination higher up on the arm without the surgical gloves indicating activity.

(3) Continuous Monitoring

In some cases, the nature of the work may be hazardous enough to require the presence of a health physicist during the entire process. This is called continuous monitoring. This type of monitoring is also good practice each time a new technique is being used. The advantage here is that the worker can devote full concentration to the job and the health physicist will take care of the necessary monitoring. Also, the worker has the benefit of health physics advice at each step of the job. This type of monitoring is also desirable if the operation requires frequent movement of materials and items into, and out of, the glovebox system. The presence of the health physics monitor there will facilitate the necessary surveying and preclude delays in the operation.

(4) Air Sampling

Air sampling is carried out to assess the condition of the work-area environment. It also allows a check on the effectiveness of the control design and/or work practices in regard to contamination control. Since internal dosimetry is difficult to perform, common practice has been to limit the concentration of radionuclides in air. Although the DAC values are used as standards that should not be exceeded, the goal is to keep airborne levels ALARA. The DAC value (refer to DOE Order 5480.11)[31] is used as an index of control, and airborne concentrations well below this level imply satisfactory control. To confirm that this is the case, routine air samples are collected. When air samples reveal increased levels of airborne activity, this is viewed as a potential breakdown of the control system. Levels above allowable values for short periods do not constitute a severe hazard themselves, but such levels over extended periods would be of concern. In the former case, it is not so much the level but the apparent loss of control that should be the major concern. Once control is lost, levels may go even higher as more material is released. Air sampling methods, as discussed in Section 14, are used for routine air samples. These may be supplemented by an in-place monitor, utilizing a surface barrier detector to detect specific energy α emissions, as discussed in 16.B.5.a.(2).

Samplers may be located in the general work area to routinely monitor the overall operation. In other cases, samplers may be positioned close to a source of potential air contamination, such as hoods or gloveboxes. Personal air samplers have also been used to obtain air samples in the breathing zone of the individual. Thought must be directed to the type of sample needed in regard to sampler placement. Large differences have been obtained in the results of some of these methods for sampling a given atmosphere (see Section 14.D). The frequency of air sampling in an area will depend upon the form of the radionuclides, the nature of the work, and the presence or absence of fixed air monitors.

C. Internal Exposure Control

1. Personnel Monitoring Measures

There are many problems in assessing the uptake, distribution, and retention of radionuclides (particularly plutonium) and the subsequent internal exposure that results.[9,32-36] To estimate the internal-exposure rate, a measure of relevant organ burden is needed. Given the organ burdens and excretion data, the total dose equivalent may be estimated. The control methods discussed above attempt to exclude the entry of radioactive matter. Nevertheless, such entry may occur, so that methods are needed to estimate the burden and subsequent radiation-dose equivalent.

A common method used to appraise internal exposures is a bioassay program. Routine urine and/or fecal samples are collected at some frequency, reduced chemically, and counted for the radionuclide content. In general, soluble radionuclides retained in the body are excreted mainly through the urine, whereas insoluble radionuclides are found mostly in the feces. Data obtained from a number of samples can be used to estimate organ burdens. In the event of a suspected intake, it may be necessary to collect all excreta for the first few days following an accident.[33]

Chest counting of individuals by whole-body counters is used to estimate lung deposition for radionuclides which emit photons of sufficient energy to be detected despite absorption by body tissue. In recent years, whole-body counters of special design have been used to detect plutonium in the body.[37] These devices are able to detect the average 17 keV x rays from ^{239}Pu and the 60 keV γ rays from ^{241}Am. Since these photons are easily absorbed in dense substances, the method is useful for lung-burden determinations, but not for bone burdens. To correctly assess the Pu lung burden by this method, for an intake equal to a few ALI values, requires that the ^{239}Pu/^{241}Am ratio be known.[9] This technique, used in conjunction with other bioassay samples, has improved the estimate of intake in the case of inhalation events. Multiple Ge detectors, operated in series, show promise of being able to measure plutonium in lung directly.

Wound monitors have been developed that can be used to estimate the amount of radioactivity in wounds.[9,29] These instruments also count the low-energy x rays from the plutonium. These devices are useful for cases in which plutonium may be embedded in the tissue. Then, alpha radiation will be totally absorbed and not detectable, whereas the x rays can still be detected. These monitors aid the physician treating the wound since the plutonium should be removed and may have to be excised. Normal skin contamination by radionuclides, such as plutonium, can usually be detected by portable α survey meters.

On occasion, other samples such as nose swabs, sputum, or blood may be needed in order to help in assessing the intake.[9]

2. Therapeutic Measures

A number of substances have been found useful for increasing the urinary elimination of metals from the body.[29,32,38,39] Of these, diethylene-triamine-penta-acetic acid (DTPA) has proven the most effective agent.[38] However, treatment with this substance is most effective only in the early stages following intake. It is presumed that once the heavy metal is bound in an organ, DTPA will have little effect.[29] In the early stages, when the level of soluble heavy metal in the blood is high, DTPA is able to increase the excretion rate. For a heavy metal, such as plutonium, this results in a loss of deposited plutonium. When plutonium is removed, the subsequent total dose may be greatly reduced. This feature is most important since soluble plutonium is so tightly bound in bone, ensuring long, continuous irradiation.

In recent years, pulmonary lavage (lung irrigation) has been tried as a means of removing inhaled matter from lungs.[29] In one application of this method to humans, three lavages removed about 1/8 of the estimated initial lung burden.[40] Since there are risks in this procedure; namely, administration of anesthesia and potential adverse reaction in the individual, the use of this technique requires careful assessment.[29]

3. Protective Equipment

Protective equipment should be supplied to the worker to increase the degree of protection afforded by other safety measures.[3, 41-45] Its purpose is to protect the worker from contamination and to aid in the control of contamination. The use of such equipment guards against pickup of external contamination on the worker's person or clothing and intake of contamination into the body. Also, the equipment is removed by the individual in the given area so that the contamination can be confined to that area.

Types of protective clothing generally used are laboratory coats, coveralls, plastic suits, pants and shirts. Coveralls or pants and shirt are preferred, since these call for a change of personal clothing. This precludes loss of personal clothing in a contamination incident or spill. Gloves and safety shoes may also be provided to round out the basic scheme. In some cases, one may provide laboratory coats and shoe covers at the zone entry for use by observers or those not normally working in the area. Any handling operation with radioactive material, particularly plutonium, should be done with protective gloves on. Other more elaborate types of clothing may be used.[41,42]

Other necessary protective devices include respiratory equipment. These may be needed for special operations, such as repair work or modifications, as well as for an emergency. The use of these devices on a continuing basis in place of proper source control of the radioactive work is undesirable.[45] One class of respirators, the full facepiece type, processes the air the worker breathes in from his surroundings.[41] These are suitable for particulate matter, and some may provide protection up to 100 times the allowable concentration in air.[46] In the other class of respirators, which includes self-contained breathing units, the worker does not breathe the air from his surroundings. This type is suitable for both gases and particulate matter and should be used for particulate concentrations greater than 100 times the DAC in air. Here again, the function performed and the form of the radioactive material used will affect the type and required use of the above protective equipment.

Utilizing a respirator fit-testing program, one may be able to demonstrate a greater protection factor than that stated above. This requires that the individual be tested, wearing his respirator, in an enclosed chamber where the penetration of the test aerosol can be measured.[6,45] In this case, the protection factor is actually measured and documented. Even in these cases though, prudent practice has been to use the more conservative factor found in Reference 43.

D. External Exposure Control

The external dose rates from some materials used in enclosures can be significant. In particular, ^{239}Pu, with large percentages of ^{240}Pu, ^{241}Pu and ^{242}Pu can exhibit relatively high external radiation fields. The external radiation field is a sensitive function of the isotopic composition.[9] For plutonium that contains less than 5% on a weight basis of isotopes other than ^{239}Pu, the external field is not a great problem. Above this weight percent, the external exposure hazards increase and eventually require special controls, such as reduced work time, shielding, or semiremote handling.[18,47,48] Of course, recently irradiated uranium capsules may contain fission products in amounts comparable to the plutonium content. These would present very serious external exposure hazards that would require extensive shielding and totally remote handling, such as in hot cells. This latter consideration also applies to handling large amounts of transplutonium matter.

In the case of a plutonium sample, the external field is comprised of low-energy x rays, low- and high-energy γ rays, and fast neutrons. Beta radiation is absorbed either by the enclosure itself or by the enclosure gloves. X rays and γ rays are reduced to some extent by the enclosure and by using leaded gloves. However, the photon dose rate increases with surface area of the plutonium. Thus, a thin layer of plutonium on the floor area of the enclosure may increase the photon dose rate. The presence of ^{241}Pu will mean a γ-dose-rate (> 40 keV) increase for years due to ^{241}Am buildup. Neutron dose rates depend upon the mass of the sample in the case of spontaneous fission. The dose rate from neutrons may also be significantly increased if low-Z number substances are present as impurities.[9]

In the early handling of plutonium, extremity exposures were of little concern. With the advent of high-exposure plutonium, hand exposures, as well as whole-body exposures, became important, and methods of reducing such exposures had to be provided.[47,48] For transplutonium substances, dose rates are such that whole-body doses must be greatly reduced for all but small amounts of these substances.[49]

1. Exposure Control Practice

Among the practices that may be used to limit the external exposure received by the worker are:[48,49]

- Limit the amount of material in the enclosed system.
- Reduce the total worktime allowed in handling radionuclides.
- Process the material as soon as possible after chemical separation.
- Remove waste matter as quickly as possible so that surface deposits do not build up.
- Use remote handling devices, such as tongs, forceps, etc., as much as possible.[2]
- Rotate personnel for those jobs having the higher exposure rates.

2. Shielding

A number of substances have been used as shields to reduce the external radiation dose. Initially, the normal thickness of material on the enclosure provided enough shielding to reduce the low-energy x rays to low enough levels. As the plutonium isotopic composition shifted, the use of leaded gloves became common practice. These gloves were used to limit hand exposure, but their thickness, in lead equivalents, was restricted since at some point handling operations became too difficult. Common practice has been to limit thickness to .8 mm (30 mil) leaded gloves (0.1 mm lead equivalent). As both γ and neutron dose rates became larger, it was necessary to go to more potent shielding to achieve the needed dose reduction. In the case of gloveboxes, only so much shielding can be included or added (approximately .15 m) and still allow ease of manual handling. Therefore, at some point, hot cells and remote handling devices must be used.

In the case of x and γ rays, high-Z-number substances are most useful as shields. However, a shield material does not absorb equally for all energies of the incoming photons. The transmission expresses the fraction of the incoming photons of a certain energy that pass through a shield of given thickness. Table 16.2 gives photon-transmission values for some common materials used in gloveboxes.

Table 16.2 Transmission of X and Gamma Rays in Selected Materials[16]

Material	Transmission 17 keV	 43 keV	 60 keV	 100 keV
1/2-in. Lucite (Plexiglas)	0.28	0.74	0.81	0.82
30 mil lead gloves (0.1 mm lead equivalent)	0	0.38	0.67	0.54
1/4-in. aluminum	0	0.45	0.59	0.78
1/8-in. steel	0	0	0.046	0.43

As may be noted from the table, as photon energy increases, the shield thickness must also be increased to retain the same shield transmission factor. For transplutonic isotopes, which have more photons in the range above 100 keV than plutonium, the photon shielding problem becomes more troublesome.[16] Attenuation factors for PuO_2 sources in several materials can be found in Reference 9.

In the case of neutron shields, the choice of a proper shield substance is more complex. For a fast-neutron spectrum, the shield must quickly remove the neutrons, yet be thick enough to absorb photons produced in radiative capture.[50] In very thick shields, the neutrons will be removed and still enough matter will be available to reduce the capture gamma ray hazard. In thin shields, one may reduce the neutron hazard but increase the γ hazard.

For fast neutrons, low-Z number substances, such as hydrogen, are most useful as shield materials.[6] One may compare neutron shielding substances on the basis of their hydrogen content. On this basis, polyethylene and water are among the better neutron shields, while concrete is adequate if the water content is approximately 7%. Reference 51 contains data on the shielding ability for some of these materials for fast neutrons. Transmission factors for thin neutron shields are difficult to obtain, since one must deal with a spectrum of neutron energies (see Section 8.E.1). In thin shields, the spectrum changes rapidly over short penetrations, so that the shield effect may depend upon the initial spectrum assumed. In thick shields, removal theory for neutrons may be applied. Reference 52 contains shielding information useful for transplutonium material (^{252}Cf).

3. Health Physics Services

Health Physics provides advice, recommendations, consultations, computations, hazard analysis and safety review and approval of operations and designs in an effort to reduce exposures to ALARA. In addition, presentation of indoctrination lectures to new personnel, refresher training of long-term personnel, and discussions and review of procedures with operations personnel help to foster an ALARA approach to carrying out operations. Frequent observation of the radiation handling practices in a given area lead to ideas for performance improvement which help to reduce personnel exposure.

Other health physics control measures cover monitoring services with survey instruments to define the radiation field and personnel monitoring programs to assess exposure. In addition to these features, fixed monitors that respond to external radiation fields above certain preset levels may be used. The response may include visual and audible alarms to indicate accident situations requiring prompt action.

a. Monitoring Services

A number of survey instruments (see Section 12) are used to estimate the external radiation field. For photons, beta-gamma survey meters (Geiger counters) or ion chamber devices are most often used. Neutron fields may be evaluated by devices that measure the neutron fluence rate (such as the PNC or long counter) or approximate the dose equivalent rate (such as the neutron rem meter). Fluence rate units may be related to the estimated dose equivalent rate by means of the relationships in Table 4.3. These values have been adapted from Reference 51.

For a fast neutron spectrum, such as in fission or (α,n) processes, an average value for the neutron energy is estimated, and the corresponding allowable fluence rate from Table 4.3 for that energy is used for the spectrum. In practice, the allowable fluence rate value for a mean energy of 1 MeV is often used to estimate the dose equivalent rate. Any moderation of fast neutrons then provides a safety factor. This is particularly true when the field is measured after passing through hydrogen-rich shields.

Estimates of the total dose rate from neutrons and gamma rays can be used to assess the hazard. If the dose rates greatly exceed the allowed standards, a severe hazard may exist. In these instances, actions that will greatly limit exposure during handling will have to be taken. For not so severe hazards, only minor changes may be called for.

b. Personnel Monitoring Devices

In the past, the most common device to monitor exposure was the film badge. These devices could be supplied with separate films that responded to beta and photons, and neutrons, respectively. In Section 13, we discussed several alternate devices which are replacing film as the method of choice. These included TL, RPL and Exoelectron devices for photons and beta, and albedo and track etch devices for neutrons. In particular, TL dosimeters are now extensively used and are recommended for plutonium monitoring.[9,25] A pocket dosimeter (see Section 13.C) is often used with the film badge or other monitoring device to allow visual checking of the exposure while it is being received. These devices can be read and recharged during stages of an operation to pinpoint the steps which result in the highest doses. Alternatively, they may be used to determine daily $\beta\gamma$ exposure , and the readings recorded. In some cases, audible dosimeters, which give off intermittent chirp-like sounds and light flashes, may be worn. These devices respond with increasing signal rate as the exposure rate increases.

Finger rings and wrist bands containing TL dosimeters are used for monitoring extremity exposures. Film has been replaced in ring devices to avoid the problem of light leaks, which produce erroneous response. This is not a concern with TL substances.

A problem in the use of film for neutron monitoring by track counting has been poor response in the intermediate energy region (see Section 13.8.a.(1)). The NTA film in common use will not respond to neutrons below about 0.5 MeV. For moderated neutrons, this film may miss a large portion of the neutron dose.

Since much work with plutonium or transplutonic radionuclides leads to moderated neutrons coming through the shielding, personnel neutron monitoring presents a problem. The use of a simple ratio of n/γ doses to estimate the neutron dose is undesirable, unless this ratio is evaluated for each specific process.[9] Albedo neutron dosimeters may be used, but since these are severely energy dependent, one must calibrate their response for the spectrum of interest. Track etch recorders using the plastic recorder CR-39 show promise of covering the energy range of 0.1-18 MeV.[53]

E. Nuclear Safety

Much of the work done in gloveboxes and hot cells involves plutonium and transplutonic radionuclides. Because of the fissionable nature of many of these materials, both with respect to thermal and fast fission, a nuclear safety program is also required. Many of the principles and protective measures discussed in Section 15.I will be applied in achieving such a program.

F. Waste Removal Practices

Waste removal practices set up for different forms of contaminated waste should be observed. Dry waste is often put in metal containers (sphincter cans), sealed in plastic bags, and removed through bag ports. This waste may then be placed in large, covered, metal drums, which can be sealed when filled, and removed for disposal (see Section 16.J.). Liquid waste is handled by piping to waste tanks or containers, by use of small bottles or other containers, and by treatment (evaporation and absorption in vermiculite) that changes the liquid waste to solid waste. Gaseous waste is often planned for in the design features of the exhaust system.[1,4] Pyrophoric wastes are often burned within an enclosure (passivated) before being disposed of.

Waste handling rules should specify the correct method of disposal for items that do not fit standard containers. Also, the correct use of waste containers as well as proper waste segregation should be explained (see Section 16.J.3).

G. Emergency Procedures

Much has been said in regard to preventing certain incidents, yet they still seem to occur. A number of causes can be assigned or postulated in each case, but the cogent feature is that, despite all that can be done, accidents still occur. For this reason, one must accept the premise that certain accidents are going to occur. To properly deal with these situations at any level requires both a well-thought-out plan and trained individuals to execute the details.[54] As one might expect, the more complex the incident, the more detailed will be the plan and the more substantial will be the group involved.

The philosophy that should apply in an emergency situation will reflect these goals:[4]

1. Protect personnel from injury and hazards, and quickly identify exposed personnel.
2. Avoid actions that might create greater hazards.
3. Bring the emergency under control and limit the extent of environmental releases.

4. Prevent property damage.
5. Limit the extent of any contamination.

To deal with an event and form a plan of action will require an initial analysis of the accident potential. Thought should be given to the type and nature of events likely to occur. These may include criticality accidents, contamination release in fires or explosions, or high-radiation fields.[4,54] At this stage, extra precautions should be taken to reduce the likelihood of the event, if possible. One should ponder the extent of the area and estimate the number of people involved. Key areas should be identified, and photographs of the suspected trouble spots may be prepared. Other building data and plans to aid emergency response groups unfamiliar with the area may also be included.

Once the events are defined, written procedures for immediate actions in these emergency situations can be prepared. There may be many facets to the plan, depending upon the nature of the event. However, at every level these plans should (1) be easy to understand, (2) specify actions to be taken, (3) detail responsibilities, (4) define communication lines, and (5) denote sites of emergency equipment. The plans should cover such aspects as evacuation plans, reporting emergencies, and the types and meaning of alarm systems. Assembly points and decontamination centers, and the proper routes to these areas, should be detailed. Drills and training of personnel should be carried out to test response and familiarize workers with their part in these plans. Review of these drills should be carried out to point out weak spots in the scheme so that these may be corrected.

When immediate local actions are not able to cope with an event, a major emergency plan is needed. This will require the response or assistance of many groups; medical people, health physicists, communications personnel, security forces, firemen, utility men, and others. The major plan designates a director whose function is to coordinate the actions to ensure a team effort. He should be assisted by other designated personnel whose responsibility is to provide expertise in certain aspects of emergency control. This group should include someone who is familiar with the area in which the emergency has taken place.

Among the functions that must be carried out in the event of a major radiation incident are:

(1) Evaluation of the situation in regard to degree of hazard.
(2) Assistance in the evacuation or recovery of personnel.
(3) Surveying of persons for contamination and/or radiation.
(4) Accounting of personnel who may have been involved in the incident.
(5) Planning of actions or procedures to bring the emergency under control.
(6) Treatment and decontamination of any injured workers.
(7) Provision of instruments, clothing, portable decontamination units, and other specialized equipment.
(8) Collection of personnel data and preparation of public information.

Following any incident, a review of the sequence of events during the emergency response should be conducted with a view toward improving performance and attaining greater safety in the future. An evaluation of the emergency measures and response of emergency groups may indicate ways of improving the handling of future emergencies.

Since facilities may be vastly different, specific details of an emergency plan must be worked out on an individual basis. The intent here is to simply point out some of the aspects that must be treated in an actual plan. References 54 and 55 discuss several aspects in regard to emergency planning.

H. Hot Cells

Many operations involving radioactive substances must be carried out in shielded enclosures called "hot cells" or "caves."[3] Hot cells protect workers from intake of these substances and from exposure to penetrating radiation.[2] The main parts of a cell are the shielding walls, cell liner, services, ventilation system, and, in some cases, a drainage system.[13] Special-purpose cells may also include viewing windows, remote-handling devices (manipulators), hoists or cranes, and access ports. Figure 16.5 shows the layout of a multicell facility.

To keep the working areas free of airborne activity, cells may be hermetically sealed or maintained under a negative pressure. The latter method is generally desired for large multipurpose cells. An air flow prevents the buildup of troublesome vapors and removes heat from equipment and lights.

The containment required is related to the state and toxicity of the substance being handled. Gases may be trapped or, if the amounts are small or the half life is short, they can be dispersed via tall stacks.

The surfaces of some solids may oxidize and produce fine airborne particles. Some of the more toxic substances are also pyrophoric, and a fire would produce radioactive smoke. Alpha emitters, being highly toxic, require good containment. However, these substances can be handled in thin-walled, in-cell gloveboxes. For processes involving large amounts of both alpha and gamma emitters, remote handling must be done in cells that provide adequate shielding and a high degree of containment. In these cases, an in-cell glovebox may still be used, in which the hot cell manipulators, sealed in vinyl plastic which also forms the roof of the glovebox, are used to perform the operations.[3]

1. Shielding

Materials used for hot cell walls include concrete, steel, lead, and water. Concrete is by far the most commonly used material, mainly because of the low cost. It can be obtained in a range of densities-from

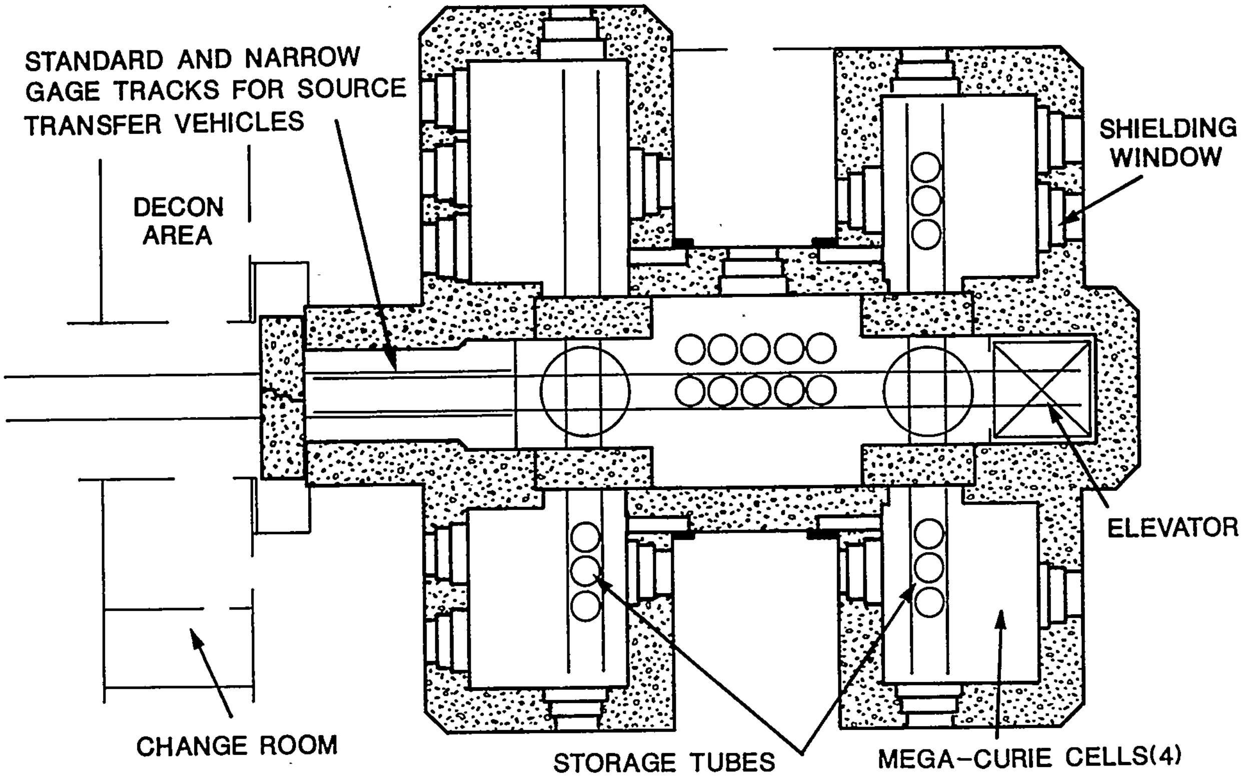

Figure 16.5 Multicell hot cell facility in the form of an H.

2.2 to 5.9 Mg/m³. The higher-density concretes are used, in most cases, for monolithic shielding structures.[56] The added cost is justified by reduced space requirements and smaller remote handling devices.[57]

The choice and thickness of shielding material for a given high-level gamma cell are often based upon a certain amount of γ activity. For example, a megacurie hot cell is designed to provide sufficient shielding so that 10^6 Curies (37 PBq) of a 1 MeV γ emitter can be handled safely with respect to exposure. The activity may be assumed to be at a point.[3,56] This is a conservative approach since a finite source is always less intense. The unshielded dose rate D_o at the position of the worker is approximated by

$$\dot{D}_o = \frac{1.264\text{x}10^{-13}\, n_\gamma\, C\, E_\gamma}{r^2}\ \text{(Gy/h)}, \qquad 16.1$$

where $n_{\gamma}C$ is the γ activity in γ/s, C is the source activity in Bq, E is the energy in MeV/γ, and r is the distance between the source and the worker in meters.

In passing through the shield, the gammas undergo absorption and scattering processes. This requires the use of a buildup factor, b, in the shielding calculations.[56,58] The value of b varies with γ energy, shield material, source and shield geometry, and depth of shield penetration. The attenuation of monoenergetic gamma rays by a given substance is then obtained from the simplified expression.

$$\dot{D} = \dot{D}_0 b e^{-\mu x}, \qquad 16.2$$

where $\dot{D}$ is the reading at a point outside the shield, D_0 is the reading at the same point without the shield, μ is the attenuation coefficient, x is the shield thickness, and b is the buildup factor for a given value of μx (see Section 8.D.2).

Buildup factors for point isotropic sources for common shielding substances are given in the references.[59,60]

A further complication is introduced for cells designed to handle transplutonic radionuclides. For these materials, the neutron emission is significant enough to warrant attention. For these radionuclides, it is generally sufficient to assume a neutron energy spectrum similar to a fission spectrum. One is then concerned with fast neutrons, thermal neutron capture gammas, and activation of the shield itself. Sometimes, in high-level γ cells, the concrete is thick enough (1-2 m) to adequately attenuate the neutrons, but other components of the cell, such as steel doors or the windows may offer inadequate shielding.[49]

The walls of a hot cell should always be checked for leakage prior to initial use.[56] A high-energy γ source is often used for this purpose. It should be placed in all possible future source positions. The leakage survey can be made with rate meters, dosimeters, or film. Large x ray films have been used to conveniently map large areas during the leakage survey. If neutrons are expected, one will need to perform neutron leakage measurements also.

2. Ventilation

Ventilating systems control the temperature and limit the spread of contamination both inside and outside hot cells. Cells are maintained at negative pressure relative to the working area to prevent the escape of contamination.[3,9] When possible, airflow patterns are set up so that air is carried to a cleaning system. This exhaust air may contain contaminants and must be cleaned before release.

Air cleaning can be achieved with filters (for particulate matter), with scrubbers (for particulates and gases), or by adsorption (for gases).[15] Prefilters are often used to extend the life of the more costly final filters. The use of filters on air inlets may offer some protection-for example, if the airflow is momentarily reversed due to an explosion or fire.[15]

When doors or access ports are opened, the air exchange should increase to keep the inlet velocity above a minimum value. A second fan is often used for this purpose. This fan turns on automatically when the cell is opened.[57]

An airflow can be maintained in completely sealed cells. The cell air is pulled through a filter system and recirculated back into the cell.[15]

3. Viewing Facilities

Windows, periscopes, mirrors, and television are some of the means employed for viewing remote operations. The choice of viewing method depends upon the process to be performed, the information required and the skill and desires of the operator.[56]

a. Windows

A solution of about 78% zinc bromide in water has been used widely for liquid windows.[3,56,57] The $ZnBr_2$ solution (ρ = 2.5 Mg/m^3) is highly transparent, rather stable under irradiation, and relatively cheap. If the solution is exposed to air or radiation, coloration occurs with a corresponding decrease in light transmission. This type of window can receive a total dose of about 10^5 Gy (10^7 rad) before discoloring enough to require a reducing agent or replacement of the zinc bromide.

Lead glass (ρ = 3.27-6.22 Mg/m^3) can be used with zinc bromide. This combination can produce a window having a shielding capacity for γ equivalent to that of the cell wall.

b. Periscopes, Mirrors and Television

Periscopes are often used for detailed examination of the work within a hot cell. A periscope is a lens system that conveys images from the objective end of the system (inside the cell) to the eye of the worker. Mirror systems can augment direct viewing and permit one to follow the progress of simple operations with fairly low activity substances in open-top cells.

Closed-circuit television is useful for performing widely spaced operations, for viewing inaccessible areas, and for viewing by more than one person. It is hampered, however, by the lack of depth perception and the small field of view.[2,3,56]

4. Remote Handling Devices

In many cases, handling radioactive substances with the bare hands is unwise. Therefore, many operations are carried out with remote handling devices (manipulators). Many such devices have been developed, the particular design depending upon the process to be performed.[3]

Tongs can be used for simple operations with substances of low activity. They have only one motion, that of grasping, and provide distance as protection for the worker.

The ball-joint manipulator is often used in small hot cells.[3,56] A rod having a handle at one end and tongs at the other, moves through a ball joint to adjust the reach. This shielding ball joint is mounted in the cell wall. The rod axis can also be moved, but is confined to a cone-shaped volume having an apex angle of about 70-90°. This device is useful for radiochemical or simple mechanical functions.

The more complex processes performed in many large hot cells require the use of general-purpose manipulator systems, often called master-slave manipulators.[2] These manipulators have a master arm (control arm) outside the shielding wall and a slave arm (working arm) within the cell (see Figure 16.6). The arms are connected (with remote control linkages) so that motion applied to the handle drives the slave arm in a like manner. These master-slave units have at least seven independent motions-three along the x, y, and z coordinates, three rotational, and one for gripping objects. They may be obtained with arms having unequal lengths and also different distances between the two arms. Snap-in type tongs (or fingers) can be changed remotely. The slave arms

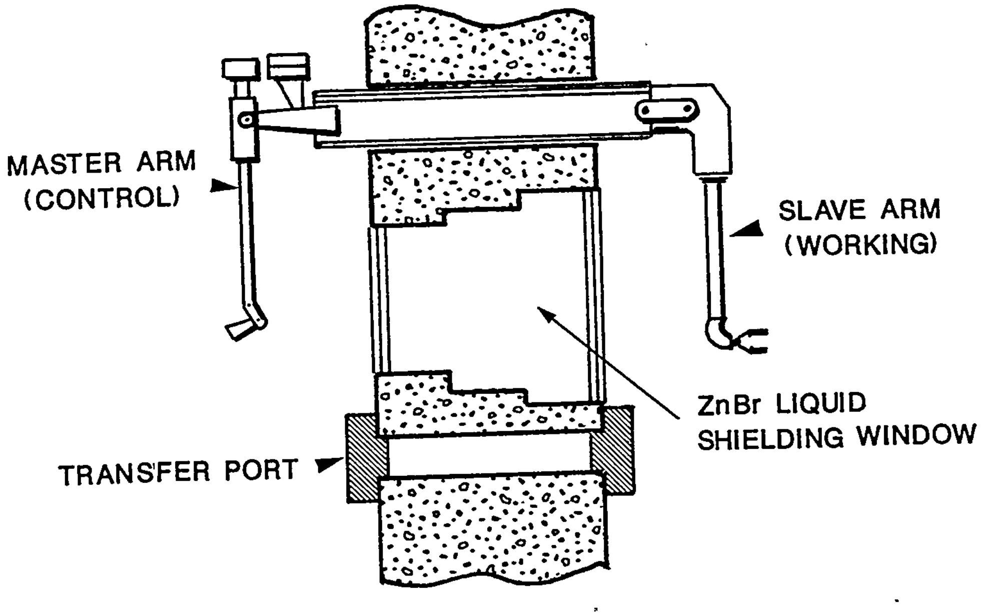

Figure 16.6 Section through a hot cell wall.

may be obtained with or without sealed boots (flexible rubber or plastic bags which are sealed to the wrist of the device at one end and sealed by a glove ring mechanism at the wall penetration).

Some of the many types of remote-handling devices are further discussed in the literature.[3,56]

5. Monitoring Hot Cell Operations

Hot cell operations require monitoring services because of the radiation exposure problems that may be encountered in handling the high-level sources used in these facilities. Monitoring services (radiation protection surveys) are provided on a routine basis and/or upon request. They can be provided while an experiment is in progress, or during cell transfers, decontamination, filter changes, etc. These services include measurements of radiation fields to set working time limits or to check the adequacy of shielding walls, and the collection of air samples plus direct and smear surveys of surfaces to detect and/or control environmental contamination. Samples of the cell exhaust can be checked to determine and control releases to the atmosphere.

Some of the various cell operations that may require monitoring services are briefly discussed below.

a. Cell Transfers

Movement of items into or out of a cell (transfers) can be accomplished with pouches or bags, transfer drawers, map tubes, shielded casks, etc. Items brought out of a cell may be covered with loose activity. Therefore, these items are normally surveyed before complete removal from the transfer system. Gamma-emitting objects are generally checked remotely with a survey meter in the cell before they are withdrawn. The meter reading shows whether the object should be extracted directly into a shielded carrier.

Highly toxic substances are often placed in pouches that are sealed upon removal. The sealed end is a likely spot for contamination and therefore should be surveyed. This bagging technique is also useful for cells that must be maintained under an inert atmosphere.[61]

b. Decontamination Operations

Hot cells may need to be decontaminated to install new equipment when the nature of the experimental work changes, to make repairs that cannot be done remotely, or to prevent the activity from reaching unmanageably high levels.[3] This cleanup process may require consideration of such things as personnel exposure to radiation, contamination control, criticality, and the handling and disposal of active wastes, as in the case of glovebox systems.

c. Filter Changes

Air cleaning filters used in hot cell exhaust systems must be changed periodically. This changing operation can present a contamination or personnel exposure problem because of the accumulated waste matter.[56] Push-through filter-exchange systems, with or without shielding, have been developed to overcome some of these problems. In some facilities, the first of a series of filters can be changed remotely with the cell manipulators. However, these protective measures are not available for many of the filter exchange operations; therefore, these operations are usually covered by radiation surveys. In some cases, respirators may be required.

6. Control Measures

Many of the health physics control measures and services discussed in 16.B.5, 16.C, 16.D and 16.E also apply in the case of hot cells or shielded facilities. That is, a program of routine checks and surveys needs to be performed to continually evaluate the effectiveness of the control program.

I. Decontamination

As has been discussed earlier, operations in gloveboxes and hot cells lead to the accumulation of high levels of loose radioactivity, which is contained in the enclosure. In addition, the use of certain forms of radioactivity in hoods will also lead to levels of loose activity on the inside hood surfaces. The levels which can be tolerated will be related to the containment and protection which the enclosure provides. Depending upon the quality, nature and form of the radionuclide used, and the operations involved, the degree of necessary containment varies. Even for adequate containment, at some point, the buildup of activity in the enclosure reaches an undesirable level and cleanup is indicated.[3] In other cases, the operation and/or the radionuclides used may change so that one may desire to clean up the old radioactivity in the system so as not to affect the result of experiments with the new radionuclide. In some instances, loss of control occurs leading to a release of radioactivity from the enclosure into the worker's environment (often referred to as a "spill"). This requires location of the source of leakage, re-establishment of the required degree of containment, and cleanup of released radioactive material. The cleanup phase may also involve equipment, clothing and personnel which were involved in the release of material.

The general term used for the radioactivity in the above situations is contamination. The term contamination is generally used to refer to the presence of unwanted radioactivity in any place where it may present the potential to harm personnel, spoil experiments, or render products and equipment unsuitable, or unsafe, for further use.[62] Loose contamination is usually surface contamination which can be easily removed, or rubbed off the surface to which it clings, and transferred to other sites. The term "fixed contamination" is a relative term used to describe contamination which is not easily removed. Since it is not really fixed to the surface, in time, this type of contamination may also become loose contamination.

The control of contamination at the source is a prime objective of a radiation protection program. This action limits the spread of contamination and helps to reduce cleanup time and costs, as well as loss of equipment and facilities, due to decontamination. Many features of contamination control have already been discussed in this section. When such actions fail, or incidents occur, or even when certain modifications are being made to a facility, the need for decontamination may arise.

If undetected or not properly removed, radioactive substances may enter the body, be taken home on contaminated clothing and shoes, be spread to other parts of the facility, or interfere with, or spoil, sensitive equipment or experiments. For these reasons, contamination should be promptly removed as soon as it is discovered. This reduces the chance of dispersal to other areas. The proper response in finding and removing contamination represents a prominent part of effective contamination control.

1. Principles of Decontamination

Although it is desirable to remove all contamination, there may be situations in which only a degree of decontamination is required.[42,43] Certainly all loose contamination should be removed, but sometimes the cost and effort required to remove the relatively fixed contamination can be considerable.[42] In some cases, it may be more costly to decontaminate an article than to simply replace it. In other cases, a certain level of contamination may be acceptable since the item or piece of equipment will be used inside a contaminated enclosure.[3] It may be more economical to just transfer the contaminated item from enclosure to enclosure without cleaning it. If the item is to be released, the criterion for release may allow certain limits of contamination based upon the specific radionuclide involved.[31,64] If a laboratory is contaminated in a spill though, the objective may be to remove all contamination since it may represent a hazard to personnel. However, in skin contamination cases, contamination levels below allowable levels may be tolerated if drastic removal methods would result in skin damage.[42]

The approach to decontamination is often complex because of the many available elements which have radioactive isotopes, as well as the form of the contamination: solid, in solution, or carried by a gas or vapor. In addition to the chemical, physical and radiochemical nature of the contaminant, the material and surface characteristics of the substrate affect decontamination. Generally, radioactive materials are held on a surface by physical adsorption (most fission products and all heavy natural radionuclides),[63] or adsorbed from suspensions and deposited on substrate (pores and indentations).[62] To remove the contamination will often require the use of a specific chemical to dissolve the particular contaminant.[63]

For a given choice of surface material, the case of decontamination will be related to the manner in which the surface is contaminated and the particular decontaminating chemical agent chosen. For metallic surfaces, the contamination tends to become incorporated in the metal making removal difficult.[63,65] Organic surfaces (paints, plastics and textiles) and vitreous surfaces (glass, porcelain) have a capacity for ion-exchange which is probably the most important contamination mechanism. All porous surfaces are easily contaminated and difficult to decontaminate. When metal and porous materials are used, strippable coatings are usually employed to seal the surface. During decontamination, the coating is often removed with the contaminant so that a new strippable coating must be applied.

In order to provide ease of decontamination, an ideal surface should have these features:[63]

(1) Be non-absorbent since porous materials are very difficult to decontaminate:

(2) Contain as few acidic groups as possible since these groups are chemically reactive;

(3) Have a low moisture content;

(4) Be protected from exposure to solvents or chemicals which attack the material;

(5) Possess sufficient chemical resistance to withstand decontaminating agents;

(6) Be capable of withstanding abrasive action;

(7) Be smooth with no cracks and ledges;

(8) Be resistant to heat and radiation.

Since no one material exhibits all these features, compromises have to be made with respect to use of materials which have deficiencies with respect to decontamination. As mentioned above, often the permanent surface will be covered by a temporary surface which can be easily removed for decontamination purposes. Among the more frequently used strippable coatings are latex paint, PVC or PVA sheet, coated paper or polyethylene.[63]

With respect to the choice of decontaminating agent, almost all detergents will do the job.[42,63] The constituents of a detergent which aid in the decontamination process are:[63] a wetting agent to emulsify oils and grease, a suspending agent to capture dirt and prevent redeposition, and complexing agents to react with the contaminants to form compounds. These agents have the virtue of cleaning the surface without damaging the integrity of the surface. However, repeated applications can cause damage in polyethylene and PVC.[63] Acids and other aggressive chemical agents, such as chromic, sulfuric, nitric, phosphoric and citric acid and sodium hydroxide, aqua regia and acetone may be used on various surfaces, but these remove some of the surface material with the contaminant.[42] This approach may be undesirable for equipment that must be reused. Once damaged, these surfaces then tend to collect contamination easily. If acid or other chemicals are not effective, then sandblasting with a fine sand may be tried. This latter method may result in contamination of the sandblaster as well as the abrasive agent.

A number of commercially available decontamination agents are available on the market under various trade names. Summaries of decontamination approaches can also be found in the literature.[66,67]

2. Decontamination Approaches

a. Working Areas

Initial decontamination efforts should be directed toward removal of loose contamination. In the case of floors or other horizontal surfaces, the surface may be dry vacuumed with special vacuum cleaners equipped with HEPA filtering of the exhaust. This procedure will eliminate a lot of the easily removed contamination. Then the surface may be wet mopped first with soap and water or detergent, and vacuumed up. For smaller contaminated areas, swabbing may be used. The use of swabs (detergent-soaked rags) should be with a wiping motion, not rubbing, and the wet rags should be frequently discarded as radioactive waste. Following this, the area should be resurveyed and remaining contaminated areas marked. If fixed contamination is indicated by the survey results, the area should be washed with a solution containing complexing agents. If contamination remains after several attempts, then removal may be undertaken with abrasive material (metal polish, abrasive creams, steel wool, etc.).[42] If the contamination persists, the surface may need to be removed by scabbling.[68] However, an alternative is that the level may be low enough that a seal may be used (paint, concrete, or other material) over the radioactivity. Application of any specific procedure, and the decontaminating agent used, depends upon surface characteristics.

b. Equipment

Wash all glassware with chromic acid cleaning solution or with concentrated nitric acid as a routine procedure following use. If these leave the glassware still contaminated, mineral acids or solutions of ammonium citrate, trisodium phosphate, or ammonium bifluoride may be tried. If decontamination is difficult, it is usually more convenient to replace the items. In order to prevent accidental return to stock or to other use, break discarded glass equipment before disposal.

Metal objects may sometimes be decontaminated with dilute mineral acids, a 10% solution of sodium citrate, or ammonium bifluoride. The use of strong acids on metal tools may corrode them, thereby causing greater difficulty in future decontamination procedures. If other procedures fail, hydrochloric acid may be used on stainless steel. Since this will remove some of the surface, use it only as a last resort, unless the equipment is to be discarded as decontaminated scrap. Brass polish is an excellent decontaminant for brass. Oxalic acid generally is satisfactory for rusty surfaces. Titanium dioxide paste is a good agent for removing fission products from metallic surfaces; do not let the paste harden, for it is then extremely difficult to remove.

Plastics may be decontaminated with ammonium citrate, dilute acids, or organic solvents.

c. Clothing

Generally, clothing used during decontamination operations consists of plastic garments and/or paper suits (Tyvek) which are discarded at the end of the operation when the worker leaves the contaminated area. Figure 16.7 shows a worker, involved in volume reduction of highly-contaminated plutonium gloveboxes, dressed in a plastic suit which is worn over paper coveralls. In this case, plastic garments, paper suit, gloves and shoe covers will all be discarded.

For regular protective clothing, if the contamination levels are not too high, the garment may be washed in a washing machine, utilizing soaps or chemical agents (detergents with certain additives, such as citrus acid) to attempt decontamination. For very high levels of contamination and/or involving highly hazardous radionuclides, it is generally more advantageous to discard the clothing, rather than to try to clean it.

For operations involving high hazard material, the clothing which is supplied is usually restricted to that specific control area. This helps in the control of contamination since the individual is required to change into other clothes before exiting the area.

d. Personnel

With respect to personnel decontamination, one should follow this guidance: if a few decontamination efforts do not work, call in a physician and/or decontamination efforts should be discontinued when the skin starts to become thin and reddened. If the integrity of the skin is damaged, then absorption of radioactive material may occur leading to potential internal deposition of radioactive material. All personnel decontamination procedures should be reviewed and approved by the plant physician.

Figure 16.7 Worker dressed for tent entry.

(1) Hand Washing

One procedure for washing contaminated skin and hands is the following:

(a) Wash thoroughly for two to three minutes by the clock. Use tepid (not hot) water and a mild soap or detergent, such as Tide or pHisoderm or Radiacwash. Cover the entire surface of the contaminated area with a good lather. Rinse off completely with water. Repeat the process at least three times. Do not use abrasive or highly alkaline soaps or powders.

(b) Eight-minute surgical scrub. If the above procedure is not enough to remove all dirt and contamination, scrub the hands for a period of at least eight minutes by the clock with a liquid or cake soap, hand brush, and tepid water, being sure to brush the entire surface of the hands, especially around the nails and between the fingers. Light pressure should be exerted on the brush - do not press so hard that the bristles are bent out of shape. Eight minutes is usually a sufficient time to allow three complete changes of tepid water and soap. Each one of these three washings should be so thorough that the brush will cover all areas a minimum of four strokes. A convenient routine is to start by scrubbing one thumb, being sure to brush all surfaces, proceed to the space between the thumb and first (index) finger and similarly to each finger and the webs between the finger.

Give attention to the palm and the back of the hand and, finally, scrub the nails and cuticles before proceeding in the same manner with the other hand.

(c) Use an appropriate radiation survey instrument to verify the removal of contamination from the dried area, then use lanolin cream to soften the hands and prevent chapping.

(d) Discard the hand brush and towels after they have been used for the removal of contaminated material (use an active waste can).

(2) Titanium Dioxide to Remove Fission Products

Titanium oxide (TiO_2) may be used as a paste or slurry made by shaking the powder into the wet palm of the hand until a good paste is formed. Run tap water over the hands continually so that the paste is kept wet, and apply this later thoroughly to all hand surfaces, especially around the finger nails, for a minimum time of two minutes.

Rinse off thoroughly with luke-warm water and follow by a thorough washing with soap and water and a hand brush. If any of the paste is left under the nails after washing, it will form a rather hard cake which is difficult to remove.

(3) Potassium Permanganate to Remove Plutonium

(a) Mix an equal volume of a saturated solution of potassium permanganate ($KMnO_4$) with 1% sulfuric acid solution (0.2 N). Pour this weak acid solution over the wet hands, covering the nails and cuticles thoroughly. Rub the entire surface lightly with a hand brush without applying enough pressure to bend the bristles out of shape.

Use running water (tepid, not hot), and rinse off after the application has thoroughly covered the hands. Do not continue this procedure for more than two minutes. This process will stain the skin a deep brown.

(b) Use a freshly prepared 5% sodium acid sulfite solution in the same manner as above, using the hand brush and tepid running water for a two-minute period. This solution will remove the brown stain on the skin. (It is convenient to keep labelled packages containing 10 g $NaHSO_3$ on hand and dissolve this amount in 200 ml of water.)

The complete procedure above may be repeated several times without appreciable harm to the skin _if_ each washing is limited to two minutes.

An abundance of water is essential. To wash other skin surfaces such as neck, face, ears, etc., the solutions may be applied with absorbent cotton. If another person is manipulating the solution, rubber gloves should be worn as protection from both contamination and permanganate staining.

Recommendations for a skin decontamination kit with respect to supplies needed can be found in Reference 59.

e. Glovebox Approach - General

The initial decontamination step is to remove all items within the box, either for separate treatment or for disposal as active waste.[3] This usually requires that these items be bagged in plastic and removed from the enclosure through a glove port or transfer area, if available. Following this procedure, sweeping or vacuuming of the box surface takes place. The debris collected by these procedures is then also bagged out. Swabbing of the walls and floor of the glovebox is then performed. Alternatively, a water spray may be used in some cases to wash down the walls if criticality safety considerations permit. Liquid from this separation is often collected by siphoning into a liquid waste container. The remainder is then wiped up with rags, and, if feasible, these rags are allowed to dry by evaporation, before being bagged out in a long pouch which is attached to one of the glove ports. Several such washings may be required in order to reduce the contamination levels sufficiently. Radiation surveys of the glovebox surfaces following each washing determine the effectiveness of the decontamination effort. Since the glovebox system is re-used for radioactive material, it is not necessary to clean the box down to very low activity levels. In some cases, as shown in Figure 16.8, it may be necessary to construct a temporary enclosure in order to remove the equipment from a glovebox without creating a loss of control situation which will contaminate the area. In this operation, in order to dismantle a fairly large piece of equipment, an enclosure was constructed on the front face of the glovebox to insure that airflow was into the enclosure. The face of the glovebox was then removed to dismantle the large piece of contaminated equipment.

f. Hot Cell Approach - General

Hot cells may need to be decontaminated to install new equipment when the nature of the experimental work changes, to make repairs that cannot be done remotely, or to prevent the activity from reaching unmanageably high levels.[3,56] This cleanup process may require consideration of such things as personnel exposure to radiation, contamination control, criticality, and the handling and disposal of active wastes. The first phase, general cell cleanup, is usually carried out by remote means. Protective coatings, active scrap, small tools, loose debris, etc., can be placed in waste-disposal drums or picked up with a vacuum cleaner. This can be followed by the wiping of machinery and any remaining "hot spots." Liquid wastes are often absorbed in vermiculite or other inorganic substances and discarded along with the solid waste. This first phase is more effective if the work is guided by a high-range survey meter. The high-level waste is removed before direct decontamination takes place.

Cell entries require the wearing of protective clothing (coveralls, caps, shoe covers, and gloves) and respiratory devices. Disposable suits and shoe bags made of paper or plastic film are often worn over the above items if the cell contamination is severe. They are removed and discarded as waste when the person leaves the cell.

Respirators are often worn for short work periods. However, an air-supplied head hood offers more comfort as well as a higher degree of protection for an extended work period. One-piece suits may be used in handling more toxic substances. The suit is also removed and discarded as radioactive waste when the worker leaves the cell. Health Physics surveys of the worker as he discards potentially contaminated clothing, help to control the spread of contamination by discovering

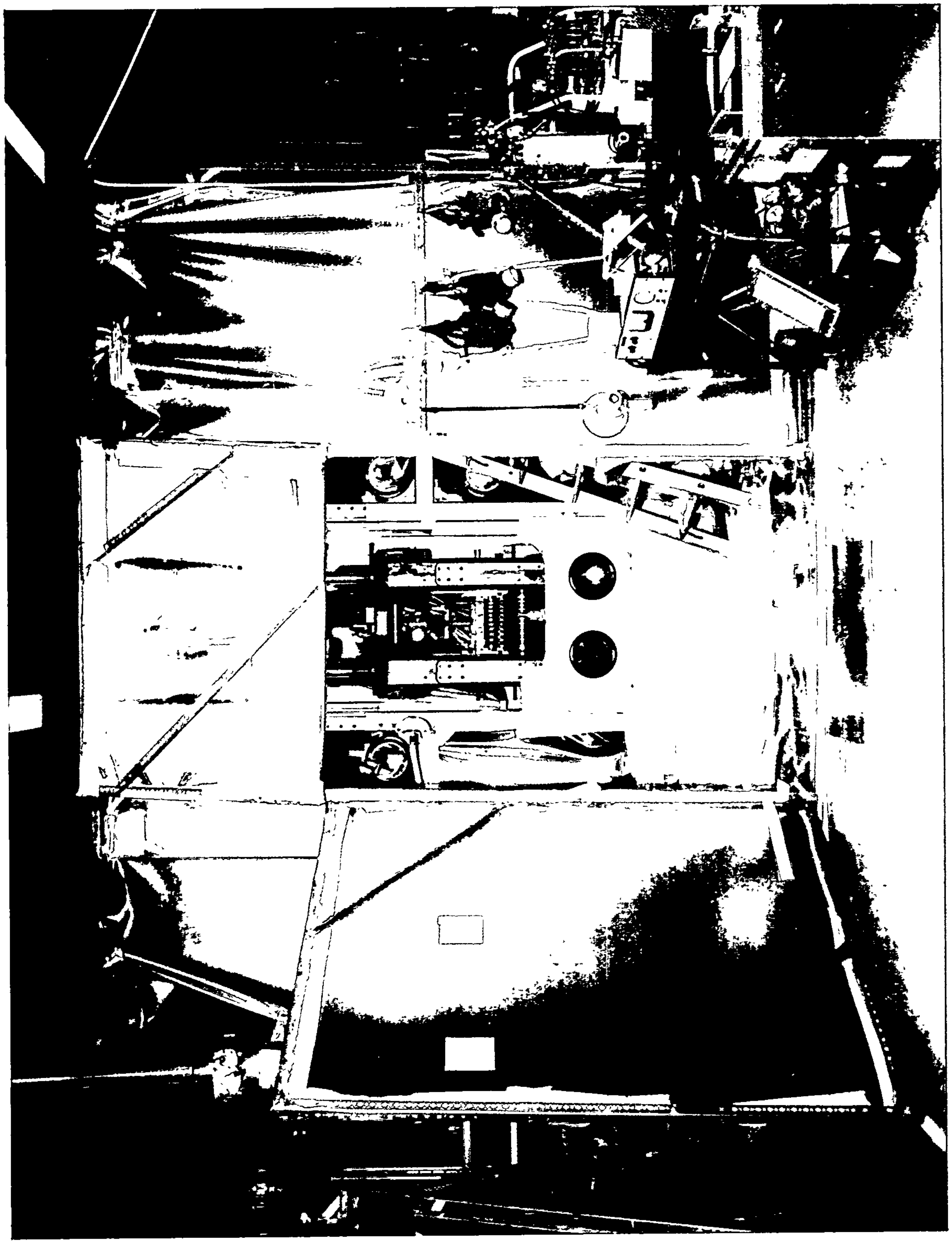

Figure 16.8 Separate temporary contamination control enclosure attached to a glovebox.

contaminated items prior to the worker leaving the cell. There is usually more than one worker in the cell at a given time, and this allows each to assist the other in removal of potentially contaminated garments. The Health Physics person is used to check each worker before he exits from the transition zone (between contaminated areas and clean areas).

Direct decontamination is often guided by in-cell radiation surveys. Items or areas of highest intensity are removed or cleaned first. This lowers the radiation field, which, in turn, reduces the exposure to the decontamination crew and health physics surveyor. The remaining tools and equipment can now be removed to an adjacent room or cell where they are further wiped down. Disposition of these items depends upon the success in cleaning, which is determined through surveys. The items may be sent to a "hot lab" equipment storage area pending future use, or dismantled and cleaned again using more effective methods and equipment.

In-cell containment boxes are often used within the cell for operations which have a high potential for spreading contamination. These units generally have such high levels of activity that discarding the entire box upon completion of the experiment may turn out to be more economical than decontamination.[3] Since these units may need to be somewhat dismantled, monitoring service is also required for this operation.

Communication between the crew in the cell and those on the outside of the cell is hindered by the protective equipment and the surroundings. Respirators, coveralls, plastic suits, the noise of supplied air, shielding walls, and distance all act to preclude effective communication. For this reason, it is important that the crew have some means, such as radio contact, of communicating with personnel outside the cell.

J. Waste Disposal

A consequence of using radioactive material is that ultimately one must consider what must be done to dispose of this material safely. Of all the possible radionuclides in use, many of these have sufficiently long half lives so that the material will still be around long after a given experiment has been completed. This poses a problem since there is no way to treat, or process, the material in order to alter the half life of the radionuclide. When the half life is relatively short, one may retain the material in storage until the activity decays away. When the waste contains highly radioactive, long-lived radionuclides, special disposal methods are required. The proper disposal of radioactive waste is a matter of concern and is regulated by federal and state regulatory agencies.

1. Types and Sources of Wastes

Radioactive wastes may be classified according to their physical state as solid, liquid and gaseous. Practically all operations with radioactive materials produce small amounts of solid waste in the form of contamination of the materials or equipment used in the operation.[69] In certain chemical studies and other experiments, solutions of radionuclides are used which result in the production of liquid waste. In these studies, exchange reactions may produce solid radioactive waste in the form of precipitates or treatment columns (ion exchange resins). Certain chemical reactions or thermal reactions can lead to the production of radioactive waste when their useful life is over. Decontamination of enclosures may often produce both solid and liquid radioactive wastes to be dealt with. Certain experiments may result in the release of radioactive gases followed by the production of both solid and liquid radioactive waste. Sometimes, large pieces of contaminated equipment may need to be disposed of following the shutdown of an experiment or facility.

Considering the three basic forms of radioactive waste, other categories or classifications may also affect the method of disposal: soluble or insoluble, combustible or noncombustible, with respect to solids. The radionuclide content with respect to the half lives and radiotoxicity of the constituents will be important factors in the treatment of the radioactive waste. In addition, the activity levels will also dictate the potential disposal mechanism which can be used.

There are four main activities which are the basic sources of radioactive waste production. The first is the mining, milling, feed preparation and fuel manufacturing activities, called the "front end" of the nuclear fuel cycle. Wastes from these activities generally result in small amounts of natural radioactivity (^{238}U, ^{235}U) products - ^{230}Th and ^{226}Ra in tailings being of most concern.[70] Second, the largest source of radioactive waste in the sense of contained activity, is the fuel irradiation and processing activities. The chemical processing of the irradiated fuel results in highly radioactive waste containing fission products, activated reactor materials, corrosion products and chemicals. The third source is activation of non-fuel materials in reactors or and accelerator components. This includes samples inserted into reactors or accelerators, as well as reactor and accelerator structural materials and impurities in coolants. The fourth source is the use of radionuclides in medical, radiopharmaceutical, industrial and scientific research applications. It should be noted that an operating power reactor is a potent source of radiation and that release of some of this radioactivity occurs under normal operating conditions.[70] To minimize those releases, certain cleanup systems are utilized which remove radioactive material from the effluents so that the discharge of radioactive effluents is ALARA. References 70-72 discuss some of these systems for PWR and BWR units. In the removal of the radioactive products by these systems, residues and resins containing the removed radioactivity become solid radioactive waste which must be dealt with.

2. Disposal Philosophy

The objectives of a radioactive waste management program are to determine the steps necessary for the safe disposal of radioactive waste and to see that these steps are followed. The necessary control mechanisms may be quite complex and there may be many types of waste to deal with. However, there are two basic approaches which have been generally applied, in the case of radionuclides which do not have short half lives; these are:[42]

(1) Concentrate and Contain - Reduce the volume of the radioactive waste to concentrate the radionuclide and then safely store the waste in a controlled area (contained). For very highly radioactive material, the waste may be solidified for long-term storage. This approach

is generally applied in the case of high levels of long-lived activity for which indefinite storage is indicated. By reducing the volume of contaminated material, one lessens the cost of disposal. These wastes are generally stored in isolated areas to minimize the chance of contact with humans or ecological systems.

(2) Dilute and Disperse - Use air or water to dilute the given concentration so that when released it will be within the release criterion. This requires that the initial concentration be of low activity to begin with. By dispersing the diluted concentration into the atmosphere or into a waterway, a further reduction in the concentration is achieved. Disposal by this method is regulated by federal standards (for example, the DAC values for allowable offsite concentrations in DOE Order 5480.xx) which also require that one consider possible ways to keep releases well below these limits (ALARA).

3. Solid Waste Disposal

A variety of radioactive waste products which vary in half life and activity may be encountered when dealing with solid radioactive wastes. This includes such materials as paper, rubber gloves, glassware, metal tools, animal carcasses, plastics and large equipment items. The cost of disposal will depend directly on the volume and weight of the radioactive waste produced. To minimize the waste volumes requires advanced planning, facility and equipment design and control of work methods. It is essential to separate ordinary nonradioactive trash from solid radioactive waste, at the point of origin. For this reason, solid radioactive waste containers should be clearly identified with the radiation symbol and easily distinguishable from ordinary trash containers. A number of suitable containers should be distributed through a work area. Figure 16.9a shows some solid radioactive waste containers. The basic collectors are a 28.3 l (1 ft^3) fiber drum and an 18.9 l (5-gallon) pail. A stainless steel secondary container (Blickman can) is generally supplied with the containers. The cover of the secondary is opened by stepping on a treadle. The sliding cover is supplied with a mechanical spring to ensure the return of the cover to the closed position. For cases in which larger quantities of slid waste may be produced, a 127.4 l (4.5 ft^3) cardboard carton (see TV carton in Figure 16.9b) may be used. For waste expected to produce readings > 2 mGy/h (200 mrad/h) at the receptacle surface, a shielded secondary container (approximately .05 m lead equivalent) can be supplied (see Figure 16.10). Larger shielded containers are also available.

The waste producer is required to provide documentation of the identity and estimated quantity of radioactivity and see that the waste is properly labeled and contained. Because disposal is governed by the most restrictive limit for radionuclides in a mixture, and waste acceptance criteria state that some materials are prohibited from the waste, and waste volumes should be kept to a minimum to reduce waste handling costs, radioactive wastes are segregated at the source. Solid waste is segregated into combustible and noncombustible, as well as compressible and noncompressible. One further segregation of solid wastes is made. If the waste consists of α emitting material of ^{233}U or radionuclides with $Z \geq 93$, of half life > 20 y, it is designated transuranic (TRU) waste if the radionuclide concentration is > 3.7×10^6 Bq/kg (100 nCi/g). Secondaries are labeled to indicate the category of waste permitted in the container (see Figure 16.8).

By segregating compressible waste from noncompressible, one is then able to compact the waste to achieve a volume reduction and a subsequent cost saving. Combustible waste in some cases may be effectively treated by incineration to greatly reduce the waste volume. It is necessary to separate TRU and non-TRU waste since these will be disposed of differently. At present, TRU waste is sent to an above-ground interim storage site for subsequent shipment to the Waste Isolation Pilot Plant (WIPP).

Special requirements must be met for certain solid waste such as biological materials (animal carcasses, other biological material). These materials tend to form gases or produce liquids as they decompose. No liquid is permitted as free liquid (not absorbed in a host material) in solid waste and there are limits on the allowable pressure in a container (< 7 psig). To prevent bacterial action which could cause gas buildup

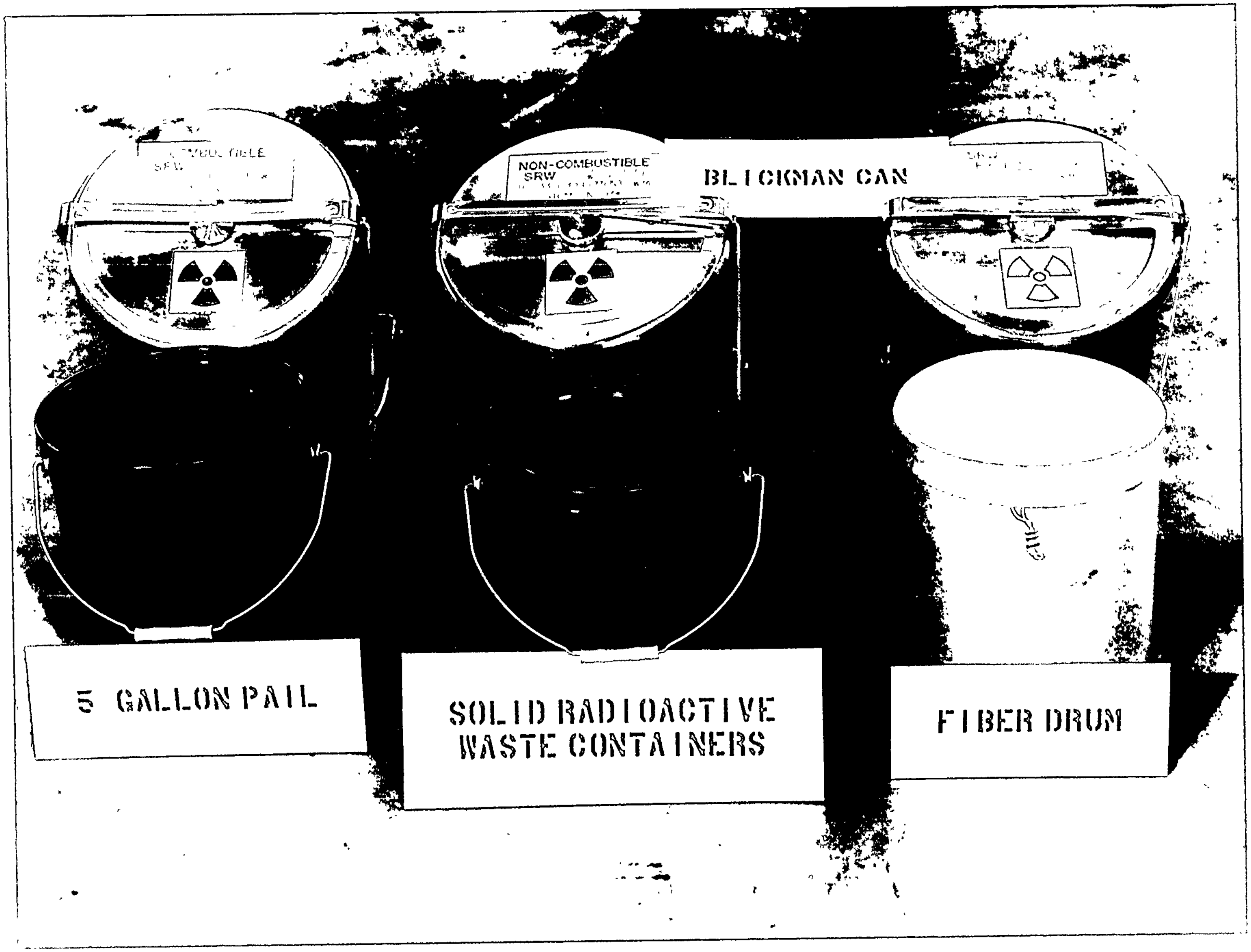

Figure 16.9a Solid radioactive waste containers.

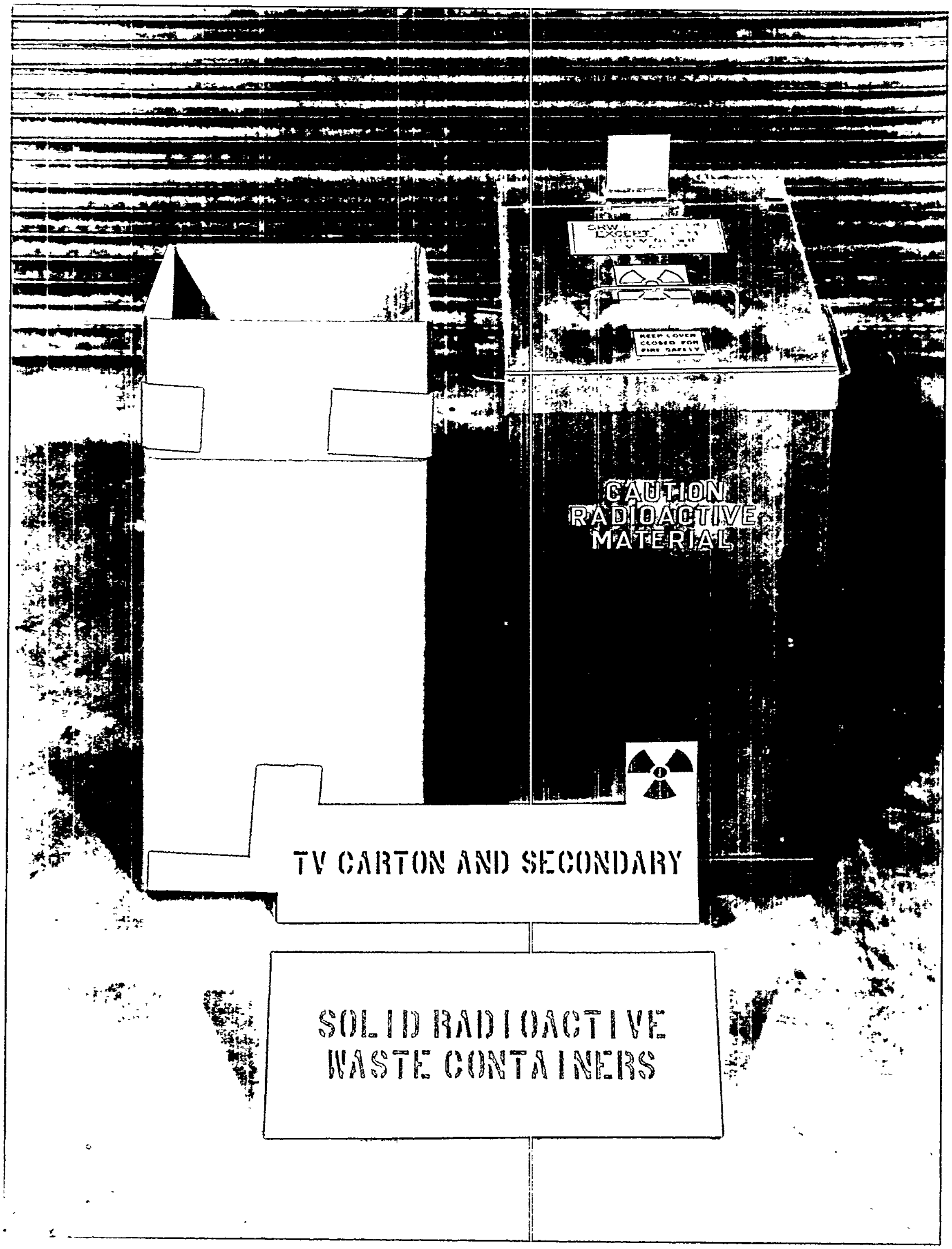

Figure 16.9b TV carton secondary container.

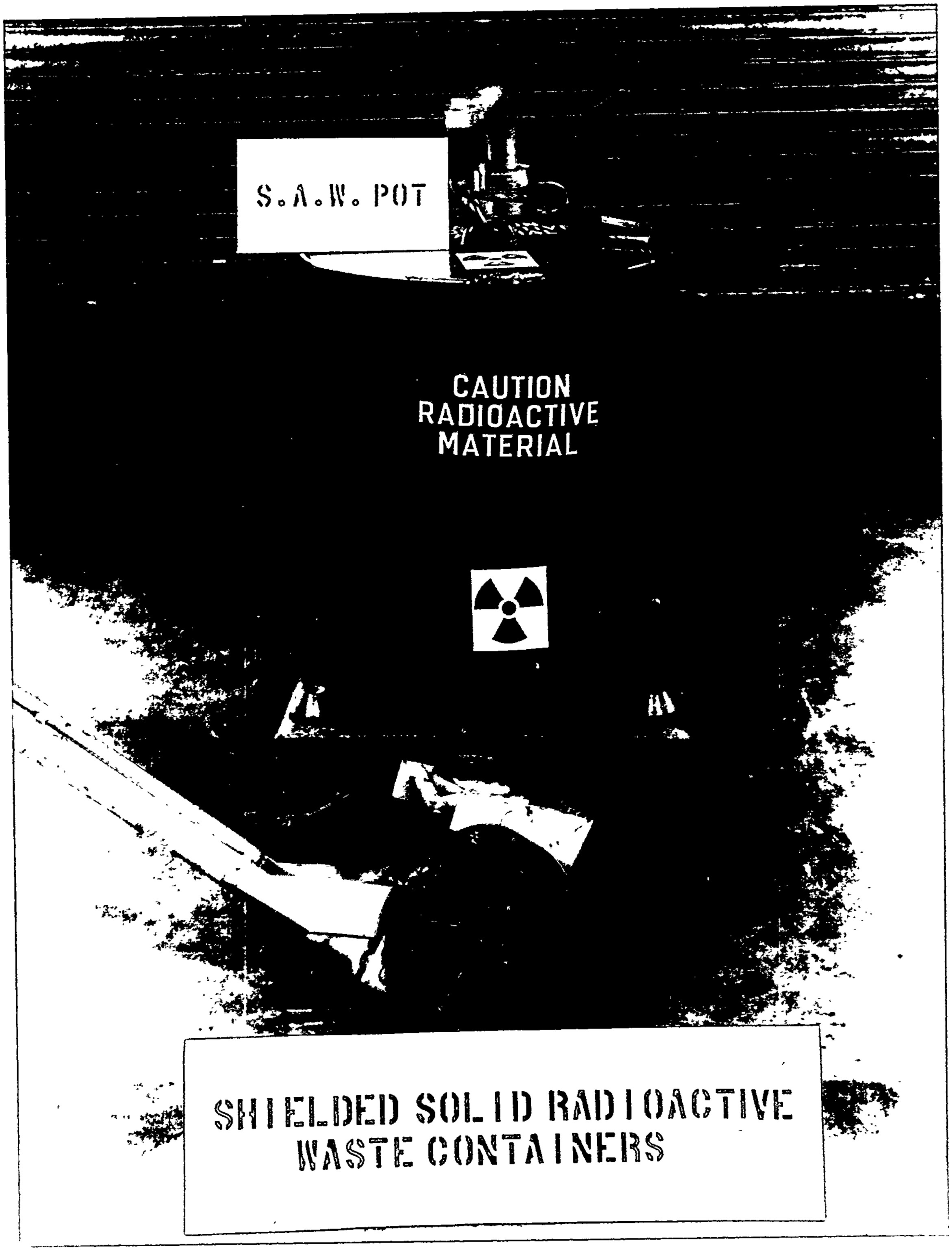

Figure 16.10 Shielded SRW container.

and/or the formation of liquids, preservatives such as slaked lime and dessicants or absorbents should be used and the wastes sealed in plastic bags prior to being placed in the container.

Bagged waste, such as that from gloveboxes or hot cells, is often TRU waste. These are generally handled in metal containers, either the 18.9 l (5-gallon) pails or paint cans, which can be sealed, or 208 l (55-gallon) drums which are sealed at the time of pickup (see Figure 16.11). Not shown in the figure are 114 l (40-gallon) drums which are also used. These wastes are generally highly radioactive and/or radiotoxic α wastes (Pu, Am, etc.), which require more care in their handling. All wastes reading > 0.2 mGy/h (200 mrad/h) are sealed in such metal pails or metal drums.

When containers are full, or when the reading on a container approaches the limit, health physics surveys the container for both the external radiation reading and for possible surface contamination (by smear survey). If survey results are within the limits, the waste container can be picked up. Pickup consists of removing the fiber drum insert, sealing the lid of the drum and removing it from the area to a waste storage area. In the case of pails or metal drums, the metal container cover is sealed when the waste is picked up.

Several treatment methods are available for solid radioactive waste disposal. The most widely used method is burial. This may be at a commercial burial site, at a designated DOE contractor site or sometimes on the facility site. Fiber drums and other containers which are brought to the waste storage area are transferred to bins for shipment out to a DOE-owned burial site. Figure 16.12 shows a number of such bins loaded on a truck, prior to leaving for the burial site. These bins are low-level radioactive waste and do not require shielding of the bins.

For compressible waste, the material may first be put into a baler to reduce the volume (average reduction factor of about 4) before the material is put into the bin. Baling is not done if the material reads > 0.2 mGy/h (200 mrad/h), and/or contains α activity and other than normal uranium and/or contains biological material.

Incineration can also achieve a dramatic volume reduction and is often used for low level radioactive wastes. One must exercise care in

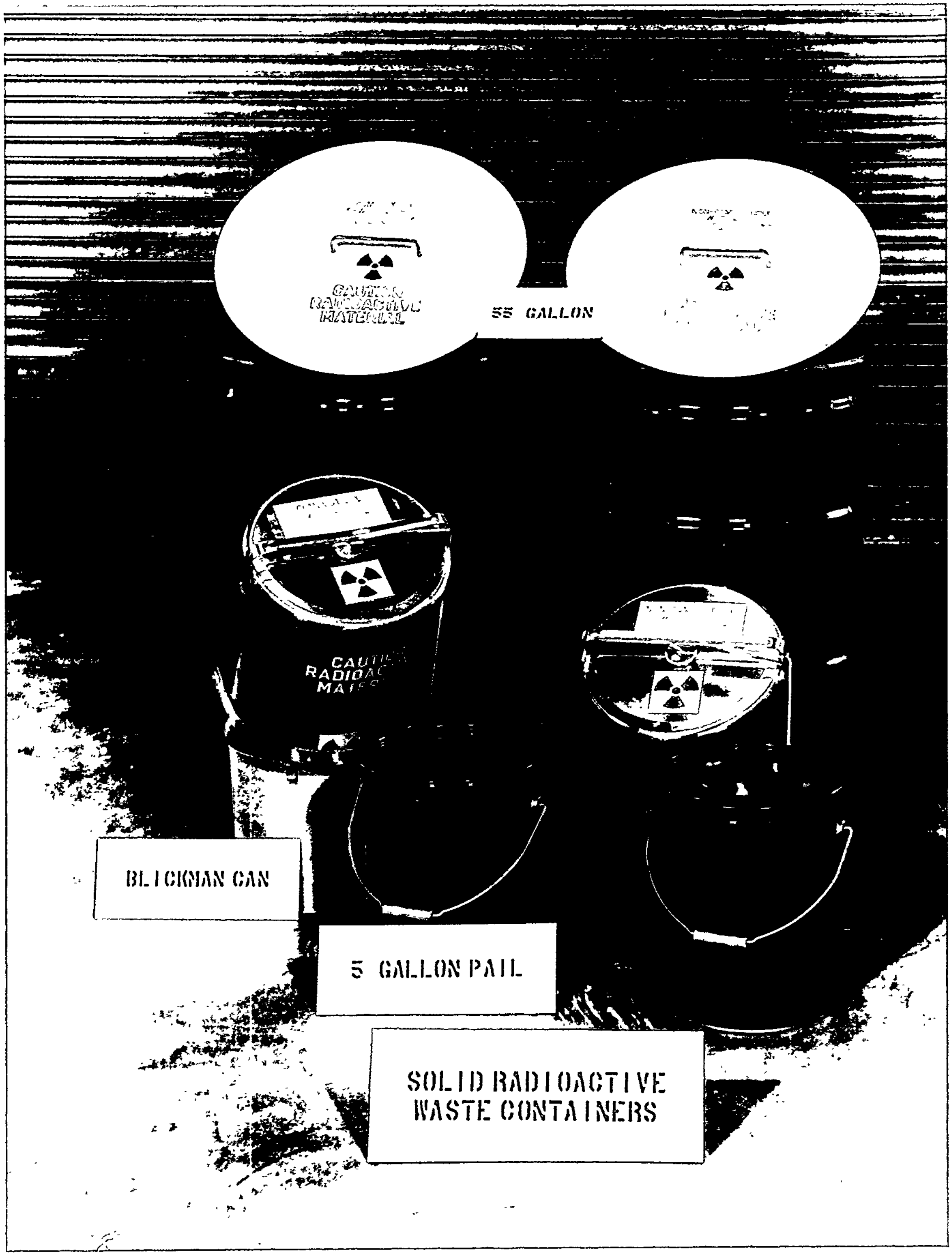

Figure 16.11 TRU solid radioactive waste container.

the use of incineration when volatile radionuclides (such as ^{125}I) may be involved. This could result in airborne ^{125}I being carried off during incineration. A futher problem may occur when incinerating vials of liquid scintillation samples. Sometimes, the combination of the plastic vial and the scintillation cocktail results in incomplete combustion.[69]

Finally, for some radionuclides, the half lives will allow storage of the waste until sufficient decay of the radionuclides occurs so that the waste can be considered trash. Then, such waste can be disposed of as trash in a landfill.

TRU wastes are put into bins similar to those shown in Figure 16.12, except that the bin is painted white to indicate TRU waste. These bins are not buried but are stored above ground. One further refinement for TRU waste is that the waste must be packaged in rigid containment. This may be accomplished for bins by using plywood liners. Non-TRU wastes are packaged in the regular bins and shipped out for burial.

4. Liquid Waste Disposal

The treatment of liquid wastes is generally more expensive than that of solid wastes. So, it is desirable to keep liquid waste to a minimum. Moreover, the treatment of liquid waste ultimately leads to residues which are converted to solid wastes since disposal sites are reluctant to accept waste in liquid form. The basic approach in control of liquid wastes is also segregation at the source, collection of concentrated wastes in containers, and treatment by various methods to remove radioactivity from the waste. The nature of the waste needs to be identified at the source. If the waste is acidic, it should not be mixed with alkaline wastes.[69] Precipitation may occur and the heat from the chemical reaction could result in the release of active aerosols. Non-aqueous wastes should be kept separate from aqueous solutions or this could cause problems later in the treatment of these wastes. Liquids containing organic compounds must not be mixed with those containing inorganic compounds.

Collection of the waste is generally by polyethylene carboys. These are usually protected by a steel secondary container to provide a

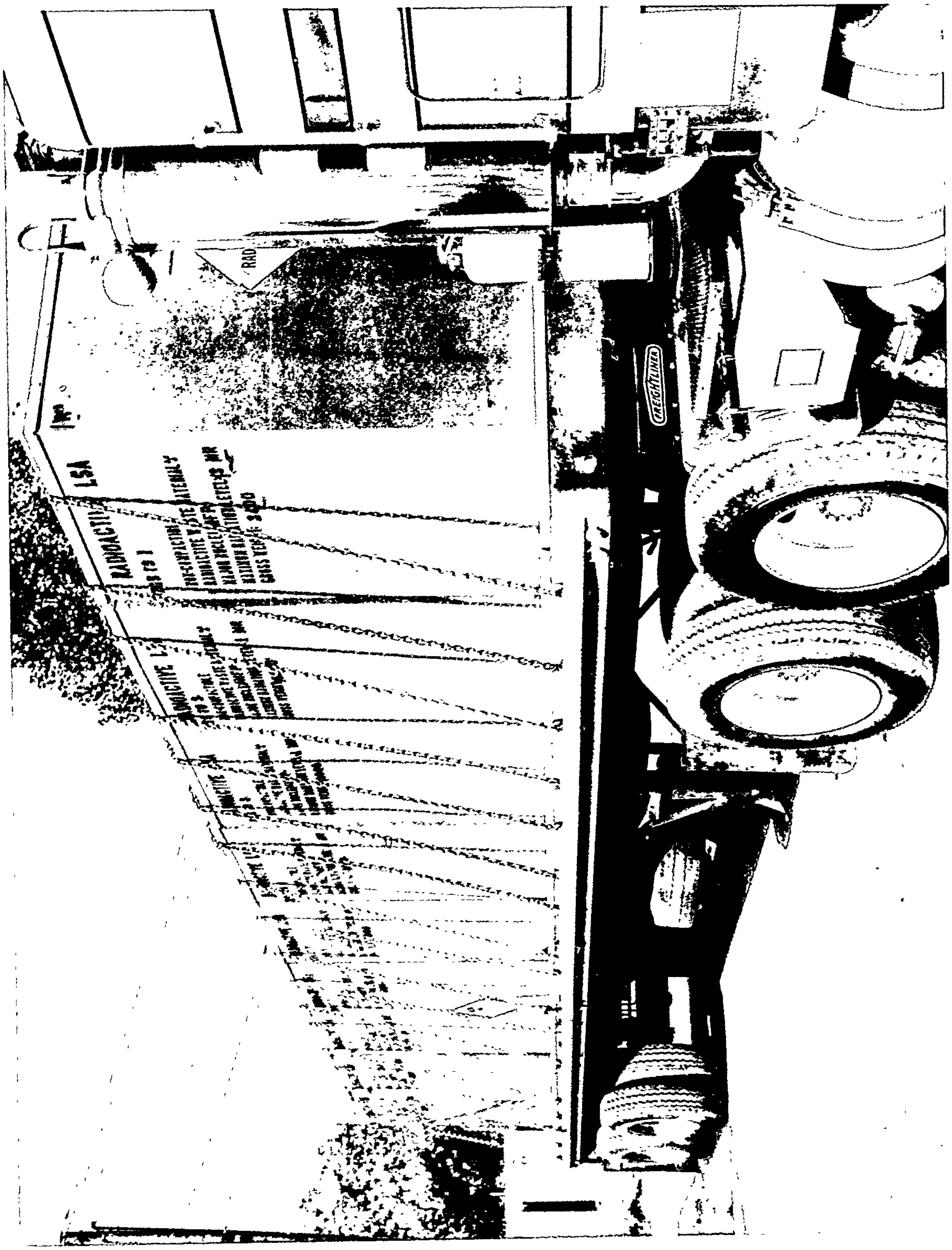

Figure 16.12 Shipment of Low Specific Activity (LSA) bins.

second line of defense in case of leakage. Because of the possible breakage factor, glass is often avoided. However, some non-aqueous solutions may chemically attack polyethylene, so these should not be used for non-aqueous solutions. Liquid waste containers should be kept closed to prevent evaporation.

The experimenter is supplied with these radioactive liquid waste containers (see Figure 16.13). These can be unshielded with a stainless steel secondary as shown in the figure. Or, the containers can be supplied with a shielded secondary. The user is expected to supply information with respect to the content of the liquid wastes. In particular, the identity and quantity of the radionuclides and volume of each entry, are to be recorded, as well as any other pertinent information regarding the potential chemical activity or hazards of the solution. When the containers are full, they are surveyed by health physics to determine radiation level and potential surface contamination. The containers are then collected and processed by various means (discussed later) and the effluent discarded via a waste treatment plant.

A second system which is used consists of retention tanks (see Figure 16.14), which are connected to the sink drains in all areas in which radioactivity is used. These are 5.7 m^3 (1500 gallon) capacity glass-lined tanks, which are generally operated in pairs. No radioactivity is permitted in laboratory drains, so this system provides a positive backup control to guard against mistakes. When one tank fills up, flow is diverted to the other tank. A sample from the full tank is obtained and measured for radioactivity. If the radioactive content is less than the allowable discharge level, the tank is directly discharged to the waste treatment plant. If above the permitted level, the contents of the tank are pumped out into a tank truck and taken to be processed as liquid radioactive waste. The effluent is then discarded via the waste treatment plant. This consists of essentially four 265 m^3 (70,000 gallon) hold-up tanks, where the pH is adjusted and final discharge control is maintained. That is, a radioactivity check is made before final discharge to ensure that the radioactivity level is within the limits for discharge.

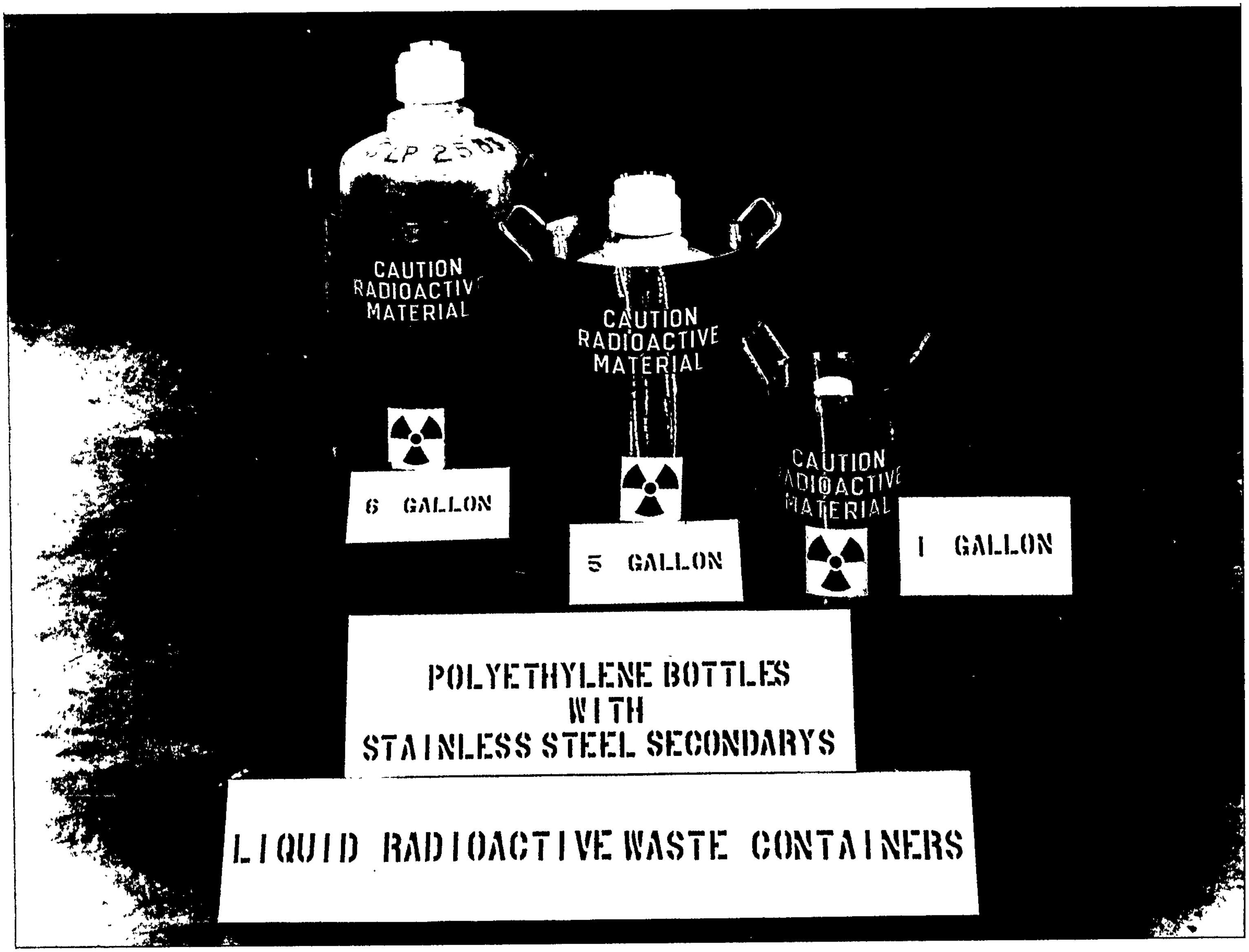

Figure 16.13 Liquid radioactive waste (LRW) containers.

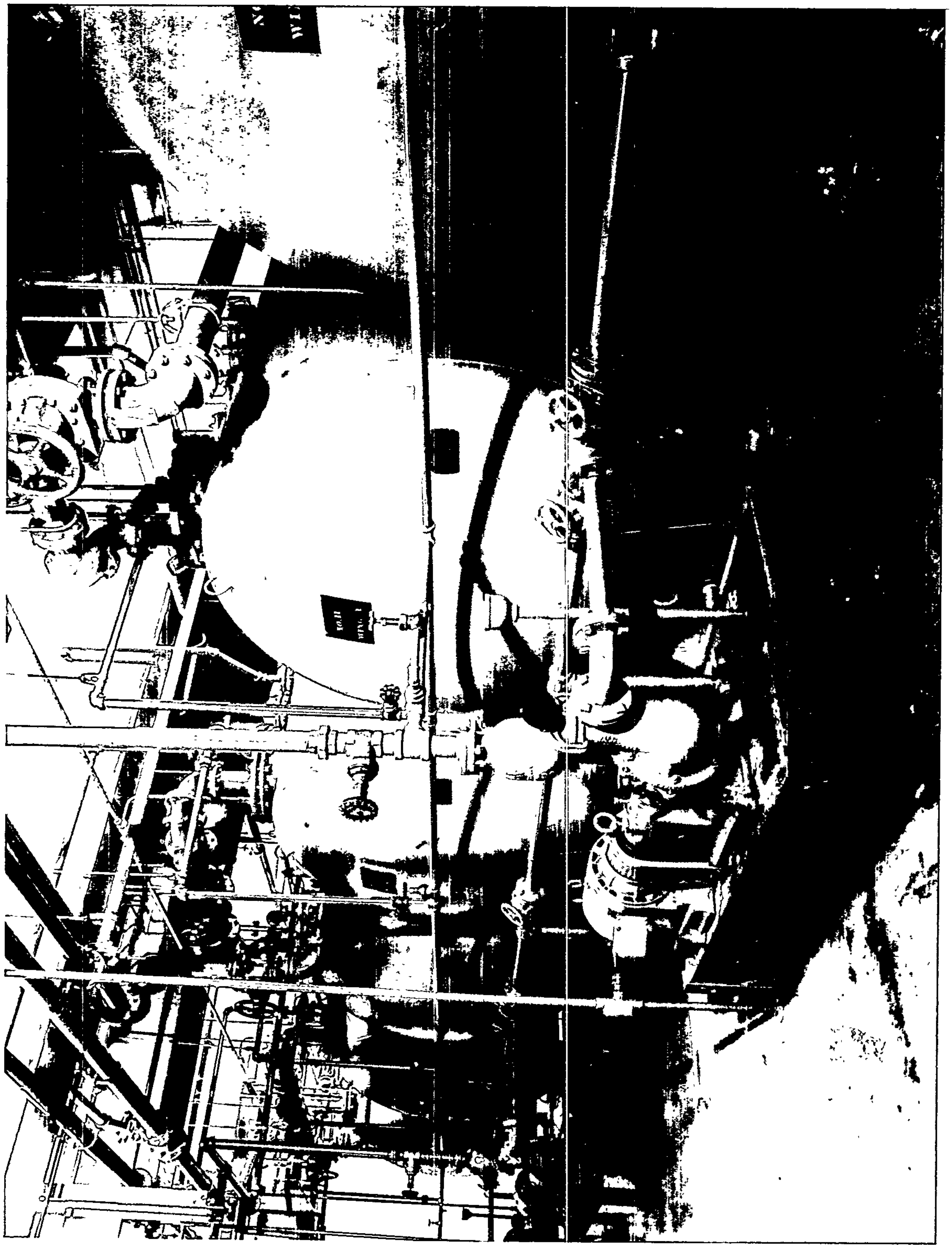

Figure 16.14 Typical retention tank system.

The treatment of liquid radioactive wastes may be by chemical methods, ion exchange, evaporation, incineration in the case of organic solvents (scintillation counting samples), and storage and decay (applicable to short half life radionuclides). Chemical methods are suitable for the case in which the required reduction in activity is not too large (approximately 90% removal can be achieved). Chemical methods include coagulation and precipitation. In each of these processes, the residue or sludge which is formed is converted to solid waste, generally by solidification in concrete or cement. In the ion exchange process, the liquid is passed through a resin which concentrates the contaminant in the bed or column.[73] The resin can then be treated as solid active waste. Evaporation is useful when the liquid has a low content of dissolved solids. In the evaporation process, the liquid is boiled off and the resultant concentrate, which will contain almost the entire radioactive content can then be converted to a solid waste by absorption in a porous material, such as vermiculite. Incineration is mainly of use in the case of vials containing organic solvents, such as are produced during medical tests (radio-immunoassay) or during other liquid scintillation applications. The successful use of the incineration method requires that careful design of the incinerator be undertaken.[69]

A number of the above treatment methods are used in order to obtain the most economical means of treatment of the liquid radioactive waste. However, if incineration is not available, scintillation liquids are normally packaged and stored until shipped to a hazardous waste disposal facility to be incinerated. Many scintillation cocktail fluids are classified as hazardous waste as defined by 40 CFR 261,[74] Subparts C and D. These fluids must not be absorbed in vermiculite. When liquid immobilization by absorbents, such as vermiculite, is used, at least twice as much absorbent as is actually needed for complete absorption is added.

5. Gaseous Waste Disposal

Gaseous wastes include radioactive gases, vapors or fumes produced during an experiment, the leakage of a particular radioactive gas being used in an operation, fission product gases or particulates which are carried by air from a reactor, activation products produced in an accelerator which are entrained in the exhaust, and particulates arising from surface abrasion in machining. In mines, the release of radon occurs during the operations involved in mining uranium ore. In the latter case, control can usually be obtained by proper design of ventilation to the mine to reduce the radon concentrations. The exhausted radon gas is vented to the area outside of the mine and is quickly dispersed so that it adds little to the concentration in that area. Moreover, the short half life of the radon daughter chain assures that most of these products have completely decayed within a few hours.

Control of airborne particulates is often accomplished either by providing local exhaust ventilation or by individually hooding or enclosing the operation. Local exhaust ventilation is accomplished by the use of flexible exhaust tubing (called elephant trunks) in the immediate vicinity of the operation to catch the particulates as they are formed and direct the particles into the exhaust system. The exhaust stream may then be passed through a filtering system before discharge. Local exhaust ventilation is used when the activity level and the radiotoxicity of the material is low. When large amounts of radioactivity, and/or highly hazardous materials are involved, the process is usually performed in an enclosure, which will then be followed by tandem high efficiency filtration (HEPA filters) of the exhaust prior to discharge. One should note, that when the filters are changed, the replaced filter, itself, becomes solid radioactive waste because of the trapped radioactive particulates.

For radioactive gases, special systems need to be supplied to remove these before discharge. Activated charcoal filters have proven successful for removal of radioiodines. Silica gel and other adsorbing materials have been used for tritium water vapor removal. For some processes, one may need to use caustic scrubbers or adsorbers to remove chemical fumes and gases.

In some applications, such as in reactors, it is possible to design a holdup system which delays the release of effluents. For those radioactive gases with short half lives, this approach results in a significant reduction in the radioactive content. Following holdup, the effluent is then passed through filters to trap particulates before being exhausted through tall stacks. The tall stacks utilize the dispersion characteristics of the atmosphere to achieve the desired concentration reduction at the site boundary.

For many applications, the required dilution and dispersion may be achieved by controlled release of the radioactive gas. That is, a quantity of radioactive gas is slowly released over a period of time such that the concentration at the site boundary or the point of potential exposure of the population is well within allowable limits. This method takes advantage of the initial dilution in the exhaust volume and the dispersion in the atmosphere.

K. Transportation of Radionuclides

In addition to shipment of radioactive wastes by certain commercial carriers, a large number of shipments involving radioactive materials for various uses also takes place. A major portion of these shipments involve small or intermediate quantities of radionuclides in relatively small packages. Many of these radionuclides are used for medical applications-either diagnosis or therapy. However, others are used by schools, laboratories and industry. Many of these radionuclides are shipped by rapid delivery services, air freight or air express.

Prior to 1966, the Interstate Commerce Commission was charged with regulating the transportation of hazardous materials between the states. In 1966, The Department of Transportation (DOT) was created and given the regulatory responsibility for safety in the transportation of all hazardous materials. This responsibility includes shipments by all modes of transport, except for postal shipments. Since that time, the regulations governing packaging and shipment of radioactive materials from the safety standpoint can be found in 49 CFR Parts 100-199.[75] Amendments appear in the Federal Register as they are implemented. The Department of Energy requirements for packaging and shipment of fissile and other radioactive materials are contained in DOE Order 5480.1, Chapter III.[76] In addition, the U.S. Nuclear Regulatory Commission, which has responsibility for safety in the possession, use and transport of byproduct, source, and special nuclear material, have requirements for their licensees regarding transportation of radionuclides in 10 CFR 71.[77] Table 16.3 which has been taken from Reference 78, summarizes the federal regulations with respect to the transportation of radioactive materials.

TABLE 16.3

SOURCES OF FEDERAL REGULATIONS

U.S. Department of Transportation's Hazardous Materials Regulations Title 49, Parts 100-177 and 178-199	
49 CFR 106	Rulemaking Procedures
49 CFR 107	Hazardous Materials Program Procedures
49 CFR 171	General Information, Regulations and Definitions
49 CFR 172	Hazardous Materials Tables and Hazardous Materials Communication Regulations
49 CFR 173	Shippers - General Requirements for Shipments and Packagings
49 CFR 174	Carriage by Rail
49 CFR 175	Carriage by Aircraft
49 CFR 176	Carriage by Vessel
49 CFR 177	Carriage by Public Highway
49 CFR 178	Shipping Container Specifications
49 CFR 179	Specification for Tank Cars
Title 10	
U.S. Nuclear Regulatory Commission	
10 CFR 71	Packaging of Radioactive Materials for Transport and Transportation of Radioactive Materials Under Certain Conditions
Title 39	
U.S. Postal Service	
Domestic Mail Manual, U.S. Postal Service Regulations, Part 124, Postal Regulations for Transport of Radioactive Matter are published in U.S. Postal Service Publication #6, and in the U.S. Postal Manual	

Much of the material in the rest of this discussion has been adapted from Reference 78. Table 16.4 lists pertinent sources for international regulations. The most recent DOT regulations, as reviewed in Reference 78, are very similar to the IAEA regulations referred to in Table 16.4.

TABLE 16.4

AVAILABILITY OF INTERNATIONAL REGULATIONS

1. "Regulations for the Safe Transportation of Radioactive Material," as amended, Safety Series #6, 1985 Revised Edition - International Atomic Energy Agency (IAEA), Vienna, Austria. Available from UNIPUB, 1180 Avenue of the Americas, New York, New York 10038.
2. International Civil Aviation Organization (ICAO), Technical Instructions for the Safe Transport of Dangerous Goods by Air, 1983 Edition. Available from INTEREG, P.O. Box 60105, Chicago, Illinois, 60660.
3. International Maritime Organization (IMO), formerly Intergovernmental Maritime Consultative Organization (IMCO). International Maritime Dangerous Goods (IMDG) Code.
4. International Air Transport Association (IATA), Restricted Articles Regulations, 25th Edition, plus Supplement and Amendment issued March 1, 1981, effective December 1982. International Air Transport Association, 2000 Peel Street, Montreal, Quebec, Canada H3A 2R4.

Many of the NRC Agreement states, as well as other states, have required shippers to conform to DOT regulations with respect to intrastate shipments.

For purposes of the federal regulations, radioactive materials are defined as those materials which spontaneously emit ionizing radiation and have a specific activity in excess of 7.4×10^4 Bq/kg (2 nCi/g) of material. Materials with radioactive concentrations below this are not regulated by either DOT or IAEA with respect to transport of radioactive materials.

1. Packaging Requirements

Three factors which influence the packaging requirements are the radionuclide(s) involved, the quantity of the radionuclide, and the form of the radionuclide material. The term special form is used to describe radioactive materials which if released, may present an external radiation hazard, but are sufficiently contained so that no loose radioactivity is expected to be released. This implies some characteristic such that loose radioactivity will not be released. For example, a radioactive source in the form of a metal bar, or an encapsulated source such that radioactivity will only be released by destruction of the capsule. Normal form refers to any radioactive materials which do not meet the qualification of special form. Generally, the regulations allow substantially larger quantities of special form materials to be transported. Depending upon the relative hazard of the radionuclide, the precautions to be taken will vary. In 49 CFR 173.435, information on more than 250 radionuclides can be found. In addition, procedures for dealing with unlisted or unknown, or mixtures of radionuclides can be found in Section 173.433. The quantity of the radionuclide allowed to be transported under certain conditions is also related to the relative hazard that the material represents and is related to the potential internal hazard which could result from a release.

a. Type A Packaging

In the present scheme of packaging requirements, every radionuclide is assigned a limit for the total amount of radioactivity of that radionuclide which can be transported in a given type of package. So, there is a quantity listed for the material when it is in normal form, as well as, when it is in special form. These quantities (shown in Table 16.5 for selected radionuclides) are the limits for what is referred to as Type A packaging. In the table, the A_1 value refers to the radionuclide in special form and the A_2 value is that for normal form. If a given quantity of a radionuclide in a specific form exceeds the limit listed for that form, then Type B packaging is required.

For a mixture of radionuclides, one may usually use the ratio rule:

$$\left(\frac{\text{Activity}}{A_i}\right)_1 + \left(\frac{\text{Activity}}{A_i}\right)_2 + \ldots \leq 1, \qquad 16.3$$

in which A_i represents the appropriate limit for the particular form (special or normal) for radionuclide 1, and the Activity_1 is the activity of that component in the mixture.

TABLE 16.5

TYPE A PACKAGE QUANTITY LIMITS FOR SELECTED RADIONUCLIDES (ADDITIONAL RADIONUCLIDES ARE LISTED IN 49 CFR 173.435)

Symbol of Radionuclide	Element and Atomic Number	A_1(Bq)* (Special Form)	A_2(Bq)* (Normal Form)
^{14}C	Carbon (6)	3.7×10^{13}	2.22×10^{12}
^{137}Cs	Cesium (55)	1.12×10^{12}	3.7×10^{11}
^{99}Mo	Molybdenum (42)	3.7×10^{12}	7×10^{11}
^{235}U	Uranium (92)	3.7×10^{12}	7.4×10^{9}
^{226}Ra	Radium (88)	3.7×10^{11}	1.85×10^{9}
^{201}Pb	Lead (82)	7.4×10^{11}	7.4×10^{11}

*Divide Bq by 3.7×10^{10} to obtain Ci,

NOTE 1: Quantities exceeding Type A package limits require Type B packaging.

To qualify as a Type A package, the package must be adequate to prevent loss or dispersal of the radioactive contents, and to maintain its radiation shielding ability, if the package is subjected to normal conditions during transport. Typical Type A packages are shown in Figure 16.15, and these conform to the DOT Specification 7A (49 CFR 178.350). Each shipper must make his own assessment and certification of the particular package design with respect to the performance criteria and be prepared to provide a complete certification and documented safety analysis to show the requirements have been met.

Low specific activity (LSA) materials present a reduced hazard because of limited radioactivity. Some LSA materials are designated by name, such as uranium ores and concentrates or unirradiated natural or depleted uranium. For other radionuclides, concentration limits apply-for ^3H_2O, the limit is 1.85×10^{14} Bq/m^3 (5 mCi/ml). The allowable concentration values for other radionuclides are related to their normal form values (A_2). If the activity is uniformly dispersed, the LSA concentration limits are:

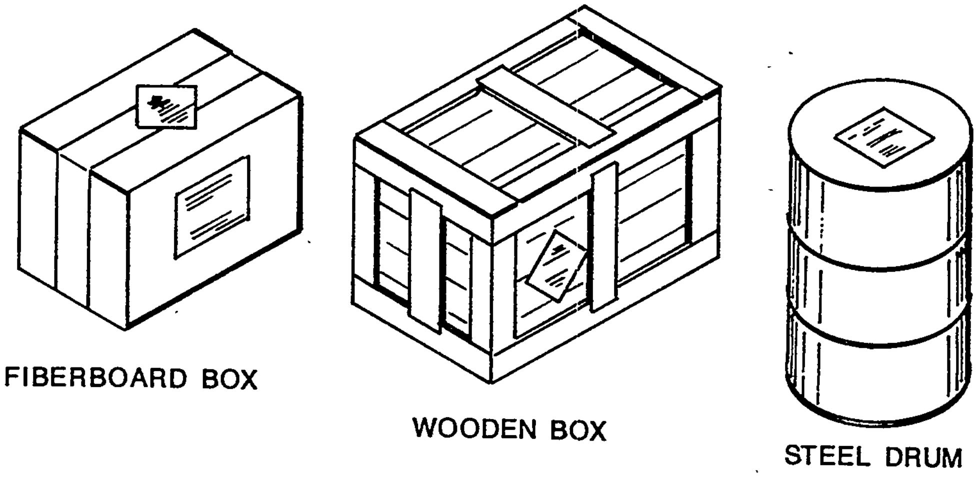

Figure 16.15 Examples of Type A packaging.

$A_2 \leq 1.85\text{x}10^9$ Bq (.05 Ci)
Activity/kg = $3.7\text{x}10^6$ Bq/kg (0.1 μCi/g)

$1.85\text{x}10^9 \leq A_2 \leq 3.7\text{x}10^{10}$ Bq (1 Ci)
Activity/kg = $1.85\text{x}10^8$ Bq/kg (5 μCi/g)

$A_2 \geq 3.7\text{x}10^{10}$ Bq (1 Ci)
Activity/kg = $1.11\text{x}10^{10}$ Bq/kg (300 μCi/g)

When mixtures are present, one may use the ratio rule, similar to that in equation 16.3. In this case, one adds up all the material in one category of A_2 values and divides by the concentration limit for that category, and proceeds in this manner for each A_2 category. If the summation of the ratios for each of the three categories is less than one, the material can be considered LSA. Most radioactive shipments will end up in the LSA category. These shipments usually are sent out in essentially Type A packaging. These shipments are referred to as "non-exclusive use" transportation, and must conform to the LSA limits given above. Exclusive use means that the consignor has exclusive use of the transport conveyance. No loading or unloading can be carried out unless under the direction of the consignor or the consignee. If the transporter uses an "exclusive use" vehicle, the packaging restrictions are relaxed and need not meet Type A specifications. However, the material must be in a strong, tight package and the vehicle must be a closed transport vehicle. In addition, exclusive use LSA shipments <u>must</u> have packages marked "Radioactive LSA" (see Figure 16.12) and the vehicle <u>must</u> be placarded "Radioactive Material."

b. <u>Type B Packaging</u>

Type B packaging must meet all the requirements for a Type A package and must withstand certain serious accident damage tests. The package must show no loss of containment and only limited loss of shielding. Typical Type B packages are shown in Figure 16.16. As indicated in the figure, these containers must pass the accident tests in 10 CFR 71.[77] These include the following:

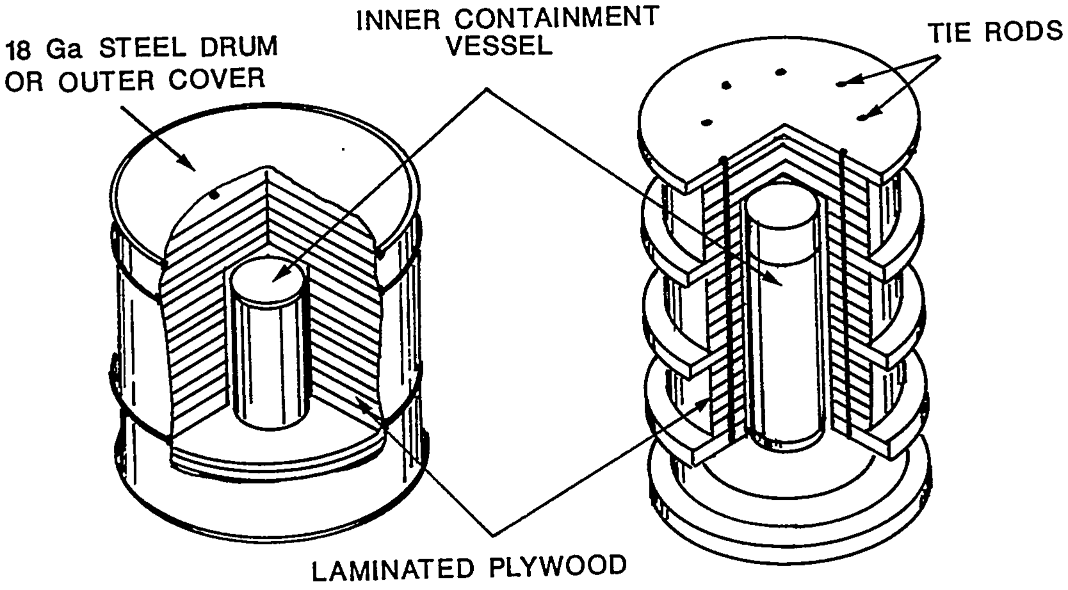

Figure 16.16 Typical Type B packagings.

(1) A 9.1 m (30 ft) free drop onto an unyielding surface.

(2) A puncture test which is a free drop, over 1 m (approximately 40 inches) onto a .15 m (6 in.) diameter steel pin.

(3) Thermal exposure at 1475°F for 30 min.

(4) Water immersion for 8 hours (for fissile material packaging only).

Except for a limited number of Type B packages described in the regulations (i.e. DOT-6M), all Type B packaging designs require prior approval of the USDOE or NRC before use.

c. Fissile Material Packaging

Over and above radionuclide content, the shipping of fissile material requires certain packaging and shipment procedures to ensure against a criticality incident. Specific requirements are discussed in 49 CFR 173.451 through 173.459 of the DOT regulations and in 10 CFR 71 of the USNRC regulations. The packaging must ensure against nuclear criticality under both normal and hypothetical accident test conditions, and must also prevent loss of contents during transport. Fissile materials are classed into one of three groups (see Table 16.6), according to the degree of control needed to assure nuclear criticality safety.

TABLE 16.6

SHIPMENT CONTROLS FOR FISSILE RADIOACTIVE MATERIALS
49 CFR - SECTION 173.455

1. Fissile Class I - Packages may be transported in unlimited numbers (Transport Index is based only on external radiation levels).

2. Fissile Class II - Number of packages limited by aggregate maximum of transport indexes of 50, (50 unit rule). No single package may exceed a transport index of 10. Transport index shall be based on criticality or external radiation level basis, whichever is most restrictive.

3. Fissile Class III - Shipments of packages which do not meet the requirements of Fissile Class I or II. Controlled by specific arrangements between the shipper and carrier. (See Section 173.457(b).)

2. Radiation Limits

The regulations prescribe radiation limits for the external dose rate for radioactive material packages during transport. These limits are shown in Table 16.7, adapted from 49 CFR 173. The term transport index, which appears in the table, is defined in terms of the maximum dose rate at 1 meter from any accessible exterior surface of the package. It is expressed in units of 10 μSv (1 mrem/h). So, a maximum reading of 100 μSv/h (10 mrem/h) at a meter away from the exterior surface of the package has a transport index (TI)=10. As indicated in Table 16.7, no single package should have a transport index greater than 10, unless it is being shipped by an "exclusive use" vehicle. The TI per package limit is also reduced to 3 when the package is shipped aboard passenger-carrying aircraft.

Table 16.7

RADIOACTIVE MATERIALS PACKAGES MAXIMUM RADIATION LEVEL LIMITATIONS
(49 CFR 173.441(a) AND (b))

RADIATION DOSE RATE AT ANY POINT ON EXTERNAL SURFACE OF ANY PACKAGE OF RADIOACTIVE MATERIAL MAY NOT EXCEED:

A. 2 mSv/h (200 mrem/h).
B. 0.1 mSv/h (10 mrem/h) AT ONE METER (TRANSPORT INDEX MAY NOT EXCEED 10).

UNLESS THE PACKAGES ARE TRANSPORTED IN AN "EXCLUSIVE USE" CLOSED TRANSPORT VEHICLE (AIRCRAFT PROHIBITED) - THEN THE MAXIMUM RADIATION DOSE RATES MAY BE:

A. 10 mSv/h (1 rem/h) ON THE ACCESSIBLE EXTERNAL PACKAGE SURFACE.
B. 2 mSv/h (200 mrem/h) AT EXTERNAL SURFACE OF THE VEHICLE.
C. 0.1 mSv/h (10 mrem/h) AT TWO METERS FROM EXTERNAL SURFACE OF THE VEHICLE.
D. 20 μSv/h (2 mrem/h) IN ANY POSITION OF THE VEHICLE WHICH IS OCCUPIED BY A PERSON.

In general, the total value of the TI in a shipment is limited to 50, except for "exclusive use" vehicles and cases in which special arrangements have been made between the shipper and the carrier, satisfying requirements of 49 CFR 173.403(i) and 173.441(b).

The transport index is also used to limit the amount of fissile material in one location under non-exclusive use conditions. In this case, the shipper determines an appropriate TI in accordance with the criteria in 49 CFR 173.455(b). Note that this value of the transport index is not based on the external radiation level but on considerations of nuclear safety. The package may also have an external radiation reading. The TI of the package is then taken as the higher of the two TI values.

In addition to the external radiation level limit, there are also prescribed limits for the level of removable contamination on the package. These limits are shown in Table 16.8 and refer to wipes or smears taken on the package surface. These limits of contamination also apply to the release of the transport vehicle following either "exclusive use" transport or a bulk shipment of LSA material. Health physics surveys and smear samples are provided in order to determine compliance with these limits.

TABLE 16.8

REMOVABLE EXTERNAL RADIOACTIVE CONTAMINATION-WIPE LIMITS

Contaminant	Maximum Permissible Limits	
	Bq/m^2	(dis/min-cm^2)
Beta/gamma-emitting radionuclides; all radionuclides with half-lives less than ten days; natural uranium; natural thorium; uranium-235; uranium-238; thorium-232; thorium-228 and thorium-230 when contained in ores or physical concentrates	3.7×10^3	22
All other alpha-emitting radionuclides	3.7×10^2	2.2

3. Warning Labels

Each package of radioactive material, unless excepted, must be labeled on two opposite sides, with a distinct warning label. Figure 16.7 shows the three classes of labels used for radioactive materials.

The labels are used to alert personnel that the package contains radioactive materials and special handling may be required. A label with an all white background (White I in Figure 16.17), indicates a low external radiation level and the package requires no special handling. If the upper half of the label is yellow (Yellow II in Figure 16.17), the package may have an external radiation level or fissile properties which must be considered during transport. If the package has a yellow label with three stripes, the transport vehicle must be placarded ("Radioactive").

On all the labels, the vertical bars are red, Each label is diamond-shaped, 0.1 m on each side, and has a black solid line border. The following information must be entered on the applicable blank spaces of the label: the name of the radionuclide (the most restrictive radionuclide if a mixture), the activity of the radionuclide and the transport index.


RADIOACTIVE–WHITE I **RADIOACTIVE–YELLOW II** **RADIOACTIVE–YELLOW III**

Figure 16.17 Radioactive package labels.

The determination of the proper label to use is based upon criteria which are contained in 49 CFR 172.403. These labeling criteria are listed in Table 16.9 for radioactive materials packages and for fissile material packages. Note that the labeling criteria requires the radioactive label for fissile materials, and the appropriate label differs for the fissile class.

4. Limited Quantities, Instruments and Articles

Some packages are excepted from some of the requirements of Type A packaging if they contain only limited quantities of radioactivity. These limits may be applied to materials, instruments and articles. The basis for most of these limited quantity values are the A_1 and A_2 values for the specific radionuclide. The activity limits are shown in Table 16.10, and A_1 or A_2 values for a given radionuclide can be found in 49 CFR 173.435.

TABLE 16.9

RADIOACTIVE MATERIALS PACKAGES LABELING CRITERIA
49 CFR-SECTION 172.403

Transport Index (T.I.)	Radiation Level at Package Surface (RL)	Fissile Criteria	Label Category
N/A	RL≤ 5μSv/h (.5 mrem/h	Fissile Class I Only No Fissile Class II or III	White - I
T.I.≤1.0	5 μSv/h ≤RL≤500μSv/h	Fissile Class I Fissile ClassII with T.I.≤1.0, No Fissile Class III	Yellow - II
1.0<T.I.	500 μSv/h (50 mrem/h)≤RL	Fissile Class II with 1.0<T.I., Fissile Class III	Yellow - III

TABLE 16.10

ACTIVITY LIMITS FOR LIMITED QUANTITIES, INSTRUMENTS AND ARTICLES

Nature of contents[1]	Instruments and Articles		Materials
	Instrument and article Limits[1]	Package Limits	Package Limits
Solids			
Special form	10^{-2} A_1	A_1	10^{-3} A_1
Other forms	10^{-2} A_2	A_2	10^{-3} A_2
Liquids			
Tritiated Water			
$<3.7\text{x}10^{12}$ Bq/m^3	-	-	$3.7\text{x}10^{13}$ Bq
$3.7\text{x}10^{12}$ to $3.7\text{x}10^{13}$ Bq/m^3	-	-	$3.7\text{x}10^{12}$ Bq
$>3.7\text{x}10^{13}$ Bq/m^3	-	-	$3.7\text{x}10^{10}$ Bq
Other liquids	10^{-3} A_2	10^{-1} A_2	10^{-4} A_2
Gases			
Tritium[2]	$7.4\text{x}10^{11}$ Bq	$7.4\text{x}10^{12}$ Bq	$7.4\text{x}10^{11}$ Bq
Special form	10^{-3} A_1	10^{-2} A_1	10^{-3} A_1
Other forms	10^{-3} A_2	10^{-2} A_2	10^{-3} A_2

[1] For mixture of radionuclides see Section 173.433(b).

[2] These values also apply to tritium in activated luminous paint and tritium adsorbed on solid carriers.

The packaging exceptions include not having to provide DOT specification packaging, shipping papers, certification, marking or labeling. Conditions which must be met include:

(1) Activity limits per package and, if appropriate, per instrument or article;

(2) The materials must be packaged in strong, tight packages that will not leak ANY of the radioactive material during conditions normally incident to transportation;

(3) The radiation level at any point on the external surface of the package cannot exceed 5 μSv/h (0.5 mrem/h);

(4) The external surface of the package must be free of significant removable contamination;

(5) For instruments and articles, the radiation level at 0.1 m (4 inches) from any point on the surface of the unpackaged instrument or article may not exceed 100 μSv/h (10 mrem/h); and

(6) A prescribed description of the contents on a document which is in, or on, the package, or forwarded with it.

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QUESTIONS

16.1 Explain "containment and concentration."

16.2 When should an inert gas be used in a glovebox?

16.3 What factor(s) determine that ordinary gloveboxes may not be adequate for radionuclide work?

16.4 What device may be used instead of a hot cell when the principal hazard is alpha emission?

16.5 What kind of airflow pattern should be designed for glovebox work?

16.6 What are the primary goals of a safety design system?

16.7 What are the four main features to be considered for plutonium- like radionuclides, in devising enclosure safety systems?

16.8 Why is the layout of a facility an important factor in contamination control?

16.9 In addition to design and administrative aspects of control, what actually determines the necessity of additional containment? Why?

16.10 What purpose do fixed monitors perform?

16.11 Name some of the fixed monitors and their specific purposes.

16.12 What is the advantage of continuous monitoring?

16.13 What is the common method used to appraise internal exposure?

16.14 What radionuclide ratio must be known to assess the lung burden of plutonium-239 when using whole body counters?

16.15 What substance is found effective in increasing the urinary elimination of metals? At what time is it most effective?

16.16 Under what circumstances does plutonium exhibit high external radiation fields? What practices may be used to limit external exposure?

16.17 What are the goals that must receive attention in an emergency situation?

16.18 What functions must be carried out in the event of a major radiation incident?

16.19 What factors determine the needed thickness of a particular shielding material?

16.20 What purpose do liners serve in hot cells?

16.21 Why, in determining the thickness of a hot cell wall, should the most intense source that will be used be considered a point source?

16.22 What are some of the individual factors that determine the buildup factor, b?

16.23 How should the walls of a hot cell be checked for leakage?

16.24 What principal purposes are served by ventilating systems?

16.25 Why are filters on air inlets to hot cells advisable?

16.26 List some problems that occur in providing and using windows for viewing the inside of hot cells.

16.27 Compare the advantages of the ball-joint manipulator with the advantages of a master-slave manipulator.

16.28 In addition to routine external monitoring of hot cells, for what other events is hot cell monitoring of great importance?

16.29 What hazards may exist in changing air filters for hot cells?

16.30 When is the term contamination generally used?

16.31 List some of the surface decontamination agents frequently used.

16.32 What procedure is generally used for decontaminating skin and hands?

16.33 What compound may be used to remove fission products?

16.34 For what contaminant is a mixture of $KMnO_4$ and H_2SO_4 used to remove contamination?

16.35 What precautions are to be taken to remove large pieces of contaminated equipment from gloveboxes?

16.36 What precautions are deemed important when making an entry into a hot cell?

16.37 What are three basic forms of radioactive waste and how are they produced?

16.38 What are the four main activities that are basic sources of radioactive waste production?

16.39 Explain:
a) concentrate and contain
b) dilute and disperse.

16.40 What responsibility lies with the waste producer as far as documentation is concerned?

16.41 What is transuranic waste?

16.42 What is the reason to use preservatives for certain solid wastes?

16.43 What problems must be carefully considered when incinerating radioactive waste?

16.44 Why is the acidic liquid waste not mixed with alkaline wastes?

16.45 What purpose do retention tanks serve?

16.46 What special systems need to be supplied to remove radioactive gases?

16.47 What is the limiting specific activity to designate a substance as radioactive for transportation purposes?

16.48 What is "ratio rule"? How is it applied?

16.49 How does Type B packaging differ from Type A packaging?

16.50 Define transport index.

SECTION 17 - PARTICLE ACCELERATORS

The first artificial transmutation studies were performed by Lord Rutherford in England in 1919. He used a radium source to bombard various gases with alpha particles. The reaction that took place with nitrogen was

$$^{14}_{7}N + {}^{4}_{2}He \rightarrow {}^{18}_{9}F \rightarrow {}^{17}_{8}O + {}^{1}_{1}H.$$

The sum of the masses of the final products is greater than that of the initial products; therefore, energy supplied by the kinetic energy of the alphas was absorbed in this process. During the next decade, many such experiments were attempted with other radioactive substances as the source of energetic alphas.

It became clear during this time that many other reactions might be produced with higher energy alphas or other particles with the same energies. Over and above this, many theories were expressed because of interest in the binding energy of the nucleus. This required a tool with which to probe into the nucleus in order to prove or disprove the predictions. For these reasons and others, methods to produce high energy particles to carry on this work were studied.

A. Accelerators

The accelerators built shortly thereafter gave rise to studies that added much knowledge to the field of physics. They were of value in revealing many new and unique particles. With the advent of these machines, an entire new field in physics has evolved - that of high-energy physics. This field at present is quite complex; however, with the aid of new designs and new principles, more data are being collected which may, in time, lead to a more complete understanding of properties of the nucleus.

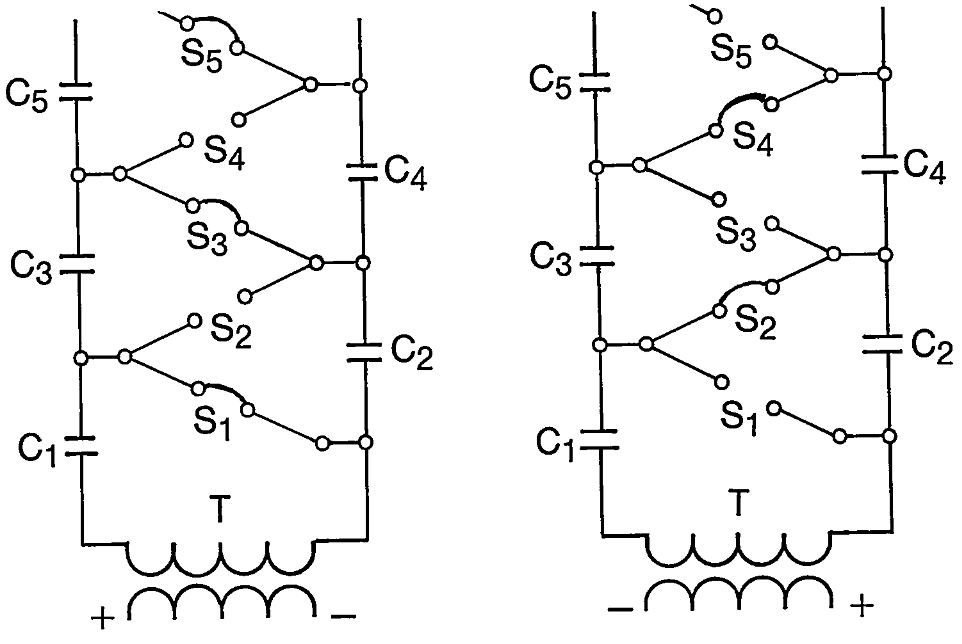
Moreover, these machines have greatly enhanced the capacity to produce radioactive substances. They have also added to a much clearer picture of the interaction processes of nuclei and charged particles or ions. The machines have produced many of the elements beyond uranium.

1. Cockcroft-Walton Accelerator

The first important use of an accelerator resulted from the studies of J. D. Cockcroft and E. T. Walton in 1929.[1] Other devices had been used to produce high-energy ions, but theirs was the first with which a nuclear transformation was obtained by artificial means. The era of particle accelerators began with the work of Cockcroft and Walton. Since the development of their machine, great progress has been made toward increasing the energy of ions by means of high voltages.

Let us assume that a voltage is applied between two plates (electrodes). An ion between the plates is acted upon by the field and drawn toward one of these plates. The energy of the ion increases as a result of this action. The unit used to measure this increase is the electron volt (eV), see Section 1.F. If we can achieve a high enough voltage between the plates, we can get high-energy ions. For instance, suppose we had a parallel plate system with a voltage of 3 kV between the plates. Then, the energy of an ion with one unit of charge would be increased by 3 keV in moving from one plate to the other.

The basic purpose of the Cockcroft-Walton machine is to obtain a high voltage in order to obtain a beam of high-energy charged particles or ions. Cockcroft and Walton produced high-energy ions with a voltage multiplier (see Figure 17.1).[1] A transformer T provides the high-voltage alternating potential. The capacitors C_1, C_2, $C_3, \ldots$, are all of equal value. Switches S_1, S_2, $S_3, \ldots$, are in reality high-voltage rectifier tubes. These tubes work so that they conduct during only half of the alternating cycle. Therefore, they act like automatic switches. During half of the alternating cycle, switches S_1, S_3, S_5, etc., are closed; that is, these tubes are conducting. At the same time, switches S_2, S_4, S_6, etc., are open or nonconducting (Figure 17.1a). During this half-cycle, C_1, C_3, C_5, etc., are charged. During the next half-cycle, C_2, C_4, C_6, etc. are charged. Moreover, during the latter half-cycle, some of the charge on C_1 is shared with C_2 and likewise for C_3 and C_4, C_5 and C_6, etc. For a set of two tubes and two capacitors, the voltage supplied by the transformer is doubled. This



(a) Charging Half-Cycle for C_1, C_3, C_5, etc.

(b) Charging Half-Cycle for C_2, C_4, C_6, etc.

Figure 17.1 Voltage multiplier. (Adapted from Reference 1)

occurs because the voltage built up on the capacitor pair is in series. The set is referred to as a voltage doubler.[2] The total output voltage of the machine is approximately twice the number of voltage doublers times the transformer voltage.

Cockcroft and Walton produced a potential of about 800,000 volts using a transformer with 100,000 volts across the secondary coil.

The voltage thus attained is then applied to an evacuated tube. Ions are urged into one end and accelerated by the potential down the length of the tube, see Figure 17.2. All types of ions can be accelerated. A gas-discharge tube is often used as the ion source. To ensure as large a beam current as possible, the accelerator tube must be operated under a vacuum. The vacuum reduces scattering of the ions.

Although the energy produced in this type machine is not as high as in other accelerators, this machine does have advantages. Fairly large ion currents, which are useful for experimental work, can be obtained at constant voltage. The machine is relatively simple with no moving parts. The high voltage, which is limited only by breakdown, can be varied by changing the input voltage from the transformer.

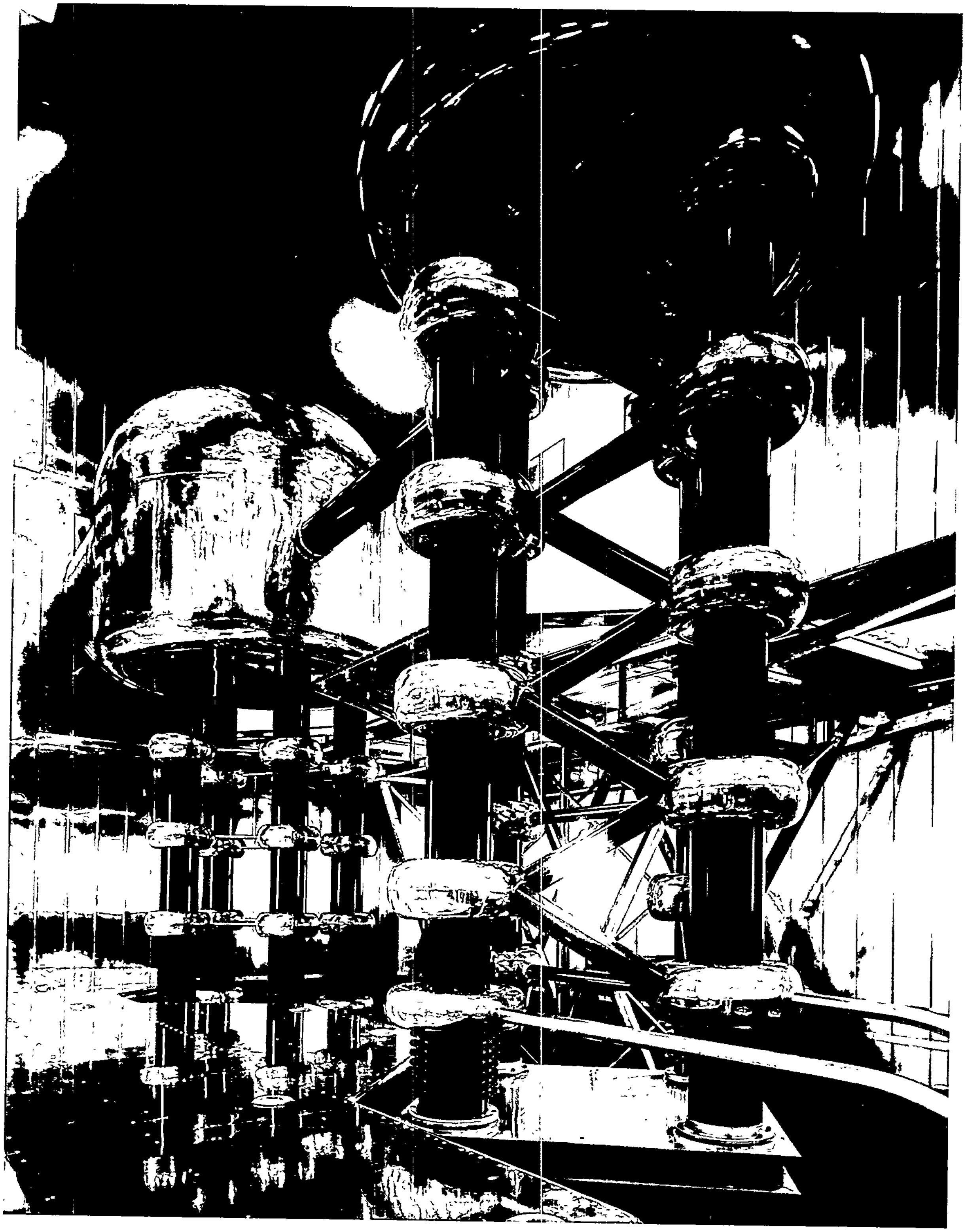


Figure 17.2 A 750 kev Cockroft-Walton used as a pre-accelerator at the IPNS-1.

The work of Cockcroft and Walton spurred the design and building of machines that could produce higher and higher energy particles. Such particles were required to probe the nucleus in an effort to "see" its structure. A Cockcroft-Walton machine can produce energies up to about 3 MeV. Present day use of such machines are primarily as preaccelerators and as neutron generators for 14 MeV neutrons.[3]

The advance in the field is indicated by the fact that in only about 60 years, machines have been built to produce particles of greater than 1 TeV (10^{12} eV).[4]

2. Van de Graaff Electrostatic Generator

a. General

Higher voltage limits were attained with the advent of the electrostatic generator by R. J. Van de Graaff in 1931. The principle of the machine is based on two facts: (1) A conducting sphere, or other hollow body, can accept a charge despite its own voltage; (2) electrical discharges occur readily at sharp points.

The electrostatic generator consists of a belt and pulley arrangement.[2] The belt, made of silk, paper, rubber, or other non-conducting fabric, is driven at high speed by a motor. A d.c. potential (20- 50 kV) is applied between a set of needle points, A, and a rounded surface, B (see Figure 17.3). The intense electric field at the needle points causes a corona discharge. The discharge occurs when we have high electric fields and high surface density. It causes the ions to speed toward or away from the points, depending upon the direction of the field. In the device shown in Figure 17.3, electrons are drawn from the belt toward the needle points. This leaves a positive charge on the belt. The moving belt takes the positive charge up to the second set of needle points, D, and rounded surface, C. In this case, electrons from the needle points are "sprayed" onto the belt. This places a positive charge on the metal sphere, called the corona cap. If a wide belt is used and the belt

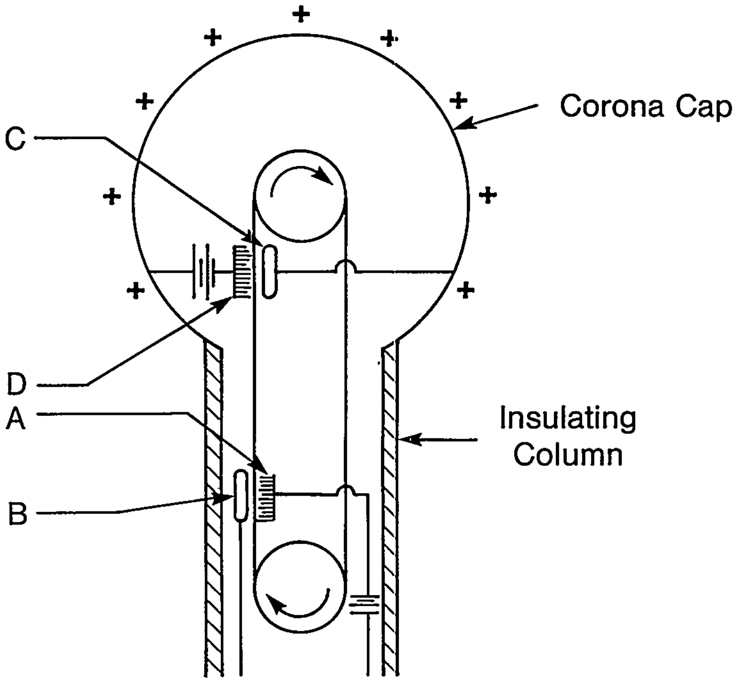


Figure 17.3 Van de Graaff electrostatic generator. (R.E.Lapp / H.L.Andrews, NUCLEAR RADIATION PHYSICS, 2/e, 1954, p.255 Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, NJ)

is driven at high enough speed (up to 60 miles/h), large charges can be built up and maintained on the cap.

The voltage of the cap with respect to earth increases until it reaches a certain value. Then, electric charge begins to leak away from the sphere surface as fast as it is taken from the belt. The limiting potential difference obtained depends upon the radius of the cap, the smoothness of the cap surface, and the pressure and moisture content of the air about the cap. The maximum voltage can be increased and the leakage decreased by placing the machine in a gas-tight steel shell, filled with a gas (nitrogen or freon) at high pressure (about 20 atm).

The potential developed by the device can then be applied to the ions that are to be employed. The voltage is applied to an evacuated accelerating tube, and ions are fed into the tube at the high-potential end. Some problems are encountered in the design of these tubes. The tubes have to be long enough (about 6 m) to avoid a discharge across the length of the tube when the potential is applied. In many cases, they are made up of sections of some insulating substance, such as glass or porcelain.

It is rather easy to continuously vary the output voltage of the Van de Graaff (VdG) accelerator. The output voltage also can be held constant to an accuracy of about 0.1%. These controls over the voltage are of value in the study of the energy dependence of certain nuclear processes. The machine can accelerate all ions, but is generally used for protons or heavier particles. Most of these devices are designed to produce particles up to about 10 MeV.

b. Tandem Van de Graaff Machine

The two-stage Tandem Van de Graaff machine is a device which can double the energy of the accelerated particle.[5] Positive ions are urged into one end of this machine, where they pick up electrons and thereby become negatively charged. These ions are then accelerated to the high-voltage terminal. At this point, the negative charges are stripped off and the positive ions are further accelerated back to ground potential. The energy thus gained by the ions is twice the potential of the high-voltage terminal. Ion beams of about 20 MeV are easily obtained with these devices.

The Cockcroft-Walton machine and the two Van de Graaff machines described above use a static means of acceleration. This means that a d.c. potential is used to directly accelerate the ions. These devices are referred to as direct accelerators. In this type of device, the acceleration path is linear, electrostatic or electromagnetic forces provide the acceleration, the system may be used for any charged particle, the beams are usually direct current and the energy limit is generally a few MeV (except for the Tandem VdG).

3. Linear Accelerators

a. Linear Radio-frequency Accelerator

The linear accelerator (or Linac as it is commonly referred to), employs a dynamic or alternating potential to accelerate the ions. Sloan and Lawrence built the first machine of this type in the

United States in 1931. Their work was restricted to the use of heavy mercury ions because high-frequency power was not available at that time.

In this device, a series of hollow metal cylinders (called drift tubes) of increasing length are arranged in a straight line as shown in Figure 17.4. The first, third, fifth, etc., tubes are connected to one terminal, A, of an a.c. generator, which runs at constant frequency. The second, fourth, sixth, etc., tubes are in contact with the other terminal, B. As a result, all odd-numbered tubes carry a negative charge when even-numbered tubes are positively charged, and vice versa.[5] That is, alternate tubes carry opposite potentials.

Assume that the source releases a positive ion when the first tube is in its negative half-cycle. The electric field between the source and the tube accelerates the ion until it enters the tube. Once inside, the ion travels at a constant velocity or "drift" through the tube, since this is a field-free region. If the ion emerges just as the

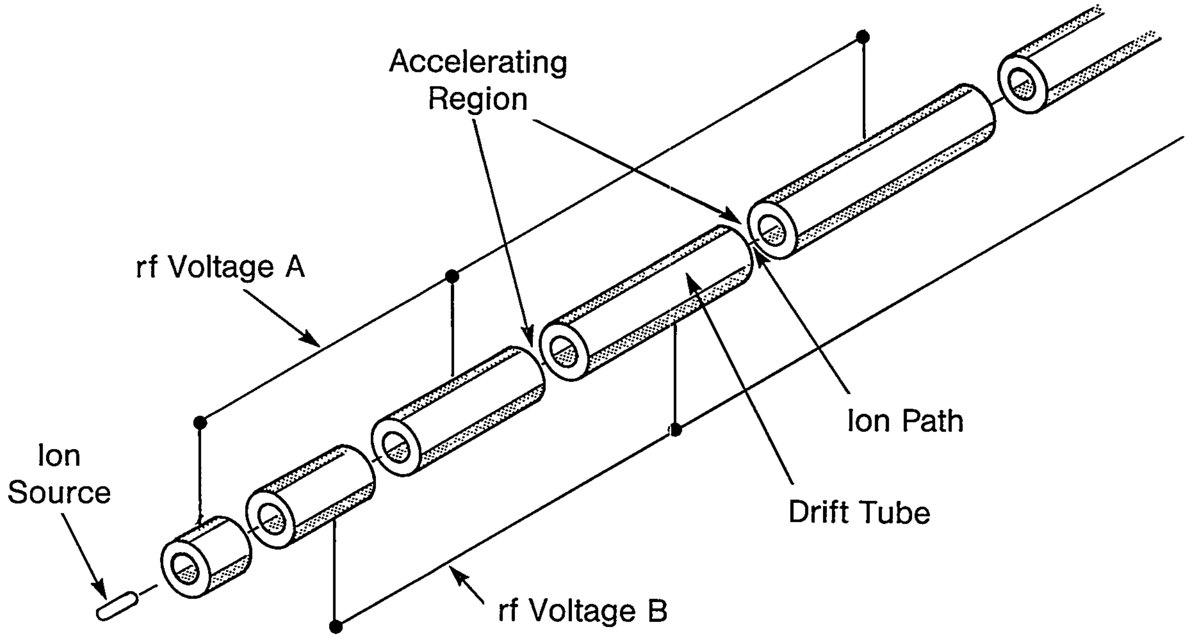


Figure 17.4 Linear radio-frequency accelerator. (R.E.Lapp / H.L.Andrews, NUCLEAR RADIATION PHYSICS, 2/e, 1954, p.257. Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, NJ)

potential on the first and second tubes is reversed, the ion is accelerated once again in the gap between the two tubes. As the ion enters the second tube, its velocity is greater than that in the first tube. Thus, it will travel a given distance in a shorter time. However, the frequency of the a.c. potential is fixed and the time between half-cycles is fixed. The ion must spend the same time in each tube so that it arrives at the next gap in phase with the a.c. potential. To continue this acceleration, and increase the energy of the ion, each successive drift tube (or gap and tube) must be longer than the preceding one. The length must be properly proportioned to the speed of the ion. Then, the time of travel through this distance will be equal to one half-period of the a.c. cycle. As the ion velocity nears the speed of light, there is no further need to increase the tube length. At this point, the ion velocity approaches a limit. Thus, the required length of the tubes becomes essentially constant. The final ion energy is equal to the sum of all increments it gained in crossing the gaps.

Sloan and Lawrence used about 36 drift tubes. The overall length of this accelerating system was nearly 1.8 m. With an applied potential of 79 kV, their machine produced mercury ions with an energy of 2.85 MeV.[1]

Interest in this type of machine decreased shortly before World War II. During the war years, radar research produced very high-frequency oscillators of considerable power. New rf tubes, which could be applied to Linacs, were produced. This renewed interest in the machines.

b. Linear Electron Accelerator

With the advent of radar and the increased use of microwaves (electromagnetic waves of wavelength from 0.01 to 1 m), new methods of acceleration have been applied to the postwar Linacs.

High-frequency microwaves can be sent through hollow metal tubes (called waveguides). A number of disks may be placed in a waveguide, which then acts like a filter to pass a certain band of frequencies. This is referred to as a disk-loaded waveguide.[1] The propagation of the waves depends upon the spacing between, and the hole size of, the disks. Since these waves have an associated electric field, they can be used to accelerate particles. A particle fed into the guide at the proper phase, is carried along with the wave like a surf rider on the crest of an incoming water wave. This type machine was designed by W. W. Hansen and others at Stanford University in 1948.[2]

Its overall length was approximately 3.7 m. This was a traveling-wave type machine, which produces a pulsed beam of electrons. In 1966, a 3.22 km (2 mile) long electron linac was constructed at the Stanford Linear Accelerator Center. This accelerator is designed to produce electrons of energy up to 40 GeV.

c. Linear Proton Accelerator

L. W. Alvarez at the University of California built a Linac for protons at about the same time the linear electron accelerator was built. His device is a standing-wave type machine. Drift tubes are located along the axis of the guide. The length of the tubes is such that protons receive a maximum acceleration when they cross a gap. The drift tubes act as a shield for the protons. That is, when the field between tubes is in the wrong direction, the protons are coasting through the tubes. Protons of 4 MeV were fed into the machine. They reached an energy of 32 MeV as they passed through the 12.2 m accelerator.

Figure 17.5 is a picture of the 50 MeV proton linac of the Alvarez type (200 MHz, 124 drift tubes), which is used as a preinjector for the Intense Pulsed Neutron Source at ANL. Protons (750 keV) from a Cockcroft-Walton type accelerator are introduced into the proton accelerator (approximately 34 m long) and accelerated to an energy of 50 MeV before being injected into the Rapidly Cycling Synchrotron (RCS). This linac system was originally constructed and used for injecting protons into the Zero Gradient Synchrotron, a 12 GeV proton machine which has now been closed down.

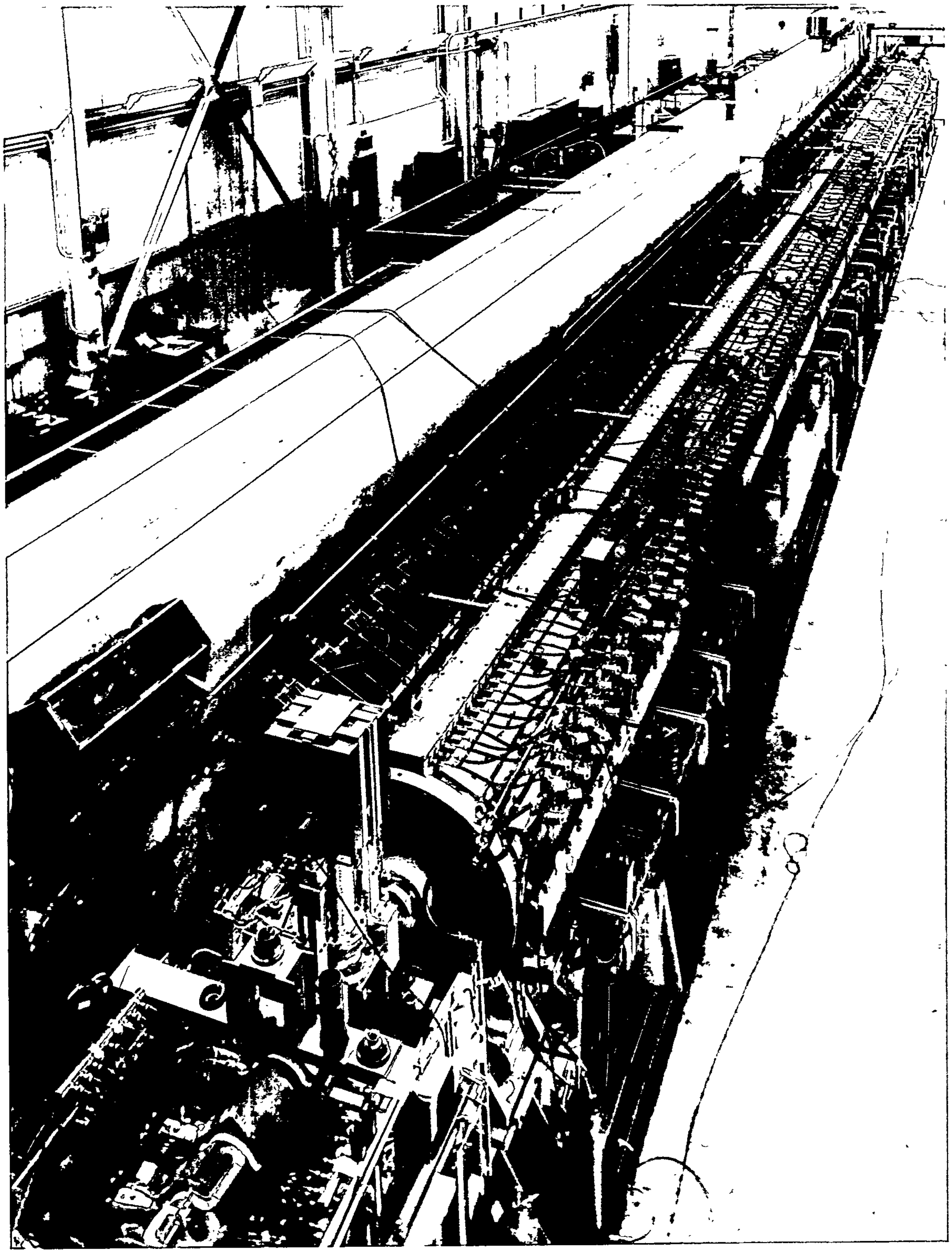


Figure 17.5 A 50 MeV proton linac used for IPNS-1.

4. Cyclotron

In the above machines, high-energy particles are produced by accelerating them in a straight line. A new method for obtaining such particles was conceived by E. O. Lawrence at the University of California in 1930.[1] He and his co-workers built a unit that employed a magnetic field to cause particles to move in a spiral of increasing radius. This machine was known as a magnetic resonance accelerator. Since 1936, this device has been referred to as a cyclotron.

The basic design consists of two electrodes, which may look like the two halves of a pill box (see Figure 17.6). In the early models, these electrodes were shaped like the letter D; therefore, they are called dees. The dees are mounted inside a vacuum chamber, located between the poles of a large magnet. A high-frequency alternating voltage is connected to the dees.[5]

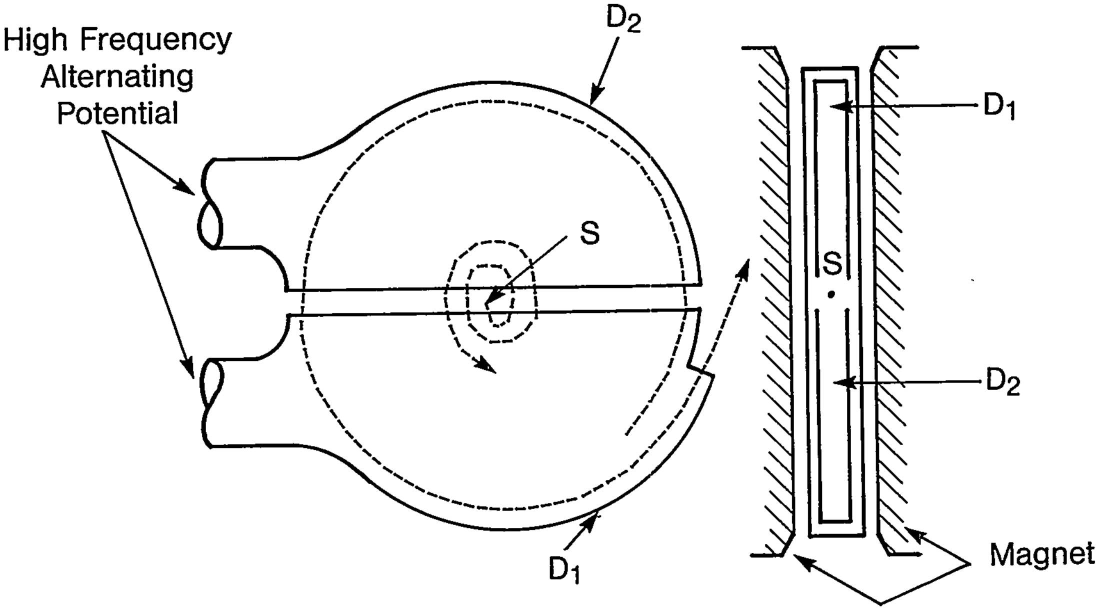


Figure 17.6 Cyclotron (Adapted from Reference 1)

Assume that a positive ion is released from the source, S, when D_1 is negatively charged. The electric field between the dees urges the ion toward and into D_1. Inside D_1, the ion is free of electric forces and travels at a constant speed. However, the magnetic field forces the ion into a semicircular path, which directs the ion back toward the gap. As the ion reaches the gap, the potential reverses. The electric field once again accelerates the ion across the gap. The ion plunges into D_2 with more kinetic energy than it had in D_1. To remain in phase with the alternating voltage, the ion must spend the same time in each dee. To do this, the faster-moving ion travels a greater distance in D_2. This means that the radius of the path in D_2 is greater than in D_1. As this action continues, the ion energy increases. Ions follow a flat spiral path until they reach point P, where they can be deflected out of the chamber to bombard a desired target.

If the charge and mass of the ion plus the magnetic field remain constant, all ions take the same time to traverse the path in each dee. Therefore, the needed frequency of oscillation can be set to the nature of the ion and the strength of the magnetic field. Proper adjustment causes the moving ion to be in phase with the changing voltage. Thus, the ion is always accelerated as it crosses the gap, and its energy increases.

The highest energy a certain ion can acquire in a given cyclotron is independent of the voltage applied to the dees. If the voltage is small, the ion revolves many times before reaching the outer edge of the dees. When the voltage is high, the number of turns is small. The energy limit for a given ion depends upon the relativistic mass increase with energy and the radial decrease in the magnetic field.

Cyclotrons are referred to in terms of the diameter of the magnet pole faces. Thus, a "60-inch cyclotron" would imply a machine with pole faces 1.5 m in diameter; the diameter of the dees being slightly less than this.

In addition to applications in nuclear physics, recent interest in the cyclotron principle involves the possibility of clinical use. The machine may be used to produce an intense neutron or proton beam for radiotherapy, or to activate materials for tracer studies and diagnostic uses.[6]

5. Betatron

Certain studies require the use of high-energy electron beams. The cyclotron cannot be adapted with ease to this purpose because relativistic effects cause the mass of an electron to increase at fairly low energies.[1,2] A device for the acceleration of electrons was built by D. W. Kerst at the University of Illinois in 1940. The principle of magnetic induction is employed in this machine (called a betatron).[7] According to this principle, a changing magnetic flux produces an electomotive force (emf) in a conductor. This induced emf can be used to produce the desired beam.

The betatron has a ring-shaped tube, referred to as the doughnut, between the poles of an electromagnet. The magnet is energized by an alternating pulsed current. As the current varies, the magnetic field produced also varies. Electrons from a hot wire are urged into the doughnut just as the field is starting to rise. Two effects are produced by the changing magnetic flux: (1) An emf is produced, which imparts energy to the electrons; (2) a radial force acts on the electrons to keep them moving in a circular orbit. The electrons are accelerated only during one-quarter of the a.c. cycle. During this time, the field is changing from zero to its highest intensity. However, the electrons make a few hundred thousand complete turns within this time. Since they gain energy with each turn, they are able to reach high energies. When the electrons attain the desired energy, they can be directed toward an x ray target or out of the doughnut through a thin window.

An electron moving in a circle loses more and more energy as its speed increases. Since the electron is being accelerated, it will radiate bremsstrahlung.[6] The higher the energy of the electron, the more energy it will radiate. Also, the bremsstrahlung will be highly peaked in the forward direction (see Section 3.B.3). The amount of energy lost per turn, E, by the electron, is given by[7,8]

$$E = 8.847 \times 10^{-11} \frac{E^4}{r} \text{ (keV/turn)} \qquad 17.1$$

in which E is the electron energy in MeV and r is the radius of curvature of the magnetic field in m. Losses become important in these machines when E is about 100 MeV. Because of such losses, betatron energies have to be limited to about 500 MeV.[2] Kerst built a betatron that produced 300 MeV electrons.

Interest in the betatron with respect to nuclear studies has not grown because higher energies can be achieved with Linacs and synchrotrons. However, betatrons in the energy range from about 10-50 MeV have been useful in therapy applications.[6]

6. Synchrotron

a. Proton Synchrotron

The synchrotron concept removed some of the limitations imposed on other cyclic machines. Cyclotrons, for instance, can produce protons that are limited to about 25 MeV.[7] In a synchrocyclotron (FM cyclotron), the cost factor or solid-core magnets restricts the proton energy to about 1 GeV.

Some synchrotrons have been built to produce high-energy protons; others were designed for electrons. Our present discussion is concerned with proton machines.

A typical synchrotron consists of a more or less circular vacuum chamber with a ring-shaped magnet and short straight sections (see Figure 17.7).[9] The protons are accelerated at one or more points along their path by an rf electric field. This field must be synchronized with the motion of the protons. As the proton gains energy, its speed increases. Therefore, the frequency of the accelerating potential must also increase during the time of acceleration. In this machine, then, the strength of the magnetic field and the frequency of the rf electric field are changing during the acceleration phase. The radius of the circular

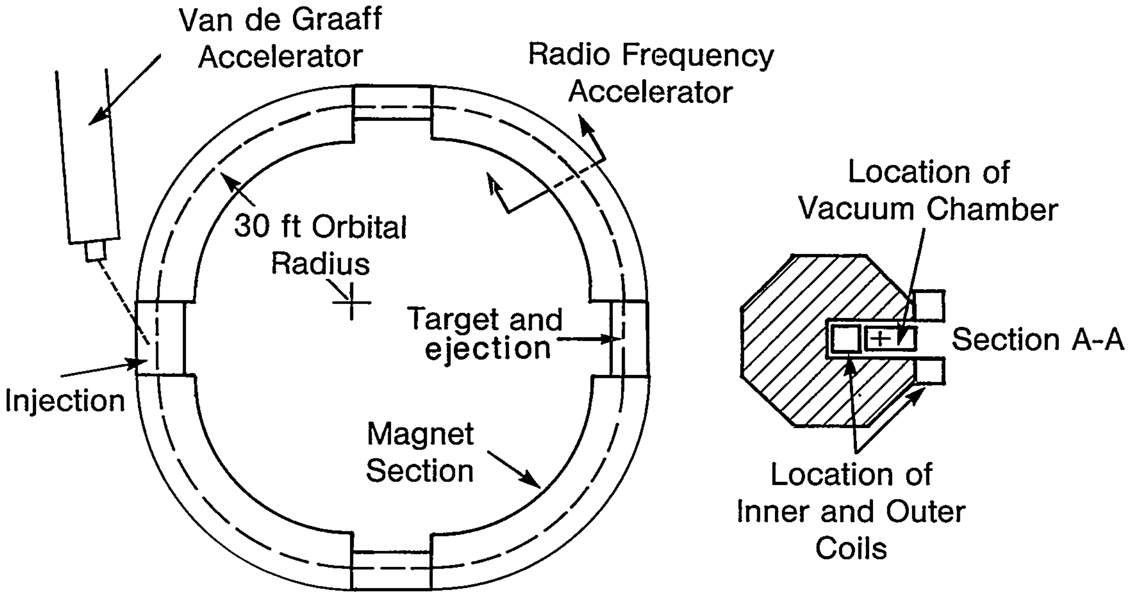


Figure 17.7 Synchrotron (the Cosmotron). (M. H. Blewett et al, Review of Scientific Instruments 24, No. 9 (1953). Origional prints by Brookhaven National Laboratory.)

path remains fixed. Therefore, the final energy of the protons depends upon the maximum magnetic field that can be attained.

The protons are usually accelerated to a certain energy in another type of machine and then injected into one of the straight sections of the synchrotron. They are injected when the rising magnetic field has reached the minimum value required for the energy of the incoming protons. The proton beam enters the vacuum chamber at a radius slightly larger than that of the path it eventually follows. The beam spirals inward slightly as the field continues to increase. The protons spiral in until they reach the desired orbit. At this time they also reach the point of the rf potential. Therefore, they receive an acceleration. The beam then travels in a more or less constant orbit as the magnetic field increases to offset the energy acquired by the protons. To gain more and more energy, the protons have to get a certain "push" at each acceleration point. This means that the protons move faster in a given revolution than in the previous one. Therefore, the frequency of the rf voltage must be increased so that when the protons arrive again at the point of acceleration they receive the same push as before. The changing magnetic field keeps the radius of the path constant. The changing frequency of the rf voltage results in the same acceleration being imparted to the protons during each revolution.

The very high-energy protons produced in these machines are not extracted as a continuous stream but rather as a bunch (pulse) of protons. After one bunch of protons is brought to full energy, the magnetic field and the frequency of the accelerating voltage must return to their starting values.[5] At this time, another bunch of protons can be brought up. The repetition rate in some of the larger machines is on the order of once every few seconds.

The Cosmotron at Brookhaven National Laboratory and the Bevatron at the University of California produced protons of 3 and 6 GeV, respectively. The Alternating Gradient Synchrotron at Brookhaven, which produces protons of 33 GeV, has a circumference of about 800 m (approximately ½ mile).[7] In order to go to higher energies, in the 100-1000 GeV range, and keep the machine size within reason, new concepts have been utilized. At Fermilab, the proton synchrotron (circumference approximately 6.3 km), which became operational in 1972, reached a proton energy of 400 GeV and a beam intensity of 2×10^{13} protons/pulse. By utilizing the concept of superconducting magnets (magnets cooled to very low temperatures, near -273°C), the magnetic fields have been doubled (and also the beam energy) within the 1980s, without having to construct a new tunnel (main ring). In addition to the expectation that this accelerator will exceed 1 TeV (1000 GeV), the concept of colliding beams is anticipated. Two storage rings will be used to accumulate protons and anti-protons, respectively, for colliding beam experiments.[4] The advantage of this mode of operation is that the total available energy when two particles collide equals the sum of the energies of the two particles.[1] For a conventional beam striking a stationary target, the available energy is roughly proportional to the $\sqrt{E}$ of the incoming particle. Storage rings are needed to increase the total number of available particles to increase the probability of collisions. Not yet fully operational, this system is referred to as the Tevatron Collider.

b. Electron Synchrotrons

In the betatron, the maximum energy attainable was limited by the radiation loss (bremsstrahlung) of the electrons circulating in the beam orbit (see equation 17.1). Because only a certain limited amount of energy per turn can be imparted to the electron, eventually the radiative loss per turn catches up to the energy added per turn, because of the changing radius.

In the electron synchrotron, the electron is accelerated in an orbit of constant radius, constrained by the changing magnetic field.[7] That is, the electron is introduced from a preaccelerator at a low enough energy for the magnetic field to capture the electron in orbit. At the high end, the electron is extracted because of the limitation in the variability of the magnet. The increase in the magnetic field during the acceleration cycle matches the rate of increase in momentum of the electron, supplied by the rf acceleration. Because the mass of the electron is small, the velocity of the electron quickly approaches the speed of light. At that point, the velocity of the electron remains constant, and only a fixed frequency rf accelerating voltage is needed. If an alternate system is used to accelerate the electron to a few MeV, then no frequency modulation[2] is needed after injection, since the electron speed will be approximately c.

As in the case of the proton synchrotron, a limitation is reached in which the economic factor dominates the considerations with respect to energy increases. However, in the case of electron synchrotrons, the emphasis has shifted from the value of the energy of the electron to the usefulness of the bremsstrahlung produced. With respect to equation 17.1, it is seen that the higher the electron energy for a given system (r=constant), the more energy is radiated as bremsstrahlung. Because of the intensity of the photons emitted (ph/s), the bremsstrahlung has become an important tool in the study of surface effects in materials, among other applications. The most powerful rotating anode x ray generators produce only about 10 W of x rays, in a narrow band around a particular energy. The bremsstrahlung (or synchrotron radiation as it is called) is a spectrum of x rays, which provides several orders of magnitude higher intensity.[8]

(1) Synchrotron Radiation Characteristics

The synchrotron radiation is emitted essentially in a narrow beam, and the intensity of the radiation is very high. The radiation is emitted in a half-angle described by equation 3.14,

$$\theta = \frac{m_o c^2}{E} \sim \frac{1}{2E} \text{ (rad), in which E is in MeV.}$$

From this relationship, it is apparent that for an electron in the GeV range, the major portion of the bremsstrahlung is emitted in a very small cone in the forward direction.

The energy of the radiation which is emitted covers a spectrum of energy. However, the spectrum is such that half of the emitted power is above a certain energy, and half below. This energy value is called the critical energy, and is given by[8]

$$E_c = 2.218 \frac{E^3}{r} \text{ (keV)}, \qquad 17.2$$

in which E is the energy of the electron in GeV and r is the radius of curvature of the magnetic field in m. For r equal to 25-30 m and E in the GeV range, the critical energy is in the x ray region. In the past, the term "light source" has been used to describe these devices. With the design of these machines approaching several GeV to 10 GeV, the majority of the useful radiation is in the x ray region, not the visible spectrum. For example, if r=40 m and E=7 GeV, the critical energy is

$$E_c = 2.218 \frac{(7)^3}{40} = 19.0 \text{ keV.}$$

So, with the increased energy of the designs, the machines are really high intensity x ray sources.

In general, these machines make use of a storage ring. A beam of electrons is preaccelerated (usually a linac) to some energy (typically a few 100 MeV), and injected into a booster synchrotron. This machine brings the electron energy up to the desired value and injects the beam into a storage ring. The storage ring is a large circumference ring which has an associated rf accelerating potential. As the electrons move around the ring, bending magnets move the electron beam in the desired orbit. The loss per turn due to bremsstrahlung is balanced by the energy per turn supplied by the rf accelerating potential. If there were no other losses, the beam could be sustained indefinitely. However, other losses do occur so that, after a time of about 4-12 h depending upon the machine, the beam must be replenished or "topped off." The losses are due to such things as collision of electrons with molecules in the vacuum chamber, oscillation of particles out of the beam path, and interaction of the particles in the beam. As the vacuum gets better, the lifetime of the beam is generally enhanced.

As the beam circulates, bremsstrahlung is given off each time the electron is accelerated, such as in the bending magnets. It is also possible to design sections (insertion devices) in which a number of magnets are present with alternating fields. This, in effect, causes a number of particle accelerations, increasing the total bremsstrahlung output. The synchrotron radiation from the bending magnets and the insertion devices is directed from the storage ring to the experimental area.

(2) ANL Advanced Photon Source

Work has begun on the design of the ANL Advanced Photon Source. This device will utilize 7 GeV positrons in a 1060 m circumference storage ring to produce the synchrotron x ray radiation. The maximum positron current will be 300 mA and the machine will require the conversion of electrons to positrons in a tungsten target, and the subsequent acceleration of the positrons in a linear accelerator (see Figure 17.8). A "booster" synchrotron then, accelerates the positrons from 450 MeV to the injection energy (7 GeV) and injects the positron pulse into the storage ring. Injection of positron pulses will be continued until the maximum current is reached. A number of insertion devices (see Figure 17.8) referred to as "wigglers" or "undulators" are also being planned.[10] The machine is expected to be operational in the 1990s.

c. Pulsed-Neutron Source

One further application of synchrotrons is to accelerate certain particles to a given energy and then direct the beam into a suitable target to produce other desirable particles. This is the purpose of the ANL "Intense Pulsed-Neutron Source (IPNS)." The layout is shown in Figure 17.9. A Cockcroft-Walton preaccelerator, A, accelerates H^- ions

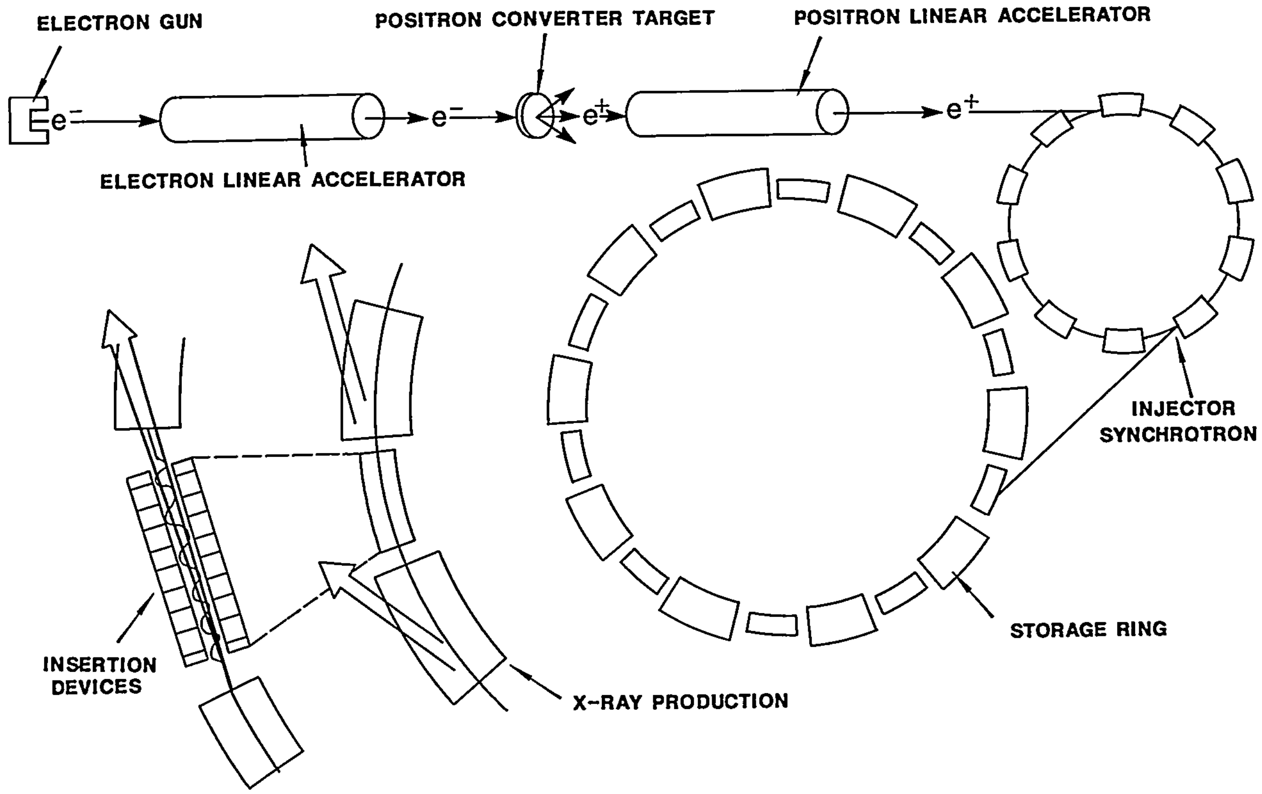


Figure 17.8 Proposed Components of the Advanced Photon Source.[10]

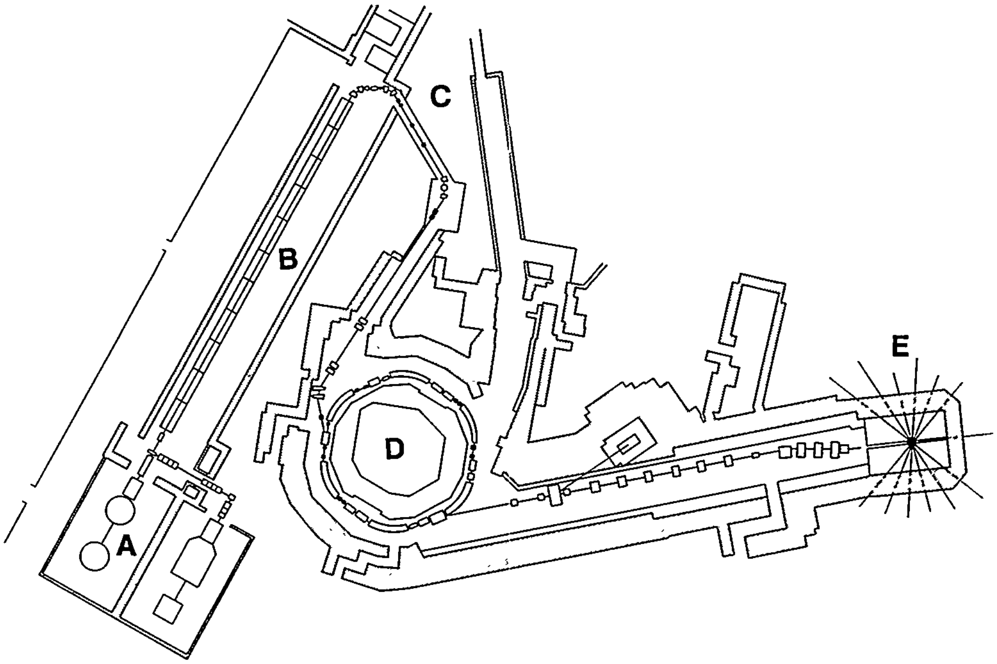


Figure 17.9 The IPNS-1 facility for producing pulsed neutron bursts.

to 750 keV, and these are injected into the 50 MeV, 200 MHz Alvarez type Linac, B, featuring 124 drift tubes driven at a single feed point by a 5 MW rf transmitter. The beam is transported from the Linac by a transport line, consisting of eight dipole bending magnets and sixteen quadrapole focusing magnets, C, into the rapidly cycling synchrotron (RCS), D, where it is stripped to an H^+ beam and accelerated to 450 MeV. The full energy beam is transported down a beam line to a depleted uranium target, E, in which the neutron pulses are produced by fast fission and spallation reactions. About 14 neutrons per proton can be achieved so that neutron fluence rates available are in the range 10^{16}-10^{17} n/m^2s, for a repetition rate of 30 Hz.[11] The main experimental use of this facility is for neutron scattering experiments. At present, plans call for the use of an enriched uranium target which is expected to increase the neutron fluence rate by a factor of three.

Other uses of accelerators, such as potential breeding of fissionable material have also been proposed.[12]

B. Elementary Particles

1. General

There seems to be a never-ending quest by scientists to find an answer to the question, "What are the elementary particles?" That is, what are the basic particles out of which our entire universe is composed?

Much of the present knowledge concerning subatomic particles was obtained from studies with accelerators. The early machines produced particles that could invade the more massive nuclei. However, observations of the framework of nuclei were difficult, and approximations had to be made. Higher energies were required to obtain more detailed and useful data. At present, many people are interested not only in the nucleus but also in the makeup of the neutrons and protons themselves. To "see" nucleon structure as clearly as we are able to see the structure of larger bits of matter, machines must produce particles with energies in the GeV range.[1,7]

The term "high-energy physics" is commonly used to refer to the branch of physics that deals with elementary particles and the forces involved in their mutual interactions.[1] This seems to place emphasis upon the means employed rather than the objects under study. A more appropriate term would be "elementary-particle physics" because the particles themselves are being studied rather than the energies involved.

Cosmic-ray research made vital early contributions to this branch of physics. High-energy particles are found in cosmic rays, but they are rare and uncontrolled. The more recent advances in the field have resulted mainly through the use of high-energy machines. For our purposes, high-energy machines can be arbitrarily taken as accelerators producing particles with E>100 MeV. The energies involved in these studies allow the scientist not only to see the particles in greater detail, but also to create new ones. It is well known that an electron-positron pair is produced from gammas with E>1 MeV. If the energy is high enough, products are formed that are many times more massive and more penetrating than the electron-positron pair.

The list of particles considered to be elementary has changed considerably over the years as higher energy projectiles have become available to probe deeper into the structure of nuclear matter. With the advent of colliding particles, finer detail has been achieved, and experimental resolution improvements have led to a smaller number of fundamental entities.[13,14] Combinations of these fundamental entities lead to the large number of observed particles. One group of fundamental particles, called leptons, are not affected by the strong force which binds the nucleus together. Six types (flavors) of leptons have been identified: electron (e^-), muon (μ^-), tau (τ^-), electron neutrino (ν_e), muon neutrino (ν_μ) and tau neutrino (ν_τ). The first three carry an identical -1 charge but differ in mass. The latter three are electrically neutral with essentially zero mass for ν_e and ν_μ, but ν_τ has not been directly observed. In addition to the leptons, each has a corresponding antiparticle. The group of particles known as leptons are all lighter in mass than the proton.[14]

Among particles which do experience the strong force in the nucleus are hadrons (among which are protons, neutrons and mesons). Although hundreds of species of hadrons have been catalogued in high energy reactions, hadrons are not elementary particles. The hadron species are now believed to be combinations of five fundamental particles called quarks.[13] These are called up (u), down (d), charm (c), strange (s) and bottom (b). In addition, the top (t) quark is also believed to exist. The electric charge carried by a quark is a precise fraction of the charge on an electron. The d, s and b quarks have a -1/3 charge, while the u, c and presumed t quarks have a charge of +2/3. The corresponding antiquarks have charges of the same magnitude but opposite sign. Hadrons do not show fractional charge since quarks combine in such a way that only integral charges results. For example, a proton is viewed as composed of two u and one d quarks for a total charge of +1.

The present scheme states that of those 12 fundamental particles, the electron, the electron neutrino, the u and d quarks are found in ordinary matter. The other particles are observed only briefly as the result of high-energy collisions.[13] In addition, the photon may also be classed as a fundamental "particle."

2. High Energy Processes

The radiations of concern from high energy processes include protons, neutrons, $\pi^{\pm}$ and other mesons, electrons, muons, and x and gamma rays.[14-18] In general, the radiation environment which emerges from the shielding of high-energy accelerators is dominated by neutrons and photons.[14,15] Although the processes which initiate the release of secondary products differ, high-energy protons initiating hadronic cascades and high-energy electrons producing electromagnetic cascades,[14] the fields which emerge from the thick shielding are rather similar.

a. Hadron Cascades

In the first stage of a high-energy process, the proton strikes the target and a number of hadrons (p, n, $\pi^{\pm}$, other), as well as fragments, may be instantly ejected.[2,14] Depending upon the mass number, A, of the fragmented ejected matter, the process is referred to as fragmentation or spallation.[2] If $A>10$, the process is called fragmentation. Spallation refers to processes in which smaller bits of matter, which include nucleons and other hadrons, are emitted. The hadrons ejected in the above processes are sometimes called cascade particles. The ejected particles in this stage move primarily forward.

In a primary interaction, cascade protons and neutrons, as well as π^{+}, π^{-}, and π° mesons, are "knocked" out of the struck nucleus.[14] Hence, this stage is often called the knock-on phase.[2] As a result of this bombardment, the residual nucleus is left in a highly excited state. Such a nucleus may give off a series of protons, neutrons, and gammas during de-excitation. This is often referred to as the _evaporation_ phase.[14]

The particles which are knocked out of the struck nucleus are initially energetic enough to produce the same type of reactions in other nuclei. These secondaries are characterized by an attenuation length about equal to the primary and the ability to produce a multiplicity of particles when they interact. So the number of particles in the cascade continues to grow as the number of reactions increase. However, the average energy carried by particles decreases as the number of interactions increases, since only so much energy was initially available. Eventually, a point is reached in the medium at which multiplication of the cascade can no longer be sustained because the energy of the particles is now too low. The cascade then begins to be attenuated exponentially (see Figure 8.5). At this point, interactions result in reduced energy of the particles without ejection of more particles and eventual absorption of particles.

Among the products formed in the hadronic cascade which may be of concern to health physicists are the mesons. A meson is a particle whose rest mass is between that of an electron and that of a proton. K mesons (called kaons) and π mesons (called pions) are formed in the cascade process and released with high energy. Some of these decay to muons (not a meson) before they lose all their energy in the propagation of the cascade. Muons are also produced by photons in muon pair production, which has an energetic threshold of 106 MeV. This reaction is unimportant except perhaps in high-energy electron accelerators. π^{o} mesons decay into two photons, which can initiate electromagnetic cascades or showers (see below).

The muon, which decays to an electron, has a mean lifetime of 2.2 μs and a mass equal to about 208 m_e. Muons do not readily interact in matter and primarily lose energy by ionization.[14] For this reason, muons have long ranges in any kind of matter.[20,21] Consequently, these particles are not easily shielded and dominate the shielding requirements for proton accelerators above 100 GeV in regions downstream of beam targets. Shielding design aspects for high-energy proton machines are discussed in References 15 and 19.

As mentioned above, the residual nucleus is left in an highly excited state following the collision, and loses this excitation energy by emitting particles, mainly neutrons and protons. These are referred to as evaporation particles, which are emitted isotropically, with an average energy somewhere around 1-4 MeV for the neutron component.

This energy is generally distributed locally and does not contribute to the cascade.[14] The evaporation neutron component is not a shielding consideration if the shield is thick, but may dominate the radiation field in relatively thin shields.[22]

b. Electromagnetic Cascades

The cascades, or showers, produced by photons, electrons, or positrons develop more simply than the hadronic cascades. The energy lost by leptons (e^-, e^+, $\mu^\pm$) is basically by two mechanisms: collision and radiation production.[14] At high energies, the energy lost by bremsstrahlung is the more important component. Since photons are produced at all energies up to the energy of the electron producing the photons, the energy is further propagated into the medium (see Figure 8.1). The high-energy photons then produce e^--e^+ pairs (pair production) or muon pairs which multiply the shower products in addition to further developing the cascade. As in the case of hadronic cascades, as the number of shower particles increases, the average energy per particle decreases. At some point in the medium, the loss due to ionization will become greater than the loss due to radiation production (see equation 3.17), and the cascade will begin to decrease in magnitude.

The cascade, or electromagnetic shower, also applies to any incident high-energy photon, or to a π° produced in a high-energy reaction. Since either of these may result in the production of a high-energy pair, the shower can be initiated. As pointed out above, the shower or cascade continues as long as the radiation loss dominates. An approximate expression for the particle energy E_{CRIT}, at which the radiation loss equals the collision loss is given by[14,19]

$$E_{CRIT} = \frac{800}{Z+1.2} \text{ MeV} \qquad 17.3$$

in which Z is the atomic number of the medium.

Another parameter used to describe the radiation loss in a medium is the radiation length, X_o. This is defined as the thickness of material which will reduce the energy of the electron by a factor of e due to radiative losses. Values of the radiation length in representative materials can be found in References 14, 19 and 26.

In the course of development of the shower for very high-energy e^- (in the GeV range), the photons are so energetic that photo-distintegration reactions occur. In this, the photon is absorbed by the nucleus and, subsequently, it emits one or more hadrons, such as n, p or pions. The threshold for pion production by photons is about 140 MeV.[7] These emitted particles, in turn, have enough kinetic energy to produce similar reactions, so that in addition to the bremsstrahlung radiation, there is also a high energy neutron component which can dominate the shielding requirements.

An additional source of neutrons is the giant resonance neutrons. In this process, the nucleus is excited by an incoming photon, resulting in the emission of a neutron. These neutrons are emitted isotropically, and have an average energy of about 1-2 MeV. As in the case of evaporation neutrons, these are not significant if the shield is thick, but may dominate for thin shields.

C. Radiation Sources

The radiations emitted as a result of accelerator operation vary in kind and degree of intensity. Therefore, shielding demands also vary, depending upon a number of factors, such as type of machine and particle being employed, beam energy and intensity, target substance, shielding substance, and layout of the area. Some of the sources of these radiations are discussed below.

From the discussion of cascade processes, the radiations of concern are neutrons and photons outside the shielding, except that for $E>20$ GeV or so, muons also need to be considered. Inside the accelerator structure, induced activity is produced by heavy charged particles (p, d, α), electrons (e^- and e^+) and photons, and evaporation neutrons.

Neutrons are not formed in electron accelerators until the electron acceleration energy is about 10-15 MeV. For proton or deuteron machines, neutrons can be produced with particles of much lower energy. When e^- are accelerated to greater than 140 MeV, π production by photons begins to occur and this leads to high energy neutrons in electron accelerators. For proton machines, the high-energy neutron component is the most important up to about 20 GeV.

1. Beam Interactions

The primary beam, which may consist of electrons, protons, or deuterons, is often brought out of an accelerator chamber. This is frequently done by sending the beam through pipes or thin foil windows. Since this beam may be quite penetrating, it poses a shielding problem and constitutes a radiation hazard.[23] Moreover, there is the problem of scattered rays. The penetration of a 500 MeV proton, for example, is about .18 m of lead. Although the shielding problem is not too severe in this case, the scattered secondary radiation may present quite a problem.

In considering the primary beam, one may find that a charged particle beam is relatively easy to stop in a given material but large amounts of secondary radiation and heat may need to be dealt with. For example, a 1 kW beam of e^- of a few MeV can produce neutron yields of 10^{11} n/s, or produce x rays equivalent to about 200 TBq of ^{60}Co.[24] The heat load from a 1 kW, 10 MeV e^- beam of 100 mm^2 area will produce a heat rise of about 50°C/s in a 10 mm thick target of ρ = 1000 kg/m^3. So, the beam power, as well as the energy of the particle, is an important consideration. Beam dumps must be designed to take both the heat load as well as effectively stopping the radiation. Another consideration is that the secondaries may be more numerous and more penetrating than the primary radiation. Some machines are designed to use the secondaries (x, γ, n) produced when the primary beam strikes a suitable material. For lower energy machines, the output is almost isotropic, but shifts to the forward direction in higher energy machines. Sometimes, positive ion machines have a problem of backstreaming electrons which produce x rays.

Neutrons are produced in almost all types of accelerators. They arise from beam interactions with the target and structural materials of the machine. Neutron yields and energy spectrum vary widely depending upon the beam energy and the type of process that takes place. Neutrons can be produced by bombarding low Z targets with positive ions accelerated to $E \geq 100$ keV. Electron accelerators, utilizing certain targets (^2H, ^9Be), can produce neutrons by (γ, n) reactions, with electron energies as low as 1.67 MeV. Generally, neutron production in most materials occurs at around 8-10 MeV and all elements produce neutrons when bombarded with particles of 20 MeV or greater per nucleon.[25]

When a high-energy proton beam strikes a material, nuclear cascades take place which result in a number of radiations. As discussed above, the secondaries released when particles in the GeV range interact with matter, are sufficiently energetic to knock out additional secondaries. These cascade particles, which include neutrons, have energies in the hundreds of MeV to GeV range. Mesons can also release additional neutrons and protons. In proton machines, the threshold for pion production by nucleons is around 290 MeV.[7] Although pions interact strongly with nucleons, many of these decay to muons before they are absorbed. Since π° mesons decay into two photons, an electromagnetic cascade is also initiated. However, the attenuation length of the electromagnetic cascade is significantly shorter than that of the hadronic cascade and generally does not influence the radiation field outside thick shielding.[14] Nuclei left in an excited state following interaction with cascade particles emit more neutrons (evaporation neutrons of E approximately 1-4 MeV). These are emitted isotropically and are usually effectively shielded in thick shields containing sufficient hydrogen. The high-energy neutron component ($E>150$ MeV) controls the character of the radiation field for proton accelerators above this energy.[14] The attenuation length of these neutrons is determined by the inelastic collisions, and the inelastic cross section is relatively constant above 150 MeV for a given material. The attenuation length, λ, can be expressed as a function of the mass number A of the material by[19]

$$\lambda = 385\ A^{0.31}\ \mathrm{kg/m^2}. \qquad 17.4$$

For example, estimate the attenuation length in steel (assume iron, A = 55.847) for the high-energy component:

$$\lambda = 385\ (55.847)^{0.31} = 1340\ \mathrm{kg/m^2}.$$

Since $\rho_{Fe} \sim 7900\ \mathrm{kg/m^3}$, the attenuation length in meters would be $1340/7900 \sim 0.17$ m. Note that the attenuation length, λ, refers to the attenuation of the radiation field during the exponential attenuation phase. This occurs only after the radiation field has reached an equilibrium in the shield which is beyond a few mean free paths in the shield (see Figure 8.5).

In high-energy-electron machines, the bremsstrahlung produced in the shower is generally the most important radiation of concern, until the energy goes above 140 MeV. At this point, pion production occurs and high-energy neutrons start to become a problem. In the GeV range, the shield considerations may be dominated by stopping this high-energy component, as well as the bremsstrahlung. Although bremsstrahlung is highly peaked in the forward direction for high-energy electrons, there is a transverse component which needs to be considered, in some cases. In low energy machines, the maximum bremsstrahlung intensity is nearly at right angles to the electron direction. Reference 23 contains information concerning the estimation of emission rates for electron accelerators of $E<100$ MeV. For higher energy machines, bremsstrahlung emission rates can be estimated from the information contained in Reference 26.

2. Stray Radiation

The term stray radiation is used to denote the radiation fields which are found outside of the accelerator shielding. Just as in the case of leakage radiation at reactors, shielding must be checked out at an accelerator to determine that it is adequate.[27] There may be additional sources of radiation though, which are not due to inadequate shielding.[23,28,29]

Misalignment of the beam through changes in, or failure of, magnets, can cause the beam to strike at a certain point and produce a secondary radiation leading to a field outside the accelerator. Sometimes accelerator components may shift from their normal position and intercept the beam causing an unusual radiation field.

When electron beams strike a target to produce x rays, some electrons are reflected back by the target substance. If these strike accelerator components, x rays are produced which can stream back down the accelerator tube. Since x rays in the target are emitted in all directions, backward directed x rays in the accelerating tube can occur.

In direct accelerators, a poor vacuum in the accelerating tube can lead to electron dark currents,[23] even if the ion source is not on. This can happen if the machine voltage is on, since any ions in the system, however produced, may be accelerated. When positive ions impinge on the accelerating tube, electrons may be released which will be back accelerated toward the positive high-voltage terminal.[24] In tandem machines, the mid-terminal contains a foil for changing the charge on the ion. If errant ions collide with the structural parts of the foil apparatus, neutron radiation can be produced.

Sometimes, in linear electron accelerators, electrons back-scattered from a target can be back-accelerated by the standing wave field used to produce the forward beam.

3. Induced Radioactivity

Radioactivity induced in various parts of a machine by the beam or scattered radiation may present a hazard to personnel during maintenance work. The greater part of the exposure received by accelerator workers can be attributed to the induced radioactivity in the accelerator components or structure.[14] The shielding, targets, vacuum chamber, magnets, rf cavities, cooling water and air in the accelerator room may become radioactive to some degree. At high energy accelerators, radionuclides which have Z and A equal to, or less than, the sum of these numbers for the target and the projectile may be produced.[30] The activities thus produced have a wide range of half lives, but some are so short that they are insignificant with respect to protection considerations. Since materials used in accelerators are limited, the most important materials with respect to activation are iron, stainless steel, copper, aluminum, concrete and plastics. Table 17.1, from Reference 14, lists a number of pertinent radionuclides produced in these materials. Most of the products result from ordinary nuclear reactions although some are from spallation, fragmentation, or capture reactions.[14]

Table 17.1 Radionuclides Commonly Identified in Solid Materials Irradiated Around Accelerators*

Irradiated Material	Radionuclides (half life in parentheses)
Plastics, oils	^{7}Be (54 Cd), ^{11}C (20 min)
Concrete, aluminum	As above, plus: ^{22}Na (2.6 y), ^{24}Na (15 h), ^{32}P (14.3 d), ^{42}K (12 h), ^{45}Ca (165 d)
Iron, steel	As above, plus: ^{44}C (3.9 h), ^{44m}Sc (2.4 d), ^{46}Sc (84 d), ^{47}Sc (3.4 d), ^{48}V (16 d), ^{51}Cr (28 d), ^{52}Mn (5.6 d), ^{52m}Mn (21 min.), ^{54}Mn (300 d), ^{56}Co (77 d), ^{56}Mn (2.6 h), ^{57}Co (270 d), ^{58}Co (72 d), ^{60}Co (5.3 y), ^{57}Ni (36 h), ^{55}Fe (2.9 y), ^{59}Fe (45 d).
Copper	As above, plus: ^{65}Ni (2.6 h), ^{61}Cu (3.3 h), ^{64}Cu (12.8 h), ^{63}Zn (38 min), ^{65}Zn (245 d)

*Taken from ICRU Report 28 - Basic Aspects of High Energy Particle Interactions and Radiation Dosimetry.

In contrast to the activation in high-energy accelerators, some accelerators do not have a significant problem with activation.[19] With the exception of neutron generators, which always present the potential for activation, other accelerators below 5-10 MeV do not have significant activation. Table 17.2, from Reference 27, points out the relative importance of induced activity in low-energy accelerators.

TABLE 17.2 Potential Induced Radioactivity in Particle Accelerators

Particle	Energy Range	Induced radioactivity in Target	Vicinity
Electrons	below 1.67 MeV	none	none
Electrons	1.67 to 10 MeV	limited	very slight
Electrons	above 10 MeV	probable	suspect
Protons, Helium ions	below 1 MeV	limited	none
Protons, Helium ions	1 to 10 MeV	limited	suspect
Deuterons, tritons	any energy	limited	suspect
All ions of light atomic weight	above 10 MeV	probable	suspect

For accelerators in which induced activation occurs, the general behavior of the radiation field is an initial rapid drop (approximately a factor of 2) in the reading due to decay of very short-lived products, followed by a similar drop in the reading within 2 days. In some cases, this fact may be useful in scheduling repairs or maintenance in order to reduce exposures.

In addition to the radionuclides in structures as indicated in Table 17.1, activation of air may lead to concentrations of ^{15}O, ^{13}N, ^{11}C and ^{41}Ar at a majority of accelerators.[14] At some facilities, ^{7}Be and ^{3}H are also important. These concentrations do not persist for long periods following shutdown of the accelerator, and the short half lives of the major products also act to reduce the potential hazard.

Of the activities induced in cooling water, ^3H, ^7Be, ^{11}C and ^{15}O have been found to be the more prevalent radionuclides.[14] As pointed out previously, ^3H may be collected by cold traps and counted by liquid scintillation methods. Air samples containing ^7Be or ^{11}C can be analyzed in conventional counting systems or by gamma spectrometry.

4. Skyshine

Skyshine is another problem to be considered. The term "skyshine" refers to secondary radiation, which first proceeds upward and is then scattered back toward the earth's surface after one or more collisions with air nuclei.[31] Initially described by Lindenbaum in this manner, the term has been extended to include radiation which proceeds downward and is ground scattered back to the surface. In our case, the term refers to the air scattered component which produces a radiation level at some distance from the accelerator. The need for roof shielding became apparent when high radiation levels were observed near accelerators, such as the Cosmotron and Bevatron. These machines were originally constructed with no roof shielding, and when high intensity operations were attempted it was found that roof shielding was needed.[32] Roof shielding may be needed over an entire accelerator or only over the target area, depending upon the intensity and modes of skyshine production. Such shielding is very important for the high-intensity, high-energy machines. Studies at GeV accelerators have shown that the skyshine drops off inversely with distance; the decrease becoming faster beyond about 150 m. The required shielding over a GeV machine may be comparable to the required thickness of a side wall shield.

In their review of skyshine experiments, Rindi and Thomas[32] reached several conclusions: the radiation intensity decreases at least as fast as r^{-2}, the neutrons are the dominant radiation at large distance r, the neutron spectrum is in equilibrium for well-shielded accelerators in the GeV region, and the skyshine intensity (n/m^2s) can be estimated from

$$\phi(r) \sim \frac{a\,Q\,e^{-r/\lambda}}{4\pi r^2}, \qquad 17.5$$

in which Q is the neutron source strength, n/s, r is the distance from the accelerator to the point of interest, in m, a is an empirical buildup factor and λ is an attenuation parameter, in m. Values of a and λ for several experimental measurements can be found in Reference 32.

5. Klystrons

High-voltage klystrons are used for the rf power source for some of the Linacs. They operate at both high voltage (100-250 kV) and high current (100-300 A). These tubes do generate very high x ray emission since e^- beams are accelerated by the high voltage. A shielding thickness of 25-50 mm of Pb has usually been found adequate to protect personnel.[23] Because of the irregular shape of the klystron, care must be taken in the placement of the shielding to avoid leakage of x rays. Surveys to detect leaks (preferably using some passive detector) should be employed.

D. Special Considerations

Ozone, a toxic gas, is produced in addition to radioactive gases in irradiated air.[23] This and other noxious gases can be formed by any type of ionizing radiation. Ozone production is usually negligible during x ray therapy or radiography because little of the e^- beam power is converted to x rays. On the other hand, direct electron beam therapy could result in hazardous ozone production. The concentration of the gas will depend upon such factors as beam power, length of operation time, target substance, and ventilation rate. It may be necessary to supply local exhaust ventilation to quickly remove the gases.

Tritium and other radioactive materials are often used as targets. In particular, tritium is used extensively in small neutron generators and can lead to some radiation concerns.[33,34] When used as a target, ^3H gas is released from the target surface and can end up in pumps, adsorbed on the inner surfaces of the accelerator tube, and in the vacuum system.[3] For this reason, it is customary to assume the entire system becomes tritium contaminated and proceed accordingly. Also, tritium targets should always be stored in sealed containers, in well ventilated areas.

The shielding initially erected around some of the machines was found to be inadequate as beam energies and intensities were increased.[17,31] More shielding was added or, in some cases, the original shielding was replaced by higher-density substances. These required changes were costly and inconvenient or, in some cases, unsatisfactory. The added shielding can cause space problems or overburden building foundations.

The design of shielding for high-energy machines is now generally considered a vital and costly factor in the planning stage. Since these machines are pure research tools, a flexible shielding design is often required to account for future developments in the experimental program or increases in beam intensity.

Each accelerator facility will need a safety system which will depend upon the type of accelerator, its use, and the details of the particular site.[27] The needs of a particular facility will also be a function of the number of beams which are available. The objective of the safety system is to prevent injury or damage by radiation beams. This requires the use of fail-safe devices and passive elements in the system.

Accelerator controls, interlocks, and warning devices are all elements of the safety system. Such devices will prevent the unauthorized production of radiation and protect personnel. Although each system must be individually designed, Reference 27 discusses the general features of a desirable safety system.

E. Radiation Protection Surveys

Many surveys are required during the first stages of an operating accelerator. The type and intensity of the radiations may vary quite a bit with changes in machine operation. Therefore, frequent surveys are made of the environment to assess the hazard in all areas under various machine conditions and to assess the integrity of the shielding. Studies have shown that the mixed fields around a large machine include protons, neutrons, pions, gamma rays, electrons, and muons. However, neutrons are the main component of these fields. Behind thick shields, more than 50% of the total dose equivalent is caused by fast neutrons (up to ~ 10 MeV).[14,16] The balance of the dose is due to thermal neutrons, gamma rays, and high-energy particles.

Among the types of surveys required are:

(1) During tune up and initial operation - Fast neutron and γ radiation fields should be measured.

(2) During routine operations - Conditions may be different from the tune up phase, so measurements should be made of the radiation field in occupied areas, the skyshine component, and residual radioactivity in targets, accelerator parts, and experimental equipment.

(3) For experiments which require unusual source or target conditions.

(4) Following operational changes - Type and intensity of radiation may vary following changes in power level, experimental arrangements, or shielding.

(5) During target removal following bombardment.

(6) During maintenance operations - The beam may strike any part of the internal structure so that activation could be found in unsuspected places. Most of the exposure received at accelerators occurs during maintenance when the accelerator is shutdown.

(7) Periodic surveys of airborne particulate matter should be made to assess potential airborne hazards.

(8) Routine smear surveys should be carried out to assess the degree of contamination control.

(9) During decontamination - Surveys of the cleanup during progress as well as the final results should be accomplished.

(10) Special surveys - Performed on request from experimenters and others.

To assess the exposure hazard, measurements that can be converted to units of dose equivalent, H, need to be made (see Section 4.A.5). The product of an absorbed dose and a suitable Q expresses the irradiation in terms of a common scale for all ionizing radiations. As shown in Section 4, Table 4.1, the Q values are related to the LET of a particle of initial energy E. When the particle moves through a medium, the energy changes and so the LET changes. When the radiation field is composed of a number of particles with a spectrum of energy, the quality factor must be an average or effective value. This is the situation one usually encounters when trying to determine $\overline{Q}$ for a neutron field.

In many cases, the health physicist must evaluate the hazard due to a mixed field of neutrons and γ rays. When the neutron energy spectrum is known, the dose equivalent, H, due to this component may be obtained from the product of the measured absorbed dose and the appropriate Q. Values of Q for monoenergetic neutrons as a function of neutron energy up to 400 MeV are contained in Section 4, Table 4.3. If the spectrum information is lacking, an upper limit for the dose equivalent can be taken as 20 times the sum of the neutron doses, plus the γ dose. A simpler but more conservative approach is to measure the total absorbed dose and use a Q of 20.

When the energy spectrum of the neutron field is known, an appropriate value of H may be computed by the use of conversion factors (see Figure 17.10). The conversion factor g(E) is the fluence rate of particles of energy E that produces a maximum $\dot{H}$ of 10 μSv/h (1 mrem/h). Then, for a spectrum of energy,

$$\dot{H} = \int_{E_1}^{E_2} \frac{\phi(E)\, dE}{g(E)}, \qquad 17.6$$

in which $\phi(E)$ is the differential fluence rate of neutrons between E_1 and E_2, and $g(E)$ is the appropriate conversion factor for that energy region. In many instances one knows only a crude spectrum, such as

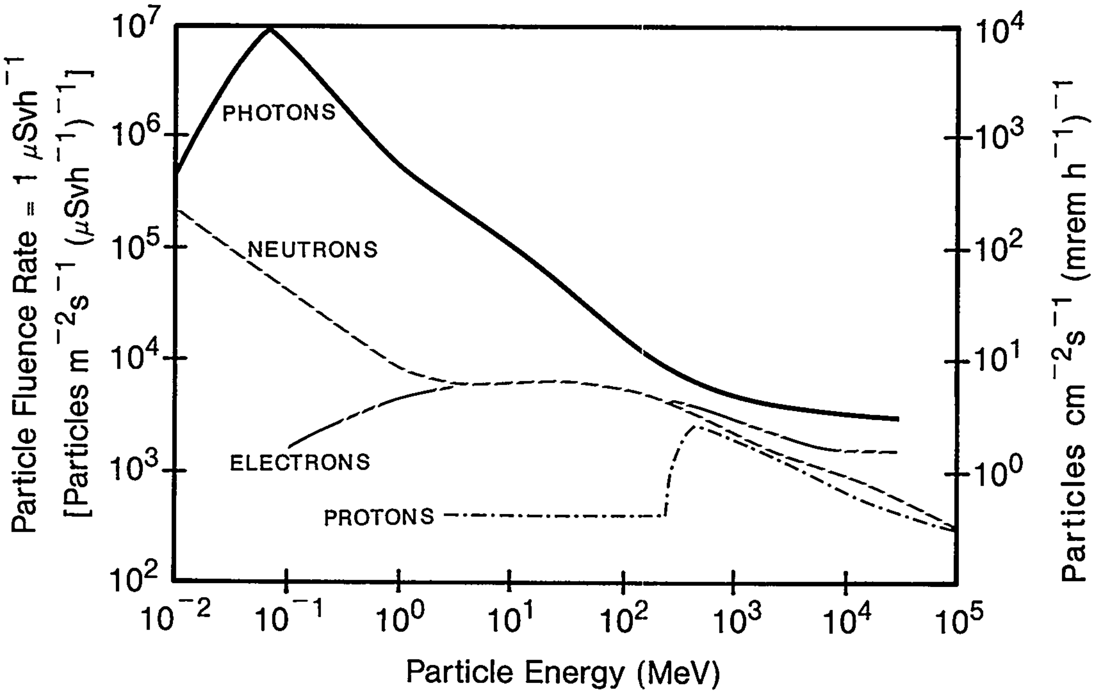


Figure 17.10 $\dot{H}$ conversion factors as a function of energy for different particles. (From Patterson and Thomas, Reference 19, Copyright 1973, used by permission of Academic Press).

the total fluence rate in several energy intervals, so that the dose equivalent can only be approximated by:

$$\dot{H} = \sum_{i=1}^{n} \frac{\phi(E_{i+1}-E_i)}{g(E_{i+1}-E_i)}, \qquad 17.7$$

in which $\phi(E_{i+1} - E_i)$ is the total fluence rate in the energy range $E_{i+1}-E_i$, and $g(E_{i+1}-E_i)$ is the average conversion factor in the same region. The γ dose equivalent component can be determined by an ion chamber measurement.

For a well-shielded accelerator, the radiation field may generally be considered as consisting of low-LET components (Q=1) and neutrons with a broad spectrum of energies.[14] The neutron component can then be determined by using thermal neutron detectors (BF_3, LiI, ^{6}LiF or activation detectors) in 0.05-.3 m diameter moderators (see Section 12.B.4.c, 12.B.4.e, 12.B.5.d, 12.B.6.b). These devices give a response which is close to the neutron dose equivalent and are referred to as neutron dose equivalent meters.

Another method which may be employed is based upon a gas-filled, TE proportional counter used as a detector and special electronic processing to determine both the absorbed dose rate and the dose equivalent rate.[19] From the determination of H and D, the ratio H/D gives the mean value of the quality factor $\overline{Q}$. Other means of estimating $\overline{Q}$ for a neutron field with a spectrum of energies is to use two instruments, one which is energy dependent in its response and the other energy independent. From the response ratio, one is often able to determine an effective energy and thus arrive at an appropriate $\overline{Q}$.

For situations in which one can reasonably expect the neutron spectrum to be constant, a Rossi LET spectrometer[19] can be used to determine the value of $\overline{Q}$ at a given location. Then, subsequent measurements with a TE chamber will determine the absorbed dose and by application of $\overline{Q}$, H can be found.

F. Radiation Survey Instruments

For many of the lower-energy accelerators, conventional survey instruments, such as used around reactors, are used (see Section 12). The type of radiation and wide range of energies encountered near the higher-energy machines may require a family of survey instruments.[19]

1. Survey Devices

A tissue equivalent (TE) ionization chamber may be employed to measure the absorbed dose. The TE chamber reading is independent of the type and energy of the radiation. A graphite-air ion chamber has been used with the TE chamber to obtain an estimate of the absorbed dose due to gammas and charged particles (protons, pions, etc.) plus that due to neutrons.[14,35] However, it is not easy to resolve the dose components. Among the problems in this method is the fact that the graphite chamber also responds quite well to neutrons. Although this twin-chamber method lacks the desired accuracy, it can provide an indication of the nature of the radiation problem.

Proportional counters with plastic walls have been used to measure neutron dose in the presence of γ rays and electrons.[36] Voltage pulses are amplified and added, and the resulting meter reading is proportional to the energy absorbed in the gas-filled unit.

The spherical neutron foil monitor, composed of thermal-activation foils within a paraffin sphere, has been employed for neutron survey purposes.[37] Fluence, dose equivalent, and average neutron energy can be determined with this unit (see 12.B.6.b). A drawback of this unit is the time delay; the foils have to be removed and counted to obtain survey results. To overcome this problem, G-M detectors have been used in place of the foils. The G-M probe is wrapped in silver foil, and the activity induced in the foil is counted between beam pulses.

The Bonner multisphere system can be used to determine the spectrum shape, fluence information and the dose equivalent. A number of spheres are used in conjunction with the LiI detector - 5.08 cm, 7.62 cm, 12.7 cm, 20.32 cm, 25.4 cm, 30.48 cm, and 45.72 cm. Readings are also obtained with the bare and Cd-covered detector. The resolution obtained by this system is poor and the data needs to be unfolded by computer analysis to arrive at the spectrum. In addition, the complex operation of the system precludes its operation as a routine survey device. This system can be used to estimate $\overline{Q}$ in a given location.

The fluence rate of high-energy neutrons (>20 MeV) has been estimated from a measure of the ^{11}C activity induced in a plastic scintillator.[14,19] The 20.3 min activity, resulting from the $^{12}C(n,2n)$ ^{11}C reaction (22 mb reaction cross section), is counted in a photomultiplier system and converted to units of fluence rate. ^{11}C is also produced by (p,pn) and (γ,n) reactions. However, in the applications outside shielding, the radiation field producing the ^{11}C is essentially neutrons.

Scintillation crystals have also been used for neutron monitoring. Plastics with a high hydrogen content are loaded with phosphors, and the light produced by proton recoils is measured with a photomultiplier unit. The crystal discriminates against x and γ rays.

Neutron spectra and total fluence above a certain energy can be roughly estimated with threshold detectors. The activity produced in foils through neutron-induced processes provides the data regarding regions of the neutron spectrum. For energies above 1 MeV, sulfur, aluminum, carbon and bismuth, with thresholds of approximately 2.5, 6, 20 and 50 MeV respectively, are used.

2. Performance Limitations

Personnel engaged in survey work around the large machines should be aware of the performance limitations of survey devices.[38] High dose rates during pulses may cause ion chamber instruments to operate at less than saturation voltage, and underestimate readings. If enough high voltage is supplied, the effect will be much reduced. Also, the chamber can be calibrated in the pulsed field. Pulse counters are limited by coincidence losses as the pulse length becomes shorter. If a radiation detector is influenced by variations which occur in time >1 ms, then the duty cycle describes beam variations. The duty cycle is the fraction of time in which the useful beam is delivered,[19] and may be expressed by

$$\text{duty cycle} = (\text{pulse width})(\text{repetition rate}) = \Delta tf. \qquad 17.8$$

If the response occurs in $\Delta t < \sim 1\ \mu s$, then variations caused by the rf must also be considered. Detection will suffer if the counter has a long resolution time. The actual count rate, R, will be related to the observed count rate, R_o, by[19]

$$R = \frac{Ro}{1-\frac{\tau Ro}{f\Delta t}}, \qquad \Delta t \gg \frac{\tau Ro}{f} \qquad 17.9$$

in which τ is the resolution time of the counter. From equation 17.9, it is seen that the count rate is underestimated and the error will be largest for small duty cycles. For this reason, a detector which integrates the pulses is often used.

The rf which is used at some accelerators may also be a problem. Unless properly shielded, the rf may be picked up by the instrument's circuit and registered as erroneous readings.[39]

Magnetic fields also affect the response of some instruments. Rate meters should not be used in strong magnetic fields.[39] Magnetic shielding of photomultipliers in scintillation counters is often required to prevent magnetic effects on the photomultiplier efficiency.[38]

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QUESTIONS

17.1 For what type of studies are particle accelerators particularly suitable?

17.2 What value do particle accelerators have other than that of being research tools?

17.3 What is the basic purpose of the Cockcroft-Walton machine?

17.4 What are the advantages of the Cockcroft-Walton machine

a) in general and
b) relative to synchrotrons and cyclotrons?

17.5 What advantage was gained from

a) the development of the Van de Graaff electrostatic generator over the Cockcroft-Walton machine and

b) the development of the tandem Van de Graaff electrostatic generator over the original Van de Graaff design?

17.6 On the basis of the technique for propelling particles, how do the linear accelerators differ from the Cockcroft-Walton machine and Van de Graaff machines?

17.7 Why are the tubes near the high energy end of an electron linear accelerator nearly the same size, while tubes for accelerating protons and heavy ions increase in length from the low to the high energy end?

17.8 How do the cyclotron, synchrotron, betatron and similar accelerators retain the particles until they have gained the energy desired? How would this retention technique work with neutrons?

17.9 What limits the practical aspect of accelerating electrons in the betatron or any other electron accelerator?

17.10 How do the cyclotron and the synchrotron differ in operation? (Consider the magnetic fields and the particle paths.)

17.11 How does the rate and mode of accelerated particle emission differ between the synchrotron machines and the earlier accelerators.?

17.12 What is synchrotron radiation?

17.13 What are the characteristics of synchrotron radiation?

17.14 How does a "light source" differ from synchrotron radiation from high energy machines?

17.15 What is a "wiggler"?

17.16 What is the mechanism of producing pulsed neutrons? For what main purpose are these neutrons used?

17.17 What term refers to the branch of physics that deals with elementary particles and the forces involved in their mutual interactions? What more appropriate term could be used?

17.18 What are "leptons"? How many types of "leptons" have been identified? What are their common physical characteristics?

17.19 What are "hadrons"? What are they believed to be made of? What are their common physical characteristics?

17.20 What are the radiations of concern in high energy processes?

17.21 What term is used to indicate the particles knocked out of a nucleus as the result of bombardment by accelerator particles?

17.22 What terms are used to indicate the processes in which smaller bits of matter including nucleons are emitted as the result of bombardment by accelerator particles?

17.23 Why do mesons constitute a radiation hazard?

17.24 Explain the "knock-on" and the "evaporation" phase in accelerator particle interactions?

17.25 What particle is produced in almost all types of particle accelerators?

17.26 Why are the original particles produced by an accelerator beam called cascade particles?

17.27 Define radiation length.

17.28 Explain photodisintegration.

17.29 Why is it impossible to produce a free π meson with an accelerator particle of 100 MeV?

17.30 How may induced radioactivity be a hazard inside a shielded accelerator?

17.31 What nuclides are of concern as air activation products and water activation products?

17.32 Explain the term "skyshine."

17.33 When and why may accelerator beam targets be a hazard?

17.34 Why may the power source for an accelerator be a radiation hazard even though it has no radioactive materials and is completely isolated from the accelerator beam and its products?

17.35 When thick shields are used, what radiation constitutes the largest component of the total dose that may be received?

17.36 When should radiation surveys for particle accelerators be initiated?

17.37 When information concerning the neutron spectrum produced in accelerator work is not available what quality factor should be assumed?

17.38 Why are the conventional radiation instruments limited in their ability to assess the radiation hazards of particle accelerators?

17.39 What toxic and radioactive gases may be produced during the operation of a particle accelerator? What is the source of these gases?

17.40 How can high energy neutrons be detected in a spectrum of neutron energies?

PROBLEMS

17.1 Starting from equation 1.25 ($r = \frac{mv}{Be}$), derive an expression for kinetic energy (non-relativistic) for positive ions.

Answer: $E_k = \frac{e^2 r^2 B^2}{2m}$

17.2 A 100 eV electron (mass = 9.11×10^{-31} kg) is circulating in a plane at right angles to a uniform magnetic field of 10^{-4} tesla. Compute

a) the velocity of the electron,
b) the radius of the trajectory of the electron

Answer: a) 5.93×10^6 m/s
b) 0.338 m

17.3 Find the orbit radius of 20 MeV proton in a field of 1 tesla.

Answer: 0.65 m

17.4 A beam of 10^{12} electrons of energy 7 GeV, bent into a circle by an effective magnetic field of radius of curvature of 160 m. What is the power lost in watts? (Hint: Assume the electrons are moving at the speed of light.)

Answer: 63.5 kW/turn

17.5 a) A 6 GeV electron beam is to be accelerated by a magnetic field such that the critical energy of the bremsstrahlung would be 16 keV. What should be the radius of curvature of the magnetic field?

Answer: 29.9 m

b) What is the approximate emission half-angle in degrees of the synchrotron radiation for the above problem?

Answer: 4.77×10^{-3} degrees

17.6 A 270 MeV n-beam intensity is to be reduced by a factor of 10 using iron. What is the minimum thickness of iron needed for this purpose? (Atomic weight of Fe=55.85, density=7700 kg/m^3).

Answer: 0.4 m

17.7 A 50 MeV proton beam is stopped by a thick copper target. The neutron fluence rate at 1 m is measured to be 1.06×10^9 n/m^{2}s. The average energy of the neutrons measured is approximately 1 MeV. Estimate the neutron dose rate at 10 m. (Hint: use Table 4.3 in Section 4).

Answer: ~ 1.4×10^3 μSv/h

17.8 Compute the annual skyshine dose to an individual member of the public due to the high energy neutrons from a well shielded positron synchrotron light source given the following parameters and data:

Electrons in the ring:	4.5×10^{12}
Neutron production rate:	0.12 n/e
Beam life time:	10 hours
Distance to the fence:	500 m
Constants:	a: 7
	λ: 330 m
Operation time:	6000 h/year

Assume a conversion factor of 3×10^3 n/m^{2}s = 1 μSv/h and the roof shielding is 2 attenuation lengths.

Answer: ~ 2 μSv.

SECTION 18 - X RAY PRODUCING MACHINES AND SEALED GAMMA SOURCES

A. Introduction

In Section 8, the basic principles of external radiation protection were introduced. These principles are time, distance and shielding. In that section, the topic of shielding was discussed in some detail. In most cases, with respect to radiation producing machines, the shielding is already incorporated into the system. This is not true for sealed γ sources, such as those used in radiography. In these cases, the source is usually delivered in a shielded container and must be stored in a shielded container. The basic difference between an x ray radiation producing machine and a sealed source is that when the machine is not running, no radiation is produced. For a sealed γ source, which incorporates a radionuclide of a certain half life, the radiation is emitted continuously so that one must be continually concerned with providing the necessary shielding.

With respect to machines which produce x rays, the manufacturer provides what is deemed to be adequate shielding of the device before it is sold. Presumably then, one acquires a machine or device which gives adequate reduction in the leakage radiation field to meet the appropriate standards. That is, the manufacturer is required by the regulations to provide adequate shielding. However, prudent practice requires that any device or source which involves radiation, and is supposedly shielded for that radiation, be surveyed to determine the adequacy of any shielding which is supplied. Also, one must survey containers which house sealed γ sources to be certain that the shielding afforded by the container is adequate to reduce the radiation field to an appropriate level.

This section deals with x radiation producing machines or devices as well as sealed γ sources used in radiography or other applications with respect to the radiation protection aspects involved in the use of these radiation devices and sources.

B. X Ray Machines

In order to provide x rays efficiently, certain requirements must be met.[1,2] These include a source of electrons, which are accelerated through a large potential difference and allowed to strike a target material which absorbs the electrons. To accomplish this acceleration, the above components must be enclosed in an evacuated chamber or envelope.[1] As discussed in Section 3.B.3, a high energy electron which is stopped in a given medium loses a part of its energy in bremsstrahlung production. The x rays which are produced have a spectrum of energies from 0 up to the kinetic energy of the electron. In addition to the continuous distribution of x rays (see Section 3.E), characteristic x rays may also appear in the spectrum (see Figure 3.9). The magnitude of the bremsstrahlung produced and the energy of the characteristic x rays depend upon the absorbing material. Both of these factors increase with the Z of the material. By adding absorber material, called filters, to intercept the x rays produced in the target of an x ray tube, the energy spectrum shape can be altered. These filters preferentially absorb the lower energy photons, and can produce a peaked spectrum which may be useful for energy calibration of detectors.[3]

1. Production

In 1913, Coolidge made an improved x ray tube which formed the basis of modern x ray tubes.[4] A typical conventional x ray tube design is shown in Figure 18.1. A heated filament (usually tungsten) boils off electrons, which are repelled by the negatively charged cathode.[2] The electrons are focussed by a focussing cup which is also negatively charged. As the electrons move toward the anode, they gain energy determined by the high voltage applied between the cathode and the anode.

Electrons strike the tungsten target and x rays are produced which exit through a thin window (sometimes Be) shown at the bottom of the tube in Figure 18.1. The intensity any spectral distribution of the x rays

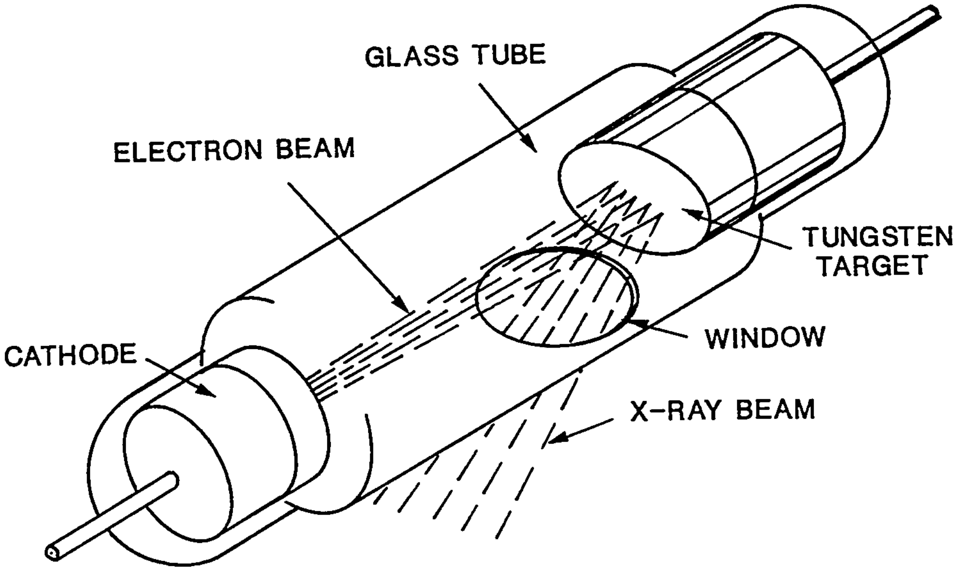


Figure 18.1 Typical conventional X-ray tube.

is influenced by the voltage between the cathode and the anode. To obtain x rays with energy useful for diagnostic purposes, high filament currents and voltages between 40-140 kV must be used.[1] For operating voltages below approximately 70 keV, and tungsten targets, only the bremsstrahlung is produced; no characteristic radiation. One may compute the minimum wavelength λ_{min} of the photons, expressed in units of angstroms, by the use of modified equation 3.31:

$$\lambda_{min} = \frac{12.4}{kVp} A^{\circ},$$

in which the maximum, or peak, voltage (kVp) is expressed in kV.

The glass envelope of the tube provides an enclosed structure which is evacuated. The vacuum is necessary to reduce e^- collisions which would greatly scatter the electrons as well as helping to prevent filament deterioration. By removing air, oxidation of the filament is slowed. When tubes become gassy, the filament deteriorates rapidly.

2. Efficiency

Even though the improvements by Coolidge increased the efficiency of x ray production, most of the electron energy loss in the target appears as heat, not radiation, for voltages in the diagnostic range. The efficiency may be defined as the ratio of the energy converted to x rays divided by the energy delivered to the target. Mathematically, one may estimate the efficiency from[2,5]

$$\text{eff} \sim 1\text{x}10^{-9}\ ZV, \qquad 18.1$$

in which Z is the atomic number of the target (Z=74 for tungsten), and V is the tube voltage in volts.

EXAMPLE: Compute the efficiency for a 70 kV machine with a W target.

$$\text{eff} \sim 1\text{x}10^{-9}(74)(7\text{x}10^{4}) = 518\text{x}10^{-5}$$
$$=0.52\text{x}10^{-2}=0.52\%$$

From the above example, it is seen that most of the electron energy goes into heat. This causes the target to reach a very high temperature. This heat must be conducted away from the target to its surroundings or the target would quickly melt. Several methods have been used to cool the anode so that it does not burn up. Cooling oil, cooling water or air blown over cooling fins have been successfully employed.[1] One other method is to distribute the heat by using a rotating anode in conjunction with an oil bath cooling system. In this application, the anode may be rotated at speeds up to 10^4 rpm.

3. Quality - HVL

Because a spectrum of photon radiation is emitted in an x ray tube, the penetration of the components of an x ray beam will vary. If there were no filtration, or absorption, of the beam, the x ray spectral distribution would be given by Kramers' Rule (equation 3.32), and the distribution would look like the solid line in Figure 3.10a. Any absorber which is present will alter the distribution by preferentially absorbing the lower energy photons. This will lead to a distribution shown by the dotted line in Figure 3.10b.

An unfiltered beam is unsuitable for use either in diagnostic or deep therapy applications of x rays.[2] In the case of diagnosis, the low energy photons will fail to penetrate through the tissue to the recording device (film, screen, other). So, the patient dose will be higher, but no information will be obtained from these low energy photons. In the case of deep therapy, the low energy photons will be absorbed near the surface of the body and will not penetrate to the deep organs. This means that the photons are being wasted on healthy tissue rather than exposing the tissue of the tumor. For the above reasons, x ray tubes are usually supplied with a certain amount of filtration, and for some applications more filtration is added to achieve a desirable effect.[2]

Since the composition of the beam changes with the addition of filtration, the quality (penetrating ability) of the beam changes with depth in a material. To specify the quality of the beam, the half value layer (HVL) in a standard material is used. For energies in the range 120-400 kV, copper is usually the standard, while below 120 kV, aluminum is used.[2] Lead is generally used from 1-50 MV.[1] A typical curve of exposure rate versus filter thickness is shown in Figure 18.2. Although the curve is plotted on semi-log paper, a straight line is not obtained. This happens because the quality (or spectral makeup) of the beam changes with depth as the lower energy photons are removed. For thick enough filtration, the curve flattens out and approaches a straight line, which is what occurs when one uses a monoenergetic source. At these thicker filtrations, essentially all of the lower energy components have been removed, and the beam is then nearly mono-energetic.

The HVL for a given choice of filtration can be determined from the experimentally determined curve. Suppose one wishes to operate the

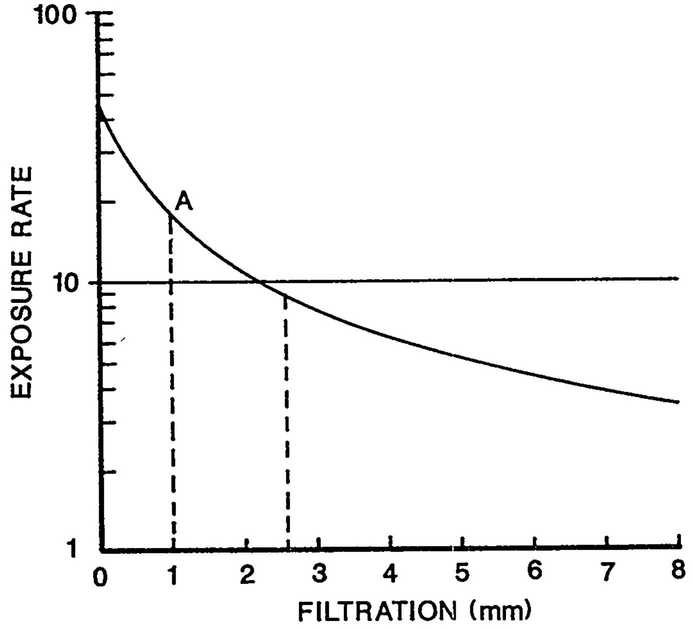


Figure 18.2 Measured attenuation curve for an x-ray machine.

machine at the exposure rate obtained with 1 mm of added filtration. The exposure rate for that filtration is about 18 (point A in Figure 18.2). The thickness at which the exposure rate drops to one-half this value (9) is about 2.6 mm - 1.0 mm = 1.6 mm, for operation of the machine with 1 mm added filtration. The half value layer, together with the exposure rate at a particular location, allows determination of the absorbed dose rate at other sites in an irradiated medium.[1]

4. Analytical X Ray Studies

In addition to diagnostic and therapy uses, x rays are used in analytical studies to examine the microstructure of materials.[6] In these studies, the x rays which are utilized are the characteristic x rays, rather than the bremsstrahlung. Two of the major ways in which x rays are used in these studies are diffraction and fluorescence analysis.[6,7] In both of these methods, the bremsstrahlung will still be an exposure concern.

a. X Ray Diffraction

The atoms in crystalline material have been found to be arranged in symmetric patterns or structures. There exist regular spacings between the planes of atoms in these crystal structures. Moreover, the spacings are close in value to the wavelengths of characteristic x rays emitted by certain target material. By utilizing these x rays to bombard a crystal sample, the scattered (diffracted) x rays which emerge form regular patterns (Bragg reflections) which can be analyzed.[4] The relationship between the lattice spacing of the crystal and the wavelength of the incident x ray, can be related to the angle at which the photon is scattered. Since the scattering pattern from the crystal is characteristic of that material, this method can be used to identify certain crystalline materials. In addition, changes in structure due to compound formation can also be used to identify two slightly different compounds.

The method may be applied in a number of ways.[8,9] A beam of x rays can be directed at a thin sample and the scattered x rays can be detected on a photographic plate behind the sample. For thick samples, the x rays which are back-scattered are used. Even powder specimens have been used with electronic detection of the diffracted x rays.

Diffraction studies require a well-regulated source of nearly monoenergetic low-energy x rays. Typical diffraction tubes operate at 25-50 kVp,[8] which may be operated at 10-50 mA at the lower voltages. The tube is often supplied with a very thin Be window, with little or no filtration, which allows large fluence rates of "soft" x rays to pass through the window, resulting in radiation fields up to > 1 kGy/min (> 10^5 R/min) at the tube housing port.[10] Beam dimensions at the sample may be significantly less than 10^{-4} m^2. These devices may be supplied with as many as four exit ports which may be used simultaneously. A number of different target materials are used, and these are listed in Table 18.1, obtained from Reference 6, along with the characteristic energies of the target emissions. Copper is the most popular target used.[8]

Table 18.1 Characteristic X Rays from Common Diffraction Tube Targets

Target	Characteristic X Rays, keV K_α	K_β
Copper	8.01	8.92
Molybdenum	17.5	19.6
Iron	6.41	7.07
Tungsten	8.40*	9.57**
Chromium	5.42	5.94
Nickel	7.48	8.24
Cobalt	6.94	7.65
Silver	22.2	25.2

*L_α **L_β

b. Fluorescent X Ray Spectroscopy

This analysis is used to determine the elemental composition of a given sample. The methodology used is to irradiate a sample with a high intensity x ray beam in order to excite the characteristic x rays (fluorescence x rays) of the atoms in the sample. The resultant x rays may be analyzed by using a system which detects the energy of the x rays. Alternatively, the x rays from the sample may be directed to a single crystal x ray diffractometer which will analyze and record the wavelengths of the photons. Since high intensity beams are used, at energies above those used in diffraction units, these devices are normally completely enclosed and interlocked. Although this should be an increased safety factor in the case of these instruments, there is a tendency for individuals to attempt to circumvent interlocks, so that increased safety may not be achieved.

Typical acceleration potentials for fluorescence analysis are in the range of 25-100 kVp. Energies less than 5 keV are unimportant because these photons are easily attenuated. So, the bremsstrahlung continuum can be assumed to extend from 5-100 keV, with an intensity maximum in the range 20-30 keV. Superimposed on the continuous spectrum will be characteristic x rays, which constitute less than half the diffraction tube output.[10]

5. Sources of Radiation Exposure

A number of potential sources of radiation are associated with the operation of various x ray machines.[10] These include:

(1) The primary beam.

(2) Leakage radiation through defects in the device.

(3) Penetration of the primary beam through the device.

(4) Secondary emission from material exposed to the primary beam.

(5) Scattered radiation.

In all machines, the primary beam represents the major potential hazard because of the intensity and energy in the beam. Usually the primary beam is adequately shielded and no problem exists, but any defect in the equipment could lead to the release of hazardous amounts of radiation. For x ray diffraction machines, both the primary, as well as the diffracted beams, are small in size and well-collimated. Although photons in a diffraction beam can be easily shielded, they also represent a potential for severe exposures. These photons are also easily absorbed in tissue and; therefore, can lead to very high doses in a short period of time. Leakage which is due to cracks or small openings also lead to small, but intense beams. Because of the small extent of these beams, they may be missed in general surveys if extreme care is not exercised. As was mentioned above,the primary radiation in a diffraction device can produce a radiation field in excess of 1 kGy/min (approximately 10^5 R/min), but one should be aware that the <u>diffracted</u> beam <u>can</u> be on the order of about 0.7 Gy/h (approximately 80 R/h). In addition, the scattered (diffracted) beam may appear at almost any angle with respect to the main beam.[10] With regard to diagnostic and therapy machines, most applications require that a protective barrier be between the operator and the device.[11] That is, the radiation field which is produced by these devices is such that the operator needs to be isolated and protected while the device is in use. The use of lead aprons and shields is recommended, especially for medical fluoroscopy which represents the largest potential source of occupational dose due to the use of x rays.

6. Safety Devices

In addition to the engineered safety mechanisms, such as leak proof joints, collimators, beam traps and safety shutters, a number of other safety approaches and/or safety devices are generally employed to enhance the safety aspects of x ray equipment. These consist of such things as flashing lights to indicate when the primary x ray beam is on. Warning signs, or interlocks, are used on switches to prevent unauthorized personnel from operating the equipment. Either physical guards, or interlocks, are used to prevent entry of parts of the body into the primary beam path. Shutter status lights or indicators which automatically indicate the position of each shutter will prevent inadvertent operation with shutters open. Warning signs, located on tube housings, placed near operating switches and at prominent places in an exposure room, help to keep personnel and others aware of the potential hazards of the equipment. Securing of unused tube ports in a closed position in a way which will prevent casual opening will also tend to reduce incidents involving exposures from unused ports which are inadvertently opened. Since research projects may involve frequent modifications of analytical x ray equipment, there is a potential for increased radiation levels following these modifications. Care is needed, and surveys should be made following these adjustments, to ensure that no new beam leakage paths been introduced.

Interlocks should be such that if the beam is shut off when the interlock is activated, the operating switch at the control panel will need to be reset before operation can be resumed. Also, all safety devices require periodic testing to insure proper operation.

7. X Ray Survey Instruments

Surveys of medical x ray installations often involve the use of beam measuring instruments to determine the exposure rates obtained for different tube voltages. These types of measurements involve precision instruments and require that the response be reasonably energy independent above about 20 keV.[12] For these surveys, an ion chamber, such as a condenser chamber or precision dosimeter, is usually the instrument of choice. These chambers can be designed for energy below 150 keV, but can also be used for high energies if supplied with a build-up cap. For medical x ray facilities, quality assurance measurements which are required include exposure reproducibility, beam quality, accuracy of mA and exposure time, table top exposure rate and kVp. Approaches to the survey of medical fluoroscopy units are discussed in References 12-16.

For health physics surveys of diffraction units and/or fluorescence devices of the analytical type, the energy dependence of the instrument may be the most important factor affecting the accuracy of the measurement.[10, 17,18] Ion chambers with a thin window (approximately .01 kg/m^2) can be used for low energy x rays when accurate exposure rates need to be measured. Some ion chambers respond correctly down to the range around 20 keV, but are inaccurate below this. As mentioned in 18.B.4.b, the x ray energy range of concern is effectively from 5-100 keV, with the maximum intensity in the 20-30 keV range. For diffraction devices which have iron or chromium anodes, it may be necessary to obtain calibration factors in order to correctly read exposure rates at the low end of the range. For other cases in which the detection of the radiation is more important than knowing the actual value of the radiation field, end-window GM survey meters or thin scintillation counters may be used. The audible capability, either speaker attachment or headset, should be used in these surveys. Providing a metal cover with a small hole in it will aid in locating the origin of leaks.[18]

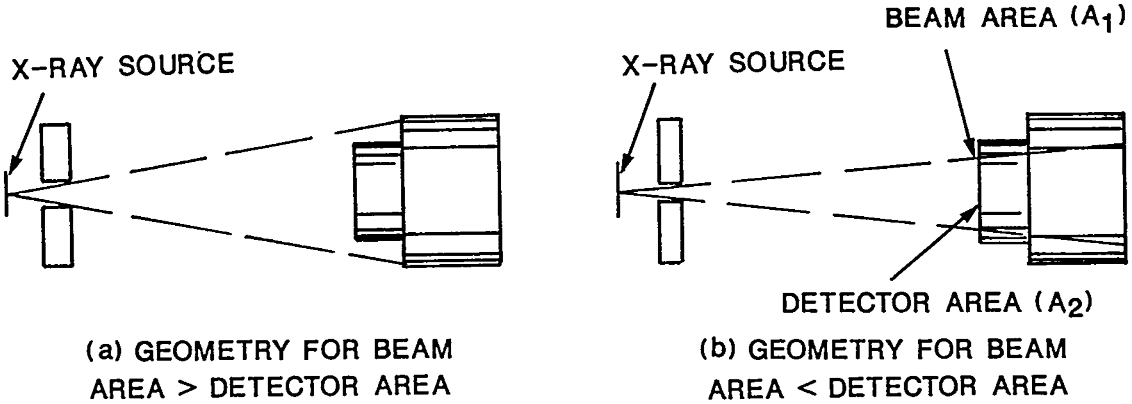
Actually, a combination of instruments, such as a GM survey meter and an ion chamber instrument, may be the better approach.[18] The sensitivity of the GM allows easy detection of beams of small cross sectional area ($< 10^{-4}$ m^2), which may be emitted from diffraction equipment. Once one finds these leaks, it is often convenient to eliminate the radiation by added shielding. For this reason, determination of the exposure rate is of little importance. For larger beams, the energy dependence of the GM will preclude its use to accurately determine the radiation field.[10]

Both thin-window ion chamber dosimeters and survey meters are used for cases in which it is necessary to determine the exposure rate. For determination of beam exposure rates, difficulties may be encountered because of the small beam size. This may dictate the use of TL devices to adequately measure the radiation field. Sometimes, one may be able to back off to a certain distance, so that the ion chamber is completely within the beam, to make the measurement. For low energy x rays then, one will need to correct for attenuation by air in the intervening distance. Operational checks of thin-window ion chambers for response to low energy x rays can be made by using ^{55}Fe sources. This radionuclide decays by electron capture, emitting a 5.9 keV x ray.[10]

Since most ion chamber survey instruments have rather large chamber dimensions, one must be aware that this means low sensitivity, especially for beams of small cross sectional area.[10] In addition, the chamber is usually calibrated under conditions in which the entire chamber is within the x ray beam dimensions (see Figure 18.3a). This means that one must correct the reading for the case in which the beam cross sectional area, A_1 in Figure 18.3b, is less than that of the detector chamber (A_2 in Figure 18.3b). This correction factor, f, is given by[10]

$$f = \frac{\text{Detector Area}}{\text{Beam Area}} = \frac{A_2}{A_1}. \qquad 18.2$$

For extremely small beams ($< 10^{-4}$ m^2), it is recommended[10] that the radiation field be averaged over 10^{-4} m^2. This means that the correction factor f in equation 18.2 reduces to the value of the sensitive area of the detector (expressed in cm^2). This is reasonable, since for these very small beams it is unlikely that repeated exposures would involve the same exposed area. X ray film is often used to obtain an estimated size of leakage beams when one is interested in exposure rate measurements. On occasion, fluorescent screens can also be used, but are



(a) GEOMETRY FOR BEAM AREA > DETECTOR AREA

(b) GEOMETRY FOR BEAM AREA < DETECTOR AREA

Figure 18.3 Geometry for calibration and for beam area less than detector area.

limited in application since their low sensitivity requires a rather high intensity of x rays.

C. Electron Machines

In x ray machines, the electron beam is purposely made to produce x rays. In certain machines utilizing electron beams, the production of x rays is incidental to the use of the electron beam. Among the electron machines of concern are electron microscopes and electron beam welders. X rays are produced in these devices when electrons in the primary beam, or back-scattered electrons, strike metal parts of the machines.[5] Since the radiation of concern is x rays, much of what was covered in Section 18.B above, with respect to the production, spectrum and efficiency of x rays, applies to the x rays from these machines. Electron microscopes may be operated in the range of 50-100 kVp, while electron beam welders range up to about 150 keV. Some electron microscopes have been built with acceleration potentials up to 1000 kVp.

Because many different materials are used in the construction of the devices, characteristic x rays may also be produced in materials such as Al, Fe, stainless steel, Cu, brass, Mo and Pt. The apertures of electron microscopes are made of higher atomic number materials (Mo, Pt). However, a larger fraction of the electron beam is more likely to strike the lower Z number materials. The K absorption edges of some of the materials are high (for example, Pb at 88 keV). The significance of this was sometimes overlooked and adequate shielding was not provided. This has led to discovery of leakage problems with some of these units in the past.[5] For this reason, the shielding adequacy of any device should be thoroughly checked before the unit is put into full service.

The more likely sites of radiation leakage are the electron gun, the camera and viewing chamber ports, the specimen chamber and column, and, at lesser frequency, the high voltage tank.[5] Much of the radiation leakage results from back-scattered electrons, rather than from primary beam bombardment.

Survey instruments for these electron beam machines can be the same as suggested for diffraction and fluorescence machines. That is, a sensitive, low energy detector, such as an end window GM survey meter or thin scintillation counter, can be used for location of leaks. An ion chamber device, with a thin window, can then be used to obtain exposure rate readings.

D. Sealed Gamma Sources

A number of sealed γ ray sources are employed in medical, industrial and research applications. Protection aspects for the use of these sources are discussed in general in References 16, 19-22. Medical applications, such as teletherapy machines, utilize ^{60}Co and ^{137}Cs sources which may produce dose rates on the order of 1 Gy/min (100 rad/min) at the patient position. These units are located within shielded treatment rooms, and the operator station is such that negligible radiation is received by the operator. Specifications for the proper maintenance and necessary checks of teletherapy equipment are given in ANSI N449-1974.[23]

In addition to the above uses, radiation therapy is also carried out using brachytherapy sources (mainly ^{182}Ta, ^{192}Ir, ^{125}I, ^{51}Cr, ^{198}Au, ^{90}Sr and ^{252}Cf). In this method, an encapsulated source is used to deliver γ (β, neutron) radiation, either by surface, intercavitary or interstitial application, to limited portions of the body containing tumors. Characteristic of these sources and principles of radiation protection are discussed in References 22 and 24.

Industrial uses of sealed gamma sources include thickness and level gauging, bore hole logging and radiography. Among the radionuclides most frequently used in these applications are ^{60}Co, ^{137}Cs, ^{192}Ir and ^{252}Cf.

Research applications include a variety of studies, including irradiation of animals and specimens, flow gauging, calibration of instruments and nuclear studies. Many different radionuclides are used depending upon the factors being studied or the necessary emission characteristics required for a given experiment. Sometimes, it is the radiation properties of the source itself which are being studied.

1. Radiographic Applications

The technique of using photons (x rays or γ rays) to produce photographs of objects on film is called radiography. In this application, the photons are passed through an object and are incident upon a photographic film. Due to regions of different density, and/or thickness, in the exposed object, the absorption of photons by the object will vary. This difference in the absorption in the exposed object results in different intensities of photons incident on the film. When developed, the film will have areas with varying degrees of contrast, some lighter, some darker. This contrast difference results in a picture of the different physical structure throughout the object.

Radiography requires the use of high activity γ ray sources and is generally used for checking the adequacy of pipe welds and seams, thickness of pipes and walls, and for locating voids in such things as shields or other supposedly solid construction. Because these applications may utilize sources of ^{60}Co or ^{192}Ir of activity up to several TBq (100-200 Ci), the sources represent a significant potential for radiation exposure. ^{192}Ir ($T_{½}$ = 74.02 d) gives a "softer" radiation than ^{60}Co ($T_{½}$ = 5.27 y) so it does not require as much shielding as ^{60}Co. The presence of more low energy photons enhances the quality of the radiographs because the greater absorption of these photons produces more contrast on the film. When thick objects needs to be radiographed, ^{60}Co is the recommended source. Because of the lighter required shielding of ^{192}Ir units, these are used extensively for applications which require portability, such as in shipyards, reactors or other field inspection applications.

The sources are supplied in portable projection systems, popularly referred to as γ "cameras." These cameras are usually constructed of steel and contain either lead or depleted uranium as shielding material. Figure 18.4a shows the design of a typical projector used for up to 3.7 TBq (100 Ci) of ^{192}Ir. The source is contained in a stainless steel capsule which is part of a small section of cable which attaches to the drive cable (see figure 18.4b). These source attachments are commonly referred to as "pigtails." In turn, the drive cable is connected to a hand crank which allows the source to be cranked out to the end of the guide tube (SNOUT) for the exposure, and cranked back to the shielded camera following the radiograph. Some projection devices have been designed so that the drive cable cannot be disconnected unless the source is in the proper shielded position in the camera. Generally, the projection units for ^{60}Co will require heavier shielding so these units are generally mounted on wheels.

Depending upon the application, the source may, or may not, be used with a collimator at the end of the guide tube. Common collimators which are used consist of shielding which limits the irradiation angle to about a 60° cone.

Because of the significant source activities involved, the dose rates are fairly high and additional precautions need to be taken. This is particularly true, since there are no warning lights on the equipment. To be sure that the source is either in, or out, one must rely on survey

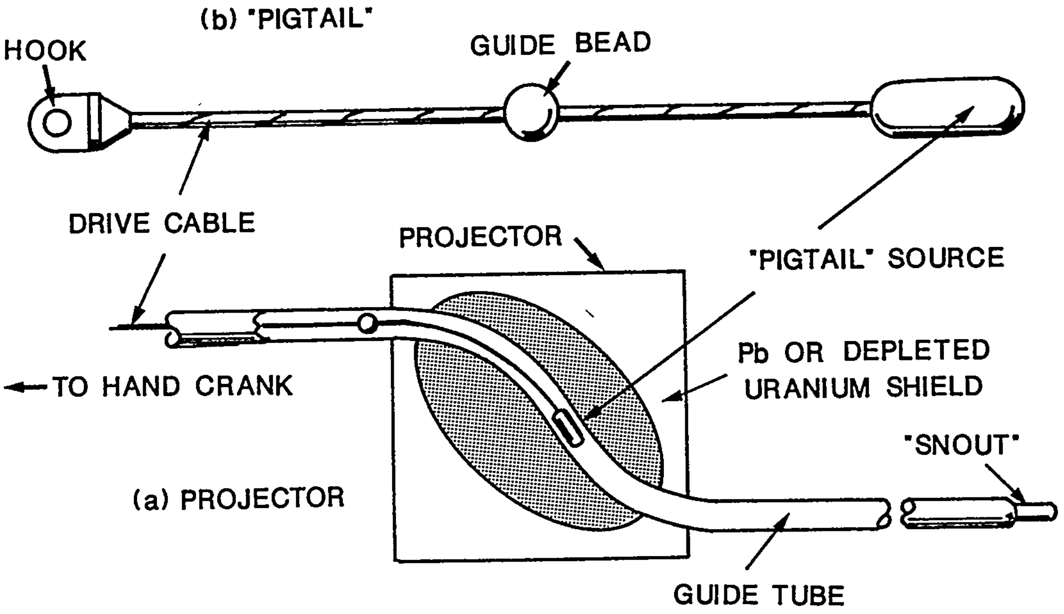


Figure 18.4 Radiography projector and "pigtail" source.

instruments. The cameras are supplied with locks to prevent unauthorized use, but should also be physically secured (locked up) when not in use. None of these projection systems (cameras) should be used unless a properly operating, and calibrated, survey meter is available to indicate the source location. Surveys must be made both prior to cranking the source out and following return of the source to the shielded position in the camera. The survey before the source is cranked out serves to indicate the expected radiation field when the source is in the properly shielded position. The survey following retrieval should include the camera itself, as well as the guide tube.

An additional consideration in the use of sealed sources, which is not a factor in x ray machines, is the potential for damage to the source or defects in the encapsulation or cladding. Both of these occurrences could lead to the release of a portion of the source material. Although high activity sources are usually doubly-encapsulated, there still is the need to document that no leakage is occurring. Depending upon the initial activity in the source, one needs to set up a program for leak testing of sealed sources. For high activity sources, the frequency of leak checks should be no less than every six months.[27] For activities which are significantly lower than those in radiography sources, the frequency may be relaxed to once per year. In general, a leak check smear which indicates greater than approximately 190 Bq (0.005 μCi) is considered indicative of a leaking source.

Among the frequent causes of exposures to radiographers is failure to make an adequate survey following a radiographic exposure. Sometimes, a malfunctioning survey meter has been the cause. In other cases, either no survey or an improper survey has been the reason. Human error is also found, especially in cases of lack of training, stress on the job or where fatigue sets in. Cases have occurred in which the source has been left exposed; sometimes not fully retracted, other times because it jams in the guide tube. On occasion, radiographers have simply forgotten to retract the source. In some instances, exposures have occurred because the radiographer failed to lock the camera following use.

The radionuclides used in radiography are generally under the control of the U.S. Nuclear Regulatory Commission. As such, the regulatory requirements as set out by the USNRC for a radiographer's license can be found in Reference 25. For an additional discussion of desirable safety measures in the use of radiography sources, see Reference 26. Reference 28 contains criteria for the radiation safety training of radiographers. In addition, it lists several references concerning radiation protection practices in industrial radiography.

2. Research Applications

In the use of sealed sources in research applications, the same principles of protection apply as previously discussed. There is a need to make surveys of the source in its shielded enclosure to determine that the shield is adequate. When the source is being used, one must survey the immediate area for primary radiation fields, and also adjacent rooms, on the sides, above and below the source room to ensure that scattered radiation is not irradiating someone in a nearby location. All sealed sources need to be properly shielded when not in use, and checked for leakage at periodic intervals.

As was the case with radiography sources, leak testing at an appropriate frequency must be carried out. The same criteria for determining leaking sources given in Section 18.D.1 may be used.

Rules for handling these sources with respect to radiation protection should be formulated and actively pursued. Routine surveys by radiation protection personnel should be carried out to ensure that proper procedures are followed and that no contamination is being released. Implicit in what was discussed above, is that an appropriate storage area is designated for these sources. The identity, source strength, and location of these sources must be documented (by inventory).

In addition, discussions with the user with a view toward reducing exposure in the operation should be continually carried on so that the ultimate goal of ALARA may be realized.

Since most of the sources used in radiography or in research have energies above approximately 200 keV, conventional survey instruments as described in Section 12 may be used for the various surveys discussed in this subsection. If accurate determination of radiation fields are required, one should use a precision ion chamber, such as are used for therapy beam measurements. More discussion on monitoring and instruments for x and gamma rays can be found in Reference 27.

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QUESTIONS

18.1 What is the basic difference between an x ray machine and sealed γ source as far as radiation production is concerned?

18.2 Why is it necessary to survey a shielded source?

18.3 What radiations are produced when an accelerating beam of electrons strikes a target?

18.4 What is the purpose of filters in an x ray machine?

18.5 Why are the filament and target of an x ray tube enclosed in an evacuated chamber?

18.6 Define the efficiency of an x ray tube.

18.7 List some methods used to prevent melting of the anode in an x ray tube.

18.8 Why are unfiltered x ray beams unsuitable for medical use?

18.9 How is the quality of the beam specified?

18.10 List some research uses of x ray diffraction.

18.11 What are fluorescent x rays? How are they generally produced?

18.12 Explain some of the potential sources of radiation associated with the operation of x ray machines.

18.13 What are some of the safety devices employed in x ray equipment.

18.14 What are the required quality assurance measurements for medical x ray installations? What instruments are used for this purpose?

18.15 When do ion chamber survey measurements need correction?

18.16 What are likely sites of radiation leakage in electron microscopes and electron beam welders?

18.17 In what areas are sealed γ sources employed?

18.18 Name some of the radionuclides used as brachytherapy sources.

18.19 What are the sources used in radiography? Name some of the applications of radiography.

18.20 What are commonly referred to as "pigtail" sources?

18.21 Why do, in general, iridium sources require less shielding than cobalt sources?

18.22 What is the primary purpose of leak testing the sealed sources periodically?

PROBLEMS

18.1 The maximum energy of an x ray produced by an x ray machine is 100 keV. What is the minimum wavelength of the photons?

Answer: 0.124 A.

18.2 Starting from the equation $E = h\nu$, (see Section 3), derive the equation for λ_{min}.

18.3 The efficiency of an x ray machine with tungsten target is rated as 0.30%. What is the tube voltage?

Answer: 40 kV

18.4 How many electrons flow from cathode to anode each second in an x ray tube with a current of 100 mA? If the tube voltage is 100 kV, at what rate is energy deposited on the anode?

Answer: 6.25×10^{17} electrons, 10^4 W.

18.5 Attenuation measurements for an x ray beam from an 100 kVp x ray machine yield the following results:

Added Filtration mm Al	Percent Attenuated
0.0	20
1.0	40
2.0	60
3.0	72
4.0	80
5.0	85

Plot the data of filtration and transmission on semilog graph paper and determine:

a) the HVL for 1 mm added filtration
b) the HVL for 2 mm added filtration.

Answer: a) 1.8 mm b) 2 mm

18.6 An x ray source is 5×10^{-3} m behind a shield block with an orifice of 10^{-3} m radius. An ion chamber 0.1 m from the shield block reads 150 cpm. If the diameter of the detector is 0.05 m, what should be the correct reading?

Answer: 212.6 cpm.

APPENDIX A

SELECTED CONSTANTS

NAME		VALUE
Avogadro's Number (Physical Scale) .	N_o	6.022×10^{23} molecules/mole
Base of Natural Logarithms. .	e	2.71828
Curie	Ci	3.7×10^{10} dis/s
Electron Charge	e	1.602×10^{-19} coulombs
Energy Equivalent of Atomic Mass Unit .	uc^2	931.68 MeV
Energy Equivalent of. . . . . Electron Mass	mc^2	0.511 MeV
Mass, Alpha Particle.	m_α	6.60×10^{-27} kg = 4.002603 u
Mass, Electron.	m_e	9.1095×10^{-31} kg = 0.000549 u
Mass, H-Atom.	m_H	1.6736×10^{-27} kg = 1.007825 u
Mass, Neutron	m_n	1.6750×10^{-27} kg = 1.008665 u
Mass, Proton.	m_p	1.6726×10^{-27} kg = 1.007276 u
Mass Unit, Unified Atomic . .	u	1.6606×10^{-27} kg = 1.0000 u
Microcurie.	μCi	10^{-6} curies = 3.7×10^{4} dis/s
Micro-microcurie. . . . . . . (picocurie).	$\mu\mu$Ci pCi	10^{-12} curies=3.7×10^{-2} dis/s
Millicurie.	mCi	10^{-3} curies = 3.7×10^{7} dis/s
Planck's Constant	h	6.626×10^{-34} Js
Pi.	π	3.1416 ...
Velocity of Light	c	$2.998 \times 10^{8} \sim 3 \times 10^{8}$ m/s
Wavelength Associated . . . . with 1 eV	λ_o	12397.67 Angstroms

APPENDIX B

RADIATION QUANTITIES AND UNITS

Quantity	Symbol	SI Unit	Unit Definition	Former Units	Definition
Absorbed dose	D	Gray (Gy)	1 Gy = 1 Jkg^{-1} = 100 rad	rad	1 rad = 10^{-2} Jkg^{-1} = 100 ergs/g
Activity	A	Becquerel (Bq)	1 Bq = 1 s^{-1}	curie (Ci)	1 Ci = $3.7x10^{10}$ s^{-1}
Dose equivalent	H	Sievert (Sv)	1 Sv = 1 Jkg^{-1}	rem	1 rem = 10^{-2} Jkg^{-1}
Exposure	X	--	1 C kg^{-1}	roentgen (R)	1 R = $2.58x10^{-4}$ Ckg^{-1} = 1 esu/cm^3 air
Kerma	K	Gray (Gy)	1 Gy = 1 Jkg^{-1} = 100 rad	rad	1 rad = 10^{-2} Jkg^{-1} = 100 ergs/g

APPENDIX C

SIGNS AND SYMBOLS

ALPHABETICALLY BY SYMBOL

A	Activity (radio); area; ampere; atomic mass number; amplification factor
$\tilde{A}$	cumulated activity
A°	Angstrom unit
A_o	activity, original
A_s	saturation activity
A_{LL}	long-lived activity
AF	attenuation factor
$\frac{AF(T \leftarrow S)}{M}$	specific absorbed fraction
ALI	annual limit on intake
a	acceleration, linear; distance
B	magnetic field; atomic stopping number; transmission factor
Bq	Becquerel
b	build-up factor
C	capacitance; concentration; coulomb; source activity; activity per unit mass
Ci	curie
c	velocity of light in vacuum
D	density, film; deuterium; absorbed dose
$\bar{D}_{50,i}$	50 y absorbed dose following intake
DAC	derived air concentration
d	density, general; deuteron; distance, linear; diameter
$d\bar{\epsilon}$	mean energy imparted
E	energy, electric field intensity
e	base of natural logarithms; electron; electron charge; lepton
E_T	threshold energy
e, e^-	charge of the electron
$^{0}_{-1}e$	electron; beta particle
$^{0}_{+1}e$	positron
e^+	charge of the positron
F	force; fraction; Faraday's constant
F_{CD}	correction factor for Cd-covered foil
f	frequency; flow rate, covered foil
f_i	correction factor for parameter i
G	geometry
g(E)	fluence rate conversion factor
H	dose equivalent; pulse height
$^{1}_{1}H$	proton
$H_{WB,L}$	annual whole body limit

$H_{50,T}$	committed dose equivalent
HEPA	high efficiency particulate air filter
HVL	half value layer
h	Planck's constant
$h\nu$	photon energy; quantum
I	intensity of radiation; intake
I_o	initial intensity
I_R	resonance integral
J	joule
K	kerma; constant; atomic shell
^oK	degrees Kelvin; absolute temperature
K.E., E_k	kinetic energy
k	constant; number of trials
k_∞	multiplication factor
kg	kilogram
L	non-leakage factor, angular momentum; atomic shell
L_C	decision limit
L_D	detection limit
L_∞	linear collision stopping power
LD_{50}	median lethal dose
l	length; mean free path; quantum number
M	mass, atomic shell
m	mass; quantum number; meter
m_e	rest mass of electron
m_H	mass of the hydrogen atom
m_n	mass of the neutron
m_p	mass of the proton
$m\mu m$	millimicron
mv	momentum
N	any number; atomic shell
N_a	Avogadro's number
N_o	Number of radioactive atoms at zero time; number original
n	neutron; number, any; number, original; emission fraction; quantum number
${}^{1}_{o}n$	neutron
O	opacity
P,p	momentum; pressure; probability; proton; resonance escape probability; atomic shell
P_g	gauge pressure
P_r	residual partial pressure
P.E.	potential energy
Q	electric charge; quality factor; reaction energy in MeV; atomic shell
q	charge, failure probability
R	range, (radiation); ratio; rate, count; activity ratio; resistance; universal gas constant; rate; Roentgen

R_α	range of alpha
RBE	relative biological effectiveness
r	radius; radial distance
S	observed standard deviation; linear stopping power; absolute sensitivity
$\frac{S}{\rho}$	mass stopping power
SEE(T←S)	specific absorbed energy
s	distance; second; quantum number
T	temperature; mean life; total time; occupancy factor; reactor period
$T_{½}$	half life, physical
T_b	half life, biological
T_{eff}	half life, effective
t	temperature, confinement time; time; triton; thickness
t_C	temperature, °centigrade
t_F	temperature, °fahrenheit
U	use factor
U_S	total transformations in source organ in 50 years
u	unified atomic mass unit
V	volume; potential; potential drop; volt
v	velocity, linear or particle
vol	volume
W	work; eV/ip; workload
w_T	weighting factor in organ dose
w/o	weight percent
wt	weight
X	exposure
x	absorber thickness
$x_{1/10}$	tenth value layer
Y	yield
Z	atomic number
Z_{eff}	effective atomic number
α	alpha particle; internal conversion coefficient
β	ratio v/c; delayed neutron fraction
$\beta, \beta^-, {}^{0}_{-1}\beta$	beta ray
$\beta^+, +{}^{0}_{1}\beta$	positron
Γ	specific gamma ray constant; fission products residual dose rate; first count factor
γ	gamma ray
Δ	finite increment; mean energy emitted per disintegration
Δk	excess reactivity
δ	secondary emission ratio
ϵ	dielectric constant; fast fission factor; electron capture; collection efficiency

ϵ_I	intrinsic efficiency
η	average fast neutrons per thermal neutron
θ	angle between incident & scattered radiation; bremsstrahlung angle, plane angle
i	subscript; index
κ	pair production coefficient
λ	decay constant; wave length; mean free path; attenuation length
λ_b	biological elimination constant
μ	attenuation coefficient; micro; muon; lepton
μ/ρ μm	mass attenuation coefficient
μ_R/ρ	mass removal cross section
μ_{en}/ρ	mass energy-absorption coefficient
μ_{tr}/ρ	mass energy-transfer coefficient
μm	micron
μs	microseconds
ν	frequency (wave motion quantum theory); neutrino
$\bar{\nu}$	anti-neutrino
Π	product of
π	pi, pi meson
ρ	density, general or vapor; reactivity
Σ	sum of; macroscopic cross section
σ	area; barn (cross section); theoretical standard deviation; Compton collision coefficient
σ_a	absorption cross sections in barns; Compton absorption coefficient
σ_r	relative standard error
σ_s	Compton scatter coefficient; scattering cross section in barns
σ_t	total cross section, barns
σ_{act}	activation cross section, barns
σ_{eff}	effective cross section, barns
σ_{fis}	fission cross section, barns
τ	resolving time; photoelectric coefficient; lepton
τ_o	neutron generation time
Φ	fluence
$\Phi(T \leftarrow S)$	Specific absorbed fraction
ϕ	work function; fluence rate; time increment
$\phi(T \leftarrow S)$	absorbed fraction
χ	symbol used in χ^2 test
Ψ	energy fluence
ψ	energy fluence rate
Ω	solid angle

APPENDIX D

SIGNS AND SYMBOLS

Mathematics

$+$	plus, addition, positive
$-$	minus, subtraction, negative
$\pm$	plus or minus, positive or negative
$\mp$	minus or plus, negative or positive
$\div, /, \_$	division
$x, \bullet, ()()$	multiplication
$()\ []$	collection
$=$	is equal to
$\neq$	is not equal to
$\equiv$	is identical to
$\cong$	equals approximately, congruent
$>$	greater than
$\ngtr$	not greater than
$\geqq$	greater than or equal to
$<$	less than
$\nless$	not less than
$\leqq$	less than or equal to
$::$	proportional to
$:$	ratio
$\sim$	similar to

$\propto$	varies as, proportional to
$\rightarrow$	approaches
∞	infinity
$\therefore$	therefore
$\int$	integral of
$\sqrt{\ }$	square root
$\sqrt[n]{\ }$	nth root
a^n	nth power of a
$\log, \log_{10}$	common logarithm
$\ln, \log_e$	natural logarithm
e	base of natural logs, 2.71828
π	pi, 3.1416
$\angle$	angle
$\perp$	perpendicular to
$\parallel$	parallel to
n	any number
$\lvert n \rvert$	absolute value of n
$\bar{n}$	average value of n
a^{-n}	reciprocal of nth power of a = $[1/a^n]$
n°	n degrees
n'	n minutes, n feet
n''	n seconds, n inches

Signs and Symbols - (cont'd.)

f(x)	function of x
Δx	increment of x
dx	differential of x
Σ	summation of
sin	sine
cos	cosine
tan	tangent
Π	product of

Greek Alphabet

Alpha	A	α	Nu	N	ν
Beta	B	β	Xi	Ξ	ξ
Gamma	Γ	γ	Omicron	O	o
Delta	Δ	δ	Pi	Π	π
Epsilon	E	ϵ	Rho	P	ρ
Zeta	Z	ζ	Sigma	Σ	σ
Eta	H	η	Tau	T	τ
Theta	Θ	θ	Upsilon	T	υ
Iota	I	ι	Phi	Φ	ϕ
Kappa	K	κ	Chi	X	χ
Lambda	Λ	λ	Psi	Ψ	ψ
Mu	M	μ	Omega	Ω	ω

APPENDIX E

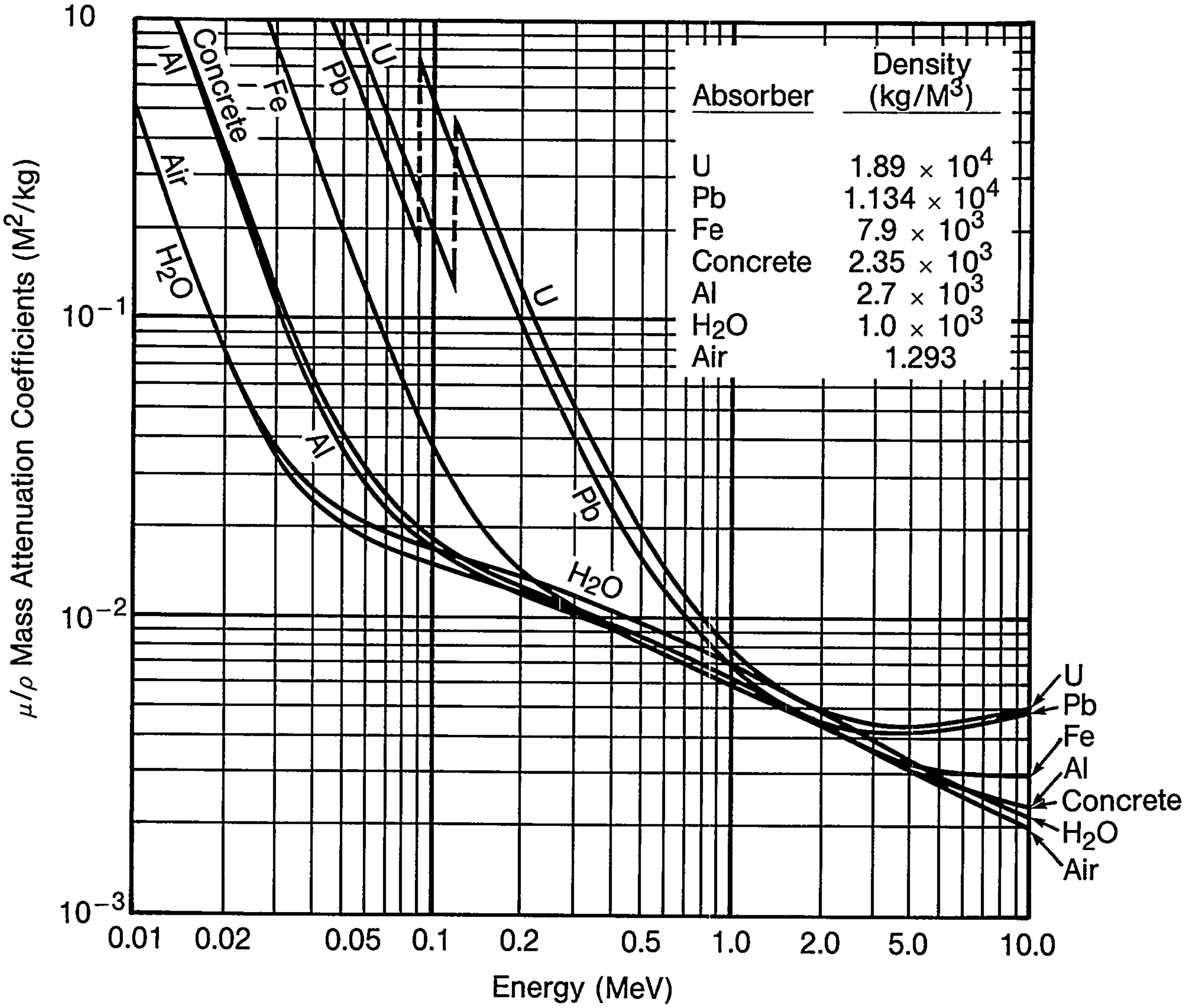
MASS ATTENUATION COEFFICIENTS (μ/ρ) & MASS ENERGY-ABSORPTION COEFFICIENTS (μ_{en}/ρ)

The mass attenuation coefficient, μ/ρ, gives the probability of photon interactions along the photon beam path. It is found from the linear attenuation coefficient, μ, by dividing μ by the density ρ of the absorber. The units of the coefficient are m^2/kg ($=cm^2/g$ divided by 10). Data for the coefficients was obtained from J. H. Hubbell, Int. J. Appl. Radiat. Isot. 33, 1269 (1982).

The mass energy-absorption coefficient, μ_{en}/ρ gives the probability of energy being absorbed as the photon interacts along the beam path. Corrections for the energy carried away by photons (scattered, bremmstrahlung, fluorescence, and annihilation) have been made in arriving at μ_{en}/ρ. The coefficient is expressed in units of m^2/kg* for a particular photon energy and absorber material. The value of the mass energy-absorption coefficient for tissue is approximately equal to that of water below about 0.06 MeV.

Data for the graph was also obtained from information contained in J. H. Hubbell, Int. J. Appl. Radiat. Isot. 33, 1269 (1982).

*Note: to convert m^2/kg to cm^2/g, multiply m^2/kg by 10



Absorber	Density (kg/M^3)
U	1.89×10^4
Pb	1.134×10^4
Fe	7.9×10^3
Concrete	2.35×10^3
Al	2.7×10^3
H_2O	1.0×10^3
Air	1.293

Figure E.1 Mass attenuation coefficients.

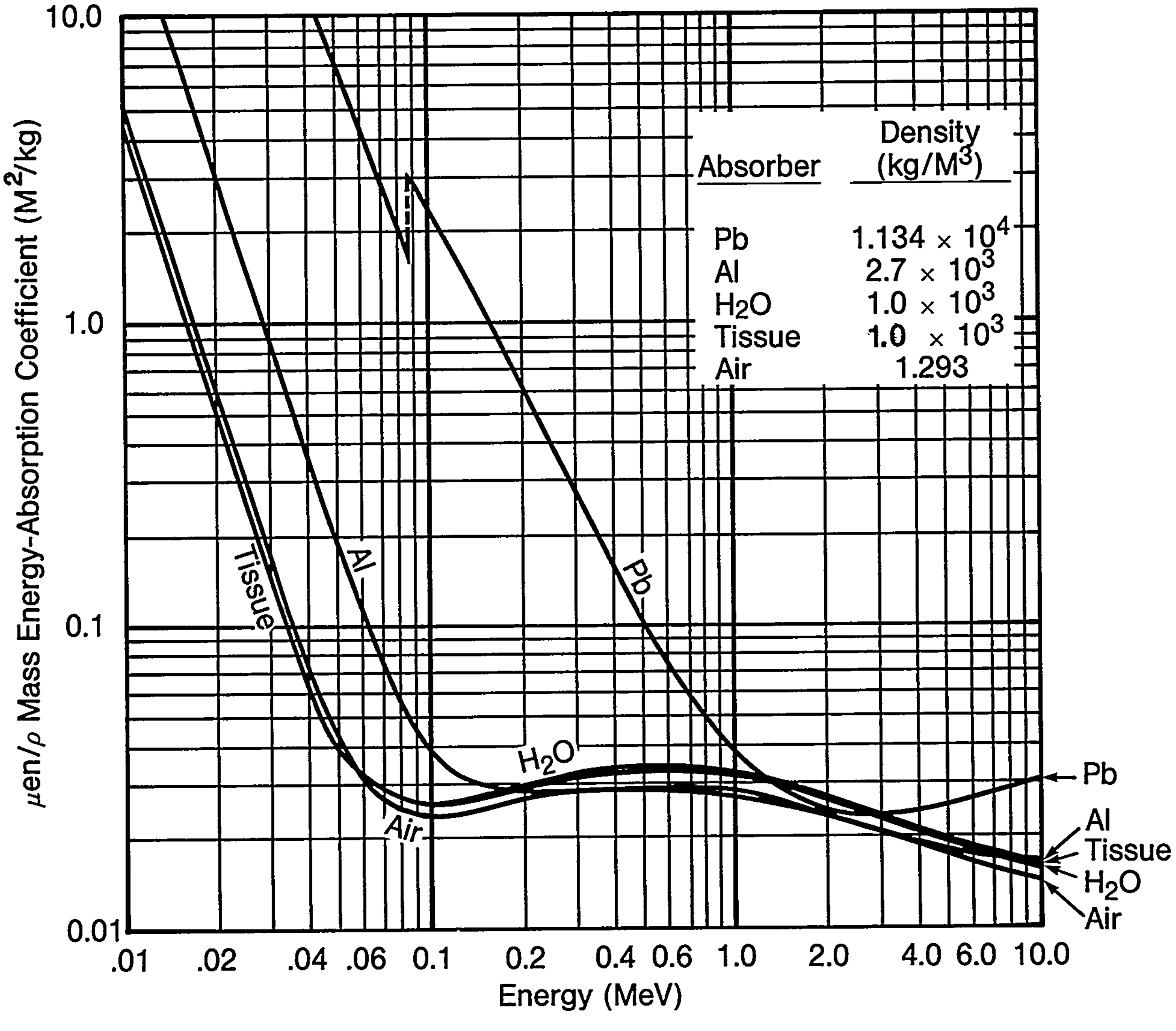


Figure E.2 Mass energy-absorption coefficients.

E-4

APPENDIX F

SELECTED MATHEMATICAL TOPICS

1. Exponents

On occasion, it is necessary to multiply a number by itself several times. This can be cumbersome to write out if a large number of such multiplications must be accomplished. In order to simplify writing this, these expressions are often written in exponential form. This consists of expressing the number, or quantity in the case of an algebraic symbol, with a superscript to the right of that number or quantity. This superscript is called the exponent (or power), and expresses the number of times the number or quantity (called the base) is multiplied by itself. For example, let $B \cdot B \cdot B$ be the quantity which is multiplied by itself. In exponential notation, this is written

$$B \cdot B \cdot B = B^3$$

If the quantity $B \neq 0$, then $B^0 = 1$ by definition.

When two of the same quantity, each raised to a certain power, are multiplied, the resultant power equals the sum of the exponents or powers. That is,

$$B^n B^m = B^{n+m}$$

To extend this, if a a given quantity which is raised to a certain power (B^n) is multiplied by itself m times, then

$$(B^n)^m = B^{nm}$$

For example,

$$(B^2)^4 = B^2 \cdot B^2 \cdot B^2 \cdot B^2 \cdot = B^8$$

This rule may also be extended to multiplication of different quantities with the same exponent:

$$(B^n)(C^n) = (BC)^n$$

A quantity or number which has a minus sign in the exponent signifies the reciprocal of that quantity to the plus power:

$$1/B^2 = B^{-2}$$

The rules for combining quantities with negative powers, or for combining negative with positive powers, are the same as for combining those with positive exponents,

$$B^{-n}B^{-m} = B^{-(n+m)}$$

$$B^nB^{-m} = B^n/B^m = B^{n-m}$$

$$(B^{-n})^{-m} = B^{nm}$$

A quantity can be expressed with a fractional exponent. This occurs when one quantity raised to a power equals another quantity

$$B^n = K$$

in which n is a positive integer. The quantity B is called the nth root of K. The above equation can be written in terms of a fractional exponent:

$$B = K^{1/n}$$

For example,

$$K^{5/2} = \sqrt{K^5}$$

$$2^{\frac{1}{2}} = \sqrt{2}$$

$$3^{2/3} = \sqrt[3]{3^2}$$

a. Exponential Equations

An exponential equation is one in which the unknown appears as an exponent. The exponential is the value which results when a given number, called the base, is raised to some exponent, x. In the Napierian system, the quantity e is used as the base. The value of the base e to six significant figures is 2.71828.

Exponential tables using the Napierian base give the value of e^x or e^{-x}. These tables are used in solving problems in which the rate of change of one quantity with respect to another is a constant fraction. Applications such as radioactive decay and radiation attenuation involve the use of these tables.

In general, the exponential equation has the form

$$y = e^x \text{ or } y = e^{-x}.$$

In the case of radioactive decay, we write

$$N/N_o = e^{-\lambda t} \text{ or } A/A_o = e^{-\lambda t},$$

in which N/N_o or A/A_o stands for y, and λt stands for x. Similarly, for attenuation problems the form becomes

$$I/I_o = e^{-\mu x},$$

where I/I_o

now stands for y and the product μx replaces x.

Example: A radioactive sample has a decay constant $\lambda = 0.1d^{-1}$. Given the present activity (A_o = 600 dis/min), find the activity 12 days later.

$$A/600 = e^{-\lambda t} = e^{-(0.1)12} = e^{-1.2}.$$

From five-place tables, the value 0.30119 is given for $e^{-1.2}$, then,

$$A/600 = 0.30119 \text{ and } A = (600)(0.30119) = 181 \text{ dis/min.}$$

Assume that we were asked the activity at time t = 12 days earlier. In this case, A =600 dis/min and we write

$$600/A_o = e^{-\lambda t} \text{ or}$$

$$A_o/600 = e^{\lambda t}.$$

Again $x = \lambda t = 1.2$, so that we locate x = 1.2 in the five-place table and read the value 3.3201 under the e^x column. Then,

$$A_o/600 = 3.3201 \text{ and } A_o = (600)(3.3201) = 1992 \text{ dis/min.}$$

With the increased use of calculators, the use of exponential tables has given way to use of calculators. If the calculator has an e^x key, one simply enters the value of x (positive or negative) and presses the e^x key to obtain the value of e^x in the display. For calculators with the INV lnx keys, one enters the value of x and presses the INV lnx keys to obtain the value of e^x in the display.

b. Power Functions

A number of exponential equations involve a base which is not the quantity e. These equations are referred to as power function relationships. These are generally expressed by

$$z = ay^x,$$

in which a and x are constants, y and z are variables. In this case, y is a variable base. The exponential quantity y^x is easily solved on present day calculators. One simply enters the value y and presses the y^x key. Then enter the exponent value x and press the = key to have the quantity evaluated in the display. In addition, most calculators have a separate x^2 key for use when one is simply squaring a number x and a $\sqrt{x}$ key for finding the square root of a given number x.

2. Common Logarithms (Base 10)

A common logarithm is the exponent which expresses the power to which the base 10 must be raised in order to equal a given number. The exponent is called the common logarithm of that number. The common logarithm consists of the characteristic (the figures to the left of the decimal point) and the mantissa (the figures to the right of the decimal point).

The characteristic gives the order of magnitude of the given number, both by the figure and its algebraic sign. A positive figure indicates the number of factors of 10 and a negative figure, the number of factors of 1/10. A zero characteristic denotes a number which lies between one and 10. The mantissa gives the significant figures in the logarithm of the given number without regard to the decimal point placement. To find the common logarithm using logarithm tables, convert the number to scientific notation. For example, let $N = 0.765$. Find the common logarithm of N. Write N as $N = 7.65 \times 10^{-1}$. The mantissa of the logarithm can be found from a suitable table,[1] by first locating 76 under the N column and reading 88377 in the S column immediately to the right of 76 in the same row. The characteristic is $\overline{1}$ for 0.765. Note that for a number with n digits to the <u>left</u> of the decimal, the characteristic of the logarithm of that number is n-1. For a positive number less than one, the characteristic is $\bar{n}$, if the first digit (other than zero) appears in the nth place to the <u>right</u> of the decimal. The common logarithm (often written $\log_{10}$ or simply log) of 0.765 is then $\bar{1}.8837$. If the number were 7.65, then $\log_{10} 7.65 = 0.8837$. If the number were 76.5, $\log_{10} 76.5 = 1.8837$. Note that the algebraic sign applies only to the characteristic, not the mantissa.

As an alternate notation, a logarithm with a negative characteristic may be changed to one with a positive characteristic. A common method is to write the logarithm as

$$\log_{10} 0.765 = 0.8837 - 1 = -0.1163.$$

This notation is used on calculators to indicate numbers with a negative characteristic. When one enters the number 0.765 and presses the log key on the calculator, the display will indicate -0.1163 for the first four places.

Properties which make logarithms useful are:

$$\log_{10} (ab) = \log_{10} a + \log_{10} b$$
$$\log_{10} (a/b) = \log_{10} a - \log_{10} b$$
$$\log_{10} (1/a) = \log_{10} 1 - \log_{10} a = 0 - \log_{10} a$$
$$\log_{10} a^n = n \log_{10} a$$
$$\log_{10} a^{-n} = -n \log_{10} a$$
$$\log_{10} \sqrt[n]{a} = 1/n \log_{10} a$$

The antilog is the given number that the logarithm represents. From our previous example for 76.5:

$$\log_{10} 76.5 = 1.8837.$$

The antilog of 1.8837 is the given number 76.5. To find a number when given the logarithm of the number is the inverse of the previous process for finding the logarithm.

In the antilog process, locate the mantissa of the logarithm in a common logarithm table. The first two significant figures are found in the same row as the mantissa, but to the left under the N column. the third significant figure is the number at the top of the column in which the mantissa is located. Given the common logarithm 2.6064, find the antilog to three significant figures: In a logarithm table find 6064, read 40 to the left under the N column as the first two figures. In the same column as 6064, read 4 at the top of the column for the third figure. The antilog is then 404, exclusive of the decimal point placement.

The characteristic of the logarithm allows one to locate the decimal point. In our example, the characteristic is positive so that the number of figures to the left of the decimal point will be one more than the value of the characteristic. That is, for a characteristic of +2, there will be three figures to the left of the decimal point. The final answer for the antilog in our example is then

$$\text{antilog } 2.6064 = 404.$$

On a calculator the procedure is to enter the logarithm (2.6064), and press the INV and log keys and the antilogarithm will be shown in the display.

3. Natural or Napierian Logarithms

A natural or Napierian logarithm is the exponent which expresses the power to which the base e must be raised to equal a given number. The exponent is called the natural logarithm (often written ln or $\log_e$) of the number. If the given number is less than unity, the logarithm is negative. If the number is greater than unity, the logarithm is positive. The logarithm of unity is zero.

Natural logarithm tables contain the natural logarithms of numbers from 1.00 to 9.99. Logarithms of smaller or larger numbers with the same significant figures are found by use of the property

$$\ln A = \ln(N \times 10^k) = \ln N + k\ln 10,$$

in which A is a number larger than 9.99 or smaller than 1.00, N is a number with the same significant figures as A but lies between 1.00 and 9.99, and k is the power of ten by which N must be multiplied to equal A. Included with the table are the natural logarithms of 10^k, up to $k = 9$.

To use the table to find the natural logarithm of 3.97, look up the first two digits under the column marked N. That is, find 3.9 in the N column. Along the same row to the right of 3.9, the logarithm is read as 1.3788 under column 7. On a calculator with a lnx key, one simply enters the value of the number for which the natural logarithm is to be found. The logarithm is then found by pressing the lnx key. The calculator may display the logarithm to more digits than are found in the logarithm tables.

The properties listed in F.2 for common logarithms apply also to natural logarithms. Actually, the logarithm properties may be applied to <u>any</u> base.

To find the logarithm of a number to a different base, the following may be used

$$\log_x N = (\log_y N)(\log_x Y).$$

For change of base involving 10 and e, the above expression requires the use of these relationships

$$\log_{10} e = 0.43429$$

$$\ln 10 = 2.3026$$

4. Equations

An equation expresses that a given quantity is equal to some other quantity. An arithmetic equation involves the use of numbers; for example

$$4 + 5 = 9.$$

In this, the quantity to the left of the equal sign is equal to the quantity on the right of the sign. When the equation contains one or more unknowns, which are designated by symbols, the equation is called an algebraic equation. For example,

$$2x + 8 = 18$$

is an algebraic equation in which the symbol x represents the unknown value. Not all values of x will satisfy the above equality. Finding all the possible values of the unknown which satisfy the above relationship is referred to as solving the equation. For some equations, called identities, all values of the unknown are solutions. For other equations, as in the above example, only specific values are valid solutions. In the example, the solution $x=5$ is the only value for which the equation is true.

In approaching algebraic equations, the following operations may be useful:

$$a + b = b + a$$
$$ab = ba$$
$$(a+b)+c = a + (b + c)$$
$$(ab)c = a(bc)$$
$$a + o = a$$
$$a.o = o$$

Certain principles which govern the manipulations used to solve algebraic equations are:

1. An operation performed on one side of the equation must be performed on the other side.

2. A term may be transposed from one side of the equation to the other side by changing the sign of the term:

   $a + b = c$
   $a = c - b$

   In the example, the transposition of b is equivalent to the operation of adding -b to each side of the equation.

3. A quantity used as a multiplier on one side of the equation may be transposed as a divider to the other side of the equation, and vice versa. For example,

   $bc = d$
   $c = d/b$

   This maneuver corresponds to the operation of dividing each side of the equation by b.

a. Linear Equations

An algebraic equation in which the unknown(s) are to the first power is called a linear equation. A frequent type of linear equation encountered describes the relationship between two unknowns x and y. This is the linear equation of a straight line:

$$y = a + bx.$$

This form of the straight line equation is referred to as the slope-intercept form. The constant a is called the intercept, which is the value of y when x=o. The slope b is a measure of the relative change in y values for a change in the x values. It is expressed as the ratio of the difference in the two y values to the corresponding difference in the x values, $b = y_1 - y_o / x_1 - x_o$, in which x_1, y_1 and x_o, y_o are the two points. A special case of the linear equation is a straight line through the origin. For this case, a = o.

Some equations, such as radioactive decay or exponential attenuation, which are not straight lines can be put into the form of a straight line for ease in graphing. For example, consider the activity equation

$$A = A_o e^{-\lambda t}$$

Take the natural logarithm of each side of the equation. This gives

$$\ln A = \ln A_o - \lambda t,$$

since ln e = 1. Now, if one considers $\ln A = y$, $\ln A_o = a$, t as the unknown x, then λ, the transformation constant, is b, the slope of the line. The negative sign indicates that the variable y (ln A) decreases with increases in the variable x (t). By graphing on semilog paper (see Section 2.c), the plot will be a straight line.

b. <u>Quadratic Equations</u>

In a quadratic equation, the highest power of any unknown is 2. This type of equation is expressed by the general form

$$az^2 + bz + c = o,$$

in which a, b and c are constants and $a \neq 0$. The process of solving a quadratic equation is called finding the roots of the equation.

If the linear term in z is missing, the equation then reduces to an expression for the square of the unknown. Consider

$$\begin{aligned} 12\ z^2 - 768 &= 0 \\ 12\ z^2 &= 768 \\ z^2 &= 64 \\ z &= \sqrt{64} = \pm 8 \end{aligned}$$

Sometimes the left side of the equation can be factored into the product of several simpler expressions. When this can be accomplished, the result is an equation in which a number of quantities (or factors) are multiplied together to give the product zero. Thus, any one or all of the factors could be zero. Solutions are obtained by setting each factor equal to zero and solving for the value of z. For example, solve

$$z^2 + 6z + 8 = 0.$$

The expression on the left side of the equality sign can be factored into the product $(z + 4)(z + 2)$, so that the equation may be written

$$(z + 4)(z + 2) = 0.$$

Setting each factor equal to 0 and solving gives

$$\begin{aligned} z + 4 &= 0, \quad z = -4 \\ z + 2 &= 0, \quad z = -2. \end{aligned}$$

When factoring of the equation is not possible, the quadratic formula can be used to solve for the roots. In fact, it can be used to find the roots of the equation, regardless of whether one can find factors or not. The quadratic formula is

$$z = \frac{-b \pm \sqrt{b^2 - 4\ ac}}{2a}.$$

One root is found using the positive value of the square root of $b^2 - 4ac$, and one is found using the negative value.

5. Proportionality

When one finds that the value of one quantity is related to the value of another quantity, such that if the one is increased or decreased by a certain factor, the other also increases, or decreases, by the same factor, the two quantities are said to be directly proportional to one another.

In Section 1.F, some examples of direct proportions are presented. These include Ohm's Law, which states that the current in a circuit is directly proportional to the applied voltage, and the relationship between charge buildup on a capacitor and the applied voltage. In each case, the relation can be transformed into an equation by specifying a constant of proportionality. Consider a capacitor, then

$$Q \propto V,$$

in which Q is the symbol for the charge which builds up on the plates of the capacitor, $\propto$ is the symbol used to indicate proportionality and V is the applied voltage. To transform this expression into an equation, C, which is called the capacitance, is specified as the constant of proportionality, and the relationship becomes equation 1.16,

$$Q = CV$$

Note that this equation has the same form as that of a straight line through the origin, with C being the slope of the line. Other direct proportions may involve powers of the quantities, such as $A = 2\pi r^2$, or roots, such as $r = 1.3\text{x}10^{-15}\ A^{1/3}$.

In a statement of direct proportion between two quantities, it is not always necessary to know the value of the constant of proportionality in order to use the relationship. For example, the radius of a circle is directly proportional to the circumference, so that

$$C \propto r$$
$$C = kr.$$

Given two circles, one with $C_1 = 31.416$, $r_1 = 5$, find C_2, if $r_2 = 8$. This may be solved by forming ratios of the quantities:

$$\frac{C_2}{C_1} = \frac{k\ r_2}{k\ r_1} = \frac{r_2}{r_1},$$

since k, the constant of proportionality, is always the same value. Substituting values gives

$$\frac{C_2}{31.416} = \frac{8}{5}$$
$$C_2 = 1.6(31.416) = 50.266.$$

If two quantities are inversely proportional to each other, then an increase by some factor of one quantity results in a corresponding decrease in the other quantity. This may be expressed as

$$A \propto 1/B$$
$$A = C/B,$$

in which C represents the constant of proportionality. As in the case of direct proportions, an inverse proportionality may involve higher powers of the quantities. Such an example is the inverse square law (Section 3.D). For a point source of radiation, the intensity at point A, a distance r_A from a source may be written

$$I_A \propto 1/r_A^2 = k/r_A^2,$$

in which k represents the constant of proportionality. At point B, a distance r_B from the point source, the intensity will be

$$I_B = k/r_B^2.$$

Taking the ratio of these two expressions eliminates k. This allows one to use the known values of the intensity at some distance from the source to compute the unknown intensity at a different distance from the source, using

$$I_A/I_B = r_B^2/r_A^2,$$

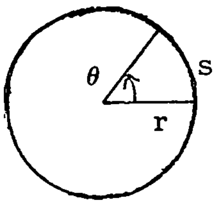
which is equation 3.26d in the text.

6. Geometry

Several concepts of plane and solid geometry are useful in health physics applications, particularly those dealing with angles, areas and volumes.

Angles may be expressed in degrees or radians. A right angle is a plane angle of 90°, a circle contains 360°, and the sum of the interior angles of a triangle is 180°. Subdivisions of the degree are minutes and seconds. There are 60 seconds (60") in 1 minute (1′) and 60 minutes (60′) in one degree (1°).

Angles may also be expressed in terms of radians. In the figure, the angle θ, in radians, is equal to the arc length s,



subtended by the angle, divided by the radius r of the circle, or

$$\theta = \frac{s}{r}.$$

Since the total arc length around the circle is the circumference, which is equal to $2\pi r$, the total angle, in radians, in a circle is

$$\theta_{total} = \frac{2\pi r}{r} = 2\pi \text{ radians.}$$

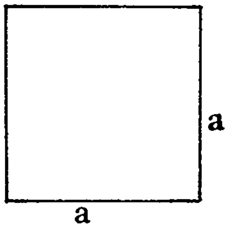
The relationship between radians and degrees is then

$$360° = \theta_{total} = 2\pi \text{ radians}$$

$$1° = \frac{2\pi}{360} \text{ radians}$$

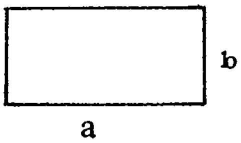
Several useful expressions for the area and the perimeter of plane figures are:

1. Square



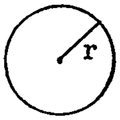
Area = a^2, perimeter = 4a

2. Rectangle



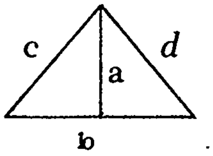
Area = ab, perimeter = 2a + 2b

3. Circle



Area = πr^2, circumference = $2\pi r$

4. Triangle



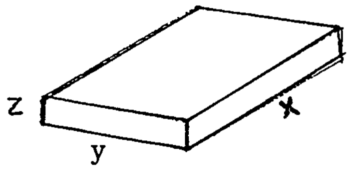
Area = 1/2 ab, perimeter = b+c+d

Plane geometry is concerned with figures that have at most, two dimensions. Solid geometry is concerned with figures that have up to three dimensions. Several expressions of use for certain common geometric solids are:

1. rectangular solid

Area = $2xz+2yz+2xy$

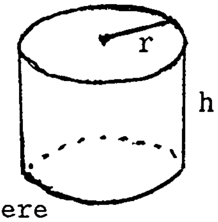
Volume = xyz



2. cylinder

Area = $2\pi rh+2\pi r^2$

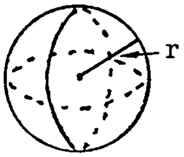
Volume = πr^2h



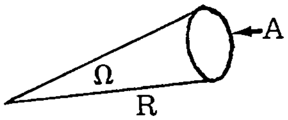
3. sphere

Area = $4\pi r^2$

Volume = $4/3\pi r^3$



One other important concept from solid geometry, of use in health physics, is the solid angle, Ω. The solid angle is measured by the ratio of the surface area A, of the portion of a sphere enclosed by a cone which forms the solid angle Ω (see figure), to the square of the radius R of the sphere:



$$\Omega = \frac{A}{R^2}$$

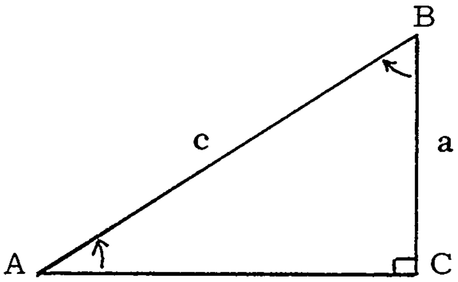
The unit of solid angle is the steradian, sr. The total solid angle Ω_{total} in a sphere is given by:

$$\Omega_{total} = \frac{A}{R^2} = \frac{4\pi R2}{R^2} = 4\pi sr.$$

The concept of the fractional solid angle, called the geometry (see Section 11.H.4), is used to determine the fraction of radiation emitted from an isotropic point source which reaches the detector.

7. Trigonometry

Trigonometric functions are defined in terms of the ratios of the sides of a right triangle. A right triangle contains an angle of 90°, called a right angle. In order to have a right angle, one of the legs of a triangle must be perpendicular to one of the other legs of the triangle. In the figure, the 90° angle is C and the side c of the triangle opposite the right angle is called the hypotenuse.



The functions of angle A defined by the ratios of the sides are:

$$\text{sine } A = \frac{a}{c} = \frac{\text{opposite side}}{\text{hypotenuse}} \qquad \text{cosecant } A = \frac{c}{a} = \frac{\text{hypotenuse}}{\text{opposite side}}$$

$$\text{cosine } A = \frac{b}{c} = \frac{\text{adjacent side}}{\text{hypotenuse}} \qquad \text{secant } A = \frac{c}{b} = \frac{\text{hypotenuse}}{\text{adjacent side}}$$

$$\text{tangent } A = \frac{a}{b} = \frac{\text{opposite side}}{\text{adjacent side}} \qquad \text{cotangent } A = \frac{b}{a} = \frac{\text{adjacent side}}{\text{opposite side}}$$

The trigonometric functions are useful for solving for unknown sides or angles in the triangle by utilizing the given information concerning the rest of the triangle. Values for the sine, cosine and tangent of a given angle can be obtained on calculators which have the keys sin, cos, and tan, respectively. The cofunctions listed above are simply the reciprocal of the function value. In addition, given a right triangle in which the value of the sine is known, the angle can be found by entering the value of the sine and pressing the INV sin keys.

One other useful relationship regarding right triangles is the Pythagorean theorem. Using the figure above, this relationship is

$$a^2 + b^2 = c^2,$$

or the square of the hypotenuse equals the sum of the squares of the other sides.

Two properties of the trigonometric functions which apply to any triangle are called the law of sines and the law of cosines. Using the terminology of the previous figure, the law of sines may be written:

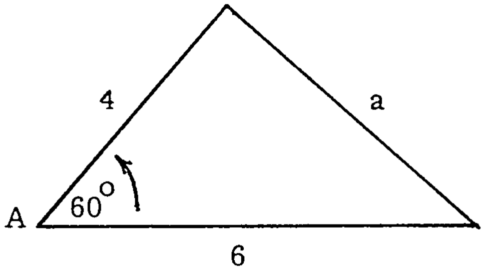
$$\frac{a}{\sin A} = \frac{b}{\sin B} = \frac{c}{\sin C},$$

in which sin is used as an abbreviation for sine.

Using the abbreviation cos for cosine, the law of cosines may be stated:

$$a^2 = b^2 + c^2 - 2bc \cos A.$$

If one knows the value of two sides and the included angle of a triangle, the other side can be determined. For example, given the two sides and the included angle in the figure, solve for the value of side a:



$$a^2 = b^2 + c^2 - 2bc \cos A$$
$$= (6)^2 + (4)^2 - 2(6)(4) \cos 60°$$
$$a^2 = 28$$
$$a = \pm\sqrt{28} = \pm 5.29$$

In this case, it is clear from the physical picture that a=5.29 is the solution.

REFERENCES

1. Kruglak, H. and Moore, J.T., Theory and Problems of Basic Mathematics, Schaum's Outline Series, McGraw -Hill Book Co., New York, NY (1973).

APPENDIX G

LIST OF ELEMENTS, AND CHART OF THE NUCLIDES*

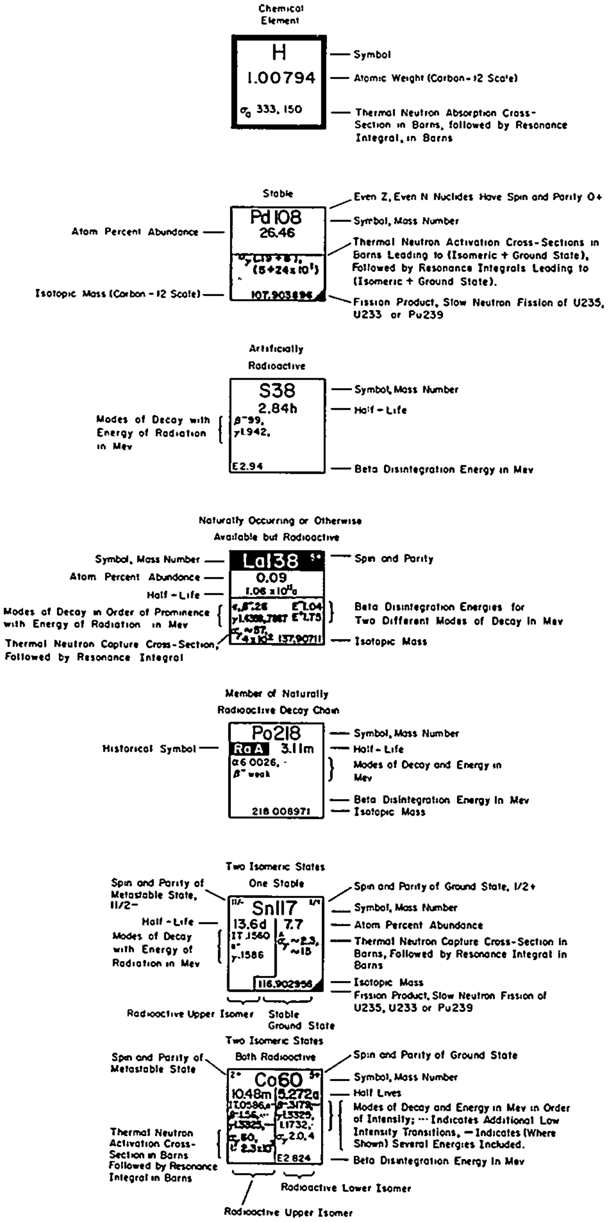
*Used with permission of Knolls Atomic Power Laboratory, Schenectady, New York. Operated by the General Electric Company for Naval Reactors, the United States Department of Energy.

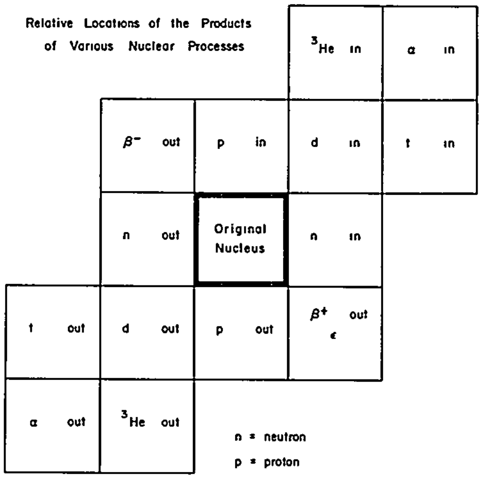
LIST OF ELEMENTS

Atomic Number	Symbol	Name	Atomic Number	Symbol	Name
0	n	neutron	53	I	iodine
1	H	hydrogen	54	Xe	xenon
2	He	helium	55	Cs	cesium
3	Li	lithium	56	Ba	barium
4	Be	beryllium	57	La	lanthanum
5	B	boron	58	Ce	cerium
6	C	carbon	59	Pr	praseodymium
7	N	nitrogen	60	Nd	neodymium
8	O	oxygen	61	Pm	promethium
9	F	fluorine	62	Sm	samarium
10	Ne	neon	63	Eu	europium
11	Na	sodium	64	Gd	gadolinium
12	Mg	magnesium	65	Tb	terbium
13	Al	aluminum	66	Dy	dysprosium
14	Si	silicon	67	Ho	holmium
15	P	phosphorous	68	Er	erbium
16	S	sulfur	69	Tm	thulium
17	Cl	chlorine	70	Yb	ytterbium
18	Ar	argon	71	Lu	lutetium
19	K	potassium	72	Hf	hafnium
20	Ca	calcium	73	Ta	tantalum
21	Sc	scandium	74	W	tungsten
22	Ti	titanium	75	Re	rhenium
23	V	vanadium	76	Os	osmium
24	Cr	chromium	77	Ir	iridium
25	Mn	manganese	78	Pt	platinum
26	Fe	iron	79	Au	gold
27	Co	cobalt	80	Hg	mercury
28	Ni	nickel	81	Tl	thallium
29	Cu	copper	82	Pb	lead
30	Zn	zinc	83	Bi	bismuth
31	Ga	gallium	84	Po	polonium
32	Ge	germanium	85	At	astatine
33	As	arsenic	86	Rn	radon
34	Se	selenium	87	Fr	francium
35	Br	bromine	88	Ra	radium
36	Kr	krypton	89	Ac	actinium
37	Rb	rubidium	90	Th	thorium
38	Sr	strontium	91	Pa	protoactinium
39	Y	yttrium	92	U	uranium
40	Zr	zirconium	93	Np	neptunium
41	Nb	niobium	94	Pu	plutonium
42	Mo	molybdenum	95	Am	americium
43	Tc	technetium	96	Cm	curium
44	Ru	ruthenium	97	Bk	berkelium

Atomic Number	Symbol	Name	Atomic Number	Symbol	Name
45	Rh	rhodium	98	Cf	californium
46	Pd	palladium	99	Es	einsteinium
47	Ag	silver	100	Fm	fermium
48	Cd	cadmium	101	Md	mendelevium
49	In	indium	102	No	nobelium
50	Sn	tin	103	Lw	lawrencium
51	Sb	antimony	104	Rf	rutherfordium
52	Te	tellerium	105	Ha	hahnium

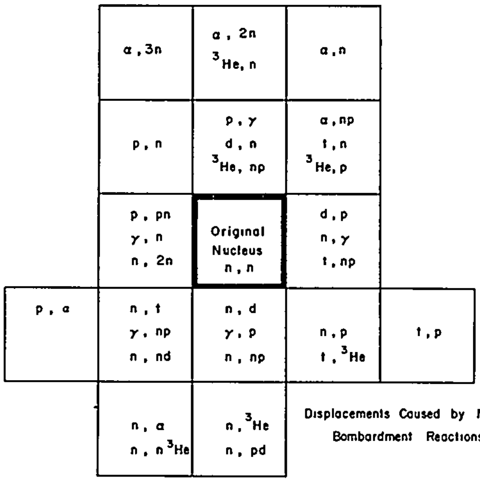
CHART OF THE NUCLIDES





Relative Locations of the Products of Various Nuclear Processes

n = neutron
p = proton
d = deuteron
t = triton (³H)
α = alpha particle
β⁻ = negative electron
β⁺ = positron
ε = electron capture



Displacements Caused by Nuclear Bombardment Reactions

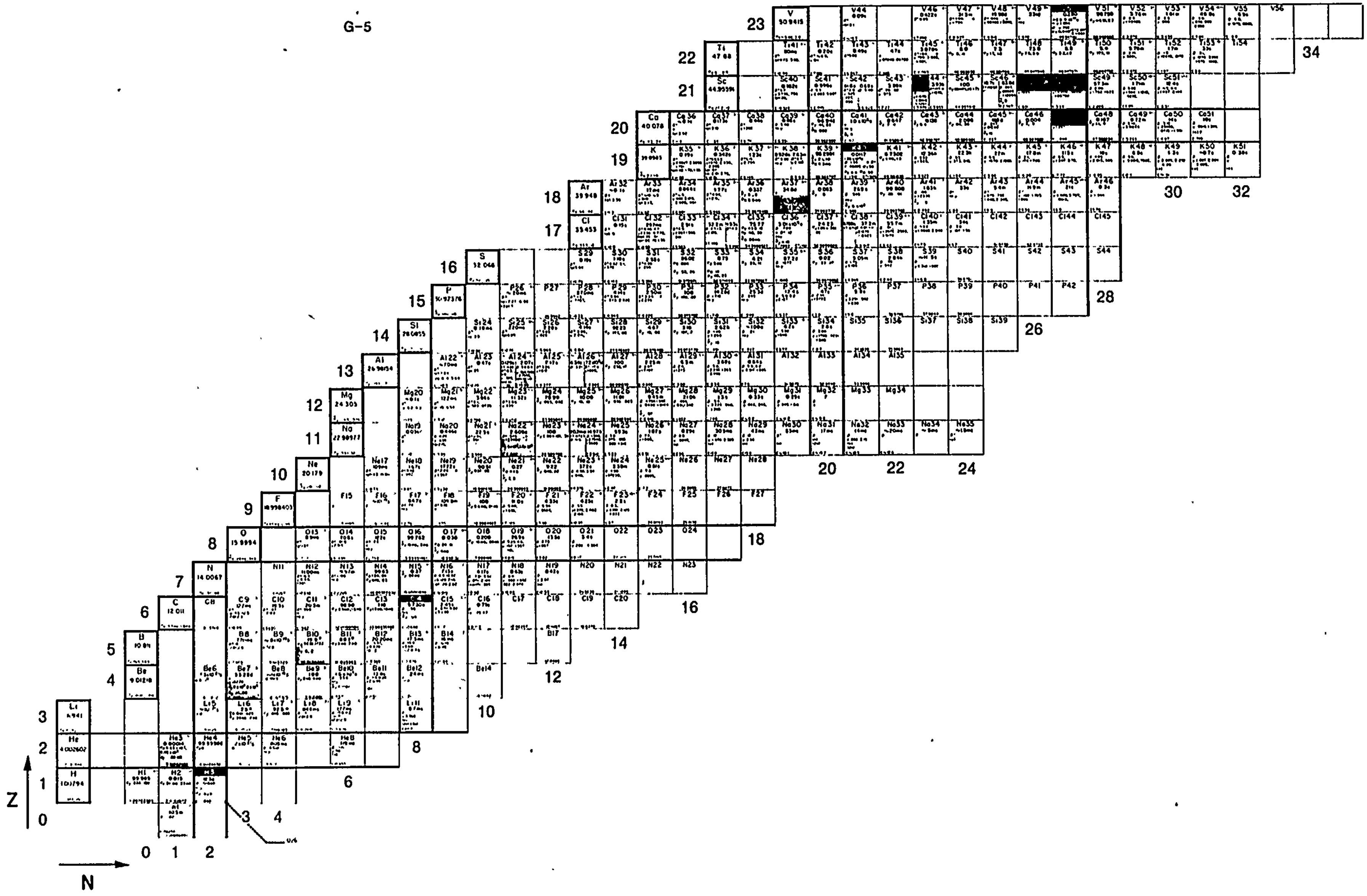
SYMBOLS

TIME

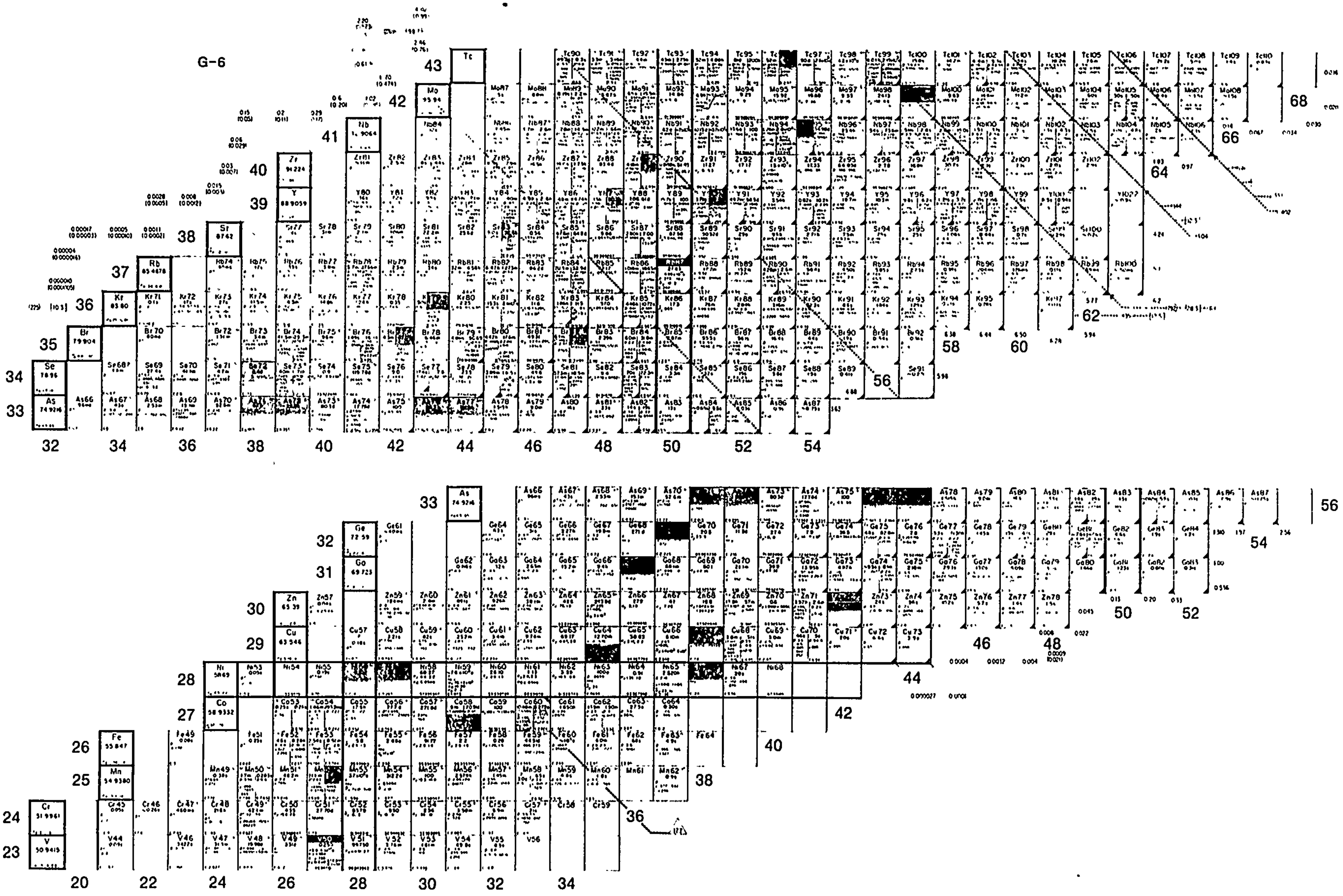
μs	microseconds (10^{-6}s)
ms	milliseconds (10^{-3}s)
s	seconds
m	minutes
h	hours
d	days
a	years

RADIATIONS AND DECAY

α	alpha particle	IT	isomeric transition
β⁻	negative electron	D	radiation delayed
β⁺	positron	SF	spontaneous fission
γ	gamma ray	E	disintegration energy
n	neutron	e⁻	conversion electron
p	proton	β⁻β⁻	double beta-decay
ε	electron capture		


Z
N
H
He
Li
Be
B
C
N
O
F
Ne
Na
Mg
Al
Si
P
S
Cl
Ar
K
Ca
Sc
Ti
V

G-6



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Tc
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Mo
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Nb
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Zr
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Y
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Sr
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Rb
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Kr
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Br
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Se
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As
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Ge
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Ga
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Co
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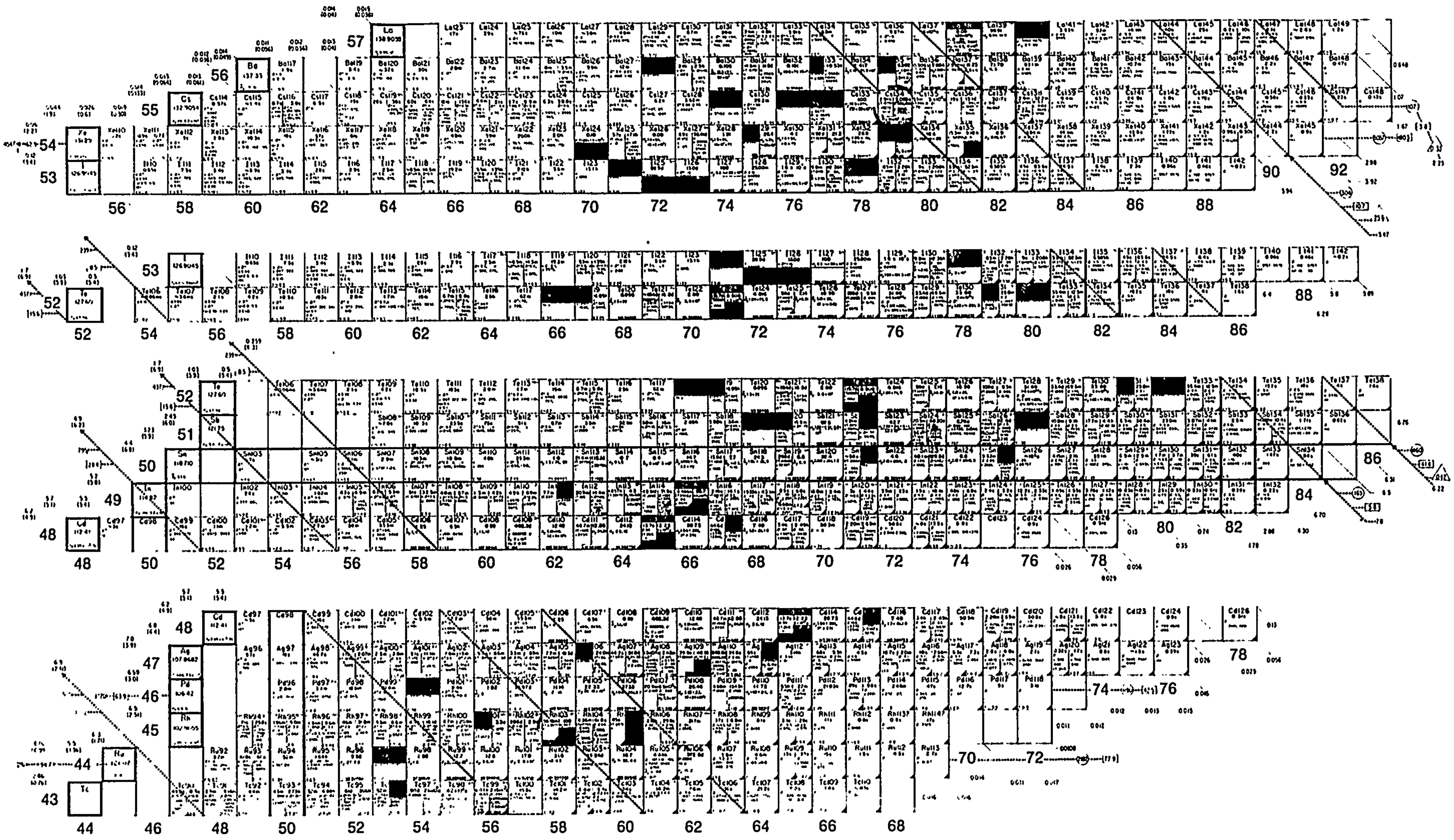

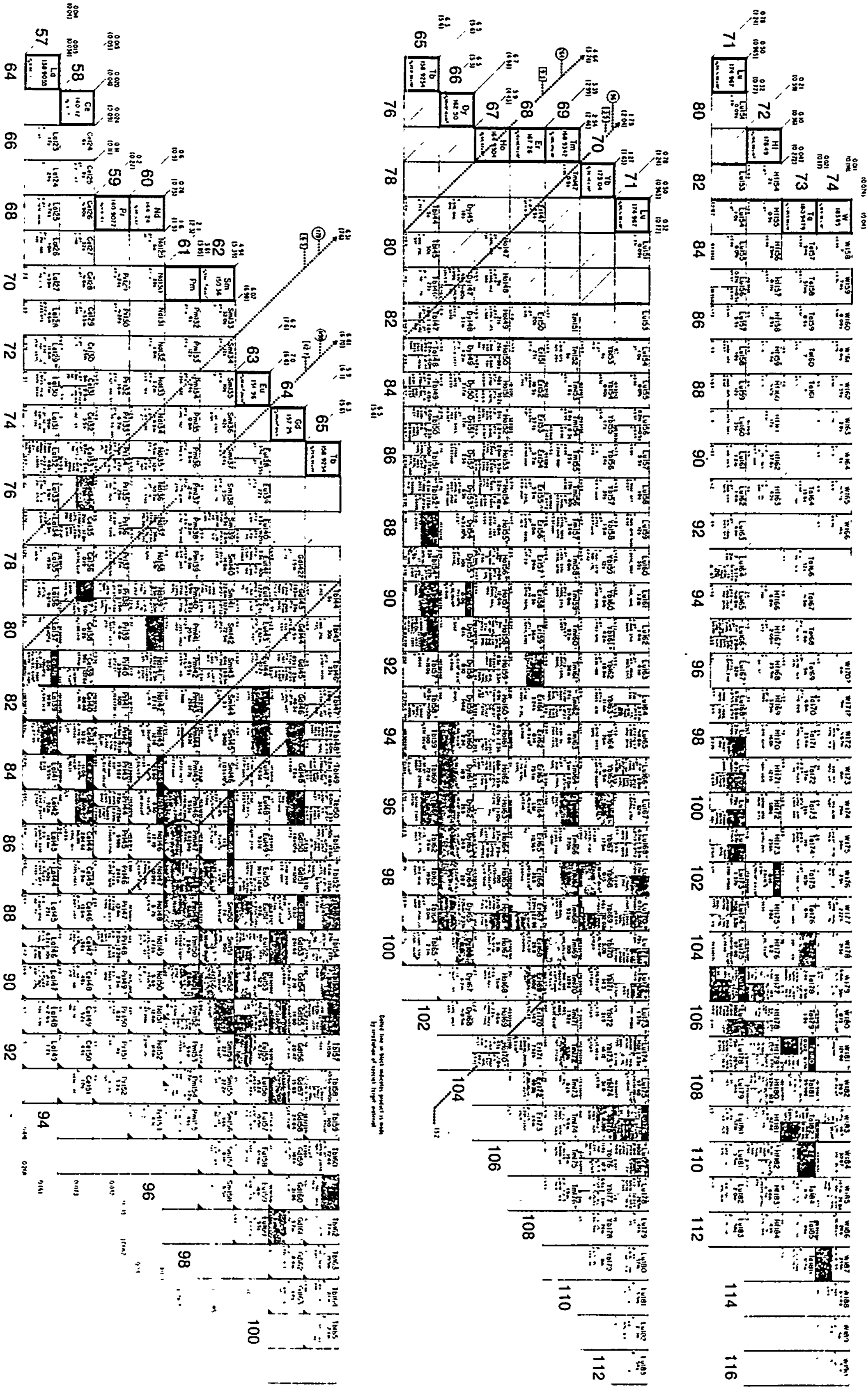
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La
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Ce
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Pr
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Nd
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Pm
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Sm
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Eu
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Gd
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Tb
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100
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Tb
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Dy
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Ho
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Er
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Tm
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Yb
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Lu
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112
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Lu
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Hf
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Ta
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W
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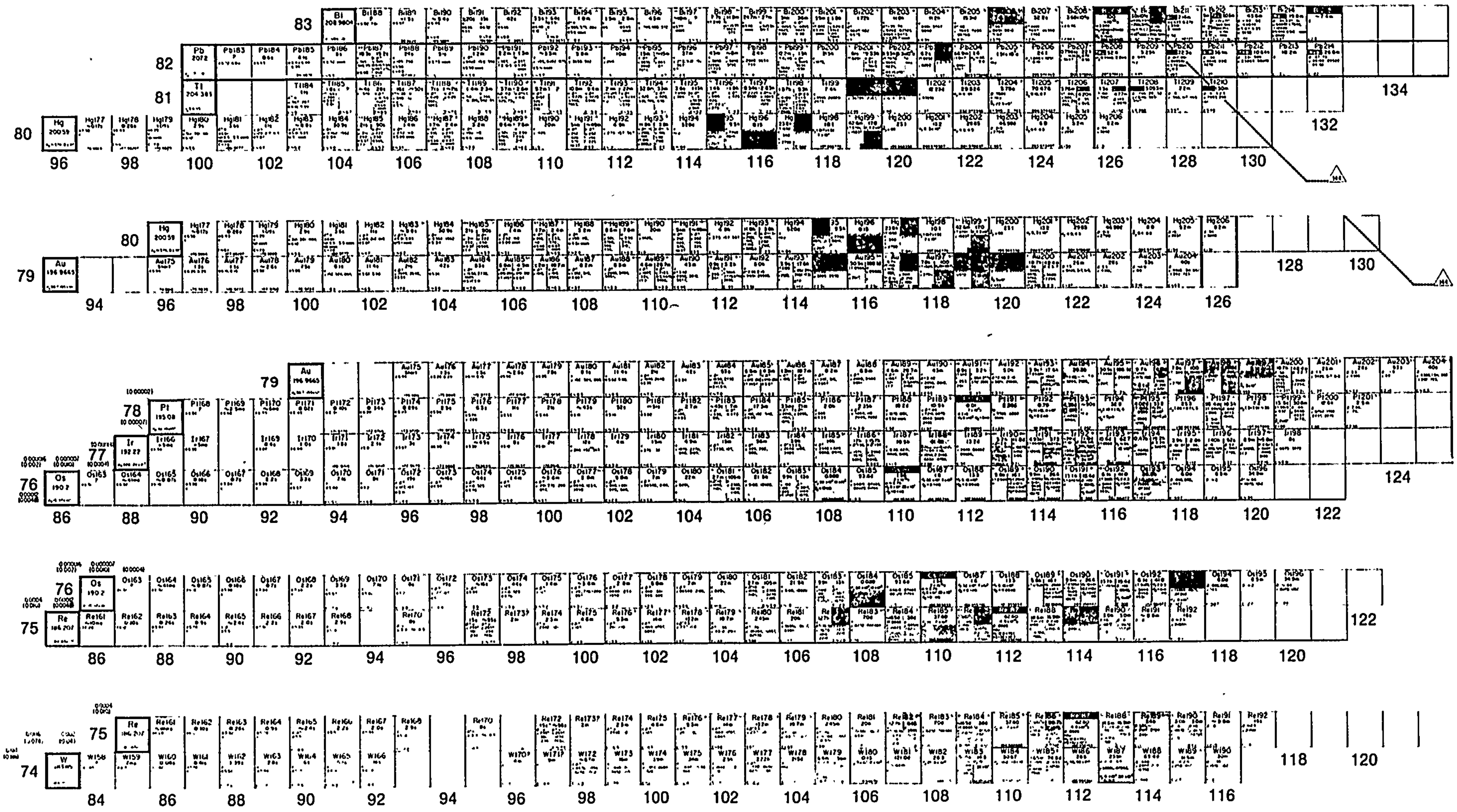
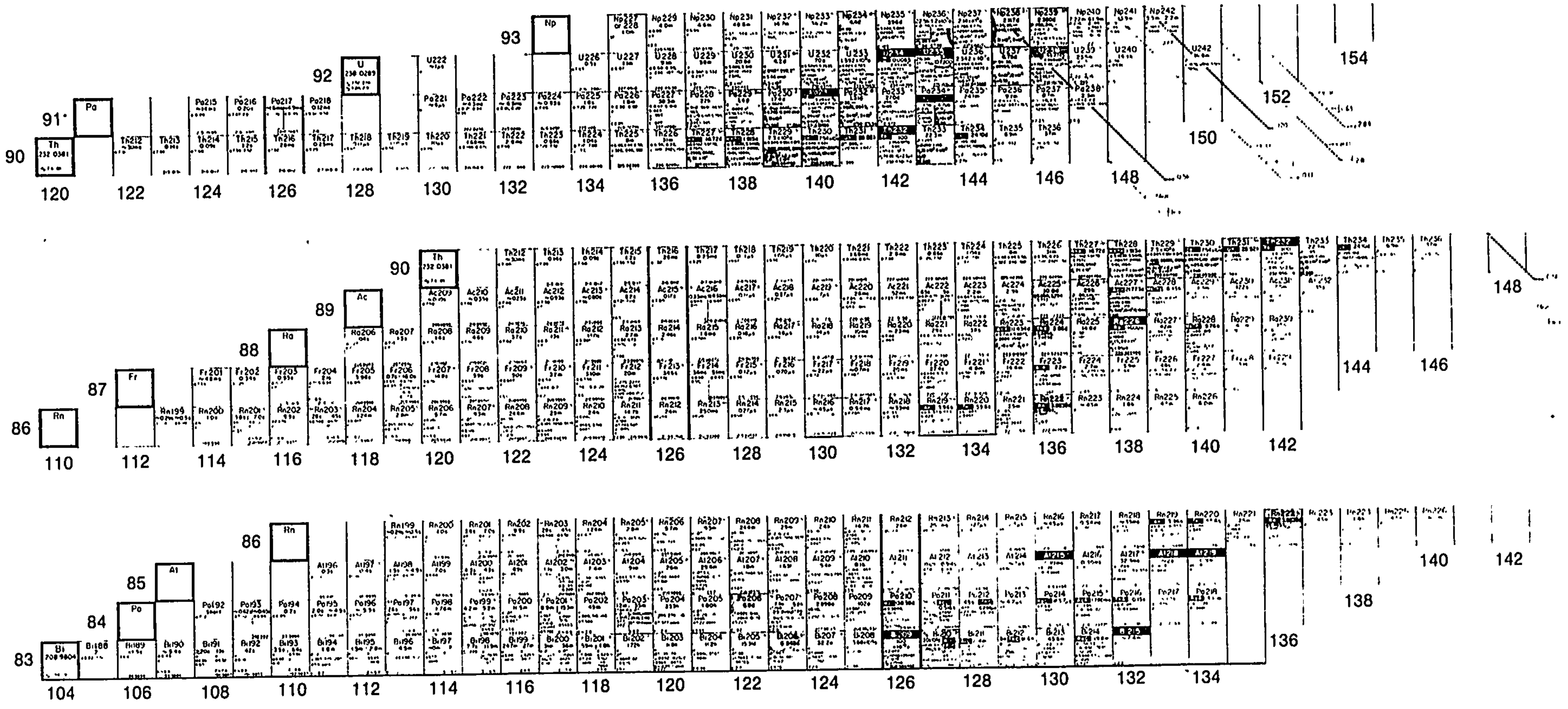
G-9

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Th
Pa
U
Np
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152
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89
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Rn
Fr
Ra
Ac
Th
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142
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146
148
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86
Bi
Po
At
Rn
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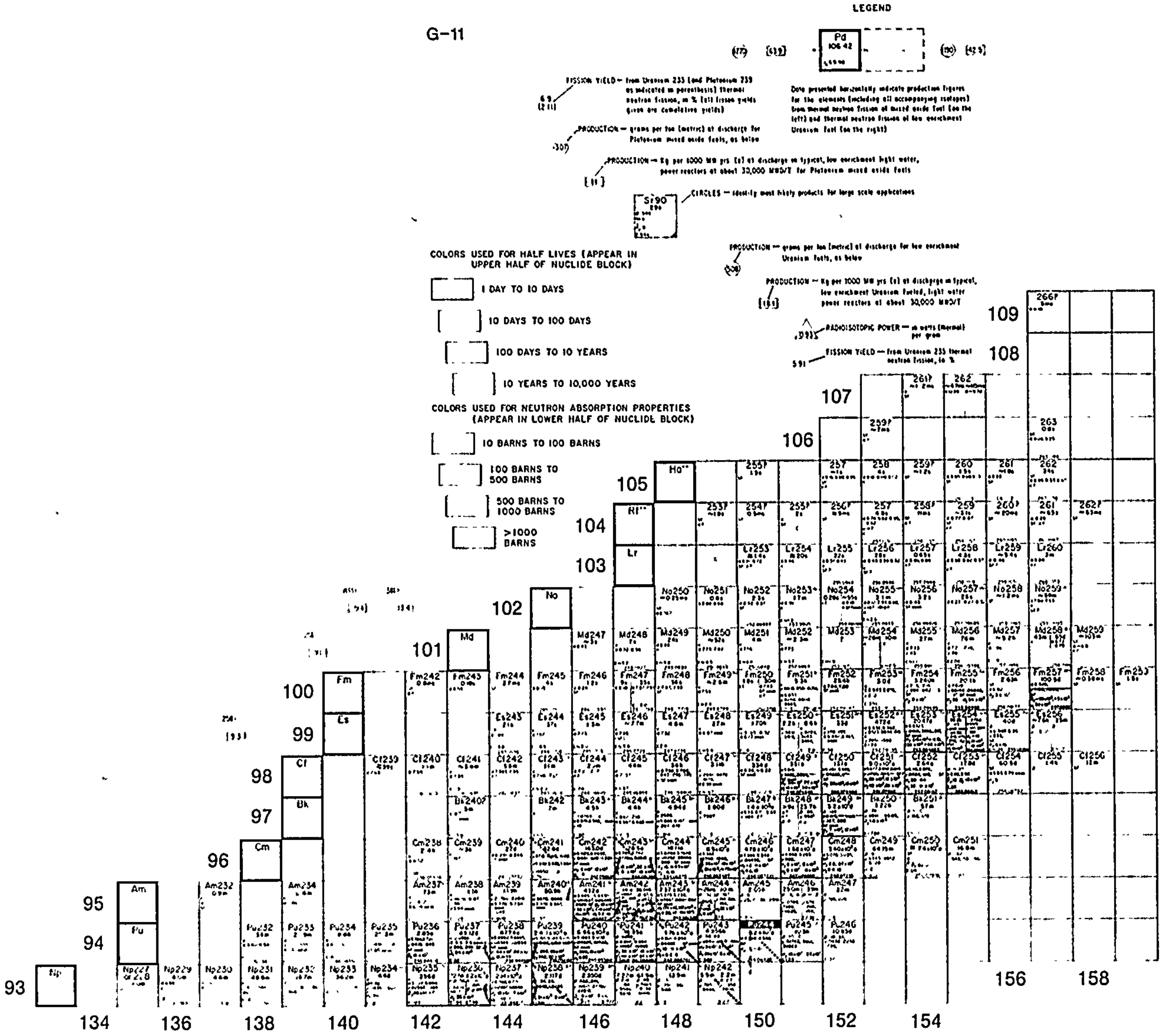
LEGEND
Pd
106.42
(17.7)
[63.9]
(110)
[42.5]
FISSION YIELD — from Uranium 233 (and Plutonium 239 as indicated in parenthesis) thermal neutron fission, in % (all fission yields given are cumulative yields)
PRODUCTION — grams per ton (metric) at discharge for Plutonium mixed oxide fuels, as below
PRODUCTION — Kg per 1000 MW yrs (e) at discharge in typical, low enrichment light water, power reactors at about 30,000 MWD/T for Plutonium mixed oxide fuels
Data presented horizontally indicate production figures for the elements (including all accompanying isotopes) from thermal neutron fission of mixed oxide fuel (on the left) and thermal neutron fission of low enrichment Uranium fuel (on the right)
Sr90
CIRCLES — identify most likely products for large scale applications
PRODUCTION — grams per ton (metric) at discharge for low enrichment Uranium fuels, as below
PRODUCTION — Kg per 1000 MW yrs (e) at discharge in typical, low enrichment Uranium fueled, light water power reactors at about 30,000 MWD/T
RADIOISOTOPIC POWER — in watts (thermal) per gram
FISSION YIELD — from Uranium 235 thermal neutron fission, in %
5.91
COLORS USED FOR HALF LIVES (APPEAR IN UPPER HALF OF NUCLIDE BLOCK)
1 DAY TO 10 DAYS
10 DAYS TO 100 DAYS
100 DAYS TO 10 YEARS
10 YEARS TO 10,000 YEARS
COLORS USED FOR NEUTRON ABSORPTION PROPERTIES (APPEAR IN LOWER HALF OF NUCLIDE BLOCK)
10 BARNS TO 100 BARNS
100 BARNS TO 500 BARNS
500 BARNS TO 1000 BARNS
>1000 BARNS
93
94
95
96
97
98
99
100
101
102
103
104
105
106
107
108
109
Np
Pu
Am
Cm
Bk
Cf
Es
Fm
Md
No
Lr
Rf
Ha
134
136
138
140
142
144
146
148
150
152
154
156
158
Np229
Np230
Np231
Np232
Np233
Np234
Np235
Np236
Np237
Np238
Np239
Np240
Np241
Np242
Pu232
Pu233
Pu234
Pu235
Pu236
Pu237
Pu238
Pu239
Pu240
Pu241
Pu242
Pu243
Pu244
Pu245
Pu246
Am232
Am234
Am237
Am238
Am239
Am240
Am241
Am242
Am243
Am244
Am245
Am246
Am247
Cm238
Cm239
Cm240
Cm241
Cm242
Cm243
Cm244
Cm245
Cm246
Cm247
Cm248
Cm249
Cm250
Cm251
Bk240
Bk242
Bk243
Bk244
Bk245
Bk246
Bk247
Bk248
Bk249
Bk250
Bk251
Cf239
Cf240
Cf241
Cf242
Cf243
Cf244
Cf245
Cf246
Cf247
Cf248
Cf249
Cf250
Cf251
Cf252
Cf253
Cf254
Cf255
Cf256
Es243
Es244
Es245
Es246
Es247
Es248
Es249
Es250
Es251
Es252
Es253
Es254
Es255
Es256
Fm242
Fm243
Fm244
Fm245
Fm246
Fm247
Fm248
Fm249
Fm250
Fm251
Fm252
Fm253
Fm254
Fm255
Fm256
Fm257
Fm258
Fm259
Md247
Md248
Md249
Md250
Md251
Md252
Md253
Md254
Md255
Md256
Md257
Md258
Md259
No250
No251
No252
No253
No254
No255
No256
No257
No258
No259
Lr253
Lr254
Lr255
Lr256
Lr257
Lr258
Lr259
Lr260
253?
254?
255?
256?
257
258
259?
260
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260?
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262?
259?
261?
262
263
266?

INDEX

ACKNOWLEDGEMENTS

It is always difficult to remember to properly thank those individuals who contribute to the production of a manual such as this. So many people become involved at various levels of activity lending support to the successful completion of such a long-range program that one may easily overlook some of those who did make a contribution. Nevertheless, one must make the attempt to acknowledge those who have contributed, even at the risk of omitting some who may be deserving of mention.

I am especially grateful to Dr. Rao Veluri, who undertook the revisions and additions to the questions and problems at the end of the various sections. Thanks to Dr. Robert Wynveen for his support and encouragement in undertaking this project and during the long period of revision and rewrite. To the previous authors of ANL-7291, Martin Schumacher (USNRC), Stanley Lasuk (USNRC) and Hiram Hunt (UNLV), I am grateful for their contribution which formed the basis for the rewrite of the present book. I would like to thank Dave Reilly for the diligence and patience shown in producing the wonderful drawings for this manual, and Al Forst of the Graphic Arts Section for his kind help and suggestions. A very special thank you is due Mrs. Linda Chamberlin for her tireless efforts in preparing and retyping the several stages of the manuscript.

I wish to acknowledge the assistance of Dr. Marcus Weseman (ORAU) of TRADE for providing the means to survey the DOE contractors to obtain planning information for the rewrite. I am grateful to the members of the Special Interest Group for Radiation Protection Training of TRADE for their interest in, and encouragement for, the revision and update of ANL-7291. Thanks to Ron Kathren (PNL) for his review, comments, and suggestions in regard to the proposed outline and form of the revision. In addition, a thank you to all the reviewers: Jack Selby (PNL), Bryce Rich (INEL), Roscoe Hall (SRP), Jay MacLellan (PNL), John Robinson (WINCO), Emily Copenhaver (ORNL), Charles Flood (BNL) and members of their staff who participated in the reviews, for their suggestions, corrections and comments. Because of their input, I believe the result is a better product.

Finally, I would like to acknowledge the guidance given by Mike Marelli, formerly of Ed Vallario's office, and Bette Murphy of DOE in helping to get this manual into a final form. I am especially grateful to Ed Vallario for his enthusiastic support of this project, for the opportunity to undertake this revision, and his continued interest in improving the quality of health physics in the DOE system.

Harold J. Moe

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