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Argonne National Laboratory

RADIATION SAFETY TECHNICIAN TRAINING COURSE

by

H. J. Moe, S. R. Lasuk,
and M. C. Schumacher

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H. J. Moe, S. R. Lasuk,
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Industrial Hygiene and Safety Division

September 1966

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RADIATION SAFETY TECHNICIAN TRAINING COURSE

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SECTION 1 - BASIC INFORMATION

A. Atomic Structure

An atom is the smallest unit of an element. A list of the different elements is given in the "Periodic Chart of the Atoms." There are 92 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, found in the periodic chart. Hydrogen, for example, has the symbol H; oxygen has the symbol O; etc.

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 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.

The radius of the nucleus is given approximately by

$$r \text{ (in cm)} = 1.5 \times 10^{-13} A^{1/3}.$$

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^{14} gm/cc.

The atomic mass of an element may be expressed in atomic mass units (amu). The amu, 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 weight divided by Avogadro's number N_a (= 6.025×10^{23} atoms/gm-atom). The reciprocal of Avogadro's number ($1/N_a$) is equal to the mass, in grams, of 1 amu, i.e., 1 amu = 1.66×10^{-24} gm.

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

$$N = \frac{m}{A} N_a,$$

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

$$N = \frac{m}{A} N_a = \frac{1}{238}(6.025 \times 10^{23}) = 2.53 \times 10^{21} \text{ atoms.}$$

The gram-atomic weight or gram-molecular weight is often used. 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 gm of ^{238}U; a gram-molecular weight (often called a mole) of H_2O is 18 gm. A gram-atomic weight of an element contains 6.025×10^{23} atoms; a gram-molecular weight contains 6.025×10^{23} molecules.

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 Fig. 1.1. The electrons rotate in elliptical and circular orbits about the nucleus. These are well-defined orbits (shells) which are called energy or quantum levels; the diameters of these orbits are large in comparison with the nuclear diameter. The shells are denoted by either a letter (K, L, M, N, O, P, Q) or quantum

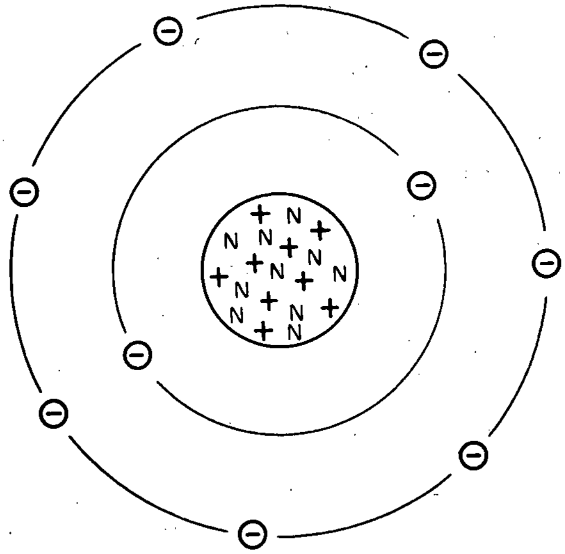

Fig. 1.1

The Stable Fluorine Atom ($^{19}_{9}F$). The nucleus contains 9 protons and 10 neutrons.

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, 1, 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.

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 (positive) 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 or removed. 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 Fig. 1.2 for the process of electron removal from an atom) results in an ion pair: the free electron and the positively charged residue.

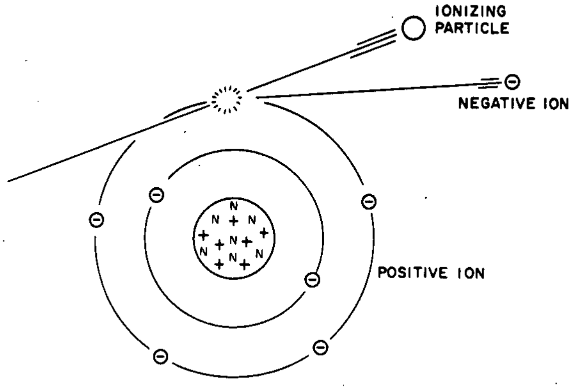

Fig. 1.2
The Ionization of an Atom

C. Isotopes

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 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.

Some isotopes 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 exists between the protons, the Coulomb force, which tends to pull the nucleus apart. Such forces are relatively long-range forces. In addition, there are very short-range attractive forces between the neutrons and protons which tend to hold the nucleus together. The net effect of these forces will determine whether a certain combination or ratio of neutrons and protons is stable. The lighter elements, in which the neutrons and protons are about equal, exhibit stability. Above $Z = 20$, an excess of neutrons is required for a nuclide to be stable.

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 cgs system is the gram. There are about 454 gm 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 cm/sec in the cgs 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:

$$F = ma.$$

If the acceleration is constant,

$$F = \frac{m(v - v_0)}{t} = \frac{mv - mv_0}{t}.$$

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 cgs system is the dyne: a dyne is the force which will give to one gram an acceleration of 1 cm/sec^2, that is, 1 dyne = 1 gm-cm/sec^2.

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: $W = Fd$. 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

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

The unit of work or energy in the cgs system is the erg. The erg = dyne-cm = gm-cm^2/sec^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.

Two important concepts in physics are Conservation of Momentum and Conservation of Energy. When no external force acts on a system, the total momentum of the system remains constant. 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.

F. Charged Particles

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

$$F = \frac{q_1 q_2}{kr^2} \text{ dynes},$$

where k is the dielectric constant.

Charges which are alike, repel; charges which are unlike, attract. The constant k depends upon the medium between the charges but is taken as unity if the medium is air.

The unit of charge in the cgs system is the electrostatic unit (esu) or statcoulomb, defined in terms of Coulomb's law: one esu is that charge which will repel an exactly equal charge with a force of one dyne when the two are one centimeter apart in a vacuum. The magnitude of the charge on the proton and electron is 4.8×10^{-10} esu. Although the statcoulomb is a small unit itself, it takes 2.08×10^9 electrons or protons to equal a charge of one esu.

When two parallel plates are connected to a battery (see Fig. 1.3), 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:

$$E = F/q.$$

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

$$W = Fd = Eqd;$$

but

$$E = V/d$$

also, so that

$$W = \frac{V}{d}qd = Vq,$$

and

$$V = W/q.$$

If W is in ergs and q in esu, then V will be in statvolts. When V is expressed in Volts, the relationship

$$\text{Volts}/300 = \text{statvolts}$$

may be used.

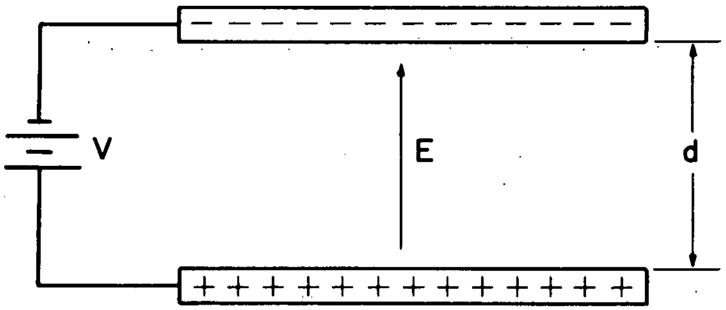

Fig. 1.3

The Electric Field E Is Perpendicular to the Plates. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition © 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

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. 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

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

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.

The usual unit of energy is the erg. 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 erg by: $1 \text{ eV} = 1.6 \times 10^{-12}$ erg.

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

$$Ve = \frac{1}{2} mv^2,$$

where Ve is expressed in ergs. This relationship is valid if the velocity of the electron is not comparable with 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.

G. Relativistic Corrections

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

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

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

$$m = \frac{m_0}{\sqrt{1 - \beta^2}},$$

where $\beta = v/c$.

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

The relativity theory considers a rest mass m_0 and a mass m which an object has by virtue of its speed v. From statements 2) and 3), we get

$$E = mc^2 = m_0 c^2 / \sqrt{1 - \beta^2},$$

where E represents the total energy content of an object of rest mass m_0 moving with a speed v. The energy content of the rest mass is found by setting $v = 0$, namely, $E_0 = m_0c^2$. The kinetic energy of the object is then

$$E_k = E - E_0 = m_0c^2 \left[\frac{1}{\sqrt{1 - \beta^2}} - 1\right].$$

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:

$$F = Hev,$$

where H is the magnetic field strength, e is the charge on the particle, and v its speed. The force will be perpendicular to H and v and the particle will tend to move in a circle of radius r (see Fig. 1.4).

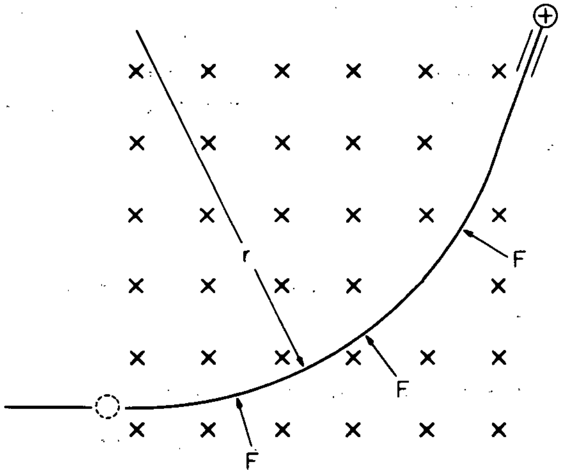

Fig. 1.4

A Charged Particle Traveling Perpendicular to a Magnetic Field Which is Directed into the Plane of the Paper. (Richards, Sears, Wehr and Zemansky, Modern University Physics, 1960, Addison-Wesley, Reading, Mass.)

In considering circular motion the force is given by

$$F = mv^2/r.$$

Accordingly, for a magnetic field acting on a charged particle moving at right angles to the field,

$$Hev = mv^2/r.$$

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SECTION 2 - NATURAL RADIOACTIVITY

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. Further investigation showed that this property of emitting radiation is a specific property of the element and that certain quantitative relationships exist. In particular, the atoms producing these radiations are unstable and decay at characteristic rates to form new atoms.

By the use of magnetic fields, it was shown that there are three distinct types of radiation (see Fig. 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 particle but a form of electromagnetic radiation, similar to light.

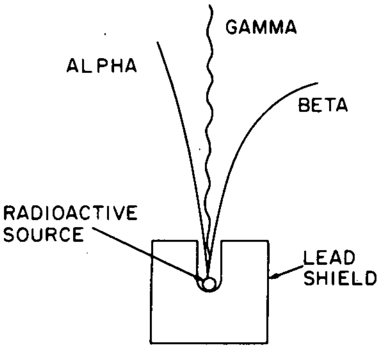

Fig. 2.1

Radiation Deflection in a Magnetic Field. The field is perpendicular to and directed into the plane of the paper. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition (Ⓒ) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

Since alpha particles were deflected in a different direction in the magnetic field than were beta particles, the two were oppositely charged. From the direction in which the alpha particle was deflected, it was shown to be positively charged, whereas the beta particle is negatively charged. Further investigation revealed that the alpha particle is actually a helium nucleus ($^4_2$He - two positive charges, mass number 4) and the beta particle is actually an electron (e^- - one negative charge, mass of 0.000549 amu).

B. Radioactive Decay

When an atom 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

$$^A_Z X \longrightarrow {}^{A-4}_{Z-2}Y + {}^4_2He.$$

When the decay is by beta emission, the atomic number Z increases by 1, but the mass number A remains the same:

$$^{A}_{Z}X \longrightarrow {}^{A}_{Z+1}Y + \beta^{-}.$$

The emission of a gamma ray does not affect either the atomic number or the mass number.

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 disintegrate in a given time interval. If one plots the number of atoms present at time t versus the time on linear graph paper (see Fig. 2.2), the curve obtained will indicate an exponential or logarithmic relationship. If the number present versus the time is plotted on semilog paper (see Fig. 2.3), a straight line is obtained. This indicates that radioactive decay is an 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:

$$\frac{\Delta N}{\Delta t} = \frac{N_1 - N_0}{t_1 - t_0} \propto N,$$

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

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

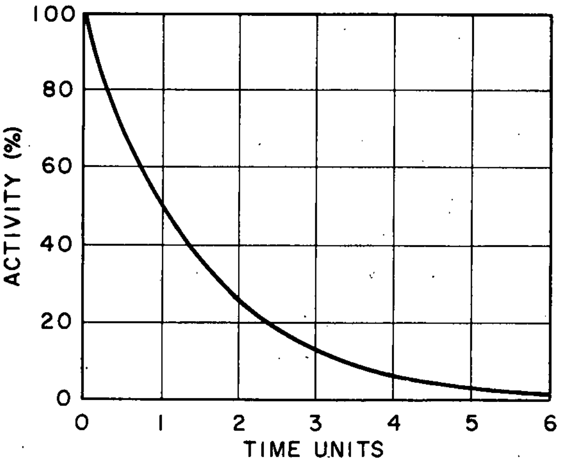

Fig. 2.2. Radioactive Decay, Linear Plot

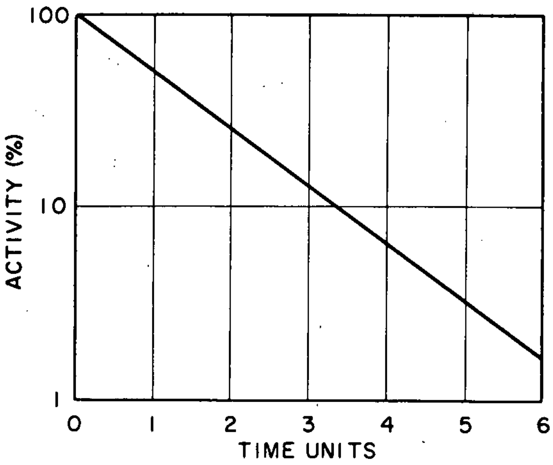

Fig. 2.3. Radioactive Decay, Semilog Plot

where λ, called 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:

$$N = N_0 e^{-\lambda t},$$

where N is the number of atoms left at time t from a sample of N_0 radioactive atoms initially present.

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

$$\ln N = \ln N_0 - \lambda t.$$

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_0.

As an example, consider the data:

Number of Atoms	Time (min)
80,000	0
29,432	1
10,824	2
3,984	3
1,464	4
536	5
200	6

Plot a curve on semilog and on linear paper, and determine the decay constant λ 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 $t = 2$ min. 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 2 min. We now obtain:

$$\ln N = \ln N_0 - \lambda t;$$

$$\ln N - \ln N_0 = -\lambda t;$$

$$\ln N/N_0 = -\lambda t;$$

$$-2 = -\lambda(2\ \text{min});$$

$$1\ \text{min}^{-1} = \lambda.$$

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

$$\frac{dN}{dt} = A = \lambda N.$$

In other words, the activity A 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,

$$N = N_0 e^{-\lambda t};$$

$$\lambda N = \lambda N_0 e^{-\lambda t};$$

$$A = A_0 e^{-\lambda t},$$

where A_0 is the activity (decay rate) of the sample at $t = 0$.

C. Half-life

The decay constant λ is closely related to the concept of the half-life of a radioisotope. The half-life $T_{1/2}$ is defined as the time required for a given isotope to decay to one-half of its initial value. Consider an initial activity A_0. At some time $t = T_{1/2}$, the activity will be equal to $A_0/2$. From our general relationship, then,

$$A = A_0 e^{-\lambda t};$$

$$\frac{1}{2} A_0 = A_0 e^{-\lambda T_{1/2}};$$

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

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

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

$$T_{1/2} = \frac{\ln 2}{\lambda} = \frac{0.693}{\lambda}.$$

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

1) half-life of emission,

2) energy of the emission.

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

D. Curie

The activity of a radioactive substance is often designated by the Curie (abbreviated Ci). The Curie is defined as the activity of a radioactive nuclide disintegrating at the rate of 3.7 x 10^{10} dis/sec. The Curie is not a measure of dose; it merely states the amount of a particular radioisotope disintegrating per unit time.

The definition of the Curie refers to disintegrations per unit time. Oftentimes, the number of disintegrations will differ greatly from the number of radiations emitted, as for example, in the case of ^{60}Co. For each ^{60}Co atom which disintegrates, a beta particle and two gamma rays are emitted. In this case, then, a Curie of ^{60}Co would emit 3.7 x 10^{10} β/sec and 7.4 x 10^{10} γ/sec. If the activity of a sample is to be accurately calculated from experimental data, the decay scheme of the radioisotope must be taken into account.

In counting a β emitter, some of the γ rays will also be counted. If one merely takes the counting rate and converts this to a disintegration rate, without considering the decay scheme, one will usually overestimate the activity of the source. For example, consider a counter which counts all the betas and 2% of the gammas. In counting a 1-μCi ^{60}Co source, the observed counting rate would be 3.7 x 10^4 β counts/sec + 0.02 (7.4 x 10^4)γ counts/sec = 3.8 x 10^4 counts/sec. If one converts this activity to Curies without regard to the decay scheme, the activity would come out as 1.04 μCi.

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.

For purposes of internal dose calculations, the National Committee on Radiation Protection defines a Curie of natural uranium and natural thorium in a different manner in NBS Handbook #69. A Curie of recently extracted uranium is equal to the sum of 3.7×10^{10} dis/sec from ^{238}U, 3.7×10^{10} dis/sec from ^{234}U, and 1.7×10^{9} dis/sec of ^{235}U. By this definition, then, a Curie of natural uranium is equal to 7.57×10^{10} dis/sec. A Curie of recently extracted thorium is equal to the sum of 3.7×10^{10} dis/sec from ^{232}Th and 3.7×10^{10} dis/sec from ^{228}Th. Thus, a Curie of natural thorium is equal to 7.4×10^{10} dis/sec.

E. Specific Activity

The specific activity is defined as the activity of one gram of a substance. It is usually expressed as Ci/gm of the radioelement. The shorter the half-life of the emitter, the greater is found to be its specific activity. The specific activity can be calculated from the equation

$$\text{SP. A.} = \lambda N,$$

where λ is the decay constant and N is now the number of atoms in one gram of the radioelement. ^{60}Co has a half-life of 5.24 yr. Calculate the specific activity:

$$\text{SP. A.} = \lambda N = \frac{0.693}{T_{1/2}} \frac{1\text{ gm}}{A} N_A$$

$$= \frac{0.693}{5.24\text{ yr}} \frac{(1)\ 6.025 \times 10^{23}}{\left(3.1 \times 10^{7} \frac{\text{sec}}{\text{yr}}\right) 60} = 4.22 \times 10^{13} \frac{\text{dis}}{\text{sec-gm}}$$

$$= \frac{4.22 \times 10^{13}}{3.7 \times 10^{10}} = 1.14 \times 10^{3} \frac{\text{Ci}}{\text{gm}}$$

F. Decay Chains

In general, most radioactive substances do not decay to form a stable isotope, 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

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

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

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

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

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

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

$$N_2 = \frac{N_1^0 \lambda_1}{\lambda_2 - \lambda_1}\left(e^{-\lambda_1 t} - e^{-\lambda_2 t}\right),$$

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 the last equation reduces to

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

However, since

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

we find

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

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.

When the parent activity is extremely long-lived, as is the case with ^{238}U ($T_{1/2} = 4.5 \times 10^9$ yr), then $\lambda_1 \ll \lambda_2$, $e^{-\lambda_1 t} \approx 1$, and $\lambda_2 - \lambda_1 \approx \lambda_2$, so that the original expression for N_2 reduces to

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

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

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

This state is known as secular equilibrium.

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

G. Decay Curve of a Mixture

The problem of identifying the half-life of a radioisotope 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 radioisotopes. 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 longer-lived component of the mixture. The shorter-lived products contribute greatly to the decay rate initially; but as these 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 paper (see Fig. 2.4) and determine the half-lives for all the activities in the sample.

cpm	t (hr)	cpm	t (hr)
10,000	0	370	14
5,778	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		

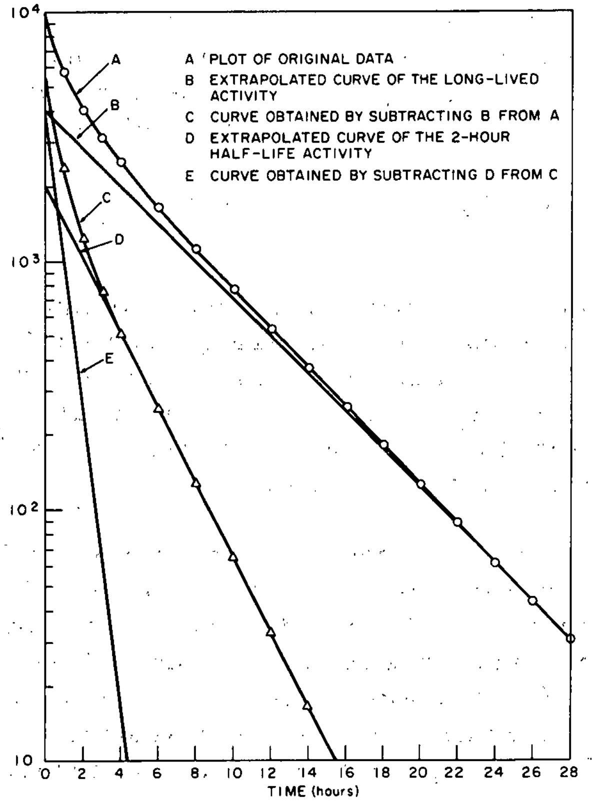

Fig. 2.4

Decay Curve Using Example Data Above

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 cpm. The half-life is determined directly from the graph: find the time at which the activity has dropped to 2000 cpm. This occurs at 4 hr. 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, cpm = 4000; t = 12 hr, cpm = 500.

$$\ln \frac{N}{N_0} = \ln \frac{1}{8} = -\ln 8 = -12\lambda;$$

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

But

$$T_{1/2} = \frac{\ln 2}{\lambda} = \frac{\ln 2}{\ln 2}(4) \text{ hr} = 4 \text{ hr}.$$

Take points on the extrapolated curve and subtract their values from the corresponding points on the experimental curve. Plot these differences versus the corresponding times. Consider the points for t = 0: the value on the experimental curve is 10,000 cpm, the value on the extrapolated curve is 4000 cpm. The difference, 6000 cpm, is plotted for t = 0. Similarly, for the point t = 1 hr, the value on the experimental curve = 5778 cpm, and the value on the extrapolated curve = 3400 cpm. The difference, 2378 cpm, is plotted for t = 1 hr. 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 2000 cpm. The half-life of this activity as determined from the graph is 2 hr.

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 obtained. The initial activity of this component is 4000 cpm, and its half-life as determined from the graph is 30 min.

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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.

A. Alpha Particles

The alpha particle is a helium nucleus ($^4_2$He) emitted with a discrete energy and a characteristic half-life from each alpha emitter. Rutherford and Royds showed that the alpha particle is a helium nucleus by collecting α particles from radon in a glass tube with electrodes. After allowing time for build-up, a source of high voltage was connected to the electrodes. When a discharge in the tube occurred, the spectrum obtained showed characteristic helium lines.

Examination of cloud-chamber photographs (see Fig. 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 α particles emitted by a given radioisotope 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.

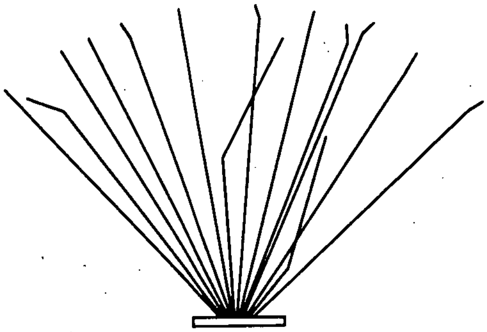


Fig. 3.1

Illustration of Alpha Tracks as Seen in a Cloud Chamber. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

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

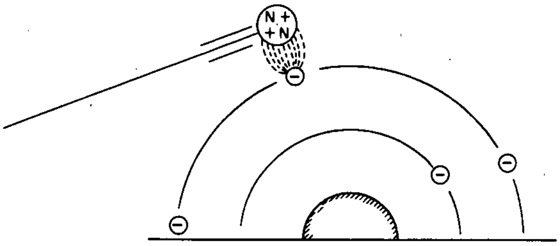


Fig. 3.2. Electrostatic Interaction

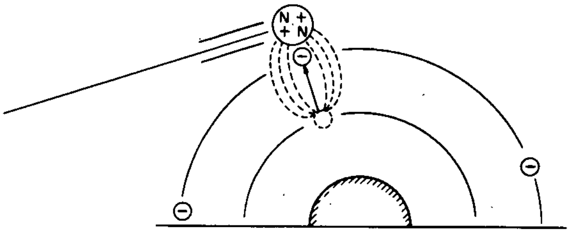


Fig. 3.3. Excitation

1. Specific Ionization

The W value is defined as the average energy needed to create an ion pair in a given substance. This value is different for alpha particles 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 eV/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 ionization is defined as the number of ion pairs formed per cm of path. The actual value of the specific ionization for α particles is not a constant. The specific ionization depends upon the velocity of the alpha particle, as well as on the atomic number and mean ionization potential of the absorbing substance. The range of specific ionization in air extends from about 10,000 to 70,000 ip/cm.

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

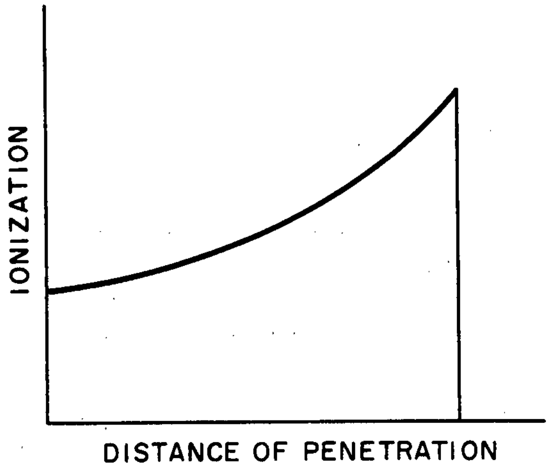


Fig. 3.4

The Ionization Increases as the Velocity of the Alpha Particle Decreases with Depth of Penetration. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

The specific ionization (SP. I.) well within the range of the alpha particle is given by

$$\text{SP.I.} = \frac{dE/dx}{W},$$

where dE/dx is the linear stopping power of the medium (eV/cm). The linear stopping power 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 alpha particles of very low energy. Therefore, one cannot express the specific ionization near the end of the path of the alpha particle.

The average specific ionization can be found from the expression:

$$\overline{\text{SP.I.}} = \frac{\text{No. of ip formed}}{\text{cm of path}} = \frac{E_\alpha}{WR}\ \frac{\text{ip}}{\text{cm}}.$$

Here, E_α is the energy of the α in eV, and R is the range of the α in cm of the substance.

2. Stopping Power

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

$$\frac{dE}{dx} = \frac{4\pi e^4 z^2}{m_0 v^2} NB,$$

where N is the atoms/cc of the medium, B the atomic stopping number (which is a function of the speed v of the particle, the atomic number, and the mean ionization potential of the medium). The linear stopping power is often expressed in units of MeV/cm. The mass stopping power, S/ρ, is the linear stopping power divided by the density ρ of the substance:

$$\frac{S}{\rho} = \frac{1}{\rho}\frac{dE}{dx}\,\frac{\text{MeV}}{\text{gm/cm}^2}\,.$$

In health physics work, a quantity of importance is the ratio of two stopping powers or the relative stopping power. For alpha 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

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

where A is the effective atomic weight.

3. Range

Because of 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 alpha particles in air at 15°C and 76 cm Hg can be found with the aid of the relationships

$$R_a \text{ (in cm)} = 0.56\ E \text{ (for } E < 4 \text{ MeV)}$$

$$= 0.318\ E^{3/2} \text{ (for } 4 < E < 7 \text{ MeV)}.$$

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

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

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

$$B_m/B_{air} \approx \frac{\sqrt{A_m}}{\sqrt{A_a}}\,.$$

From this relationship, then, since

$$\left(\frac{S}{\rho}\right)_m = \frac{B_m A_a}{B_a A_m} \approx \frac{\sqrt{A_m}}{\sqrt{A_a}}\,\frac{A_a}{A_m} = \frac{\sqrt{A_a}}{\sqrt{A_m}},$$

we find for the range in any substance

$$R_m(\text{in cm}) \approx \frac{3.2 \times 10^{-4}\sqrt{A_m}}{\rho_m}\,R_a,$$

where $\sqrt{A_a} = 3.82$ and $\rho_a = 1.226 \times 10^{-3}$ gm/cc at 15°C and 76 cm of Hg. This relationship is usually good within ±15%. The quantity $\sqrt{A_m}$ for a compound or a mixture can be found from the expression

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

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

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

$$R_{air}\,\rho_{air} \approx R_{tissue}\,\rho_{tissue}.$$

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 = 4.195 MeV) in UO_2 (ρ = 10.9 gm/cc).

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

$$R_a = 0.318\,E^{3/2} = 0.318\,(4.195)^{3/2} = 2.73 \text{ cm in air.}$$

$$R_{UO_2} = \frac{3.2 \times 10^{-4}\,(11.52)\,2.73}{10.9} = 9.25 \times 10^{-4} \text{ cm} = 0.009 \text{ mm.}$$

4. Relative Hazard

The fact that alpha particles have short ranges in dense substances makes the alpha particle much less of a hazard to humans than other external radiations. The alpha particle of highest energy emitted by natural radioactive substances will just penetrate the dead layer of skin on the human body. Consequently, for most α emitters, no living tissue is damaged and no external hazard will exist.

On the other hand, once inside the body, the short range of the α particle becomes very important. In this case, the alpha 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. For this reason, alpha 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 idea of beta emission has been extended to include emission of positrons, and beta particles are considered to be positrons and electrons. The rest mass of these particles is the same; they have equal but opposite charges.

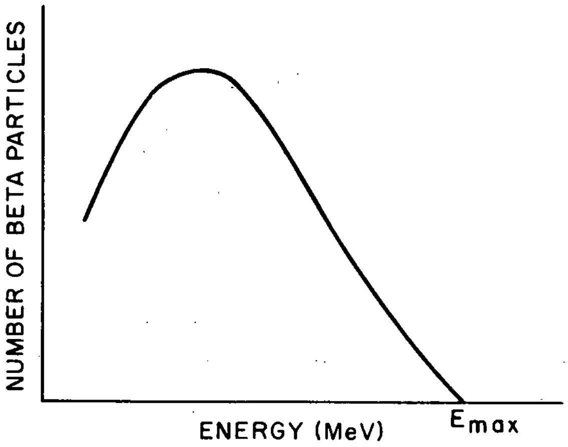


Fig. 3.5. Typical Beta Spectrum. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

Unlike alphas, betas are not emitted with discrete energies but show a continuous energy spectrum (see Fig. 3.5). The atom emits electrons of all energies up to some maximum value. This maximum value for the beta spectrum may be found in nuclide tables and is characteristic of that atom.

The average energy 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^+).

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.

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 alpha. But because of its smaller mass (~ 1/7300 that of an alpha) and lower charge (1/2 that of an alpha), the interactions take place at less frequent intervals. Consequently, electrons do not produce as many ions per centimeter of path as do alphas.

1. Specific Ionization

The W value for electrons in air is 34 eV/ip. The specific ionization values for electrons are much lower than those for alphas. Roughly, the specific ionization for electrons varies inversely as the square of the speed of the electron for energies up to 10 MeV. Specific ionization values in air for electrons range from 60 to 7000 ip/cm.

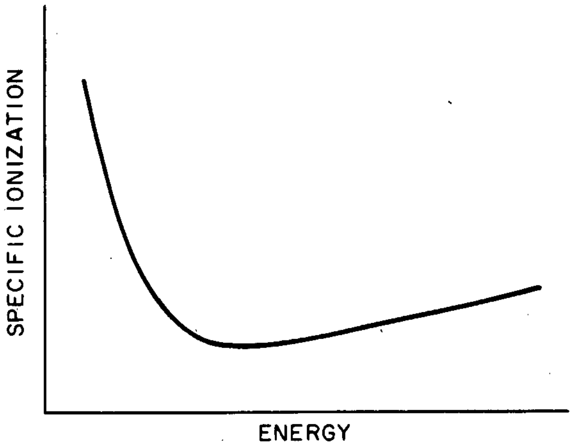


Fig. 3.6. The General Shape of the Specific Ionization vs. Energy Curve for β Particles. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

At low energy, the specific ionization decreases as energy increases (see Fig. 3.6). It reaches a minimum value above 1 MeV. Then, the specific ionization increases as the energy increases due to relativistic effects.

The specific ionization 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 ionization for electrons increases as the energy decreases, the value obtained for the spectrum should be higher than that for monoenergetic electrons.

Similar to the result obtained for alphas, the specific ionization well within the range of an electron is given by

$$\text{SP.I.} = \frac{dE/dx}{W},$$

where dE/dx is the linear stopping power of the substance, usually expressed in eV/cm. Also, the average specific ionization for <u>electrons</u> can be found from the expression

$$\overline{\text{SP.I.}} = \frac{\text{No. of ip formed}}{\text{cm of path}} = \frac{E_e}{WR}\ \frac{\text{ip}}{\text{cm}},$$

where E_e is the energy of the electron in eV and R the range in centimeters of the substance.

The specific ionization of betas is given approximately by

$$\text{SP.I} \approx 33 + 63\,E_{max}^{-0.9} \quad \text{when } 0.05 < E_{max} < 2 \text{ MeV}$$

(see Item 3 in Bibliography for this Section).

2. Stopping Power

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

$$\frac{dE}{dx} = \frac{2\pi e^4 NZ}{m_0 v^2} B',$$

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 X-ray spectrum which is obtained in these interactions is called bremsstrahlung (literally, braking radiation). 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 greater.

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 Fig. 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.

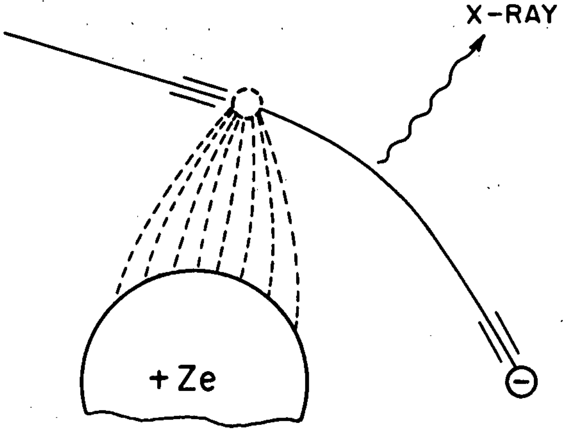


Fig. 3.7. Bremsstrahlung

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.

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

$$F \approx 7 \times 10^{-4} ZE_k,$$

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.

In the case of a beta source, the fraction of beta energy converted to X rays is approximately

$$F \approx 3.33 \times 10^{-4} ZE_{max},$$

where E_{max} is the maximum energy of the beta spectrum. In this case, a 1-MeV beta 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 the maximum energy of the beta for purposes of shielding design. This will normally result in a safety factor in the calculations.

4. Range

The stopping power for electrons decreases as the atomic number Z of the absorber increases. This results because substances of high Z have fewer electrons per gram and these are more tightly bound. 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. These two effects tend 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 mg/cm^2, i.e., $t(\text{cm})\rho(\text{mg/cm}^3)$ = Range (mg/cm^2). The range is then nearly independent of the type of absorbing substance.

Katz and Penfold (Ref. 20) have expressed the relationship between the range (mg/cm^2) and the energy (MeV) of the electron in mathematical form.

$$R\,(\mathrm{mg/cm^2}) = 412\,E^{1.265-0.0954\ln E} \quad (0.01 < E < 2.5\ \mathrm{MeV})$$

or

$$\ln E = 6.63 - 3.2376\,(10.2146 - \ln R)^{1/2},$$

and

$$R\,(\mathrm{mg/cm^2}) = 530\,E - 106 \qquad (E > 2.5\ \mathrm{MeV}).$$

These expressions can be used to find the range of electrons in any substance in terms of mg/cm^2. As far as a continuous beta spectrum is concerned, it turns out that the range of the beta is equal to the range of a monoenergetic electron whose energy is the same as the maximum energy of the beta 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 mg/cm^2 of any substance?

$$R = 412\,(1.5)^{1.265-0.0954\ln 1.5} = 412\,(1.5)^{1.265-0.0954(0.40547)}$$

$$= 412\,(1.5)^{1.265-0.0386} = 412\,(1.5)^{1.226};$$

$$\ln R = \ln 412\,(1.5)^{1.226} = \ln 412 + 1.226\ \ln 1.5$$

$$= \ln 4.12 + 2\ln 10 + 1.226\ \ln 1.5 = 1.41585 + 2\ln 10 + 0.49711$$

$$= 1.91296 + 2\ln 10 = 6.51813;$$

$$R \approx 677\ \mathrm{mg/cm^2}.$$

5. Exponential β-particle Absorption

Suppose beta 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 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 absorption coefficient. The equation describing β absorption is

$$I = I_0 e^{-\mu t},$$

where I_0 is the initial intensity, I is the intensity after passing through a thickness t of absorber (in cm), and μ is the absorption coefficient (in cm^{-1}). Experiments have shown that the mass absorption coefficient $\mu/\rho\,(cm^2/gm)$, where ρ is the density of the substance in gm/cc, is almost independent of the atomic weight of the absorber. Thus, one can express the mass absorption coefficient by

$$\frac{\mu}{\rho} \approx 17\,E_{max}^{-1.43}\ cm^2/gm$$

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

$$I = I_0 e^{-(\mu/\rho)x},$$

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

6. Relative Hazard

A beta source may constitute an external hazard, depending upon the energy of the betas. It takes a beta of about 70-keV energy to penetrate the dead layer of skin and do damage to living tissue. Normally, beta radiation is thought of as only a slight external hazard. A beta source can be fully shielded by a small thickness of aluminum.

An external hazard may result from the production of X rays in the shield in stopping the betas from high-energy sources. The amount of bremsstrahlung can be reduced by choosing a shielding substance of low atomic number Z.

The beta particle is not as great an internal hazard as is the alpha particle. A beta will have a much greater range in tissue, but the specific ionization will be less than that of alpha. Not as much energy is given up in a small volume of tissue as by an alpha. Thus, it will not be as effective in causing damage.

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 the water, one can infer that energy is being propagated. In this case, the wave disturbs the surface of the water.

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. 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 of the wave is given by

$$V \text{ (in cm/sec)} = \nu\lambda,$$

where ν is the number of waves/sec and λ is cm/wave.

D. Electromagnetic Waves

Electromagnetic waves are of concern to the health physicist. When an electromagnetic wave moves through matter, there is set up an electric field E perpendicular to the direction of propagation and a magnetic field H perpendicular to both the electric field and the direction of motion (see Fig. 3.8). Heat, light, X rays, and γ rays are electromagnetic waves which differ only in wave length. Our work concerns X and γ rays.

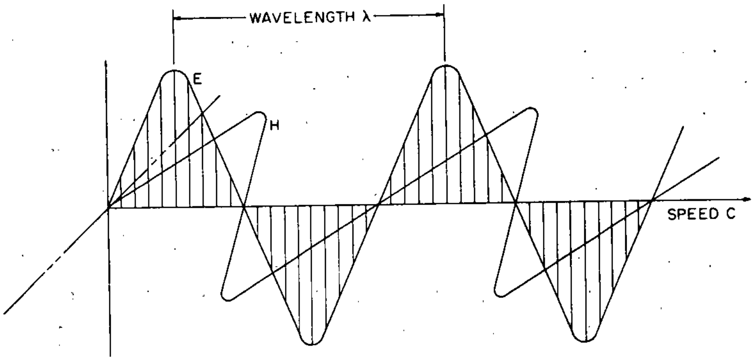


Fig. 3.8. Electromagnetic Wave. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. Ⓒ 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

The speed of an electromagnetic wave in a vacuum is 3×10^{10} cm/sec. 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, Å; one angstrom unit equals 10^{-8} cm.) X and γ rays have wavelengths in the range from 10^{-6} to 10^{-11} cm.

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 6.625 x 10^{-27} erg-sec. The energy E of radiation of frequency ν is given by

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

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 is often given in units of MeV/cm^2-sec. 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

$$I_A \propto 1/r_A^2$$

or

$$I_A = k/r_A^2.$$

At point B, the intensity is similarly given by

$$I_B = \frac{k}{r_B^2}.$$

From these two equations we get

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

or

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

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.

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 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/sec) 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_A, all the energy emitted per unit time will pass through the surface of this sphere. The intensity about the point A is then

$$I \text{ (in MeV/cm}^2\text{-sec)} = S/\text{Surface area of Sphere} = S/4\pi r_A^2 .$$

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.

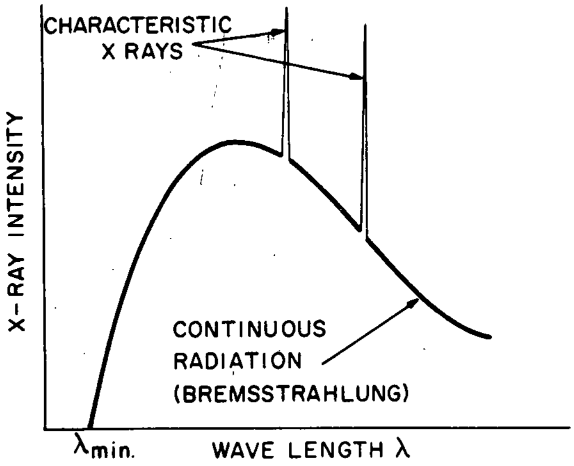


Fig. 3.9. Typical X-ray Spectrum. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. © 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

The same properties were found for gamma 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 then the bremsstrahlung also appear (see Fig. 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.

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 Å. 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 an electron beam strikes a target, a large portion of its energy is lost as heat. Part of the energy is lost in X-ray production, 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

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

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

$$E = Ve,$$

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

$$\lambda_{min}\ (\text{in cm}) = \frac{hc}{Ve} = \frac{(6.6 \times 10^{-27})(3 \times 10^{10})}{\frac{V}{300}\,(4.8 \times 10^{-10})} = \frac{12.4 \times 10^{-5}}{V}$$

or

$$\lambda_{min}\ (\text{in Å}) = 1.24 \times 10^{4}/V.$$

Here, λ_{min} is the wavelength of the highest-energy X ray produced for a given accelerating voltage V. If V is increased, a higher-energy X ray can be obtained. The spectrum of X rays contains only 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. The intensity of the radiation is a function of both the accelerating voltage and the beam current.

Gamma rays are emitted from the nucleus of a radioactive atom. Most atoms which decay by beta emission give off gamma rays. Many atoms which decay by alpha emission also give off gamma rays. Gamma rays are emitted when the nucleus of an atom has an excess energy above its lowest energy state after emitting either an alpha or beta. In fact, these rays may be emitted from any nucleus in an excited state.

1. Interactions with Matter

The transfer of energy to matter by alpha and beta 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.

The three main ways in which X and γ rays 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.

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 as it moves. The photon may strike an electron in an orbit of an atom of the substance. In the photoelectric effect (see Fig. 3.10), all of the 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 in the medium. The energy relationship between the photon and the electron is

$$E = h\nu = E_k + \phi,$$

where E_k is the kinetic energy of 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

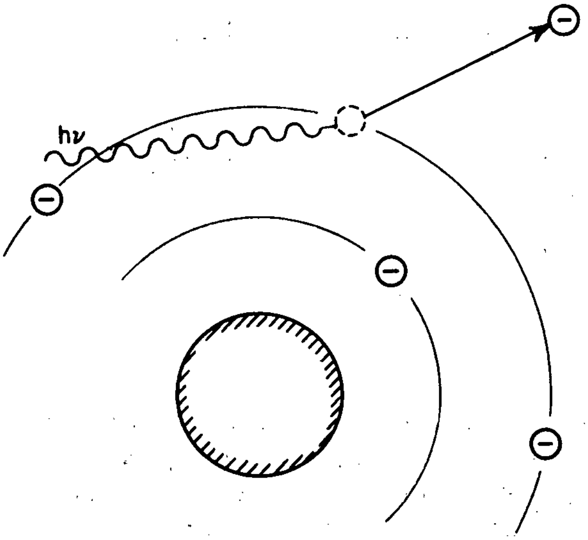


Fig. 3.10. The Photoelectric Effect. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

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).

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 is emitted. If the photoelectron is removed from the K shell, then all of the characteristic X rays may appear. These X rays are often called fluorescent radiation.

Sometimes the photons strike electrons in the outer orbits of an atom and eject these electrons (see Fig. 3.11). These 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 electrons.

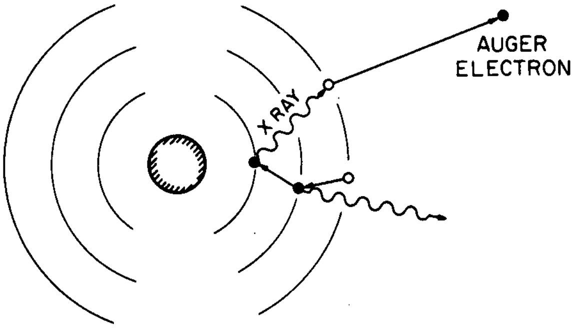


Fig. 3.11. Process Whereby a Photon Ejects an Auger Electron

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 <u>internal conversion</u>. At one time this form of decay was thought to be an internal photoelectric effect. The present view is that it simply is an alternative mode of decay, that is, the excited nucleus may emit a γ 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. Thus, α_K is number of K electrons/number of unconverted γ rays. Thus, the total coefficient α is given by

$$\alpha = \sum_i \alpha_i .$$

3. Compton Effect

In the case of the Compton effect (see Fig. 3.12), a γ ray impinges upon an electron and gives up only part of its energy. The result is that a photon of lesser energy 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.

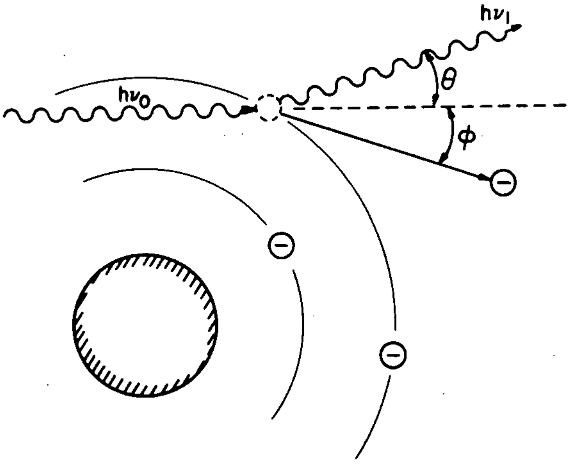


Fig. 3.12. The Compton Effect. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

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

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

(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 not 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 ($\theta = 180°$). In this case, the energy of the electron is given by

$$E_e = \frac{E_\gamma^2}{E_\gamma + \dfrac{m_0c^2}{(1 - \cos\theta)}} = \frac{E_\gamma^2}{E_\gamma + \dfrac{0.51}{2}} = \frac{E_\gamma^2}{E_\gamma + 0.256} \text{ (in MeV)},$$

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

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

The Compton effect is important for gamma-ray energies between about 200 keV and 5 MeV in most light elements. The Compton effect decreases with increasing gamma 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 γ 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 Fig. 3.13) 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 0.51 MeV, the photon must have an energy $\geqq$ 1.02 MeV for pair production to occur.

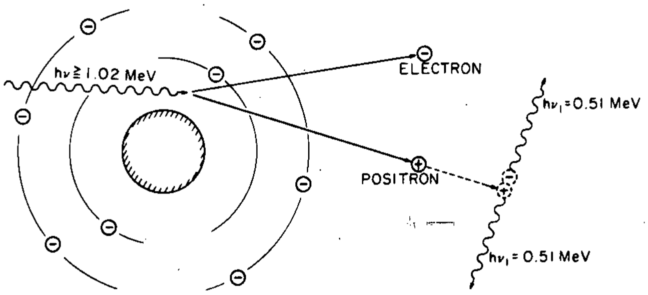


Fig. 3.13. Pair Production and Annihilation. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radition Physics, second edition. © 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

When this process does occur, energy of the photon beyond 1.02 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 process called annihilation. In this process, the mass of the particles is changed into two photons of 0.51 MeV each, emitted in nearly opposite directions. They may interact further in the substance through the photoelectric or Compton effects.

Since this process does not occur at all unless the gamma energy is $\geqq$ 1.02 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.

The three processes: photoelectric effect, Compton effect, and pair production, account for the main photon interactions with matter. Both the photoelectric and Compton effects decrease with increase of gamma energy. Pair production increases with gamma energy. The result of these effects is that each substance will have minimum absorption at some gamma energy.

In health physics, we are concerned with the absorption of radiation by matter so that we can protect people and equipment from these rays. The three processes produce electrons which then 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.

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 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.

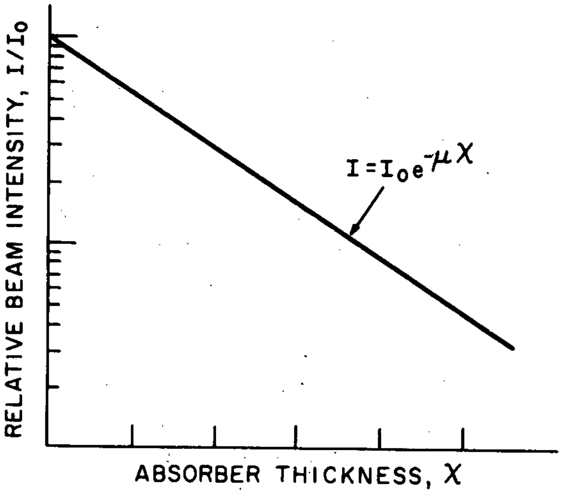


Fig. 3.14. Absorption of X and γ rays. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

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 Fig. 3.14). 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 the equation

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

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

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

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 gamma and the absorbing substance. Thus, μ for a given material is a constant only for a given energy of X and γ radiation. One must consult attenuation coefficient curves in order to find the value of μ for a given energy of X or γ rays in a certain substance. Such curves can be found in the literature (Ref. 11). In these curves, one will more often find the mass attenuation coefficient plotted.

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

$$\mu_m \text{ (in cm}^2\text{/gm)} = \mu/\rho,$$

where ρ is in gm/cc. Thus,

$$\mu = \mu_m\rho \text{ (cm}^{-1}\text{)}.$$

What is the linear attenuation coefficient for a 1-MeV γ in lead? The density of lead is 11.3 gm/cc and $\mu_m = 0.046 \text{ cm}^2\text{/gm}$:

$$\mu = \mu_m\rho = (0.046)(11.3) = 0.52 \text{ cm}^{-1}.$$

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

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

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:

$$(\text{Weight Fraction})_i = k_i A_i / A,$$

where k_i is the number of the 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 γ rays in H_2O. The mass attenuation coefficients for H and O are 0.126 and 0.0635 cm^2/gm, respectively. Then,

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

$$= 0.0635 \frac{k_O A_O}{A_{H_2O}} \rho_{H_2O} + 0.126 \frac{k_H A_H}{A_{H_2O}} \rho_{H_2O}$$

$$= 0.0635 \frac{(1)(16)}{18} (1) + 0.126 \frac{(2)(1)}{18} (1) = 0.0704 \text{ cm}^{-1}.$$

6. Half-value Layer

A quantity of usefulness in quick shielding estimates is the half-value layer. 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

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

7. Mean Free Path

The mean free path λ is the average distance which a photon travels before interacting. The mean free path is equal to the reciprocal of the attenuation coefficient:

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

If the thickness of an absorber is equal to one mean free path, then the intensity of the gamma 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.

8. $\mu - \sigma_s$

Up to this point, all the concepts have been based upon a narrow beam of γ rays. In this case, if a gamma 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 wide beam of radiation (see Fig. 3.15), 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. The question arises whether as many are scattered into the path as are scattered away from it. The net affect is difficult to assess.

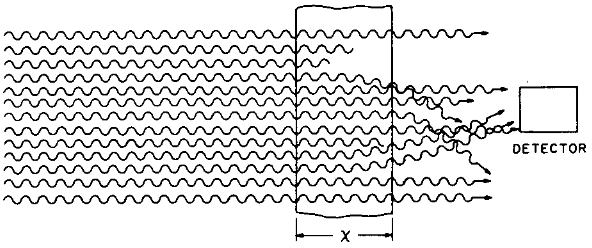


Fig. 3.15. Wide Beam of γ Rays in an Absorber

One approach to this problem assumes that all scattered radiation remains along the path. This is done by taking the total coefficient μ and subtracting the probability σ_s that scattering occurs. This new coefficient, $\mu - \sigma_s$, is then used in place of μ. The attenuation of the beam is then given by

$$I = I_0 e^{-(\mu - \sigma_s)x}.$$

By using $\mu - \sigma_s$, one assumes that any energy scattered out of the path is replaced by energy scattered into the path. This new coefficient gives the probability of energy being absorbed per unit path length.

In most cases of a wide beam of radiation passing through an absorbing substance to a detector, some of the scattered radiation is lost from the beam and never reaches the detector. The use of the coefficient $\mu - \sigma_s$ in these cases leads to an overestimation of the flux reaching a detector. The use of the coefficient μ leads to an underestimation of the flux reaching a detector. In health physics work, the coefficient $\mu - \sigma_s$ is preferred since it will introduce a safety factor in calculations.

9. Buildup Factor

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

$$b = \frac{\text{true flux}}{\text{calculated flux}} = \frac{I}{I_0 e^{-\mu x}}$$

or

$$I = I_0 b e^{-\mu x}.$$

In this case one can calculate the true flux I if the value of the buildup factor is known. The buildup factor corrects for the underestimation of scattered radiation reaching the detector. Curves of buildup factors are available in the literature. The use of the buildup factor concept will be discussed in the section on shielding.

10. Relative Hazard

X and γ rays are of most concern as potential external hazards. These rays have great ranges in air and other substances, and may present a significant hazard at fairly large distances from a source. In most cases shielding is needed to insure that no hazard exists. The shield which is required is much greater than for either alpha or beta radiation. Also, scattered rays from a source may contribute to the hazard, that is, a shield may provide good protection in one direction, but scattered rays from this shield may cause a hazard to exist in another area. Any cracks or breaks in an otherwise adequate shield may lead to streaming or beams of radiation.

Because X and γ rays do penetrate substances to such a high degree, the damage done to tissue will extend throughout the body. Thus, the more radiosensitive tissues are exposed when one is in an external gamma field. Damage to these organs affects the body more than damage to less critical tissues, such as skin. For this reason, X and γ rays are more of an external hazard than either alpha or beta rays.

On the other hand, X and γ radiation is not as great an internal hazard as is alpha or beta radiation. As is the case with beta radiation, the long range of gamma in tissue means that less energy will be given up in a small volume of tissue. A photon will give up energy only when it interacts. It does not produce a continuous path of ionization as do alpha and beta particles. For this reason, a small organ in the body will receive less damage from a gamma source than an equal alpha or beta source.

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 alpha 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 beta emission ($E_{max} = 0.78$ MeV) with a half-life $T_{1/2} \approx 12.8$ min.

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.

Some of the properties of these sources are:

1. (α,n) reactions - These sources are often prepared by making an intimate mixture of an alpha emitter and finely divided powder of the target substance. ^{226}Ra, ^{210}Po, and ^{239}Pu are the common alpha emitters 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. Source strength is often expressed in terms of the amount of the α emitters.

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 gamma activity, but give a lower yield than Ra-Be. Po-Be sources also have a low gamma 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.

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 on page 10, NBS Handbook #72.

2. (γ,n) reactions - Some nuclei will emit neutrons when exposed to gamma rays. Most sources do not emit gamma rays with energy greater than 4 MeV. The target substances are thus limited to beryllium and deuterium.

The gamma background is always high in these sources. A further drawback is the short half-life of the gamma emitters which are used. The emission rate for these sources is closely proportional to the amount of the gamma emitter. These sources emit nearly monoenergetic neutrons.

The characteristics of some (γ,n) sources are given on page 11, NBS Handbook #72.

3. 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). 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.

4. Reactors - The fission process in reactors produces neutrons with a spectrum of energy. A reactor provides a significant source of neutrons since, on the average, 2.5 neutrons are emitted per fission.

5. Spontaneous fission sources - The neutron spectra from these sources are similar to that produced in reactors. However, their expense precludes their use as standard sources at the present time. The characteristics of these sources are given on page 12, NBS Handbook #72.

2. Neutron Energy

The types of interactions which neutrons undergo depend quite strongly on the energy of the neutron. As in the case of gamma rays, certain effects predominate depending on the energy of neutrons. It is often advantageous to treat neutron interactions in terms of the energy range in which they predominate. The NCRP has classified neutrons according to the following energy scheme:

1. 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 2.2×10^5 cm/sec (2200 m/sec). The energy of thermal neutrons is then

$$E = \frac{1}{2}mv^2 = \frac{1}{2}(1)(1.66 \times 10^{-24})(2.2 \times 10^5)^2 = 4.02 \times 10^{-14} \text{ ergs}$$

$$= 0.025 \text{ eV}.$$

2. Intermediate neutrons - This energy region of neutrons goes from 0.5 eV to 10 keV. Neutrons in this interval are often called "resonance neutrons."

3. Fast neutrons - Fast neutrons range in energy from 10 keV to 10 MeV.

4. Relativistic neutrons - This range includes neutrons of energy greater than 10 MeV.

3. Interactions with Matter

The study of neutron interactions with matter 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 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. Since the advent of high-energy accelerators, more of these reactions are being found and studied.

For fast neutrons, elastic scattering 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 is pictured as a billiard-ball type of collision in which energy and momentum will be conserved (see Fig. 3.16).

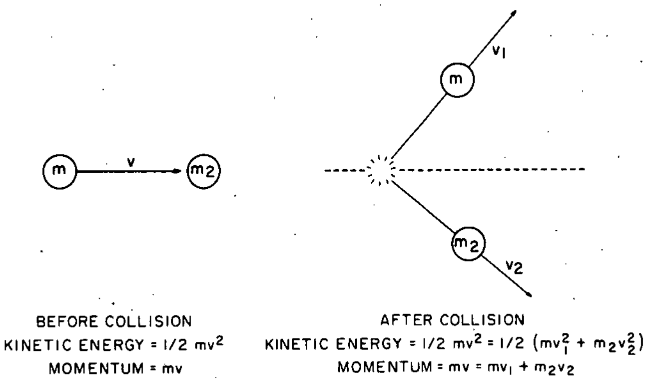
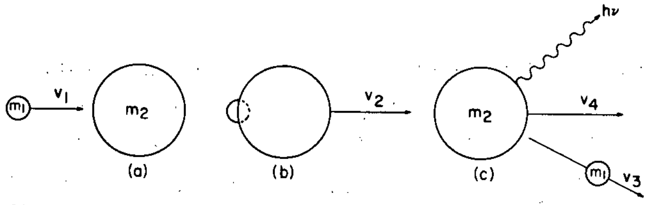


Fig. 3.16. Elastic Collision. Energy and momentum are conserved.

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 of energy E_n, the average energy lost in a single collision with hydrogen is $\sim 2/3\ E_n$. As the mass of the nucleus becomes very large, the neutron will lose hardly any energy. For this reason, substances with high Z are poor for use in slowing down neutrons.

In the energy range above 0.5 MeV, inelastic scattering begins to occur. This process is pictured as a putty-ball type of collision (see Fig. 3.17). In the process, the neutron raises the nucleus to an excited state and comes off with a loss in energy. The excited nucleus often returns to the ground state by emitting the excess energy as a gamma 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).



BEFORE COLLISION

MOMENTARILY, BOTH MASSES ARE UNITED

AFTER COLLISION EXCESS ENERGY IS EMITTED AS ELECTROMAGNETIC RADIATION

Fig. 3.17
Inelastic Collision

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 elastic scattering is still dominant for intermediate neutrons. The phenomenon of resonance absorption also occurs in this region. When a neutron has an energy which is equal to a nuclear energy level of the absorbing substance, the probability of absorption increases greatly. Thus the substance will absorb more neutrons of this energy than of 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 neutron is further decreased, it approaches the thermal-neutron range. The dominant process for thermal neutrons is capture, in which the neutron becomes part of the absorbing nucleus. The compound nucleus thus formed must then get rid of excess energy, usually by emission of gamma 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 leads 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 which causes ionization in the medium. In the case of capture or nuclear reactions, ionization occurs due to the emission of charged particles from the excited nucleus. If an (n, γ) capture takes place, the resulting ionization will depend upon the gamma interacting in the medium. The loss of energy by neutrons, as well as by gamma rays, is not a continuous process as it is for alpha and beta particles. The neutron, or the gamma ray, will move through matter with little interference until an interaction takes place. When this occurs, then energy will be lost.

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^{-24} cm^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:

$$\sigma_t = \sigma_{\text{scatter}} + \sigma_{\text{capture}} + \sigma_{\text{fission}} + \cdots\cdot$$

One should not construe that the cross section is 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: 1.5×10^{-13} cm. For an impact to just occur, the particle centers cannot be more than 3×10^{-13} cm apart (see Fig. 3.18). The impact area would then be a circle of area $\pi(3 \times 10^{-13})^2 = 28.3 \times 10^{-26}$ cm^2. But the cross section for capture in hydrogen for 0.01-eV neutrons is 48 b, or 4800×10^{-26} cm^2.

It is clear that the effective area is much greater than the impact area. One can see from this that σ is much more than simply a geometric area. Curves for σ_t can be found in the literature.(21,22)

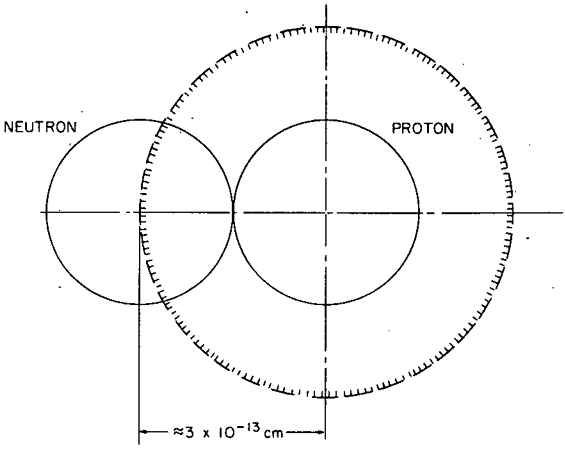


Fig. 3.18
Impact Area

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 macroscopic cross section Σ, is desired. The total macroscopic cross section Σ_t is related to the total microscopic cross section σ_t by the equation:

$$\Sigma_t = N\sigma_t \text{ cm}^{-1},$$

where N is the number of atoms per cubic centimeter of the substance.

Let us picture a cubic centimeter of a substance in terms of a cross-sectional area of 1 cm^2 and a thickness 1 cm. 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 1-cm thickness.

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. By analogy, the mean free path λ for neutrons is given by

$$\lambda = 1/\Sigma_t.$$

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

5. Neutron Absorption

Assume a narrow beam of neutrons is passing through a substance. Let k be the number of neutrons per cubic centimeter in the beam. Assume, further, that each neutron has a speed v. Then the neutron flux ϕ is given by

$$\phi = kv \text{ (in n/cm}^2\text{-sec)}.$$

The flux gives the number of neutrons passing through one square centimeter per unit time. In terms of our previous picture of one cubic centimeter of a substance, the probability of a reaction in passing through 1 cm of the substance is given by Σ_t. The number of reactions per unit time in this centimeter of the substance will be

$$\frac{\text{number of reactions}}{\text{cc-sec}} = \phi\Sigma_t.$$

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

$$\Delta\phi = -\phi\Sigma_t\Delta x.$$

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:

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

This form is the well-known expression for an exponential relationship. When integrated, then is obtained

$$\phi = \phi_0 e^{-\Sigma_t x},$$

where ϕ_0 is the initial flux 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.

As is the case with gamma rays, this type of relationship is valid only if one can consider the scattered neutrons as being removed from the beam. For a narrow beam, this condition may be assumed. For a wide beam, some of the 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.

6. Removal Cross Section Σ_R

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 Σ_R is then used to calculate neutron attenuation.

The macroscopic removal cross section is given by

$$\Sigma_R = \frac{0.602\,\rho}{A}\,\sigma_r \text{ (in cm}^{-1}\text{)},$$

where σ_r is the microscopic removal cross section in b/atom. It has been found that σ_r is about equal to 3/4 of the total cross section, σ_t, at 8 MeV, except for hydrogen.

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. The concept is valid for fast neutrons in the range 2-12 MeV. In this energy range, neutrons which suffer inelastic collisions in a heavy substance will lose a large part of their energy. In this case, they then may be captured by hydrogen. If they suffer an elastic collision with a heavy nucleus, they lose hardly any energy. However, an elastic collision with hydrogen causes a large transfer of energy. In this case 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.

If the quantity Σ_R/ρ is plotted against the atomic weight A of an element, a smooth curve results. One can obtain the removal cross section for any element from this plot. Such a curve can be found in NBS Handbook #63 on page 79. 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

$$\Sigma_R(\text{compound}) = \sum_{i=1}^{n} (\Sigma_R/\rho)_i \rho_i,$$

where $(\Sigma_R/\rho)_i$ is the value obtained from the curve in Handbook #63 for the i^{th} element, and ρ_i is the density of the i^{th} element as it appears in the compound.

The concept of removal cross section is based on data obtained with thick shields. When thin shields are used, scattered neutrons will not be removed as well. For this type of problem, a buildup factor must be used. The attenuation expression will then be

$$\phi = \phi_0 b e^{-\Sigma_R x}.$$

The buildup factor for neutrons will be discussed in Section 5.

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 may then become radioactive. Many artificially radioactive substances are produced when other substances are exposed to the neutron flux in a reactor. Absorption occurs mainly for thermal neutrons. For this reason, most artificial radioisotopes have been produced in reactors. When a nucleus absorbs a thermal 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.

Assume a sample containing N atoms is exposed to a thermal neutron flux 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

$$\text{formation rate} = \phi \sigma_{act} N.$$

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

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

where N^1 is the number of radioactive atoms present and λ is the decay constant. The net rate of growth is thus rate of growth = formation rate - decay rate.

The corresponding differential equation is

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

If the flux is constant for the time t that the sample is irradiated, the equation can be integrated to give

$$N_t^1 = \frac{\phi\sigma_{act}N}{\lambda}(1 - e^{-\lambda t}),$$

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

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

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

$$A = A_t e^{-\lambda t_1} = \phi\sigma_{act}N(1 - e^{-\lambda t})e^{-\lambda t_1}.$$

The activity A_t of the sample in the flux 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 $t = 0$ to 1 for $t = \infty$. For any time t, this term gives the fraction of A_s which the irradiated sample reaches.

The equation is valid if all target atoms receive the same flux. In effect, this means that 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.

Let us take an example of the use of this expression. A one-gram tungsten sample is exposed for 3 days at an average thermal-neutron flux of 10^{11} n/cm^2-sec in a reactor. Given that σ_{act} is 40 b and that the half-life of ^{187}W is 24 hr. What is the activity of the sample 12 hr after irradiation? Also, determine A_s.

$$A = \phi\sigma_{act}N(1 - e^{-\lambda t})e^{-\lambda t_1}.$$

$$N = (m/A)N_a = (1/186)6.025 \times 10^{23}; \quad t = 3\text{ d}, \quad t_1 = 0.5\text{ d}.$$

$$\sigma_{act} = 40 \text{ b} = 40 \times 10^{-24} \text{ cm}^2.$$

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

$$\phi = 10^{11} \text{n/cm}^2 \text{ sec}.$$

Therefore,

$$A = 10^{11}(40 \times 10^{-24})\left(\frac{6.025 \times 10^{23}}{186}\right)(1 - e^{-0.693(3)})\, e^{-0.693(0.5)}$$

$$= 8.02 \times 10^9 \text{ dps} = 0.217 \text{ Curie}$$

and

$$A_s = \phi\sigma_{act}N = 10^{11}(40 \times 10^{-24})\left(\frac{6.025 \times 10^{23}}{186}\right)$$

$$= 1.296 \times 10^{10} \text{ dps} = 0.35 \text{ Curie}$$

8. Relative Hazard

Along with gamma rays, neutrons are of most concern as potential external hazards. Since neutrons possess no charge, they travel great distances in air and other substances. Thus, it is more difficult to provide shielding for neutrons than for alpha or beta particles. As for gamma rays, scattered neutrons can add to the hazard.

Because the neutron moves through matter rather freely until it interacts, the damage done in tissue extends throughout the body. Just as for gamma rays, the more radiosensitive tissues will receive damage from an external neutron field. The damage done in tissue is a function of neutron energy. Fast and relativistic neutrons (energy <30 MeV) produce many elastic collisions in tissue. Since almost 2/3 of the body atoms are hydrogen, about 80-95% of the energy given up in tissue by fast neutrons occurs in elastic scattering by hydrogen. The recoil protons which are produced will lose their kinetic energy by ionizing tissue. For this reason, fast neutrons are more hazardous than neutrons of lower energy.

Intermediate and thermal neutrons produce damage in tissue as the result of absorption. In this case, the $H(n,\gamma)D$ and $^{14}N(n,p)^{14}C$ reactions are important. For a large mass of tissue, the $H(n,\gamma)D$ process is more important then the $^{14}N(n,p)^{14}C$ process. For a small mass, the reverse is true. Since a gamma 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 a gamma. For a large organ, the gamma dose will be greater.

Since some of the neutron interactions can produce heavy recoil particles, protons, and gammas, the transfer of energy to tissue will vary. However, the effectiveness of neutrons in producing damage is from 2.5-10 times that of gamma rays. On this basis, the neutron is more hazardous than gamma rays. However, each radiation is significant as an external hazard; one cannot ignore the gamma field if a neutron field is also present.

The neutron is not treated as an internal hazard. Since there are no natural neutron-emitting sources, there is little chance of a neutron source being fixed in the body.

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SECTION 4 - CONCEPTS OF RADIATION UNITS AND DOSE DETERMINATIONS

A. Definition of Terms

The organization which selects and defines the units and quantities 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. Early reports of this group giving a history of the development of the units can be found in the literature (Ref. 1-9). The current reports of the ICRU can be found in NBS Handbooks 84-89. Much of the information used in this section is taken from Handbook 84.

It is the desire of the ICRU to restrict the use of the term dose. In previous reports, three types of doses were discussed, namely, exposure dose, absorbed dose, and RBE dose. The term exposure is now used in place of exposure dose. The term absorbed dose is retained. Dose equivalent replaces the term RBE dose. The definitions of these terms are:

1. Exposure - The exposure X is the quotient of ΔQ by Δm, where ΔQ is 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:

$$X = \Delta Q/\Delta m.$$

The special unit of exposure is the Roentgen (symbolized as R): 1 R = 2.58×10^{-4} C/kg, where C stands for Coulomb. This unit is numerically equal to the old value for R, 1 esu/cc of air. The use of the Roentgen is restricted solely to exposure.

2. Absorbed Dose - The absorbed dose D is the quotient of ΔE_D by Δm, where ΔE_D is the energy imparted by ionizing radiation to the matter in a volume element containing mass Δm

$$D = \Delta E_D/\Delta m.$$

The special unit of absorbed dose is the rad. Its use is reserved solely for absorbed dose. By definition, 1 rad = 100 ergs/gm = 10^{-2} Joule/kg.

3. Dose Equivalent - The dose equivalent (denoted as DE) is defined as the product of absorbed dose and other necessary modifying factors. The ICRU wishes to reserve the term RBE for use in radiobiology only. The term quality factor (symbolized as QF) should be used for protection purposes. The product of an absorbed dose and a suitable quality factor expresses the irradiation in terms of a common scale for all ionizing

radiations. The distribution factor (denoted as DF) is used to correct for nonuniform distribution in the case of internally deposited radionuclides. Thus, the dose equivalent is given by

$$DE = D(QF)(DF)....$$

DE is numerically equal to the dose in rads times the appropriate factors. The unit of dose equivalent is the rem.

B. Quality Factor

For the present, the values of the quality factors which are to be used correspond to the previous values of RBE used for protection work (see Ref. 22). However, there is a need to distinguish these values from the RBE values used in experimental work. Hence the new name quality factor.

In the future, these values may be revised. One limitation already present is that the QF values are not applicable in the case of acute exposures at high dose rates, such as in an accident. These values apply only for the case of normal protection work. The values in present use are:

Type of Radiation	Quality Factor (QF)
X ray, γ ray, β^-, β^+	1
β^-, β^+ for $E_{max} \leqq 0.03$ MeV	1.7
α particles	10
heavy recoil atoms	20
neutrons (See page 89 below)	

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 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 volume 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 volume of air then, the electrons which are produced will give up part of their energy outside this volume before being stopped. However, this loss of electrons may be compensated by electrons entering the given volume 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 electronic equilibrium is reached, that is, the energy imparted to a given small volume of air by all the electrons which traverse this volume equals the energy imparted to the medium along the total path of all electrons produced in the volume by the photons.

The collection of the ions which are produced in a given small volume of air under equilibrium conditions is then a measure of the exposure. The exposure may be expressed in units of 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. Because of historical reasons, the Roentgen is retained as a special unit of exposure. In the past, one Roentgen corresponded to 1 esu/cc of air. In terms of the new units for exposure,

$$1\ \text{R} = \frac{1\ \text{esu}}{1\ \text{cm}^3\ \text{of air}}\left(\frac{1\ \text{C}}{3\times 10^{9}\ \text{esu}}\right)\left(\frac{1\ \text{cm}^3\ \text{of air}}{1.293\times 10^{-6}\ \text{kg}}\right) = 2.58\times 10^{-4}\,\frac{\text{C}}{\text{kg}},$$

in which 1 cc of air at 0°C and 76 cm Hg = 1.293×10^{-3} gm.

D. Exposure Rate - Isotropic Point Source

Assume an isotropic point source of C Curies of a γ emitter of energy E_γ at the point 0 (see Fig. 4.1). Our interest is to obtain an expression for the exposure rate at the point A, distant r from the source, in terms of Roentgens per unit time. Since the source is isotropic, the intensity I_A at the point A will be given by

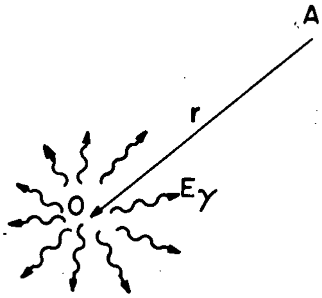

Fig. 4.1
Isotropic Point Source

$$I_A = S/4\pi r^2,$$

where S is the total energy per second emitted by the source. If S is expressed in MeV/sec,

$$S = 3.7\times 10^{10}\left(\frac{\text{dis}}{\text{sec-Ci}}\right) n_\gamma\left(\frac{\text{fraction of }\gamma}{\text{dis}}\right)$$

$$C(\text{Ci})\cdot E_\gamma\left(\frac{\text{MeV}}{\gamma}\right) = 3.7\times 10^{10}\, n_\gamma C E_\gamma.$$

Then

$$I_A = 3.7\times 10^{10} n_\gamma C E_\gamma/4\pi r^2 = 2.944\times 10^{9}\, n_\gamma C E_\gamma/r^2 \text{ in } (\text{MeV/cm}^2\text{-sec}).$$

The exposure X is defined in terms of the ionization produced in a volume of air. The exposure rate is given by

$$\text{Exposure rate} = \Delta X/\Delta t.$$

A special unit of exposure rate is the R/hr.

To find the ions produced in air, the energy given up by the beam must be known. The term I_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 transferred to the substance. The product $I_A\ {}_{en}/\rho$ gives the amount of energy transferred per unit time per unit mass. If one assumes that all this energy is used to produce ion pairs, then

$$\frac{\text{ion pairs}}{\text{gm-sec}} = \frac{I_A}{W}\left(\frac{\mu_{en}}{\rho}\right) 10^6,$$

where W = 34 eV/ip for electrons in air, μ_{en}/ρ is equivalent to $(\mu - \sigma_s)/\rho$ (cm^2/gm) as given in Section 3, and the factor 10^6 converts MeV to eV. This leads to

$$\frac{\text{ion pairs}}{\text{gm-sec}} = 2.944 \times 10^{15}\,\frac{n\gamma C E\gamma}{34 r^2}\left(\frac{\mu_{en}}{\rho}\right) = 8.66 \times 10^{13}\,\frac{n\gamma C E\gamma}{r^2}\left(\frac{\mu_{en}}{\rho}\right).$$

Conversion to exposure rate units gives

$$8.66 \times 10^{13}\,\frac{n\gamma C E\gamma}{r^2}\left(\frac{\mu_{en}}{\rho}\right)\left(\frac{\text{ip}}{\text{gm-sec}}\right) 3.6 \times 10^3 \left(\frac{\text{sec}}{\text{hr}}\right)$$

$$1.6 \times 10^{-19}\left(\frac{C}{\text{ip}}\right)\left(\frac{10^3\ \text{gm}}{\text{kg}}\right) = 49.882\,\frac{n\gamma C E\gamma}{r^2}\left(\frac{\mu_{en}}{\rho}\right),$$

in C/kg-hr. Then,

$$\text{Exposure rate in R/hr} = 49.882\,\frac{n\gamma C E\gamma}{r^2}\left(\frac{\mu_{en}}{\rho}\right)\frac{C}{\text{kg-hr}}\,\frac{1}{2.58 \times 10^{-4}}\left(\frac{\text{kg - R}}{C}\right) =$$

$$1.93 \times 10^5\,\frac{n\gamma C E\gamma}{r^2}\left(\frac{\mu_{en}}{\rho}\right)$$

from an isotropic point source of a gamma emitter of activity $n\gamma C$ (in Ci), energy $E\gamma$ (in MeV), and mass energy-absorption coefficient μ_{en}/ρ, at a point a distance r cm 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

$$e^{-\sum_{i=1}^{j} (\mu_{en}/\rho)_i x_i} = e^{-(\mu_{en}/\rho)_1 x_1} e^{-(\mu_{en}/\rho)_2 x_2} \ldots e^{-(\mu_{en}/\rho)_j x_j},$$

where x_i is expressed in gm/cm^2.

The general expression becomes

$$\text{Exposure rate (in R/hr)} = 1.93 \times 10^5 \frac{n_\gamma C E_\gamma}{r^2}\left(\frac{\mu_{en}}{\rho}\right) e^{-\sum_{i=1}^{j} (\mu_{en}/\rho)_i x_i}.$$

If the only substance of concern is air, then certain simple relations can be deduced. For example, the attenuation in a few feet of air is negligible and no attenuation factor need be included. Also, μ_{en}/ρ can be taken as 0.0275 cm^2/gm in the range $0.2 < E_\gamma < 2$ MeV, so that

$$\text{Exposure rate (in R/hr)} \cong 6\, n_\gamma C E_\gamma \text{ at 1 ft and } 0.2 < E_\gamma < 2.0 \text{ MeV}$$

$$\text{Exposure rate (in R/hr)} \cong 0.53\, n_\gamma C E_\gamma \text{ at 1 m and } 0.2 < E_\gamma < 2.0 \text{ MeV}.$$

These simple formulae are often 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 μ_{en}/ρ for the gamma obtained.

E. Exposure Rate - Flux Density

Along the same line, one may wish to express the exposure rate in terms of the flux density of photons in a beam of radiation. If one assumes ϕ photons/cm^2-sec of energy E_γ in the beam, the resulting expression is

$$\text{Exposure rate (in R/hr)} = 6.57 \times 10^{-5}\, \phi E_\gamma (\mu_{en}/\rho),$$

where E_γ is in MeV and μ_{en}/ρ is in cm^2/gm. In this case, no attenuating substance is included, and the exposure rate as given would be well within the beam dimensions to avoid edge effects.

F. Absorbed Dose

The exposure concept is concerned with the production of ions. The concept of absorbed dose focuses attention on the energy imparted to matter in a volume element. Since one can presume that the biological effects result only 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. The ICRU defines energy imparted as the difference between the sum of all energy entering the volume and the sum of all energy leaving the volume, minus the energy equivalent of increased rest mass from nuclear reactions within the volume (see Ref. 10).

Whereas the exposure refers to photons, the absorbed dose may be used for any ionizing radiation. Thus the concept can be used for α, β, γ, and neutrons.

G. Absorbed Dose Rate - Isotropic Point Source of Photons

Consider again an isotropic point gamma-emitting source. As we saw in Section D above, $I_A\ (\mu_{en}/\rho)$ gives the amount of energy transferred per unit time per unit mass. If this absorbed dose rate in air is given in MeV/gm-sec, we may convert it as follows:

$$\frac{\text{ergs}}{\text{gm-hr}} = I_A \left(\frac{\mu_{en}}{\rho}\right)\left(\frac{\text{MeV}}{\text{gm-sec}}\right) 1.6 \times 10^{-6} \left(\frac{\text{ergs}}{\text{MeV}}\right) 3.6 \times 10^{3} \left(\frac{\text{sec}}{\text{hr}}\right).$$

Since the rad is a special unit of absorbed dose, we also have

$$\frac{\text{rad}}{\text{hr}} = 5.76 \times 10^{-3}\, I_A \left(\frac{\mu_{en}}{\rho}\right)\left(\frac{\text{ergs}}{\text{gm-hr}}\right) \frac{1}{100} \left(\frac{\text{gm-rad}}{\text{ergs}}\right)$$

$$= 5.76 \times 10^{-5}\, \frac{2.944 \times 10^{9}\, n_\gamma C E_\gamma}{r^2} \left(\frac{\mu_{en}}{\rho}\right)$$

$$= 1.7 \times 10^{5} \frac{n_\gamma C E_\gamma}{r^2} \left(\frac{\mu_{en}}{\rho}\right).$$

This gives the absorbed dose rate from an isotropic point source of a gamma emitter of activity $n_\gamma C$ Curies, energy E_γ, and mass energy-absorption coefficient μ_{en}/ρ, at a point a distance r from the source. The general expression for the absorbed dose rate in rad/hr is then

$$\frac{\text{rad}}{\text{hr}} = 1.7 \times 10^{5} \frac{n_\gamma C E_\gamma}{r^2} \left(\frac{\mu_{en}}{\rho}\right) e^{-\sum_{i=1}^{j} (\mu_{en}/\rho)_i x_i}$$

The relationship between the absorbed dose rate and the exposure rate at any point in a medium irradiated by photons is

$$\frac{\text{rad}}{\text{hr}} = (0.869)\left[\frac{(\mu_{en}/\rho)_{\text{medium}}}{(\mu_{en}/\rho)_{\text{air}}}\right]\left(\frac{\text{R}}{\text{hr}}\right).$$

From this, if the only substance is air, one gets

$$\text{rad/hr} = 0.869\ (\text{R/hr}).$$

H. Absorbed Dose Rate - Flux Density of Photons

The absorbed dose rate in terms of the flux density of photons in a beam of radiation is given by

$$\text{rad/hr} = 5.76 \times 10^{-5}\ \phi E_\gamma\, \mu_{en}/\rho,$$

where ϕ is the flux density in photons/cm^2-sec, E_γ is the energy of the photons in MeV; and μ_{en}/ρ is in cm^2/gm. This expression is valid well within the beam dimensions.

I. Absorbed Dose Rate - Isotropic Point Source - Alpha and Beta Radiation

If an isotropic point source of C Curies is an alpha or a beta emitter, then the absorbed dose rate at the point A (see Fig. 4.1) is given by

$$\frac{\text{rad}}{\text{hr}} = \frac{0.17\ n_p C}{r^2}\left(\frac{S}{\rho}\right),$$

where n_p is the fraction of particles of energy E (in MeV) and S/ρ is the mass stopping power for a particle of energy E in eV/(gm/cm^2). Since neither alpha nor beta radiation has 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.

J. Mass Stopping Power

For purposes of calculation, the mass stopping power S/ρ for use in the previous formula is given by K. Z. Morgan (Ref. 23). For alpha particles,

$$\left(\frac{S}{\rho}\right)_\alpha = \frac{(\text{SP.I.})_{\text{air}} W}{\rho_{\text{air}}}\left(\frac{S}{\rho}\right)_m \left(\text{in } \frac{\text{eV}}{\text{gm/cm}^2}\right),$$

where $(SP.I.)_{air}$ is given in ip/cm, ρ_{air} is the density of air at which the specific ionization was measured, W is 35 eV/ip for alpha particles in air, and $(S/\rho)_m$ is the relative mass stopping power of the medium with respect to air.

Figure 4.2, based on data from Ref. 23, gives the mass stopping power for alpha radiation in air in terms of $MeV/(gm/cm^2)$ as well as the relative mass stopping power $(S/\rho)_t$ of tissue with respect to air. For substances other than tissue,

$$(S/\rho)_m \approx \sqrt{A_{air}}/\sqrt{A_{medium}}$$

(see Section 3).

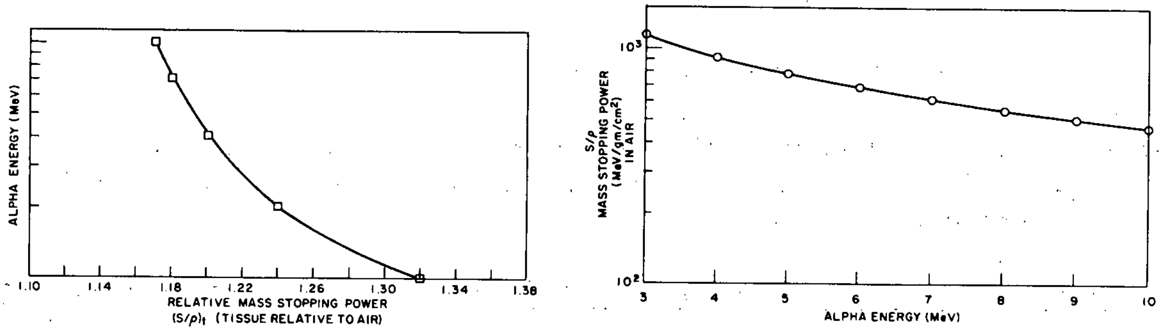

Fig. 4.2. Mass Stopping Power of Alpha Radiation as a Function of Energy (Based on information in Radiation Hygiene Handbook, section 14, McGraw-Hill Book Co., Inc., New York, 1959)

This information allows one to calculate roughly the absorbed dose rate from an alpha source, but only well within the range of the particle in the medium. The reason for this is that the specific ionization 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 alpha particle, the specific ionization, as well as the stopping power, changes very rapidly, and the above information is no longer of value.

In the case of beta radiation, a further difficulty is encountered, namely, the existence of a spectrum of energy. Also, the scattering of beta particles is quite extensive. In Section 3, we saw that, well within the range of the beta, the absorption could be taken as exponential. The mass absorption coefficient for most substances was given as

$$\mu/\rho \approx 17\ E_{max}^{-1.43}\ (cm^2/gm)$$

for $0.05 < E_{max} < 5$ MeV. The mass stopping power for beta radiation may then be roughly expressed, in $eV/gm\text{-}cm^2$, as

$$(S/\rho)_\beta \approx 5.67 \times 10^6 E_{max}^{-0.43} e^{-17(E_{max}^{-1.43})x},$$

where E_{max} is in MeV and x is in gm/cm² of the medium. Again, these expressions are of value only well within the range of the beta.

Although these expressions for the stopping power of alpha and beta radiations are only rough approximations, the need for more accurate expressions is not grave. Since the range of charged particles from radioactive substances in most materials is quite short, neither alpha nor beta radiation presents much concern as external sources of radiation. For this reason, as well as the fact that these radiations are easily shielded, it is not expedient that one have exact expressions.

K. Absorbed Dose Rate - Flux Density of Particles

The absorbed dose rate for a flux density of alpha or beta particles in a beam of radiation is given by

$$\text{rad/hr} = 5.76 \times 10^{-11} \phi_p (S/\rho),$$

where ϕ_p is the particles/cm²-sec and S/ρ is the mass stopping power for a particle of energy E (in MeV). In this case, the values of S/ρ for alpha particles can be found by the same method as used for a point isotropic source. Figure 4.3 gives the mass stopping power for beta radiation in air in terms of MeV/(gm/cm²). One may assume the value 1.14 for the relative mass stopping power $(S/\rho)_t$ of tissue with respect to air for use in the above expression. This figure is based on information found in Ref. 23.

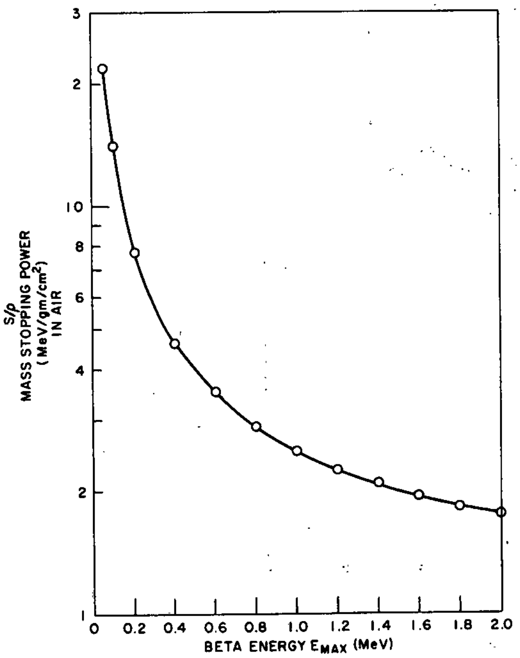

Fig. 4.3

Mass Stopping Power of Beta Radiation as a Function of Energy (Based on information in Radiation Hygiene Handbook, section 14, McGraw-Hill Book Co., Inc., New York, 1959)

L. Absorbed Dose 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 pointed out. 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.

In NBS Handbook #88, an expression is given for the absorbed dose in the case of fast neutrons. The results are applicable to a pure material when the scattered neutrons can be neglected. A curve has been drawn for the dose in pure hydrogen, carbon, nitrogen, and oxygen. This curve has been reproduced as Fig. 4.4. The absorbed dose indicated for the Standard Man is based on the percentage composition as agreed upon by the ICRP.

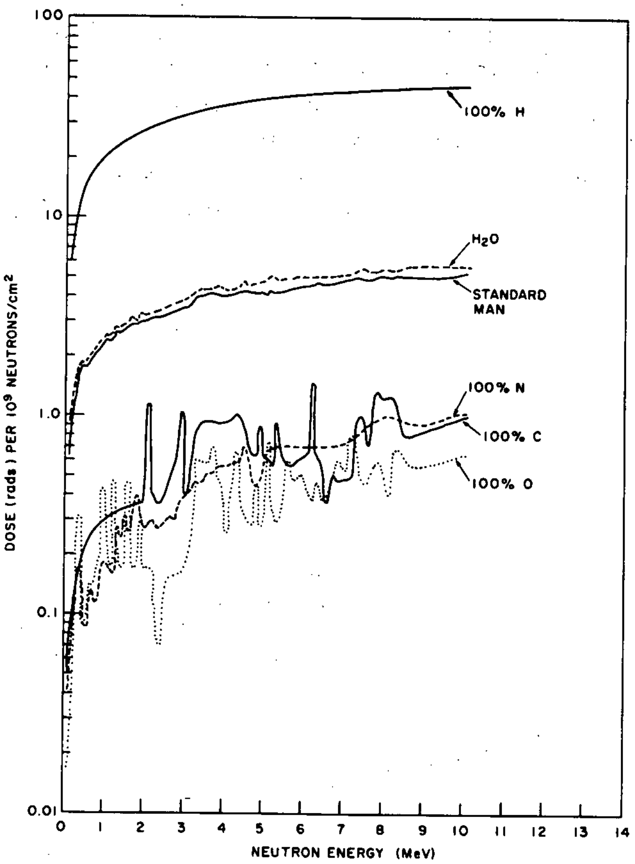

Fig. 4.4. Absorbed Dose for Neutrons as a Function of Energy (From NBS Handbook 88, U. S. Govt. Printing Office, Washington, D. C., 1963)

The neutron dose was calculated by multiplying the value of absorbed dose for each member at the given neutron energy by the percentage composition by weight of that member and summing the contributions:

$$D = \sum_{i=1}^{n} (w/o)_i \text{ [rads for pure element]}_i$$

The absorbed dose is given in terms of the particle fluence 10^9 n/cm^2. The fluence Φ represents the time-integrated flux density, so that

$$\phi = \Phi/t,$$

where t is in seconds, Φ is in n/cm^2, and ϕ is in n/cm^2-sec.

No similar expressions are available for intermediate or thermal neutrons. Resonance absorption plays a key role in these energy regions, so that the presence of certain elements can greatly alter the absorbed dose.

There is a lack of useful formulae for calculating the absorbed dose rate from neutrons. In terms of instrument measurements, the neutron flux density can be measured rather readily. The Monte Carlo calculations of Snyder which relate the neutron flux density for a given energy to the dose equivalent are thus more useful in terms of protection work. These calculations are the basis of the following table, which is taken from NBS Handbook #63, p. 15, and which has been revised according to the new concepts of the ICRU:

MAXIMUM PERMISSIBLE NEUTRON FLUX DENSITY

Time-average flux density for 40-hr week to deliver either 100 or 300 mrem

Neutron Energy, MeV	Quality Factor	n/cm^2-sec for 100 mrem	n/cm^2-sec for 300 mrem
Thermal	3	670	2000
0.0001	2	500	1500
0.005	2.5	570	1700
0.02	5	280	850
0.1	8	80	250
0.5	10	30	90
1.0	10.5	18	55
2.5	8	20	60
5.0	7	18	55
7.5	7	17	50
10	6.5	17	50
10-30	-	10^a	30^a

a = suggested limit.

If one has a measure of the neutron flux density, then this table can be used to find the dose equivalent which this flux density would represent over a 40-hr week. When little information on the neutron energy is available, one should assume a quality factor of 10. The dose equivalent rate from a flux density of ϕ n/cm^2-sec of a given energy can be found by use of the table. In this case, a simple proportion would exist,

$$\frac{\mathrm{DE}}{100\ \mathrm{mrem}} = \frac{\phi}{\phi_{\mathrm{MPL-100\ mrem}}}$$

This gives the dose equivalent of the flux density ϕ if the time in the neutron field were 40 hr. The dose equivalent rate would then be

$$\frac{\mathrm{mrem}}{\mathrm{hr}} = \frac{\mathrm{DE}}{40\ \mathrm{hr}} = \frac{\phi}{\phi_{\mathrm{MPL-100\ mrem}}}\ 2.5\ \frac{\mathrm{mrem}}{\mathrm{hr}}.$$

In general, this 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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SECTION 5 - 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.

Three factors which determine the total exposure one receives in a given radiation field are:

(1) time of exposure;

(2) distance from the source;

(3) amount of shielding present.

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 strong radiation field. In this case, the work procedure 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.

Assume that work must be done in a field of 1 R/hr(= 1,000 mR/hr). If we are allowed 60 mR/day, how long can a person work in this area so as not to exceed this limit?

$$t(\text{hr}) \times 1{,}000(\text{mR/hr}) = 60(\text{mR}).$$

Then

$$t = 60/1{,}000 \text{ hr} = 0.06 \text{ hr} = 0.06 \text{ hr} \times 60 \text{ min/hr}$$

$$= 3.6 \text{ min}.$$

This means that an individual would receive a daily permissible exposure in 3.6 min while working in this radiation field.

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 measurements in air or vacuum.

For other than point sources, the intensity will decrease inversely with distance, but not necessarily as the square of the distance.

However, we are 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. We can eliminate 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.

The time, distance, and shielding factors are used individually or in combination to minimize external radiation hazards.

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.?

The answer to some of these questions may be determined from the discussion and examples which follow.

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.

B. Alpha Radiation

Alpha particles, which lose energy rapidly in any medium because of their relatively high specific ionization, can be stopped by very thin absorbing materials. A few sheets of paper or thin (less than 1/64 in.) aluminum foils will absorb alphas from α-emitting sources. The most energetic alphas will travel only a few centimeters in air. The outer layer of skin ($\sim$7 mg/cm^2) on our body will absorb α particles up to $\sim$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.

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 ionization, which means that its penetration in any absorber will be much greater than that of an α particle.

A 1-MeV β particle will travel about 10 ft 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.

The thickness and choice of material for shielding from beta radiation depends upon: (1) stopping the most energetic β, and (2) bremsstrahlung.

The shielding thickness which is necessary to stop β particles of a given energy will decrease with increasing density. For example, ~0.1 in. of aluminum (ρ = 2.7 gm/cc) will stop 1.5-MeV betas, whereas only ~0.024 in. of lead (ρ = 11.3 gm/cc) 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 pot) is further discussed in Section D.

D. Gamma Radiation

Gamma rays do not lose energy continuously, as will α or β particles, in traversing a medium. As a result, γ rays are much more penetrating than α or β radiation. Gamma radiation is attenuated in matter by: (1) the photoelectric effect, (2) Compton effect, and (3) pair production.

Materials of higher density, such as Pb, U, Th, Au, and W, are best suited for γ shields. However, the use 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.

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, we can choose a shield thickness which will reduce the intensity to nondetectable or nonhazardous levels.

Three relationships (see Fig. 5.1) can be used for calculating shield thickness:

$$I = I_0 e^{-\mu x}; \tag{A}$$

$$I = I_0 e^{-(\mu - \sigma_s)x}; \tag{B}$$

$$I = I_0 b e^{-\mu x}. \tag{C}$$

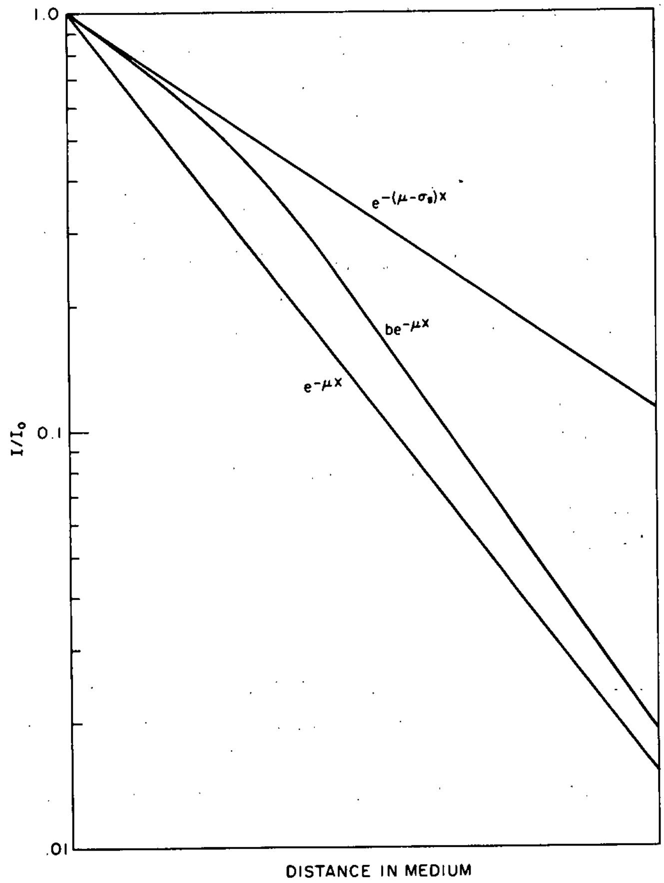


Fig. 5.1. Relationships for Calculating Shield Thicknesses

In radiation protection work we prefer equation (B), since this equation incorporates a safety factor in our calculations (see Fig. 5.1). The attenuation coefficient $(\mu-\sigma_s)$ assumes that all scattered gammas reach our point of interest. This means that we will overestimate the intensity I when radiation I_0 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.

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 flux I; hence, we get a more accurate value for the needed shield thickness.

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.

A half-value layer is determined as follows:

$$I = I_0 e^{-(\mu-\sigma_s)x}.$$

Then,

$$I/I_0 = 1/2 = e^{-(\mu-\sigma_s)x_{1/2}}$$

or

$$\ln \frac{1}{2} = -0.693 = -(\mu - \sigma_s)\, x_{1/2}.$$

From this

$$x_{1/2} = 0.693/(\mu - \sigma_s).$$

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.

If we substitute $0.693/x_{1/2}$ for $\mu - \sigma_s$ in the exponential expression, we obtain

$$I/I_0 = e^{-0.693(x/x_{1/2})}.$$

From this

$$\ln (I_0/I) = (x/x_{1/2})(0.693) = (x/x_{1/2}) \ln 2 = \ln 2^{x/x_{1/2}},$$

where $x/x_{1/2}$ is the number of half-value layers. If we let $n = x/x_{1/2}$, then $I_0/I = 2^n$.

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 cms of lead are required?

(a)

$$I_0/I = 16 = 2^n.$$

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 $\left(\frac{\mu - \sigma_s}{\rho}\right)_{Pb}$ is obtained from the absorption coefficient versus energy curve:

$$\left(\frac{\mu - \sigma_s}{\rho}\right)_{Pb} = 0.045 \frac{cm^2}{gm};\ 0.045 \frac{cm^2}{gm}\left(11.3 \frac{gm}{cc}\right) = 0.509\ cm^{-1} = (\mu - \sigma_s)_{Pb}$$

Then

$$1 \text{ half-value layer} = 0.693/(\mu - \sigma_s)_{Pb} = 0.693/0.509 \text{ cm},$$

and

$$4 \text{ half-value layers} = 4 \times (0.693/0.509) = 5.45 \text{ cm of lead}.$$

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 1 R/hr. What thickness of lead will be required, as a lining inside the box, to reduce the reading at the surface to 2 mR/hr? Assume an effective γ energy of 0.8 MeV.

$$I_0/I = 1{,}000/2 = 500 = 2^n.$$

Then

$$n = \frac{\ln 500}{\ln 2} = \frac{6.215}{0.693} \approx 9.$$

Therefore, 9 half-value layers are required.

$$\left(\frac{\mu - \sigma_s}{\rho}\right)_{Pb} = 0.057 \frac{cm^2}{gm};\ 0.057 \frac{cm^2}{gm}\left(11.3 \frac{gm}{cc}\right) = 0.644\ cm^{-1} = (\mu - \sigma_s)_{Pb}$$

Hence,

$$9 \times 0.693/0.644\ cm = 9.68\ cm \approx 4\ in.\ of\ lead.$$

2. Buildup Factor

In calculations which include the buildup factor b, we take into account the scattered photons resulting from the Compton effect and the "uncollided γ flux" which reach our detector. The value of b will vary with radiation energy and shield material, and also with source geometry and depth of shield penetration.

Curves for b in a number of materials can be found in the Radiological Health Handbook (Ref. 14) and other references. Frequently, the value of b is plotted on semilog paper against a quantity called the relaxation length. The relaxation length is that thickness of absorber which will result in a reduction of 1/e in the initial beam intensity I_0, i.e., $I = 0.368\ I_0$.

We use the narrow-beam relationship, $I = I_0\, e^{-\mu x}$, to determine the value of b. The number of relaxation lengths is given by $\ln(I_0/I)$ to a first approximation. One may iterate to obtain a better approximation.

Example 3. A 1-MeV γ point 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?

$$I_0/I = e^{\mu x}$$

or

$$\ln I_0/I = \mu x.$$

Then

$$\ln 20 = 2.99573 \approx 3 \text{ relaxation lengths.}$$

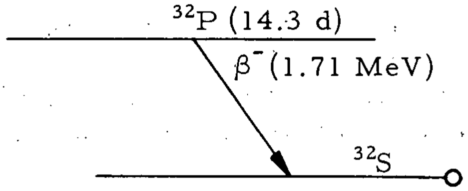
Now consult the curve for a point isotropic source in lead. From this we see that b = 2.

To illustrate the use of both b and calculations involving bremsstrahlung, we have the following example.

Example 4. A 100-Ci ^{32}P source will be shipped in a cylindrical container 3 in. in diameter and 4 in. deep. The I.C.C. regulations require a reading of ≤10 mR/hr at 1 meter. Calculate the container wall thickness using equations (A), (B), and (C) on page 96.

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 for ^{32}P is as follows:



The fraction of beta energy converted to bremsstrahlung is given by $F = 3.33 \times 10^{-4}$ ZE. For the all lead container:

$$F = 3.33 \times 10^{-4}(82)(1.71) = 0.047.$$

Now

$$100 \text{ Ci}(0.047) = 4.7 \text{ Ci of bremsstrahlung.}$$

The R/hr at 1 meter, unshielded is 0.53 CE = 0.53 (4.7)(1.71) = 4.26. For E = 1.71 MeV, $(\mu/\rho)_{Pb} = 0.049$ (cm²/gm), $(\mu - \sigma_s/\rho)_{Pb} = 0.03$ (cm²/gm), and b is found from the curve. Number of relaxation lengths = $\ln(I_0/I)$ = $\ln(4{,}260/10) = 6.05444$; therefore, $b \approx 3$. The density of lead is 11.3 (gm/cc).

$I = I_0 e^{-\mu x}$; $I_0/I = e^{\mu x}$; $4{,}260/10 = e^{0.049(11.3)x}$;

$\ln 426 = 0.554\,x$; $x = 6.054/0.554 = 10.93 \text{ cm} = 4.3 \text{ in.}$ (A)

$I = I_0 e^{-(\mu - \sigma_s)x}$; $I_0/I = e^{(\mu - \sigma_s)x}$; $4{,}260/10 = e^{0.03(11.3)x}$;

$\ln 426 = 0.339\,x$; $x = 6.054/0.339 = 17.86 \text{ cm} \approx 7 \text{ in.}$ (B)

$I = I_0 b e^{-\mu x}$; $I_0 b/I = e^{\mu x}$; $4{,}260\,(3)/10 = e^{0.049(11.3)x}$;

$\ln 1{,}278 = 0.554\,x$; $x = 7.153/0.554 = 12.91 \text{ cm} \approx 5.1 \text{ in.}$ (C)

Using the above value for the wall thickness, the weight of the container would be about 781 lb. In using $(\mu - \sigma_s)$ to calculate the amount of lead required, we get a container with a weight of 1,655 lb. This is more than twice the amount of lead that is actually needed.

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 (Ref. 4), the effective atomic number Z_{eff} for a compound, in reference to the bremsstrahlung formula, is given by

$$Z_{eff} = \frac{N_1 Z_1^2 + N_2 Z_2^2 + N_3 Z_3^2 + \ldots}{N_1 Z_1 + N_2 Z_2 + N_3 Z_3 + \ldots},$$

where N_i is the number of i^{th} atoms per cc of atomic number Z_i. For Lucite, $C_5H_8O_2$, $Z_{eff} = 5.85$.

The range of ^{32}P betas in Lucite is approximately 1/4 in. The fraction of beta energy converted to bremsstrahlung is

$$F = 3.33 \times 10^{-4}\,(5.85)(1.71) = 0.0033.$$

Since the source activity is 100 Ci, the amount of bremsstrahlung is

$$100 \text{ Ci } (0.0033) = 0.33 \text{ Ci}.$$

It is difficult to derive a function which describes the spectrum which is emitted. However, one can assume that the bremsstrahlung consists of monoenergetic X rays equal to the maximum beta energy.

In this case, the exposure rate is

$$\text{R/hr at 1 meter} = 0.53\ \text{CE} = 0.53\ (0.33)(1.71) = 0.299\ \text{R/hr}$$

$$\approx 300\ \text{mR/hr}.$$

Once again b is found from the curve:

$$\ln 300/10 = 3.4\ \text{relaxation lengths}.$$

From the curve, b = 2.2.

$$I = I_0 e^{-\mu x}; \quad I_0/I = e^{\mu x}; \quad 300/10 = e^{0.049(11.3)x};$$

$$\ln 30 = 0.554\ x; \quad x = 3.401/0.554 = 6.14\ \text{cm} \approx 2.4\ \text{in.} \qquad \text{(A)}$$

$$I = I_0 e^{-(\mu - \sigma_s)x}; \quad I_0/I = e^{(\mu - \sigma_s)x}; \quad 300/10 = e^{0.03(11.3)x};$$

$$\ln 30 = 0.339\ x; \quad x = 3.401/0.339 = 10.03\ \text{cm} \approx 3.9\ \text{in.} \qquad \text{(B)}$$

$$I = I_0 b e^{-\mu x}; \quad I_0 b/I = e^{\mu x}; \quad 300(2.2)/10 = e^{0.049(11.3)x};$$

$$\ln 66 = 0.554\ x; \quad x = 4.19/0.554 = 7.56\ \text{cm} \approx 3\ \text{in.} \qquad \text{(C)}$$

Hence, a small amount of Lucite will appreciably reduce the lead requirement. This container will have about 1/3 as much lead as is necessary for the "all lead" container. Again, if $\mu - \sigma_s$ is used in the calculations, the amount of lead is almost twice 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 $\mu - \sigma_s$ can be used. For calculations in air, $\mu - \sigma_s$ can always be used. The buildup factor is important when absorbers are used in which the scattering can be quite extensive.

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 the neutrons until they are absorbed in the medium. In many cases, penetrating γ rays are produced as a result of neutron absorption. The cross sections for these interactions are not as well known as for γ rays. The data is especially lacking for those energies where the penetration is highest and cross section lowest. This contributes to the difficulties in calculating neutron attenuation.

The problem of neutron shielding is further complicated by the fact that substances which effectively attenuate neutrons are generally poor gamma shields. Neutrons are more hazardous to a biological system than gammas; therefore, the neutron flux emerging from a shield should be smaller than the γ flux. Fast neutrons are most difficult to attenuate and therefore are the main concern in the shielding problem. Almost all neutrons are fast upon release from a source.

Elements of low mass number are ideal neutron moderators. Therefore, hydrogen (in the form of water, plastic, or paraffin), beryllium, and carbon (in graphite) 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 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. A 2.2-MeV γ is produced upon neutron capture by hydrogen.

Concrete is an effective 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.

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.

The attenuation of a narrow beam of neutrons passing through a substance is determined from the relationship

$$\phi = \phi_0 e^{-\Sigma_t x},$$

where ϕ_0 is the incident flux density, Σ_t is the total macroscopic cross section, and x is the absorber thickness. In using Σ_t, we assume all scattered neutrons to be 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 thick shields which contain hydrogen, the removal cross section Σ_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 hydrogenous shield. In these cases, the removal cross section itself accounts for buildup.

In the case of thin shields, information concerning buildup is very sparse at this time. A conservative approach is suggested in NBS Handbook 63, in which a buildup factor of ~5 for water or paraffin is given for shields at least 20 cm thick.

When the shield is thin, even one which contains hydrogen, the scattered neutrons may contribute significantly to the flux at a point of interest. Since the removal cross section is based on measurements through thick shields, the concept does not adequately apply to thin shields, and a buildup factor must be included.

The flux from an isotropic neutron point source can be approximated by the formula

$$\phi = \frac{\phi_0 b e^{-\Sigma_R t}}{r^2},$$

where

ϕ_0 = flux at 1 cm;

r = distance from source to point of interest (cm);

b = buildup factor;

Σ_R = macroscopic removal cross section for the given source and shield material;

t = shield thickness (cm).

A list of macroscopic removal cross sections can be found in NBS Handbook 63.

Example 5. What is the flux at a point 1 meter away from a neutron source which is surrounded by a 20-cm sphere of water? The source is a Po-Be source with a flux of 10^6 n/cm^2/sec at 1 cm. The value of Σ_R for water is 0.103 cm^{-1} (from NBS Handbook 63).

$$\phi = \frac{10^6(5)\ e^{-0.103(20)}}{10^4} = 64\ \text{n/cm}^2\text{/sec.}$$

If we ignored the buildup factor, ϕ = 13 n/cm^2/sec.

Shield design will eventually become possible through purely calculational methods; however, at present, it is partly an "art." Many of the currently designed shields make use of data from experimental studies.

F. Shielding Materials

Water, concrete, steel, and lead are the more common materials used for shielding purposes.

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; (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.

Water, because of its high hydrogen content per unit volume and low cost, is widely used as a neutron shield; however, it is a poor γ absorber. In cases where a liquid is undesirable, plastics, wood, or paraffin can be used. Polyethylene is probably the most commonly used plastic.

Concrete is an adequate 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 neutron absorber, and the capture reaction in boron yields a relatively low-energy (0.5 MeV) gamma.

High-density materials are necessary for shielding the γ radiation. Lead is a valuable gamma 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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SECTION 6 - BIOLOGICAL EFFECTS 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. The job of a tissue is to perform a special body function. A cell is composed of many elements; hydrogen, oxygen, carbon, and nitrogen are the main components. For the present, our interest will be directed to the basic unit of structure in the body, namely, the cell.

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 per cent water. The vital part of the cell is the nucleus, usually an oval body 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.

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. 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. Arranged linearly along the chromosomes are the genes which determine hereditary characteristics. 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 this 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 different 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 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, some of the parts will be charged. These fragments are called radicals and ions, and are not chemically stable. Because the cell has a high water content, the most important radicals are those formed from water molecules. 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. 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. Although many factors are important in assessing the total damage, it seems likely that most cell functions and structures are somewhat impaired by radiation.

Much work of a theoretical nature has been done concerning the threshold question. The threshold dose would be that dose below which no effects are observed. If such a concept is valid, then there would exist a dose of radiation such that no damage would be done. At the present time there is not enough empirical data to prove or disprove the existence of a threshold dose. For this reason, one should always assume that, however small the amount of radiation, it will produce biological effects.

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. 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 effects which can be transmitted to a future generation (see Ref. 1). 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. Somatic Effects

The somatic 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 somatic effects:

1) nature or type of radiation - some types of radiation are more effective in producing damage;
2) the absorbed dose - this is a function of the energy absorbed per gram of tissue;
3) time distribution - a lethal dose given in a short time may not be lethal if protracted over a long time;
4) dose distribution - is the total body involved or only a specific organ?

All these factors combine to make the effects on different organs differ for changes in any of the parameters. Indeed, even the age of the individual enters the picture, since children are more affected by radiation than are adults.

The 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 doses, there are three basic forms of early or acute damage. In the range above a few thousand rad, the dose is fatal within minutes to hours. 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 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 gastro-intestinal 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 <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 >200 rad. In the range above 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 gamma-ray data. As such, the conversion to rem for other types of radiation is not justified in this instance.

The clinical effects which follow acute exposure to total body doses of ionizing radiation are given in summary in Table 6.1. This table is taken from Ref. 2 and may be of use as a guide. Note the virtual absence of any symptoms in the range below a dose of 100 rems. Some people would be expected to have mild symptoms in the range 50-100 rems because of differences among individuals. Below 50 rems, however, no symptoms at all are expected.

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: this expresses the dose at which 50 per cent of those exposed would die. For man, the best estimate places the LD_{50} at 300 to 500 rad. Of course, in this range all would have severe symptoms. Note that this dose refers to short-term total body radiation.

The previous paragraphs dealt with the overall effect of very high doses. In each case, the site of major damage leading to death was discussed. Other organs and systems will also be involved along with those whose damage may be conducive to death. The following breakdown attempts to point out some of the somatic effects caused 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, ~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. In cases of severe irradiation, the platelets drop in number within about a week. 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

Table 6.1*

SUMMARY OF CLINICAL EFFECTS OF ACUTE IONIZING RADIATION DOSES

Range	Subclinical Range, 0 to 100 rems	Therapeutic Range, 100 to 1,000 rems			Lethal Range, over 1,000 rems	
		100 to 200 rems	200 to 600 rems	600 to 1,000 rems	1,000 to 5,000 rems	Over 5,000 rems
		Clinical surveillance	Therapy effective	Therapy promising	Therapy palliative	
Incidence of vomiting	None	100 rems: 5% 200 rems: 50%	300 rems: 100%	100%	100%	
Delay time	-	3 hr	2 hr	1 hr	30 min	
Leading organ	None	Hematopoietic tissue			Gastrointestinal tract	Central nervous system
Characteristic signs	None	Moderate leukopenia	Severe leukopenia; hemorrhage; infection; purpura Epilation above 300 rems		Diarrhea; fever; disturbance of electrolyte balance	Convulsions; tremor; ataxia; lethargy
Critical-period post-exposure	-	-	4 to 6 wk		5 to 14 d	1 to 48 hr
Therapy	Reassurance	Reassurance; hematologic surveillance	Blood transfusion; antibiotics	Consider bone-marrow transplantation	Maintenance of electrolyte balance	Sedatives
Prognosis	Excellent	Excellent	Good	Guarded	Hopeless	
Convalescent period	None	Several weeks	1 to 12 mo	Long	-	
Incidence of death	None	None	0 to 80% (variable)	80 to 100% (variable)	90 to 100%	
Death occurs within	-	-	2 mo		2 wk	2 d
Cause of death	-	-	Hemorrhage; infection		Circulatory collapse	Respiratory failure; brain edema

*The Effects of Nuclear Weapons, U. S. Government Printing Office, Washington 25, D. C. (April 1962).

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.

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).

3. Digestive Tract

The digestive tract, or alimentary canal, consists of the mouth, pharynx, esophagus, stomach, and the small and large intestines. In an adult man, the canal may be as much as 30 ft 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 chemical phases of digestion begin. The radiosensitivities 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. In fatal cases, infection, failure of food absorption, and dehydration from diarrhea are the causes.

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. If the dose is high enough, the effect will be sterility. It requires a larger dose to produce permanent sterility in the male than in the female. In most cases, the dose required to produce permanent sterility is in the region of fatal doses. In man, partial sterility can be induced at doses around 150 rad.

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.

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.

The effect of radiation on the brain is to alter function rather than structure for doses up to LD_{50}. At higher doses, brain damage may occur directly or through lack of blood supply due to damage of blood vessels.

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 per cent iodine, and is essential for growth and development. Damage to the thyroid, or to the other two glands, have marked affects in the body.

The thyroid is radioresistant from the standpoint of external radiation. It can be severely damaged if radio-iodine is inhaled, since iodine will concentrate in the thyroid. Damage causes a decrease in production of thyroxine, which produces a lower metabolism rate. Muscle tissue may then fail to absorb enough oxygen and health can be badly impaired.

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 lens opacities. The formation of cataracts is thus a late somatic effect. Acute effects in other eye structures occur only after high doses.

Neutrons are more effective than X rays in producing cataracts. Doses of X rays greater than 500 rad will produce significant cataract, but after only 200 rad of mixed gamma and neutrons, lens opacities have been reported.(1)

The susceptibility to cataract formation depends somewhat on age. Radiation is more likely to cause cataracts in younger people.

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.

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 stomach.

External radiation is not too effective in causing damage to these organs. Most damage is caused by internal exposure from radioisotopes which concentrate in the liver.

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. 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.

11. Circulatory System

The heart and blood vessel system are damaged seriously only for very high doses of radiation.

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). Skin cancer is a late effect of chronic irradiation at high dose rates.

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.

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 radioisotopes, 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.

E. Late 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. Neoplasms

Animal studies indicate that the incidence of cancer is a function of the dose and the time duration of the dose. Many types of cancer are produced in man, most likely in the skin, the lungs, the bones, and the bone marrow (leukemia). The latent period is long for radiation-induced cancers, of the order of years.

2. Tissue Effects

Among the more prominent late effects in tissues are cataracts and sterility. Radiation-induced cataracts are slowly progressive over a period of time, but may stop or even regress. Sterility is a late effect which may be either permanent or temporary. In some cases, fertility will return in a few years.

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. Much more data are needed before any conclusions can be drawn.

4. Growth and Development

Effects on the embryo depend upon the dose as well as the age of the fetus. The younger the fetus, the more it is affected. 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.

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. At present, all that is really known is that radiation does increase the incidence of these effects.

F. 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.

For this reason, much of the present knowledge is based on work with animals. 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.

Recent studies have shown that some hereditary effects in man are caused by chromosome damage. Up to the present time, the lack of knowledge about the harmful traits has made estimates of the magnitude of these effects quite doubtful. 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.

As in the study of late somatic effects, further progress in the problems which concern hereditary effects await much needed data from careful experimentation. Perhaps the study of those groups which have been exposed to higher doses may yield useful data. In any case, there remain many problems of real concern which must be overcome.

There are many areas in which the surface has only been scratched. It is only through further study that many of the factors which play a role in hereditary effects will be ascertained.

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SECTION 7 - BACKGROUND RADIATION

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 five main components: cosmic radiation, the radioactivities of the earth's surface, of air, of water, and of the human body. One might add that a man-made source (that is, fallout from nuclear weapons testing) influences 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.

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) that outside the earth's atmosphere, cosmic rays consist of 87% protons, of 12% alpha particles, and of 1% other heavier nuclei at latitudes above 55°. These particles may have energies in the range from a few BeV to greater than 10^{17} 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. This consists of two belts of radiation (electrons and protons) which are observed at high altitudes. These belts are symmetrical with respect to the magnetic equator.

The first band occurs at about 1000 km and extends from 30° N to 30° S. The intensity in the belt increases with altitude until about 3000 km and then drops off. The energy spectrum is not very well known at the present time. These belts of radiation are being studied because of the dose rates they would produce during space travel.

The second band starts at about 12,000 km and reaches its maximum at 15,000 km. It extends from 60° N to 60° S. It is thought that dose rates in this band are higher than those in the inner belt, but that shielding would reduce the dose rate considerably.

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 are emitted: mesons, 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 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 the earth's surface, the secondary cosmic rays consist of mesons mainly (hard component), electrons and photons (soft component), 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 50° over that at the equator. At sea level, the effect is small for the ionizing component (10%) but large for the neutron component (150%).

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.

At sea level and high latitudes, the ionization is about 1.96 ip/cm^3-sec, giving a dose rate of about 28 mrad/yr for soft tissue. Patterson et al.(2) have given an estimate of 25 mrem/yr as the dose rate from neutrons. The total dose rate at sea level would then be on the order of 50 mrem/yr at middle latitudes.(1) This dose rate would be expected to increase with altitude and decrease with latitude.

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 gamma-ray dose from radioactive elements of the uranium and thorium series and from radioactive ^{40}K.

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 gamma dose rate for the year exceeds one rad.

For other areas, studies have been made in an attempt to measure the average gamma dose rate to the world population. Not enough data have been collected as yet to make a firm estimate. The studies which have been made include values for dose rates both indoors and outdoors. By means of these data and the assumption that most of a person's time is spent inside buildings, the estimated mean dose rate turns out to be about 50 mrad/yr.

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 alpha as it decays, the gas ^{222}Rn is formed. In the thorium chain, the decay of ^{224}Ra results in the 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 given 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.

The external gamma dose rate from ^{222}Rn and ^{220}Rn is based upon an average concentration of 0.5 $\mu\mu Ci/\ell$ of ^{222}Rn and 0.02 $\mu\mu Ci/\ell$ of ^{220}Rn. The tissue dose rate turns out to be about 4.5 mrem/yr. Many attempts have been made to try to assess the dose rate to the lungs from these products. This estimate depends quite a bit on the assumptions made in regard to the many factors which enter into the respiration process. Data from Sweden seem to indicate that the lungs may receive a higher dose from natural sources than any other body tissue.

Among other radioactive products which are found in air are ^{14}C, ^{3}H, and ^{7}Be. None of these products add a significant amount to the background dose rate.

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 large amounts 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 though, except in cases of very high water concentrations. Tables of ^{226}Ra and ^{222}Rn concentration, as well as other data, can be found in Ref. 1.

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, ^{226}Ra, ^{228}Ra, and ^{14}C are the main radioisotopes 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. The content in the body organs of man varies widely. Based on an average content of 0.2% by weight in man, the dose rate to the gonads turns out to be about 20 mrem/yr.

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 concentration is taken as 75 $\mu\mu$Ci. Of this amount, 80% is assumed to be in the skeleton. The content of ^{228}Ra is taken as 50 $\mu\mu$Ci. The dose rate produced by these products is shown in Table 7.1.

The average whole-body content of carbon is taken as 18%. 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 average dose rate turns out to be about 1 mrem/yr total body.

Table 7.1 gives a summary of the present estimates for the dose rate from all background sources.

Table 7.1*

BODY TISSUE DOSE RATES DUE TO EXTERNAL AND INTERNAL IRRADIATION FROM NATURAL SOURCES OF RADIATION IN "NORMAL REGIONS"

Source of Irradiation	Dose Rate, mrem/yr		Bone Marrow
	Gonad	Haversian Canal	
External Irradiation:			
Cosmic Rays (Including Neutrons)	50	50	50
Terrestrial Radiation (Including Air)	50	50	50
Internal Irradiation:			
^{40}K	20	15	15
^{226}Ra and decay products (35 per cent equilibrium)	0.5	5.4	0.6
^{228}Ra and decay products (equilibrium)	0.8	8.6	1.0
^{210}Pb and decay products (50 per cent equilibrium)	0.3	3.6	0.4
^{14}C	0.7	1.6	1.6
^{222}Rn (absorbed into bloodstream)	3	3	3
Total	125	137	122

*This table taken from Report of the United Nations Scientific Committee on the Effects of Atomic Radiation (1962).

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 due to other background sources, it will be treated in this section.

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 25 miles.

The fallout which occurs may be described as local or world-wide. 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 hr after the blast.

If the height of the burst is such that the fireball does not touch the surface, then 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 30,000 to 55,000 ft. 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 yr 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. It should be noted that in a nuclear war, local fallout would by far present the most danger.

G. Effect 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 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, 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. These are: ^{131}I, ^{89}Sr, ^{90}Sr, and ^{137}Cs. Another product, ^{14}C, which is produced by neutron absorption by nitrogen atoms in air, is also of interest.

Both ^{131}I and ^{89}Sr, because of their short half-lives are of concern only as short-term hazards. On the other hand, ^{90}Sr, ^{137}Cs, and ^{14}C are important from the standpoint of long-term hazards. At the present time, the impact of world-wide fallout on natural background levels has been quite small. The estimates of the average dose to a person in the United States during the next 70 yr are placed between 0.4 and 0.9 rem to the bone. Values for bone marrow and gonads are even less than this. Thus, the contribution to the background dose rate from the testing of nuclear weapons does not seem to be very serious. Danger would come from a large increase in the fission product inventory in the stratosphere. This condition could result from large-scale, long-term atmospheric testing of weapons.

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SECTION 8 - 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 6). 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 some 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.

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 present 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 since it was 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 was found that 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. What may have been safe with one type of source in a given type of treatment might then turn out to be not so safe for some other source.

During this early period, much of the use of radiation was centered about the medical field. 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. With the advent of radiation uses in industry, the need for standards and exposure limits became more acute. Larger numbers of people were 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 complex factors which enter into radiation effects.

2. Tolerance Dose

In the early 1920's, a movement for better standards and stricter control of radiation began.(2) By this time, some of the long-term effects of exposure to high dose rates were known to radiologists and technicians. A great deal of effort was expended in many countries in an attempt to improve protection standards. The concept of a tolerance dose arose out of these studies as well as a search for a better unit of dose.

3. 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.(3) 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.(4) At first known as the International X-ray and Radium Protection Commission, this group is now called the International Commission on Radiological Protection (ICRP). The task of this group has been to discuss and review basic protection principles, and to then prepare recommendations of safe practices based on their findings. 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 its early days, the ICRP did not have a wealth of data from which to draw. Because the need for exposure limits was a pressing matter, attention was directed to this aspect. 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 1937-50. 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). Acting on the suggestion of the chairman of the ICRP, this group was formed in the United States in 1929.(5) The work of this body is coordinated by the National Bureau of Standards. The recommendations of the Committee can be found in the National Bureau of Standards Handbooks. The first of these reports, dealing with X-ray protection, was published in 1931 as NBS Handbook 15.(6) In 1935, this handbook was revised and issued in 1936 as NBS Handbook 20.(7) This report contained a recommendation of a tolerance exposure level of 0.1 R/day. Handbook 23,(8) dealing with radium protection, was issued in 1938. The work of the NCRP was disrupted by the advent of the war. However, its recommendations as outlined in Handbooks 20 and 23 served as the basis for protection practices during the days of the Manhattan project. 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 were born which could lead to serious problems. Internal exposure 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 problems more effectively. A review of new data led to certain changes and additions to the old recommendations. By 1948, the reports of the subcommittees were ready. These recommendations formed the bulk of subject matter for NBS Handbook 59.(9)

4. Permissible Dose

Of prime importance was the adoption of the term permissible dose. Data from biological studies showed that one could not assume that all effects had a threshold dose. In the case of gene damage, effects could be expected at very low doses. The term tolerance dose seemed to suggest that this was a dose below which no effects would occur. The NCRP felt that a change in terminology was warranted to get away from this idea. Since some effects could be expected even from very low doses, the NCRP felt it was prudent to point this out and suggest lower limits of exposure. The figure of 0.3 R/week was recommended.

The ICRP then met in 1950 and based its new recommendations on the large body of data which had been collected by the NCRP. The ICRP also lowered the exposure level to 0.3 R/week. Their report(10) gave recommendations for maximum permissible concentrations for a few radioisotopes. In addition, the Commission set up a number of committees to deal with specific areas of protection problems.

At both of these meetings, the subject of genetic effects was discussed. Since data was lacking on key points no firm decisions could be reached.

5. Total Accumulated Dose

The report(11) issued by the ICRP after its 1953 meeting contained much more matter than any previous report. This was due to the fact that a number of the committees made very detailed recommendations. The Commission met again in 1956. At this time the recommendation was made that the maximum permissible exposure level be lowered to 0.1 R/week. At this meeting, concern was expressed over long-term exposure at low dose rates, as well as genetic effects. Again, these were areas for which data could not be gathered except through long-term studies. Also, large numbers of subjects would be required, since the damage would be nonspecific. Thus, the damage due to radiation would show up only on a statistical basis. For these reasons, the level was lowered and also the total accumulated dose was restricted. The latter results if one accepts the recommendation that the total accumulated dose D be governed by $D = 5(N-18)$ rem, where N is the age in years.

Because of the growth of the field and the uses of radiation which might involve exposure to large populations, it was felt that this matter should be treated. On the basis of recommendations made by the Genetics Committee of the National Academy of Sciences,(12) the ICRP has suggested limits for population groups also.

The recommendations which were worked out at this meeting have been published(13-16) and have served as a basis for many of the present limits. However, work continues in the field, and from time to time new reviews will be made and perhaps the limits will be changed again. At any rate, one should not be disturbed by the fact that the limits always seem to be reduced. The basic premise which is the guideline in all protection work is to see that exposures are kept at the lowest practical levels. Regardless of what has been the permitted levels, protection efforts to date have resulted in an average exposure well below these levels.(2)

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, in recent years the National Academy of Sciences-National Research Council has undertaken the study of biological effects.(12, 18) This group consists of about 140 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 from which regulations can be developed. On a world-wide basis, the United Nations has established a Scientific Committee. Their reports(19,20) on the effects of atomic radiation have helped greatly to supply much needed background knowledge.

6. Federal Radiation Council

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 acts to advise the President concerning radiation matters and to provide guidance for all Federal agencies in setting standards and in working with the States.(21) The Secretary of Health, Education, and Welfare is the chairman of this group. Up to this time, the Council has issued three staff reports.(22-24)

7. Radiation Protection Guides

The concepts of "Radiation Protection Guide (RPG)" and "Radioactivity Concentration Guide (RCG)" have been adopted by the Council for Federal use. The Radiation Protection Guide is defined as "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 as far below this guide as practicable."(22) The import of this definition is to focus on the concept of an acceptable risk. The FRC feels that the term "maximum permissible dose" is often misunderstood. In order to convey the meaning that there can be no level of exposure without proper thought about the reasons for permitting the exposure, the Council decided on the new term.

The RCG is the concentration of radioactivity in the environment which is determined to result in whole body or organ doses equal to the Radiation Protection Guide. This term is advocated instead of the ICRP term "maximum permissible concentration."

The RPG values recommended by the FRC appear in the Council's first report.(22) These values are intended as interim levels for normal peacetime operations:

RADIATION PROTECTION GUIDES*

Occupational Exposure	Condition	Dose (rem)
1. Whole body, head and trunk, blood-forming organs, gonads, lens of the eye	Accumulated dose	5(N-18); N > 18 years
	13 weeks	3
2. Skin and thyroid	Year	30
	13 weeks	10
3. Hands and forearms, feet, and ankles	Year	75
	13 weeks	25
4. Bone	Body burden	0.1 μgm ^{226}Ra or its equivalent
5. Other organs	Year	15
	13 weeks	5
Population		
1. Individual	Year	0.5 (whole body)
2. Average	30 years	5 (gonads)

*Adapted from Ref. 22.

In the second report,(23) the FRC provides guidance concerning radioactive substances which may be deposited in the body. The recommendations are designed to limit exposure of members of population groups from radioactivity in the environment. Although most of the recommendations are quite general, specific guidance is provided for exposure of population groups to ^{226}Ra, ^{131}I, ^{89}Sr, and ^{90}Sr. However, no recommendations are made concerning Radioactivity Concentration Guides (RCG). The matter is left to the Federal agencies to develop the concentration values needed for control actions.

8. General Philosophy of the Federal Policy on Radiation Matters

These two reports form the basic general philosophy of the Federal policy on radiation matters. Each Federal agency has the responsibility to determine specific regulations in its area of jurisdiction. In some cases, these guides may 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."(22-23)

B. 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.(17) In this respect, the Act authorizes 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 is concerned with the development of regulatory safety standards.

In regard to the licensing program, the AEC has set forth its regulations in the FEDERAL REGISTER. These are contained under Title 10, Code of Federal Regulations. The parts of Title 10 which deal with health and safety are: 2, 7, 8, 20, 25, 30, 40, 50, 55, and 70. Part 20, Standards for Protection against Radiation, deals specifically with the regulations for control of radiation hazards by the licensee.

In addition to this, the AEC has issued regulations which pertain to its own activities as well as to those of contractors not subject to licensing. These appear in the AEC MANUAL, Chapter 0524 - Standards for Radiation Protection. The standards are based upon those which apply to licensees and those recommended by the FRC. The AEC defines contractor as "an organization whose activities are conducted under a contract containing the standard safety, health and fire protection clause AECPR 9-7.5004-8."

As part of its duties, the AEC 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 AEC personnel 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 or contractor is required to correct this.

Many of the states have taken up the task of setting up their own safety standards. In this respect, the AEC has been directed to assist the states to assure that the state and Commission programs are compatible. An outline of programs which have been set up can be found in Ref. 17.

Safety in the shipment of radioactive substances is regulated by five agencies. These are the Interstate Commerce Commission, Coast Guard, Federal Aviation Agency, Post Office Department, and the AEC. Because of the large increase in shipments in recent years, a move to coordinate their activities is under way. In 1957, the Interagency Committee on Transportation of Radioactive Materials was formed.(25)

This group has members from each of the five agencies and also from the Bureau of Explosives of the Association of American Railroads. The purpose of this body is to assure that consistent and uniform regulations are issued. At present this group is engaged in a review and revision of the present transportation rules. Reference 17 contains excerpts which cover the salient points of these rules.

This concludes the brief outline of those 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. However, these changes will reflect a conscious effort to attain added safety in the field.

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SECTION 9 - INTERNAL DOSE CALCULATIONS

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. Among the many factors of main concern that enter into such a calculation are: the shape of the organ, the type of radiation, 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 (μCi/gm) 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.

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. As it turns out, 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. This is pointed out in the recent report of the RBE Committee of the ICRP and ICRU.(4)

For the purposes of protection work, the present method of dose determination, proposed by the ICRP,(2) provides an adequate estimate. One must keep in mind that as knowledge is gained, 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 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 many biological factors, most of which are not quantitatively known. Much work is needed in this area to improve the sparse 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 method of calculation suggested by the ICRP(2) is not very precise. It can only yield a good estimate in cases where there are sufficient data to back up the assumptions made. 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 maximum permissible concentration (MPC) values are very useful. 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.

At the present time, the FRC is in the process of studying the problem of setting concentration limits.(3) When completed, their recommendations will provide guidance for the Federal agencies. Until these guides are ready, it is useful to follow the recommendations of the ICRP. The rest of this section will be devoted to the methods used by the ICRP to calculate body burdens and MPC values. The current report of the ICRP which deals with internal dose can be found in Ref. 2. Much of the information which follows has been taken from this.

Let us focus on the nature of the problem. One is given a maximum permissible exposure level (MPL) for a certain organ. A knowledge of the mass and size of the organ is needed. For a certain radionuclide of known decay scheme and energy of emissions, one can calculate an organ burden. This will be the amount of the radionuclide in the organ which delivers the stated dose equivalent rate.

There are many ways in which the radionuclide may reach this organ. For most purposes, only two are of concern: inhalation and ingestion. The rest of the problem is then to relate the concentration in air and/or in water to the amount of the radionuclide which ends up in the organ. Many factors enter into this part of the problem.

From the standpoint of absorption into the body, the solubility of the radioactive substance in body fluids plays a major role. Whether a given substance will be soluble in body fluids or not is not easy to tell. However, as a mode of entry, ingestion is of concern only for soluble isotopes. Also, in the case of elements not required by the body, absorption by ingestion is poor.

In those cases where absorption does occur, one is concerned with the transfer of the substance to the organ, that is, only a fraction of the ingested material will go into the 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.

In regard to inhalation, both soluble and insoluble matter must be treated. Another factor is particle size. The movement and retention of matter in the respiratory tract is a function of the particle size. Because of the complex relationship which exists, estimates of the transfer of radioactive matter by this mode of entry are poor. More data are needed before firm estimates can be made.

At present, the ICRP assumes a general model for use in the calculations. In this model, one-quarter of that which is inhaled is assumed to be immediately exhaled. One-half of the material is trapped by the upper respiratory tract and swallowed. As in the case of ingestion, a portion of the soluble matter will reach the organ of interest. The final one-quarter of the inhaled matter reaches the lungs. For a soluble substance, this will be taken up by the blood stream. A fraction of this matter is then transferred to the organ of concern.

For insoluble matter, the model is somewhat the same. Of that amount which reaches the lungs, it is assumed that half is worked up and swallowed within 24 hr. In this case then, five-eighths of the material is swallowed with no absorption. The one-eighth which remains in the lungs is slowly taken up by body fluids.

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 growth 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 a separate compartment in which the elimination of a substance proceeds at a constant rate. This exponential model for elimination means that each organ has a half-life for biological elimination of a given substance. This is only very roughly true and in some cases a power series may be a better model.(2) However, it provides a simple means of obtaining a rough estimate of the retention in a certain organ.

Given the intake rates for air and water, one can then solve for the MPC values. The calculation consists of finding the daily intake which would not result in an accumulation greater than the organ burden. This daily intake is assumed to continue over a period of 50 yr. This means that the dose equivalent rate in the organ is not greater than the allowed limit at the end of this period of exposure. In the ICRP method, then, the net amount of the radionuclide in the organ during the 50-yr exposure time approaches a limit which is equal to the organ burden. For many radioisotopes, the limit will be reached long before the end of the exposure time. When this occurs, an equilibrium state is reached in which there is no further increase in accumulation in the organ. For other radioisotopes, this equilibrium is not reached in the 50-yr period. In this case, continued exposure will result in further accumulation. The ICRP uses 50 yr since the working period for most people extends from age 18 to about age 65.

B. Standard Man

Throughout the above outline of the calculation of MPC values, 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 "standard man" concept. The standard man represents a set of internationally agreed-upon values for the many characteristics of man which are needed for internal dose calculations. These values can be found in Tables 6 through 11 of the ICRP report.(2) Since the values are intended to represent an average adult, no account is taken of the differences which occur among individuals. The published values of MPC's(2,5,6) have been calculated using the standard man data.

C. Critical Organ

One of the problems of setting MPC values is to pick out the organ which will be most affected by the given radionuclide. In many cases, a substance will be taken up by a number of body organs. The MPC value should be such that none of these organs accumulate an amount in excess of the organ burden. In dealing with this facet of the problem, the ICRP uses the concept of "critical organ:" the critical organ is taken to be that organ of the body whose damage by the radiation results in the greatest damage to the body.

A number of factors affect the choice of a critical organ. Some organs are more essential to the body than are others, and slight damage to these may affect the body more than heavy damage to some of the others. Since there is a difference in radiosensitivity among the organs, this must also be taken into account. In cases such that little absorption into the body occurs, the greatest harm may be produced in the organ of entry. In most cases studied by the ICRP, the critical organ turns out to be that organ which receives the largest accumulation of the radioactive substance. However, the ICRP has included the MPC values for organs other than the critical organ in its report.(2) The critical organ and MPC values for a given radionuclide appear in boldface type. These are the values which the ICRP recommends, and they apply primarily to occupational exposure. Occupational exposure refers to the exposure of a person who normally works with radiation as part of his job.

D. Effective Energy per Disintegration

In order to find the organ burden for a given radioisotope, a knowledge of the energy absorbed by the tissue is needed. In an assumed radioactive process, the energy which is released may be divided among a number of products. Some or all of these products will then lose energy in passing through tissue. The total energy ΣE which is absorbed per disintegration will be equal to the sum of the energy lost by each of the products. These products may include X, γ, α, β^-, β^+, e^-, and recoil atoms. The actual products which are involved in a given process depend upon the decay scheme of the radioactive substance. Also, one may have to take into account decay chains, that is, if the parent decays to a radioactive daughter, then one must deal with the energy absorbed in tissue as a result of the radioactive process of the daughter product also.

Since one is more concerned with the effect of the absorbed energy on tissue, it is not enough merely to find the total energy absorbed. For this reason, the ICRP uses the concept of the "effective energy per disintegration." This includes the total energy absorbed modified by certain factors which take into account the effect on tissue. One can then express on a common scale, namely, the dose equivalent, the effectiveness of the energy absorbed in tissue.

The general form used by the ICRP to indicate the effective energy per disintegration is $\Sigma EF(RBE)n$. In any given calculation, some of the factors may be unity or may not even apply. Thus, simpler forms of the above may appear, i.e., ΣE.

The symbols under the summation sign are used to denote:

1. The Energy Absorbed per Disintegration, E

This quantity must be found for each of the products that result from the radioactive process. The total energy absorbed will then be the sum of the contributions from each of these products. One can find the amount contributed by each of the products with a knowledge of the decay scheme and the use of these equations:

a. <u>β^- Radiation</u>

$$E = 0.33\ fE_{max}\left(1 - \frac{Z^{1/2}}{50}\right)\left(1 + \frac{E_{max}^{1/2}}{4}\right)\frac{\text{MeV}}{\text{dis}}.$$

In this equation, Z is the atomic number of the substance which emits the β^- and f is the fraction of the disintegrations in which a beta of maximum energy E_{max}(MeV) is emitted. In this case, it is assumed that all the beta energy is given up in the tissue.

b. <u>β^+ Radiation</u>

$$E = 0.33\ fE_{max}\left(1 + \frac{E_{max}^{1/2}}{4}\right) + 2f(0.51)(1 - e^{-\sigma x})\ \frac{\text{MeV}}{\text{dis}}.$$

The symbols f and E_{max} have the same meaning as for β^-. The positron gives up all its energy in the organ when it combines with an electron to form two 0.51-MeV gamma rays. The γ rays may then be absorbed in the organ and add to the total energy. The symbol x stands for the radius of the body organ (cm). Values for each organ can be found in Table 8 of Ref. 2. The symbol σ stands for $(\mu - \sigma_s)$ in tissue. Values for $\left(\frac{\mu - \sigma_s}{\rho}\right)$ versus the photon energy can be found in Ref. 8. Since ρ for tissue is about 1 gm/cc, $\left(\frac{\mu - \sigma_s}{\rho}\right)$ for tissue equals $(\mu - \sigma_s)$.

c. <u>γ Rays</u>

$$E = fE_\gamma(1 - e^{-\sigma x})\ \frac{\text{MeV}}{\text{dis}},$$

where E_γ is the energy of the gamma and the other symbols have been discussed before. This expression assumes that no internal conversion is taking place.

d. <u>Internal Conversion, e^-</u>

$$E = f\left[(E_\gamma - \eta)\left(\frac{\alpha_k}{1 + \alpha_k}\right) + \eta\left(\frac{\alpha_k}{1 + \alpha_k}\right)(1 - e^{-\sigma x}) + E_\gamma\left(\frac{1 - e^{-\sigma x}}{1 + \alpha_k}\right)\right]\frac{\text{MeV}}{\text{dis}}.$$

When internal conversion must be taken into account, the above form, which gives the total absorbed energy for the entire process, should be used. Again, E_γ is the energy of the gammas which are emitted, in MeV. The term $(E_\gamma - \eta)$ gives the energy of the conversion electron, where η is the binding energy in MeV of the daughter product. The symbol α_k is the conversion coefficient for the K shell. This is the ratio of K electrons to the "unconverted" gamma rays. If the process also occurs in the L and M shells, one must take this into account also. In cases for which α_L and α_M are unknown, the above can be used to yield a conservative value for the energy absorbed.

e. α Radiation

$$E = fE_\alpha \frac{\text{MeV}}{\text{dis}},$$

where E_α is the energy of the alpha in MeV.

f. Recoil Atoms

$$E = \frac{fE_\alpha m_\alpha}{m_r} \frac{\text{MeV}}{\text{dis}}.$$

In the process of alpha emission, the atom recoils and its energy must be included in the total for alpha decay. In the expression, m_α and m_r are the mass of the alpha and the recoil atom, respectively.

g. K and L Capture X Rays

$$E = f\eta(1 - e^{-\sigma x}) \frac{\text{MeV}}{\text{dis}}.$$

The above form assumes that only one X ray is emitted and that it has an energy equal to η. Actually, a number of X rays and/or Auger electrons may be emitted, but the total energy of all these products will equal to η. This simple form gives an answer which is conserative and saves undue effort.

2. Radioactive Chain Factor, F

The radioactive chain factor appears in the calculations when one must deal with a decay chain. The ICRP assumes that only the parent is present when the radionuclide is taken into the body. However, the effect of daughter products which are formed by decay in the organ are taken into account through the use of the factor F, which is defined as the ratio at time t of the number of disintegrations per unit time of the daughter atoms to the number of disintegrations per unit time of parent atoms in the critical organ. The product of F_i and the absorbed energy of the i^{th} daughter gives the weighted contribution of the i^{th} daughter. The sum of the contributions from the parent and all the daughters will then give the equivalent energy absorbed if the decay chain were to act as a single radionuclide.

The factor F_0, where the subscript zero refers to the parent, equals one. The factor for the ith daughter can be found from:

$$F_i = \frac{\prod_{j=1}^{i} T_j/T_j^r}{1 - e^{-\lambda_0 t}} \sum_{n=0}^{i} \frac{T_n^i(1 - e^{-\lambda_n t})}{\prod_{\substack{p=0 \\ p \neq n}}^{i} (T_n - T_p)}.$$

In this equation, λ_0 is the total decay constant of the parent (see page 147) and λ_i is the total decay constant of the ith daughter. The time t is taken as 50 yr, and T_i is the total half-life of the ith daughter. The symbol T_i^r stands for the radioactive half-life of the ith daughter.

Table 5a of the ICRP report(2) contains a listing of the effective energies for many decay chains. Values for the factor F_i are also contained in this table for each daughter nuclide in the chain.

3. Relative Biological Effectiveness, RBE

It was pointed out in Section 4 that the ICRU wishes to restrict the use of the term RBE to radiobiology only. In regard to protection purposes, the term "quality factor (QF)" is suggested to replace the former term RBE.

As used in experimental work, the RBE is often defined as the inverse ratio of the absorbed doses of two kinds of radiation that produce the same effect. Other factors which affect the response must be the same or the results will not give a measure of the RBE. Thus, in order to talk about an RBE one must define exposure conditions as well as the effect studied.

For protection work, the dose levels which are used are well below those employed in experiments. Also, more than one effect must be taken into account for different types of radiation. Over and above this, one cannot control the many factors which may affect the exposure. For this reason, a distinction should be made between RBE and the factor used in protection work.

The ICRP endorses the use of the term quality factor for protection purposes.(4) This factor is applied without regard to exposure factors, and for all biological effects. The values of QF which are to be used are the same as those formerly termed RBE values. They are not intended to imply what the concept RBE conveys. They do provide a gross quality correction for the type of radiation so that one may use a common scale for all radiation. For internal dose considerations, the QF values are:

QF = 1 for β^-, β^+, e^-, X and γ rays;

= 1.7 for β^-, β^+, e^- with $E_{max} \leqq 0.03$ MeV;

= 10 for α;

= 20 for recoil atoms.

4. Relative Damage Factor, n

In the case of bone-seeking radionuclides, the amount of the nuclide permitted in the bone is based upon a direct comparison with ^{226}Ra. However, some radionuclides appear to produce more damage to the bone than an equal dose of ^{226}Ra. To account somewhat for this fact, the ICRP recommends the use of the relative damage factor, n. This factor applies only in the case of bone-seeking nuclides. Subject to the stated conditions, the value of n is:

n = 1, if (a) the parent of the chain is an isotope of radium, or
(b) the energy component originates as X or γ radiation;

= 5, for all other cases.

Table 5 of the ICRP report(2) contains a listing of the value of ΣEF(RBE)n for many of the radionuclides. The sample calculations at the end of this section contain examples of the use of the factors to find ΣEF(RBE)n.

E. Other Factors

Table 12 of the ICRP report(2) contains much other information needed to make the calculations. Although many sources were reviewed for needed data, little was learned concerning these factors. For this reason, many approximations have been made for the distribution fractions and biological half-lives found in this table.

Values of the effective or total half-life are also listed in Table 12. Since the process of decay and elimination are both exponential, each will have a certain value for the decay constant. The total fraction lost by the organ per unit time, λ, will be the sum of the two individual decay constants:

$$\lambda = \lambda_r + \lambda_b.$$

From the definition of half-life,

$$\frac{0.693}{T} = \frac{0.693}{T_r} + \frac{0.693}{T_b}.$$

From this, we derive

$$T = \frac{T_b T_r}{T_b + T_r},$$

where the subscripts b and r refer to biological and radioactive, respectively. The total or effective half-life, T, is then the time for one-half of the nuclide to be removed from the organ. If the half-life for either process is very large compared with that of the other, T approaches the value of the smaller of the two as an upper limit. In most cases, the radioactive half-life is well known, but the biological half-life is often poorly known. This can be traced to the lack of continuous exposure data. Thus, many of the values have been extrapolated from single dose studies and may not represent too well the situation for the continuous case.

F. Maximum Permissible Body Burden

Early in this section, the term organ burden was mentioned as the amount in the organ which would deliver the stated dose equivalent rate. Since a nuclide may be deposited in a number of organs, the amount in the total body becomes important. The ICRP uses the term "Maximum Permissible Body Burden" (MPBB) and the letter q to denote the amount of a radionuclide, in μCi, allowed in the total body. The body burden q is the amount in the total body such that the allowed amount in the critical organ is not exceeded. In all cases, the organ burden for a given nuclide will be some fraction of the body burden q. The value of this fraction will depend upon how the nuclide is distributed in the body. If the substance is taken up by a number of organs, each will have its own fraction (f_2) of the body burden. For the organ which is taken as the critical organ for the assumed nuclide, the product qf_2 gives the organ burden. For the other organs which receive the radionuclide, the product qf_2 for these organs results in less than the organ burden.

There are two criteria which the ICRP uses to determine body burdens:

(1) for bone-seeking radionuclides - the body burden is estimated by a direct comparison with ^{226}Ra and its daughter products. However, the radionuclide in question must emit significant amounts of particulate radiation. In the event it does not, then it falls under the second criterion.

(2) for all other radionuclides - the body burden is set to limit the weekly dose equivalent received by a given organ. These limits are:

   (a) 0.1 rem/wk to the gonads and total body;
   (b) 0.6 rem/wk to the skin and thyroid;
   (c) 0.3 rem/wk to all other soft tissue.

One should note from the first criterion that a bone-seeker which emits only X and γ rays is not compared with ^{226}Ra. Inasmuch as X and γ rays have a much greater range in tissue than do particles, the nearby tissue will be exposed almost to the same extent as bone. Consequently, the body burden should be limited by part c above.

G. Body Burden Based on Radium

The body burden q for ^{226}Ra has been set at 0.1 μg (~0.1 μCi),(11) as based upon direct observations of humans.(12) Down through the years, the best source of data about internally deposited radionuclides has been the studies concerning radium deposition in humans. This wide experience has led to the adoption of radium as the standard for the chronic effects of bone-seekers on humans.

Chronic exposure data furnishes the value of q directly. However, these data are lacking for most radionuclides, so that a comparison method is used in many cases to find q. In this method, the energy given up in bone by the radionuclide is set equal to the energy given up by 0.1 μCi of ^{226}Ra and its daughter products. The relative damage factor n is used to modify the energy term of the given radionuclide. This allows for the greater effectiveness of some radionuclides as compared with radium.

For a given radionuclide X, one can write

$$\left[qf_2\Sigma EF(RBE)n\right]_X = \left[qf_2\Sigma EF(RBE)n\right]_{Ra}.$$

In this equation, $q = 0.1$ μCi, $f_2 = 0.99$, and $\Sigma EF(RBE)n = 110$ MeV/dis for radium. When the symbols are replaced by these values, q is given in μCi by

$$q_X = \frac{11}{[f_2\Sigma EF(RBE)n]_X}.$$

The average absorbed dose rate to the bone from 0.1 μCi of ^{226}Ra and its daughters turns out to be 0.06 rad/wk. In terms of the dose equivalent rate, the figure is 0.56 rem/wk. Although this dose equivalent rate is higher than that allowed for most other organs of the body, medical data have shown that 0.1 μCi of radium is a safe burden.(12)

H. Body Burden Based on Dose Equivalent Rate

The lack of data for other than bone-seeking radionuclides precludes the use of a standard for these cases. One must calculate the body burden based upon the equivalent dose rate in the critical organ.

For a given organ of mass m (in gm), the permissible dose equivalent rate will be R rem/wk. If one permits a body burden of q (in μCi) in the total body, then qf_2 is the amount in the assumed organ. The amount allowed in the organ will be that which gives R rem/wk, or

$$R\left(\frac{\text{rem}}{\text{wk}}\right) = \frac{qf_2(\mu\text{Ci})\ 2.22 \times 10^6}{m(\text{gm})}\left(\frac{\text{dis}}{\text{m}\mu\text{Ci}}\right) \Sigma E(\text{RBE})\left(\frac{\text{MeV}}{\text{dis}}\right) 1.6 \times 10^{-6}\left(\frac{\text{ergs}}{\text{MeV}}\right)$$

$$1.008 \times 10^4\left(\frac{\text{m}}{\text{wk}}\right)\frac{1\ \text{rad}}{100(\text{ergs/gm})} = \frac{358\ qf_2 \Sigma E(\text{RBE})}{m}.$$

One then solves for q, which gives

$$q = \frac{2.8 \times 10^{-3}\ mR}{f_2 \Sigma E(\text{RBE})} \quad (\text{in } \mu\text{Ci}).$$

The value of R for a given organ was discussed on page 149. The fraction f_2 and the mass m are given in Table 12 of the ICRP report.(2)

I. MPC in Air and Water - Single Radionuclide

Let us consider for the moment a concentration C_i (μCi/cc) of a radionuclide in air or water. In this case, the subscript i denotes either method of intake. Assume that C_i remains constant during the time a person is exposed to the concentration. The average breathing and water intake rates (cc/day) of the standard man are also assumed to be constant. This means that the rate of uptake P_i (μCi/day) of the radionuclide in a given organ will be constant.

If there were no elimination from the organ, the amount in the organ would increase at a constant rate. However, elimination does occur and the fraction eliminated per unit time is given by $\lambda (= 0.693/T)$ for the given organ. The rate of elimination will not be a constant. It will depend upon the amount of the radionuclide which is in the organ at any given time.

The rate of change of the amount in the organ $\left[\frac{d(q'f_2)}{dt}\right]$ will be given by

$$\frac{d(q'f_2)}{dt} = P_i - \lambda(q'f_2),$$

where $q'f_2$ stands for the amount in the organ at any time t. This differential equation has the solution

$$q'f_2 = \frac{P_i}{\lambda}(1 - e^{-\lambda t}),$$

for the condition that $q'f_2 = 0$ when $t = 0$. This equation relates the amount in the organ, $q'f_2$, to the rate of uptake, P_i, at any time t. If q' is set equal to the body burden q for the given organ, then qf_2 will be the allowed amount in the organ. One can then solve for the allowed rate of uptake, P_i, which would result in an organ burden qf_2 at any time t:

$$P_i = \frac{\lambda q f_2}{1 - e^{-\lambda t}} = \frac{0.693\, qf_2}{T(1 - e^{-0.693t/T})}.$$

Let S_i stand for the rate of intake (cc/day) of either air or water. Also, let f_i be the fraction of that inhaled or ingested which gets to the organ of reference. Then, the rate of uptake P_i (μCi/day) is given by

$$P_i = C_i(\mu\text{Ci/cc})\ S_i(\text{cc/day})\ \ f_i(\text{fraction arriving at organ}).$$

The <u>allowed</u> rate of uptake for the given method of intake results when C_i is replaced by $(MPC)_i$. From the two expressions for P_i, one arrives at

$$(MPC)_i = \frac{0.693\, qf_2}{S_i f_i T(1 - e^{-0.693t/T})}\ \frac{\mu\text{Ci}}{\text{cc}},$$

where t and T are in days.

One can then solve this equation for the Maximum Permissible Concentration in air and/or in water. The value of the factor S_i depends on the method of intake and the type of exposure. For the case of continuous exposure, $S_a = 2 \times 10^7$ cc/day and $S_w = 2.2 \times 10^3$ cc/day, where the subscripts a and w denote inhalation and ingestion, respectively. For occupational exposure (40 hr/wk), $S_a = 6.9 \times 10^6$ cc/day and $S_w = 750$ cc/day. The MPC values based on a 40-hr/wk exposure become

$$(MPC)_a = \frac{10^{-7}\, qf_2}{f_a T(1 - e^{-0.693t/T})}\ \mu\text{Ci/cc};$$

$$(MPC)_w = \frac{9.2 \times 10^{-4}\, qf_2}{f_w T(1 - e^{-0.693t/T})}\ \mu\text{Ci/cc},$$

in which

f_a = the fraction of inhaled radionuclide reaching the organ of reference and

f_w = fraction of that taken into the body by ingestion which is retained in the critical organ.

Values for both the fractions, f_a and f_w, can be found in Table 12 of the ICRP report.(2)

In the above expressions, the ICRP has set t = 50 yr in order to obtain the recommended values of $(MPC)_a$ and $(MPC)_w$. It should be noted that this assumes that the worker is exposed to a constant concentration of the radionuclide throughout the 50-yr period in order to reach the organ burden. For short-term exposures, the concentrations can be many times the MPC values without creating a serious hazard.

J. MPC Formulae for a Radioactive Series

If the radionuclide is the parent of a decay chain, one must consider the contribution of the daughter products. In this case, the dose equivalent rate R rem/wk will be given by

$$R\left(\frac{\text{rem}}{\text{wk}}\right) = \frac{358}{m} \sum_{i=0}^{k} (qf_2)_i E_i (RBE)_i n_i,$$

where the sum is taken over the k members of the chain. This may be written as

$$R = \frac{358}{m} (qf_2)_0 \sum_{i=0}^{k} \frac{(qf_2)_i}{(qf_2)_0} E_i (RBE)_i n_i.$$

Since

$$F_i = \frac{(qf_2)_i}{(qf_2)_0},$$

we may rewrite this expression as

$$R\left(\frac{\text{rem}}{\text{wk}}\right) = \frac{358}{m} (qf_2)_0 \sum_{i=0}^{k} E_i F_i (RBE)_i n_i.$$

The body burden q_0 for the parent is then given by

$$q_0 = \frac{2.8 \times 10^{-3}\ mR}{(f_2)_0 \sum_{i=0}^{k} E_i F_i (RBE)_i n_i}\ \mu Ci.$$

The weighted energy sum in this expression allows one to treat the whole chain as a single radionuclide. The product $(qf_2)_0$ may then be replaced in the general MPC formula by the above value. The MPC values for occupational exposure are then

$$(MPC)_a = \frac{10^{-7}\ qf_2}{f_a T(1 - e^{-0.693t/T})} = \frac{2.8 \times 10^{-10}\ mR}{f_a T_0 (1 - e^{-0.693t/T_0}) \sum_{i=0}^{k} E_i F_i (RBE)_i n_i}\ \frac{\mu Ci}{cc}$$

and

$$(MPC)_w = \frac{9.2 \times 10^{-4}\ qf_2}{f_w T(1 - e^{-0.693t/T})} = \frac{2.58 \times 10^{-6}\ mR}{f_w T_0 (1 - e^{-0.693t/T_0}) \sum_{i=0}^{k} E_i F_i (RBE)_i n_i}\ \frac{\mu Ci}{cc}.$$

In these relationships, T_0 is the effective half-life for the parent of the decay chain. If bone is the critical organ, R = 0.56 rem/wk when particulate radiation is involved. For other organs, R has the values discussed on page 149. For continuous exposure, replace 2.8×10^{-10} and 2.58×10^{-6} by 9.6×10^{-11} and 8.8×10^{-7}, respectively. An example of the use of the above formulae can be found at the end of this section.

K. MPC of Noble Gases - Submersion Exposure

The problem of setting MPC values for inert gases involves external 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. One is then concerned with the exposure of a person surrounded by a semispherical infinite cloud of radioactive gas. The radius of the cloud is then so large that one can assume that the energy emitted per unit volume equals the energy absorbed per unit volume. One can further assume that only one-half of the total solid angle contributes to the irradiation. For occupational exposure, the time that the worker is exposed is taken as 40 hr/wk for 50 out of 52 wk. The dose equivalent rate R(rem/wk) in tissue will be given by the expression

$$R\left(\frac{\text{rem}}{\text{wk}}\right) = \frac{1}{2}\left(\frac{S}{\rho}\right)_t C\left(\frac{\mu\text{Ci}}{\text{cc}}\right) \frac{3.7 \times 10^4}{\rho_a(\text{gm/cc})} \left(\frac{\text{dps}}{\mu\text{Ci}}\right) \Sigma E(\text{RBE})\left(\frac{\text{MeV}}{\text{dis}}\right) 1.6 \times 10^{-6}\left(\frac{\text{ergs}}{\text{MeV}}\right)$$

$$1.385 \times 10^5\left(\frac{\text{sec}}{\text{wk}}\right) \frac{1 \text{ rad}}{100(\text{ergs/gm})} = 41\left(\frac{S}{\rho}\right)_t \frac{C\Sigma E(\text{RBE})}{\rho_a},$$

where $C(\mu\text{Ci/cc})$ is the concentration of the radionuclide, $\rho_a(\text{gm/cc})$ is the density of air, and $(S/\rho)_t$ is the relative mass stopping power of tissue with respect to air. If R is set equal to the allowed dose equivalent rate, then C can be replaced by $(\text{MPC})_a$:

$$(\text{MPC})_a = \frac{\rho_a R}{41(S/\rho)_t \ \Sigma E(\text{RBE})} \frac{\mu\text{Ci}}{\text{cc}}.$$

In the case of a γ or high-energy $\beta(E_{max} \geqq 0.1 \text{ MeV})$ emitter, the total body will receive the exposure. Then, $R = 0.1$ rem/wk, and for β and γ,QF(RBE) = 1. The value of $(S/\rho)_t$ can be taken as 1.13 and ρ_a as 0.0012 gm/cc. Using these values, one gets

$$(\text{MPC})_a = \frac{(1.2 \times 10^{-3})(0.1)}{41(1.13) \ \Sigma E} = \frac{2.6 \times 10^{-6}}{\Sigma E} \frac{\mu\text{Ci}}{\text{cc}}.$$

If low-energy radiation is involved, then skin becomes the organ of concern and

$$(\text{MPC})_a = \frac{1.56 \times 10^{-5}}{\Sigma E} \frac{\mu\text{Ci}}{\text{cc}}.$$

One should note that since this is an external source of exposure, the total dose equivalent varies directly with the exposure time. If the individual is exposed continuously to the radioactive cloud, replace 2.6×10^{-6} and 1.56×10^{-5} by the values 5.9×10^{-7} and 3.5×10^{-6}, respectively. From the protection standpoint, one would be concerned with concentrations many times these MPC values, even for short term exposures.

L. MPC Values for a Mixture of Radionuclides

Suppose that the air or water contains a mixture of radionuclides. Assume that both the identity and concentration of each of the members is known. Let C_i represent the concentration of the i^{th} member of permissible concentration $(\text{MPC})_i$. Then, if n radionuclides are present and the following inequality holds:

$$\frac{C_1}{(\text{MPC})_1} + \frac{C_2}{(\text{MPC})_2} \cdots + \frac{C_n}{(\text{MPC})_n} \leqq 1,$$

the mixture can be taken as within the limits.

For the cases in which either the identity or the concentration of one or all the members is not known, one may refer to the AEC Manual, Chapter 0524 - Standards for Radiation Protection. Annex 1 contains the rules which apply in these cases.

M. Dose Equivalent from Single Intake

At times one is faced with the problem of a single intake of a radionuclide rather than with long-term exposure. Two facets of the problem must be taken into account. First, for a large intake in a short time, a good portion of the substance may just pass through the body without absorption. For this condition, a portion of the gastrointestinal (GI) tract may receive the most exposure. This part of the problem is treated somewhat in the ICRP report(2) and also in Ref. 9 and 10. Also treated is the problem of continuous exposure when the GI tract is the critical organ. This occurs most often when the inhaled or ingested substance is insoluble.

The other part of the problem concerns the portion of the radioactive material which is absorbed. A certain amount of the substance then ends up in some of the body organs. In the case of acute exposure, the radionuclide may be distributed much differently than in the case of long-term exposure, that is, one may find larger amounts of a bone-seeking radionuclide in other organs than in bone following an acute intake. This occurs because of the difference in distribution and retention. For a single intake, a larger fraction of the substance may be taken up by soft tissue than by bone. However, this matter may be quickly eliminated from soft tissue, whereas the portion in bone may be very slowly eliminated. Over a long period of time the amount in the bone will build up. This makes the bone the organ of concern for long-term exposure. For the acute intake case, then, one is interested in the short-term distribution in the body.

Let us assume that one has a measure of the amount $I_0(\mu Ci)$ of a radionuclide in a given organ. If this is a single intake, our concern is only with this amount, that is, we ignore any past accumulation and assume no further accumulation. We will use the compartment model to describe the elimination of this product. The amount $I(\mu Ci)$ which is present at any time t after the intake is completed will be given by

$$I(\mu Ci) = I_0 e^{-\lambda t}.$$

The dose equivalent rate in the given organ will be given by

$$\frac{d\,DE}{dt}\left(\frac{rem}{day}\right) = \frac{I(\mu Ci)\ 2.22 \times 10^{6}}{m(gm)}\left(\frac{dis}{m\text{-}\mu Ci}\right)\ \Sigma E(RBE)n\left(\frac{MeV}{dis}\right)\ 1.6 \times 10^{-6}\left(\frac{ergs}{MeV}\right)$$

$$1.44 \times 10^{3}\left(\frac{m}{day}\right)\frac{1\ rad}{100\ (erg/gm)} = \frac{51.15\ I\Sigma E(RBE)n}{m}.$$

The dose equivalent rate at any time t in terms of the initial amount I_0 is given by

$$\frac{d\,DE}{dt}\left(\frac{\text{rem}}{\text{day}}\right) = \frac{51.15\ I_0 \Sigma E(RBE)n}{m}\ e^{-\lambda t}.$$

The total dose equivalent DE can be found by integrating the above expression with respect to t:

$$DE = \frac{74\ I_0 T \Sigma E(RBE)n}{m}\ (1 - e^{-0.693t/T})\ \text{rem}.$$

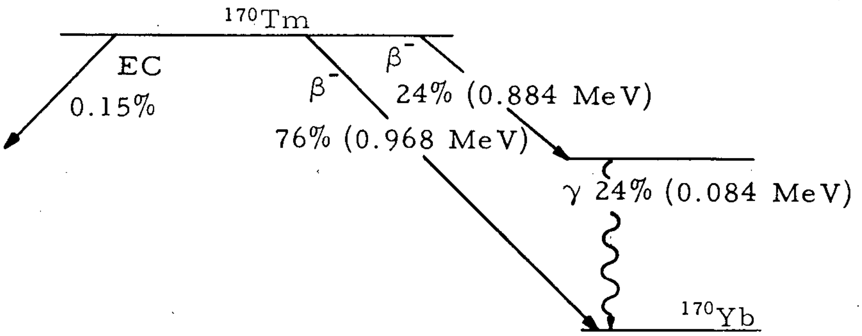
This equation gives the total dose equivalent DE delivered in the time t.

This completes the present discussion on internal dose. For a more thorough treatment of the problem one may refer to the ICRP report(2) and the many sources of reference contained therein.

In an attempt to illustrate the use of the formulas presented and discussed in this section, a number of example calculations are included.

N. Sample Calculations of $(MPC)_a$ and $(MPC)_w$

(1) Calculate $(MPC)_a$ and $(MPC)_w$ for ${}^{170}_{69}\text{Tm}$ for bone as the critical organ. Referring to page 746, Ref. 14, one obtains the decay scheme for ${}^{170}\text{Tm}$:



The electron capture may be ignored in this case. One can write:

$$\Sigma EF(RBE)n = \sum_{i,j}^{n} f_{\gamma_i} E_{\gamma_i} (1 - e^{-\sigma_i x})\ F_i (RBE)_i n_i + f_{\beta_j^-} E_{\beta_j^-} (0.33) \left(1 - \frac{Z^{1/2}}{50}\right)$$

$$\left(1 + \frac{E_{\beta_j^-}^{1/2}}{4}\right) F_j (RBE)_j n_j ,$$

where $F_i = F_j = 1$; $(RBE)_\gamma = (RBE)_{\beta^-} = 1$; $n_\gamma = 1$; $n_{\beta^-} = 5$; $Z = 69$; $x = 5$ cm (from Table 8 - Ref. 2); and σ_i is obtained from Ref. 8. Then,

$$\Sigma EF(RBE)n = (0.24)(0.084)[1 - e^{-0.0265(5)}]\,(1)(1)(1) + (0.76)(0.968)(0.33)$$

$$\left(1 - \frac{\sqrt{69}}{50}\right)\left(1 + \frac{\sqrt{0.968}}{4}\right)(1)(1)(5) + (0.24)(0.884)(0.33)$$

$$\left(1 - \frac{\sqrt{69}}{50}\right)\left(1 + \frac{\sqrt{0.884}}{4}\right) x\ (1)(1)(5)$$

$$= 0.002 + 1.3 + 0.4 \cong 1.7\ \text{MeV/dis.}$$

Since the critical organ is bone and there is particulate radiation involved, the body burden is based on a comparison with ^{226}Ra:

$$q = \frac{11}{f_2 \Sigma EF(RBE)n}$$

or

$$qf_2 = \frac{11}{1.7} = 6.47\ \mu\text{Ci}$$

in the critical organ. Thus,

$$(MPC)_a = \frac{10^{-7}\, qf_2}{Tf_a(1 - e^{-0.693t/T})}$$

for occupational exposure. From Table 12 of Ref. 2, $T = 113$ days, and $f_a = 0.16$ for bone as the critical organ. For $t = 50$ yr,

$$[1 - e^{-0.693t/113}] \cong 1.$$

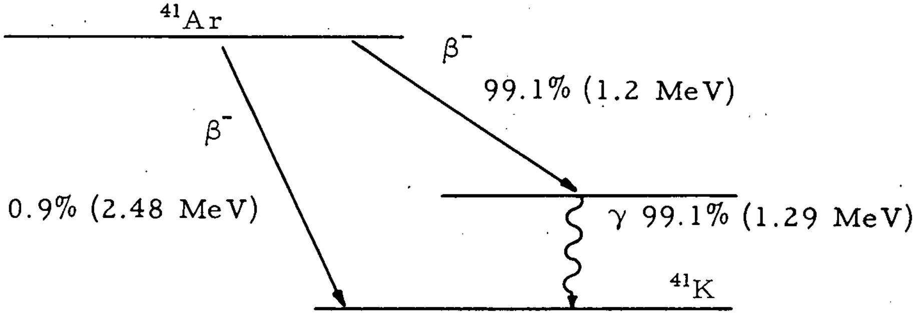
Therefore,

$$(MPC)_a = \frac{10^{-7}(6.47)}{113(0.16)(1)} \cong 4 \times 10^{-8}\ \mu\text{Ci/cc}.$$

Similarly, $f_w = 6.5 \times 10^{-5}$ from Table 12, so that

$$(MPC)_w = \frac{9.2 \times 10^{-4}\, qf_2}{Tf_w(1 - e^{-0.693t/T})} = \frac{9.2 \times 10^{-4}(6.47)}{113(6.5 \times 10^{-5})(1)} \cong 0.8\ \mu\text{Ci/cc}.$$

(2) Calculate the $(\text{MPC})_a$ for ${}^{41}_{18}\text{Ar}$. The decay scheme appears on page 618 of Ref. 14:



Since argon is an inert gas, one can use the formula for submersion, in which the dose is based on external rather than internal exposure. For this case, one can write

$$\Sigma E = \sum_{i,j}^{n} f_{\gamma_i} E_{\gamma_i}[1 - e^{-\sigma_i x}] + f_{\beta_j^-} E_{\beta_j^-}(0.33)\left(1 - \frac{Z^{1/2}}{50}\right)\left(1 + \frac{E^{1/2}_{\beta_j^-}}{4}\right).$$

The total body becomes the critical organ and the gamma travels through air. One must use σ_i for air, and x is then the radius of the gas cloud. For a cloud of large volume, $[1 - e^{-\sigma_i x}] \cong 1$. Then,

$$\Sigma E = (0.99)(1.29)(1) + (0.009)(2.48)(0.33)\left[1 - \frac{\sqrt{18}}{50}\right]\left[1 + \frac{\sqrt{2.48}}{4}\right]$$

$$+ (0.99)(1.2)(0.33)\left[1 - \frac{\sqrt{18}}{50}\right]\left[1 + \frac{\sqrt{1.2}}{4}\right]$$

$$= 1.28 + 0.01 + 0.46 \cong 1.8 \text{ MeV/dis.}$$

Therefore,

$$(\text{MPC})_a = \frac{2.6 \times 10^{-6}}{\Sigma E} = \frac{2.6 \times 10^{-6}}{1.8} = 1.4 \times 10^{-6}\ \mu\text{Ci/cc}.$$

(3) Find the occupational $(\text{MPC})_a$ and $(\text{MPC})_w$ for ${}^{147}_{60}\text{Nd}$ plus daughters considering bone as the critical organ. The decay chain is

$${}^{147}_{60}\text{Nd} \xrightarrow{\beta^-,\ \gamma} {}^{147}_{61}\text{Pm} \xrightarrow{\beta^-} {}^{147}_{62}\text{Sm} \xrightarrow{\alpha} {}^{143}_{60}\text{Nd}$$

The factor F for each of the products appears in the effective energy term. Our first step will be to determine this factor for the daughter products. The factor for the parent, F_0, is taken as 1. The others are obtained from the expression

$$F_i = \frac{\prod\limits_{j=1}^{i} T_j/T_j^r}{1 - e^{-\lambda_0 t}} \sum_{n=0}^{i} \frac{T_n^i(1 - e^{-\lambda_n t})}{\prod\limits_{\substack{p=0 \\ p \neq n}}^{i} (T_n - T_p)}.$$

Hence,

$$F_{Nd} = F_0 = 1.$$

$$F_{Pm} = F_1 = \frac{T_1}{T_1^r(1 - e^{-\lambda_0 t})} \left[\frac{T_0(1 - e^{-\lambda_0 t})}{T_0 - T_1} + \frac{T_1(1 - e^{-\lambda_1 t})}{T_1 - T_0}\right].$$

$$F_{Sm} = F_2 = \frac{T_1 T_2}{T_1^r T_2^r(1 - e^{-\lambda_0 t})}\left[\frac{T_0^2(1 - e^{-\lambda_0 t})}{(T_0 - T_1)(T_0 - T_2)} + \frac{T_1^2(1 - e^{-\lambda_1 t})}{(T_1 - T_0)(T_1 - T_2)} + \frac{T_2^2(1 - e^{-\lambda_2 t})}{(T_2 - T_0)(T_2 - T_1)}\right].$$

From Table 12 - Ref. 2,

$T_0 = 11.2$ d, $T_1 = 570$ d, $T_2 = 1500$ d,

$T_0^r = 11.3$ d, $T_1^r = 920$ d, $T_2^r = 4.8 \times 10^{13}$ d.

For t = 50 yr, the terms $(1 - e^{-\lambda_n t}) \approx 1$, and

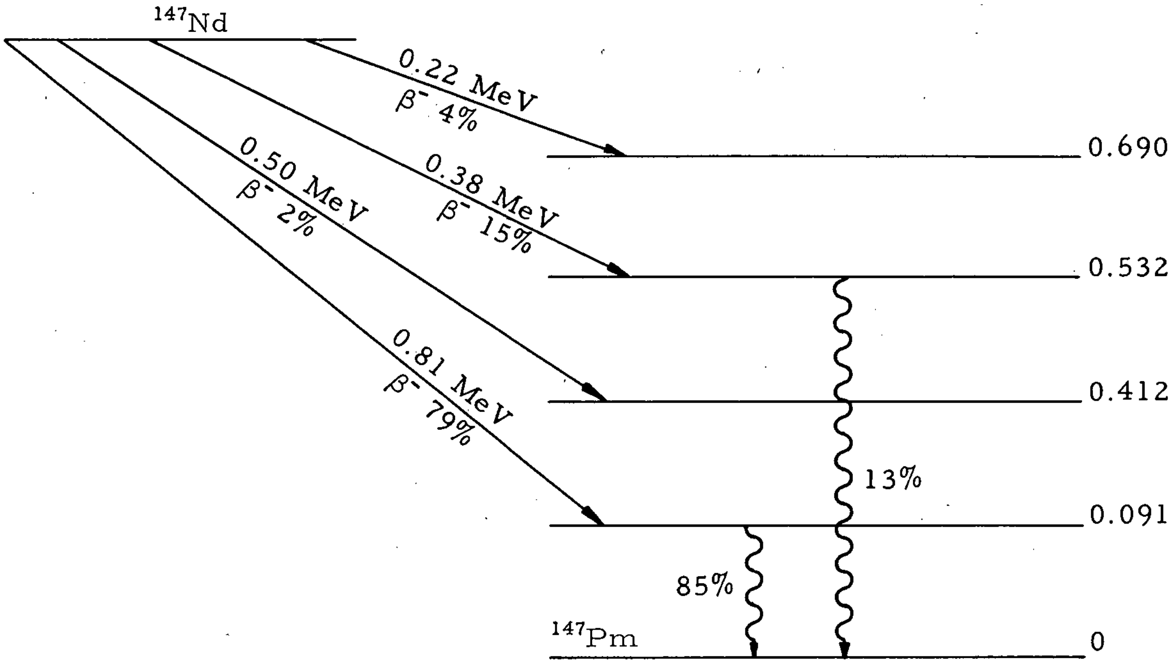
$$F_1 = \frac{T_1}{T_1^r}\left[\frac{T_0}{T_0 - T_1} + \frac{T_1}{T_1 - T_0}\right] = \frac{T_1}{T_1^r} = \frac{570}{920} = 0.62$$

$$F_2 = \frac{T_1 T_2}{T_1^r T_2^r}\left[\frac{T_0^2}{(T_0 - T_1)(T_0 - T_2)} + \frac{T_1^2}{(T_1 - T_0)(T_1 - T_2)} + \frac{T_2^2}{(T_2 - T_0)(T_2 - T_1)}\right]$$

$$F_2 = \frac{(570)(1500)}{(920)(4.8 \times 10^{13})}\left[\frac{(11.2)^2}{(11.2 - 570)(11.2 - 1500)} + \frac{(570)^2}{(570 - 11.2)(570 - 1500)} + \frac{(1500)^2}{(1500 - 11.2)(1500 - 570)}\right]$$

$$= 1.94 \times 10^{-11}[0.00015 - 0.63 + 1.63] = 1.94 \times 10^{-11}.$$

The effective energy of the chain will be the sum of the terms for each member of the chain. Again, from Ref. 14 - page 726,



Other gammas are emitted but have not been included because their contribution is small. One then gets

$$[\Sigma EF(RBE)n]_{Nd} = \sum_{i,j}^{n} f_{\gamma_i} E_{\gamma_i}(1 - e^{-\sigma_i x})\ F_i(RBE)_i n_i + f_{\beta_j^-} E_{\beta_j^-}(0.33)\left(1 - \frac{Z^{1/2}}{50}\right)$$

$$\left(1 + \frac{E^{1/2}_{\beta_j^-}}{4}\right) F_j(RBE)_j n_j .$$

$$= 0.85(0.091)(1 - e^{-(0.026)5})(1)(1)(1) + (0.13)(0.532)$$

$$(1 - e^{-(0.032)5})(1)(1)(1) + (0.79)(0.81)(0.33)\left(1 - \frac{\sqrt{60}}{50}\right)$$

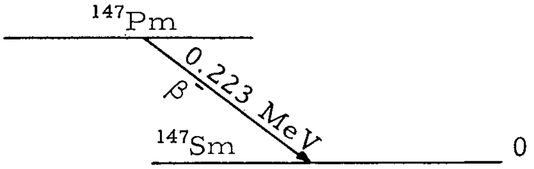
$$\left(1 + \frac{\sqrt{0.81}}{4}\right)(1)(1)(5) + (0.02)(0.5)(0.33)\left(1 - \frac{\sqrt{60}}{50}\right)\left(1 + \frac{\sqrt{0.5}}{4}\right)$$

$$(1)(1)(5) + (0.15)(0.38)(0.33)\left(1 - \frac{\sqrt{60}}{50}\right)\left(1 + \frac{\sqrt{0.38}}{4}\right)(1)(1)(5)$$

$$+ (0.04)(0.22)(0.33)\left(1 - \frac{\sqrt{60}}{50}\right)\left(1 + \frac{\sqrt{0.22}}{4}\right)(1)(1)(5).$$

$$[\Sigma EF(RBE)n]_{Nd} = 0.01 + 0.01 + 1.09 + 0.02 + 0.09 + 0.01 = 1.23 \text{ MeV/dis.}$$

The decay scheme of ^{147}Pm is found on page 727 of Ref. 14:



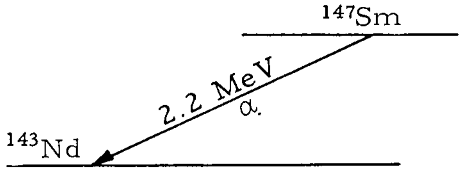
From this, one gets:

$$[\Sigma EF(RBE)n]_{Pm} = f_{\beta^-} E_{\beta^-} (0.33)\left(1 - \frac{Z^{1/2}}{50}\right)\left(1 + \frac{E^{1/2}}{4}\right) F_{Pm}(RBE)_{\beta^-} n_{\beta^-}$$

$$= (1)(0.223)(0.33)\left(1 - \frac{\sqrt{61}}{50}\right)\left(1 + \frac{\sqrt{0.223}}{4}\right)$$

$$(0.62)(1)(5) = 0.22 \text{ MeV/dis.}$$

The decay of ^{147}Sm is by alpha emission to stable ^{143}Nd (p. 729 - Ref. 14):



The effective energy term will include the contribution from the recoil atom:

$$[\Sigma EF(RBE)n]_{Sm} = f_\alpha E_\alpha F_{Sm}(RBE)_\alpha n_\alpha + f_\alpha E_\alpha \frac{m_\alpha}{m_r} F_{Sm}(RBE)_r n_r$$

$$= (1)(2.2)(1.9 \times 10^{-11})(10)(5) + (1)(2.2)\frac{4}{143}$$

$$(1.9 \times 10^{-11})(20)(5)$$

$$= 2.09 \times 10^{-9} + 0.12 \times 10^{-9} = 2.21 \times 10^{-9} \text{ MeV/dis.}$$

The effective energy term for the chain then becomes

$$\sum_{i=0}^{k} E_i F_i (RBE)_i n_i = [\Sigma EF(RBE)n]_{Nd} + [\Sigma EF(RBE)n]_{Pm} + [\Sigma EF(RBE)n]_{Sm}$$

$$= 1.23 + 0.22 + 2.21 \times 10^{-9} = 1.45 \text{ MeV/dis.}$$

The $(MPC)_a$ and $(MPC)_w$ for a decay chain are obtained from:

$$(MPC)_a = \frac{2.8 \times 10^{-10}\, mR}{f_a T_0 (1 - e^{-0.693t/T_0}) \sum_{i=0}^{k} E_i F_i (RBE)_i n_i} \ \mu Ci/cc,$$

and

$$(MPC)_w = \frac{2.58 \times 10^{-6}\, mR}{f_w T_0 (1 - e^{-0.693t/T_0}) \sum_{i=0}^{k} E_i F_i (RBE)_i n_i} \ \mu Ci/cc.$$

The parameters in this case become:

$$\text{for bone as the critical organ} \quad \begin{cases} R = 0.56 \text{ rem/wk} \\ m = 7 \times 10^3 \text{ gm} \end{cases}$$

$$\text{with Nd as the parent of the chain} \quad \begin{cases} T_0 = 11.2 \text{ d} \\ f_a = 0.09 \\ f_w = 3.5 \times 10^{-5} \end{cases}$$

One then solves the expressions with t = 50 yr. This gives $(1 - e^{-0.693t/T_0}) \cong 1$, and one arrives at

$$(MPC)_a = \frac{2.8 \times 10^{-10}(7 \times 10^3)(0.56)}{(0.09)(11.2)(1)(1.45)} \cong 8 \times 10^{-7} \ \mu Ci/cc,$$

and

$$(MPC)_w = \frac{2.58 \times 10^{-6}(7 \times 10^3)(0.56)}{(3.5 \times 10^{-5})(11.2)(1)(1.45)} \cong 20 \ \mu Ci/cc.$$

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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 he first have a measure of the radiation field in regard to some standard. Then, he can evaluate his results with respect to the relative hazard in terms of this yardstick. Based upon this analysis, he 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 radiation protection guides (RPG).(1) These guides serve as the standards that should not be exceeded without careful thought about the reasons for so doing. To compare the radiation field to these standards, we must have a means of measuring the field in terms of defined units.(2) 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.(3-8) 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.(4) 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.(7) Other special processes, such as the Čerenkov effect(4) 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 alpha and beta, 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.

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,(2) 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.(5,8,9)

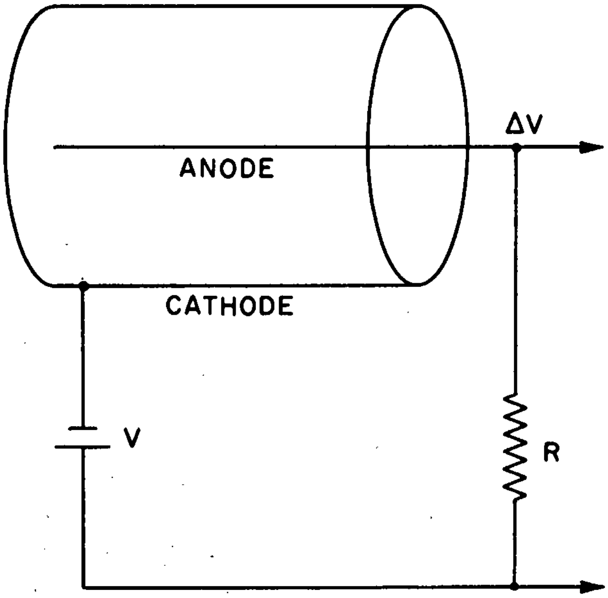


Fig. 10.1. Two-electrode, Gas-filled Chamber

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 Fig. 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 wall 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 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 <u>detect</u> 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 Fig. 10.2, which gives the relative pulse size produced by both α (solid curve) and β (dashed curve) in a given chamber.

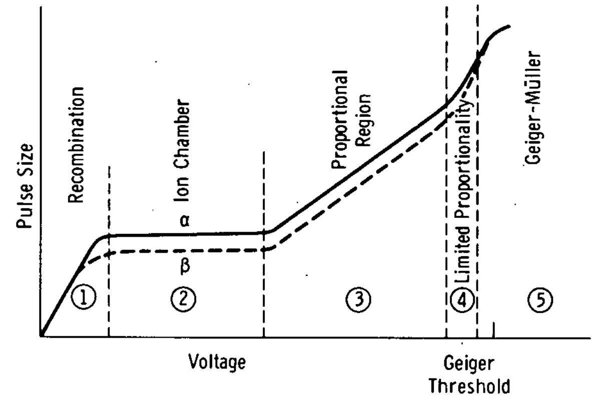


Fig. 10.2. Pulse Size as a Function of Voltage in Gas-filled, Two-electrode Chamber

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 <u>saturation</u> 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 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 Fig. 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

$$E = \frac{V}{r \ln (r_2/r_1)}, \qquad (10.1)$$

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 from 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 always 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 ΔV may be expressed as

$$\Delta V = \frac{Q}{C} = \frac{Ne}{C}\ (\text{Volts}), \tag{10.2}$$

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

$$\Delta V_p = \frac{AQ}{C} = \frac{ANe}{C}\ (\text{Volts}). \tag{10.3}$$

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 to 10^6 can be achieved.(4)

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 alpha and beta. Let us assume that the gas is air, and that an alpha and a beta each have about 3.5 MeV to expend in the chamber. If the alpha 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 amplication factor $A = 10^3$, the voltage pulse would be

$$\Delta V_p = \frac{ANe}{C} = \frac{10^3(10^5)(1.6 \times 10^{-19})}{16 \times 10^{-12}} = 1.0 \text{ Volt}.$$

For the same total path length, the beta produces much less ionization. In this case, the value should be around 100 ion pairs. The pulse size is then

$$\Delta V_p = \frac{ANe}{C} = \frac{10^3(10^2)(1.6 \times 10^{-19})}{16 \times 10^{-12}} = 1.0 \times 10^{-3} \text{ Volts}.$$

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 Fig. 10.2, the final pulse size from alpha as well as beta approaches the same value.

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.

Other gas-filled chamber devices used to detect radiation by the ionization method are the cloud chamber(15) and the spark chamber.(16,17) Since these are used mainly to detect particle tracks, they are not of interest for our purposes.

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 one mm or less.(18) 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,[(18,19)] 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.

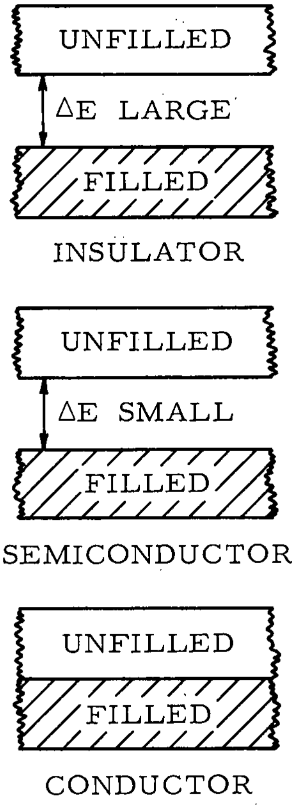


Fig. 10.3

Band Structure in Conductors, Semiconductors, and Insulators. (Adapted from Progress in Nuclear Physics, Vol. 9,[(18)] Copyright 1964, Pergamon Press, Ltd., used by permission.)

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.[(18)] 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 10 eV or so); in a semiconductor, the band gap may be of the order of 1 eV. Because of the small ΔE, 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.

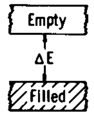
(1) Electron-Hole Pairs

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.

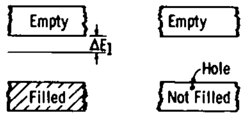
If a strong electric field is applied to the crystal, the electron in the conduction band moves under the action of the applied field. Also, an electron in the valence band moves into the vacancy left by the initial electron. In this sense, then, the hole also moves under the action of the field. That 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.(18)

(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 Fig. 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 Fig. 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 phosphorus, added to silicon reduces the band gap to only 0.045 eV,(18) in effect. Since conduction in this instance occurs by the movement of negative charges, the substance is called an n-type semiconductor.



(a) "Pure" Semiconductor



(b) n-Type (c) p-Type

Fig. 10.4

Effect of "Doping" with a Desired Carrier Type. (Adapted from Progress in Nuclear Physics, Vol. 9,(18) Copyright 1964, Pergamon Press, Ltd., used by permission.)

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 substance is referred to as an intrinsic semiconductor. In this case, the impurities compensate for each other. The charge carriers are then due to thermal excitation of electrons across the band gap.(4) Thus, a truly intrinsic semiconductor amounts to the concept of a pure semiconductor.

(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 effect is significant and requires very high purity and crystalline perfection to reduce its effects.(18) Based upon practical values for applied electric fields to the crystal, only silicon and germanium are suitable substances at present.

b. Conduction Counters

The first use of solid counters was the crystal or conduction counter.(21) 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.

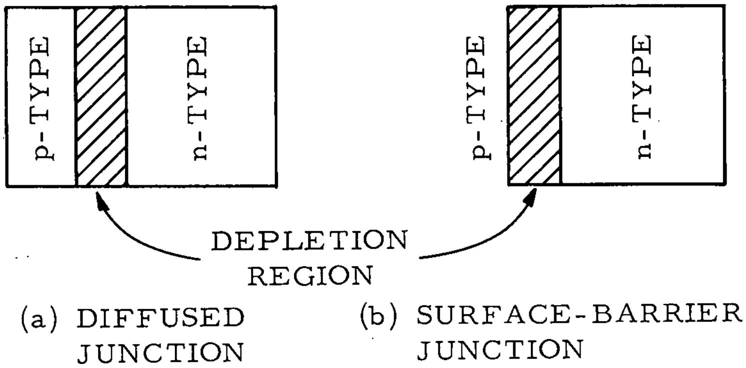
A charged particle entering the crystal loses energy by creating electron-hole pairs. The energy transferred is generally large compared to that needed to raise an electron to the conduction band. Thus, more pairs may result from the excess energy of these highly excited states. The presence of the field 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 semiconductor junction counters has caused interest in the crystal counter to wane.

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 have met with limited success. However, by means of the junction counter, we can obtain a region that is almost free of carriers. With proper design, this region may extend to as much as 1 cm 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 a 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 (Fig. 10.5a). This gives rise to the depletion layer, a space-charge region in which a potential difference of about 0.5 V exists. It is formed by the 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 depleted of 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 may extend over as much as 1-3 mm. This is about the practical limit for this type of unit.



(a) DIFFUSED JUNCTION (b) SURFACE-BARRIER JUNCTION

Fig. 10.5. Diffused and Surface-barrier p-n Junctions. (From Nuclear Radiation Detection by W. J. Price, Copyright 1964, Second Edition, McGraw-Hill Book Company, used by permission.)(4)

The surface-barrier type junction is depicted in Fig. 10.5b. This junction is prepared in most instances by spontaneous oxidation of the chemically etched surface of an n-type substance. In this case, the surface states that result, in effect, form a p-layer. Again, by use of the reverse-bias technique, this junction may be used to detect radiation.

One other type of junction counter that has been prepared is the p-i-n type. This offers the advantage of an increase in sensitive volume, since the sensitive region may extend up to 1 cm. This type counter has an intrinsic region between p and n surface layers. One device that features this design is the lithium-ion-drifted counter.(19) Lithium, a donor, is diffused into p-type silicon. 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, 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 counter.

C. Scintillation Devices

In recent years, the scintillation method of counting has reached a new level of importance. Lord Rutherford first used this method in his studies on alpha scattering. At that time, the scintillations were counted visually, and 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, 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 Fig. 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, organic liquid solutions, organic solid solutions, 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; and (4) the spectral distribution of the light should match the response features of commercial photocathodes.(4)

For equal energy, an alpha produces only about one-tenth the light output as a beta in organic crystals, organic liquid, and organic solid (plastic) phosphors.(25) 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. The present overall energy conversion is approximately 15-60 eV/light photon.

Organic and noble-gas counters have small decay times, τ.(7) The decay time for the organic phosphor trans-stilbene ($C_{14}H_{12}$) is about 5×10^{-9} sec. Inorganic phosphors such as sodium iodide ($\tau = 0.25$ μsec) 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 we 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 is contained in the literature.(4,9,22,24,25)

3. Detection of Pulses

Light flashes, or pulses, from the phosphor can be detected by means of a photomultiplier 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 about 0.5-2 keV/photoelectron is expended in most of these systems.

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 cesium antimonide (Cs_3Sb). 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 Fig. 10.6). The unit may contain as many as nine or ten 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 ten-stage tube in which the given voltage is 100 Volts, the total voltage above the cathode at the collecting plate would be 1000 Volts.

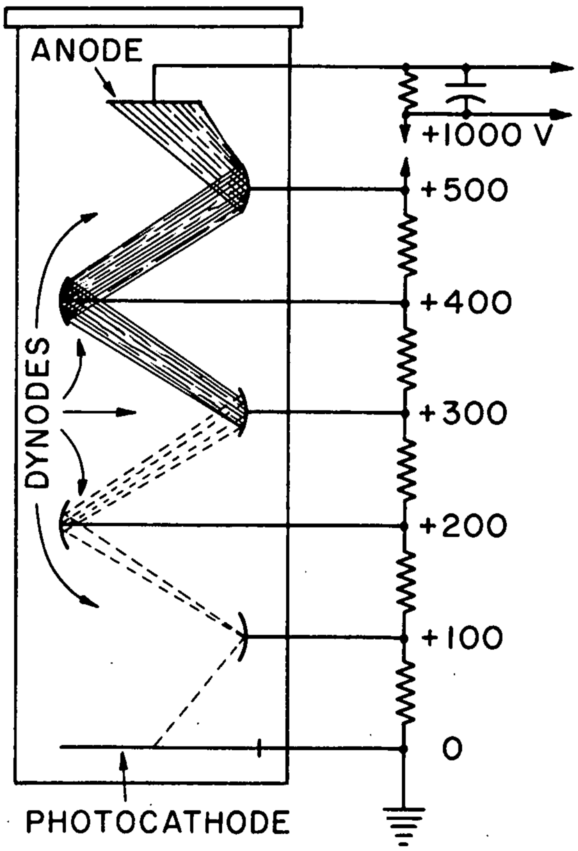


Fig. 10.6. Illustrating Photomultiplier Action. (From ORNL-2808/ORINS-30, Fig. 5, p. 7.)[26]

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 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 often in the range of three to five. The number of electrons freed for each incident electron is called the secondary-emission ratio, σ.[7] This ratio is also a function of the dynode surface substance, as well as the energy of electrons that bombard it.

At each stage, the input current is increased by the factor σ. The total multiplication in a system of n stages is then σ^n. In a ten-stage system in which $\sigma = 4$, the ratio of the output current to the primary photocurrent is greater than 10^6. That is, $\sigma^n = 4^{10} > 10^6$. In this way, much larger currents may result than would be obtained in ion-chamber devices.

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. The 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 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, we need a substance 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.(4,7,8,34) 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 separately in Section 13.

Chemical devices, both solid and liquid, have been used mainly to measure the absorbed dose. That is, the changes produced by the radiation can be simply related to the energy absorbed in these systems.(8) These products are more stable and somewhat easier to identify in liquid systems than in others.(7) 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. Systems with G values less than 20 are of most use in the kilorad range. We need a G value of about 500 to measure as low as 500 mrad.(7)

We may class these devices in two broad groups: (1) those using water as a solvent, and (2) other systems.(4) 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 routine use at present.(8) 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 10^3-10^5 rad.(8)

The ceric sulfate units are prepared by adding ceric sulfate to 0.1-0.8 sulfuric acid. The reduction of the ceric to cerrous ions may also be measured spectrophotometrically. This system works well in the range 10^5-10^7 rad.

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 color change may be measured in terms of dose by many means.(7) Reference 8 describes other systems and contains data on the response of these units to ^{60}Co gamma rays.

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.(37) 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(7,8,36) 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 Fig. 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 and is also called the band-gap energy. We may 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.

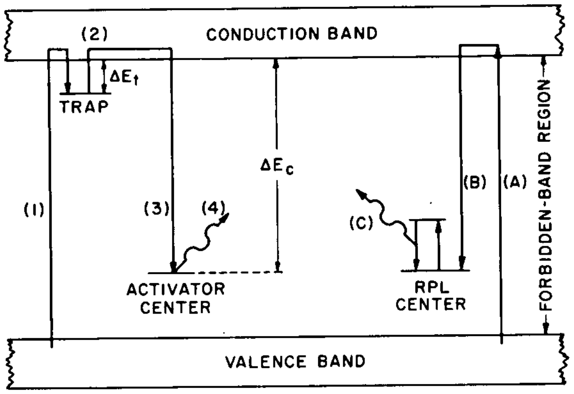


Fig. 10.7. Trap Effects in Solids Leading to Thermoluminescence and Radiophotoluminescence. (Adapted from Personnel Dosimetry Techniques for External Radiation (Symposium), Madrid, O.E.C.D./ENEA (1963).(36)

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 10^3 to 10^6 rad. Plastics have been used with success in the range 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 the unit is exposed to as well as the wavelength of the measuring light. This problem can be overcome in several ways;(37) 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.(37) 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 the inorganic color-change devices. The useful range is from 10^5 to 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 in Fig. 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. In recent years, attempts have been made to use radiophotoluminescence devices for personnel monitoring.(38,39) 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 in Fig. 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.(35) 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. 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. 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. When the flux density is constant, this flux density can also be found.(4,8,40) In this method, then, a stable substance is exposed to a neutron flux density for a known time span. The substance is then removed, and the induced activity is measured. If the flux density ϕ remains constant over the time t, then the activity A_t that results in the sample at the end of the exposure time is

$$A_t = \phi\sigma_{act}N\left(1 - e^{-0.693t/T_{1/2}}\right). \tag{10.4}$$

In Eq. 10.4, 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 flux density 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, $T_{1/2}$, 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.(4) All these choices depend on the specific problem.

2. Activation Detector Applications

Many of the present activation detector applications use the detector substance in the form of thin foils. 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. In this method, a foil is sandwiched between cadmium sheets. The thickness of the cadmium defines a cutoff energy. 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.

b. Resonance Measurements

In many 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 has been measured for many substances, and a table of these values is contained in Ref. 4. The method consists in exposing a cadmium-covered foil to the neutron fluence. The induced activity that results can then be related to the total fluence in the energy interval.(41) The thickness of the cadmium defines the lower end of this interval.

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. We can then select a substance in which no activation results until the neutron energy is above a certain value.(8) 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. In these instances, we may be able to replace the actual cross section and threshold energy by "effective" values. A table of these effective values for a number of threshold detectors is contained in Ref. 4.

The threshold detector can be used to obtain the total fluence above the threshold energy. By using a number of threshold devices, we can obtain the total fluence in the energy interval between any two threshold energies. With the proper choice of foils, we may then be able to estimate the neutron spectrum.

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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 mean-level detection system.(1) 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. Mean-level System

The mean-level 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.(1)

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 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 particles. Thus, the number of pulses at the output of the system is directly related to the number of particles. That is, the number of particles entering 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 lights on either scale-of-2 (binary) or scale-of-10 (decade) units. A mechanical register, which records a count every time the entire scaling circuit is lit up, may also be included. This 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 Fig. 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

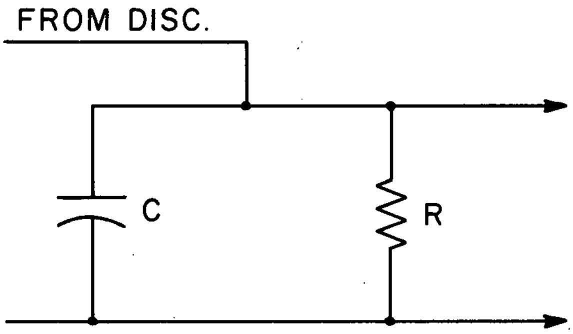


Fig. 11.1. Count-rate Meter Circuit

$$\text{leakage rate} = \frac{q'}{RC}, \qquad (11.2)$$

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, or

$$nq = \frac{q'}{RC} = \frac{V}{R}, \tag{11.3}$$

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 by 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.

D. Counter Plateau

The discriminator circuit in a pulse counter overcomes the effects of the noise in the system. However, 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 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 Fig. 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). If the voltage is increased still further, the counter may jam.

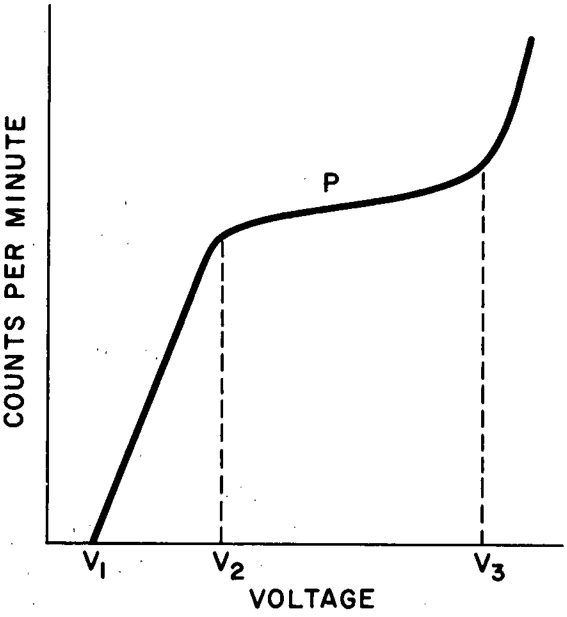


Fig. 11.2. Counter Plateau in Geiger Region

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 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 5% per 100 Volts in new tubes. 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 page 199), 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 our 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 alpha, beta, and gamma, the plateau region would indicate voltages at which all the alpha pulses pass the discriminator. An operating voltage in this region would allow us to count only the larger alpha 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 β-γ pulses become large enough to pass the discriminator. For further voltage increase, we would obtain a second plateau region. This region corresponds to voltages at which all pulses pass the discriminator. Because of the beta energy spectrum, the initial beta pulses may vary greatly in size. For this reason, the β-γ plateau is not as flat as the alpha plateau.

In the above system, we can obtain the alpha count and the total count. We then find the count due to β-γ pulses by subtracting the two counts. In some designs, the β-γ as well as the alpha count can be obtained separately. With the proper detector, fast neutrons alone can be counted in the presence of gamma 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 radioisotope is the source of the detected radiation, besides how much is there. To obtain this information often requires knowledge of the half-life and energy of the emission. We can find the half-life by following decay of the sample with the counter. This is one way of gaining more information from the count rate alone. We can determine the energy of the radiation by means of pulse-height analysis.

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.(2) The design of this is such that the size of the output pulse is closely proportional to that of the input pulse.

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 are collected at exactly the same time. Because of this, the voltage across the capacitor varies with time (see Fig. 11.3). Pulse is the term used to imply that this voltage varies with time. The presence of a resistance in the 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).

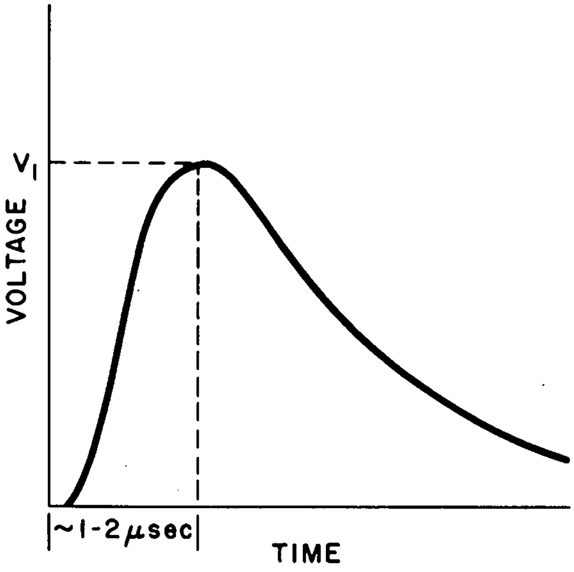


Fig. 11.3. Variation of Pulse Size with Time

These pulses are fed into the linear amplifier, where the gain is independent of the pulse height. Each pulse is then increased in size and moves on to the pulse-height analyzer.

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 reject those below this size. The pulse size that is accepted is controlled by a potentiometer. This is quite often a variable resistance used to bias a tube so that it does not conduct. The tube conducts if the incoming pulse height is large enough to overcome the bias, and the pulse is then counted. If the pulse size is not large enough to overcome the bias, no count results. In a pulse-height analyzer, this system is often called the window-level discriminator.

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 Fig. 11.4. As we lower the level setting from the high end (100), no pulses are recorded. This occurs when 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 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 distribution of pulse heights. That is, one pulse size occurs more frequently than others that are larger or smaller than this size. The 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."

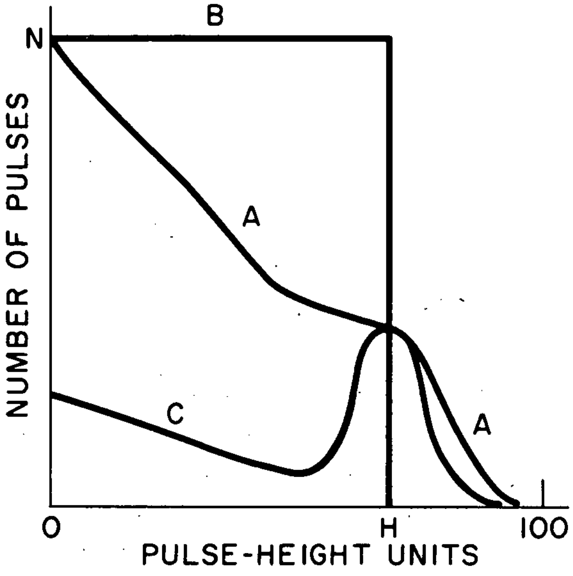


Fig. 11.4. Types of Pulse-height Spectrum

b. Window

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 window-width system. 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 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.(8) 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 as much as 10% of the window level. The choice of the width in a given case depends upon the spectrum information desired.

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 Fig. 11.4. 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 its 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 a 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 easily.

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 gamma spectroscopy. The subject is also discussed in the literature.(1-3)

F. Resolving Time

In the previous treatment of counting systems, it was noted that the pulse is not collected instantly (see Fig. 11.3). 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. 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.

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 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.

1. Resolving-time Correction

A knowledge of the resolving time of the system permits us to correct for counting losses at high rates.(6,7) Suppose a source has a true count rate N cps. Then, a system with a certain τ should give an actual count of n cps. The counts missed in unit time (M) would be given by

$$M = N - n. \qquad (11.4)$$

For each of the n cps that the system records, there is a time, τ, in which the system does not respond. 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

$$M = Nn\tau. \tag{11.5}$$

Thus,

$$N - n = Nn\tau, \tag{11.6}$$

and

$$N = \frac{n}{1 - n\tau}. \tag{11.7}$$

Now, suppose that τ is 200 μsec, and the observed counting rate n from a source is 60,000 cpm. By use of Eq. 11.7, the true count rate N can then be found as follows:

$$N = \frac{n}{1 - n\tau},$$

$$n = 6 \times 10^4 \text{ cpm},$$

$$\tau = 2 \times 10^{-4} \text{ sec} = 3.33 \times 10^{-6} \text{ min};$$

therefore

$$N = \frac{6 \times 10^4}{1 - (6 \times 10^4)(3.33 \times 10^{-6})} = \frac{6 \times 10^4}{1 - 0.2} = 7.5 \times 10^4 \text{ cpm}.$$

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, we can calculate the count rate at which losses become excessive.

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(1)

$$\tau = \frac{n_1 + n_2 - n_{12} - b}{n_{12}^2 - n_1^2 - n_2^2}. \qquad (11.8)$$

Now, assume that the above procedure has been carried out. Let the results be: b = 30 cpm, n_1 = 6000 cpm, n_{12} = 12,800 cpm, and n_2 = 7000 cpm. Solving for τ gives

$$\tau = \frac{6000 + 7000 - 12{,}800 - 30}{(12.8 \times 10^3)^2 - (6 \times 10^3)^2 - (7 \times 10^3)^2} = \frac{170}{78.84 \times 10^6} = 2.16 \times 10^{-6}\,\text{min},$$

or

$$\tau \cong 130\ \mu\text{sec}.$$

The above method of finding τ assumes that the products, $n_1\tau$, $n_2\tau$, and $n_{12}\tau$ are much less than 1.(1) In addition, the value of τ is taken to be independent of the count rate. In some cases, τ may decrease as the count rate increases.(7) For these reasons, it is often wise to use other methods to determine τ.(7,10) 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.

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 100 to 500 μsec.(6) 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.(9) 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.

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 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.(5)

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; alcohol, amyl acetate, xylene, and ethyl formate have been used. Counting tubes that contain these gases are called organic-quenched. This type of counter has a limited useful life of about 10^{10} 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 often has a short plateau with a large slope.(1)

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.(4) 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 radioisotope is not always available, we must have counters sensitive to alpha, beta, and gamma. 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.(10)

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 both the type of detector and the nature 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 1%.(5,9) 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 effect 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.(2) 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 β-γ 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.(2) Low background for alpha proportional counters is not as big a problem as for β-γ 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. Such an arrangement used with a standard shield reduces the background dramatically.

Coincidence circuits have also been used to reduce background for special counting applications.(2) 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.

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 alpha and beta 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.(7,10) For other cases, some of the information contained in Ref. 14 may be used. In many applications, only a relative value of the source strength is needed, and the value of these factors need not be known.

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,

$$G = \frac{\Omega}{4\pi}, \tag{11.9}$$

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 in the cone of solid angle $4\pi G$.

Assume that the detector has a circular aperture of radius r and that the point source, 0, is located on the axis a distance, a, from the counter (see Fig. 11.5). The geometry then depends upon the plane angle, θ. The larger this angle becomes, the larger the solid angle. The geometry for this setup may be written as

$$G = \frac{1}{2}\,(1 - \cos\,\theta) = \frac{1}{2}\left(1 - \frac{a}{\sqrt{a^2 + r^2}}\right) \tag{11.10}$$

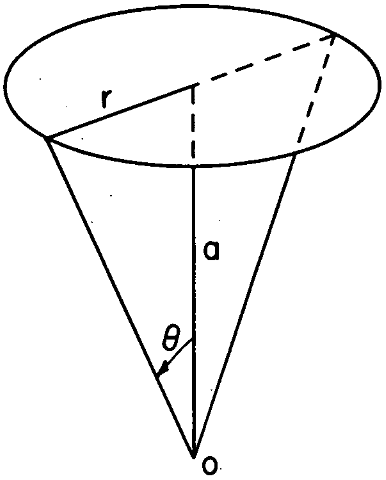


Fig. 11.5. Geometry for a Circular Aperture and a Point Source

Sometimes the point source is not located on the axis as shown in Fig. 11.5. 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 contains modified forms of Eq. 11.10 that allow us to calculate the geometry. In addition, a reference is quoted for tables of solid angles subtended by disks and cylinders.

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 of unknown activity A (dpm) with a beta 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 (cpm). Let us assume that the count rate is low enough so that no resolving time losses occur. We may then write

$$n = SA, \qquad (11.11)$$

where S is the absolute sensitivity and is found from

$$S = \epsilon_I G f_b f_w f_s. \qquad (11.12)$$

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

$$A = \frac{n}{S}. \qquad (11.13)$$

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.

6. Yield

The determination of the absolute sensitivity is not easy. For this reason, we use the relative sensitivity, which is found by means of 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

$$Y = \frac{n}{R} = \frac{\text{count rate (cpm)}}{\text{emission rate } (\beta/\text{min})}. \qquad (11.14)$$

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.

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 to be expected. However, the time to observe any finite number of counts varies and this gives rise to an error in an observed counting rate.(15) This error is inherent in the decay process and not a feature of the counting instruments.

1. Poisson Distribution and Gaussian Error Curve

For many counts taken on a small 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,

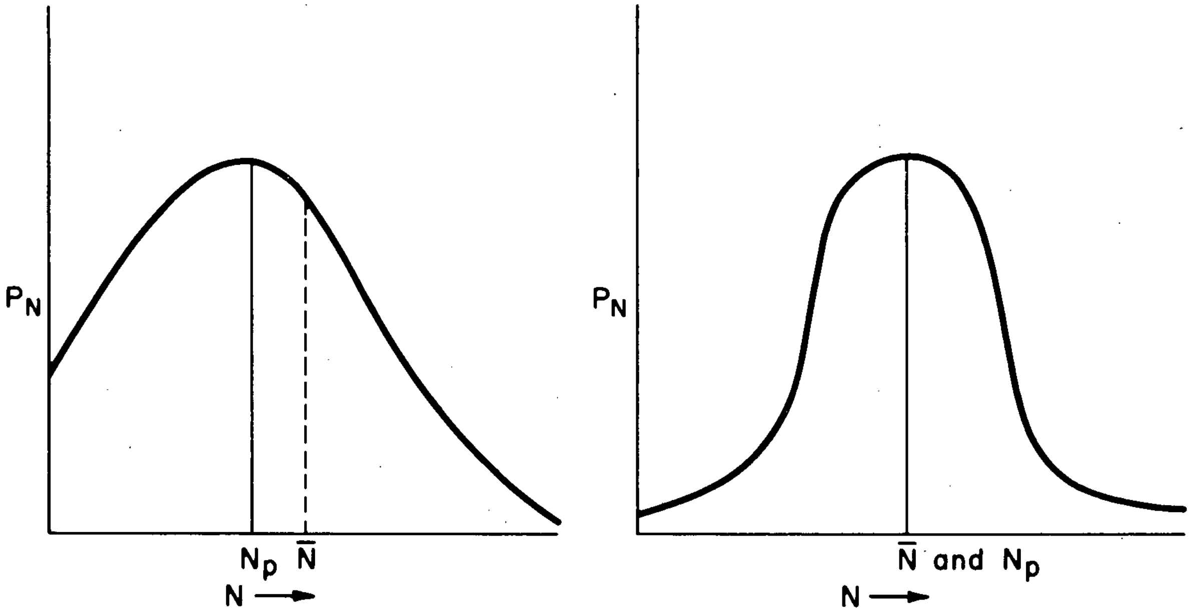
$$P_N = \frac{\overline{N}^N e^{-\overline{N}}}{N!}, \tag{11.15}$$

in which $\overline{N}$ is the average number of counts in the given interval. This gives a curve that is nonsymmetric with respect to the maximum value (see Fig. 11.6a). 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, $\overline{N}$.

If we obtain a large sample, we expect more decays to occur, on the average, in the same given time as before. In this case, $\overline{N}$ would be much larger. For this condition, the distribution approaches

$$P_N = \frac{1}{\sqrt{2\pi\bar{N}}} e^{-(N-\bar{N})^2/2\bar{N}} \qquad (11.16)$$

which gives a Gaussian error curve (Fig. 11.6b). This curve is more symmetric with respect to the maximum so that the most probable value, 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 samples, the variations in the counts become smaller. Therefore, for a given count time, we get a more precise count for the larger samples.



(a) Poisson Distribution (b) Gaussian Error Curve or Normal Distribution

Fig. 11.6. Probability Distributions for Low and High Values of N

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 (Part E above), 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.(15) If an average $\overline{N}$ counts are observed in a given time, then the standard deviation of the count is

$$\sigma = \sqrt{\overline{N}}. \tag{11.17}$$

The standard deviation expresses the range of values about the average count in which about two-thirds of all similar counts should fall. Since one-third of the counts are outside this range, we have a measure of the reproducibility of the type of count taken. If the source is large enough, a single count, N, can be used to calculate σ. Suppose we count a source for 20 min and obtain 20,000 counts; then,

$$\sigma = \sqrt{N} = \sqrt{2 \times 10^4} = 141 \text{ counts.}$$

We would then express the result as 20,000 ± 141 counts for this source counted for 20 min. Now, suppose we make a large number of counts on this source, all for 20 min. 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.

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

$$\text{Error} = K\sigma = K\sqrt{N}. \tag{11.18}$$

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 in 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, we should always specify which error is being used. For example, a sample is counted for 10 min, 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;

$$\text{Error} = K\sigma = K\sqrt{N};$$

$$9/10 \text{ Error} = 1.6449\sqrt{10^4} = 164.49 \text{ counts}.$$

Our result is then 10,000 ± 164 counts at the 90% confidence level.

4. Standard Deviation of a Count Rate

In general, our interest is in the standard error in the count rate, n. In this case,

$$n = \frac{N}{t},$$

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

$$\sigma_n = \sqrt{\frac{n}{t}}. \tag{11.19}$$

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 cpm 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

$$\sigma = \sqrt{\sigma_1^2 + \sigma_2^2}, \qquad (11.20)$$

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

$$\sigma = \sqrt{N_1 + N_2}. \qquad (11.21)$$

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 min. The source count turns out to be 6000 counts for 20 min, so that

$$\sigma = \sqrt{N_1 + N_2} = \sqrt{6000 + 300} = 79.4 \text{ counts}.$$

Once again, in terms of the count rate, the standard deviation is

$$\sigma_n = \sqrt{\frac{n_1}{t_1} + \frac{n_2}{t_2}} \qquad (11.22)$$

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 total count or count rate:

$$\sigma_r = \frac{\sqrt{N_1 + N_2}}{N_1 - N_2} = \frac{\sqrt{n_1 t_1 + n_2 t_2}}{n_1 t_1 - n_2 t_2}. \tag{11.23}$$

By using the K values in Table 11.1, we can extend the concept to include the other errors. That is,

$$\text{Relative Error} = K\sigma_r. \tag{11.24}$$

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(12)

$$\frac{t_1}{t_2} = \sqrt{\frac{n_1}{n_2}}. \tag{11.25}$$

We can illustrate the point with an example. Let a source give a counting rate of about 260 cpm in an area where the background is 92 cpm. The question is: How long must we count the sample to obtain a relative probable error (rpe) of 4%?

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

$$\frac{t_1}{t_2} = \sqrt{\frac{n_1}{n_2}} = \sqrt{\frac{260}{92}} = 1.663.$$

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 rpe of a count rate

$$\text{rpe} = 0.6745\sigma_r = \frac{0.6745\sqrt{\dfrac{n_1}{t_1} + \dfrac{n_2}{t_2}}}{n_1 - n_2}.$$

Suppose we choose to substitute for t_2, that is, $t_2 = t_1/1.663$, so that

$$0.04 = \frac{0.6745\sqrt{\dfrac{260}{t_1} + \dfrac{92(1.663)}{t_1}}}{260 - 92}$$

$$= \frac{0.6745\sqrt{\dfrac{413}{t_1}}}{168}$$

then,

$$9.963 = \sqrt{\frac{413}{t_1}}$$

$$(9.963)^2 = \frac{413}{t_1}$$

$$t_1 = \frac{413}{99.261} = 4.16 \text{ min},$$

and

$$t_2 = \frac{t_1}{1.663} = \frac{4.16}{1.663} = 2.50 \text{ min},$$

so that

$$t = t_1 + t_2 = 6.66 \text{ min}.$$

This is the time we must count the sample to obtain a relative probable error of 4%. 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

$$0.04 = \frac{0.6745\sqrt{\frac{184}{t_1} + \frac{92(1.414)}{t_1}}}{184 - 92}$$

$$= \frac{0.6745\sqrt{\frac{314}{t_1}}}{92},$$

$$(5.456)^2 = \frac{314}{t_1}$$

$$t_1 = \frac{314}{29.77} = 10.55 \text{ min},$$

$$t_2 = \frac{t_1}{1.414} = \frac{10.55}{1.414} = 7.46 \text{ min},$$

and

$$t = t_1 + t_2 = 18.01 \text{ min}.$$

In this case, the total counting time is about $2\frac{3}{4}$ 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 61 min to obtain the same rpe! Reference 14 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.

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 would be "too consistent." A means of checking counter performance is provided by the chi-square test of Pearson.(16) The essentials of this method are outlined in Ref. 14.

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 <u>smallest</u> value of

$$\frac{\sqrt{n_2}}{Y}, \qquad (11.26)$$

As before, n_2 is the background counting rate, and Y is the yield of the counter.

Many other topics concerning the statistics of nuclear detection are contained in Chapter 3 of Ref. 1.

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SECTION 12 - HEALTH PHYSICS INSTRUMENTS

Up to now, much has been said in regard to how we detect radiation. In many instances, all that is required is a counter that records the events occuring 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 Ref. 1, which contains a survey 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.(1-3) 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

$$E_m = \left(\frac{S}{\rho}\right)_m WJ_g, \tag{12.1}$$

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 $(S/\rho)_m = 1$,

$$E_m = WJ_g, \tag{12.1a}$$

and the cavity need not be so small. This simple expression will be valid if a number of conditions(2,3) 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,(4) the "walls" of the chamber are really air. This device has been used as a primary standard to measure the unit of exposure, the Roentgen. In most devices used for survey work, though, we cannot have true walls of air. Likewise, "tissue" walls are also difficult to obtain. Thus, for a chamber designed to measure rads in some medium, we must attempt to match the wall and gas to this 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.(3) 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 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 Ref. 6.

d. Charged-particle Equilibrium

For a beam of photons or neutrons that strikes a chamber wall, 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 Fig. 12.1). The curve goes through a broad maximum, reaches a peak value at thickness t', and then slowly decreases with further increase in t.

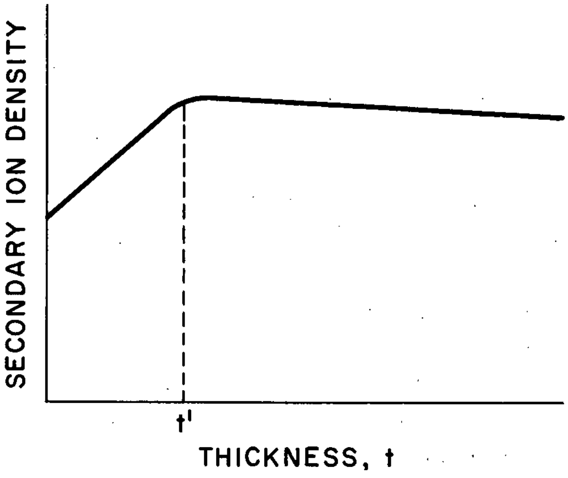


Fig. 12.1. Equilibrium Thickness, t', for Charged-particle Equilibrium (adapted from Ref. 1)

The thickness, t', is called the equilibrium thickness and is equal to the maximum range of the secondaries produced. At this thickness, a state of charged particle equilibrium (CPE) exists. That is, for every secondary that leaves a small volume about a point at this thickness, one with about equal energy would enter.(1) 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.(2) 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.(1)

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.(1)

e. Kerma

A report(7) by the ICRU defined the term kerma (K). This concept can be used to replace the old term "first-collision dose" used for photons and neutrons.(2) The kerma is the energy per unit mass transferred to charged particles by the uncharged particles passing through a substance. For a small medium in air, the kerma for the medium replaces what once was called the "first-collision dose." When CPE exists at a point in the substance, the kerma is equal to the absorbed dose. This assumes that bremsstrahlung losses are small. For transient equilibrium cases, the kerma is just slightly less than the absorbed dose.(7) The absorbed dose may be obtained by calculating the kerma(1) 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 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, causes the dose to depend upon the size and shape of the object.(8) For a nondirectional field (not a beam), the true absorbed dose in a large tissue mass may even be less than the kerma.(9)

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.(2,9) This factor may be much greater for lower-energy neutrons and high-energy photons.(8-10)

An air-equivalent ion chamber, designed to measure the exposure in Roentgens, 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.(7) The absorbed dose is then

$$D_{air} = 0.87X, \tag{12.2}$$

where X is the exposure in Roentgens.

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.(11,12)

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.(1)

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.(13) 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.(2,14) 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.(1,2)

A useful gamma dosimeter, which makes use of a Geiger-Müller counter, has been reported.(15) This device is shielded with tin and lead and can be used to measure only gamma in a mixed field.

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.(1)

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 in a state of development. Certain aspects of these devices appear to offer great promise for their future widespread use.(13) 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.(1) 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.

The features of many of these devices are summarized in Table 7 of Ref. 2.

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 be 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.(1-3,8) These doses are then weighted by the proper quality factor and summed to obtain the dose equivalent.(16,17) Second, we can measure the total absorbed dose with a tissue-equivalent chamber. Then, if the dose delivered at each level of LET (linear energy transfer) is measured by some other method,(9) the dose equivalent can be found. Third, we can measure the flux density and energy spectra of the components of a mixed field, then we can compute the absorbed dose.(8) 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.(18) Hurst and Ritchie(19) describe a device using threshold detectors, which can be used to obtain an estimate of the neutron absorbed dose.

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(7) 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(20) 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 QF 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 QF be related to LET_∞, the stopping power in water.(17) The stopping power gives the energy loss per unit distance regardless of where this energy is absorbed.(7) A table of the LET-QF relationship, taken from Ref. 17, is shown in Section 17 (Table 17.1).

b. Distribution in LET of Dose

The dose equivalent may be found then by measuring the absorbed dose at each LET, multiplied by the QF for that value of LET, and summing these products. The distribution in LET of dose 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.(21) 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.

c. Summary

In its report,(17) the ICRP favors the use of the dose-equivalent concept to express its recommendations. For this reason, much effort is being devoted to the problem 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.(22-27) 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.(24) 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 survey instruments have been developed and used in the past 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 some of the many useful instruments are discussed here. More about the features of many of these devices is contained in Refs. 22, 25, 26, and 27.

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 may 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 (see page 216). 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.(23) This device measured the total charge collected (see Fig. 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 movable 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.

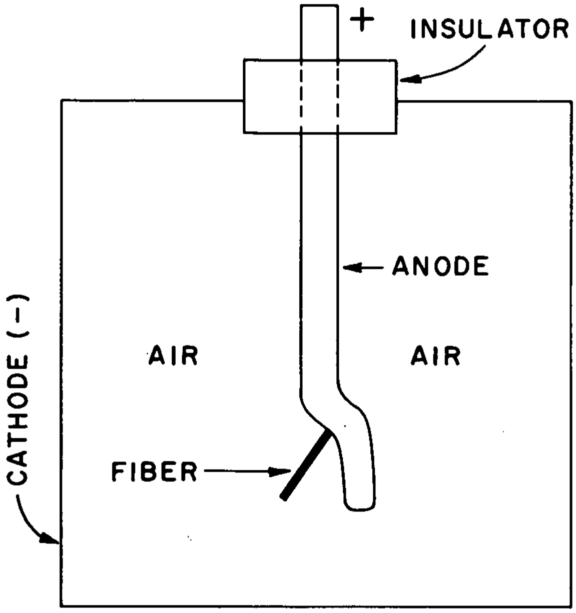


Fig. 12.2. Quartz-fiber Type Electroscope

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.

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.(3) 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.(5) 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.

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 enable us to cover exposure ranges from 0 to 2500 R.(22) These devices can achieve a reading accurate to within 3% in the design range.

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 has been developed by Rossi and Failla.(11) Further studies in this field have lead to other plastics, whose response to radiation is close to that of tissue.(29) The composition of tissue may be taken as $(C_5H_{40}O_{18}N)_n$.(2) It has been found(11) that a mixture of methane, carbon dioxide, and nitrogen in the proper proportions produces a TE gas. 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.

The chamber size and shape may vary with the purpose of the measurement. The TE plastic walls of these chambers are usually about $\frac{1}{4}$ in. thick. This thickness precludes the entry of low-energy charged particles, but is necessary for reliable chamber function.(11) 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 8 in. have been used for this purpose.(30) Small-volume devices (thimble chambers) are used to measure very high doses.(11)

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.(11) The sensitivity can be increased if the chamber is combined with a vibrating-reed electrometer.(30) 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 of 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 only ~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, Fig. 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 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 value of the resistance in the chamber circuit may be varied by means of a switching arrangement, which allows the choice of any one of three or four resistors. This allows the device to be used over a number of ranges of intensity. The higher-value resistors give the greater sensitivities (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.(5) 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.(5)

Ion-chamber troubles most often arise from failure of the insulation.(5) 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.

a. Juno Survey Meter

The Juno survey meter is a portable instrument used to measure the exposure rate of photons. It can also be used to distinguish qualitatively between alpha, beta, and photons. It consists of an ion chamber, which contains sliding filters to allow rejection of either alpha or beta or both. The case material is aluminum, and the central electrode is generally aluminum, coated with graphite. A Teflon bushing insulates the central electrode from the rest of the case. The gas in the chamber is air at atmospheric pressure. The chamber volume is large, on the order of 30 cu in.

The window, usually nylon, rubber hydrochloride, or aluminized Mylar about 1 mg/cm^2 thick, forms one wall of the chamber. The inside of this window is coated with graphite to make a good conducting surface. Since the windows are fragile, care must be exercised to be sure that they remain intact. A broken window disrupts the electric field and renders the instrument useless. The window, of course, imposes a directional restriction on the use of the survey meter to detect alpha and beta.

With all shields open, the device responds to alpha, beta, and gamma. A shield of cellulose acetate about 0.01 in. thick slides in front of the window to reject alpha. The response is then due only to beta and gamma. An aluminum filter of about 3/32-in. thickness is then used to cut out the beta response.

With the shields closed, the Juno measures the photon exposure rate to within 10%.(22) The aluminum wall and electrode with their graphite coatings are air-equivalent substances. For this reason, the meter scale is calibrated in exposure rate (mR/hr). The chamber gives correct readings for photons in the energy range 0.3 to 3 MeV. Below 0.3 MeV, the chamber is energy-dependent, and readings will overestimate the exposure rate.

For most monitoring purposes, the energy dependence of this device is the most serious source of error.(3) Below 50 keV, a much thinner chamber is required.(28) From about 50 to 250 keV, the overestimate results in a safe reading. The instrument is normally supplied with full-scale ranges between 50 and 25,000 mR/hr. Special models have also been designed for higher ranges.

The Juno is designed to measure the exposure rate and is calibrated in a uniform photon field. For this reason, do not attempt to measure alpha or beta dose rates with this device. The device can be calibrated with a number of known alpha sources in order to measure alpha activity quantitatively. The meter reading would then only be used as a relative current scale. A chart could then be prepared to give the source activity (dpm) in terms of the scale reading.

b. Samson Survey Meter

The Samson survey meter is an air-ionization chamber, which can be used to detect alpha, beta, and gamma. The chamber responds to these, but the device is only calibrated for alpha. No sliding shields are supplied so that we must use an external filter to see if other than alpha is present. This may be as simple as a sheet of paper. In this way, we can obtain a qualitative reading of the β-γ field. The instrument can also be equipped with an external probe for β-γ readings.

The Samson uses a balanced circuit with an electrometer tube and a high-gain amplifier.(22) Again, a zero-set control is used to balance the circuit. The chamber is large, as in the Juno, and the chamber window is protected by a stainless-steel screen. The screen is perforated to allow about 80% open area. Again, the window acts as one of the chamber walls and is usually rubber hydrochloride or nylon. The inside is coated with Aquadag (graphite), which alphas of >2 MeV can penetrate.

c. Jordan Radector

The Jordan Radector has a large, probe-type ion chamber designed to accurately measure a very wide range of exposure rates. The chamber wall is steel, and the gas in the chamber is argon at 10 atm. The electrometer tube is placed within the chamber, which is hermetically sealed. The chamber can be used for photons and high-energy beta. The instrument has a logarithmic response meter, which allows us to read exposure rates from 0.5 mR/hr to 500 R/hr. The probe is often supplied with a long cable, and the device can be used remotely. Since the window for beta is on the end of the probe, the response is highly directional.

The device is calibrated for gamma response in the photon energy range of 80 keV to 1.2 MeV. In this range, the Radector has a relative accuracy of ±15%. The instrument responds to beta with $E > 150$ keV. The reading for beta should be taken only as a relative reading. The approximation is fair for beta with energy between 150 keV and 1 MeV.

Features of some of the other ion-chamber devices, such as the Cutie Pie, Zeus, and Victoreen Roentgen Rate Meter, are described in Refs. 22-27. Reference 31 contains a discussion of the general aspects of survey devices.

3. Geiger-Müller Survey Meters

The Geiger-Müller survey meter employs a G-M tube and count-rate meter to obtain a simple and sensitive detecting device. The tube is contained in a sliding shield, and the entire unit is the probe. A cable connects the probe to the instrument case.

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.(32) The anode is an axial wire (tungsten) of about 0.02 to 0.1 mm in diameter.(33) These tubes are usually operated at reduced gas pressure of about 7 to 20 cm Hg.(23) The gas-fill varies, but some type of self-quenching gas (see Section 11, par. 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. However, some counters "saturate" in a very high radiation field. 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.

The G-M survey meter may be used to detect beta and gamma. The tube thickness is about 30 mg/cm^2. This is much too thick to allow alpha penetration, but the device detects beta of $E > 0.2$ MeV. The sliding shield (~1500 mg/cm^2) rejects all beta from normal sources. For low-energy beta, an end-window type G-M survey meter can be used. In an end-window tube, one end of the cylinder has a very thin covering, which can be used as a window. Mica, a few mg/cm^2 thick, is quite often used as the window substance. These devices respond to beta of $E >$ ~30 keV(22) and alpha of $E >$ ~3 MeV.(24) Since the window is very fragile, care must be taken in using these devices.

End-window counters are used extensively to detect ^{14}C, which gives off a low-energy beta. We must be always 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.

b. Energy Dependence

The G-M survey meter is not a precise instrument for exposure-rate measurements. The response of this device is not directly proportional to the energy absorbed in the sensitive volume.(33) The energy absorbed per unit mass in any medium is a function of the photon energy fluence (F) and the mass energy-absorption coefficient (μ_{en}/ρ).(7) This is true if the secondaries do not have too high an energy. The count rate for a G-M survey meter depends upon the efficiency of the counter (counts per incident photon). This, in turn, is a function of the cathode material. 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. To overcome this energy dependence, the cathode material must have a response nearly proportional to the photon energy.(33) 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 0.3-3 MeV.(33,34) Below 0.3 MeV, the response of any cathode material is highly energy-dependent, and none of the counters are suitable for exposure-rate readings.

Wagner and Hurst(15) 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.

c. Application

At present, the G-M survey meter is best used as a device to detect radiation. We cannot measure exposure rates below RPG levels very well with ion-chamber instruments. For this reason, the G-M counter serves as a useful, low-level survey device. Such counters are available from several commercial companies. Usually, these devices are designed to cover the range 0-20 mR/hr, although higher ranges are available. For the glass-wall type tubes, it is good practice to coat the tube with an opaque spray. This helps avoid spurious readings due to light leakage into the tube.

For aid in detection, these devices also provide for the use of headset earphones. 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.(33) The counters are operated at atmospheric pressure or slightly above. In many cases, the gas is allowed to flow through the chamber.(23) Methane is a popular gas 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 or gas is 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.

Troubles sometimes arise 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 ionization. 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.(27) Aluminized Mylar 0.85 mg/cm^2 thick, forms the chamber window, which has an area of 61 cm^2. The probe is connected by cable to the small, lightweight ($\sim 4\frac{1}{2}$ lb) case that houses the count-rate meter and voltage supply. The device uses transistors, rather than tubes, in the circuits. A jack enables earphones to be used.

The meter has three scale ranges, allowing measurements up to 10^5 cpm based upon 50% geometry. The device measures only alpha, even in the presence of β-γ. Both the discriminator and high-voltage circuits can be adjusted to allow detection of low-energy beta.

The use of air as the counting gas causes the probe to be sensitive to pressure and humidity, which may subject the scale reading to much fluctuation. The device should be allowed to warm up at least 2 minutes before use.

b. Gas Proportional Alpha Counter

The Eberline Gas PAC employs the same circuits as the Eberline PAC. In the Gas PAC, the chamber gas is propane, and the device is used as a gas-flow counter. In this way, the humidity and pressure effects can be overcome, and the probe gives a higher counting efficiency. Since propane contains hydrogen, the Gas PAC also responds to a 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 9 lb. The gas bottle holds about a 24-hr 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.

c. BF₃ Proportional Neutron Counter

Boron-lined or BF_3 gas proportional counters can be used to measure thermal-neutron flux densities. These instruments can be designed to be insensitive to high gamma fields (~10^4 photons/n).(18) 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 flux density. 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.(23) These devices are often less efficient than the BF_3 counters because of absorption of alpha and lithium in the counter wall.

These 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 cpm/unit flux density when the device is calibrated in a known thermal ϕ.

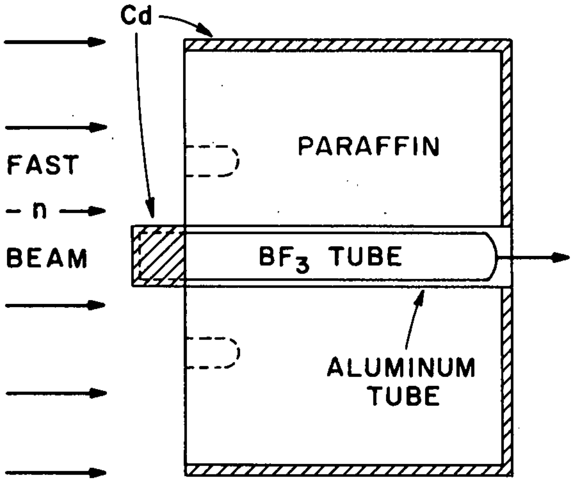


Fig. 12.3. Typical Setup for Long-counter Application

d. Long Counter

BF_3 detectors have been used not only to measure thermal flux density 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,(36) is referred to as a "long counter." In this design (see Fig. 12.3), 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 neutron energy should affect only the depth in the counter at which they are detected.(18) 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 1 ft long and 8 in. 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 Fig. 12.3) 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.

In recent years, DePangher has modified the design features of this device to improve the performance. Some details about this device are contained in Ref. 1. The basic principle of the long counter has also been used in a smaller, more portable device, which is commercially available.(27)

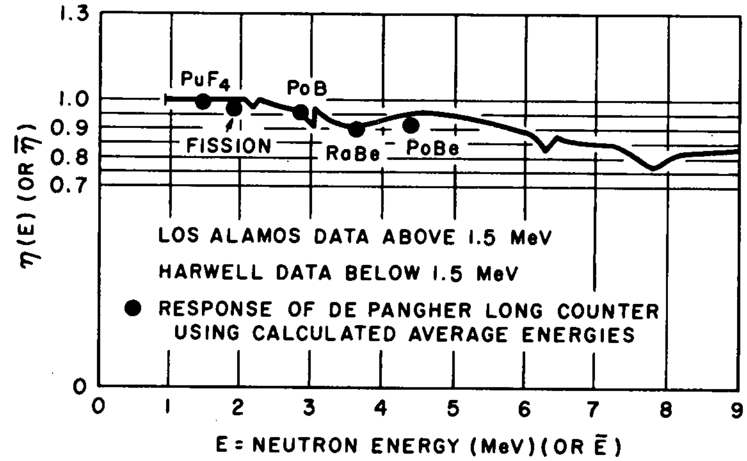


Fig. 12.4. Long-counter Response (from NBS Handbook 85(1))

The long counter derives its name from the fact that the response is relatively flat over a wide neutron-energy range (see Fig. 12.4). 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 flux-density measurements in neutron beams of wide spectrum variations.

The long counter must be calibrated for both fast and thermal flux densities. 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.(18) The response is given as cpm/unit flux density 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 flux-density readings obtained.(18) Although designed to measure only neutrons, the device will respond to high gamma fields (~5-10 R/hr).

The long counter measures only the flux density and does so without regard to neutron energy. For this reason, we can not 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 flux density values in NBS Handbook 63(10) may be used to estimate the dose equivalent. 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(2,8,24) or dose equivalent.(38) The device mentioned in Ref. 38 follows the NCRP dose-equivalent curve within about 25% for neutron energy between 0.15 and 5 MeV. This instrument is a tissue-equivalent proportional counter with a transistorized count rate meter circuit. More details about the features of this device are contained in Ref. 22.

5. Scintillation Survey Meters

Scintillation counters are used extensively in many phases of radiation measurements.(23,39,40) Many devices have been developed to fill a special need in regard to a certain study. 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 report (see p. 240).

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(1,41) can be used for neutrons, depending upon the energy spectrum of the field.(42)

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 usually consists of an amplifier, a count-rate meter, and a battery supply.

For alpha and gamma devices, the instrument is best used simply to detect radiation. The response of these counters to gamma is such that readings are limited to <5 mR/hr. The energy dependence for these units precludes their use as exposure-rate meters.(22) Because of the low range, gamma scintillation counters are not used too extensively for survey work. For neutrons, the units can be designed to give an approximate dose-equivalent rate.

b. 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 1.5 mg/cm^2.(27) The active area of these instruments tends to be nonuniform in response near the edges. This may be due to poor light transfer by the light pipe. Also, we must protect the lighttight screen from damage, since even a pinhole admits enough light to give an erroneous reading.

The meter gives a reading in cpm and may provide a choice of three or four ranges. Some of these devices tend to saturate in a high alpha field.(27) This occurs when the photomultiplier can no longer resolve the pulses. The instrument may include an earphone jack to provide an aural indication.

c. Neutron Scintillation Counter

Many substances have been employed to measure neutron fields with scintillation counters.(42) 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 hydrogenous substance and detects the proton recoils;(2) 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.(43)

One example of this type of device grew out of studies directed toward neutron-spectrum measurements.(44) The phosphor was a small ^{6}LiI (Eu) crystal, which makes use of the ^{6}Li (n,α) T reaction. This process has a high cross section for thermal neutrons (~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 interactions occur, some of the fast neutrons lose a large part of their energy and become thermal neutrons. For a large enough moderator, a certain thermal-neutron distribution results. That is, the thermal flux density increases from an initial value at the front surface to a maximum value, and then decreases with depth to the rear surface.(45) The depth at which this maximum occurs is a function of the neutron energy. If we now place the phosphor in a moderator of a certain size, the response varies with the energy of the incoming neutrons.

In the original study, the crystal was mounted at the center of several cadmium-covered polyethylene spheres. 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 2 to 12 in. The response of a given-diameter sphere was studied as the neutron energy was varied. The response curve for the 12-in.-diam sphere had about the same shape as the dose-equivalent curve (see Fig. 12.6 later). This aspect was investigated for a 10-in.-diam sphere. The response of this device approximates the dose-equivalent curve in the energy range from thermal to 7 MeV.(46) In this device, the scaler was replaced by a count-rate meter.

Commercial versions of this system are now available from Texas Nuclear Corporation. Their device uses the 10-in. polyethylene sphere and can be obtained with a rate meter and a decade scaler. The gamma rejection is good, but the device is rather heavy. A carrier is also available so that the instrument can be made semiportable.

The basic scheme used in this method has been adapted to other designs. Reference 47 describes a similar device using a paraffin sphere and a boron-loaded scintillator. This unit is more suited to the measurement of intermediate neutrons.

The use of spherical geometry in these devices tends to make the response nondirectional. In this way, we can overcome a difficulty that hinders long-counter uses.

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 flux density. 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.(2,18) 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.(48)

a. High-energy Neutron Monitor

The threshold-detector technique has been used in a device to monitor high-energy neutrons and protons.(49) In this method, a cylinder of Pilot-B plastic scintillator of 5-in. diam and 5-in. 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 min. 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 may include a single-channel analyzer to provide optimum signal-to-background ratio.(30) 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.

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 Fig. 12.5. The diameter of the sphere is 12 in. and a symmetrical array of three sensors is placed at the center. Six other sensors (2 in. in diam) are located on the rectangular coordinate axes at the same distance from the center. In this design, the foils are 1 in. below the moderator surface and 5 in. from the sphere center. The actual detector is composed of two 12-in.-diam, 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.

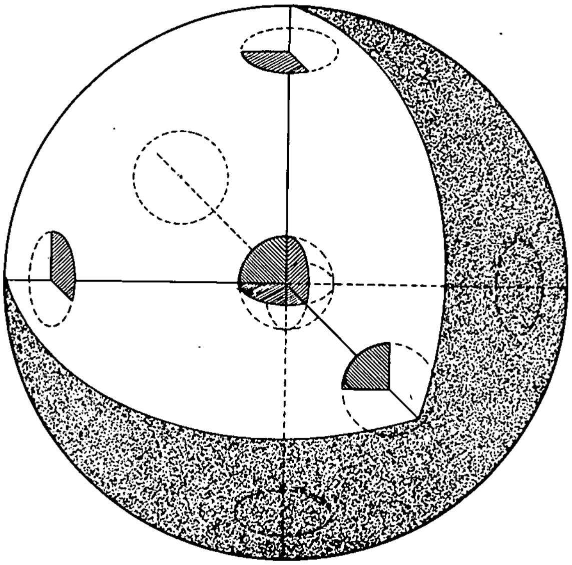


Fig. 12.5. Sensor Geometry for the Spherical Neutron Foil Monitor (from ANL-7085(45)).

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 on page 233. In the foil monitor, the foil array replaces the ^{6}LiI (Eu) scintillator as the thermal detector. The response of the ANL device is compared to the dose-equivalent rate curve in Fig. 12.6. Also shown are the response curves for the other spherical detectors of Bramblett et al.(44) and Hankins.(46)

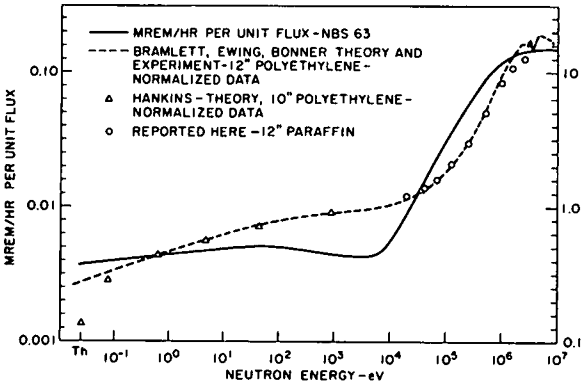


Fig. 12.6 Response Curves for Three Spherical Neutron Detectors Compared to the Dose-Equivalent Rate Function (from ANL-7085(45)).

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.

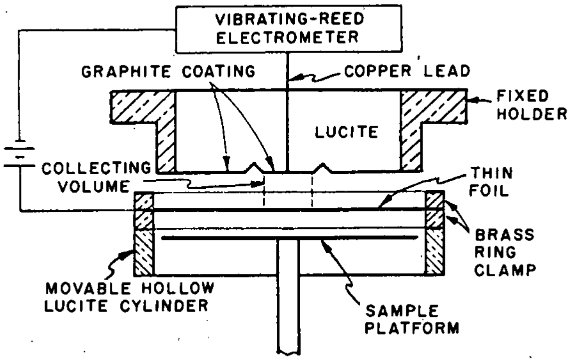
This device is used to obtain a simultaneous measure of the fluence, dose-equivalent, and effective neutron energy. The unit may be used to measure fields as low as a few mRem/hr. However, since the foils must be removed and counted, the results of a survey are delayed. The technique has been used in remote monitors by using small, silver-wrapped G-M counters in place of the foils.(30)

C. Special-purpose Instruments

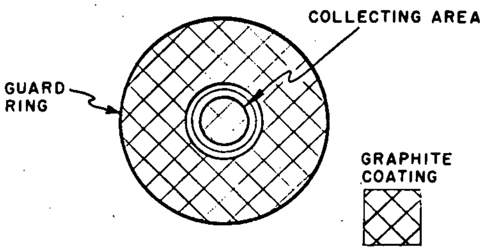
Along with survey instruments, some special devices are also useful in a radiation-safety program. These units provide a complement to the survey devices and in some cases may supply information that survey instruments cannot supply. We will treat only a few of these special-purpose units, which are of use at the present time.

1. Extrapolation Chamber

The extrapolation chamber was devised by Failla(50) 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 Fig. 12.7). A thin foil (doubly aluminized Mylar) serves as the accelerating electrode of the chamber. This foil is clamped between two flat 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 Fig. 12.7b). A source is placed on a platform (see Fig. 12.7a) 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 Fig. 12.7a) as a function of the plate spacing. The currents are very small (~10^{-15} Amp) so that the current output is read on a vibrating-reed electrometer.



(a) SIDE VIEW



(b) TOP ELECTRODE VIEW

Fig. 12.7. Extrapolation Chamber (from ANL Model)

The current may then be plotted as a function of the plate spacing (see Fig. 12.8). 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 Fig. 12.8 is proportional to the absorbed dose rate. The value of the proportionality constant depends upon the design features of the device.

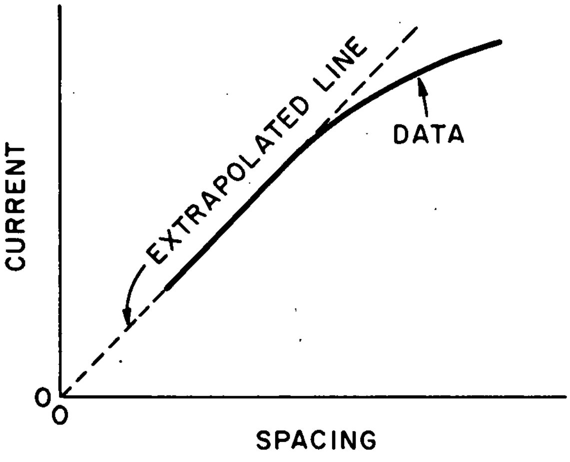


Fig. 12.8. Plot of Extrapolation-chamber Data

The chamber has been most useful for measuring surface dose rates of alpha and beta. For beta, Lucite provides a good tissue-like scattering pattern.(51) 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. Beta can be measured in the presence of alpha by using a thin cover substance to shield out the alpha. We can even obtain the beta dose rate through other shield substances, such as rubber gloves.(52)

2. Spectrometry Units

In most cases, we can identify a radioisotope 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. 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). 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 device, we must adjust the channel level over the range of pulse sizes to record the distribution. This is accomplished with a "sweep" system, which varies the window level at a given rate. On the other hand, in a multichannel analyzer, we can record counts simultaneously in all the channels. In this way, we can obtain the spectrum much faster. In each case, a scaler, a count-rate meter, and a display device are usually included.

For low counting rates, the statistics of a single-channel unit are poor unless we use very long counting times.(24) For this reason, many units presently use multichannel units, which are now available in compact devices using transistorized circuits.

The spectrometer may then 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. The extent of the information we can obtain from these units depends upon the design features in the device.

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. LET Spectrometer

The LET spectrometer employs a 2- or 8-in.-diam tissue-equivalent chamber filled with tissue-equivalent gas. The output is fed to a 400-channel pulse-height analyzer. This device is based upon the unit described by Rossi and Rosenzweig.(53) 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,(21) 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 QF > 1 (mainly positively charged particles).(1) The gas pressure in the chamber is kept low 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 Fig. 12.9. Assume a given beam of fast neutrons impinges on the chamber wall (1/4-in. 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.

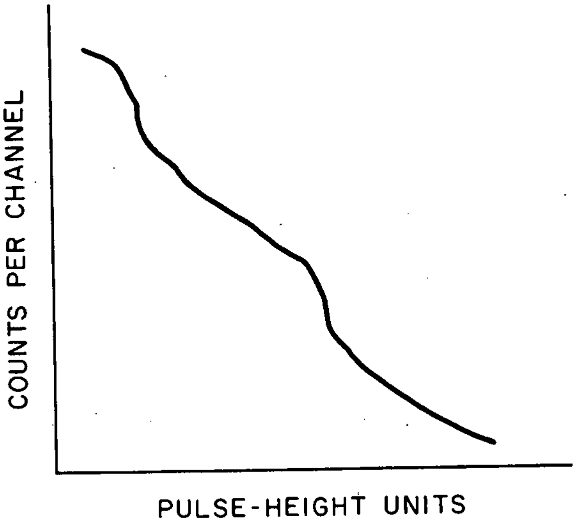


Fig. 12.9. LET Spectrometer Pulse-height Distribution

To convert the pulse-height curve above into the distribution in LET of dose (see page 219) requires a mathematical analysis that accounts for the difference in path lengths.(21) This analysis can be set up as a computer program, which performs the needed treatment.(30)

b. Junction-detector Alpha Spectrometer

ANL now uses an alpha spectrometer, which employs a silicon surface-barrier junction detector.(54) The detector is a p-n junction with an active area of 300 mm^2 and a depletion layer of about 125 microns at 50 Volts. The unit can detect alpha up to 15 MeV, protons up to 3.7 MeV, and beta up to 0.165 MeV. 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 junction device are amplified and sent to a 512-channel analyzer. The vacuum system improves the shape of the pulse-height curve. However, the sample must be plated on a thin planchet to decrease energy absorption.

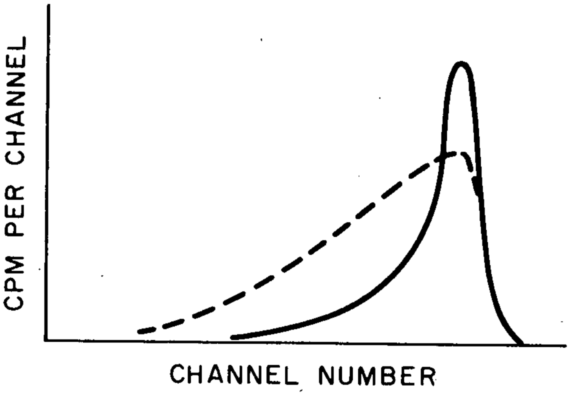


Fig. 12.10. Alpha Pulse-height Spectrum in Silicon Detector

The pulse-height spectrum obtained when a thin alpha source is measured in a vacuum is shown as the solid line in Fig. 12.10. We obtain very good resolution of the pluse sizes since little energy is lost by absorption in the source or along the path. The dotted portion in the figure is obtained if 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.

c. Gamma-ray Scintillation Spectrometers

At present, the scintillation method is quite popular for gamma spectroscopy. The most common phosphor used in this method is a NaI crystal, with a small (~0.1%) thallium impurity added.(55) This substance gives off large light pulses and has a high density ($\rho = 3.67$ gm/cc), 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 4 x 4-in. crystal has a diameter of 4 in. and height of 4 in. 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. For this reason, many devices use a series of batteries to obtain the total voltage needed.

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 (scaler, count-rate meter, and display system). In a single-channel unit, a sweep circuit is also included.

(1) 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.(55,56) 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 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 angle. The electron then 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 1.02 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 1.02 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 0.51-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 a NaI (Tl) crystal, the photoelectric effect is more likely to occur for photon energy less than about 250 keV.(56) 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.

(2) 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 results in the use of the total photon energy

to excite the crystal. Thus, the largest light pulse results, and this leads to the largest voltage pulse. The pulse-height peak that results from the largest light pulse is called the photopeak. The pulse-height value of the photopeak can then be related to the gamma energy, E_γ. 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 $E_\gamma - K_x$, where $K_x = 28$ keV for iodine.(55) 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.(56) 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

$$E_C = \frac{E_\gamma^2}{E_\gamma + 0.256}, \tag{12.3}$$

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 a peak. The spectrum at E_C usually appears as a gradual decrease in the distribution rather than a sharp cutoff. As the crystal size is increased, we enhance the probability of multiple scattering leading to total absorption. This results in more full-energy pulses (photopeak) and a reduced Compton distribution.

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 $E_\gamma - 0.51$ 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 $E_\gamma - 1.02$ MeV.

Several other peaks may also show up in the spectrum. The backing substance on the source sample may give rise to a backscatter peak. This peak occurs at a pulse size that corresponds to the energy, E_b, given by

$$E_b = E_\gamma - E_C. \tag{12.4}$$

A coincidence sum peak may appear when a source emits two or more gamma rays in cascade.(56) 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.(56) 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 are stopped in an absorbing substance.

(3). Pulse-height Spectrum Examples. The peaks that occur in gamma spectrum analysis can be traced to statistical fluctuations in the system. Thus, the peak consists of a distribution of pulse sizes. That is, a large number of pulses are about the same size (the peak value), but some pulses are larger and some smaller than this size. The peak may have a considerable width, and this can be expressed in terms of the resolution. The resolution is defined as the ratio of the full width at half maximum to the energy value at the photopeak. The resolution improves as we go to higher gamma energy and, for a NaI (Tl) crystal,(5) can be estimated by

$$\text{resolution} = \frac{6.5}{\sqrt{E_\gamma}}. \tag{12.5}$$

In this, the resolution is expressed in per cent, and E_γ is in 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.(55,56) 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 only the energy given up in the crystal. Figure 12.11 shows a typical pulse-height spectrum obtained for ^{137}Cs gamma in a

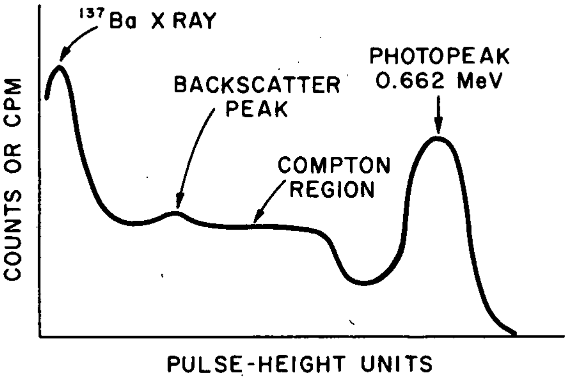


Fig. 12.11. ^{137}Cs Pulse-height Spectrum in a NaI (Tl) Crystal

NaI (Tl) crystal. The large bell-shaped peak is the photopeak from the 0.662-MeV gamma of ^{137}Cs. Since this is the highest-energy gamma emitted by ^{137}Cs, the peak from this pulse-size distribution appears at the highest position on the Pulse-height scale.

The Compton region starts at a pulse height equivalent to an energy of

$$E_C = \frac{E^2_\gamma}{E_\gamma + 0.256} = \frac{(0.662)^2}{0.662 + 0.256} \approx 0.48 \text{ MeV}.$$

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 $E_b = E_\gamma - E_c = 0.662 - 0.48 = 0.18$ 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 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 X ray, which has an energy of 32 keV, occurs when internal conversion takes place as an alternate decay mode. That is, ^{137}Cs emits a beta and goes to ^{137m}Ba. This is an excited state from which the 0.662 gamma 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.

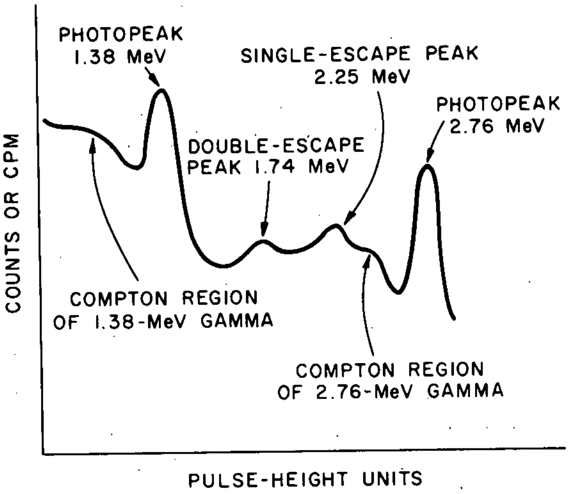


Fig. 12.12. ^{24}Na Pulse-height Spectrum in a NaI (Tl) Crystal

Our last spectrum (Fig. 12.12) is that for ^{24}Na. ^{24}Na emits a beta particle, which is followed by a 2.76-MeV gamma and a 1.38-MeV gamma in cascade. Here, the 2.76-MeV peak appears at the highest position on the pulse-height scale. This photon has enough energy to interact by the pair-production effect. The peak at 2.25 MeV is then the single-escape peak. This shows up if only one of the 0.51-MeV annihilation gammas escapes from the crystal. The peak at 1.74 MeV is then the double-escape peak. This will show up only if both gammas escape.

The shape of these peaks is affected by the presence of the Compton region for the 2.76-MeV gamma. That is, $E_c = 2.53$ MeV.

The next peak shown is the photopeak for the 1.38-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, so that the peak is still bell-shaped. The Compton region from this peak starts at a pulse height equivalent to $E_c = 1.16$ 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.(55,57) 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.(58)

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SECTION 13 - PERSONNEL MONITORING DEVICES

A. Introduction

The preceding sections discussed many of the instruments 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(1,2) revealed several features a good 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 0.1 R to greater than 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 to improve the performance of personnel monitors.

From the outset, one type of device, the film dosimeter, has been used with satisfactory results. Film dosimeters are used in almost all cases in which large numbers of people are involved. Work(3) with solid-state dosimeters shows promise of their use in the near future. However, the film dosimeter is still one of the more current devices for personnel monitoring. For this reason, most of this section will deal with film dosimetry.

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 detecting 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 widely used. The fact that it is small, rugged, and low in cost and provides a permanent record enhance its value as a dosimeter. Many fine papers in the literature(4-7) 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. Of prime concern are the components of the film itself.

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 Fig. 13.1). The base substance may be paper, glass, or cellulose. A typical X-ray film would have a base about 200 microns thick and two emulsion layers, each about 12 microns thick.(6) 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 microns in diameter in X-ray film and about one-fifth that size in neutron film.(6) The amount of silver bromide in the emulsion ranges from 30 w/o for some common films to 80 w/o for neutron films.(4) Neutron films also differ from X-ray films in that the neutron-film emulsion is almost three times as thick and often only in a single layer.

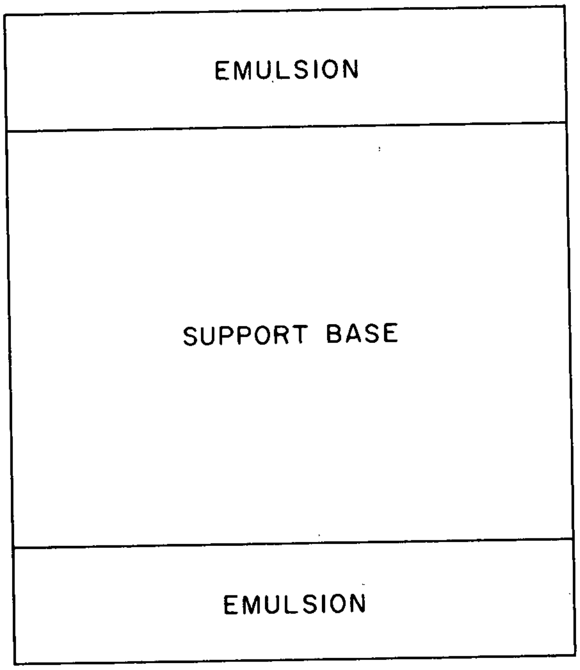

Fig. 13.1. Cross Section Through a Typical X-ray (photon) Film. [Adapted from Personnel Dosimetry Techniques for External Radiation (Symposium), Madrid, O.E.C.D./ENEA (1963).(6)]

2. Photographic Effect

In essence, the theory describing the effect of an exposing agent on a film was first advanced by Gurney and Mott.(8) More recent papers(9,10) 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.(6,9) These electrons exist for a short time in this state and then are trapped at defects in the crystal lattice. This trapping action seems to be done preferentially by silver ions. The trapping action creates a space charge, which then attracts a silver ion, resulting in an aggregate of two silver atoms. This silver center is then able to trap 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 ionization 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 firmly electrons are trapped and how stable the resulting silver clusters are. The size needed for a stable center is not too well known at present.(6) The unstable centers may regress or fade. In any case, the stable centers can be developed. During this process, the latent-image centers are reduced to silver by the chemical action of the developer. This causes the silver to plate 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).(4)

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.(5) The photographic effect is measured in terms of the diffuse transmission density, D. 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 (see page 262).

3. Properties of Film Response

Assume that an X-ray film is exposed to a photon beam. The photon action on the film produces a darkening. For a greater beam intensity, we would find a greater darkening. In other words, the film response is a function of the quantity of radiation. If film is exposed to more or less radiation, the film response is proportional to the amount incident on the film. This produces a curve that relates the film response to the exposure. A common method of representation is to plot the density versus log exposure. This type curve is often referred to as a characteristic curve (see Fig. 13.2).

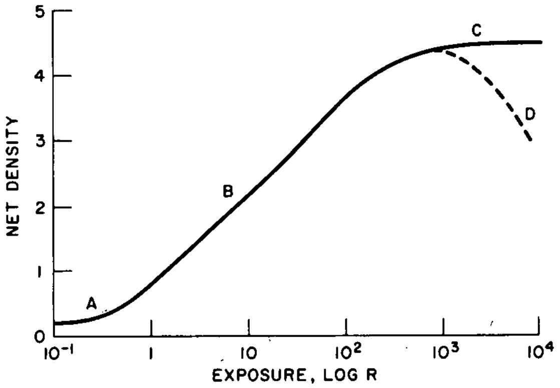

Fig. 13.2. Characteristic Curve for a Photon Film. [Adapted from NBS Handbook 57(5) and Personnel Dosimetry Techniques for External Radiation (Symposium), Madrid, O.E.C.D./ENEA (1963).(6)]

a. Characteristic Curve

The characteristic curve for many film types is shown by the solid line in Fig. 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), 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 contrast of the film is often taken to mean the slope of this straight portion. The contrast then is a measure of the ability of the film to distinguish between two nearly equal exposures. 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.

In some films, as shown by the dotted line (D) in Fig. 13.2, solarization occurs. In this case, a more or less constant value is not reached. Instead, the density begins to decrease as the exposure is increased. The cause of this effect is not yet fully understood.(6) It seems to be observed to a lesser degree when strong developers are used.

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.(4) This law fails if we are near the maximum density for a given film.(6)

Although the shape will be the same, the location of a curve along the log R 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,(12) but the location depends upon beta energy. The preceding statement applies to the case of betas with enough energy to pass through the entire emulsion (see page 260).

c. Latent-image Fading

The fact that the latent-image centers tend to fade with time, limits the time a badge may be worn. This fading is enhanced by the presence of oxygen, high humidity, and high temperature.(6) Under extremes of these factors, the fading may be quite serious. Under normal conditions, such as room temperature or below and low relative humidity,(7) the fading is slight for X-ray films. For photon films, most of the fading occurs within the first 10 hr under normal conditions.(6) 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.(13) The response of film does not vary greatly in the range from room temperature to 99°F.(7) Also, the temperature at which film is stored before or after being exposed seems to have little effect.

d. Film Speed

We can measure the sensitivity of a photon film in terms of the reciprocal of the exposure needed to produce a given density. This also is used as a measure of film speed.(14) Films are available in a wide range of speeds from a number of companies. Tables of film speeds for many films useful for radiation monitoring are contained in the literature.(4,5,7)

For most film types, the sensitivity below 100 keV may be up to 30 times that at 600 keV. This can be explained on the basis of increased absorption of photons of low energy in the film over that in air. For the region above 600 keV, the absorption in air becomes proportional to the absorption in the emulsion so that the sensitivity remains fairly constant.(5) This aspect of film response is referred to as energy dependence.

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 Fig. 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.

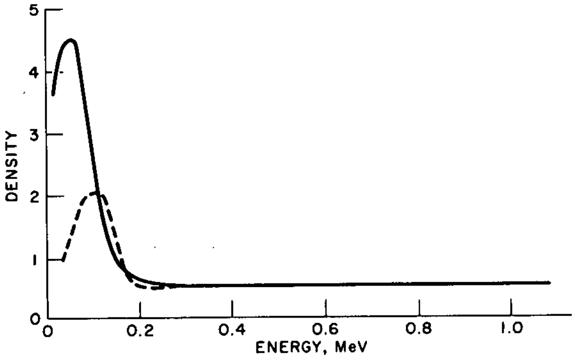

Fig. 13.3. Energy Dependence in Typical Photon Film, with and without Filter. (Adapted from NBS Handbook 57.(5))

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. That is, the response curve reflects the absorption of energy in the silver bromide. However, the exposure 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.(6) The film absorbs a larger fraction of the photons at this energy than at any other that 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.

Among the methods that have been proposed to correct for increased response, the use of filters seems to be the more popular in this country.(15) As a result, the design of the film holder itself has become quite a study project.(2,7,15,16)

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 so that they do not greatly affect the photons in the region above 0.3 MeV. They are thick enough, though, to affect those below 0.3 MeV. The use of filters then tends to flatten the response somewhat in the region below 0.3 MeV. This is roughly pictured by the dotted portion in Fig. 13.3. We may note these general features that the use of a filter produces: (1) The peak response is reduced, or flattened; (2) the location of the peak may shift; and (3) the region of uniform response may extend down closer to 200 keV.

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 easy. The problem is to find a filter that flattens the response in the range 30-200 keV.(2)

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.(2) For these reasons, different filter materials are used throughout the country.(15) Some common substances used include Cd, Sn, W, Pb, Ag, Cu, and Al. Each of these substances produces its own preferential absorption of the softer radiation. The choice of the substance used, as well as its thickness, depends 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 an indication of the radiation energy.(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 ≥100 keV.(7) A 0.5-mm lead filter is useful for E ≥ 200 keV.(17) A single shield composed of many elements has also been reported.(18)

4. Processing of Film

The manner in which the film is processed affects the final density obtained. The density is a function of these factors: (1) development time, (2) temperature and age of the developer, and (3) amount of agitation.(7) If we ignore these factors, serious errors can result.

The slope of the characteristic curve (contrast) is a function of the developing time and the nature of the developer. The type of developer is almost always the same as those for medical X-ray films.(7) These are used at approximately 68°F for about 3 to 5 min. With care, we can extend the range of some films by the use of the proper developer.

When the film is removed from the developer, it is placed in a hypo solution (fixer), which dissolves the unused emulsion. The permanence of the image is a function of the time the film is left in the fixer. In many cases, this time may vary from 5 to 15 min. To prevent the film from curling, it is then rinsed, removed, and dried at room temperature.

5. Reading and Interpretation

If a film is processed without being exposed it will still show a slight density. This density is known as the base fog. The magnitude of this effect is a function of the type of emulsion, the age of the film, and the manufacturing processes, as well as 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.

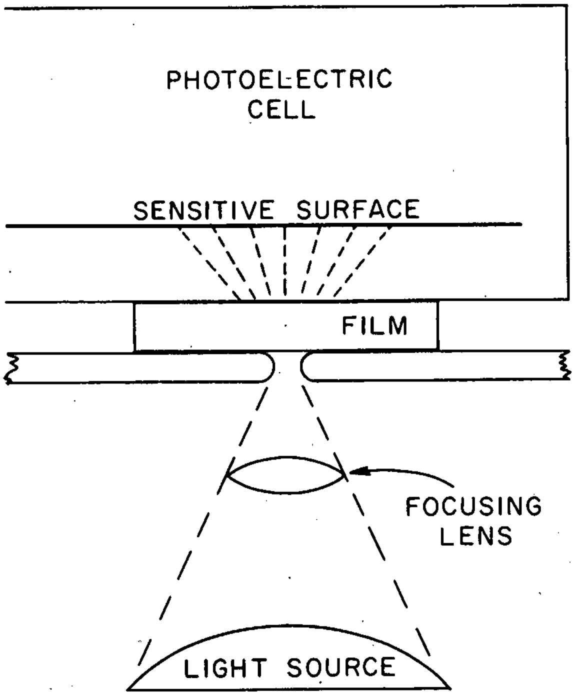

Fig. 13.4. Densitometer, Simplified Setup

a. Densitometer

The processed film is read on a densitometer (see Fig. 13.4). An intense source of light is focused on a small opening. The light passes through a small section of the film and falls upon a photoelectric cell. When light falls upon the sensitive surface of the cell, electrons are released. The number of electrons released is proportional to the amount of light that impinges on the cell surface. The electrons are collected, causing a current to flow. The current is then read on a microammeter. The scale of this meter gives the density reading in number units. The range of these devices sometimes goes as high as a density of 5.0.(7)

The densitometer can be adjusted to read 0.0 when no film is in the path of the light. That is, the cell circuit can be adjusted by means of a zero control to provide the desired meter reading. Then, when a film is in place, some of the light is absorbed. This changes the current flowing in the cell circuit, and the meter needle deflects. The greater the darkening on the film, the greater the meter reading. On most devices, experienced readers can read densities to about ±0.01. Of more importance than absolute accuracy in reading films is the reproducibility of the readings.

b. Calibration 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. By doing this, we can account for: (1) the variation of response because of energy, and (2) the variation in density because of processing factors.

In making film calibrations, we must take care to reduce sources of stray radiation, since these will affect the film. We should have a calibrated gamma source (such as ^{60}Co or ^{226}Ra) and an accurate determination of the exposure rate at the point where the film is exposed. Some calibration sources are listed in Ref. 7. The source and films should be spaced well away (at least a meter) from objects that might scatter the photons.(7)

Frequent calibrations are needed because no two batches of the same type of film have exactly the same response. Both the base fog and the calibration curves may change. Variations as high as 25% may be expected, even with careful techniques.

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.

c. Interpretation

The interpretation of the readings in terms of the exposure depends upon the number of filters used in the badge. In some units, this number may be as much as five.(15) 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. The response under the filters is affected in various degrees by the amount of soft radiation each filter transmits.

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.(6) If the radiation field consists of a mixture of beta and gamma of unknown energy spectra, then we need a multishield device.(16)

For a multifilter badge, the densities behind the filters reflect somewhat the character of the transmitted radiation. This enables us to gain 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 section 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 beta. Sometimes, the response due to beta plus gamma may result in a density so high that we may not be able to read the film.

Many multishield systems are in use at present.(15) In all of these, the absence of known spectral data requires a somewhat complex procedure to interpret the response.(2,16,19) An added feature in some multishield systems is their response to thermal neutrons (see page 263).

6. Film Holders

The film holder is often in the form of a clip-on badge, in which the shields are often 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.(15)

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. In some cases, finger exposure has been related to readings obtained from wrist badges.(20,21)

In recent years, a movement to equip badges with a number of devices to detect nuclear excursions has been gaining favor.(15) Perhaps the best example of this design is the Oak Ridge badge.(16) The unit contains both fast (>2.5 MeV)- and thermal-neutron detector units. Also included are glass rods to measure high gamma doses and a chemical dosimeter useful for visual sorting of exposed persons after an incident.

Not all film-badge systems are as elaborate as the ORNL badge. However, in installations at which a criticality might occur, nearly all have at least an indium foil for use in quick sorting of exposed persons.

7. Film Packets

Many films used for monitoring are similar to medical X-ray film and are contained in dental-size packets. No one film covers the range of exposure needed for personnel monitoring; therefore these packets contain at least two films. For example, the DuPont No. 558 packet contains a No. 519 film (40 mR to 30 R) for the low range and a No. 1290 film (9-3000 R) for the high range.

No one type of packet seems to be preferred in this country. A recent survey of 23 users showed that eight different film packets are in use.(15).

The packet material is often an opaque paper wrapper of about 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.(6)

8. Electronic Equilibrium Conditions

The film and its wrapper constitute an equilibrium thickness up to about 0.3 MeV.(5) 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 substances have been used as the equilibrium layer. When this layer consists of a substance with a higher Z number than the paper, complications may result.(5) For this reason, 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 that 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 Ref. 22.

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 recent detailed study(23) points out that we can obtain good results with proper care.

Much of what has been already said has dealt with photon films. The next few pages discuss neutron monitoring by film techniques.

9. Neutron Dosimetry

As in the case of instrument design, neutron monitoring by films 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 or photons being emitted. This occurs when elastic scattering and/or capture take place.

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 on photon film.(24) Also, elastic collision between fast neutrons and hydrogen in the emulsion and surroundings lead to recoil protons. Capture of thermal neutrons by nitrogen also produces recoil protons. These protons in turn lose their energy to the grains, which result in the film density.

Studies(25,26) have shown that the sensitivity of photon film to fast neutrons is quite low. For some of the photon films used in these studies, the relative sensitivity to fast neutrons is inversely proportional to the grain size. The thermal-neutron response in terms of fluence is roughly equal to that of 3- to 4-MeV fast neutrons. However, since the response is due mainly to photons and beta, sensitivity is not expected to vary much with grain size.

Neutron 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 the neutron effect when photons are present. For this reason, fast-neutron monitoring is often done by track-counting methods.

a. Fast Neutrons

(1) Track Counting. Track counting as a means of measuring 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 in these emulsions are much finer than in X-ray films. The emulsion also contains much more of these AgBr grains than do photon films. The latent images formed during the passage of a recoil proton 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 is the Kodak Personal Neutron Monitoring, Type A.(15) This type of emulsion is usable in the energy range 0.5 to 14 MeV, in the proper kind of packet.(27) 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. 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 protons 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.(28) The use of these emulsions is also limited by the response to gamma. The response amounts to a fogging effect, which makes it quite difficult to pick out the tracks. Thus, a gamma field that produces an exposure of a few Roentgens renders this type of film unreadable for neutrons. Attempts(29) to treat this type of film to reduce the gamma response but maintain the neutron response show promising results.

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.(16) A 40-mil-cadmium filter (~1 mm) absorbs the thermal flux so that proton-recoil tracks behind this shield should be due 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. We can protect the film and reduce this effect by sealing the film in humidity-proof bags.(6)

(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 magnification used may be about 750X. 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 seen in any one field is small and varies from field to field. For this reason, 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 study of Snyder and Neufeld.(30)

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 that we have exposed the film under proton-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 of neutrons is not now able to cover the entire range of neutron energy desired for health-physics work.

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.

In the presence of a dense photon field, though, it becomes difficult to determine what portion of the density is due to thermal neutrons. Now, if we have both a high-capture cross-section substance (cadmium) and a low one (tin), it becomes fairly easy to obtain the thermal-neutron response.(24,31) In this case, the density under the cadmium is caused by both capture gammas and the other photons present. The density under the low-capture cross-section 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.(7) 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, 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 consists of a small quartz-fiber electroscope, which forms 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 (~2 cm^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 becomes

$$Q = CV, \text{ in Coulombs.} \qquad (13.3)$$

The value of the charge put on the device is a function of the design. A typical value for the capacitance is around 3 $\mu\mu$F, and the charging voltage may be 180 Volts. In this case, the total charge that would result is then $Q = (3 \times 10^{-12})(180) = 5.4 \times 10^{-10}$ 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,

$$X = \frac{Q}{m}, \qquad (13.4)$$

in which Q is in Coulombs, and m is in kg. Therefore, for the pocket dosimeter,

$$X = \frac{Q}{m} = \frac{C\Delta V}{\rho \text{ Vol}}, \qquad (13.4a)$$

where ΔV is the drop in voltage, ρ is the density of air in kg/cm^3, and the volume, Vol, of the chamber is in cm^3. The special unit of exposure, the Roentgen, is given by $1 \text{ R} = 2.58 \times 10^{-4}$ Coulombs/kg. This leads to the following expression for the exposure in terms of Roentgens:

$$X(R) = \frac{C\Delta V}{2.58 \times 10^{-4}\, \rho \text{ Vol}} = \frac{3 \times 10^{9}\, C\Delta V}{\text{Vol}}. \qquad (13.4b)$$

The final form results when we replace ρ by its value of 1.293×10^{-6} kg/cm^3. The density of air is a function of both temperature and pressure, so that a correction term is often needed.

The range of the device is then a function of the parameters in Eq. 13.4b. Suppose we charge a dosimeter of sensitive volume 2 cm^3 and capacitance 3 $\mu\mu$F to 180 Volts. What will be the voltage change if the device is expected to read an exposure of 100 mR, full scale? The required voltage change would be

$$X = \frac{3 \times 10^{9}\, C\Delta V}{\text{Vol}} \text{ Roentgens;}$$

$$100 \text{ mR} = 0.1 \text{ R} = \frac{3 \times 10^{9}(3 \times 10^{-12})\ \Delta V}{2},$$

and

$$\Delta V = \frac{0.2}{9 \times 10^{-3}} = 22.2 \text{ Volts.}$$

The scale of the device can then be designed so that, for the above voltage change, the fiber image moves from 0 to 100 mR. 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 100 and 200 mR, as well as 1, 5, 10, and 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-hr period.

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 beta response is often poor. Most of these units are highly directional in their beta response and do not respond to beta below 1 MeV.

Some of these meters may have an inside coating of ^{10}B so that they can be used as thermal-neutron dosimeters.(34)

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.(35) 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.

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.(3,35-37)

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 10^3 to 10^6 rads. This range for plastics is between about 10^5 and 10^8 rads for most substances. 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.(36) 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 10^5 to 10^8 rads, it is also not suited for personnel dosimetry.

1. Thermoluminescence

Phosphors that have been used the most in studies of thermoluminescence include manganese-activated calcium fluoride (CaF_2:Mn)(38) and lithium fluoride (LiF).(40) Other substances, such as $CaSO_4$:Mn, MgF_2:Mn, and Al_2O_3, have also been used. 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) 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.(38)

a. Lattice Defects or Traps

Solids that have a crystal structure contain many kinds of lattice defects. 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).(38) The regions about these defects are not always able to maintain a state of neutral charge. The presence of any charge 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 in 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.(38)

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 is often measured by means of a photomultiplier tube.(38) A plot of the brightness (relative intensity) versus temperature as the solid is heated is called a glow curve (see Fig. 13.5). 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 Fig. 13.5. 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. Thus, the integrated intensity of any peak can be used to measure the dose. Also, the dose can be related to the peak height, but then we must have a standard method of heating the solid. The reason for this is that the peak height is a function of the heating rate. To use the height as a measure of the dose requires that the heating be done the same way each time. The brightness peak tends to reach a higher value for a faster heating rate. Since the light sum remains the same, there is a shorter duration of emission.

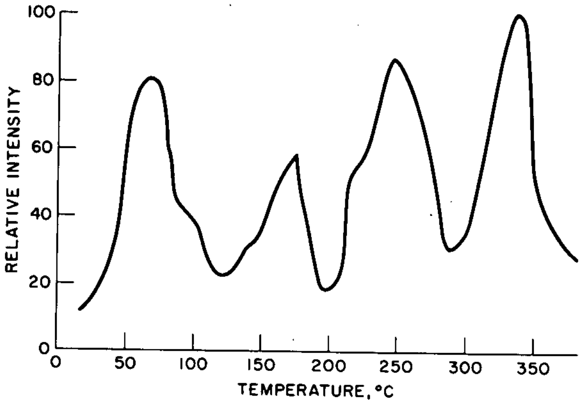

Fig. 13.5. Thermoluminescent Glow Curve. [From Personnel Dosimetry Techniques for External Radiation (Symposium), Madrid, O.E.C.D./ENEA (1963).(38)]

c. Characteristics

The response to gamma is linear in the exposure range from a few mR to greater than 10^3 R; therefore, these phosphors can be used for personnel dosimetry. The response to fast neutrons is low in these substances. The thermal-neutron response of CaF_2:Mn units is about one-fourth that of the gamma response. The gamma response of LiF is much smaller than that of CaF_2:Mn. The thermal-neutron response is much better in these units because of the presence of ^{6}Li. The thermal-neutron response is about 40 times that of gamma.(40) If we use pure ^{7}Li in the unit, the response to thermal neutrons is small. Thus, we can use a pair of units, one with ^{7}Li and one with ^{6}Li in it, to arrive at both gamma and thermal-neutron doses.

The energy dependence is related to the type of phosphor used. Cameron(40) reports that LiF is energy-independent in the range 30 keV to 1.2 MeV, within 25%. CaF_2:Mn is quite energy-dependent by itself, but the use of a tin filter has reduced this feature to within 30% down to 40 keV.(38)

These phosphors suffer from fading.(3) If we use the deeper traps in the solid, the fading is less severe.(38) 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 above 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.(37) However, LiF seems to be more prone to damage effects than CaF_2:Mn.(41) Both substances can be made into sufficiently small units for personnel-dosimetry purposes.

Thermoluminescence was first proposed for dosimetry in 1953.(42) 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 been slow. We may perhaps mention that to measure the dose, we must remove the stored-energy centers. This makes a recheck impossible.(3) The present practice in this country is to hold film badges indefinitely for recheck purposes.(15) This conflict may have slowed down the acceptance of thermoluminescence, but recent increased interest may yet lead to its common use.

2. Infrared Stimulation

A method somewhat closely related to thermoluminescence consists of illuminating the solid with a light whose wavelength is longer than that emitted during luminescence. The formation of the centers at traps is similar to what occurs in thermoluminescence. Instead of the luminescence that occurs as the result of thermal agitation, the light is emitted when the solid is illuminated by infrared light. This method was first suggested in Refs. 43 and 44. Substances such as KCl:Ag, NaCl:Ag, and SrS:Eu,Sm have been investigated. The useful dose range is about the same as for the thermoluminescent units, but the loss of dose during storage has been the most significant fault. Reference 3 reviews this method, which seems to be similar to thermoluminescence in most characteristics.

3. Radiophotoluminescence

a. General

As a result of the type of center used in the measurement, thermoluminescence and infrared stimulation both result in the loss of the stored information during readout. The radiophotoluminescent method is not subject to this feature; therefore RPL units may be reread. This

5

method has been used with success for silver-activated phosphate glass.(45) Studies(46,47) 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.(36) 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 10 rad to greater than 10^3 rad. This Schulman glass is often referred to as high-Z glass(3) because the effective atomic number is about 28. The glass was designed for use mainly as a casualty-type dosimeter.(16,36,52) In that state of limited sensitivity, the glass was not of use for routine personnel monitoring.

Yokota(50) has devised a new low-Z glass as well as a new fluorimeter. The effective Z of this glass is about 16 and is obtained by replacing the KPO_3 in the Schulman glass with $LiPO_3$. The linear response is retained, and the sensitivity to lower photon doses is greatly increased. The reported(50) gamma sensitivity of less than 50 mR makes this type glass of interest in personnel monitoring. A 1963 article(5) also discusses ways to extend the response range to 10^7 rad.

The fast-neutron response is small in RPL devices, being perhaps 0.5-1.0% of the gamma response.(53) 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 ^{6}Li and ^{10}B. We can alter the thermal-neutron sensitivity for the low-Z glass by using pure ^{7}Li. At present, the Yokota glass has a higher thermal-neutron sensitivity (10 mrem) than does the Schulman glass.

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 for the Yokota glass has been reported to be 8-12 times the ^{60}Co response.(49) 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(50) describes a number of filter arrangements for decreasing energy dependence.

If the RPL devices are kept out of sunlight, the fading in them is small, about 5% in 6 months. Sunlight greatly affects the fading, so that errors of 50% or more may result.(54)

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 8 x 8 x 4.7 mm. Fluorimeters, although rather expensive,(3) are commercially available for reading these devices.

The recent 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.

4. Conductivity Methods

Little has been done with the class of solids known as semiconductors in regard to personnel dosimetry. One reason for this may be the expense involved for large numbers of these devices. A definite reason in some cases is a lack of sensitivity. This lack of sensitivity precludes the use of CdS crystals since they must be used at high dose rates.

The silicon-junction counter has been used to measure radiations that produce a large amount of ionization. This type detector has been used for slow neutrons if the device is covered with a thin layer of ^{6}LiF or ^{10}B. The main drawback of these units has been their high cost.(3)

Recently, the silicon p-n junction counter has 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 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 gamma rays. It is not yet suitable for low-level use since its sensitivity is limited to about 5 rad. It may find use as a fast-neutron dose indicator for nuclear-accident dosimetry.

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SECTION 14 - AIR SAMPLING

A. Introduction

Air sampling is done 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.

In principle, regulations governing radiation dose equivalent to individuals should apply to internally deposited emitters as well as to external sources of radiation. Permissible levels for external radiation dose equivalent should be reduced by the dose equivalent obtained from internal emitters. However, accurate quantitative dosimetry is practically unobtainable for internal emitters. Therefore, it is common practice to keep environmental concentrations of radionuclides well below the RCG level. To confirm that this is being done, air samples are collected.

B. Nature of the Contaminant

Contaminants are generally divided into two groups:(1)

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.

Particulate matter includes solid and liquid particles ranging upward from molecular size (approximately $10^{-3}\mu$). Solids are further subdivided into dusts, fumes, and smokes, which are distinguished mainly by their mode of generation.(2) A dispersion of liquid particulates in air is called a mist or fog.

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.

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 radioisotopes (as shown in Fig. 14.1).

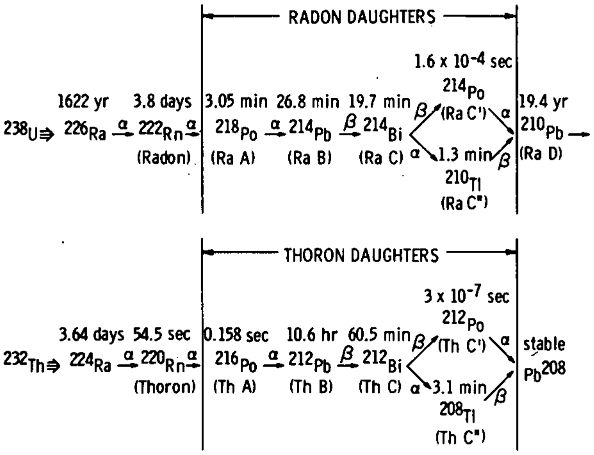

Fig. 14.1. Uranium and Thorium Decay Chains

One of the daughter elements in each chain is a noble gas - radon (^{222}Rn) in the 238 chain, and thoron (^{220}Rn) in the 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. It has been widely observed that air masses moving from over large water areas show low concentrations of these isotopes.(3)

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. The rise is attributed to a general decrease in wind velocity and increasing stability of surface conditions. The decrease in concentration is associated with vertical transport from the earth's surface as it is warmed. Seasonal as well as diurnal variations are observed.(3-5)

The radon/thoron activity ratio is also variable, as is the radon-daughter/thoron-daughter ratio. Havlovic(6) reports a value of 4.7 (six-months average) for the latter, which is in general agreement with values ranging from about 1 to 6 reported by Kawano and Nakatani.(4)

Although thorium is more abundant than uranium in the earth's crust, radon has a half-life (3.8 days) considerably longer than that of thoron (54.5 sec) and thus is better able to migrate to the atmosphere.(7)

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 ranging from 0.005 to 0.04 μ in size.(8) 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 Part H below, Analysis of a Filter Sample.

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.(9) 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 Fig. 14.2).

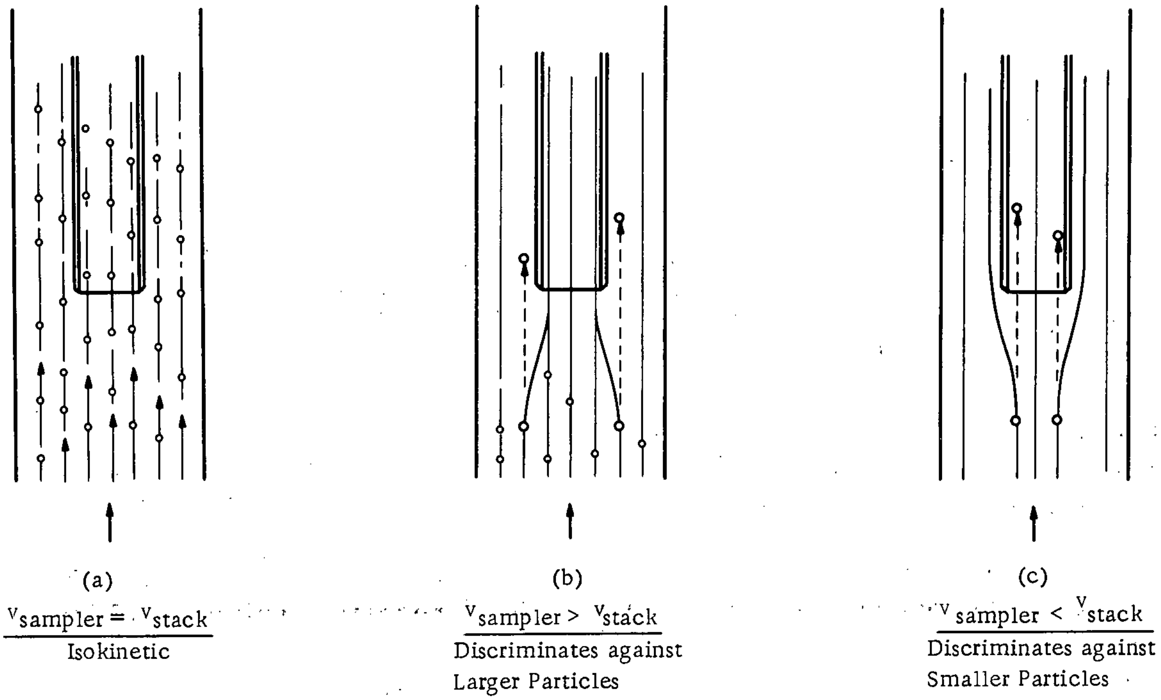

Fig. 14.2. Isokinetic Sampling

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. Sampling lines should be as short and as straight as possible to avoid plating out along the walls of the tube. The sampling head should be sufficiently far removed (at least three stack diameters) from the last bend, opening, or obstruction to avoid the effects of excessive turbulence in the airstream. The need for isokinetic sampling diminishes as the particle size in the aerosol decreases.

Secondly, the concentration at the sampling head must be representative of the concentration at the point of concern. Since this is usually the nose of an individual, the sample must be of a concentration approximating that in the breathing zone of the individual. Sherwood and Greenhalgh(10) have reported a fivefold variation 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 gloveboxes.

E. 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 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 isotope is usually its radioactivity; it will normally be the basis of any analysis made.

Finally, the choice of 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.

F. Sampling Methods and Devices

Methods and devices employed for radioactive air sampling at ANL are briefly discussed in this section.

1. Filter Samplers

Filtration is the most common method of sampling for radioactive particulates. It is simple and efficient. The mechanics of filtration are discussed in Part G 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.

The Hollingsworth Voss HV-70 filter is used at ANL for general-purpose air sampling. It is a closely formed, cellulose-asbestos paper with good efficiency (approximately 98%), low pressure drop, and good mechanical-strength characteristics. Analysis is commonly done by gross counting of alpha and beta activities in stacked proportional counters (piggyback probe). For alpha counting, the self-absorption in the paper must be taken into account.

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 ranging from about 0.01 to 5 μ, 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.(11) 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.

The 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.

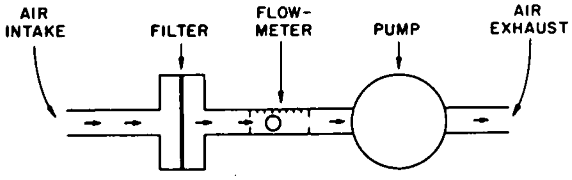

Fig. 14.3. Typical Filter Sampler Schematic Diagram

A sampler in common use at ANL 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 HV-70 paper (4 x 9 in.) at a flow rate of 15-20 m^3/hr.

The Silverman sampler, employing a positive-displacement blower, is used where samples are taken continuously over several days. It operates at flow rates up to about 35 m^3/hr using 4 x 9-in. HV-70 paper.

Portable, high-volume air samplers (Hurricane, Staplex, and Unico) are used for short collection times. With 5 x 8-in. HV-70 papers, these units sample at rates up to about 60 m^3/hr.

Membrane or cellulose fiber filters in small (1- to 2-in.-diam.) 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. The Minihead sampler(12) developed at ANL is an example of this type sampler.

Lapel samplers, operated on rechargeable batteries, have recently come into use and offer an opportunity to sample the breathing zone of an individual moving about. Flow rates of present devices range from 2 to 20 liters/min.

Once the sample is collected, the activity on the filter can be determined in several ways. Gross alpha or beta counts can be taken, and from these the filter activity can be calculated using a previously determined counter yield.

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 Part H below.

Gamma spectroscopy can be done if the contaminant has a suitable X or gamma 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. Electrostatic Precipitators

An efficient way to collect particulates is by electrostatic precipitation onto a charged electrode by an electric field at right angles to the flow of air. The intense field serves both to ionize and collect the aerosol particles.(13)

The collector, usually a glass or metal cylinder concentric to a highly charged (10-20 kilovolts) center electrode, is removed for analysis. Gross beta counts may be taken by mounting the collector coaxial to a G-M tube; the collected material can also be washed from the collector and counted.

The efficiency of collection can be made to approach 100% by the proper selection of electrode voltage, collecting-tube dimensions, and air flow.(14,15) Typical parameters would be 3-cfm air flow, 12,000 volts d.c. and a $1\frac{1}{2}$ x 7-in.-long collecting tube.

The Industrial Hygiene Group at ANL frequently uses an instrument of this type.

3. 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

$$V = V_f \frac{p - p_r}{p}, \tag{14.1}$$

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,

$$V = V_f \frac{p_g}{p}. \tag{14.2}$$

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°C and 760 mm Hg) to give

$$V_s = V\frac{p}{p_s}\frac{T_s}{T} = V\frac{p}{760}\left(\frac{273}{273 + t^\circ\,C}\right) = V_f\frac{p_g}{760}\left(\frac{273}{273 + t^\circ\,C}\right) \tag{14.3}$$

This allows comparison of different samples on a standardized basis.

The grab sampler in common use at ANL (see Fig. 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 microcuries of 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 beta particle.

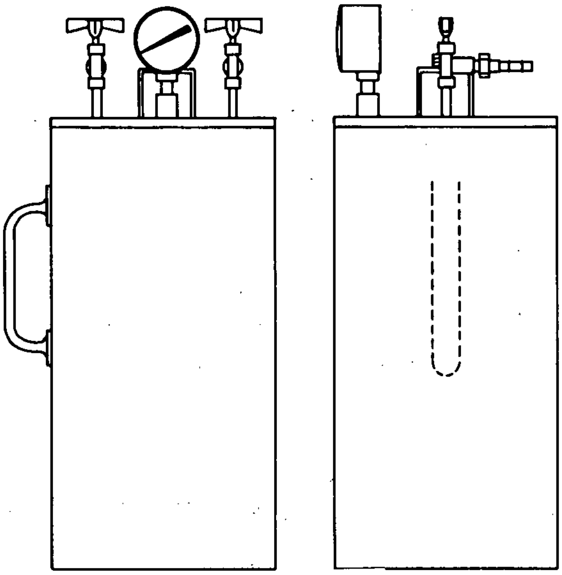

Fig. 14.4. 3.5-liter Grab Sampler

The weakness of this scheme is its low efficiency. The 3.5-liter sampler, using a 30-mg/cm^2 tube 3 in. long, has a yield of about 0.5% for ^{41}Ar.

Grab samples taken in glass bulbs or bottles can be analyzed using gamma pulse-height analysis. Yields of nearly 20% have been obtained for ^{41}Ar in a 110-ml bulb counted between two 4 x 4-in. NaI crystals.

Beta-emitting gases sometimes can be identified and quantitatively analyzed by letting them decay to particulate daughters in a sampling flask.(16) The flask can then be washed out, and an aliquot of the wash water counted in a proportional counter or end-window G-M counter. Decay curves or aluminum absorption curves may help to confirm the identity of the active nuclide.

4. Impingers

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 (Fig. 14.5) and its smaller analog, the midget impinger. The sampling rates for these instruments are 1 and 0.1 cfm, respectively.(17) Their collection efficiency becomes generally poor for particles smaller than about 0.7 μ.(15)

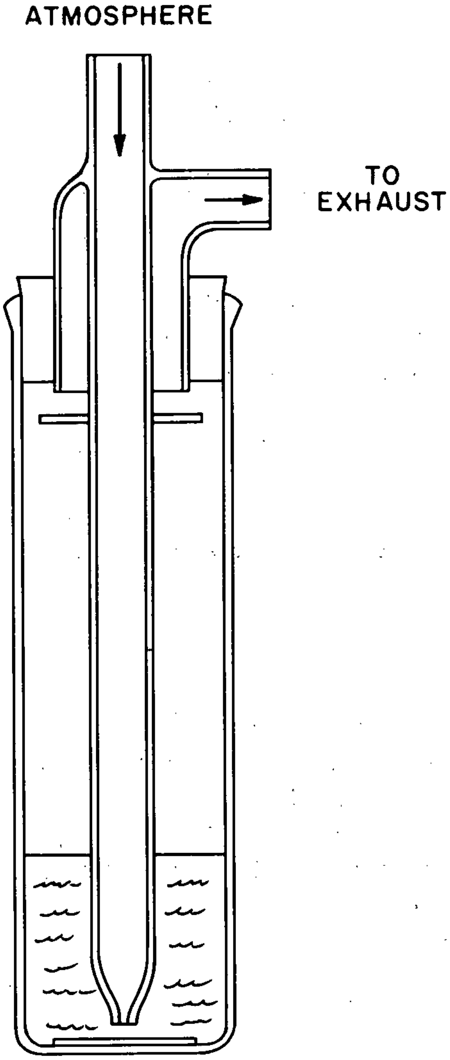

Fig. 14.5. Greenburg-Smith Impinger. Wet impingement is the principle used here.

The impinger has long been a standard instrument of the U. S. Public Health Service, and many standards of permissible concentrations of dusts are based on measurements made with it.

The cascade impactor (Fig. 14.6) consists of several (usually four) impingement stages in series. 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 in this way. However, distortion may be introduced because of the breakup of the larger agglomerates undergoing impingement. This is a problem with all impingement devices.

Impactors may be useful for sampling in the presence of natural radon-daughter activity, which is usually associated with particles smaller than 0.3 μ in diameter.(18)

5. 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 for nonreactive or insoluble gases. At ANL, 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 give the concentration of water vapor per unit volume of air. Together the two numbers give the tritiated water concentration per unit volume of air at the time of sampling.

6. Adsorbers

Adsorbers such as activated charcoal, silica gel, or alumina gel are used for collecting organic vapors and nonreactive gases and vapors

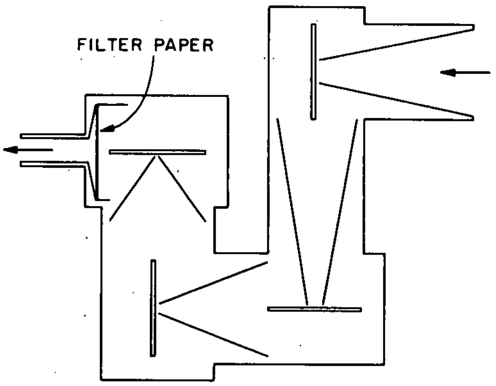

Fig. 14.6. Cascade Impactor

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.(19)

Activated carbon cartridges of the type described by Sill and Flygare(20) have been used at ANL 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.

Analysis of the cartridges used for iodine is accomplished by gamma spectroscopy.

7. Continuous Air Monitors

Continuous air monitors, often homemade assemblies of various components, are used to maintain a watch on the level of air activity in an area such as the inside of a reactor containment shell.

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.

One interesting instrument, the continuous air monitor CAM-5,(21) was designed at ANL. It uses the equilibrium counting ratio of radon daughters by simultaneously counting the beta and alpha 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 beta counting rate. Although the individual counting rates may vary, their ratios remain quite 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 alpha- or beta-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.

G. 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.(11,22,23) 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.(15,24) Theoretically, we would expect a minimum collection efficiency (maximum penetration) for particles of some intermediate size at a given airflow(25,26) 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.(27-29) When found, the maximum often occurs in the neighborhood of 0.2 μ.(25) Minimum deposition in the deepest part of the lung is observed in this region.(30-32)

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 m^3 sample is taken for ^{226}Ra, 99.9% efficiency is assumed, and a net counting rate of 220 cpm is obtained on a counter system having a 50% yield (Y) for ^{226}Ra. Then,

$$A(^{226}Ra) = \frac{220 \text{ cpm}}{Y \text{ cpm/dpm}} = 440 \text{ dpm} = 2 \times 10^{-4} \mu Ci;$$

$$C(^{226}\text{Ra}) = \frac{2 \times 10^{-4}\ \mu\text{Ci}}{20\text{m}^3(0.999)} = 1.001 \times 10^{-5}\frac{\mu\text{Ci}}{\text{m}^3} = 1.001 \times 10^{-11}\frac{\mu\text{Ci}}{\text{cc}}.$$

If the penetration were actually 1% rather than 0.1%, the concentration, C, would really be

$$C = \frac{2 \times 10^{-4}\ \mu\text{Ci}}{20\text{m}^3(0.99)} = 1.01 \times 10^{-5}\frac{\mu\text{Ci}}{\text{m}^3} = 1.01 \times 10^{-11}\frac{\mu\text{Ci}}{\text{cc}}.$$

Truly, this is a negligible error.

Now, assume that an air-cleaning filter had a concentration of 10^{-8} μCi/cc on its upstream side. An assumed efficiency of 99.9% would give a concentration of 10^{-11} μCi/cc on its downstream side, whereas an actual efficiency of 99.0% would mean a downstream concentration of 10^{-10} μCi/cc. For the isotope chosen above, this means the difference between exhausting concentrations of 1 and 10 MPC (maximum permissible concentration) for an uncontrolled area.

H. 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.

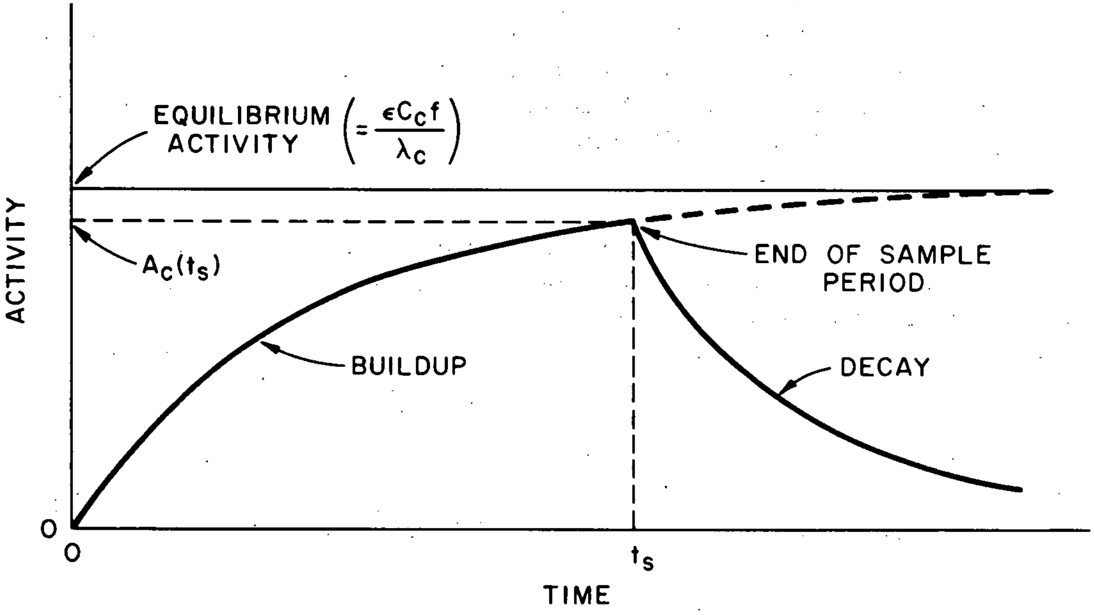

Fig. 14.7. Activity of a Single Contaminant on the Filter

The net activity on the filter due to the contaminant at any time during sampling is given by

$$A_c(t_s) = \frac{\epsilon C_c f}{\lambda_c}(1 - e^{-\lambda_c t_s}), \tag{14.4}$$

where

ϵ is the efficiency of collection of the filter,

C_c is the average concentration of contaminant in the air,

f is the flow rate of air through the sampler,

and

λ_c is the decay constant for 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 counting time is related to the activity at the end of sampling by

$$A_c(t_1) = A_c(t_s)\, e^{-\lambda_c(t_1 - t_s)}. \tag{14.5}$$

The activity is obtained from the counting rate, $R(t_1)$, on the filter by

$$A_c(t_1) = \frac{R(t_1)}{Y} = \frac{R_{tot}(t_1) - B}{Y}, \tag{14.6}$$

where

Y is the yield of the counting system for the isotope 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

$$C_c = \frac{A_c(t_1)}{\frac{\epsilon f}{\lambda_c}(1 - e^{-\lambda_c t_s})\, e^{-\lambda_c(t_1 - t_s)}}. \tag{14.7}$$

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 isotope 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 condition exists.

The upper limit of seriousness of air contamination can be estimated crudely by assuming that all the activity on the paper is due to the contaminant, and that no decay occurs on the filter since we are usually concerned with half-lives much longer than our sampling time. The scheme is plausible for beta emitters, as is seen in Table 14.1. Table 14.1 shows

Table 14.1

MAXIMUM AND MINIMUM CONCENTRATIONS IN N KENNEL DURING PERIOD 1/3/66 TO 7/8/66 IF ALL ACTIVITY ON FILTER IS DUE TO CONTAMINANT OF INTEREST

	Minimum Concentration, dpm/m^3	Maximum Concentration, dpm/m^3	Allowed Concentration of Most Harardous Emitter, dpm/m^3
Beta Emitters	2.4	72	660 (^{90}Sr)
Alpha Emitters	1.2	36	4.4 (^{239}Pu)

the maximum and minimum values of concentration derived in this manner during daily sampling in N kennel of the Biology Building between 1/3/66 and 7/8/66. The scheme is useless for hazardous alpha emitters because the spread in the background air activity is many times their low permissible concentration values. These samples were taken over 7- to 8-hr collection periods. Shorter sampling times decrease the sensitivity for long-lived emitters. Even higher levels of contamination can be masked in areas where high radon-thoron daughter levels are common. For example, under the conditions of Table 14.1, an allowable level of ^{90}Sr could be hidden within the range of variation if the backgrounds were ten times those in Table 14.1. The sensitivity would also be reduced if a prolonged condition of poor ventilation gives rise to abnormally high background levels.

A much more reliable indication of a hazardous release is given by the so-called first-count factor.

2. First-count Factor

A quick indication of a possible airborne hazard can often be based on the ratio of the activities of beta emitters to the activities of alpha emitters on the filter, determined as quickly as possible after sampling:

$$\Gamma = \frac{\mathrm{dpm/m^3}(\beta\gamma)}{\mathrm{dpm/m^3}(\alpha)} . \tag{14.10}$$

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 beta emitter is signaled and conversely, if the ratio changes significantly downward, the release of an alpha emitter is signaled.

A capsule air-sampling history of N kennel is given in Figs. 14.8 and 14.9. This kennel has been used to house dogs injected with ^{137}Cs. The left half of Fig. 14.8 shows chronologically the daily variation in the factor during a period when recently injected dogs were being housed in the kennel. The right half shows the factor for the same location some $2\frac{1}{2}$ years after the last injection. The same data are grouped in histograms in Fig. 14.9.

During the latter period, the kennel may be regarded as a clean area. This is evident from the very small variation in the first-count factor (2.11 ± 0.23). During this same 6-month period, the alpha and beta concentrations each varied by a factor of 30.

By contrast, the first-count factor varies widely during the earlier period, and its distribution shows virtually no central tendency.

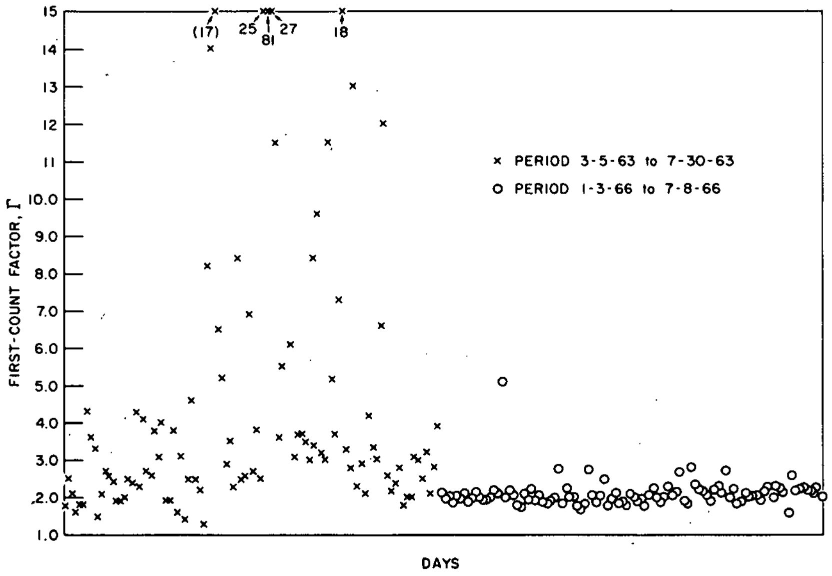

Fig. 14.8. First-count Factors from N Kennel Air Samples in Chronological Order

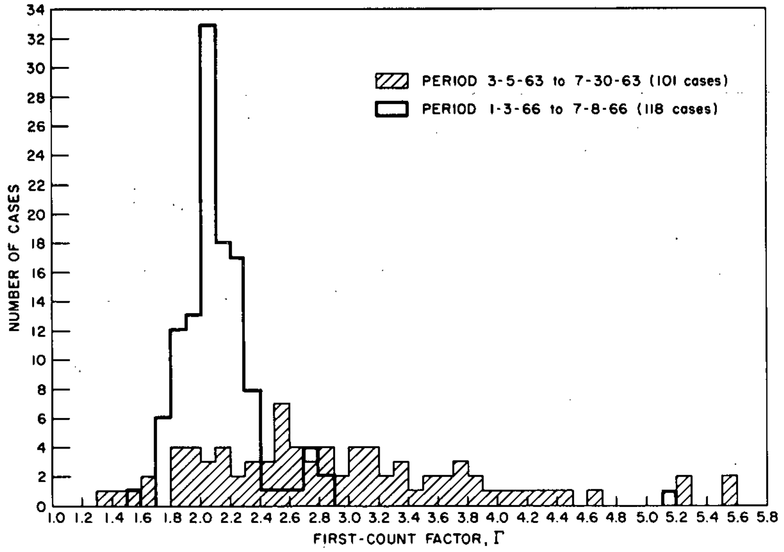

Fig. 14.9. Histograms of N Kennel First-count Factors (Data of Fig. 14.8)

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 alpha self-absorption factor. It will likely vary from system to system and is not terribly important in itself. What is important is that the number for a given counting system be well known for the condition when only naturally-occurring air contamination is present. This number, $\overline{\Gamma}$, is obtained by having an air-sampling "history" of the area.

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 beta or alpha activity. It allows the concentration to be estimated using the known mean value of the first-count factor, $\overline{\Gamma}$, for a clean area.

First, because of its strong central tendency, we assume that

$$\overline{\Gamma} = \frac{A^{\circ}_{N\beta}}{A^{\circ}_{N\alpha}}, \tag{14.11}$$

although, strictly speaking,

$$\overline{\Gamma} = \frac{1}{n}\sum_{i}^{n}\left(\frac{A^{\circ}_{N\beta_i}}{A^{\circ}_{N\alpha_i}}\right) = \left(\overline{\frac{A^{\circ}_{N\beta}}{A^{\circ}_{N\alpha}}}\right). \tag{14.12}$$

We also define

$$\Gamma \equiv \frac{A^{\circ}_{\beta}}{A^{\circ}_{\alpha}}, \tag{14.13}$$

where

Γ is the first-count factor obtained for a particular sample,

$A^{\circ}_{N\beta}$ is the natural beta activity on the filter immediately after sampling,

$A^{\circ}_{N\alpha}$ is the natural alpha activity on the filter immediately after sampling,

A°_{β} is the total beta activity on the filter immediately after sampling,

and

A°_{α} is the total alpha activity on the filter immediately after sampling.

Now, if an alpha release occurs during sampling,

$$\Gamma = \frac{A^{\circ}_{\beta}}{A^{\circ}_{N\alpha} + A^{\circ}_{c\alpha}} = \frac{A^{\circ}_{\beta}}{\frac{A^{\circ}_{N\beta}}{\overline{\Gamma}} + A^{\circ}_{c\alpha}} \tag{14.14}$$

and the activity on the filter due to this contaminant is

$$A^{\circ}_{c\alpha} = A^{\circ}_{\beta}\left(\frac{1}{\Gamma} - \frac{1}{\overline{\Gamma}}\right), \tag{14.15}$$

since in this case, $A^{\circ}_{\beta} = A^{\circ}_{N\beta}$.

If we further assume that the contaminant is long-lived compared with the duration of sampling,

$$A^{\circ}_{c\alpha} = C_{c\alpha} f t_s, \tag{14.16}$$

and

$$\frac{C_{c\alpha}}{A^{\circ}_{\beta}} = \frac{1}{f t_s}\left(\frac{1}{\Gamma} - \frac{1}{\overline{\Gamma}}\right). \tag{14.17}$$

Similarly, we find that

$$\frac{A^{\circ}_{c\beta}}{A^{\circ}_{\alpha}} = \Gamma - \overline{\Gamma}, \tag{14.18}$$

or that

$$\frac{C_{c\beta}}{A^{\circ}_{\alpha}} = \frac{1}{f t_s}\left(\Gamma - \overline{\Gamma}\right). \tag{14.19}$$

Equations 14.15 and 14.18 are plotted in Fig. 14.10 for the conditions of Fig. 14.9 ($\overline{\Gamma}$ = 2.1) and may be used to estimate C_c.

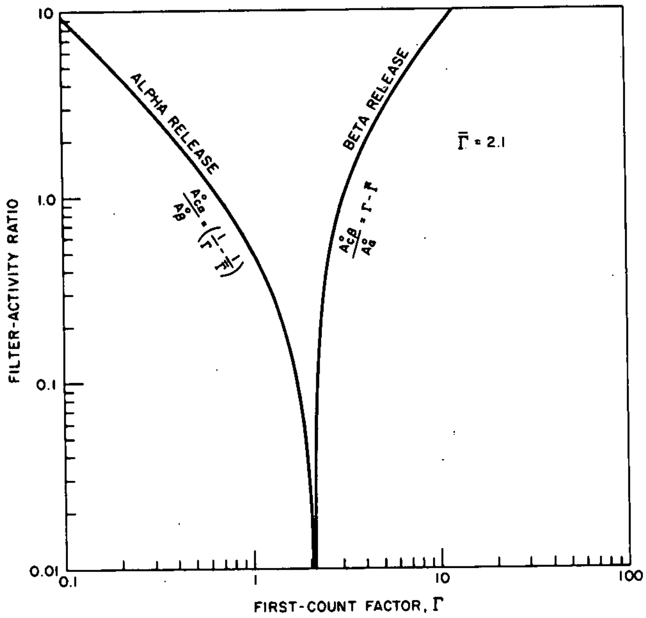

Fig. 14.10. Filter-activity Ratio vs. First-count Factor

Table 14.2 gives an idea of the sensitivity of this scheme for quickly estimating a release of activity. The values of Γ expected after sampling an allowable concentration of ^{90}Sr (660 dpm/m^3) or of ^{239}Pu (4.4 dpm/m^3) are listed for various times and several different equilibrium values of radon-daughter activity on the filter. $\overline{\Gamma}$ is assumed to be 2.1, and the sampling rate is taken as 20 m^3/hr.

It is seen that sensitivity improves with increased sampling time and with decreased natural background. Values of Γ between about 1.7 and 2.6 fall within the normal histogram for Γ in Fig. 14.9. Thus, it is apparent that this scheme is not reliable for estimating allowable concentrations of the most hazardous alpha emitters except under the restricted conditions of low background and/or long sampling times. For the unfavorable situation, no quick estimate is possible and quantitative information must await a more leisurely analysis of the air sample.

Table 14.2

CALCULATED VALUES OF Γ AFTER SAMPLING ALLOWABLE CONCENTRATIONS OF ^{239}Pu OR ^{90}Sr FOR SEVERAL EQUILIBRIUM β ACTIVITIES OF RADON DAUGHTERS ON THE FILTER

Assumed $\overline{\Gamma}$ = 2.1 and f = 20 m^3/hr

$A_{\text{Equilibrium}}$, dpm	Sampling Time, min			
	60	120	240	480
	Alpha Contaminant, Γ			
10,000	~2.04	~2.02	~1.95	1.82
5,000	1.98	1.94	1.82	1.62
1,000	1.66	1.50	1.20	0.85
500	1.38	1.16	0.85	0.52
	Beta Contaminant, Γ			
10,000	5.8	7.6	13	23
5,000	21	29	55	108
1,000	39	57	108	213
500	77	111	213	424

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.

The composite half-life of the radon daughters is about 35 min.(33) If the filter is counted about 6 hr postsampling, their 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

$$A_1 = A_{Th}(t_1) + A_c(t_1). \tag{14.20}$$

Counting again at a later time, t_2, we get

$$A_2 = A_{Th}(t_2) + A_c(t_2). \tag{14.21}$$

Now,

$$A_{Th}(t_2) = A_{Th}(t_1)\, e^{-\lambda_{Th}(t_2 - t_1)}, \tag{14.22}$$

and

$$A_c(t_2) = A_c(t_1)\, e^{-\lambda_c(t_2 - t_1)}. \tag{14.23}$$

We can solve for the unknown to obtain

$$A_c(t_1) = \frac{A_2 - A_1\, e^{-\lambda_{Th}\Delta t}}{e^{-\lambda_c \Delta t} - e^{-\lambda_{Th}\Delta t}}, \tag{14.24}$$

where

$$\Delta t = t_2 - t_1.$$

From here we can use Eq. 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 hr, which is about the composite half-life (10.64 hr) of the thoron daughters. The half-life of the contaminant should be considerably longer than 10.6 hr so that the denominator in Eq. 14.24 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),

$$A_c(t_1) = \frac{A_2 - A_1 e^{-\lambda_{Th}\Delta t}}{1 - e^{-\lambda_{Th}\Delta t}}, \tag{14.24a}$$

and

$$A_c(t_1) = A_c(t_s) = \epsilon C_c f t_s. \tag{14.25}$$

That is, the contaminant does not undergo significant decay on the filter during sampling or analysis, a condition that applies to almost all the isotopes of real concern for usual sampling times (less than 8 hr).

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:(34)

$$\frac{A_c(t_2)}{A_2} = \frac{R - \frac{A_1}{A_2}}{R - 1}, \tag{14.26}$$

where

$$R = \frac{A_{Th}(t_1)}{A_{Th}(t_2)} = e^{\lambda_{Th}\Delta t}. \tag{14.27}$$

In Fig. 14.11, $A_c(t_2)/A_2$ 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.

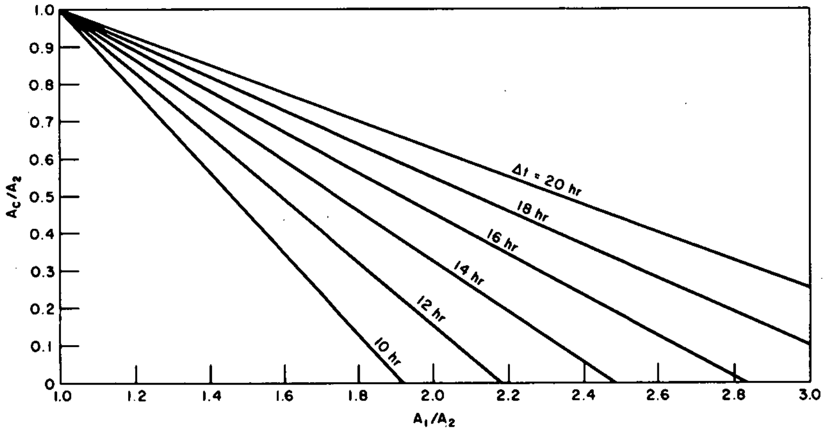

Fig. 14.11. Contaminant Activity Fraction vs Activity Ratio

Example: A filter sample is taken for 4 hr at 20 m^3/hr; ^{239}Pu is the suspected contaminant. The following disintegration rates were obtained:

A	t		
	t_s = 4 hr	t_1 = 10 hr	t_2 = 20 hr
A_β	10,000	1,000	520
A_α	5,400	1,200	1,000

Estimate the concentration of ^{239}Pu. Assume that Fig. 14.9 represents the normal situation in the area.

From the first counts, we get Γ = 1.86. Since this is within the expected range for Γ, no real information is obtained. However, from the later counts, we get

$$A_{1\alpha}/A_{2\alpha} = 1.2,$$

and from Fig. 14.11 we get

$$\frac{A_c(t_2)}{A_2} = 0.78;$$

therefore

$$A_c(t_2) = (0.78)(1000) = 780 \text{ dpm}.$$

Thus, we estimate the average concentration during sampling as

$$C_{Pu} = \frac{A_{Pu}}{ft_s} = \frac{780 \text{ dpm}}{20 \frac{m^3}{hr} \times 4 \text{ hr}} \cong 10 \text{ dpm}/m^3.$$

4. Concentration of Short-lived Emitters

As the half-life of the contaminant approaches that of the thoron daughters (10.6 hr), calculations using Eq. 14.24 become increasingly unreliable because the difference,

$$e^{-\lambda_c \Delta t} - e^{-\lambda_{Th} \Delta t},$$

becomes too small.

Fortunately, short-lived isotopes have MPC 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 Eq. 14.7. Decay during sampling must be accounted for, since concentration might otherwise be seriously underestimated.

For very short-lived isotopes ($T < 35$ 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.

I. 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 is beyond the control of the producer. Moreover, AEC regulations(35) 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.

AEC regulations also state the conditions of release that require prompt reporting.(36) Any release that exceeds 5000 times the applicable concentration, when averaged over 24 hr, must be reported immediately. Any release that exceeds 500 times the allowed value, when averaged over 24 hr, must be reported to the AEC within 24 hr.

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 also desirable 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 would be a detector such as 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 μCi/cc in the stack effluent if the radioisotopes 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 gamma emitters. Absorption curves in aluminum and comparison with appropriate standards would help identify beta emitters. A charcoal filter could be looked at for iodines or noble gases.

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 continuous air monitors, are often homemade instruments assembled for a particular job. They may be as elaborate or as simple as desired.

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SECTION 15 - REACTORS

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 field.

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 elements used as targets, Fermi and his group tried uranium. In most of the 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 $^{238}_{92}U$, 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 the problem of separating these products. In 1938, using the carrier technique, Curie and Savitch obtained a radioactive substance with the chemical properties of lanthanum. This seemed to upset the proposed view that all the products were transuranic. Then, Hahn and Strassman showed that the activity in question 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 confirmed this view and led to an estimate of the cross section for the process. Also, at this time, many investigators noted that more than one neutron was emitted during a fission process. 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, much of the research work after the summer of 1940 was kept secret. The groundwork leading to military usage was to first obtain a self-sustained chain of fission events that could be controlled. Then, the details could be worked out to result in a fission chain in a very short time. This uncontrolled chain reaction would result in a tremendous release of energy.

By the middle of 1941, Fermi and his group at Columbia had constructed a graphite-uranium cubical lattice. The purpose of this device was to study the multiplication of neutrons in the system. Late in 1941, Compton was placed in charge of this project, and the work 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 summer of 1942, Fermi's work had progressed far enough so that the self-sustained reaction was within sight. At this point, 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 as long as fuel is available. Although a reactor in its final form is more complex than simply an array of fuel, the heart of the operation is the fission process in the fuel. Let us look at this process in the fuel substance ^{235}U (see Fig. 15.1). A thermal neutron, n_{th}, is captured by the ^{235}U nucleus, and this can lead to two results. First, the uranium may just become a ^{236}U atom. This nonfission type capture occurs about 15% of the time.

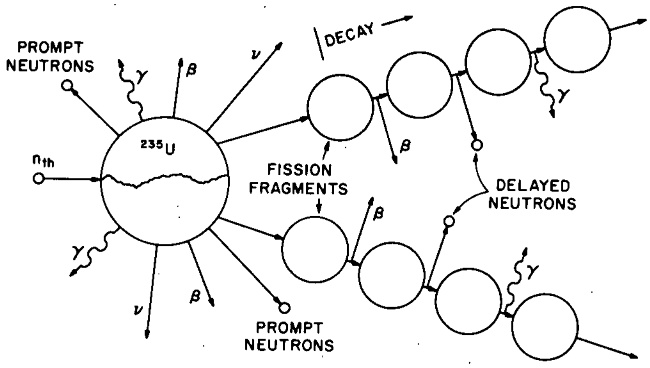

Fig. 15.1. Fission Process in ^{235}U due to a Thermal Neutron

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, gamma, as well as beta and neutrinos (ν), are also emitted. These are referred to as the prompt beta and gamma. About 9 MeV/fission of prompt betas and about 7 MeV/fission of prompt gamma are emitted in ^{235}U fission.(4) The fission fragments come off with velocities of about 10^9 cm/sec and may have ranges up to 1 in. in air.(1) Since the fragments have too many neutrons to be stable, 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 neutrons 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 can be roughly classified into six groups, which can be identified by an assigned half-life.(4) The delayed groups include those neutrons that are emitted at times $>10^{-3}$ sec after the atom splits.

During this decay phase of the process, more beta and gamma are also given off by the fragments as they decay to stable systems. The entire fission process in this case results in a release of about 200 MeV/atom. Of this, the fission fragments carry off about 165 MeV.(1,4) Note that this is the energy released in only one fission. This implies that we can obtain a tremendous amount of energy from just a small amount of uranium since it would contain so many atoms. For example, let us find the energy release in the complete fission of 1 gm of ^{235}U:

$$\text{Number of } {}^{235}\text{U atoms} = \frac{m}{A}(6.025 \times 10^{23})$$

$$= \frac{1}{235}(6.025 \times 10^{23}) = 2.56 \times 10^{21}.$$

The total energy released, E_t, will be the product of the number of atoms times 200 MeV/atom:

$$E_t = 2.56 \times 10^{21} \text{ atoms} \left(200 \frac{\text{MeV}}{\text{atom}}\right) = 5.12 \times 10^{23} \text{ MeV}$$

$$= 5.12 \times 10^{23} \text{ MeV} \left(3.83 \times 10^{-14} \frac{\text{cal}}{\text{MeV}}\right) = 1.96 \times 10^{10} \text{ cal.}$$

A gram of high explosive (TNT) releases about 1000 cal of energy.(2) A ton of TNT would then release about 9.1×10^8 cal. Then, the energy released in the fission of 1 gm of ^{235}U would be equivalent to

$$\frac{1.96 \times 10^{10} \text{ cal}}{9.1 \times 10^8 \text{ cal/ton TNT}} \approx 21 \text{ tons of TNT.}$$

When we have a large amount of uranium that 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. Fission Type

The fission process is not restricted solely to slow-neutron captures in ^{235}U. Other substances may also be fissioned by slow neutrons. Some of these may be fissioned by fast neutrons or other agents as well. Even gamma rays have been used to produce photofission.(1) Some substances, such as ^{238}U, are fissioned by fast neutrons but not by thermal neutrons. Many of the 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.(4,5)

We can distinguish between those 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 $Z \geq 73$ under the proper circumstances. In most of these materials, though, the fission process cannot be made self-sustaining. For this reason, only ^{233}U, ^{235}U, and ^{239}Pu, in which the chain reaction can be maintained, are of value for reactor use.

2. Fission Yield

The term fission yield is used to express the per cent 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 of the fission process 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 Fig. 15.2. Note the existence of 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%.(1,4) Mass numbers are used since the products in these groups are unstable and emit beta. Since there are about three beta decays, on the average, for each fragment, the atomic number varies greatly, but the mass number remains comparatively constant.

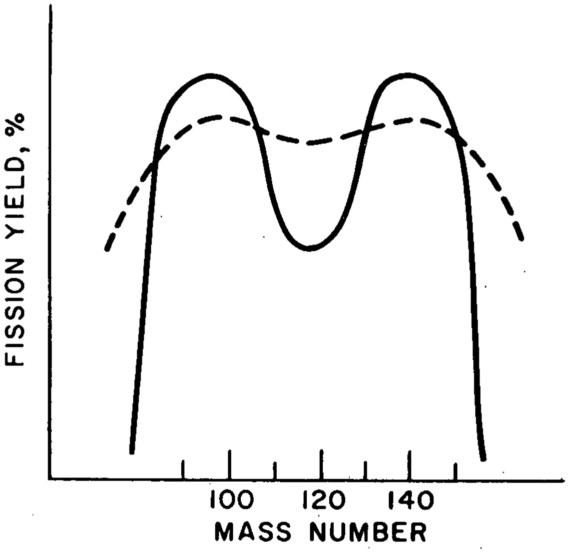

Fig. 15.2. Fission Yield in ^{235}U. (Adapted from ANL-5800.(4))

When the two unequal masses, m_L and m_H, are released in fission, the energy carried off by each is different. We can apply the principles of conservation of energy and momentum to this problem. Then we obtain

$$\frac{E_H}{E_L} = \frac{m_L}{m_H}, \tag{15.1}$$

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, if we use fast fission (14-MeV neutrons), symmetric fission occurs much more frequently. This is shown as the dotted curve in Fig. 15.2. At higher neutron energies, we can even obtain just a single peak.

3. Fission Rate

The rate at which the fissions occur depends upon many factors, which include the fuel properties as well as 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 flux density ϕ(n/cm^2-sec), the fission rate R (fissions/sec) in a given fuel substance is

$$R_f = \phi\sigma_f N, \tag{15.2}$$

where N is the total number of atoms of the fuel exposed to the neutrons. Suppose a 2-gm sample of ^{235}U is irradiated by a constant thermal neutron-flux density of 10^{13} n/cm^2-sec. Then, if σ_f = 580 barns, what is the fission rate in the ^{235}U sample?

$$R_f = \phi\sigma_f N = \frac{\phi\sigma_f m N_A}{A}$$

$$= 10^{13}(580 \times 10^{-24})\frac{(2)}{235}\,6.025 \times 10^{23} = 2.974 \times 10^{13}\,\frac{\text{fissions}}{\text{sec}}.$$

The total number of fissions in one day would then be

$$\frac{\text{fissions}}{\text{day}} = 2.974 \times 10^{13}\,\frac{\text{f}}{\text{sec}}\left(8.64 \times 10^{4}\,\frac{\text{sec}}{\text{day}}\right) = 2.57 \times 10^{18}.$$

If we are dealing with a region in which the neutron-flux density is uniform, then the fission rate can be expressed as

$$R_f = \phi\Sigma_f V, \tag{15.3}$$

where Σ_f is the macroscopic fission cross section (= $\sigma_f\rho N_A/A$), in cm^{-1}, and V is the volume of the region in cm^3.(6)

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.(7) From the fission process itself, neutrons are made available in each event. If we can ensure that at least one neutron resulting from an event causes another fission event, then the chain reaction is possible. On the other hand, we must have control over the system or the process could result in too great an energy release. The complex form a reactor may take merely 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 do exist, most reactors contain the following components:(1,7,8)

(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 does not absorb them. Light elements make good moderators since the neutron loses a larger fraction of its energy in each elastic collision with light elements than it does with heavy elements. Water, D_2O (heavy water), beryllium, and carbon have been used with success in reactor systems.

(3) Coolant. The coolant flows through the core to remove the heat produced by the fission process. That is, as the fission fragments lose their kinetic energy, this energy generally appears in the form of heat. For reactors that operate at powers greater than 1 Watt, cooling is required. Many systems have used such substances as water, liquid metal (NaK), and gases as coolants.

(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 gamma, massive shielding may sometimes be required.

2. Critical Size

To maintain the chain reaction in the system, we must conserve the neutrons produced in fission. When fission occurs, neutrons are emitted, usually two or three (2.54 on the average for ^{235}U fission(7)). These are emitted as fast neutrons, and as they move through the core many events can occur. That is, 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-flux density and the cross section for that type of event. Since other processes compete with the fission events for neutrons, we must be able to supply a large number of neutrons to keep the chain going. That is, we must have a certain amount of a fissionable substance present before a self-sustained chain reaction can take place. This amount of material is called the critical size. The critical size for any system 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.

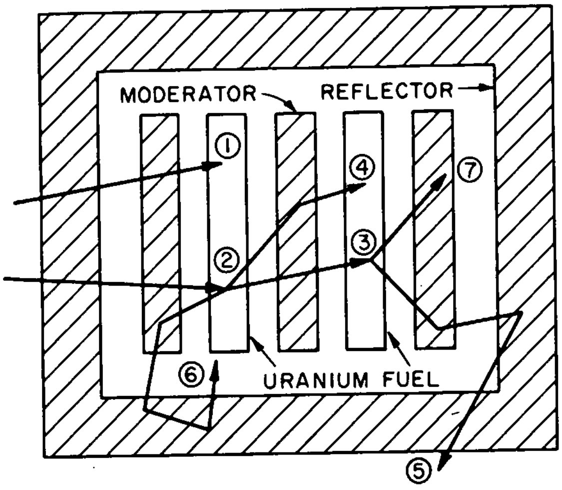

Fig. 15.3. Nuclear Assembly for Thermal Fission

Suppose we have an assembly of uranium fuel, moderator, and reflector as shown in Fig. 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 nonfission capture, which leads to ^{236}U ($T_{1/2} = 2.39 \times 10^7$ yr; 4.5 MeV alpha) 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 then 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 1 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:

$$^{238}U + {}^{1}n \rightarrow {}^{239}U \rightarrow {}^{239}Np + \beta^- \rightarrow {}^{239}Pu + \beta^-.$$

If the process of resonance capture occurs, no neutrons are released (4).

The neutrons may also 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). In the course of slowing down, the neutron may also be absorbed by some impurity atom (7). Substances such as boron, cadmium, or rare-earth impurities have high capture cross sections for neutrons. This type of capture also 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 then 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 we were to add 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 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.(8) Suppose a thermal neutron is captured in the ^{235}U fuel and produces fission. This leads to the emission of a number of fast neutrons. On the average, this number is 2.54 for ^{235}U fission. In this case, though, our neutron could have been captured without producing fission. Thus, we can 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, $\eta < 2.54$. On the average, we would 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 1 MeV and most of the neutrons are quickly reduced to energies below this in just a few collisions. At this point, we would 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,(6) resonance absorption in ^{238}U can occur. As the neutrons are reduced in energy, some may be lost in capture. Those that are not lost are further 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 as well as the type of moderator substance. Usually, some neutrons are captured so that p is less than unity. In this case, we would 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 <u>four-factor formula</u>:

$$k_\infty = \eta\epsilon pf. \tag{15.4}$$

This factor, k_∞, gives the ratio of the number of neutrons present in any generation to that in the preceding generation. 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 did not mention 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 of the neutrons diffuse out from the surface of the reactor and are lost. This "leakage" reduces the actual multiplication in the system by a factor L. This nonleakage factor is the fraction of neutrons that do not escape. We can define an effective multiplication factor, k_{eff}, for a finite system as $k_{eff} = k_\infty L$, in which $L < 1$. The value of the nonleakage 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.

We can describe the operation of the system in terms of the value of k_{eff}. If $k_{eff} = 1$, the reactor is just critical. In this case, 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.

The design problem that must be solved to obtain a reactor is to adjust the parameters so that k_{eff} is slightly greater than unity. This often requires extensive and complex mathematical computations to arrive at the correct design features. We must be able to adjust the multiplication available so that the neutron-flux density increases.(9) This allows the reactor power to be brought up to a desired level. Then, for steady-state operation, k_{eff} is adjusted so that it is unity. At this point, the power levels off since the number of neutrons remains constant in each generation.

5. Reactor Control

The problem of reactor control can be approached from the standpoint of the change in the neutron population in the system. Let us assume that k_{eff} can be made greater than one in our system 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.

Given a total N neutrons at the start of a generation, the increase in neutrons is $N\Delta k$ at the end of this generation. Let τ_0 be the generation time, that is, the average time, in seconds, between successive neutron generations. The total gain in neutrons per second is then $N\Delta k/\tau_0$. Thus, the rate of change of neutrons with time is given by

$$\frac{\Delta N}{\Delta t} = \frac{N\Delta k}{\tau_0}. \tag{15.5}$$

If we assume a number N_0 neutrons at time $t = 0$, Eq. 15.5 can be integrated to give

$$N = N_0 e^{\Delta k t/\tau_0} = N_0 e^{(k_{eff} - 1)t/\tau_0}, \tag{15.5a}$$

where N is the total number of neutrons at time t. In a thermal reactor, τ_0 is about 10^{-3} sec for the prompt neutrons emitted in fission. In a reactor that has $k_{eff} = 1.005$, let us find the increase in neutrons after 2 sec for each initial neutron;

$$N = N_0 e^{\Delta k t/\tau_0} = e^{0.005(2)/0.001} = e^{10} = 22{,}026 \text{ neutrons for each initial neutron.}$$

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, even though they make up less than 1% of the total neutrons, allows for better reactor control. In this case, since there is a time lag before the delayed neutrons are emitted, they increase the generation time τ of all fission neutrons. The mean lifetime for the delayed groups in ^{235}U fission is 0.082 sec.(9) Thus, the generation time for all fission neutrons becomes

$$\tau = 0.001 + 0.082 = 0.083 \text{ sec.}$$

In the same example as before,

$$N = N_0 e^{\Delta kt/\tau} = e^{0.005(2)/0.083} = e^{0.12} = 1.13 \text{ neutrons.}$$

In this case, the increase is not so very rapid, and we have time in which to achieve control.

a. Reactor Period

The concept of 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-flux density to increase by the factor e. Using the previous relationship for the number of neutrons at any time, we have

$$\frac{N}{N_0} = e^{\Delta kt/\tau} = e^{\Delta kT/\tau}, \tag{15.6}$$

when t = T. From the reactor period definition, when t = T,

$$\frac{N}{N_0} = e^1. \tag{15.6a}$$

From Eq. 15.6,

$$e^{\Delta kT/\tau} = e^1, \tag{15.6b}$$

so that

$$\frac{\Delta kT}{\tau} = 1, \tag{15.6c}$$

and

$$T = \frac{\tau}{\Delta k} \text{ sec.} \tag{15.6d}$$

From the standpoint of control of the reactor, the period must not be too short. If we use our example of $\Delta k = 0.005$ and $\tau = 0.083$ sec, then $T = 0.083/0.005 = 16.6$ sec. This period allows ample time for reactor control. Now, suppose the prompt-neutron generation time $\tau_0 = 10^{-3}$ sec is used; then $T = 0.001/0.005 = 0.2$ sec. In this case, the increase is so rapid that safe control of the reactor would be difficult. 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.

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. This change affects the prompt-neutron population immediately but does not affect the delayed group immediately. Such an effect upon the neutron multiplication can be described by the reactivity, ρ, which is defined by

$$\rho \equiv \frac{\Delta k}{k_{eff}}, \tag{15.7}$$

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 $\rho \cong \Delta k$. 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. Then for any $\Delta k < \beta$, the delayed neutrons control the rate of rise in the neutron population. This is known as the delayed critical condition. When $\Delta k = \beta$, the chain reaction is maintained by the prompt neutrons alone (prompt critical condition). For $\Delta k > \beta$, the multiplication of the neutron-flux density occurs quite rapidly; therefore this state should be avoided.

The reactivity value of a given reactor component may thus be used as a measure of its effect on the system. One common way in which this is done is to express the reactivity value in terms of the prompt critical condition. That is, if $\rho/\beta = 1$, the system is prompt critical and the value of the reactivity is one "dollar." A reactivity value expressed as "cents" occurs when $\rho/\beta < 1$, for a given reactivity value. In this sense, every part of the reactor has a "worth" insofar as it affects the multiplication. The changes in the control of the system can be based upon the study of reactivity values of the various system parts.

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.(1,2,6,8)

Two of the foremost built-in features that are used to advantage in many 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 design features tend to counteract the increase.

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 then more than the minimum needed for the chain reaction. A neutron absorber may also 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 poisons, and ^{135}Xe is the most troublesome.(2) Of the others, ^{149}Sm is next in importance.(6)

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. During steady operation, the ^{135}Xe concentration reaches an equilibrium value that depends upon the flux density in the system. When the system is shut down, the decay chain of ^{135}Te leads to an increase in ^{135}Xe, which reaches a peak value in about 11 hr.(10) Unless the system has been designed with sufficient reactivity, this effect hinders reactor start-up shortly after shutdown.

In many instances, the control system proper refers to the mechanical systems. For thermal reactors, the control is accomplished by use of rods containing neutron-absorbing elements (Cd, B, Hf).(10) For fast reactors, the system may contain control rods, movable fuel rods, or a movable reflector. Control rods can be grouped as 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 worth. Thus, by insertion or withdrawal, we can affect the neutron multiplication.

In addition to the normal control system, we may also have emergency systems, which allow for a quick scram. In some systems, the moderator can be quickly dumped from a core vessel to aid in quick 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.

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 so many 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 the material does not expand or buckle out of shape. Over and above this, the substance should resist corrosive agents as well as 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 system involves contributions from many sources. For this reason, the subject is much too complex to treat in such a short outline. The subject of shielding design alone requires extensive and complex calculations as well as attention to the material properties.(11,12) Information concerning the engineering aspects of reactor design is contained in Refs. 13-15.

7. Power Level

Relating the neutron-flux density to the power level is not easy, except for certain special designs.(2,6) 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 10^7 ergs/sec. The energy, in ergs, per fission is

$$200\left(\frac{\text{MeV}}{\text{fis}}\right)\ 1.6 \times 10^{-6}\left(\frac{\text{ergs}}{\text{MeV}}\right) = 3.2 \times 10^{-4}\left(\frac{\text{ergs}}{\text{fis}}\right).$$

For a power level of 1 W, the fission rate is

$$\text{fission rate} = \frac{10^7\left(\dfrac{\text{ergs}}{\text{sec-W}}\right)}{3.2 \times 10^{-4}\left(\dfrac{\text{ergs}}{\text{fis}}\right)} = 3.1 \times 10^{10}\left(\frac{\text{fis}}{\text{sec-W}}\right). \qquad (15.8)$$

Suppose we calculate the total atoms per day that are fissioned in a 1-MW (10^6 W) reactor operating for a full 24 hr:

$$1\ \text{MW} = 10^6\ \text{W} = 3.1 \times 10^{10}\left(\frac{\text{fis}}{\text{sec-W}}\right)\ 10^6\ \text{W} = 3.1 \times 10^{16}\left(\frac{\text{fis}}{\text{sec}}\right).$$

Then,

$$3.1 \times 10^{16}\left(\frac{\text{atoms}}{\text{sec}}\right)\ 8.64 \times 10^{4}\left(\frac{\text{sec}}{\text{day}}\right) = 2.68 \times 10^{21}\ (\text{atoms/day}).$$

From page 305, we found that 1 gm of ^{235}U consists of 2.56 x 10^{21} atoms. Therefore

$$\frac{2.68 \times 10^{21} \text{ atoms/day}}{2.56 \times 10^{21} \text{ atoms/gm}} \approx 1.05 \text{ gm/day of } ^{235}\text{U}.$$

Thus, a 1-MW reactor burns about 1 gm of fuel per day.

8. Reactor Types

We can class most common reactors into four main categories: research, breeder, power, and isotope production.(1,2,6,8) Research reactors may also be used as a system for training. The features of research reactors are discussed in a general way in Ref. 7. 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.(1) Power reactors convert the fission energy to electrical energy, which is then used to supply power.(1,6) The graphite reactor at Oak Ridge, which is now shut down, was used as a source of radioisotope production.(9) Reference 16 describes some aspects of several Argonne reactors.

D. 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.

1. Leakage Radiation

The fission process gives rise to a number of products: fission fragments, beta particles, gamma rays, and fast neutrons.(17-19) The decay of the fragments results in the fission-product chains, which produce more beta and gamma as well as the delayed neutrons. The beta 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, gamma 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 page 304). Capture in some other substance may result in capture gamma rays (0.5-10 MeV) or activation of the substance. The latter process may then lead to the emission of gamma 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 gamma rays. In all neutron interactions that lead to gamma 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 vary in energy from about 0.5 to 6.5 MeV.(17) 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.

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 flux densities 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 broken. 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.(18) 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 gamma 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.

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.

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.(18) The coolant passing through the system may also acquire activated scale, which can also then deposit 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-flux density 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 product is ^{16}N, which has a $T_{1/2}$ = 7 sec. Since these are gases, they tend to escape from the system, but any delay results in significant decay, which reduces the problem.

Some systems (homogeneous or liquid fuel) have a combined fuel-moderator-coolant solution. In these, the liquid is contained in a closed-loop system so that it may be pumped from the core to a heat exchanger. The fluid is highly radioactive, and the piping system may become activated from the delayed neutrons emitted from the fissioned fuel. In this type of arrangement, the entire system may require extensive shielding.

In a closed system that contains a heat exchanger, we often find ion-exchange columns or other systems that attempt 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.

3. Sources 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. 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.(20)

For an air-cooled system, the products are radioisotopes 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 gamma with a $T_{1/2}$ = 110 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.

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 developes 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.

When water is the coolant-moderator, the products include radioactive nitrogen, oxygen, fluorine, and tritium.(18,20) 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 an added problem.(21) Tritium, which has a $T_{1/2}$ = 12.26 yr and emits an 18-keV beta, is formed in the reaction $^2H(n,\gamma)^3H$. The gas itself does not add greatly to the amount of airborne activity 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 (~0.1%) by the body, but as an oxide (H_2O vapor), almost all of that taken in remains in the body. 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. Because of the weak beta emitted by this radioisotope, the monitoring of tritium presents a further problem.(21,22)

In organic-cooled reactors, the hydrogen and carbon present result in activated gases in which ^3H and ^{14}C are the main radioisotopes. For liquid-metal coolants, the source of radioactive gas products is not the coolant itself, but the gas blanket above the liquid-metal.(20)

In liquid-fuel systems, the fission gases and volatile products are released in the fluid. These must be removed from the closed system during operation. Regardless of how well the system is designed, small amounts of these products gain access to the atmosphere. If 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 systems, the water moderator is 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.(20)

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-min range, and (4) solubility to some extent in the coolant.(23)

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 (^{138}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. Some nonvolatile fission products and perhaps some of the fuel may also be released with the gases.(20,25,26) For such a release, common practice has been to assume a loss of 100% of the noble gases (xenon and krypton), 50% of the volatiles (iodine and bromine), and 1% of the solids (barium and strontium). Recent studies tend to view these figures as too conservative.(27)

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 material, 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-flux density 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.

With a large fission-product release, we can also expect some of these contaminants to deposit out on 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 radioisotopes that may be found. For a given system, though, certain products occur more frequently. In general, identification of the products is not easy.

E. Radiation Surveys

During startup of a reactor, a detailed study should be made to check the integrity of the shield design.(17,19) This survey should be performed at low 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 gamma rays are the more important radiations, the survey should be designed to obtain the levels of both neutrons and gamma 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 include the levels from each of these neutron components. In the survey, attempt to find leaks due to voids, streaming, scattering, and diffusion through the reactor structure.(19)

We should also monitor at new or unusual power levels or when fuel is removed or changed. The removal of any part of the shield, such as when the reactor top is opened or removed, also requires a survey. If new experiments are set up, or beams are brought out of the system, radiation levels should be checked. Also, surveys should be performed when samples are put into or removed from the reactor.

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.

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.

F. Reactor Survey Instruments

The instruments needed to monitor a reactor should include devices that can measure the neutron and gamma fields. Many of these portable devices have been described in Section 12. Portable devices such as G-M survey meters and ion chambers can be used for surveying gamma rays. Neutron-flux densities can be measured by BF_3 proportional counters or "long" counters. For dose-equivalent measurements of neutron levels, we can use the 10-in.-diam polyethylene sphere device discussed in Section 12 (page 234). Other instruments that can be applied have also been discussed 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. At ANL, HV70 or Millipore filter samples have been 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.(22) This method seems to be about the most satisfactory for the difficult task of monitoring tritium. For suspected iodine release, charcoal filters have been the most successful means of detection.(25)

Smear samples are taken at ANL with either HV70 or Millipore filter papers. Almost any substance may be used for beta-gamma activity. For alpha 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 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.

Smear samples can be counted for activity levels and also subjected to gamma spectrometry tests. The latter method allows for identification of the source, provided the number of activities present is not too large. If a large number of contaminants are present, the gamma spectrum may be too complicated to allow easy analysis. In some cases, alpha spectrometry can be of value, though alpha is seldom a problem around reactors.

G. 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 often used to provide data on the radiation concentration released through the exhaust system. These 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 to provide an audible signal in the event of a release.

H. Nuclear-accident Monitors

A runaway reactor does not approach anything like an atom bomb even under the most adverse malfunctions.(28,29) 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 steam explosion, which may eject parts of the core. In these instances, severe damage to the reactor vessel occurs. Over and above these excursions, the presence of fissionable material in the core or in storage can sometimes result in an unwanted criticality. These nuclear accidents are less severe in that no blast can occur. However, these effects can result: (1) high radiation levels, which can be lethal for a considerable distance; (2) the release of enough heat to melt metals in some parts of the system; and (3) contamination of the nearby area severe enough to present a hazard to personnel for perhaps months.(28)

The criticality incident occurs with no warning so that there is no time to invoke safety measures. That is, the system is subcritical at one instant and then, suddenly, critical. When one of these accidents occurs, nothing can stop the process. In many cases, a short burst of radiation occurs, but then the process is not sustained. We would like to obtain data on the absorbed dose to which a worker has been exposed in one of these bursts. Such data can be of value for a number of reasons.(30)

To obtain information on the absorbed dose, we need a measure of both the neutron fluence and spectrum and the gamma-absorbed dose. This information can be obtained by the use of one of a number of nuclear-accident dosimeters.(30) Since the placement of these units may differ from the location of the exposed worker, we need a measure of the ^{24}Na activity induced in the person's body. By means of the ^{24}Na data, the dose at the dosimeter station can be related to the dose the worker receives.

In principle, the above method is simple and easy enough to apply. In practice, though, an accident can be a complex affair in which many unforseen obstacles are present.(31,32) These factors may delay or impede the dose determination in any given event. For example, the dosimeter may be destroyed or badly contaminated in the incident. Also, the field may be so high that the unit cannot be retrieved easily. In these cases, we may have to find other activated substances that can be used to measure neutron fluence.

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SECTION 16 - HOT CELLS

Many operations involving radioactive substances must be carried out in shielded enclosures called "hot cells" or "caves." Hot cells protect workers from bodily intake of these toxic substances and from exposure to the penetrating rays.(1) The main parts of a cell are the shielding walls, cell liner, services, ventilation system, and, in some cases, a drainage system.(2) Special-purpose cells may also include viewing windows, remote-handling devices (manipulators), hoists, and access ports.

To keep the working areas free of airborne activity, cells may be hermetically sealed or maintained under a negative pressure.(3) 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.(1) 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 pyrophoric, and a fire would produce radioactive smoke. The alpha emitters, being highly toxic, require good containment. However, these substances can be handled in thin-walled 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.

A. 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.(3) It can be obtained in a range of densities - from 2.2 to 5.9 gm/cc. The higher-density concretes are used, in most cases, for monolithic shielding structures.(4) These heavy concretes are more costly than ordinary concrete; however, the added cost is justified by several factors. For example, the thinner walls permit close viewing through windows and reduce space requirements. Also, smaller remote-handling devices, which are cheaper and easier to maintain, can be used.(5)

In the construction of hot cells whose future needs cannot be adequately foreseen, metal boxes with a loose filling substance (steel punchings, sand, ores, etc.) are quite useful. With changes in shielding requirements, it is much simpler to alter this type of structure than a monolithic concrete cell.

Steel tanks filled with water can be used for shielding when the cell contains isotopes that undergo spontaneous fission or when neutrons are being produced from (α,n) processes.(3) The tank wall thickness can be varied if necessary to shield against gamma rays due to neutron capture. Concrete can also be employed for neutron shielding. However, the water content must be adequate to provide the required attenuation.

The choice and thickness of shielding material for a given high-level gamma cell are often based upon a certain amount of gamma activity. The activity is then assumed to be at a point.(4) This is a conservative approach since a finite source is always less intense. The unshielded intensity I (in R/hr) at the position of the worker is approximated by

$$I = \frac{6CE}{r^2}, \tag{16.1}$$

where C is the gamma activity in Curies, E is the energy in MeV/dis, and r is the distance between the source and the worker in feet.

The shield thickness depends upon the attenuation required to reduce the intensity to prescribed limits in the working areas. This attenuation is the ratio of the unshielded intensity to the desired intensity at the position of the worker. It is a function of the composition and density of the shield.

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.(4,6) The value of b varies with gamma 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

$$I = I_0 b e^{-\mu x}, \tag{16.2}$$

where I is the reading at a point outside the shield, I_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.

Buildup factors for point isotropic sources for common shielding substances are given in the references.(8,10,11)

Concrete is an adequate shielding substance for neutrons if the water content is at least 7 w/o. More water increases the neutron attenuation. Some concretes can lose 50% or more of their water during several years.(3) This loss requires the inclusion of safety factors in the original shielding design.

The walls of a hot cell should always be checked for leakage.(4) A high-energy gamma 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.

B. Liners

The decontamination of hot cells is a necessary but time-consuming and costly operation. It often requires more time than the experimental work itself. For ease and speed in this cleanup work, cells are customarily equipped with liners.(5,7) A liner is some type of material that covers the inner surface of the cell walls. There is a choice between a costly liner that does not take up radioactive matter readily and can be cleaned easily (e.g., stainless steel), or cheap substances that can be repeatedly removed and replaced at low cost (paint, lacquer, etc.).

Carbon steel 1/4 to 1 in. thick or stainless steel 1/16 to 1/4 in. thick is normally used to line concrete walls.(7) The seams are welded and made smooth for ease in cleaning. The quality of the weld is quite important. Faulty welds allow penetration of active cleaning fluids, which results in contamination of the radiation shield. In some cases, steel forms have been used when the concrete walls were poured. These were then left in place to serve as an inner liner and outer surface. Window frames, fittings, etc., were then welded to these surfaces.

Disposable plastic sheeting is often used for lining a cell. During the cleanup phase, this liner is folded and discarded along with most of the activity that adheres to its exposed surface.

If it is not feasible to use metal or plastic for liners, multiple layers of solvent-, oil-, and water-resistant paints are used to coat the porous concrete walls.

Recommendations for surfaces and coatings in areas in which radioactive substances are handled are contained in the references.(8,9)

C. Services

Electrical power plus various gas and liquid services are often enclosed within ducts or service tunnels. However, they may be introduced through access ports. These penetrations in the cell wall are arranged in a stepped, zigzag, or curved manner to avoid loss of shielding.(2,7) Any exposed lines, connectors, outlets, etc., must be able to remain intact under the action of corrosive atmospheres. Cells are designed so that these services can be manually controlled on the outside or with ease by remote means within the facility.

D. Ventilation

Ventilating systems can control the temperature and limit the spread of contamination both inside and outside hot cells.(7) Cells are usually maintained at reduced pressure to prevent the escape of contamination into the working areas. When possible, air-flow patterns are set up so that the air is carried to some type of cleaning system. This exhaust air may contain contaminants and therefore must be cleaned before release to the atmosphere. Particulate matter can be kept in the lower portion of the cell by having the air supplied at the top and exhausted near the bottom.(5)

Cleaning can be achieved with filters (for particulate matter), with scrubbers (for particulates and gases), or by adsorption (for gases).(3) 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 air flow is momentarily reversed due to an explosion or fire.(3)

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.(5)

Equipment that disturbs the ventilation because of its size, form, or heat production may be contained in smaller enclosures within the cell.(5,12) These units can then be tied into the cell ventilation system.

An air flow can also be maintained in completely sealed cells. The cell air is pulled through a filter system and recirculated back into the cell.(3)

If there is a loss of normal electrical power, emergency power should be available to maintain cell air flow.(7)

E. Drainage

Hot cells can be supplied with drains to remove liquid wastes.(7,12,13) Floors are sloped to floor drains, which are connected to a waste-collecting system. In some cases, a cell can be equipped with an automatic water-type fire-fighting system. This cell must have ample drainage to prevent the water from reaching doors or access ports that are not watertight.

The waste matter is often routed to large shielded storage tanks. These tanks may have a liquid-level indicator plus a maximum-level alarm. After monitoring, the contents can be sent to retention tanks or to a waste-processing system. This would depend upon the activity and the need for more processing.

F. 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.(4,14)

1. Windows

Liquid or glass windows are used for direct viewing. A liquid window is usually a tank placed in the cell wall and filled with a transparent liquid. It should have about the same shielding ability as the cell wall itself. Water can be used for viewing purposes. It is cheap, does not change color as a result of radiation, and offers good vision through a thickness of 10 ft or more.(5,14) However, a higher-density liquid is generally used to decrease the required thickness.

A solution of about 78% zinc bromide in water has been used widely for liquid windows.(4,5,14) The $ZnBr_2$ solution (ρ = 2.5 gm/cc) is highly transparent, rather stable under irradiation, and relatively cheap. If the solution is exposed to air or radiation, impurities such as iron, which is present in trace amounts, become oxidized and the bromide ion is changed into free bromine. This causes coloration and a corresponding decrease in light transmission. This coloration effect can be retarded by adding a small amount of hydroxylamine hydrochloride ($H_2NOH \cdot HCl$), a reducing agent. This type of window can receive a total exposure of about 10^7 R before discoloring enough to require more reducing agent or replacement of the zinc bromide.

The inner and outer walls of the window usually consist of two 2.5-cm-thick glass plates.(5) Double sheets of glass are used to reduce the likelihood of losing the shielding fluid as a result of an accident. Glass used for the inner wall is highly stable in regard to radiation effects.

Lead glass (ρ = 3.27-6.22 gm/cc) can be used with zinc bromide. This combination can produce a window having a shielding capacity and thickness equal to that of the cell wall.

Glass windows can be made of one or more special kinds of glass developed for shielding windows. The glasses are supplied in several sizes and with thicknesses up to about 20 cm.(5) A window requiring a greater thickness must be made of more than one plate. The plates are arranged so that they are a few tenths of a millimeter from each other. These spaces are then filled with a liquid (mineral oil or $ZnBr_2$ solution) having an index of refraction about the same as that of the glass.

Three factors to be considered in choosing the type of glass are: the original color of the glass, the color produced by gamma rays, and the stability of the induced color at room temperature.(5,14) [Coloration and other damage effects on glasses due to neutrons are not too well known.(4)]

Color is produced in glass when some of the electrons displaced by gamma rays are trapped in so-called color centers. Commercial lime glasses lose their transparency when exposed to 10^4 R or less.(5,14)

Lead glasses offer more resistance to radiation. The coloring process can be reversed by applying heat or an intense beam of light to the affected glass. The color fades as the trapped electrons are energized and return to their normal positions.

Color formation can be controlled to a certain extent by adding a small amount of cerium oxide to the glass.(4,5,14) This is known as nonbrowning glass. It is believed that the displaced electrons are removed by valence changes in the cerium. This prevents the formation of color centers.

Direct viewing is enhanced by having the work located close to, and normal with, a window surface. This reduces distortion and improves depth perception.(7)

2. Periscopes

Periscopes are often used for detailed examination of the work within a hot cell.(4) 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.(5) He can view a large field with low magnification or a small and more detailed field with high magnification. If a lens changer is used, he can choose an object in a wide field and then change lenses to obtain a closeup view. Mirrors are used to bend the optical path. A rotating mirror at the objective end of the unit allows a worker to scan the inside of a cell.

Periscopes placed at an angle through the shielding wall prevent the escape of direct radiation. A more elaborate unit, with several right-angle bends, has been designed for operations involving high-activity sources.(14)

3. Mirrors and Television

Mirror systems can augment direct viewing and permit us to follow the progress of simple operations with fairly low activity substances in open-top cells.(7,14)

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.(4,14) The latter limitation can be overcome to a certain extent with a television system in which the camera and screen follow the movements of the observer's head.(15)

Factors that often favor mirrors over television include clearer images, lack of cables, lack of maintenance, and low cost.(7)

G. Remote-handling Devices

In many cases, handling radioactive substances with the bare hands is unwise. Therefore, many operations with active matter are carried out with remote-handling devices (manipulators). Many such devices have been developed, the particular design depending upon the process to be performed.(14)

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.(4,7) A rod, having a handle at one end and tongs at the other, moves through a ball joint to change 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 60-90°. 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. These manipulators have a master arm (control arm) outside the shielding wall and a slave arm (working arm) within the cell (see Fig. 16.1). 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 may be obtained with or without sealed boots.

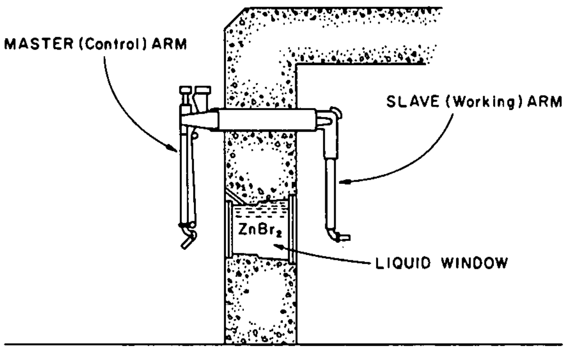

Fig. 16.1. Section through a Hot-cell Wall

Some of the many types of remote-handling devices are further discussed in the literature.(4,7)

H. 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 Safety 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.

1. 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 is to 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.(17) This bagging technique is also useful for cells that must be maintained under an inert atmosphere.(19)

2. Decontamination

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.(4,17) 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.(21) 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.(4,17,18) 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.(4) They are removed 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.(4,17) One-piece suits may be used in handling more toxic substances. The suit is removed after cleaning in one or more shower stalls.(4,17-19)

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 surveyor. The remaining tools and equipment can now be moved 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.(4,16)

Containment boxes are often used within a hot cell to limit the spread of contamination. Sometimes, discarding the entire box upon completion of the experiment may be more economical.(18) Monitoring services are generally required for this operation.

Radios or telephones can be used by the decontamination group when direct contact is hindered by respirators, suits, shielding walls, or distance.

3. 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.(4,17,18) Push-through filter-exchange systems, with or without shielding, have been developed to overcome some of these problems.(4,22,23) In some facilities, the first of a series of filters can be changed remotely with the cell manipulators.(4,17) 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.

I. Personnel Monitors

Personnel-monitoring devices worn by cell operators and support personnel at ANL include film badges and pocket dosimeters (see Section 13). Sometimes, a commercially available device, which gives off intermittent chirp-like sounds and light flashes, is also worn.(17,18) The signal rate increases with increasing exposure rate.

Hand and shoe monitors are often placed near a cell operator so that he can check himself during his work. They are also located near exit doors, and all personnel are expected to check their hands and shoes before leaving a cell complex.

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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

$$^{14}_{7}\mathrm{N} + ^{4}_{2}\mathrm{He} \rightarrow ^{18}_{9}\mathrm{F} \rightarrow ^{17}_{8}\mathrm{O} + ^{1}_{1}\mathrm{H}.$$

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.

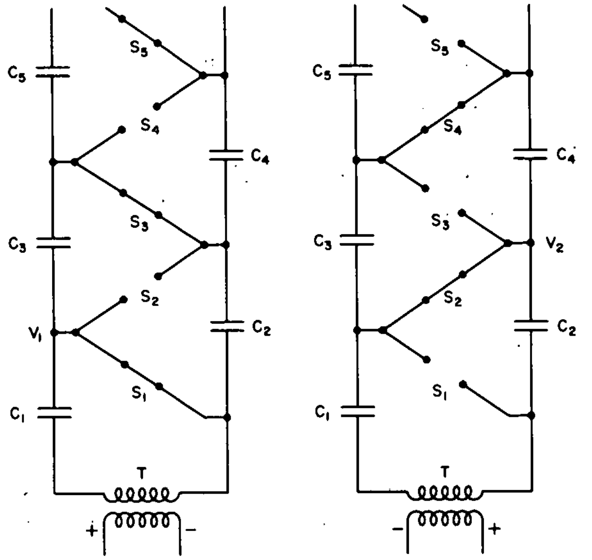
Moreover, these machines have greatly enhanced the capacity to produce radioactive substances. They have also added to a much clearer picture of the interaction processes of nuclei and charged particles or ions. The machines have produced many of the elements beyond uranium.

1. Cockcroft-Walton Accelerator

The first important accelerator resulted from the work 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 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). The eV is defined as the energy an electron gains when it is accelerated by a potential of 1 Volt. Therefore, if we can achieve a high 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 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 Fig. 17.1).(1) A transformer T provides the high-voltage alternating potential. The capacitors C_1, C_2, C_3, ..., are all of equal value. Switches S_1, S_2, S_3, ..., 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 (Fig. 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 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.



(a) Charging half-cycle for C_1, C_3, C_5, etc.

(b) Charging half-cycle for C_2, C_4, C_6, etc.

Fig. 17.1. Voltage Multiplier. [From Samuel Glasstone's "Sourcebook on Atomic Energy," Second Edition, Copyright 1958, D. Van Nostrand Company, Inc., Princeton, N. J.(1)]

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. All types of ions can be accelerated. A gas-discharge tube is often used as the ion source. To ensure as large a beam 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.

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.

The advance in the field is indicated by the fact that in only about 35 years, machines have been built to produce particles of greater than 30 GeV (GeV = 10^9 eV). Moreover, machines have been proposed for the hundreds of GeV range.

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. The belt, made of silk, paper, rubber, or other nonconducting fabric, is driven at high speed by motor.(3) A d.c. potential (20,000-50,000 Volts) is applied between a set of needle points, A, and a rounded surface, B (see Fig. 17.2). The intense electric field at the needle points causes a corona discharge. This type of discharge takes place 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 Fig. 17.2, 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. Moreover, the positive ions on the belt are removed and transferred to the sphere. This adds to the charge building

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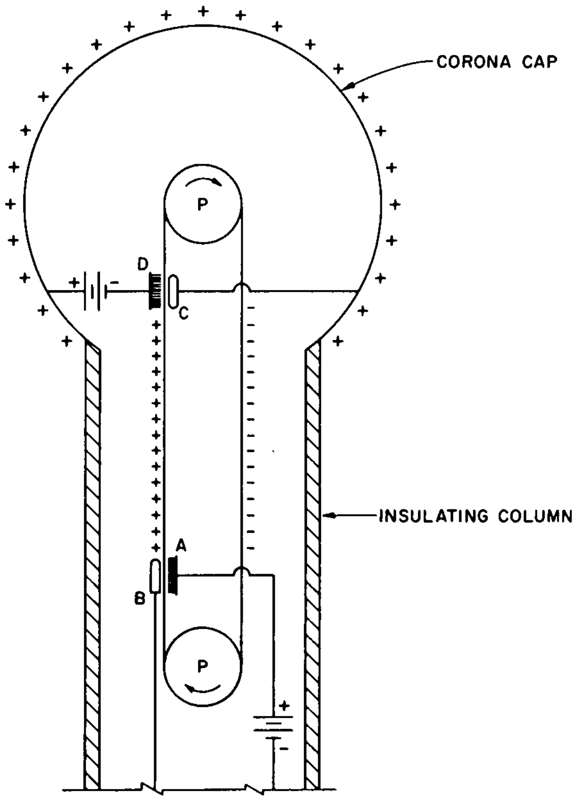

Fig. 17.2. Van de Graaff Electrostatic Generator. (R. E. Lapp and H. L. Andrews, "Nuclear Radiation Physics," Second Edition, © 1954.(2) Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

up on the cap. If a wide belt is used and the speed at which the belt is driven is high enough (perhaps as high as 60 mph), large charges can be built up and maintained on the cap.

As the machine is operated, the voltage of the cap with respect to earth increases until it reaches a certain value. Then, electric charge begins to leak away as fast as it is taken from the belt. This leakage is once again a corona discharge in air at the surface of the sphere due to the high field and the high surface density. The 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, which is then filled with a gas at high pressure (about 15 atm). Common gases used for this purpose are nitrogen and Freon.

The potential developed by the device can then be applied to the ions that are to be employed. The voltage is applied to an 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 20 ft) 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. These sections, which may be about 10 in. in diameter, are joined together by vacuum-tight seals. The common practice is to connect the produced high voltage to a hollow cylindrical electrode at one end of the tube while a like electrode, at the opposite end, is connected to ground. To reduce scattering and absorption of the ions, the tube is operated under a vacuum.

It is rather easy to continuously vary the output voltage of the Van de Graaff accelerator. Another feature of this machine is that the output voltage can be held constant to an accuracy of about 0.1%. These controls over the voltage are of value in the study of energy dependence of certain nuclear processes. The machine can be used to accelerate all ions, but is, in most cases, used for protons or heavier particles. Most of these devices are designed to produce particles of 5-6 MeV.

b. Tandem Van de Graaff Machine

The two-stage Tandem Van de Graaff machine is a recent addition to the accelerator field.(4) 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 10 MeV are easily obtained with these devices. This method can be applied to three or more stages to obtain higher-energy beams.

3. Linear Accelerators

a. Linear Radio-frequency Accelerator

The two Van de Graaff generators described above use a static means of acceleration. This means a d.c. potential is used to accelerate the ions. In the linear accelerator, (or Linac as it is commonly referred to), a dynamic or alternating potential is employed 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 Fig. 17.3. 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.(4) That is, alternate tubes carry opposite potentials.

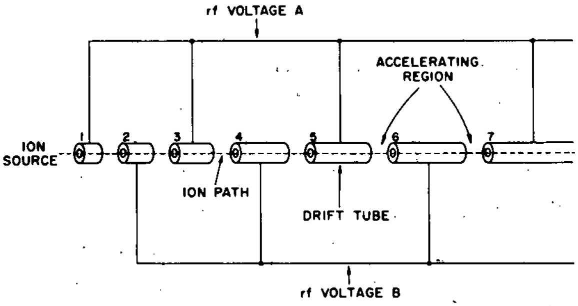

Fig. 17.3. Linear Radio-frequency Accelerator. (R. E. Lapp and H. L. Andrews, "Nuclear Radiation Physics," Second Edition, (C) 1954.(2) Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

Let us 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 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, since the frequency of the a.c. potential is fixed, the time between half-cycles is fixed. Therefore, the ion must spend the same amount of time in each tube so that it arrives at the next gap in phase with the a.c. potential. To continue this acceleration process, and thereby increase the kinetic energy of the ion, each successive drift tube (or gap and tube) must be longer than the preceding one. The length of any particular gap and tube arrangement 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. However, 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 6 ft. With an applied potential of 79,000 Volts, 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 1 to 100 cm), 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. 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.(5) Its overall length is 220 ft. This is a traveling-wave type machine, which produces a pulsed beam of electrons.

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 are fed into the machine. They reach an energy of 32 MeV as they pass through the 40-ft accelerator.

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 of the cyclotron consists of two electrodes, which may look like the two halves of a pill box (see Fig. 17.4). 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, which is located between the poles of a large magnet. A source of high-frequency alternating voltage is connected to the dees.(4)

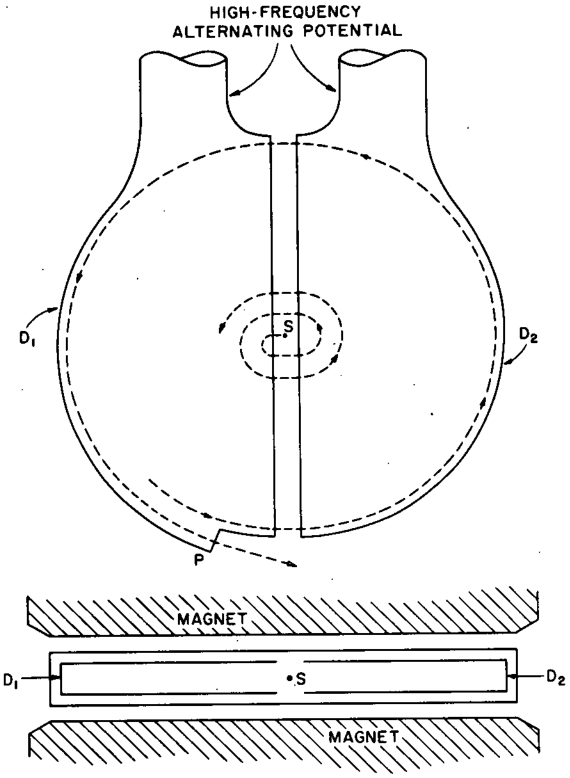

Fig. 17.4. Cyclotron. (From Samuel Glasstone's "Sourcebook on Atomic Energy," Second Edition, Copyright 1958, D. Van Nostrand Company, Inc., Princeton, N. J.(1))

Let us 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. When the ion reaches the gap, the potential of the dees is reversed. The electric field once again accelerates the ion across the gap. As a result, the ion plunges into D_2 with a little more kinetic energy than it had in D_1. To remain in phase with the alternating voltage, the ion spends the same amount of 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 given 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.

Cyclotrons are referred to in terms of the diameter of the magnet pole faces. Thus, a "60-in. cyclotron" would imply a machine with pole faces 60 in. in diameter; the diameter of the dees is a few inches less.

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) 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). According to this principle, a changing magnetic flux produces an electromotive 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 in the doughnut as a result of 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. 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 has not grown because higher energies can be achieved with Linacs.

6. Synchrotron

a. General

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.(8) In a synchrocyclotron (FM cyclotron), the cost factor restricts the proton energy to about 1 GeV.

Some synchrotrons have been built to produce high-energy protons; others were designed for electrons. Our discussion is concerned mainly with proton machines.

A typical synchrotron consists of a more or less circular vacuum chamber within a ring-shaped magnet and short straight sections (see Fig. 17.5).(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 path remains fixed. Therefore, the final energy of the protons depends upon the maximum magnetic field that can be attained.

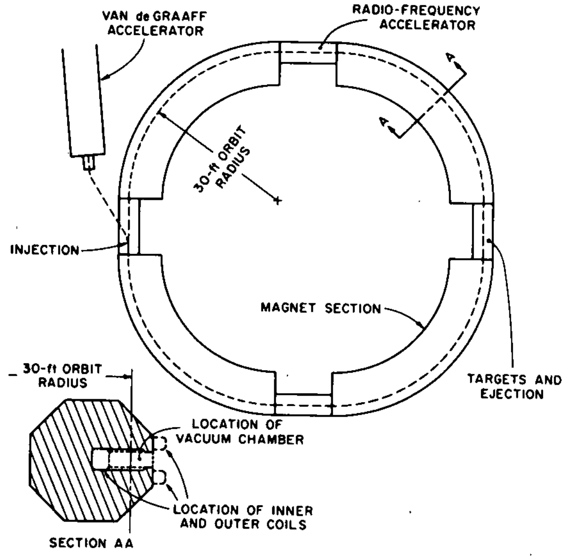

Fig. 17.5. Synchrotron. (The Cosmotron.) (M. H. Blewett et al., "Review of Scientific Instruments" 24, No. 9 (Sept 1953).(9) Original prints by Brookhaven National Laboratory.)

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 thus 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.(4) 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 produce protons of 3 and 6 GeV, respectively. These energies are so high that relativistic effects must be taken into consideration.

b. Zero-gradient Synchrotron (ZGS)

The ZGS at Argonne is able to produce 12.5-GeV protons. It was designed to accelerate 3 to 5 x 10^{12} protons per pulse and produce pulses at a rate of about one every 4 sec; however, higher repetition rates have been achieved. The magnet consists of eight sections (octants) arranged in a circle whose diameter is about 200 ft.(8) The octants are separated by straight sections, which are alternately 14 and 20 ft long. A vacuum chamber, 32 in. wide by $5\frac{1}{4}$ in. high, is enclosed within this arrangement.

Protons are injected into the ZGS chamber with an energy of 50 MeV. The entire injection system consists of a 750-keV Cockcroft-Walton and a 50-MeV Linac. The Linac is 110 ft long and contains 124 drift tubes. The machines in this injection system are designed for pulsed rather than continuous operation.

The term "zero gradient" refers to the nature of the magnetic field within the magnet (and therefore throughout the cross section of the chamber). In the ZGS, the magnetic field, at any instant, is uniform within the magnet (i.e., zero gradient). In some machines, the field decreases from the inner to the outer edges of the magnet. The field is described as having a gradient (negative) in the radial direction. A machine may have a system of magnets in which the gradient changes with successive magnets. That is, the field increases radially in one magnet section, decreases in the next section, etc. This type machine is known as an Alternating Gradient Synchrotron (AGS).

Since the field in the ZGS chamber is constant, it provides no vertical focusing effect. That is, if the protons are above or below the equilibrium orbit, the field does not act on the protons to force them back to this desired orbit. Vertical focusing is achieved by the proper shaping of magnet ends.(4) The ends of an octant are cut so that they are not perpendicular to the equilibrium orbit in any straight section. The desired focusing action is thus provided at the entry and exit of each of the eight magnet sections.

Radial focusing also takes place at the magnet ends. Inside the magnet sections, the proton beam has radial stability.

The protons follow the equilibrium orbit if they are in phase with the rf voltage. If they are out of phase, the protons spiral about the orbit as they travel through the chamber. Protons slightly out of phase with the rf voltage receive more or less acceleration depending upon whether they are behind or ahead of the protons that are in phase. Eventually, these protons are brought back into phase so that the protons in the beam tend to form "bunches." This bunching effect is referred to as phase focusing.

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?

In the last century, the atom was looked upon as the basic unit. During the early part of this century, it was found that the atom was composed of a tiny core (the nucleus) and one or more electrons. Later studies showed that the nucleus was composed of still smaller units - the proton and the neutron (nucleons). To this list we could add the light wave (photon), since light waves can act like particles and particles can act like light waves.

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.(10,11)

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.(8) 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.(11)

Cosmic-ray research made vital early contributions to this branch of physics.(8) 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. 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.

2. High-energy Processes

The radiation from high-energy processes includes protons, neutrons, π and μ mesons, electrons, and X and gamma rays.(12-14) In some of the newer machines (GeV range), such as the ZGS, protons on protons produce K mesons plus Λ, Σ, and Ξ hyperons (the so-called strange particles). Moreover, the products themselves can add to the total output through further processes.

In the first stages of a high-energy process, the particle strikes the target and a number of products may be instantly ejected.(2,15,16) Depending upon the mass number, A, of the ejected matter, the process is referred to as fragmentation or spallation. If $A > 10$, the process is often called fragmentation. Spallation refers to processes in which smaller bits of matter, which include nucleons, are emitted. The nucleons ejected in the above processes are sometimes called cascade particles.

In a primary interaction, cascade protons and neutrons, as well as π^+, π^-, and π^0 mesons, are "knocked" out of the struck nucleus. Hence, this stage is often called the knock-on phase. 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.

a. Mesons

Among the products formed in the knock-on phase that are 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. The heavier mesons are called π mesons, or pions. Their mass is about 273 times the electron mass ($273m_e$). In addition to positive and negative charges, a π meson may have no charge; the mass of this neutral π meson is $265m_e$. The lighter mesons are called μ mesons, or muons. Their mass is $207m_e$. They have a positive or negative charge. Mesons are unstable particles, which decay with a mean lifetime of about 10^{-6} to 10^{-16} sec. Charged π mesons decay into μ mesons and neutrinos; the neutral meson decays into two gammas:(17)

$$\left.\begin{array}{l}\pi^+ \rightarrow \mu^+ + \nu \\ \pi^- \rightarrow \mu^- + \bar{\nu}\end{array}\right\} \left\{\text{mean lifetime} = 2.5 \times 10^{-8}\ \text{sec}\right\};$$

$$\pi^\circ \rightarrow \gamma + \gamma \quad (\text{mean lifetime} = 2.3 \times 10^{-16}\ \text{sec}).$$

The μ mesons decay into electrons and neutrino-antineutrino pairs:

$$\left.\begin{array}{l}\mu^+ \rightarrow e^+ + \nu + \bar{\nu} \\ \mu^- \rightarrow e^- + \nu + \bar{\nu}\end{array}\right\} \left\{\text{mean lifetime} = 2.2 \times 10^{-6}\ \text{sec}\right\}.$$

It is believed that π mesons are exchanged between protons and neutrons within a nucleus. This meson exchange would account for part of the force field in the nucleus.

A third group of mesons, called K mesons or kaons, has been found. Charged kaons have a rest mass of about $968m_e$ and a mean lifetime of 1.2×10^{-8} sec. The neutral kaon has a rest mass of $976m_e$ and a mean lifetime shorter than that of the charged kaon.

b. Hyperons

An elementary particle whose mass is greater than that of the proton is called a hyperon. The hyperons are unstable with very brief mean lifetimes, as shown in the following table:(12,17-19)

Particle	Mass, m_e	Mean Lifetime, sec
Λ	2183	2.5×10^{-10}
Σ^+	2328	0.8×10^{-10}
Σ°	2332	$\sim 10^{-20}$
Σ^-	2341	1.6×10^{-10}
Ξ°	2566	1.5×10^{-10}
Ξ^-	2583	1.3×10^{-10}

The positron is often referred to as the antiparticle of the electron since it can be mutually annihilated with the electron. Particles that are commonly found in nature have counterparts that carry an opposite electric charge; these counterparts are known as antiparticles or antimatter.(2) Except for the photon and neutral π meson, all the other known elementary particles are associated with an antiparticle.(17)

C. Radiation Sources

The radiations emitted as a result of accelerator work vary in kind and degrees 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.

1. Beam Interactions

The primary beam, which may consist of electrons, protons, or deuterons, is often brought out of the accelerator chamber. This is frequently done by sending the beam through pipes or thin foil windows.(20) Since this beam is quite penetrating, it poses a shielding problem. Moreover, there is the problem of scattered rays. The penetration of a 500-MeV proton, for example, is about 7 in. of lead. Although the shielding problem is not too severe in this case, the scattered radiation may present quite a problem.

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.

When the beam strikes a target, nuclear processes take place which may result in a number of radiations. For instance, if a proton in the GeV range strikes a target nucleus, protons and neutrons with energies in the 100-MeV range are instantly ejected.(14) They, in turn, can cause processes like those caused by the primary proton. For this reason, they are often referred to as cascade particles. The evaporation protons and neutrons emitted by the excited nucleus may have energies in the 10-MeV range. The π mesons produced can also cause nuclear processes, which release more neutrons and protons. (To produce a new particle in any process, energy must be supplied that is equivalent to the rest energy of the particle.(6) Therefore about 140 MeV must be supplied to a nucleon to produce a free π meson. In proton-nucleon processes, the protons must have at least 293 MeV to produce π^+ and π^- mesons; 280 MeV is required to produce π° mesons.) The π mesons interact strongly with nuclei; however, many of them may decay before being absorbed. The μ mesons resulting from this decay process can pass through large amounts of matter before they are absorbed. In the disintegration of the target nucleus, high-energy gamma rays, which can start electron-photon cascades, are also produced. If the energy of the incoming proton is high enough, the K mesons and hyperons can be produced. The minimum energy (in GeV) required to produce these strange particles by protons on protons is as follows: K, Λ (1.58), Σ (1.78), Ξ (3.73), $\overline{K}$ (2.49), $\overline{\Lambda}$ (7.1), $\overline{\Sigma}$ (7.4), $\overline{\Xi}$ (8.9).(8) The symbols $\overline{K}$, $\overline{\Lambda}$, etc., represent the antiparticles.

X rays are one of the problems associated with electron machines. The X rays produced when the beam strikes the target or parts of the machine have an energy spectrum that extends up to the maximum energy of the beam.(20,21) Although the rays emerge at all angles, the intensities vary. The intensity in the forward direction becomes relatively greater with increasing beam energy.

2. Induced Radioactivity

Radioactivity induced in various parts of a machine by the beam or scattered rays may present a hazard to personnel during maintenance work. The activities thus produced have a wide range of half-lives, similar to fission products.(20) Initially, there is a rapid decay of the induced activity. This fact is useful in scheduling repairs. In the very high-energy machines, the dose rates from this source are usually lower because the activity is produced deep in the material.

3. 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.(22) The use of heavy-element targets increases the skyshine radiation level. 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 a GeV accelerator have shown that the skyshine drops off inversely with distance; the decrease becomes faster beyond about 500 ft. The shielding over a GeV machine may be comparable to the required thick side wall shield.

4. Klystrons

High-voltage klystrons are used for the rf power source for some of the Linacs. X rays are generated in these tubes, and shielding may be required for personnel protection.(21)

5. Radioactive and Toxic Gases

Ozone, a toxic gas, plus radioactive gases such as ^{15}O and ^{13}N, products of (γ,n) reactions in air, may be produced during machine operation.(23,24) The concentration of these gases varies, depending upon factors such as beam power, length of beam operation, target substance, and ventilation rate.

The shielding initially erected around some of the machines was found to be inadequate as beam energies and intensities were increased.(14,22) 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.

D. 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. 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 rem dose is caused by fast neutrons (up to ~20 MeV).(12,25) The balance of the dose is due to thermal neutrons, gamma rays, and high-energy particles.

To assess the hazard, measurements that can be converted to units of dose equivalent (DE) have to be made. The DE is defined as the product of the absorbed dose (D) and modifying factors.(26) The quality factor (QF) should be used for protection work. The product of an absorbed dose and a suitable QF expresses the irradiation in terms of a common scale for all ionizing radiations. The distribution factor (DF) is used to correct for nonuniform distribution of internally deposited radionuclides. Thus, the DE (in rems) is given by

$$DE = D\,(QF)\,(DF)\ldots . \tag{17.1}$$

At present, the values of QF to be used correspond to the previous values of RBE used for protection work. These values are related to the linear energy transfer (LET) of the radiation. (See Section 12 for a discussion of LET.) The relationship between QF and LET recommended for radiation-protection work is shown in Table 17.1.(27) For X and gamma rays, QF = 1. For electrons, QF = 1 except at very low energies. (When $E_{max} \leq 0.03$ MeV, QF = 1.7.) In many cases, we must evaluate the hazard due to a mixed field of neutrons and gamma rays. When the neutron energy spectrum is known, the DE due to this component is obtained from the product of the absorbed dose and the appropriate QF. Values of QF for neutrons as a function of neutron energy up to 10 MeV are contained in NBS Handbook 63.(37) If the spectrum information is lacking, an upper limit for the DE can be taken as 10 times the sum of the neutron doses, plus the gamma dose. A simpler but more conservative approach is to measure the total absorbed dose and use a QF of 10.

Table 17.1

RELATIONSHIP BETWEEN QUALITY FACTOR (QF) AND LINEAR ENERGY TRANSFER (LET)

QF	LET_∞ (keV per micron in water)
1	3.5 or less
1 to 2	3.5 to 7.0
2 to 5	7.0 to 23
5 to 10	23 to 53
10 to 20	53 to 175

[From "Report of Committee IV on Protection against Electromagnetic Radiation above 3 MeV and Electrons, Neutrons, and Protons," Pergamon Press, distributed by The MacMillan Company (1964).](27)

Figures 17.6 and 17.7 show suggested estimates of QF for neutrons and protons plus particle fluxes corresponding to a DE rate of 2.5 mrem/hr.(27)

E. Radiation Survey Instruments

For many of the lower-energy accelerators, conventional survey instruments are used (see Section 12). The type of radiation and wide range of energies encountered near the higher-energy machines requires, at present, a family of survey instruments.

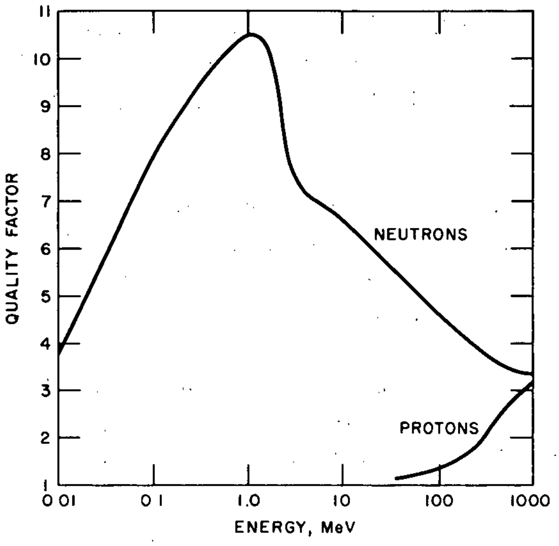

Fig. 17.6. Values of QF for Neutrons and Protons of Various Energies. [From "Report of Committee IV on Protection against Electromagnetic Radiation above 3 MeV and Electrons, Neutrons and Protons," Pergamon Press, distributed by The MacMillan Co. (1964).](27)

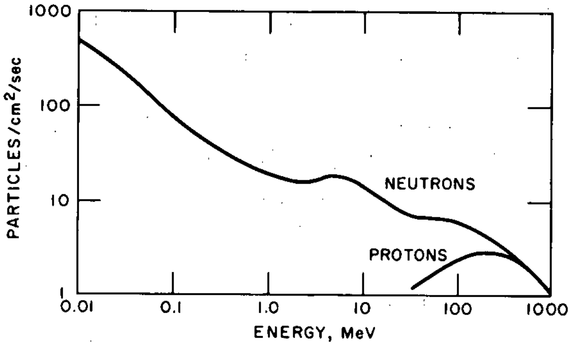

Fig. 17.7. Particle-flux Densities Corresponding to a Dose Rate of 2.5 mrem/hr. The flux values are for primary particles in beams assumed to be in equilibrium with secondary particles as a result of passage through shielding. [From "Report of Committee IV on Protection against Electromagnetic Radiation above 3 MeV and Electrons, Neutrons and Protons," Pergamon Press, distributed by The MacMillan Co. (1964).](27)

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.(28,29) 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 gamma rays and electrons.(30) Voltage pulses are amplified and added, and the resulting meter reading is proportional to the energy absorbed in the gas-filled unit.

A device composed of thermal-activation foils within a paraffin sphere has been employed for neutron survey purposes. Fluence, dose equivalent, and average neutron energy can be determined with this unit.(28,31) 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 some 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 flux of high-energy neutrons (>20 MeV) has been estimated from a measure of the ^{11}C activity induced in a plastic scintillator.(28,29,32,33) The 20.4-min activity, resulting from the $^{12}C(n,2n)^{11}C$ reaction, is counted in a photomultiplier system and converted to units of flux.

Scintillation crystals have also been used for neutron monitoring.(29) 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 gamma rays.

Neutron spectra and total flux above a certain energy can be estimated with threshold detectors. The activity produced in foils through neutron-induced processes provides the data regarding regions of the neutron spectrum.

Personnel engaged in survey work around the large machines should be aware of the limitations of survey devices. Moreover, they should be trained in the techniques of measuring and evaluating pulsed-radiation fields.

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