DOSIMETRY

Radiation dosimetry can be defined as the theory and application of principles and techniques associated with the measurement of ionizing radiation (1). In the field of radiation protection/safety, radiation dosimetry is divided into two primary categories: external dosimetry and internal dosimetry. External dosimetry applies to measurements in which the radiation source (i.e., the radioactive material) is outside the body and the measurements can be made with any number of sensitive radiation detectors. In contrast, internal dosimetry applies to situations in which the radioactive material is taken inside the body and may be incorporated into organs and tissues of the body. Internal dosimetry requires very specialized calculations involving many assumptions, usually standardized in the field because it is not possible to make direct measurements for internally deposited radioactive materials. Usually measurements are made of radioactive material excreted from the body or radiation emanating from the body using very sensitive detectors and, based on the data

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obtained, a calculation is used to assess the dose. Because of the very specialized nature of internal dosimetry, this topic will not be discussed further.

External dosimetry usually involves radiation detection instrumentation which can be used to assess one or more characteristics of the radiation field. These characteristics include measurements of the types of radiation, the total energy deposited in a radiation detector, the energy distribution (i.e., the energy spectrum) and/or the total fluence, the radiation intensity, the angular dependence of the field, the time dependence of the field, locations of the sources within the area, and many other specific parameters depending on the purpose of the measurement. Thus, the selection of an appropriate radiation detector depends on the purpose of the measurement. In addition, almost all radiation detectors (i.e., dosimeters) require careful calibration in known radiation fields before use in unknown radiation fields.

The term *dosimetry* can best be understood by remembering that the term simply means "dose measurement." In the simplest of terms, dose measurement involves the assessment of the energy deposited by ionizing radiation in a known amount of material. This article introduces the concepts associated with radiation dosimetry and provides a survey of some common radiation detectors used for this purpose. To assist in understanding the concepts presented in this article, a glossary of common terms is provided at the end. For detailed discussions of the quantities and units associated with radiation dosimetry, the publications of the International Commission on Radiation Units and Measurements (2–5) should be consulted.

INTERACTIONS OF RADIATION WITH MATTER

The basis for radiation dosimetry is obviously the ability to use a device to detect or measure the energy deposited by radiation (or a quantity which can be related to the deposited energy). The device can take many forms and the methods of detection vary widely. In order to understand how radiation can be detected and how energy is deposited in a radiation detector, it is necessary to have some understanding of the mechanisms through which radiation interacts with matter. These interaction mechanisms are dependent on the types of radiation to be measured and, in many cases, the energy of the radiation. The subsequent discussion is intended to provide some insight into the common types of radiation and the interaction mechanisms of some of the most common radiations. More detailed discussions can be found in several excellent texts (6–8).

Alpha Radiation

Alpha particles (α^{++}) are large, charged particles composed of two neutrons and two protons (some texts correctly identify this particle as a helium nucleus) and are less penetrating than the other radiations typically considered in radiation dosimetry. An alpha particle has a positive charge of two and, on an atomic scale, is quite massive. In general, a single neutron or proton is about 1840 times larger than any of the electrons orbiting the nuclei of atoms. Thus, an alpha particle is more than 7000 times larger than a single electron. This size difference plays an important role in the way alpha particles interact with matter and the hazard they present to living tissue.

For the purposes of this discussion, it can be assumed that alpha particles emitted in the decay of a specific radionuclide are monoenergetic (i.e., they all have the same kinetic energy). The kinetic energy of an alpha particle is transferred to the medium, through which it is passing, by interactions between the particle and the orbital electrons of the atoms or molecules of the material. The major energy-loss mechanisms are electronic excitation and ionization of the orbital electrons. An alpha particle has a high electrical charge (+2) but a very low velocity (because of its mass). Thus, interactions with the medium are not spaced very far apart and the alpha particle does not travel far in most materials (i.e., alpha particles have a high specific ionization and a short range). These interactions are not actually collisions with the loosely bound electrons of the atom but are electrical or coulombic in nature. Since the particle is positively charged, it exerts an attractive force on the oppositely charged, negative electrons. The force exerted by the alpha particle is dependent on the distance between it and the electron as they pass (often called the *impact parameter*). The force and the probability of an ionizing event both increase as the distance decreases. In some cases, this attractive force is not sufficient to remove the electron from the atom and the electron simply is raised to a higher orbital position (i.e., the atom is excited). If the attractive force is sufficient to separate the electron from the atom, the atom is ionized and an ion pair is created. In some cases, for close encounters, it is instructive to imagine that the alpha particle exerts a huge attractive force and rips one or more electrons from the atom.

Each interaction with an electron reduces the kinetic energy of the alpha particle. On the average, it takes approximately 34 eV to produce an ion pair in air. This is about twice the first ionization potential for most gases because, as stated, the energy transferred is shared between excitation and ionization events in the material. A typical alpha particle has a high specific ionization and may produce 20,000 to 40,000 ion pairs/cm of travel in air. In addition, the range of the alpha particle is dependent on the material through which it is traveling. Thus, even a very energetic alpha particle, with an initial energy of 5 million electron volts (MeV), will travel only about 5 cm in air. The range of the 5 MeV alpha particle in human tissue is only about 30 μm. A typical alpha particle does not have sufficient energy to penetrate the dead layer of skin on the human body. Thus, alpha particles are not of dosimetric concern if they remain outside the body, but must be given careful consideration if internally deposited (a situation which puts them in direct contact with living tissue).

In general, alpha particles have straight paths through material and discrete ranges. The energy transfers of this massive particle are small relative the total kinetic energy of the particle (until the alpha reaches the end of its travel), thus the straight path through material. Usually alpha particles associated with a specific radionuclide are characterized by specifying the mean range of the radiation. A number of empirical equations have been derived to relate the initial kinetic energy and the range of the alpha particle. Usually, these ranges are calculated for air and converted to other materials through simple relationships. For alpha particles with initial kinetic energies in the range 4 to 8 MeV, a useful equa-

tion is:

$$R = 1.24E - 2.62 \quad (1)$$

where R is the range in air (cm); and E is the alpha particle energy (MeV). The range in tissue is obtained through a very simple ratio:

$$R_{\text{air}} \times \rho_{\text{air}} = R_{\text{tissue}} \times \rho_{\text{tissue}} \quad (2)$$

where R_{air} is alpha particle range in air (cm); ρ_{air} is density of air at STP (0.001293 g cm^{-3}); R_{tissue} is alpha particle range in tissue (cm); and ρ_{tissue} is density of tissue (1 g cm^{-3}). Thus, the range in tissue is simply:

$$R_{\text{tissue}} = R_{\text{air}} \times \rho_{\text{air}} \quad (3)$$

Once the alpha particle loses its kinetic energy it acquires two free electrons to become a helium atom.

Alpha-emitting radionuclides are usually those considered to be the heavy elements: these are the elements with a large number of nucleons (neutrons and protons) in the nucleus. Typical alpha-emitting radionuclides include Ra-226, Th-232, U-235, U-238, Pu-238, and Pu-239. Usually, alpha particle emission also is accompanied by the emission of electromagnetic radiation (gamma radiation). In most cases, spontaneous fission competes with alpha emission as a mode of transformation; however, usually alpha emission is dominant.

Alpha decay results in a recoil nucleus that also slows by producing the same kind of dense ionization. In dose calculations, especially internal dose calculations, these contributions to "dose" also must be included. The alpha particle and the recoil nucleus are both assigned a quality factor (radiation weighting factor) of 20 to indicate the biological significance of these radiations.

Beta Radiation

Beta particles (β^-) are identical to electrons in that these radiations have the same mass and charge as electrons. Beta radiation is more penetrating than alpha radiation and, in certain situations, must be considered carefully in radiation dosimetry. Beta particles originate in the nuclei of atoms which are unstable and are neutron rich (i.e., having too many neutrons in the nucleus). These radiations arise when a neutron in the nucleus is converted spontaneously into a proton, an electron, and an antineutrino. Since the electron does not normally exist in the nucleus, it is ejected and generally carries a significant kinetic energy.

In contrast to alpha particles, beta radiation emitted from the nuclei of a material is not monoenergetic but has a range of energies (i.e., a continuum or a spectrum). The energy available in the transformation is shared between the beta particle and an antineutrino. The antineutrino is thought to have no charge and a near zero mass (this is still a subject of great debate). Thus, the energy of the beta radiation emitted is distributed essentially from zero to some maximum energy associated with the specific transformation. At one extreme, the antineutrino carries away all the energy while at the other the beta particle carries away all the energy. Obviously, all other energy sharing arrangements are possible and the continuum of energies is produced. Antineutrinos have high penetrating power and low interaction probabilities and, therefore, contribute little to the dose from beta radiation. For these reasons, antineutrinos are not normally considered in dose assessments.

There are a few pure beta emitters, that is, radionuclides which reach a stable state with the emission of the beta particle and antineutrino and no other radiation. These include radionuclides such as H-3, C-14, S-35, and P-32. Often, other radiations (electromagnetic) may be emitted in this transformation because emission of the beta particle from the nucleus may not have resulted in putting the nucleus in its most stable state.

In some situations, the nuclei of atoms may be unstable because they contain too many protons (i.e., proton rich). In this case, a proton is converted spontaneously into a neutron, a positron, and a neutrino. A positron (β^+) is simply a positively charged electron which is emitted from the nucleus. As with the beta particle, the energy of the transformation is shared between the positron and the neutrino and, thus, the kinetic energies of positrons emitted in the transformation of a particular radionuclide are also distributed over a continuum. As was the case for antineutrinos, the dose contributions from neutrinos produced in beta decay are not considered in dose assessments. Radionuclides which are positron emitters include C-11, O-15, F-18, Na-22, and P-30. Positron emitters are rare but have some unique characteristics which make them very attractive for use in diagnostic nuclear medicine. These radionuclides typically have short half-lives, are produced by accelerators located in the medical facility, and their uses are limited to positron emission tomography (PET). Positrons interact with material in a similar manner as do beta particles and only the differences will be discussed here.

Beta particles also interact with matter through excitation and ionization. However, in this case the interactions are those of two particles, of the same mass, with like charges and the major interaction process is repulsion rather than attraction. It is useful to think of these as scattering interactions or inelastic collisions. Since the beta particle and the electron are essentially identical, there is a net repulsive force exerted and the orbital electron is either raised to a higher orbital position (excited) or separated from the atom (ionized). The net result is the creation of an ion pair and approximately the same energy is required to create the ion pair.

Since the beta particle and the electron are the same size (mass), in contrast to alpha particle interactions, the beta particle changes direction because of each interaction. There are many scattering events as the beta particle loses its kinetic energy and the path of the particle is far from straight (many authors use the word *tortuous*). Nevertheless, the term *range* is used to provide some insight in the distance a beta particle can travel through a material. However, the range of a beta particle is best thought of as the "crow-flight distance" or the linear thickness of material rather than the total distance (i.e., path length) traveled. It should be obvious that the path length is much larger than range. A useful rule of thumb is that the range of a beta particle, in air, is about 4 meters per MeV.

Beta particles also lose energy through radiative collisions with nuclei. This phenomenon, called *bremsstrahlung production,* describes the electromagnetic energy radiated when the beta particle is accelerated due to the presence of the nucleus. Bremsstrahlung is usually important only at very high beta particle energies and in high atomic number materials. How-

ever, production of bremsstrahlung must be recognized and considered in some radiation dosimetry situations.

There are a number of empirical range–energy relationships for beta particles. One of these equations is:

$$R = 0.542E - 0.133 \quad (4)$$

where R is the beta particle range (g cm^{-2}); and E is the beta particle energy (MeV). This equation is useful for beta particles with a maximum energy greater than 0.8 MeV. Also, note the units on the range. This unit is called the density thickness of material and is a useful way to express the range. To convert to the range in a specific material in more conventional units, all one must do is divide this result by the density of the specific material. Thus, using density thickness to specify the range means that it is not necessary to specify the material, as was required in the discussion of alpha particles.

At the end of travel, after each beta particle has lost its kinetic energy, it can exist in nature simply as a free electron. However, a positron cannot exist in nature and, when this particle has expended its kinetic energy, it combines with a free electron and these two particles annihilate. This process is a good example of conversion of mass into energy since the two electron masses disappear and the rest mass energy of the particles appears as two energetic photons of 0.511 MeV each. The production of these energetic photons (called *annihilation radiation*) must be considered in radiation dosimetry measurements as well as radiation shielding design.

X rays and Gamma Radiation

X rays and gamma rays are both electromagnetic radiation. These radiations really differ only in their origin: X rays are produced during rearrangements in the electron orbitals (shells) of the atom, whereas gamma rays are produced as a result of nuclear rearrangements. Energy (or wavelength) differences are not important in that many low energy gamma rays have been discovered and it is now possible to produce very high energy X rays. Both of these radiations are called photons. A photon has been described both as a *particle* and a *bundle* of energy. This is because photons possess both particle and wave-like properties: that is, a photon possesses energy but has no mass and has a wavelength and a frequency. X rays and gamma rays typically are the most penetrating radiation of those generally discussed in radiation dosimetry. The degree of penetration is dependent on the energy of the photons and the material through which the photons are passing. Very dense materials, such as lead, are excellent shields against photon radiation.

Photons are indirectly ionizing radiations in that a primary interaction with the material must occur which produces a charged particle and it is this charged particle which produces additional ionization and excitation in the material. Photons interact with matter in many different ways but, usually, discussions are limited to three primary mechanisms: the photoelectric effect, Compton scattering, and pair production. These are three very distinct interactions and the type and probability of the interaction occurring depends on the photon energy and the material through which the photon is passing.

The photoelectric effect occurs with highest probability at low photon energies and in high atomic number (i.e., high-Z) materials. The probability of a photoelectric interaction is proportional to the atomic number of the material (Z^4) and inversely proportional to the energy of the photon ($h\nu^{-3}$). This interaction is considered relatively unimportant for photons with energies above about 1 MeV, except in very high-Z materials. The photoelectric effect can be considered to occur with the entire atom although the interaction is really with a tightly bound electron (typically the K-shell electrons). The incoming photon strikes the electron and transfers all of its energy to the electron (the photon disappears). If the energy transferred to the electron (often called a photoelectron) is greater than the electron binding energy, the photoelectron is ejected from the atom with the excess energy being manifested as kinetic energy. This charged particle then produces additional ionization and excitation through interactions with material as described previously for beta radiation.

The photon interaction produces a vacancy in the electron structure of the atom which must be filled. An electron from a higher orbit will drop into the vacancy, leaving another vacancy which must be filled. Thus, there is a cascade of electrons as each succeeding vacancy is filled. As these electron vacancies are filled, photons are emitted with energies equal to the difference between the initial and final energy levels. These photons are called "characteristic X rays" because the photons are unique to the element from which they originate.

Compton scattering is most probable for photons in the energy range 0.1 MeV to 1 MeV and in light materials (i.e., low-Z materials). Compton scattering occurs between a photon and a very loosely bound electron. The electron is in one of the outer electron orbits and is assumed to be essentially free from electrical influences from other electrons or the nucleus, since the binding energy is significantly less than the photon energy. In this interaction, the conservation of momentum and energy make it impossible for the photon to transfer all of its energy to the electron. The photon has a collision with the electron, dislodging the electron but transferring only a portion of its energy to the photon. The photon is deflected (scattered) from its original direction of travel and has a lower energy (longer wavelength). The Compton electron has a kinetic energy equal to the difference between the initial and final photon energy. The scattered photon may have additional Compton interactions or a photoelectric interaction in the material (depending on the photon energy). As with the photoelectron, the Compton electron produces additional ionization and excitation through interactions with material, as described previously for beta radiation.

Pair production occurs with highest probability for photons with high energy (i.e., typically more than a few million electron volts). This interaction is the opposite of annihilation radiation production discussed previously. Here, the photon penetrates to the near vicinity of the nucleus and has a coulombic interaction with the charged nucleus. The photon disappears and two charged particles are produced. The charged particles are electrons of opposite signs (i.e., an electron and a positron) and the excess kinetic energy of the photon is shared equally between the two particles. Since each of these particles has the rest mass energy equivalent of 0.511 MeV, the interaction is not possible for photons with energies below a threshold of 1.022 MeV. In general, even above this threshold, pair production is not important for photons below about 4 MeV. Again, these charged particles produce additional ionization and excitation through interactions with material, as described previously for beta radiation. When the positron

has expended its energy in ionization and excitation in the material, it will annihilate with a free electron, as described previously.

Neutron Radiation

Neutron interactions are strongly dependent on the kinetic energy of the neutron and the material through which it is passing. Neutrons have approximately the same mass as a proton but possess no electrical charge. Thus, neutrons can penetrate to the nucleus of an atom and interact in a number of different ways. For thermal neutrons, capture is the most important interaction. In this interaction, the neutron is captured by the nucleus of an atom of the material and the nuclear structure is transformed. Typically, the nucleus is unstable and the excess energy, in the form of radiation, may be emitted. In dosimetry for radiation protection, two thermal neutron interactions with tissue are important. First, the neutron may be captured by the nucleus of a hydrogen atom and a 2.2 MeV gamma ray is emitted (this reaction is written $^{1}H(n, \gamma)^{2}H$). A second important interaction is between the thermal neutron and nitrogen, $^{14}N(n, p)^{14}C$, which produces a 0.6 MeV proton. There are a number of other possible capture reactions which occur with a wide range of materials (typically called activation) but these will not be discussed here.

Intermediate energy neutrons are usually in the process of slowing down from higher energies. The interactions processes include scattering but capture and other nuclear reactions also may occur.

Fast neutrons are usually the most important in terms of radiation dosimetry, especially since the concern is the deposition of energy in tissue. Primary neutron interactions considered are either elastic or inelastic collisions. An elastic collision of a neutron with nucleus of an atom results in deflection of the incident particle and a transfer of a portion of the neutron energy to the struck nucleus. Energy losses depend on the size of the struck nucleus and the collision angle. Sometimes, it is possible to transfer all the fast neutron energy to the struck nucleus in a head-on collision. The most important fast neutron interaction in tissue is elastic scattering with hydrogen. In these interactions, since the neutron and the hydrogen nucleus (a proton) are essentially the same size, complete energy transfer is possible. In tissue, more than 90% of the fast neutron interactions and the energy transfer are due to elastic collisions between fast neutrons and hydrogen nuclei.

As the neutron energy increases, inelastic collisions become important. These interactions occur for neutrons with energies typically above about 1 MeV and, above about 10 MeV, elastic scattering and inelastic scattering have equal probability of occurring. In tissue, the most important inelastic interactions are those with the nuclei of carbon, nitrogen, and oxygen. Most of these interactions result in the emission of gamma rays, as the nuclei deexcite. However, in some cases, the deexcitation may include the emission of protons or alpha particles. These latter reactions typically take place with very high energy neutrons (i.e., above about 5 MeV).

In the relativistic energy range (i.e., >10 MeV), inelastic scattering is more important than elastic scattering. For high-Z materials, the elastic probability (i.e., cross-section) may be ignored entirely. But, even in this energy range, elastic collisions in tissue are still important.

THE BRAGG–GRAY PRINCIPLE

Many radiation detectors, if calibrated properly, can be used to measure the absorbed dose and the dose equivalent from exposure to ionizing radiation. Ionization chambers, filled with air, were one of the first detectors to be used to measure the absorbed dose in tissue. Accurate dosimetry with ionization chambers (or any gas-filled detector) has its foundation in the Bragg–Gray principle. This fundamental principle states that the energy deposited by secondary electrons per unit volume in a solid medium is equal to the product of the ionization per unit volume in a gas-filled cavity in the medium, the mean energy expended in the gas, and the ratio of the mass stopping powers of the secondary electrons in the medium and the gas. More succinctly, this simply means that the amount of ionization produced in the gas-filled cavity, in the medium, serves as a measure of the energy deposited in the surrounding medium.

For the above statement to be true, four conditions must be met (1):

1. The cavity must have dimensions such that only a small fraction of the energy of the charged particle is dissipated in it. This requirement simply means that only a small fraction of the charged particles contributing to the ionization will enter the cavity with a range that is less than the dimensions of the cavity.
2. Contributions of radiation interactions in the gas filling the cavity to the total ionization in the cavity should be negligible. This requirement means that ionization in the cavity should be caused by charged particles produced in the medium, as opposed to the cavity gas. Generally, this requirement is satisfied if the first requirement is satisfied.
3. The cavity must be surrounded by an equilibrium thickness of the solid medium. This is the thickness that will result in the condition called *electronic equilibrium*. Electronic equilibrium exists when electrons, produced by radiation interacting in the cavity and leaving the cavity, are replaced by electrons, produced by radiation interacting in the medium and entering the cavity, deposit a portion of their kinetic energy (see Fig. 1). Basically, the equilibrium thickness is equal to the range of the most energetic secondary electrons produced by interactions of the primary radiation. The principle of

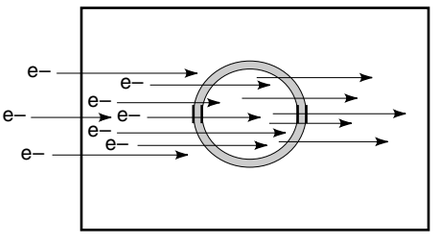

Figure 1. Illustration of electronic equilibrium as applied to radiation dosimetry. The gas-filled cavity must be small in relation to the range of the electrons generated by photon interactions in the medium. Electrons that are produced in the cavity, and leave the cavity carrying away energy, must be replaced by electrons generated in the medium which enter the cavity and deposit energy.

electronic equilibrium is employed in the free air ionization chamber (see discussion below).

4. Energy deposition by ionizing radiation must be uniform throughout the solid medium immediately surrounding the gas-filled cavity.

If these requirements are met, the energy absorbed per unit mass of the medium is related to the ionization per unit mass of the gas in the cavity by:

$$E_m = J_g \times W \times s_m \tag{5}$$

where J_g is the number of ion pairs formed per unit mass of the gas (usually expressed in units of grams); W is the average energy required to produce an ion pair in the gas; and s_m is the ratio of the mass stopping power of the medium to that of the gas in the cavity for the secondary electrons. As mentioned previously, the average energy required to produce an ion pair in most gases is about 34 eV. The factor s_m can be expressed as:

$$s_m = \frac{N_m \times S_m}{N_g \times S_g} \tag{6}$$

where N_m is the number of electrons per unit mass of the medium; N_g is the number of electrons per unit mass of the gas; S_m is the electron stopping power of the medium; and S_m is the electron stopping power of the gas. The factor s_m indicates how much more frequently ionization will occur in the medium as compared to the gas in the cavity. Therefore, measurement of the factor J_g, the ionization per unit mass of the gas in the cavity, linked with the knowledge of the values s_m and W makes it possible to determine the energy deposited (i.e., to determine the absorbed dose) in the medium. If the medium of interest is tissue, then the Bragg–Gray principle allows measurement of the absorbed dose in the irradiated tissue.

THE FREE AIR IONIZATION CHAMBER

One of the first detectors used to measure the quantity exposure due to gamma and x ray sources was the ionization chamber (see IONIZATION CHAMBERS). Early ionization chambers employed metal electrodes and crude insulators, were filled with air, and were very simple in construction. The free air ionization chamber provides a good example of the simplicity of such detectors and, at the same time, illustrates the principles so necessary to measure the deposited energy. This detector is designed to measure exposure over a specific photon energy range and serves as a standard device at national standards institutes across the world.

Basically, the free air ionization chamber is a parallel-plate detector with plate separation being one of the important variables (see Fig. 2). A potential difference is maintained between the high voltage electrode and the collecting electrode. The collecting electrode is surrounded by a grounded guard electrode to clearly define the electric field shape and limits. At one end of the detector is a collimator which serves to define the radiation beam (S) as precisely as possible. Between the plates of the detector, the collecting volume is defined by the collimator and the electric field lines

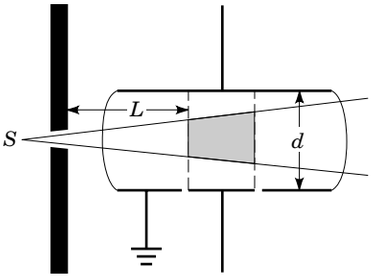

Figure 2. Schematic drawing of a free-air ionization chamber. The collimated photon source is located at S, the distance L is the necessary "thickness" of air to establish electronic equilibrium, and the plate separation is indicated by d. As the photon energy increases both L and d must increase; thus, these detectors are designed for a specific range of photon energies.

between the two electrodes. The distance between the collimator and the collecting volume (L) also is variable to ensure that electronic equilibrium exists in this volume. As the energy of the photon beam is increased the collimator must be moved farther away from the collecting volume (the range of the secondary electrons increases), and the plate separation (d) must be increased to accommodate the increase in the effective collecting volume of the chamber. These detectors have inherent limitations due to electronic equilibrium requirements which dictate changes in the plate separation and the distance between the collimator opening and the collecting volume. For some photon energies, these detectors, with appropriate shielding, can be quite large. For these reasons, free air ionization chambers are manufactured to be used in specific photon energy ranges and their use is restricted to standards laboratories.

Some corrections must be made in the use of the free air ionization chamber. These corrections include:

- Attenuation of the photons in the air between the collimator opening and the collecting volume;
- Recombination of ion pairs in the chamber;
- Changes in air density and humidity;
- Ionization produced by photons scattered from the beam; and
- Loss of ionization due to inadequate separation of the electrodes.

However, if these corrections are made, it is possible to make measurements of radiation exposure to about $\pm 0.5\%$.

In dosimetry it is not always possible to establish electronic equilibrium in the manner used in the free air ionization chamber. Usually, the outer wall of the ionization chamber is constructed using solid materials which provide an equilibrium thickness for a specific photon energy range (see Fig. 3). In addition, equivalent materials are used in some applications. That is, materials which are classified as *air-equivalent* or *tissue-equivalent* are used in certain specific dosimetry applications. This is accomplished by selecting a material which attenuates the radiation in the same manner as the equivalent material.

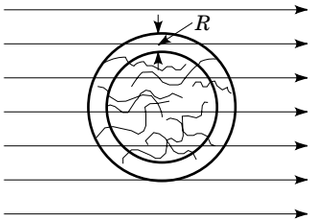

Figure 3. The "air-wall" of an ionization chamber condensed around the collecting volume of the detector. The thickness of the wall, R, is essentially equal to the distance, L, in Fig. 2.

GENERAL CONSIDERATIONS

The basic requirement of any radiation dosimeter is that it measure the dose received (i.e., energy deposited) with sufficient reproducibility and accuracy over the entire range of radiation energies, doses, or dose rates expected during its use. The dosimeter may be a standard device used to characterize a particular field (e.g., calibration of an X-ray machine) or it may be a monitoring device worn by a radiation worker to establish the occupational dose. The accuracy of the dosimeter may vary depending on the intent of the measurement and the dose levels to which the dosimeter is exposed. For example, national and international guidance on personnel monitoring indicate that, for routine exposures, an accuracy of $\pm 50\%$ is acceptable. As the exposure level increases, the required accuracy of the dosimeter becomes more restrictive and, as the doses approach the permissible exposure levels, the accuracy should be $\pm 30\%$. At higher exposure levels, such as those approaching clinical significance (i.e., life-threatening), the desired accuracy becomes $\pm 25\%$. However, these levels of accuracy would be totally unacceptable in a radiation therapy situation where accuracies of a few percent are required. A clear understanding of the specific dosimeter requirements and the ability of the dosimeter to meet these requirements is important in any radiation dosimetry program.

Regardless of the intent of the dosimetry program, it is imperative that the performance characteristics and limitations of the dosimeters be completely understood by those responsible for the program. Quantitative measurements made with a particular type of dosimeter will depend on a number of factors including:

- Variation of dosimeter response from the ideal;
- Reliability with which the dosimeter maintains its calibration or retains the recorded information; and
- Influence of environmental factors on dosimeter response.

Factors affecting dosimeter response include radiation quality, radiation intensity, energy dependence, angular dependence, and the presence of other radiations. Terms such as fading or leakage are used to describe the loss of information stored or recorded by the dosimeter. Environmental factors include temperature, humidity, dust, vapors, light, and other factors such as rough handling and radiation contamination. All of these factors may influence measurements of the absorbed dose and, potentially, lead to invalid monitoring results.

No dosimeter meets all the requirements for an ideal dosimeter. But, the strengths and weaknesses of the system in use must be completely understood to ensure proper use and interpretation of the results obtained.

IONIZATION CHAMBERS

There are two broad categories of ionization chambers used in radiation detection: passive and active (a more detailed discussion is presented in IONIZATION CHAMBERS). Passive detectors have been applied routinely to monitoring radiation exposure of individuals engaged in activities at many types of nuclear facilities. Passive dosimeters are integrating detectors since they provide only an indication of the total exposure. No dose rate information is given (but total dose divided by exposure time provides average dose rate) and, if the dosimeter is exposed beyond its useful range, no useful information is provided. Active detectors are normally used for radiation surveys in the workplace and will not be discussed here.

Some passive detectors require a number of steps to be taken to secure an indication of the radiation exposure and, ultimately, the absorbed dose. These dosimeters are often called indirect-reading or condenser-type dosimeters. Basically, the dosimeter is a right-circular cylinder capacitor. The outer electrode is a right circular cylinder of conducting material and it surrounds, but is insulated from, a central electrode (usually a thin wire). The volume between the two electrodes is filled with air. The dosimeter must be prepared for use (charged) using an external circuit, exposed to radiation, and evaluated (usually in the same device used to charge the dosimeter). The measured exposure can be obtained only at the end of the irradiation and the dosimeter has been removed from the radiation field. The exposure can be evaluated from the relationship:

$$CV = Q \tag{7}$$

where C is the electrical capacitance of the chamber; V is the change in voltage before and after the exposure; and Q is the charge collected during the exposure. If the chamber volume is known, the exposure can be calculated based on the definition of exposure. Usually, this is not necessary since the charger-reader, used to prepare the dosimeter for use, is calibrated to read directly in units of exposure. These dosimeters typically have a useful range of 0 to 200 mR with a quoted accuracy of $\pm 15\%$ over a photon energy range of a few keV up to 3 MeV. However, dosimeters of this type have been manufactured to cover a number of exposure ranges. Although it is seldom done, a conversion factor can be applied to relate exposure to the absorbed dose.

Certain types of the condenser-type dosimeters are used as secondary standard devices for calibration of the output of x-ray machines, radiation therapy sources, and radiation detector calibration sources. These detectors, often called R-chambers, are manufactured with different wall thicknesses and compositions for use in a wide range of photon fields. Usually the wall thickness is air equivalent and the thickness of the wall is adjusted for a particular photon energy range. For example, R-chambers are available for use in the energy range 6 keV to 35 keV and also for photons in the range 0.25 MeV to 1.4 MeV. Exposure range is normally controlled by selecting the chamber volume. Detectors are available for the mea-

surement of exposure from 0.001 R up to 1000 R. These detectors also are available as active detectors coupled to a high voltage/electrometer for immediate readout of the exposure rate.

Other passive dosimeters (called *direct-reading* or *self-reading* dosimeters) have incorporated a lens system into the dosimeter which allows the exposure to be evaluated visually by the wearer without the use of an external circuit. These detectors have a sealed collecting volume, filled with air, and a single electrode made in two pieces. One piece is relatively rugged and stationary, while the other electrode is made of a thin, conducting fiber (i.e., a quartz fiber), which is moveable in relation to the stationary electrode. In this dosimeter, the electrodes are charged to the same polarity and, thus, the stationary electrode and the fiber repel each other. The dosimeter is prepared for use (charged) by adjusting the voltage so that the moveable electrode casts a shadow at the zero position on an internal scale. When the dosimeter is exposed to radiation, ion pairs produced in the sensitive volume reduce the charge and the fiber moves closer to the stationary electrode. The wearer can view the change in position of this shadow through a magnifying lens system incorporated into the dosimeter, while holding the dosimeter up to any light source. This movement is directly related to the exposure received.

The direct-reading dosimeters typically are very rugged and are preferred over the indirect-reading devices. At one time, these detectors were used extensively in the nuclear power facilities to provide day-to-day monitoring of the work force. These dosimeters have been manufactured for use in similar photon energy and exposure ranges and have the same accuracy, etc., as quoted previously. Some special dosimeters have been manufactured for low-energy photons and for detecting thermal neutron radiation. However, these are special applications of the general technology and these uses are not wide spread.

Currently, passive dosimeters are not used for monitoring personnel radiation exposure in the workplace. These dosimeters have been replaced by thermoluminescence dosimeters and, more recently, by sophisticated electronic dosimeters.

THERMOLUMINESCENCE DOSIMETRY

Thermoluminescence dosimetry (TLD) is a common and popular method applied to the measurement of personnel radiation exposure. These dosimeters are used widely in nuclear facilities in the United States and, in many cases, are used to establish the "dose of record" to satisfy regulatory requirements. TLDs have many of the characteristics of the ideal dosimeter but there also are certain characteristics which influence the response of the dosimeter and significantly affect the results obtained. These characteristics must be completely understood for TLDs to be used effectively as a radiation dosimeter (9).

Some authors state that thermoluminescence (TL) has been observed for centuries, whenever certain limestones and fluorites were heated. Certainly some of the early research dates back to that of Sir Robert Boyle and others in the mid 1600s. However, it was not until the early 1950s that Daniels proposed the use of TL as a radiation dosimeter. The relation between x-ray exposure and TL had been reported in 1904 but, interestingly, no one proposed the logical application of TL to the measurement of radiation dose.

In TLD the absorbed dose is determined by observing the light emitted by the previously exposed TLD phosphor (or crystal) as it is heated in a controlled manner. The light emitted is directly proportional to the radiation energy deposited in the phosphor and, thus, to the absorbed dose. However, no TLD phosphor is an absolute dosimeter, the phosphor and the evaluation system must be carefully calibrated to establish the relationship of the light emitted and the absorbed dose. Careful calibration is required of each TL material, the badge or holding device in which the TLDs are placed, and the particular system in which the material is to be evaluated. Even though the material may be the same, each batch of TL material may have slightly different characteristics which affect its response to radiation. In addition, there are a number of possible approaches to evaluating the TLDs and each evaluation system has very specific characteristics. Calibration of a TLD system is a major task which must be completed before the system can be put in routine use for radiation dosimetry.

Some authors would suggest that the physical and chemical theories of TL are poorly understood. However, the basic phenomenon can be understood qualitatively using an explanation founded in solid-state physics. This approach is perhaps too simple when one considers the complexity of TL emission but it will serve to illustrate the fundamental processes. Consider a hypothetical energy-level diagram of an insulating material, in which the valence band is assumed to be filled and the conduction band is assumed to be empty (see Fig. 4). If this insulating phosphor is exposed to ionizing radiation, interactions of the radiation in the phosphor will free electrons from their respective atoms (ionization). These electrons will be raised from the valence band into the conduction band. The loss of electrons in the valence band creates holes, (i.e., positively charged atoms or *sites*). The electrons and holes may migrate through the phosphor until they recombine or are trapped in metastable states. These metastable states may be associated with defects in the crystal structure or, to facilitate the trapping, impurity materials may be added intentionally to TLD phosphors. These materials typically occupy interstitial positions in the lattice and, in our energy

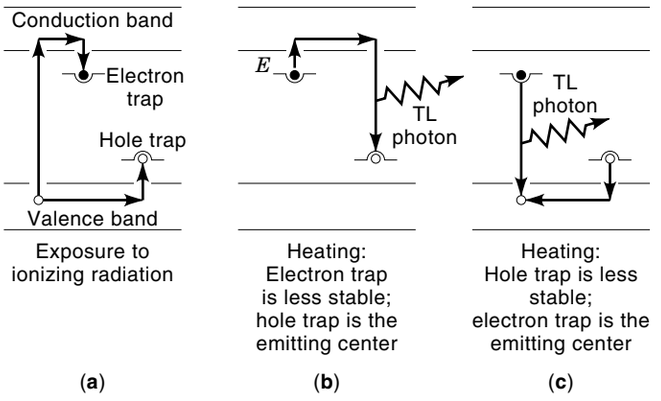

Figure 4. Schematic energy level diagram of an insulating crystal exhibiting thermoluminescence. The "traps" are located in the forbidden zone and retain the trapped electrons or "holes" until thermally stimulated.

diagram, occupy energy levels in the forbidden zone between the conduction and valence bands.

These traps prevent the electrons from returning to the valence band and, in effect, the energy which raised the freed electrons is stored in the phosphor. If the phosphor can be stimulated so that the energy is released, and that energy can be measured, the phosphor can be used as a radiation dosimeter. Usually, the stimulation is through heating the phosphor in a controlled manner (thermo) and the stored energy is released in the form of visible light (luminescence). The measured emitted light is directly proportional to the energy deposited in the phosphor and, thus, the radiation absorbed dose.

Actually, the stored energy may be released through two possible mechanisms. First, as the phosphor is heated, the trapped electrons may receive sufficient energy to release them from their traps, raising them back into the conduction band. These electrons may return to the valence band, recombine with a hole, and release the stored energy in the form of a luminescence photon. The light photon released has an energy proportional to the difference between the excited and stable electron energy levels. Second, the hole trap may be more unstable and heating of the phosphor provides sufficient energy for the hole to wander through the crystal until it can combine with a trapped electron. Again, when the hole and the electron combine, a luminescence photon is released. Often, since the two processes are similar, only the first possibility is mentioned in simplified discussions of TL theory.

The temperature required to free the electrons and cause the emission of light is related to the energy gap between the valence and conduction bands. When exposed to radiation, the deposited energy produces many trapped electrons and holes. As the temperature of the phosphor is increased in a controlled manner, the probability of releasing electrons is increased. Finally, a temperature is reached at which all the electrons have been released. Thus, the emitted light from the heated phosphor will be weak at low temperatures, pass through one or more maxima as the temperature increases and, finally, decrease again to zero. A plot of the emitted light versus the heating temperature is called a *glow curve.* More recently, as the heating cycle has become more controllable and reproducible, a glow curve may be a plot of the emitted light as a function of time (where there is a constant, known heating rate). Typical glow curves show one or more maxima (called *glow peaks*) as traps at several energy levels are emptied. The relative amplitude of these peaks indicates approximately the relative populations of electrons in the various traps.

For radiation dosimetry, either the total light emitted during a selected part or all of the heating cycle, or the height of one or more of the glow peaks, may be used to indicate the absorbed dose. Often, a single peak (called the *dosimetry peak*) is selected for use in the evaluation. Usually, the dosimetry peak is one of the more stable peaks. However, in some phosphors, only a single glow peak is present (e.g., $Al_2O_3:C$ and $CaSO_4:Mn$). Again, proper calibration is required of both the phosphor material and the device used to evaluate the TLD.

There are a large number of materials which exhibit TL and many have been studied as potential radiation dosimeters. Several of the most popular TLD materials include $CaSO_4:Mn$, $CaSO_4:Dy$, $CaF_2:Mn$, $CaF_2:Dy$, $LiF:Mg:Ti$, $LiB_4O_7:Mn$, and $Al_2O_3:C$. Of these TLD materials, LiF has been the most widely used material. This phosphor will be discussed in some detail to illustrate the considerations important to dosimetry.

Commercially available LiF (commonly called TLD-100) has been studied extensively because of its excellent characteristics for use as a radiation dosimeter. For photon radiation, these include:

- A relative constant energy response per unit exposure over a wide range of photon energies. At low energies, LiF exhibits an over-response of about 25% in the 30 keV to 40 keV energy range. In other TLD materials, this over-response can be as high as factors of 10 to 15. In addition, in special cases, the energy over-response can be reduced by using a simple energy-compensating shield.
- Even though LiF has a density of 2.64 g cm^{-3}, the effective atomic number of LiF is about 8.2 which makes this TLD material nearly tissue-equivalent (Z = 7.4 to 7.6). Other TLD materials have an effective atomic number in the range of 12 to 15.
- The main dosimetry peak ($\cong$190°C) is extremely stable and shows little loss of information (fading) when stored at room temperature. Fading is estimated to be only about 5% per year. LiF actually exhibits up to six glow peaks of various magnitudes but pretreatment annealing (i.e., heating) can reduce the influence of these less stable peaks on the measured results.
- The phosphor is useful over a wide range of exposures, typically from a few tens of mR up to hundreds to thousands of R. The actual range will depend on a number of factors which must be determined during calibration.

LiF has been used in a number of dosimetry and monitoring applications, from measurement in high-dose cancer therapy situations to many personnel monitoring applications.

LiF also has applications in neutron dosimetry because the phosphor is available in three separate formulations. LiF (TLD-100) has the natural isotopic mix of the two lithium isotopes, ^6Li (7.4%) and ^7Li (92.6%). However, two other phosphors are available, called TLD-600 and TLD-700. TLD-600 is highly enriched in the isotope ^6Li (95.6%), which has a very high thermal neutron cross-section (about 945 barns) for the (n,α) reaction. In contrast, TLD-700 is made essentially of pure ^7Li (99.96%). This isotope of lithium has essentially no sensitivity to thermal neutrons (the cross-section is 0.033 barns).

In this application, two different LiF phosphors are employed (either TLD-100 with TLD-700 or TLD-600 with TLD-700). The latter combination is preferred and will be used in the following discussion. Since TLD-600 has a sensitivity to thermal neutrons as well as photons the output signal (the glow curve) from this phosphor will represent the contributions to dose from both radiations. Since TLD-700 has essentially no sensitivity to neutrons, the output signal from this detector represents only the dose due to photons. Subtraction of the photon dose measured by TLD-700 from the neutron and photon dose measured by TLD-600 provides an estimate of the thermal neutron component of the mixed radiation

field. As with most radiation dosimetry systems, these detectors require proper calibration before use.

This technique is not limited to measurement of thermal neutron dose. The dose due to fast neutrons can be measured using the same approach. However, here the TLDs must be covered with material to absorb (i.e., capture) thermal neutrons incident on the dosimeter and alter the TLD response. One popular detector system is called the albedo dosimeter. This dosimeter takes advantage of the fact that fast neutrons incident on the human body may be moderated (i.e., slowed down) by the tissue and reflected out of the body at lower energies, in the incident direction. The albedo dosimeter is designed to measure the thermal neutrons that escape the body when it is irradiated with fast neutrons and, through calibration, provides an estimate of the fast neutron dose. Many different designs of an albedo dosimeter, incorporating any number of TLDs and a variety of materials, have been reported in the literature.

ELECTRONIC DOSIMETERS

Currently, electronic dosimeters are used widely in radiation safety/personnel monitoring applications around the world. Originally, the idea of a small electronic device, which could be worn comfortably by a worker, was restricted to detectors which simply issued an alarm (or "chirped") to warn the wearer of a potentially unknown radiation field. Even though some attempted to quantitatively relate the "chirp-rate" to the dose rate, the primary use of the device was simply to provide a warning. Obviously, this restricted use was a direct consequence of the inability to build small radiation dosimetry systems.

With the advent of microelectronics, electronic dosimeters now have replaced the direct-reading dosimeters which were used extensively in many nuclear facilities. In addition, the rapid development and incorporation of a number of attractive features into these dosimeters have caused many to consider replacing TLDs with these devices. Should this occur, the electronic dosimeter will become the dosimeter of record in terms of satisfying the regulatory requirements for personnel radiation monitoring.

This rapidly growing popularity is due to a number of features which allow better control of radiation exposure in the workplace and facilitate record-keeping. The attraction of electronic dosimeters is that they usually are coupled to a computer, through a reading device, which allows the setting of alarm points before use and the interrogation of the device after use. The data obtained from the dosimeter can be routed directly to the computer records system and, ultimately to the exposure file of the wearer. These devices will record, and display for the wearer, the accumulated dose or the dose-rate to which the worker has been exposed. In some cases, several alarm set-points are available for use.

Typically, electronic dosimeters are designed to detect photons, but dosimeters for beta and neutron radiation are being developed. Several different radiation detectors have been used in these dosimeters. The simplest detectors are small (about 1 cm^3 volume), halogen-quenched, energy-compensated Geiger–Mueller tubes. These detectors operate at about 550 V. Other dosimeters use *pin* diodes or silicon semiconductor detectors which require significantly lower voltages (about 1 V to 4 V). Typical energy response for these dosimeters is ±30% over a photon energy range of 60 keV to 1.2 MeV and the accumulated exposure range is typically 0 mR to 9,999 mR. Dosimeters are exposure rate limited, although at a very high rate, to the range of 10 R/h to 1000 R/h, depending on the detector selected for use. The entire unit is quite small, the weight of typical units ranges from 100 g to 200 g. Battery life has been reported to range from 100 h to about 6 months of use. Some dosimeters offer the capability of using telemetry so that a single radiation protection technician can monitor the activities of several workers in different locations in the facility.

However, since this field is developing rapidly, the quality of these detectors can be quite variable and many anomalies have been reported in the literature. These include wide variations in the photon energy response of detectors which are identical, poor quality control on some of the software, loss of stored information, audible alarm failures, and radiofrequency interference, to list only a few reported problems (10).

GLOSSARY

Absorbed dose. The amount of energy deposited by ionizing radiation in a material per unit mass of the material. Usually expressed in the SI unit, the gray, 1 Gy = 1 J/kg. The traditional units for absorbed dose is the rad, 1 rad = 0.01 J/kg; therefore, 1 Gy = 100 rad.

Cross section (σ). A quantitative measure of the probability of a given nuclear reaction. The concept of a nuclear cross section can best be visualized as the cross-sectional area, or "target area," presented by a nucleus to the incident particle. The unit associated with the cross section is the *barn* where 1 b = 10^{-24} cm^2.

Directly ionizing radiation. Radiation composed of charged particles that interact directly with the electrons in the medium through coulombic interactions. These radiations include alpha particles, beta particles, positrons, electrons, and protons.

Dose equivalent. The product of the absorbed dose and the quality factor. This quantity is used to express the effects of the absorbed dose from the many types of ionizing radiation on a common scale. The SI unit for dose equivalent is the sievert, 1 Sv = 1 J/kg, and therefore, 1 Sv = 100 rem. The traditional unit for dose equivalent is the rem. More recently, the International Commission on Radiological Protection has renamed this quantity the equivalent dose and has redefined it as the product of the absorbed dose and the radiation weighting factor, W_R.

Dosimeter. A device that may be worn or carried by an individual into a radiation field for the purpose of measuring the absorbed dose or the dose equivalent. The dosimeter may be either active or passive. In some cases, the dosimeter may measure the total dose over the exposure period and, in others, the dosimeter may provide dose rate information.

Effective dose equivalent. The sum of the products of the dose equivalents in organs and tissues of the body, H_T, and the individual tissue weighting factors, W_T.

Exposure. An early quantity defined as the total charge produced in air by photons interacting in a volume of air with a known mass. The currently accepted SI unit is C/kg and the

traditional unit was originally the roentgen, 1 R = 2.58 × 10^{-4} C/kg. This term also is used, in general, to indicate any situation in which an individual is in a radiation field.

Fluence. The number of particles incident on a sphere of a specified cross-sectional area. The unit associated with this quantity is m^{-2}.

Individual dose equivalent, penetrating, $H_P(d)$. The dose equivalent in soft tissue below a specified point on the body at a depth, d, that is appropriate for strongly penetrating radiation. The recommended depth, d, for monitoring in terms of $H_P(d)$ is 10 mm, and $H_P(d)$ may be written as $H_P(10)$.

Individual dose equivalent, superficial, $H_s(d)$. The dose equivalent in soft tissue below a specified point on the body at a depth, d, that is appropriate for weakly penetrating radiation. The recommended depth, d, for monitoring in terms of $H_s(d)$ is 0.07 mm, and $H_s(d)$ may be written as $H_P(0.07)$.

Indirectly ionizing radiation. Radiation composed of uncharged radiations that must interact with the material producing charged particles in order to transfer energy to the medium. These radiations include, among others, X rays, gamma rays, and neutrons).

Ionization. In this context, the process of removing (or adding) one or more electrons from (or to) an atom or molecule through interactions of radiation with the medium.

Ion pair. The resulting positively charged nucleus and the free electron produced by the interaction of ionizing radiation with the medium.

Isotope. One of two or more atoms of an element with the same number of protons in the nucleus but with different numbers of neutrons, such as C-12, C-13, and C-14, are isotopes of the element carbon. A radioisotope is an unstable isotope of an element that undergoes a spontaneous nuclear transformation by the emission of nuclear particles or electromagnetic radiation to reach a more stable nuclear state. This transformation usually produces an atom of a different element. A radioisotope is often called a radionuclide, although radionuclides are not necessarily isotopes.

Kerma. The sun of the initial kinetic energies of all charged ionizing particles liberated by uncharged ionizing particles in a specific mass of material. The units for the absorbed dose may be used for this quantity (i.e., gray or rad). Absorbed dose and kerma may be equal depending on the degree of charged particle equilibrium and bremsstrahlung production.

Linear energy transfer. The mean energy lost by a particle, due to collisions with electrons, in traversing a specified distance. The unit for this quantity is J m^{-1}.

Mass stopping power. The rate of energy loss (i.e., dE/dx) by charged particle radiation traversing the medium divided by the density of the medium.

Neutron. A fundamental constituent of the nucleus of an atom. Neutrons can be produced in a number of ways and represent a significant source of indirectly ionizing radiation. Neutrons are classified according to their energy. A *thermal* neutron is at thermal equilibrium with its environment and, in special cases, has a Maxwellian distribution of velocities. In this distribution, the most probable velocity at 295 K is 2200 m s^{-1} corresponding to an energy of 0.025 eV. *Intermediate* neutrons have energies in the range of 0.5 to 10 keV. These neutrons are also called resonance or epithermal neutrons. *Fast* neutrons have energies in the range 10 keV to 10 MeV. Neutrons in this energy range interact with material primarily through elastic collisions (i.e., billiard-ball collisions). *Relativistic* (or high-energy) neutrons have energies in excess of 10 MeV. These neutrons interact with material primarily through inelastic collisions.

Quality factor. A factor to weight the absorbed dose for the biological effectiveness of the radiation producing the absorbed dose. Currently accepted quality factors are: beta particles, positrons, electrons, x rays, gamma rays, and bremsstrahlung, $Q = 1$; protons, fast neutrons, $Q = 10$; alpha particles, recoil nuclei, $Q = 20$. Note that both the national and international bodies have recommended new factors (now called radiation weighting factors) but these have not been adopted for use in the federal regulations of the United States.

Radiation. In this context, this term is used to designate ionizing radiation. Ionizing radiation has sufficient energy to cause ionization of the atoms and molecules with which the radiation interacts.

Radiation weighting factor (w_R). A factor used to weight the absorbed dose for the radiation effectiveness in a similar fashion to the earlier use of quality factors.

Response. The ratio of the indicated reading of a radiation detection instrument to the actual value of the quantity being measured. Usually, the calibration factor is the reciprocal of the response.

Specific ionization. The number of ion pairs produced by charged particles per unit length of travel in the medium, usually expressed in units of ion pairs per centimeter.

Stopping power. The rate of energy loss (i.e., dE/dx) by charged particle radiation traversing the medium.

Tissue weighting factor (w_T):. A factor which expresses the fraction of the total stochastic risk associated with the irradiation of a particular tissue. The risk is based on the probability of producing a fatal cancer, a nonfatal cancer, hereditary effects, and shortening of lifespan in the exposed individuals.

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DQDB. See METROPOLITAN AREA NETWORKS.