Nuclear Fusion and Radiation

Lecture 9 (Meetings 24 & 27)

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Radiation Interactions with Matter

- We live in an environment awash in radiation: Radiation emitted by radioactive nuclides, electromagnetic radiation of all wavelengths, photons (far more abundant than matter in our universe), cosmic rays, neutrinos.
  - Most of this radiation such as neutrinos, light and longer wavelength electromagnetic radiation, fortunately, usually passes/interacts harmlessly through/with our tissues.
  - Shorter wavelength electromagnetic radiation (ultraviolet light, X rays, and gamma rays), neutrons and charged particles produced by nuclear reactions can cause various degrees of damage to our cells.

Radiation Interactions with Matter

- For radiation to produce biological damage, it must first interact with the tissue and ionize cellular atoms, which, in turn, alter molecular bonds and change the chemistry of the cells.

- Ionizing radiation is subdivided into two classes:

  - **directly ionizing radiation** whose interactions produce ionization and excitation of the medium. Fast moving charged particles, such as alpha particles, beta particles, and fission fragments, can ionize matter.

  - **indirectly ionizing radiation** that cannot ionize atoms but can cause interactions whose charged products, known as secondary radiation, are directly ionizing. Neutral particles, such as photons and neutrons, cannot interact Coulombically with the electrons; rather, they transfer some of their incident kinetic energy to charged secondary particles.

Attenuation of Neutral Particle Beams

- Unlike charged particles, neutral particles move in straight lines through a medium, punctuated by occasional “point” interactions, in which the neutral particle may be absorbed or scattered or cause some other type of reaction (interaction is dominated by short-range forces).

- The interaction of a given type of neutral radiation with matter may be classified according to the type of interaction and the matter with which the interaction takes place:

  – The interaction may be a scattering of the incident radiation accompanied by a change in its energy. A scattering interaction may be elastic or inelastic.

  – The interaction may be an absorption of the incident radiation. The identity of the incident particle is lost, but total relativistic momentum and energy are conserved.

Attenuation of Neutrals: Linear Interaction Coefficient

- The interaction of radiation with matter is always statistical in nature, and, therefore, must be described in probabilistic terms.

- Consider a particle traversing a homogeneous material and let $P_i(\Delta x)$ denote the probability that this particle, while traveling a distance Δx in the material, causes a reaction of type i (e.g., it is scattered).

- It is found empirically that the probability per unit distance traveled, $P_i(\Delta x)/\Delta x$, approaches a constant as Δx becomes very small, i.e.,

$$\mu_i = \lim_{\Delta x \to 0} \frac{P_i(\Delta x)}{\Delta x} \tag{1}$$

  where the limiting process is performed in the same average statistical limit as was used in the definition of the radioactive decay constant λ.

- The quantity μ_i is a property of the material for a given incident particle and interaction.

Attenuation of Neutrals: Linear Interaction Coefficient

- In the limit of small path lengths, μ_i is seen to be the probability, per unit differential path length of travel, that a particle undergoes an ith type of interaction.
- The constant μ_i is called the linear coefficient for reaction i and often referred to as the macroscopic cross section for reactions of type i.
- For each type of reaction, there is a corresponding linear coefficient. For, example, μ_a is the linear absorption coefficient, μ_s the linear scattering coefficient, and so on.
- The "total" probability, per unit path length, that a neutral particle undergoes some sort of reaction, μ_t , is the sum of the probabilities, per unit path length of travel, for each type of possible reaction, i.e.,

$$\mu_t(E) = \sum_i \mu_i(E) \tag{2}$$

- Note the dependance on the particle's kinetic energy!

Attenuation of Neutrals: Linear Interaction Coefficient

- The total interaction probability per unit path length, μ_t, is fundamental in describing how *indirectly-ionizing* radiation interacts with matter and is usually called the linear attenuation coefficient.

- It is perhaps more appropriate to use the words total linear interaction coefficient since many interactions do not "attenuate" the particle in the sense of an absorption interaction.

Attenuation of Neutrals: Uncollided Radiation Attenuation

- Consider a plane parallel beam of neutral particles of intensity I^o $\frac{\text{particles}}{\text{cm}^2\text{s}}$ normally incident on the surface of a thick slab (see Fig. below).

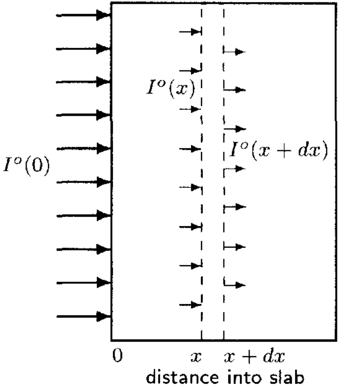

Figure 1: Uniform illumination of a slab by radiation.

Attenuation of Neutrals: Uncollided Radiation Attenuation

- As the particles pass into the slab some interact with the slab material.

- Of interest in many situations, is the intensity $I^o(x)$ of uncollided particles at depth x into the slab.

- At some distance x into the slab, some uncollided particles undergo interactions for the first time as they cross the next Δx of distance, thereby reducing the uncollided beam intensity at x, $I^o(x)$ to some smaller value $I^o(x+\Delta x)$ at $x+\Delta x$.

- The probability an uncollided particle interacts as its crosses Δx is

$$P(\Delta x) = \frac{I^o(x) - I^o(x+\Delta x)}{I^o(x)} \tag{3}$$

Attenuation of Neutrals: Uncollided Radiation Attenuation

- In the limit as $\Delta x \to 0$, we have

$$\mu_t = \lim_{\Delta x \to 0} \frac{P(\Delta x)}{\Delta x} = \lim_{\Delta x \to 0} \frac{I^o(x) - I^o(x + \Delta x)}{\Delta x} \frac{1}{I^o(x)} = -\frac{dI^o(x)}{dx} \frac{1}{I^o(x)} \quad (4)$$

which implies that

$$\frac{dI^o(x)}{dx} = -\mu_t I^o(x) \quad (5)$$

Remark: We obtained similar equation when we studied the "idealized" accelerator-based fusion reactor.

- This has the same form as the radioactive decay equation. The solution for the uncollided intensity is

$$I^o(x) = I^o(0) e^{-\mu_t x} \quad (6)$$

- Uncollided indirectly-ionizing radiation is thus exponentially attenuated as it passes through a medium.

Attenuation of Neutrals: Uncollided Radiation Attenuation

- From this result, the interaction probability $P(x)$ that a particle interacts somewhere along a path of length x is

$$P(x) = 1 - \frac{I^o(x)}{I^o(0)} = 1 - e^{-\mu_t x} \qquad (7)$$

- The probability $\bar{P}(x)$ that a particle does not interact while traveling a distance x is

$$\bar{P}(x) = 1 - P(x) = e^{-\mu_t x} \qquad (8)$$

- As $x \to dx$, we find $P(dx) \to \mu_t dx$, which is in agreement with the definition of μ_t.

Attenuation of Neutrals: Distance Before An Interaction

- From the above results, the probability distribution for how far a neutral particle travels before interacting can be derived.

- Let $p(x)dx$ be the probability that a particle interacts for the first time between x and $x + dx$. Then,

$$p(x)dx = \bar{P}(x)P(dx) = e^{-\mu_t x}(1 - e^{-\mu_t dx}) = e^{-\mu_t x}\mu_t dx = \mu_t e^{-\mu_t x} dx \quad (9)$$

- Note that $\int_0^\infty p(x)dx = 1$, as is required for a proper probability distribution function.

- This probability distribution can be used to find the average distance x traveled by a neutral particle to the site of its first interaction, namely, the average distance such a particle travels before it interacts.

Attenuation of Neutrals: Distance Before An Interaction

- The average of x is

$$\bar{x} = \int_0^\infty x p(x) dx = \mu_t \int_0^\infty x e^{-\mu_t x} dx = \frac{1}{\mu_t} \tag{10}$$

- This average travel distance before an interaction, $1/\mu_t$, is called the mean-free-path length.

- The total linear attenuation coefficient μ_t can be interpreted, equivalently, as
  (1) the probability, per unit differential path length of travel, that a particle interacts, or
  (2) the inverse of the average distance traveled before interacting with the medium.

Remark: Note the analogy to radioactive decay, where the decay constant λ is the inverse of the mean lifetime of a radionuclide.

Attenuation of Neutrals: Half-Thickness

- To shield against *photons or neutrons*, a convenient concept is that of the half- thickness, $x_{1/2}$, namely, the thickness of a medium required for half of the incident radiation to undergo an interaction.

- For the uncollided beam intensity to be reduced to one-half of its initial value, we have

$$\frac{1}{2} = \frac{I^o(x_{1/2})}{I^o(0)} = e^{-\mu_t x_{1/2}} \tag{11}$$

from which we find

$$x_{1/2} = \frac{\ln 2}{\mu_t} \tag{12}$$

Remark: Note the similarity to the half-life of radioactive decay.

Attenuation of Neutrals: Shield Thickness

Example: What is the thickness of a water shield and of a lead shield needed to reduce a normally incident beam of $1-MeV$ photons to one-tenth of the incident intensity?

Water: $\mu(1-MeV)=0.07066cm^{-1}$. Lead: $\mu(1-MeV)=0.7721cm^{-1}$.

Attenuation of Neutrals: Shield Thickness

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Water: $\mu(1 - MeV) = 0.07066 cm^{-1}$. Lead: $\mu(1 - MeV) = 0.7721 cm^{-1}$.

Solution:

For the beam intensity to be reduced to one-tenth of its initial value, we have

$$\frac{1}{10} = \frac{I^o(x_{1/10})}{I^o(0)} = e^{-\mu_t x_{1/10}} \tag{13}$$

from which we find

$$x_{1/10} = \frac{\ln 10}{\mu_t} \tag{14}$$

Therefore, with this μ_t values we find $x_{1/10} = 32.59cm = 12.8in$ for water and $x_{1/10} = 2.98cm = 1.17in$ for lead.

Radiation Interaction Rates: Flux Density

- To quantify the “strength” of a radiation field, we can use several measures.
- One of them is the **intensity**, which is the flow (number per unit time) of radiation particles that cross a unit area perpendicular to the beam direction. The units of intensity are, for example, $cm^{-2}s^{-1}$. However, in most radiation fields the particles are moving in many different directions and the concept of beam intensity loses its usefulness.
- Another measure, which is very useful for calculating reaction rates, is the particle **flux density** (see definition of cross section in Lecture 3) defined as

$$\phi(\mathbf{r}) = vn(\mathbf{r}) \tag{15}$$

Because v is the distance one particle travels in a unit time and, because $n(\mathbf{r})$ is the number of particles in a unit volume, we see that the product $vn(\mathbf{r})$ is the distance traveled by all the particles in a unit volume at $\mathbf{r}$ in a unit time. In other words, the flux density is the total particle track length per unit volume and per unit time. The units of flux density (vn) are, for example, $(cms^{-1})cm^{-3} = cm^{-2}s^{-1}$.

Radiation Interaction Rates: Reaction-Rate Density

- From the concept of flux density and cross sections, we can now calculate the density of interactions. Specifically, let $\hat{R}_i(\mathbf{r})$ be the number of i-th type interactions per unit time that occur in a unit volume at $\mathbf{r}$.

- Thus

$$\begin{aligned}
\hat{R}_i(\mathbf{r}) &= \frac{\text{distance traveled by radiation in } 1cm^3 \text{ in } 1s}{\text{distance radiation must travel for } i\text{-th interaction}} \\
\hat{R}_i(\mathbf{r}) &= \frac{\phi(\mathbf{r})}{1/\mu_i} \\
\hat{R}_i(\mathbf{r}) &= \mu_i \phi(\mathbf{r}) \qquad (16)
\end{aligned}$$

  Note that for neutron interactions, Σ_i is usually used instead of μ_i. Also note that the units of the interaction density rate $\hat{R}_i(\mathbf{r})$ are, for example, $\frac{\#}{cm^3 s}$.

  Remark: See definition of cross section in Lecture 3.

- This simple expression for reaction-rate densities is a key equation for many nuclear engineering calculations.

Radiation Interaction Rates: Radiation Fluence

- In most interaction rate calculations, the interaction properties of the medium do not change appreciably in time.
- From (16), the total number of interactions that occur in some volume V between times t_1 and t_2 by neutrals of all energies is

$$\text{Number of Interactions} = \iiint_V dV \int_0^{E_{max}} dE \int_{t_1}^{t_2} dt \mu(\mathbf{r}, E)\phi(\mathbf{r}, E, t) \quad (17)$$

  We define the fluence of radiation between t_1 and t_2 as the time-integrated flux density, namely

$$\Phi(\mathbf{r}, E) = \int_{t_1}^{t_2} dt \phi(\mathbf{r}, E, t) \quad (18)$$

  which in the steady-state case reduces to

$$\Phi(\mathbf{r}, E) = (t_2 - t_1)\phi(\mathbf{r}, E) \quad (19)$$

- Calculation of the flux density $\phi(\mathbf{r}, E, t)$ is difficult and, generally, requires particle transport calculations. Fortunately, in many practical situations, the flux density can be approximated by the flux density of uncollided source particles, a quantity which is relatively easy to obtain.

Radiation Interaction Rates: Flux Density - Isotropic Source

- **Point Source in Vacuum:** Consider a point isotropic source of that emits S_p particles per unit time, all with energy E, into an infinite vacuum.
- All particles move radially outward without interaction, and because of the source isotropy, each unit area on an imaginary spherical shell of radius r has the same number of particles crossing it, namely, $S_p/(4\pi r^2)$.
- The flux density ϕ^o of uncollided particles a distance r from the source is

$$\phi^o(r) = \frac{S_p}{4\pi r^2} \tag{20}$$

- If at a distance r we place a small homogeneous mass, such as a small radiation detector, with a volume ΔV_d, the interaction rate R^o is

$$R^o(r) = \mu_d(E) \Delta V_d \frac{S_p}{4\pi r^2} \tag{21}$$

  where $\mu_d(E)$ is the linear interaction coefficient for the reaction of interest in the detector material.
- Notice that the flux density and interaction rate R decrease as $1/r^2$ as the distance from the source is increased. This decreasing dose with increasing distance is sometimes referred to as geometric attenuation.

- **Point Source in a Homogeneous Attenuating Medium:** Now consider the same point monoenergetic isotropic source embedded in an infinite homogeneous medium characterized by a total interaction coefficient μ.

- As the source particles stream radially outward, some interact before they reach the imaginary sphere of radius r and do not contribute to the uncollided flux density.

- Using (6), the number of source particles that travel at least a distance r without interaction is $S_p e^{-\mu r}$, so that the uncollided flux density is

$$\phi^o(r) = \frac{S_p}{4\pi r^2} e^{-\mu r} \tag{22}$$

- The term $e^{-\mu r}$ is referred to as the material attenuation term to distinguish it from the $1/r^2$ geometric attenuation term.

Radiation Interaction Rates: Flux Density - Isotropic Source

- **Point Source with a Shield:** Now suppose that the only attenuating material separating the source and the detector is a slab of material with attenuation coefficient μ, and thickness t.

- In this case, the probability that a source particle reaches the detector without interaction is $e^{-\mu t}$, so that the uncollided flux density is

$$\phi^o(r) = \frac{S_p}{4\pi r^2} e^{-\mu t} \tag{23}$$

Radiation Interaction Rates: Flux Density - Isotropic Source

Example: A point source with an activity of $5Ci$ emits $2-MeV$ photons with a frequency of 70% per decay. What is the flux density of $2-MeV$ photons $1m$ from the source? What thickness of iron needs to be placed between the source and detector to reduce the uncollided flux density to $\phi^o_{max} = 2 \times 10^3 cm^{-2}s^{-1}$?

Iron: $\mu(2-MeV) = 0.3222cm^{-1}$.

Example: A point source with an activity of $5Ci$ emits $2-MeV$ photons with a frequency of 70% per decay. What is the flux density of $2-MeV$ photons $1m$ from the source? What thickness of iron needs to be placed between the source and detector to reduce the uncollided flux density to $\phi^o_{max} = 2\times 10^3 cm^{-2}s^{-1}$?

Iron: $\mu(2-MeV) = 0.3222cm^{-1}$.

Solution:

The source strength can be computed as

$$S_p(\text{photons}/s) = (5Ci)(3.7\times 10^{10}Bq/Ci)(0.7\text{photons/ decay}) = 1.295\times 10^{13}s^{-1}. \quad (24)$$

Because the mean-free-path length ($\triangleq 1/\mu$) for a 2- MeV photon in air is 187 m, we can ignore air attenuation over a 1-m distance. The flux density of 2-MeV photons 1 meter from the source is then, from (20),

$$\phi^o_1 = \frac{S_p}{4\pi r^2} = \frac{1.295\times 10^{13}s^{-1}}{4\pi(100cm)^2} = 1.031\times 10^6 cm^{-2}s^{-1}. \quad (25)$$

Radiation Interaction Rates: Flux Density - Isotropic Source

Example: A point source with an activity of $5Ci$ emits $2-MeV$ photons with a frequency of 70% per decay. What is the flux density of $2-MeV$ photons $1m$ from the source? What thickness of iron needs to be placed between the source and detector to reduce the uncollided flux density to $\phi^o_{max} = 2\times 10^3 cm^{-2}s^{-1}$?

Iron: $\mu(2-MeV) = 0.3222cm^{-1}$.

Solution:

The uncollided flux density with an iron shield of thickness t placed between the source and the 1-meter detection point is, from (23),

$$\phi^o_2 = \frac{S_p}{4\pi r^2} e^{-\mu t} = \phi^o_1 e^{-\mu t}. \quad (26)$$

Solving for the shield thickness t gives

$$t = -\frac{1}{\mu}\ln\left(\frac{\phi^o_2}{\phi^o_1}\right) = -\frac{1}{\mu}\ln\left(\frac{\phi^o_{max}}{\phi^o_1}\right) = -\frac{1}{0.3222cm^{-1}}\ln\left(\frac{2\times 10^3}{1.031\times 10^6}\right) = 19.4cm. \quad (27)$$

Radiation Doses & Hazard Assessment

- Ionizing radiation creates chemical free radicals and promotes oxidation-reduction reactions as it passes through biological tissue.

- However, how these chemical processes affect the cell and produce subsequent detrimental effects to an organism is not easily determined.

- Much research has been directed towards understanding the hazards associated with ionizing radiation.

- Consequences of exposure to ionizing radiation may be classified broadly as:
  - hereditary effects
  - somatic effects.

Radiation Doses & Hazard Assessment

- Hazards of human exposure to ionizing radiation depend on both the exposure and its duration and may be classified broadly as:
  - deterministic consequences: Acute, life-threatening exposure. Illness is certain. The scope and degree of illness depend on the radiation dose and the physical condition of the individual exposed.
  - stochastic consequences: Minor acute or chronic low-level exposure. Hereditary illness may or may not result; cancer may or may not result. Only the probability of illness, not its severity, is dependent on the radiation dose.

Dosimetric Quantities

- Dosimetric quantities are intended to provide, at a point or in a region of interest, a physical measure correlated with a radiation effect.

  - Fluence is not closely enough related to most radiation effects to be a useful determinant. If faced with exposure to a radiation field of fixed fluence, one would certainly care about the nature of the radiation or its energy.

  - Energy fluence appears to be more closely correlated with radiation effect than is fluence alone, since the energy carried by a particle must have some correlation with the damage it can do to material such as biological matter. Even so, this choice is not entirely adequate—not even for particles of one fixed type.

Dosimetric Quantities

- It is apparent that one should be concerned not so much with the passage of particles or energy through a region of material, but with the creation of certain physical effects in that material.
- Major efforts have been made to quantify these phenomena by measurement or calculation, the results to be used as indices of radiation damage.
- Any such quantification is called a *dose* if accumulated over a period of time, or a *dose rate* if the effect per unit time is of interest.
- There are a few dosimetric quantities that have been precisely defined and that are particularly useful in radiological assessment. To understand these definitions, one must appreciate the several-stage process in the passage of energy.

Dosimetric Quantities

1. Uncharged primary radiation such as neutrons or photons interact with the nuclei or the electrons of the material through which they are passing.
2. As a result of the interactions, secondary charged particles are emitted from the atoms involved, and each of these particles starts out with kinetic energy related to the energy of the primary particle and the type of interaction that led to creation of the secondary particle.
3. The secondary charged particles lose energy while traversing the material either (a) through ionization and associated processes such as atomic and molecular excitation and molecular rearrangement, or (b) through emission of photons called bremsstrahlung.
4. The uncharged primary particles may produce additional uncharged particles through scattering or other processes.

Energy Imparted to the Medium

- For a given volume of matter of mass m, the energy ϵ "imparted" in some time interval is the sum of the energies (excluding rest-mass energies) of all charged and uncharged ionizing particles entering the volume minus the sum of the energies (excluding rest-mass energies) of all charged and uncharged ionizing particles leaving the volume, further corrected by subtracting the energy equivalent of any increase in rest-mass energy of the material in the volume.

- Thus, the energy imparted is that which is involved in the ionization and excitation of atoms and molecules within the volume and the associated chemical changes.

- This energy is eventually degraded almost entirely into thermal energy.

Absorbed Dose

- The *absorbed dose* is the quotient of the mean energy imparted $\Delta\bar{\epsilon}$ to matter of mass Δm, in the limit as the mass approaches zero

$$D \equiv \lim_{\Delta m \to 0} \frac{\Delta\bar{\epsilon}}{\Delta m} \tag{28}$$

- Here $\bar{\epsilon}$ is the expected energy imparted to the medium averaged over all stochastic fluctuations.
- The absorbed dose is thus the average energy absorbed from the radiation field, per unit differential mass of the medium.
- The concept of absorbed dose is very useful in radiation protection. Energy imparted per unit mass in tissue is closely, but not perfectly, correlated with radiation hazard.
- The standard unit of absorbed dose is the *gray* (Gy), 1 Gy being equal to an imparted energy of 1 J per kg. A traditional unit for absorbed dose is the *rad*, 1 rad is defined as 100 erg per g. Thus, 1 rad = 0.01 Gy $(1erg = 10^{-7}J)$.

Kerma

- The *absorbed dose* is, in principle, a measurable quantity; but in many circumstances it is difficult to calculate the absorbed dose from radiation fluence and material properties.

- A closely related deterministic quantity, used only in connection with indirectly ionizing (uncharged) radiation, is the *kerma*, an acronym for "kinetic energy of radiation absorbed per unit mass".

- If E_{tr} is the sum of the initial kinetic energies of all the charged ionizing particles released by interaction of indirectly ionizing particles in matter of mass m, then

$$K \equiv \lim_{\Delta m \to 0} \frac{\Delta \bar{E}_{tr}}{\Delta m} \tag{29}$$

  Again the bar over the E_{tr} indicates the expected or stochastic average.

Calculating Kerma and Absorbed Doses

- The calculation of the kerma (rate) is closely related to the reaction (rate) density.
- If, at some point of interest in a medium, the fluence of radiation with energy E is Φ, the kerma at that point is:

$$\begin{aligned} K &= \#\text{ interactions per unit mass} \times \text{energy per interaction} \\ K &= \left\{\frac{\mu(E)\Phi}{\rho}\right\} \times \{Ef(E)\} = \left(\frac{f(E)\mu(E)}{\rho}\right) E\Phi \end{aligned}$$

- Here $f(E)$ is the fraction of the incident radiation particle's energy E that is transferred to secondary charged particles, and $\mu(E)/\rho$ is the mass interaction coefficient for the detector material.
- The kerma rate $\dot{K}$ is obtained by replacing the fluence by the fluence rate or flux density ϕ.
- This result for the kerma applies equally well to neutrons and photons.

Photon Kerma and Absorbed Dose

- The product $f\mu$, called μ_{tr} (*linear energy transference coefficient*), accounts only for charged secondary particles and excludes the energy carried away from the interaction site by secondary photons (Compton scattered photons, annihilation photons, and fluorescence).

- Thus for photon kerma calculations we can write

$$K = 1.602 \times 10^{-10} E \left(\frac{\mu_{tr}(E)}{\rho} \right) \Phi \tag{30}$$

for the kerma in units of Gy, E in MeV, μ_{tr}/ρ in cm^2/g, and Φ in cm^{-2}.

Photon Kerma and Absorbed Dose

- If the secondary charged particles produce substantial bremsstrahlung, a significant portion of the charged-particles' kinetic energy is reradiated away as bremsstrahlung from the region of interest.
- Even under charged-particle equilibrium, the kerma may overpredict the absorbed dose.
- The production of bremsstrahlung can be taken into account by the substitution of μ_{en} (*linear energy absorption coefficient*) for μ_{tr}.
- Then, under the assumptions of charged particle equilibrium and no local energy transfer from bremsstrahlung

$$D = 1.602 \times 10^{-10} E \left(\frac{\mu_{en}(E)}{\rho} \right) \Phi \qquad (31)$$

  for the absorbed dose in units of Gy, E in MeV, μ_{en}/ρ in cm^2/g, and Φ in cm^{-2}.
- The Dose rate $\dot{D}$ is obtained by replacing the fluence by the fluence rate or flux density ϕ.

Photon Kerma and Absorbed Dose

Example: What are the iron kerma and absorbed dose rates from uncollided photons 1 meter from a point isotropic source emitting 10^{14} 5-MeV gamma rays per second into an infinite water medium?

Photon Kerma and Absorbed Dose

Example: What are the iron kerma and absorbed dose rates from uncollided photons 1 meter from a point isotropic source emitting 10^{14} 5-MeV gamma rays per second into an infinite water medium?

Solution:

From Appendix C in the book, we can find that the total interaction coefficient for 5-MeV photons is $\mu_{H_2O} = 0.03031 cm^{-1}$. The uncollided flux density 1-meter from the source is, from (22),

$$\phi^o = \frac{S_p}{4\pi r^2} e^{-\mu_{H_2O} r} = \frac{10^{14} s^{-1}}{4\pi (100cm)^2} e^{-(0.03031 cm^{-1})(100cm)} = 38.41 cm^{-2} s^{-1} \tag{32}$$

Also from Appendix C in the book, $(\mu_{tr}/\rho)^{Fe} = 0.02112 cm^2/g$ and $(\mu_{en}/\rho)^{Fe} = 0.01983 cm^2/g$ for 5-MeV photons in iron. Then, from (30), the iron kerma rate is

$$\dot{K}^o = 1.602 \times 10^{-10} E \left(\frac{\mu_{tr}(E)}{\rho} \right)^{Fe} \phi^o = 6.50 \times 10^{-10} Gy/s = 2.34 \mu Gy/h. \tag{33}$$

Photon Kerma and Absorbed Dose

Example: What are the iron kerma and absorbed dose rates from uncollided photons 1 meter from a point isotropic source emitting 10^{14} 5-MeV gamma rays per second into an infinite water medium?

Solution:

With the assumption of charged particle equilibrium, the absorbed dose rate in iron is given by (31) as

$$\dot{D}^o = 1.602 \times 10^{-10} E \left(\frac{\mu_{en}(E)}{\rho} \right)^{Fe} \phi^o = 6.10 \times 10^{-10} Gy/s = 2.20 \mu Gy/h. \quad (34)$$

- Notice that the detection medium can be different from the medium through with the radiation is traveling.
- Also, even with the assumption of charged particle equilibrium, $\dot{K}$ is slightly larger than $\dot{D}$ since the bremsstrahlung energy emitted by secondary electrons is absorbed away from the point at which the absorbed dose is calculated; however, it is included (through the initial kinetic energy fo the secondary electrons) in the concept of the kerma.

Fast Neutron Kerma

- When fast neutrons pass through a medium, the primary mechanism for transferring the neutrons' kinetic energy to the medium is from neutron scattering interactions.

- In a neutron scatter, the scattering nucleus recoils through the medium creating ionization and excitation of the ambient atoms.

- For isotropic elastic scattering in the center-of-mass system of a neutron with initial energy E, the average neutron energy loss (and hence average energy of the recoil nucleus) is

$$f_s(E)E = \frac{2A^*}{(A^*+1)^2}E, \qquad A^* = \frac{M}{m_n} \tag{35}$$

where M and m_n denote the masses of the recoil nucleus and the neutron respectively.

Fast Neutron Kerma

- Thus, if only elastic scattering is of importance, the neutron kerma is

$$K = 1.602 \times 10^{-10} E \left(\frac{f_s(E)\mu_s(E)}{\rho} \right) \Phi \tag{36}$$

- Here K has units of Gy, when E is in MeV, Φ is in cm^{-2}, the macroscopic cross section for elastic scattering μ_s is in cm^{-1}, and the medium's mass density ρ is in g/cm^3.

- For slow or thermal neutrons, calculation of the neutron kerma is more difficult since charged particles produced by nuclear reactions must be considered.

Dose Equivalent

- The *dose equivalent* H is recognized as an appropriate measure of radiation risk when applied in the context of establishing radiation protection guidelines and dose limits for population groups.

- It is defined as the product of the *quality factor* QF and the *absorbed dose* D, i.e.,

$$H \equiv QF \times D \tag{37}$$

- QF takes into account variations in the sensitivity of the biological material to different types or energies of radiations and is meant to apply generically to those biological effect or endpoints of importance in low-level radiation exposure, namely cancer and hereditary illness.

- The standard unit of dose equivalent is the *sievert* (Sv) equal to the absorbed dose in Gy times the quality factor. A traditional unit for dose equivalent is the rem, based on the absorbed dose in rad, and $1rem$ is equivalent to $0.01Sv$.

Dose Equivalent - Quality Factor

Radiation	QF
X, γ, $\beta^{\pm}$ (al energies)	1
neutrons $< 10keV$	5
neutrons $10 - 100keV$	10
neutrons $0.1 - 2MeV$	20
neutrons $2 - 20MeV$	10
neutrons $> 20MeV$	5
protons ($> 1MeV$) [ICRP]	5
protons ($> 1MeV$) [NCRP]	2
alpha particles	20

Table 1: Quality factors

Dose Equivalent

Example: What is the dose equivalent 15 meters from a point source that emitted $1MeV$ photons isotropically into an infinite air medium for 5 minutes at a rate of 10^9 photons per second?

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Solution: We can neglect air attenuation over a distance of 15 m so that the fluence 15 m from the source is

$$\Phi = \frac{S_p \Delta t}{4\pi r^2} = \frac{10^9 s^{-1} 600 s}{4\pi (1500cm)^2} = 2.122 \times 10^4 cm^{-2}. \quad (38)$$

The dose equivalent $H = QF \times D$, where D is the absorbed dose at the point of interest. Implicit in the concept of dose equivalent is that the energy absorbing medium is tissue. If (μ_{en}/ρ) data is not available for tissue, we can approximate tissue by water (available in Appendix C in the book). Finally, since the radiation is photons, the quality factor is $QF = 1$ (see Table 1). The dose equivalent is

$$\begin{aligned} H = QF \times D &= QF \times 1.602 \times 10^{-10} E \left(\frac{\mu_{en}(E)}{\rho} \right)^{H_2O} \Phi \\ &= 1 \times (1.602 \times 10^{-10}) \times 1 \times 0.03103 \times (2.122 \times 10^4) \\ &= 10.5 \mu Sv. \end{aligned} \quad (39)$$

Effective Dose Equivalent

- In a human, different organs have different radiological sensitivities, i.e., the same dose equivalent delivered to different organs results in different consequences.
- Moreover, a beam of radiation incident on a human body generally delivers different dose equivalents to the major body organs and tissues.
- Finally, ingested or inhaled sources of radiation usually produce different doses equivalents in the various body organs and tissues.
- To account for different organ sensitivities and the different doses received by the various organs a special dose unit, the effective dose equivalent H_E, is used to describe better the hazard a human body experiences when placed in a radiation field.

Effective Dose Equivalent

- The effective dose equivalent is a weighted average of the dose equivalents received by the major body organs and tissue, namely,

$$H_E \equiv \sum_T \omega_T \overline{QF}_T D_T \equiv \sum_T \omega_T H_T \tag{40}$$

where, for organ/tissue T, ω_T is the *tissue weighting factor*, D_T is the absorbed dose, $\overline{QF}_T$ is the averaged quality factor. Here $H_T \equiv \overline{QF}_T D_T$ is the dose equivalent.

Organ	ω_T	Organ	ω_T
gonads	0.25	breast	0.15
red marrow	0.12	lung	0.12
thyroid	0.03	bone surface	0.03
remainder	0.30	TOTAL	1.00

Effective Dose Equivalent

Example: Naturally occurring radionuclides in the human body deliver an annual dose to the various tissues and organs of the body as follows: lung $36mrem$, bone surfaces $110mrem$, red marrow $50mrem$, and all other soft tissues $36mrem$. What is the annual effective dose equivalent that a human receives?

Solution:

Organ T	H_T (mrem/y)	w_T	$H_T w_T$ (mrem/y)
broch./epith.	36	0.12	4.32
bone surfaces	110	0.03	3.30
red marrow	50	0.12	6.00
Soft Tissues:			
gonads	36	0.25	9.00
breast	36	0.15	5.40
thyroid	36	0.03	1.08
remainder	36	0.30	10.80
Total: (effective dose equivalent)			39.90