SECTION 8 - EXTERNAL PROTECTION FACTORS - TIME, DISTANCE, SHIELDING

A. Factors Affecting Exposure in a Radiation Field

The philosophy inherent in any program of radiation safety is to reduce exposure, whether internal or external, to a minimum. If it is impossible or impractical to remove a source of radiation, other means must be considered for purposes of personnel protection. This brings about a certain reliance on judgment, in that one must make decisions regarding the level of exposure which is to be tolerated. Since expense and time consuming factors may be involved, the goal of radiation protection is embodied in the acronym "ALARA", which stands for As Low As Reasonably (Readily) Achievable.\(^1\) That is, one should make whatever reasonable efforts one can to reduce exposure to the lowest levels, taking into account economic and practical considerations.\(^2\)

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

1. time of exposure;
2. distance from the source;
3. amount of shielding present.

1. Time

The time factor simply means that the longer one remains in a radiation field, the greater will be the exposure received. At times, especially during emergencies, work must be performed in a high radiation field. In all cases, the work activities should be carefully planned outside the work area so that a minimum amount of time is used to complete the job. If the time required for one man to complete the job would result in an exposure beyond prescribed limits, then a team of workers should be employed. This would mean a small exposure for several people instead of a large exposure for one individual. Whenever working in a radiation field, one should attempt to complete the work quickly. One should not linger in
radiation fields discussing the work. If discussions are necessary, leave the radiation field to conduct the discussions and minimize exposure time.

For time consuming jobs in high radiation fields, the crew should rehearse, or conduct dry runs, without using radioactive materials. Rehearsing will allow one to discover where potential "bottlenecks" in the procedures might occur before someone gets exposed. By correcting the procedures, the time to complete the job will be reduced.

In some cases, one may be able to limit the amount of radioactive material being used at any one time, or one may be able to distribute the exposure over a longer time frame by limiting handling time.

Assume that work must be done in a field of $2.6 \times 10^{-4} \text{ C/kgh}$ (approximately 1 R/h). If we are allowed $5.2 \times 10^{-6} \text{ C/kg}$ (approximately 20 mR) each working day, how long can a person work in this area so as not to exceed this limit?

$$X = \dot{X}t$$

$$= t(h) \times 2.6 \times 10^{-4} \text{ (C/kgh)} = 5.2 \times 10^{-6} \text{ C/kg}$$

Then $t = \frac{5.2 \times 10^{-6}}{2.6 \times 10^{-4}} \text{ h}$

$$t = 0.02 \text{ h (1.2 min)}$$

This means that an individual would receive a daily maximum allowable exposure in 1.2 min while working in this radiation field. Suppose the job needed to be completed, and the estimated time for completion was 15 min. How many men would be needed so that no one exceeded a daily limit?

$$N = \frac{15 \text{ min}}{1.2 \text{ min/man}} = 12.5 \text{ men required}$$
2. Distance

The intensity of a radiation field decreases with distance from the source. If we consider a point source of penetrating radiation, the decrease in intensity will be inversely proportional to the square of the distance between any two radial points from the source. This "inverse square law" can be utilized only when the dimensions of source and measuring device are small compared with the distance between them and the distances are measured in air or vacuum (see Section 3.D).

For other than point sources, the intensity will decrease inversely with distance, but not necessarily as the square of the distance. Nevertheless, one should store sources far from any work area to take advantage of the reduced radiation field.

Increasing the distance from a source is often the most effective way of decreasing radiation exposure. However, one is often faced with a space problem, or a job in which the worker has to be in close proximity to the source. This essentially places a restriction on the distance factor. In these cases, one must still avoid hand contact. To reduce exposure to the body, work at arm's length. Use tongs, long-handled tools, and, if the radiation field is very high, remote control devices to handle radioactive materials. Also, one may need to employ shielding.

3. Shielding

We can reduce the radiation hazard by placing a suitable attenuating material, or combination of materials, between the source and worker. The choice and/or thickness of attenuating material (shield) will depend upon the type, and energy, of the radiation.

A shield is a medium of some thickness which will stop or effectively attenuate radiation to nonhazardous levels. The effectiveness of the shield is determined by the interactions between the incident radiation and the atoms of the absorbing medium. The interactions which take place depend mainly upon the type of radiation (α, β, x-γ
photons, neutrons, etc.), the energy of the radiation, and the atomic number of the absorbing medium. An effective shield will cause a large energy loss in a small penetration distance without emission of more hazardous radiation.

In choosing a shielding material, our first consideration must be personnel protection. However, other factors may influence our choice of material—such as; is it economical; is it too heavy; how much space are we allowed for the shield; does it have proper structural strength? Also, will it create a toxicity or contamination problem due to radiation damage; to what levels must the radiation be reduced in order to obtain accurate measurements with various instruments in the area, etc.?

When massive shields are required, cost is a very critical factor, as well may be the type of material and space limits. In this case, the transport of the radiation must be studied in more detail to arrive at optimum design features. For some applications, the shielding requirements are less severe and can be adequately estimated by simpler methods.

The time, distance, and shielding factors are used individually or in combination to minimize external radiation hazards. For more discussion about protection against external radiation, one may refer to References 3-7.

Of the three basic principles of external radiation protection (time, distance, shielding), shielding presents the most complex problem in many applications. There are so many factors influencing the design of shields that a large number of such calculations must be performed on computers. For this reason, it is hard to treat the subject from the health physics standpoint in that, the most interesting aspects of shielding are usually not the concern of the radiation protection section. However, it is of value to examine some of the quantitative relationships because they, on occasion, do become useful in practical health physics work. Certainly, the qualitative approach to shielding should be studied and understood.
B. **Alpha Radiation**

Alpha particles, which lose energy rapidly in any medium because of their relatively high ionization loss, can be stopped by very thin absorbing materials. A few sheets of paper or thin (less than 400 μm) aluminum foil will absorb alphas from α-emitting sources. The most energetic alphas will travel only a few tens of mm in air. The outer layer of skin, approximately 0.07 kg/m² (7 mg/cm²) in thickness, will absorb α particles up to approximately 7.5 MeV. Since this is dead tissue, no harmful effect is produced upon the body. Therefore, α radiation is not considered an external exposure problem.

Alphas accelerated in machines attain much higher energies but are of little consequence in regard to bremsstrahlung. These alphas bombard target matter which then emit radiation products (neutrons, gammas) which require more shielding than that needed to stop the α particles. The combination of α sources with certain light elements such as Li, Be, B, O, N, F, will result in neutron emission which adds to the shielding considerations.

C. **Beta Radiation**

The processes by which β particles lose energy in absorbers are similar to those for alphas. However, an additional process must be considered in dealing with β absorption. This is the process whereby electromagnetic radiation, called bremsstrahlung, is produced.

A β particle has a very small mass and one-half the value of the magnitude of charge on an α particle. So for a given energy, a β particle will have a much greater velocity than an α particle. As a result of these and other factors, the β particle has a lower specific energy loss, which means that its penetration in any absorber will be much greater than that of an α particle. However, the β radiation from most natural sources have ranges < 20 kg/m² (2 g/cm²).

A 1 MeV β particle will travel about 3.5 m in air. In order to penetrate the dead layer of skin, a β particle must have an energy of
about 70 keV. Beta radiation is considered a slight external hazard, but mainly a skin exposure hazard.

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

The shielding thickness which is necessary to stop $\beta$ particles of a given energy will decrease with increasing density. For example, approximately 280 $\mu$m of aluminum ($\rho=2.7\times10^3$ kg/m$^3$) will stop 1.5 MeV betas, whereas only approximately 66 $\mu$m of lead ($\rho=1.134\times10^4$ kg/m$^3$) is needed for the same purpose.

However, the production of bremsstrahlung increases with increasing atomic number of absorber. A $\beta$ source with $E_{\text{max}} = 1$ MeV will lose about 3% of its energy as bremsstrahlung when lead (Z=82) is the absorber. If aluminum (Z=13) is the absorber, the fraction is about 0.4%. Therefore, low-Z absorbers such as aluminum, plastics (lucite), or even glass are effective $\beta$ shields.

If cost and/or weight must be considered, a combination shield may be used. In this case, a low-Z absorber is used to stop the betas, followed by a high-Z absorber to attenuate the bremsstrahlung. This type of shield (as a shipping container) is further discussed in Section 8.D.3.

1. Electron Accelerator Shielding

Whereas the shielding of $\beta$ sources in general is a simple matter, the picture changes with respect to electron beams. The interactions of the electrons in the beam are the same as those of betas. However, because electrons may attain much higher energy (> 1 GeV), the secondaries which are produced become the item of greatest concern. At electron energy > 10 MeV, photons produced in bremsstrahlung begin to initiate the release of fast neutrons (see Figure 8.1). At higher electron energies, higher energy bremsstrahlung spectra occur which lead to even
higher energy neutrons, pions, photofission, as well as electron-photon showers. In the GeV range, production of nucleonic cascades (see Section 8.6.1) will occur. Much of the shielding will then be dependent upon neutron attenuation. A further problem is the residual radioactivity due to the released neutrons and those which are then captured farther on in the shield, or by nearby matter.

The problem of shielding electron beams is one of shielding for the secondary radiations produced in electron interactions. As mentioned above, the main products are bremsstrahlung (x rays), γ rays, neutrons, and residual radioactivity (β⁺, β⁻, and γ rays) for electron beams up to about 100 MeV. The approach to shielding in many respects is similar to the approach used for reactors. However, in the accelerator case, the neutron spectrum may have higher energy components than the reactor spectrum (for an electron accelerator of Eₑ = 100 MeV, the main problem is neutrons of Eₙ > 40 MeV). A second point is that electron and x ray beams are highly directional (mostly forward) so that the output end of the accelerator becomes the object of shielding concern.
Here, the interactions produce neutrons which can be treated as fairly isotropic, so much of the shielding design concerns these products. The neutrons of concern for lower energy accelerators (< 30 MeV) are referred to as "giant resonance neutrons," and have an effective energy of approximately a few MeV. As the accelerator energy increases, eventually the high energy component of neutrons ($E_n >$ approximately 100 MeV) will dominate the shielding considerations.

References 8 and 9 contain information on the shielding of electron accelerators of $E$ up to 100 MeV. Reference 9 also contains information on the shielding of electron accelerators of $E >$ 100 MeV. In addition, Reference 10 discusses approaches to neutron shielding at high energy electron accelerators.

D. Gamma Radiation

Gamma rays do not lose energy continuously, as do $\alpha$ or $\beta$ particles, in traversing a medium. As a result, $\gamma$ rays are much more penetrating than $\alpha$ or $\beta$ radiation. Gamma radiation from most radionuclides is attenuated in matter mainly by: (1) the photoelectric effect, (2) Compton effect, and (3) pair production.

Until the photon interacts, no energy is lost to matter. When the photon does interact, the result is the release of energetic electrons. The interactions discussed above are directly proportional to the atomic number ($Z$), generally to some power between 1-5. By increasing the number of available electrons, one enhances the probability of photon interactions. So, materials of higher $Z$ and higher density, such as Pb, U, Th, Au, and W, are best suited for $\gamma$ shields. The use of some of these metals is limited by their high cost and/or weight. Therefore, metals such as Fe, Pb, Cr, and Ni are used for $\gamma$ shielding. In addition, because concrete is an effective $\gamma$ and $n$ shield, concrete is often used for $\gamma$ when neutron fields are also of concern, as in the case of reactors and accelerators.
1. Calculations of Shield Thickness

Gamma-ray absorption is an exponential process. Theoretically, this means that γ rays are never completely absorbed no matter how thick the absorber. However, practically, we can choose a shield thickness which will reduce the intensity to nondetectable or nonhazardous levels.

Three relationships (see Figure 8.2) can be used for calculating shield thickness for monoenergetic photons:

I = I₀e^{−μx}; \quad μ=\text{Linear Attenuation Coefficient} \quad \text{eq. 3.39 (A)}

I = I₀e^{−(μ_{εn})x}; \quad μ_{εn}=\text{Linear Energy Absorption Coefficient} \quad \text{eq. 8.1 (B)}

I = I₀b e^{−μx}; \quad b = \text{buildup factor} \quad \text{eq. 3.51 (C)}

In radiation protection work we prefer equation (B), since this equation incorporates a safety factor in our calculations (see Figure 8.2). The attenuation coefficient μεn assumes that all scattered gammas reach our point of interest. This means that we will overestimate the intensity I when radiation I₀ moves through a shield thickness x. As a result, the calculated thickness will be greater than is necessary to reduce the intensity to prescribed levels.

Equation (A) is used for narrow (collimated) beam conditions. Here, we assume that all γ photons which deviate (scatter) from the beam never reach our detector. If this equation is used in treating a wide beam of radiation, we would underestimate the required shield thickness. This occurs because some scattered radiation would arrive at the detector. In addition, some contribution may be made by fluorescent radiation, bremsstrahlung, and e⁺ annihilation photons.

The third equation (C), is used when cost, weight, or space are factors which must be considered. If we know the buildup factor b, we calculate the true γ intensity I; hence, we get a more accurate estimate for the needed shield thickness.
Figure 8.2 Relationships for calculating shield thickness. The curve $b e^{-\mu x}$ gives the correct attenuation for distance $x$; $e^{-\mu x}$ predicts too much attenuation, $e^{-\mu enx}$ predicts too little.
For quick shielding estimates, we can use multiples of tenth value or half value layers. A half value layer \((x_{1/2})\) is defined as that thickness of a material which reduces the radiation intensity to one half its initial value. A tenth value layer, or any other thickness which reduces the intensity to some desired fraction of the initial intensity, is similarly defined (see Section 3.E.6).

A half value layer is determined as follows:

\[ I = I_0 e^{-\left(\mu_{en}\right)x} \quad \text{8.2a} \]

Then:

\[ \frac{I}{I_0} = 1/2 = e^{-\left(\mu_{en}\right)x_{1/2}} \quad \text{8.2b} \]

or

\[ \ln\frac{1}{2} = -0.693 = \left(\mu_{en}\right)x_{1/2} \quad \text{8.2c} \]

From this:

\[ x_{1/2} = \frac{0.693}{\mu_{en}}. \quad \text{8.3} \]

The above expression can be used to determine the number of half value layers needed to reduce the initial intensity to any other desired level. In Section 3.E.6, \(x_{1/2}\) was defined in terms of the linear attenuation coefficient \(\mu\), which would be the HVL for narrow beam conditions.

If we substitute \(0.693/x_{1/2}\) for \(\mu_{en}\) in the exponential expression, we obtain

\[ \frac{I}{I_0} = e^{-0.693\left(x/x_{1/2}\right)}. \quad \text{8.4a} \]

From this, taking the reciprocal of each side, and then the logarithm,

\[ \ln\left(\frac{I_0}{I}\right) = \ln\left(\frac{x}{x_{1/2}}\right), \quad \text{8.4b} \]
where \(x/x_{1/2}\) is the number of half value layers. If we let \(n = x/x_{1/2}\), then \(I_o/I = 2^n\), which is the same result as equation 3.46.

**EXAMPLE 1.** It is desired to reduce a beam of \(\gamma\) rays to 1/16 of its initial intensity. The gammas have an energy of 1 MeV and lead will be used as the shielding material. How many half value layers are required? How many \(m\) of lead are required?

(a) \(I_o/I = 16 = 2^n\)

Now

\[
\ln 16 = n \ln 2
\]

or

\[
n = \frac{\ln 16}{\ln 2} = 2.773 \approx 4
\]

Therefore, 4 half value layers are required.

(b) The value of \(\frac{\mu_{en}}{\rho}\) Pb is obtained from the energy absorption coefficient versus energy curve (Appendix E),

\[
\frac{\mu_{en}}{\rho}\text{Pb} = 0.0038\text{m}^2\text{kg}^{-1}; \quad (0.0038 \frac{\text{m}^2}{\text{kg}})(1.134 \times 10^4 \frac{\text{kg}}{\text{m}^3}) = 43.1 \text{ m}^{-1}(0.43 \text{ cm}^{-1})
\]

Then

1 half value layer = 0.693/(\(\mu_{en}\))Pb = 0.693/43.1 m\(^{-1}\)

and

4 half value layers = 4x(0.693/43.1) = 64.3 mm of lead (6.43 cm) = 0.064 m.

**EXAMPLE 2.** A radium source will be contained at the center of a wooden box. The \(\gamma\) reading through the surface of this box is \(7 \times 10^{-8}\) C/kgs (approximately 1 R/h). What thickness of lead will be
required, as a lining inside the box, to reduce the reading at the surface to $1.4 \times 10^{-10}$ C/kg (approximately 2 mR/h). Assume an effective energy of 0.8 MeV. Using equation 3.46,

$$I_0/I = 7 \times 10^{-8}/1.4 \times 10^{-10} = 500 = 2^n$$

Then, taking the logarithm of each side, and dividing by $\ln 2$ gives

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

Therefore, 9 half value layers are required. To find the Pb thickness,

$$\frac{\mu_{en}}{\rho}$$

is found from Appendix E.

$$(\mu_{en}/\rho)_{Pb} = 0.0047 \frac{m^2}{kg}; 0.0047 \frac{m^2}{kg} (1.134 \times 10^4 \frac{kg}{m^3})$$

$$= 53.3 m^{-1}(.533 \text{ cm}^{-1}) = (\mu_{en})_{Pb}$$

Hence,

$$9 \times 0.693/53.3 \text{ m}^{-1} = .117 \text{ m (4-5/8 in) of lead}$$

2. **Buildup Factor**

In calculations which include the buildup factor $b$, we take into account the scattered photons resulting from the Compton effect and the "uncollided gamma flux" which reach our detector. The value of $b$ varies with radiation energy, shield material, source geometry, and depth of shield penetration.
Values for $b$ in a number of relevant materials can be found in the literature (11-13) and values for lead are shown in Figure 8.3. Frequently, the values of $b$ are plotted on semilog paper against values of the mean free path for the initial radiation. The mean free path (mfp) is that thickness of absorber which will result in a reduction of $1/e$ in the initial beam intensity $I_0$, i.e., $I=0.368 \ I_0$. For radiation of attenuation coefficient $\mu_o$ in the medium, the number of mean free paths can be found from

$$N_{\text{mfp}} = \mu_o x,$$

in which $x$ is the thickness of the shield.

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

**EXAMPLE 3.** A 1 MeV $\gamma$ source is surrounded by a lead shield which reduces the intensity at a point outside the shield by a factor of 20 from what it would be without the shielding. What is the value of $b$ to a first approximation? Using equation 3.39

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

or

$$\ln \frac{I_0}{I} = \mu x = (\mu_o x)_1,$$

Then

$$\ln 20 = 2.99573 = 3 \text{ mfp}.$$

Now consult the 1 MeV curve for a point isotropic source in lead (Figure 8.3). From this we see that $b = 1.93$ for $\mu_o x = 3$ and $E = 1$ MeV. The value of $b_1$ is the first approximation to the actual buildup. To iterate, we use the product $b_1 I_0/I$, to arrive at a new estimate of the penetration $(\mu_o x)_2$. 
Figure 8.3  Exposure buildup factor in Pb – Point isotropic source (adapted from reference 13).
1.93 I_o/I = \text{e}^{\mu x} \\
ln 1.93 I_o/I = \ln 1.93(20) = (\mu_o x)_2 \\
\ln 38.6 = 3.65325 \sim 3.65 \text{ mfp.}

The new value \( b_2 \), for \( (\mu_o x)_2 = 3.65 \), is \( \sim 2.08 \). Repeating the iteration,

\[
b_2 \left( \frac{I_o}{I} \right) = \text{e}^{\mu x} \\
2.08(20) = \text{e}^{\mu x} \\
\ln 41.6 = 3.72810 = (\mu_o x)_3 \text{ mfp.}
\]

The value of \( b_3 \) is very nearly equal to \( b_2 \), so we stop the iteration process at this point and estimate the final \( b \) as 2.1. The value of the iteration approach is that many times one is asked to compute the thickness of a shield required to obtain a given attenuation. In cases where the scattering component of radiation may be important, one could significantly underestimate the required shield thickness by simply using equation 3.39.

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

**EXAMPLE 4.** A 3.7 TBq (100 Ci) \(^{32}\text{P}\) source will be shipped in a cylindrical container, 7.6x10^{-2} m (3 in) in diameter and .1016 m (4 inches) deep. The shipping regulations require a reading of <2.6x10^{-6} C/kg/h (10 mR/h) at 1 meter. Calculate the container wall thickness using equations (A), (B), and (C) in Section 8.D.1.

We first consider an all lead container; we will then work out the problem for a combination wall material of Lucite and lead.

The decay scheme\(^{14}\) for \(^{32}\text{P}\) is as follows

\[
\begin{array}{c}
\text{\(^{32}\text{P}(14.29\text{d})\)} \\
\beta^- (1.71 \text{ MeV}) \\
\text{\(^{32}\text{S}\)} \\
\text{0}
\end{array}
\]
The fraction of $\beta$ energy converted to bremsstrahlung is given by equation 3.16, $F = 3.33 \times 10^{-4}$ ZE. For the all lead container:

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

Now

$$3.7 \times 10^{12} \frac{\text{dis}}{s}(\frac{1}{\text{dis}})(0.047) = 1.739 \times 10^{11} \text{ ph/s of bremsstrahlung}$$

The C/kgh at 1 meter, unshielded, is $3.74 \times 10^{-15} \text{ n}_\gamma$

$$CE_{\gamma} = 3.74 \times 10^{-15} (1.739 \times 10^{11})(1.71) = 1.112 \times 10^{-3} \text{ C/kgh} \ (4.31 \text{ R/h}).$$ For $E/\lambda = 0.57$, as suggested in Section 3.B.3, $(\mu/\rho)_{\text{Pb}} = 0.0133 \ (\text{m}^2/\text{kg})$ from Appendix E. $(\rho_{\text{en}}/\rho)_{\text{Pb}} = 0.0078 \ (\text{m}^2/\text{kg})$, and $b$ is found from the curve in Figure 8.3 by interpolation. Number of mean free paths=$\ln(I/O) = \ln(1.112 \times 10^{-3}/2.6 \times 10^{-6}) - \ln 427.7 = 6.05482$; therefore $b \approx 3.1$. The density of lead is $1.134 \times 10^{4} \ (\text{kg/m}^3)$,

$$I = I_0 e^{-\mu x}; \frac{I_0}{I} = e^{\mu x}; \ln 427.7 = 0.0133(1.134 \times 10^{4})x;$$

$$\ln 427.7 = 150.8x; \ x = 6.058/150.8 = 4.02 \times 10^{-2} \text{ m (1.6 in)} \ (A)$$

$$I = I_0 e^{-(\mu_{\text{en}})x}; \frac{I_0}{I} = e^{(\mu_{\text{en}})x};$$

$$\ln 427.7 = 0.0078(1.134 \times 10^{4})x;$$

$$\ln 427.7 = 88.45x; \ x = 6.058/88.45 = 6.85 \times 10^{-2} \text{ m (2.7 in)} \ (B)$$

$$I = I_0 be^{-\mu x}; \frac{I_0 b}{I} = e^{\mu x}; \ ln 427.7(3.1) = 0.0133(1.134x) \ 10^{4})x;$$

$$\ln 1325.9 = 150.8x; \ x = 7.190/150.8 = 4.77 \times 10^{-2} \text{ m (1.9 in)} \ (C)$$

Using the above value in (C) for the wall thickness, the weight of the container would be about 46 kg (101 lbs). In using $\mu_{\text{en}}$ to
calculate the amount of lead required, we get a container with a weight of 90 kg (approximately 200 lbs). This is about twice the amount of lead that is actually needed. Note that the value obtained in (A) would represent insufficient shielding.

3. Effective Atomic Number

We will now consider the problem if lucite is used to shield the betas and lead to shield the bremsstrahlung.

According to Evans (Reference 15), the effective atomic number $Z_{\text{eff}}$ for a compound, in reference to the bremsstrahlung formula, is given by

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

where $N_i$ is the number of $i^{th}$ atoms, of atomic number $Z_i$, per cc. For Lucite (C$_6$H$_{5}$O$_2$), $Z_{\text{eff}} = 5.85$.

The maximum range of $^{32}$P betas in Lucite is approximately 0.008 m (approximately 5/16 in). The fraction of $\beta$ energy converted to bremsstrahlung is

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

Since the source activity is $3.7 \times 10^{12}$ Bq, the amount of bremsstrahlung is

$$3.7 \times 10^{12}(\text{dis})\left(\frac{\text{B}}{\text{s}}\right)(0.0033) = 1.22\times 10^{10} \frac{\text{ph}}{\text{s}} \text{ of bremsstrahlung.}$$

It is difficult to derive a function which describes the energy spectrum of the photons emitted. However, one can assume that the bremsstrahlung consists of monoenergetic x rays equal to the maximum $\beta$ energy. This will result in a safe estimate for the exposure rate, since only a small fraction of the photons will have $E = E_{\text{max}}$ (see Figure 3.10a.).
In this case, the exposure rate is:

\[ X \text{ at 1 meter} = 3.74 \times 10^{-15} \, n_\gamma C \cdot E_\gamma \]
\[ = 3.74 \times 10^{-15} (1.221 \times 10^{10})(1.71) \]
\[ = 7.81 \times 10^{-6} \, \text{C/kg h} (0.303 \, \text{R/h}). \]

Once again b is found from the curve (Figure 8.3):

\[ \ln(7.81 \times 10^{-5}/2.6 \times 10^{-6}) = \ln 30.04 = 3.403 - 3.4 \, \text{mfp} \]

From the curve b~ 2.2. Recalculating, as before,

\[ I = I_0 e^{-\mu x} ; I_0/I = e^{\mu x} ; \]
\[ 30.04 = e^{0.0133(1.134 \times 10^4 x)} ; \]
\[ \ln 30.04 = 150.8 x ; x = 3.403/150.8 = 2.26 \times 10^{-2} \, \text{m} \, (0.9 \, \text{in}). \, (A) \]

\[ I = I_0 e^{-(\mu_{en})x} ; I_0/I = e^{(\mu_{en})x} ; \]
\[ 30.04 = e^{0.0078(1.134 \times 10^4/x)} ; \]
\[ \ln 30.04 = 88.45 x ; x = 3.403/88.45 = 3.85 \times 10^{-2} \, \text{m} \, (1.5 \, \text{in}). \, (B) \]

\[ I = I_0 b e^{-\mu x} ; I_0 b/I = e^{\mu x} ; \]
\[ 30.04(2.2) = e^{0.0133(1.134 \times 10^4 x)} ; \]
\[ \ln 66.088 = 150.8 x ; x = 4.191/150.8 = 2.78 \times 10^{-2} \, \text{m} \, (1.1 \, \text{in}). \, (C) \]

Hence, a small amount of lucite will appreciably reduce the lead requirement. This container needs about 40% as much lead as is necessary for the "all lead" container. Again, if \( \mu_{en} \) is used in the calculations, the amount of lead is almost 1.7 times as much as is actually needed.
It should be remembered that the use of the buildup factor is important where economy, space, and weight must be considered, such as in the design of a reactor. If these considerations are unimportant, the simpler expression involving $\mu_{en}$ can be used. For calculations in air, $\mu_{en}$ can always be used. The buildup factor is important when absorbers are used in which the scattering can be quite extensive.

4. **Point Kernel**

In the previous examples, a combination of exponential attenuation and inverse square law drop off was used to obtain the total attenuation. This combination for a point source is referred to as the point kernel. Expressed in terms of the fluence rate, $\phi$, the point kernel may be written

$$\phi = \frac{N_\gamma b \ e^{-\mu x}}{4\pi r^2} \ \text{ph/m}^2\text{s}$$ \hspace{1cm} 8.6

In equation 8.6, $x$ is the thickness (m) of the shielding medium of attenuation coefficient, $\mu$ (m$^{-1}$), and buildup factor, $b$, $r$ is the distance (m) from the point source to the exposure point (see Figure 4.1), and $N_\gamma$ is the photon source strength (ph/s). If the source and exposure point are within the same medium, then $x=r$. When the medium is air, and $r=x$ is only a few meters, $e^{-\mu x} \approx 1$, $b=1$, and the expression reduces to equation 4.6. Suitable conversion factors may be included, on the right side of equation 8.6, to arrive at the exposure, kerma, or absorbed dose rate.

The point kernel is often used to compute the total contribution to an exposure point from an extended source. In this application, the point kernel is used to represent the contribution to the fluence rate at an exposure point, from each small element of the extended source. The total contribution is found by summing all the contributions by integrating over the source dimensions.13
One may express the broad beam attenuation in terms of a transmission (or attenuation) factor. This approach is used to estimate shielding requirements for medical equipment. If we rewrite equation 8.6, and express our quantity as exposure rate, \( \dot{X} \), then

\[
B = \frac{\dot{X}r^2}{X_0}
\]

in which \( \dot{X}_0 \) is the exposure rate at 1 m, \( r(m) \) is the distance to the point of interest, and \( B \) is the transmission factor, defined as the ratio of the radiation field with the shield present to that with the shield removed. In this case, the factor \( B \) includes both attenuation and buildup. Transmission factors can then be measured with depth of penetration, and plotted as shown in Figure 8.4. One simply reads the factor \( B \) from the curve to find the effectiveness of an iron shield of a stated thickness. Conversely, given a desired exposure rate \( \dot{X} \), and knowing the rate at 1 m, \( \dot{X}_0 \), one may compute the necessary value of the transmission \( B \) to obtain \( \dot{X} \) at a certain distance \( r \). Having computed \( B \), one may use Figure 8.4 for the relevant radionuclide to find the corresponding thickness of iron needed. For example, a 37 GBq (1 Ci) source of \(^{60}\)Co has an exposure rate, \( \dot{X} \), of approximately \( 9.27 \times 10^{-8} \) C/kgs (approximately 1.3 R/h) at 1 m. How much iron is needed to reduce the field to \( 1.8 \times 10^{-10} \) C/kgs (approximately 2.5 mR/h) at 4 m? Using equation 8.7,

\[
\dot{X} = \frac{\dot{X}_0 B}{r^2}
\]

and

\[
B = \frac{\dot{X}}{X_0} \left( \frac{r^2}{(4 \times 1.8 \times 10^{-10})} \right) = 3.11 \times 10^{-2}
\]

From Figure 8.4, when \( B = 3.11 \times 10^{-2} \), the thickness of iron needed would be approximately .119 m (4.7 in). Transmission curves for a number of radionuclides in several different absorbers can be found in Reference 16.
Figure 8.4  Transmission of gamma rays through iron. (Adapted from Reference 17).
With respect to some X ray equipment, the exposure is expressed in a different manner. The term weekly workload, \( W \), is used to designate the degree of use and is given by the product of the target current (mA) and total machine running time (min), for a certain kilovoltage machine. In addition, two other factors are used to modify the value of the workload. The use factor, \( U \), corrects for the fraction of the total time that the radiation is directed at a particular barrier. For the direct (primary) beam, the use factor is generally taken as 1 for floors and \( \frac{1}{4} \) for walls. The occupancy factor, \( T \), corrects for the fraction of time that the area is occupied while the beam is on. The value of \( T \) varies from 1 (full occupancy), to \( \frac{1}{4} \) (partial occupancy), to 1/16 (occasional occupancy). With these adjustments, we may rewrite equation 8.7, in terms of exposure \( X \), as

\[
X = \frac{WUTB}{r^2}
\]

In Reference 16, Appendix D, a number of curves are shown giving the attenuation (transmission) factor in lead and concrete for several values of machine kilovoltage. As an example, let \( W = 1.2 \times 10^6 \) mAs (2x10^4 mAmin), \( U=1 \), \( T=1 \), for a 250 kV X ray machine. Find the thickness of lead necessary to reduce the weekly exposure to 5.16x10^{-6} C/kg (20 mR) at 4 m. Using equation 8.8 to find \( B \),

\[
B = \frac{Xr^2}{WUT} = \frac{5.16 \times 10^{-6} \times 4^2}{1.2 \times 10^6 (1)(1)} = 6.88 \times 10^{-11} \frac{C}{kg \text{mAs}} \times \frac{1.6 \times 10^{-5} \frac{R}{\text{mAmin}}}{1 \frac{\text{R}}{\text{mAmin}}}
\]

Using Figure 2, Appendix D of Reference 16, we find the thickness of lead for \( B=1.6 \times 10^{-5} \) (since the units of the curve are \( \text{R/mAmin} \)). The required Pb thickness is approximately 10 mm. Notice the use of 5.16x10^{-6} C/kg in the shield design. This is 1/5 of the maximum allowable weekly exposure, and represents the application of ALARA principles. That is, designing the shield so that the exposure will be less than the maximum allowable weekly exposures. When this is done before shielding is installed, the cost is often reduced.
In addition to primary beam attenuation of x-ray machines, scattered radiation also needs to be treated. Reference 16 deals with the aspects of shielding the secondary, or stray, radiation.

E. Neutrons

Neutrons, like gammas, are a highly penetrating form of radiation. They possess no charge and, therefore, are unaffected by the electric fields of atoms in the traversed medium.

Neutron attenuation is accomplished mainly through elastic and inelastic collisions which reduce the energy of neutrons until they are absorbed in the medium. In many cases, penetrating $\gamma$ rays are produced as a result of neutron absorption. In addition, other secondary radiations, including more neutrons may be produced. The probability of scattering of neutrons is much higher than the absorption probability in many materials. The cross sections for neutron interactions tend to show more variation with energy than those exhibited for photon interactions. For these reasons, buildup factors for neutrons are a strong function of the material composition, the incident $E_n$ spectrum, and the geometry of the setup.13

The problem of neutron shielding is further complicated by the fact that substances which effectively attenuate neutrons are generally poor $\gamma$ shields. Since photons usually accompany neutrons, one may shield neutrons adequately, but not $\gamma$ rays. Neutrons are more hazardous to a biological system than gammas; therefore, the neutron fluence emerging from a shield should be smaller than the $\gamma$ fluence. However, for deep penetrations, the neutrons may be effectively removed and then a $\gamma$ problem may be encountered. Fast neutrons are more difficult to attenuate and therefore are the main concern in the shielding problem. Almost all neutrons are fast upon release from a source.

A further consideration is that neutrons are generally emitted with a spectrum of energies. Thus, the initial attenuation characteristics are determined by a number of components. Each component is then modified as it moves through the medium, since the neutrons lose energy. The result is
that as the neutrons move through the material, the lower energy neutrons are removed more easily than are the higher energy neutrons. So, as the penetration in the medium increases, the surviving neutrons are those which undergo the least interactions. This results in what is called "hardening" of the spectrum. This means that the average energy of the components shifts from a lower value to a higher value as penetration increases.

Elements of low mass number (A) are ideal neutron moderators. Therefore, hydrogen (in the form of water, plastic, or paraffin), beryllium, carbon (in graphite), and LiH are popular shield constituents. A shield composed entirely of a hydrogenous material, such as water, is satisfactory for neutron attenuation; however, intermediate or heavy elements alone are not suitable neutron shields. This is due to the fact that in a nonhydrogenous material, a larger number of collisions are necessary before absorption can take place. As a result, neutrons will penetrate to greater depths in such a shield. If iron is a shield component, additional precautions must be taken because thermal neutron capture in iron produces high-energy gammas (approximately 7.5 MeV). Also, high energy neutrons are effectively slowed down in iron until the energy is about 250 keV, in which case streaming occurs. That is, inelastic scattering is rare, and elastic scattering does not reduce the neutron energy very efficiently. So, neutrons tend to just "stream" through the medium. If the neutron energy is reduced enough so that capture is more probable, a 2.2 MeV γ is produced upon neutron capture by hydrogen. This photon must also be considered in the shielding approach.

Concrete is an effective neutron shielding material because of its water content. In addition, it is cheap, strong, and can be formed in almost any desired shape and size. Also, concrete is a better γ attenuator than water because of its higher-density additives. However, it is possible for the water content of concrete to change with time and this will affect the attenuation.

For most cases, a single shielding material will not provide adequate protection. Therefore, shields are generally composed of both neutron-attenuating material and γ absorbers.
1. **Shielding Approaches**

The attenuation of a narrow beam of monoenergetic neutrons passing through a substance is determined from the relationship in equation 3.58,

\[ \phi = \phi_o e^{-\Sigma_t x}, \]

where \( \phi_o \) is the incident fluence rate, \( \Sigma_t \) is the total macroscopic cross section, and \( x \) is the absorber thickness. In using \( \Sigma_t \), we assume all scattered neutrons are removed from the beam.

For a wide beam, some of the neutrons which are scattered away from the point of interest are replaced by neutrons scattered toward the point of interest. Therefore, the calculated value for \( \phi \) will be smaller than the measured value; hence, we overestimate the effectiveness of the attenuating substance.

In **thick** shields which contain sufficient hydrogen (60 kg/m²), the removal cross section \( \mu_R \) is used in the equation instead of the total cross section. It may also be used for a shield composed of one substance followed by a sufficiently thick hydrogenous shield. In these cases, the removal cross section itself accounts for buildup.

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

For the case of thin hydrogenous shields, one may use an empirical expression from Casper to account for the buildup of small-angle scattered neutrons. For a point \(^{235}\text{U}\) fission source in water of source strength, 1 n/s, the absorbed dose rate of fast neutrons is:

\[
D(\text{Gy/h}) = \frac{9.68 \times 10^{-11}}{4\pi x^2} \times 349 e^{-[10.503x^{0.698} - 3.08x]} \]

8.9
in which \( x \) is the distance (m) for a point fission source of neutrons. This may be extended to any hydrogenous substance, if we drop the part of the exponential term due to oxygen in water (-3.08 \( x \)), and rewrite as

\[
D(E) = 9.68 \times 10^{-11} (kx)^{0.349} e^{-10.503(kx)} \cdot 0.698
\]

in which \( k \) is the ratio of the atomic density of H in the substance to that in water. Although the above expressions are applicable to thin shields, there must still be sufficient hydrogen (approximately 7% by weight for low mass number material). The above expressions apply primarily to a fission neutron spectrum, but can be used for neutron sources which have an average energy in the range 1-2 MeV. With the proper choice of \( x \) in the above expressions, one may estimate the necessary neutron shielding for a fast neutron source with a spectrum of energies.

Reference 18 contains curves of the attenuation factors of neutrons in concrete, Nevada Test Site soil, water and polyethylene for monoenergetic neutron beam sources. In addition, this reference also has curves of the neutron tissue kerma for normally incident monoenergetic neutron beams on concrete. These curves can be useful if one has some spectrum information, since then one can compute an average transmission factor for a given neutron spectrum. That is, the average transmission factor, \( B \), is

\[
B = \frac{\sum_{i=1}^{n} F_i(E) \cdot A F_i(E)}{\sum_{i=1}^{n} F_i(E)}
\]

in which \( F_i(E) \) is the fraction of the neutron spectrum in energy group \( E \), and \( A F_i(E) \) is the average attenuation factor for the energy group. Most neutron sources one encounters consist of an energy spectrum of neutrons being emitted. To estimate the shielding and assess the dose from a neutron source are difficult tasks.
For very high energy neutrons, a multiplicity of energetic secondaries are produced in the first few mean free paths of the penetration of thick shields. Since these energetic secondaries may also produce a further number of secondaries at these energies \((E \gg 20 \text{ MeV})\), a buildup of radiation occurs during this phase. The term nucleonic cascade is used to describe this buildup in particles during the early stages of penetration (see Figure 8.5). The region in which the buildup rises to a maximum and then decreases to the value at the start of penetration is known as the transition zone. The penetration distance of this transition zone is typically \(2-4 \text{ mfp of the incident particle}\). Following this region, the attenuation of the beam is approximately exponential, with an attenuation length of \(\lambda(\text{kg/m}^2)\). This leads to a general expression for the attenuation factor, \(AF\),

\[
AF = B \ e^{-\frac{X}{\lambda}}
\]

References 10 and 13 discuss the value of the buildup factor \(B\) and the attenuation length for a number of shielding substances. For a definitive treatment of the shielding aspects of high energy proton accelerators, as well as other aspects of particle accelerator health physics, the book by Patterson and Thomas\(^{19}\) is suggested.

The previous discussion of neutron shielding dealt with fast neutrons. For thick shields, intermediate neutrons and thermal neutrons also must be considered. In particular, the intermediate neutrons may contribute appreciably to the transmitted dose. Capture of thermal neutrons may lead to a high capture \(\gamma\) dose rate. Reference 13 discusses methods of dealing with these shielding considerations.

F. Shielding Materials

Water, concrete, steel, and lead are the more common materials used for shielding purposes.
Figure 8.5  Transition zone - Radiation increases to a maximum and eventually undergoes exponential attenuation.

Some of the things which must be considered when choosing materials for a neutron shield are: (1) will it effectively decrease the neutron energy; (2) does it have a high capture cross section for thermal neutrons; and (3) will it effectively attenuate the accompanying γ radiation? In addition, radiation absorption in the shield will release energy in the form of heat. Therefore, to insure shield integrity, it may be necessary to know the temperature distribution in the shield.

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

Concrete is an adequate neutron shield material if the water content is at least 7% by weight. High-density concretes are recommended where space considerations are important. A certain amount of both neutron and γ attenuation can be obtained by use of a thinner shield of high-density concrete than is possible with ordinary concrete.

Boral, a mixture of aluminum and boron carbide (B₄C), is used for neutron shielding. Boron carbide, containing 80% boron, is a good
thermal neutron absorber, and the capture reaction in boron yields a relatively low-energy (0.5 MeV) γ. However, this photon contribution may be significant in some situations.

High-density materials are necessary for shielding the γ radiation. Lead is a valuable γ shield; however, uranium, tin, and iron (in steel) are also used. For γ rays of about 2 MeV, a given mass of lead will achieve approximately the same attenuation as an equal mass of iron. Above or below this energy, lead is superior to iron. However, lead has a low melting point and temperatures must be considered if this element is used for shielding. Lead also presents a toxicity problem if improperly handled.

REFERENCES


BIBLIOGRAPHY


QUESTIONs

8.1 What are the three factors which determine the total exposure one receives in a given radiation field?

8.2 Explain how a job can be completed within safe limits when the time required for one man to complete the job would result in an excessive exposure.

8.3 What geometrical limitations are placed on the use of the "inverse square law" for penetrating radiation?

8.4 What is the first consideration in choosing a shielding material and what other considerations cannot be overlooked?

8.5 Explain why α radiation is not considered an external exposure problem.
8.6 Why do α sources need shielding?

8.7 What is the threshold energy above which β particles must be considered by shielding?

8.8 What secondary radiation is produced when β particles are stopped by shielding?

8.9 List the radiation products that result from electrons when accelerated to high energies.

8.10 When do high energy neutrons dominate in shielding consideration for electron accelerators?

8.11 In what materials is the secondary radiation produced by β shielding the greatest?

8.12 Why are high Z materials good as γ shields?

8.13 Explain the difference between μ and μ_{en} used as exponents in shield thickness formulae.

8.14 What term is used to identify the thickness of a material which reduces the intensity of a radiation by one-half?

8.15 What term is given to the variable ratio involving radiation energy, shield material, and source geometry that is used to obtain a more accurate prediction of the probable effectiveness of radiation shielding?

8.16 What term is given to that thickness of absorber which will reduce the initial beam by a factor of 1/e?

8.17 What value must be determined when a shielding material is either a compound or a mixture before using the bremsstrahlung formula in β shielding calculations?

8.18 Explain the reasons for using the buildup factor in γ shielding.

8.19 What is a point-kernel?

8.20 What is the transmission factor?

8.21 Explain the term weekly workload in reference to x ray machines.

8.22 By what processes does a neutron lose its energy in a given medium?

8.23 Explain why a single shielding material is usually unsatisfactory for neutron shielding.
8.24 What is meant by "hardening" of the neutron spectrum?

8.25 Compare iron, water and concrete as neutron shielding materials. Which is the most effective and why?

8.26 What are the two principal processes included in the total macroscopic cross section for neutrons? For the total microscopic neutron cross section?

8.27 What factor replaces the macroscopic cross section in the neutron shielding formula in the case of a thick shield containing hydrogen?

8.28 Why should the neutron flux emerging from a shield be smaller than the gamma flux?

8.29 For what reason other than being a good neutron absorber is boron used in neutron shielding?

8.30 What are some of the drawbacks of lead as shielding material?

PROBLEMS

8.1 How long can a person work in a radiation field of $7 \times 10^{-5}$ C/kg/h if he is allowed $5.2 \times 10^{-8}$ C/kg per day? If the estimated time for the job is 15 minutes, how many persons are needed for the job?

Answers: 4 minutes 17 seconds, 4 persons

8.2 The original intensity of a 2 MeV $\gamma$ beam from a point source is $7.74 \times 10^{-2}$ C/kg/h. Find the intensity of the beam after it emerges from a 0.06 m thick piece of lead using the formulas:

a) $I = I_0 e^{-\mu x}$
b) $I = I_0 e^{-\mu_{en} x}$
c) $I = I_0 be^{-\mu x}$

Assume the following for lead as a shield for 2 MeV photons: $\rho = 1.134 \times 10^4$ kg/m$^3$, $b = 2.1$, $\mu/\rho = 4.5 \times 10^{-3}$ m$^2$/kg, and $\mu_{en}/\rho = 2.35 \times 10^{-3}$ m$^2$/kg.

Answers: a) $3.62 \times 10^{-3}$ C/kg/h
b) $1.56 \times 10^{-2}$ C/kg/h
c) $7.61 \times 10^{-3}$ C/kg/h
8.3 In problem 8.2, it is required to reduce the exposure outside the lead shield by a factor of 10,000. Use all the three formulas to estimate the thickness of lead needed.

Answers:  
   a) 0.18 m  
   b) 0.35 m  
   c) 0.21 m  

8.4 Find the approximate buildup factor using iteration and Figure 8.3, for a 2 MeV γ beam from a point source, when the original intensity is $1.3 \times 10^{-1}$ C/kgh, and the intensity upon passing through a lead shield is $1.0 \times 10^{-3}$ C/kgh.

Answer: 3.3

8.5 Find the effective atomic number for carbon dioxide, CO$_2$. Assume for the atomic numbers: Z: C=6, O=8. **Note:** Do not confuse the effective atomic number with the effective atomic mass of a substance.

Answer: 7.45

8.6 One of the organic scintillators used in neutron spectroscopy, stilbene, has a chemical formula C$_{14}$H$_{14}$N$_2$. Find the effective atomic number of stilbene. The atomic numbers of hydrogen, carbon and nitrogen are 1, 6 and 7, respectively.

Answer: $Z_{\text{eff}} = 5.5$

8.7 Cesium-134, $^{134}$Cs, emits $\beta^-$ particles of which

   a) 13% are 0.683 MeV,  
   b) 50% are 0.655 MeV,  
   c) 5% are 0.31 MeV,  
   d) 32% are 0.083 MeV.

Compute the bremsstrahlung produced by $10^8$ Bq when the $\beta^-$ particles are completely stopped in lead.

Answers:  
   a) $2.42 \times 10^6$ photons/s  
   b) $8.94 \times 10^6$ photons/s  
   c) $4.23 \times 10^4$ photons/s  
   d) $7.25 \times 10^4$ photons/s
8.8 A 250 kVp x-ray therapy unit operates with a tube current of 20 mA for an average of 20 h/week. How thick should a concrete protective barrier be such that the weekly exposure in the controlled area 3 m from the tube is less than 5.16x10^-6 C/kg? Use a use factor of 1/2 and occupancy factor of 1. What would be the thickness, if the wall were to be made of lead sheet? Note: use Figure 2 and Figure 3 of Appendix D, NCRP Report No. 49.

Answer: Concrete thickness - 0.45 m
Lead thickness - 0.01 m

8.9 Compute the remaining fluence rate of a narrow beam of 1 MeV neutrons passing through a foil of cadmium 1x10^-3 m thick. Data for cadmium: σ_t=6.2 barns, ρ=8.65x10^3 kg/m^3, A = 112.40

Answer: 97.2%

8.10 Using the data in the preceding problem except that the neutrons are thermal for which cadmium presents a total cross section of 2500 barns.

Answer: - 9.3x10^-6 or 0.

8.11 The atomic density of H in concrete is 1.375x10^28 atoms/m³, and in water, 6.69x10^28 atoms/m³. Compute the absorbed dose rate from a 5x10^8 n/s point fission source, located behind a 0.4 m thick concrete wall. Assuming Q=20, what is the dose equivalent rate?

Answer: 1.6 μGy/h
32 μSv/h