




# Chapter 7: External Radiation Protection

# Basic principles for external radiation protection

# Basic Principle for Radiation Protection

## Basic Principles

-  Maximizing distance
-  Minimizing exposure time
-  Shielding the radiation source

# Effective Dose, Weighting Factor (or Quality Factor)

Effective dose is absorbed dose modified by weighting factors for radiation type and exposed organs

## Radiation type

- X-rays and gamma radiation
- beta, electrons
- neutrons
- alpha radiation

## Weighting factor

1  
1  
3-20  
20

Quality Factor

$$\text{Effective dose} = \sum_{organ} w_{organ} (w_R \times D_{organ})$$

*whole body dose*

*Dose equivalent*

# Effective Dose

*whole body*

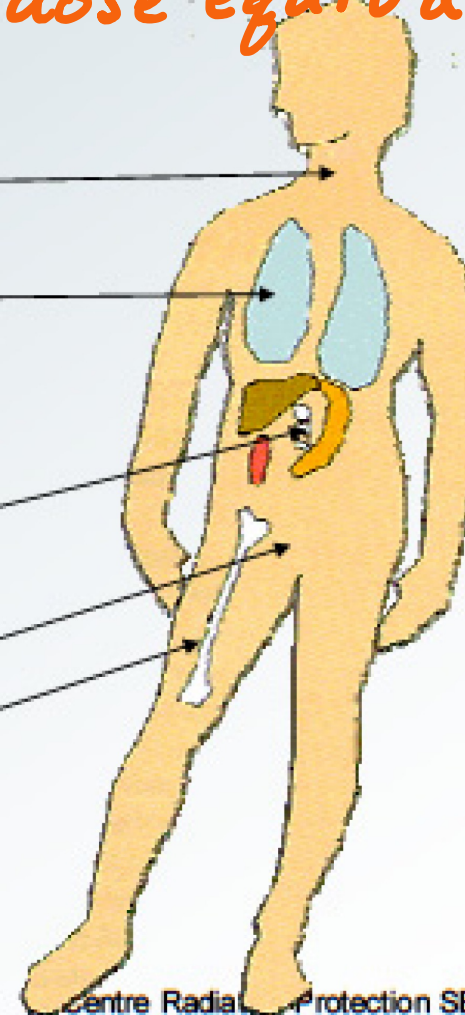
*organ dose*

$$\text{Effective dose} = \sum_{\text{organ}} w_{\text{organ}} \left( w_R \times D_{\text{organ}} \right)$$

*dose equivalent*

For calculation of effective dose:

0.05	thyroid
0.05	breast tissue
0.12	lungs
0.05	esophagus
0.12	stomach
0.05	liver
0.12	colon
0.05	urinary bladder
0.20	reproductive organs
0.01	skin
0.12	red bone marrow
0.01	bone surface
0.05	<u>rest of body</u>
1.00	total



# Deterministic and Stochastic Effects

The biological effects of radiation differs widely depending on the dose, kind of radiation. They can be divided into two general categories, stochastic and deterministic.

- ☞ **Stochastic effect** are those that occur in a statistical manner, such as cancer.
- ☞ **Deterministic effect** are those that show (a) a clear relationship between dose and effect in a given individual, and (b) a clear threshold for inducing the effect. Skin reddening and radiation-induced cataracts are examples of deterministic effects of radiation.

# Basic Limits - Occupational Exposure

- ☞ The ICRP 26 basic annual limits recommended for exposure to workers
  - ☞ To limit nonstochastic effects
    - ☞ 0.5 Sv to all tissues except the lens of the eye
    - ☞ 0.15 Sv to the lens of the eye
  - ☞ To limit stochastic effects
    - ☞ 0.05 Sv per year

# Gamma Ray Shielding Designs

- ☞ (Revisited) Approaches for evaluating gamma ray dose.
- ☞ Build-up factor
- ☞ Shielding factor
  - ☞ What is the shielding factor?
  - ☞ How to use the shielding factor to determine the shielding needed?

# **Radiation Dose Induced by Gamma Radiation**

# Radiation Dose from a Gamma-Ray Point Source

Considering an I-131 point-source of 1 MBq, how do we evaluate the exposure it delivers at a distance of 1 m?

☞ The decay of  $^{131}\text{I}$  produces gamma rays of various energies as shown below,

Quantum Energy, MeV	Photons per Transformation	Energy Absorption Coefficient for Air, $\text{m}^{-1}$
0.723	0.016	$3.8 \times 10^{-3}$
0.637	0.069	$3.9 \times 10^{-3}$
0.503	0.003	$3.8 \times 10^{-3}$
0.326	0.002	$3.8 \times 10^{-3}$
0.177	0.002	$3.4 \times 10^{-3}$
0.365	0.853	$3.8 \times 10^{-3}$
0.284	0.051	$3.7 \times 10^{-3}$
0.080	0.051	$3.2 \times 10^{-3}$
0.164	0.006	$3.3 \times 10^{-3}$

# Radiation Dose from a Gamma-Ray Point Source

For the 0.080-MeV gamma ray, we have

$$\dot{X} = \frac{5.1 \times 10^{-2} \times 8 \times 10^{-2} \times 1.6 \times 10^{-13} \times 1 \times 10^6 \times 3.6 \times 10^3 \times 3.2 \times 10^{-3}}{4\pi(1)^2 \times 1.293 \times 34}$$

$$= 1.36 \times 10^{-11} \frac{\text{C}/(\text{kg} \cdot \text{h})}{\text{MBq}}$$

☞ Repeating the calculation, we get the following results:

Quantum Energy, MeV	(C/kg)/h at 1 m
0.723	$4.583 \times 10^{-11}$
0.637	$17.787 \times 10^{-11}$
0.503	$0.598 \times 10^{-11}$
0.326	$0.258 \times 10^{-11}$
0.177	$0.126 \times 10^{-11}$
0.365	$123.400 \times 10^{-11}$
0.284	$5.569 \times 10^{-11}$
0.080	$1.361 \times 10^{-11}$
0.164	$0.339 \times 10^{-11}$
Total = $1.540 \times 10^{-9} \frac{\text{C}/(\text{kg})/\text{h}}{\text{MBq}}$ at 1 m	

# Specific Gamma-Ray Constant

- ☞ In a more generic case, considering a gamma-ray point-source of activity  $A$  (MBq), the exposure rate it will deliver to a distance  $d$  (m) away in air is given by

$$\dot{X} \left( \frac{C/kg}{h} \right) = \sum_{i=1}^I \left[ \frac{A \times 10^6 (t/s) \times 3600 (s/h) \cdot f_i (p/t) \cdot E_i (J/p)}{4\pi [d (m)]^2} \cdot \mu_i (m^{-1}) / \rho_a (kg/m^3) \right] \cdot \frac{1}{34 \frac{J/kg}{C/kg}}$$

where

$\dot{X}$ : exposure rate,  $\left( \frac{C/kg}{h} \right)$

$f_i$ : fraction of transformation that result in a photon of the  $i$ 'th energy

$E_i$ : energy carried by each photon of the  $i$ 'th group, (J)

$\mu_i$ : linear energy absorption coefficient for photons of the  $i$ 'th group, ( $m^{-1}$ )

$\rho_a$ : the density of air, ( $kg/m^3$ )

- ☞ The specific gamma-ray constant,  $\Gamma$ , for this gamma-ray source, is given by

$$\Gamma \left( \frac{C/kg \cdot m^2}{h \cdot MBq} \right) = \frac{\dot{X} \left( \frac{C/kg}{h} \right) \cdot [d(m)]^2}{A(MBq)}$$

- ☞ The specific gamma-ray constant numerically equal to the exposure rate that a source of a unit activity delivers to a unit distance away in air.

# Specific Gamma-Ray Constant

- ☞ Substitute all known constants into the generic equation, the specific Gamma-Ray Constant can be given by

$$\Gamma = 1.043 \times 10^{-6} \sum_i f_i \times E_i \times \mu_i \frac{(\text{C/kg}) \text{ m}^2}{\text{MBq} \cdot \text{h}},$$

where  $f_i$  is the fraction of the transformations that yield a photon whose energy is  $E_i$  and  $\mu_i$  is the linear energy absorption coefficient in air of the  $i$ th photon.

- ☞ For many practical situations, when photon energy is ranging from 60keV to 2MeV, the linear absorption coefficient varies little with energy, over this energy range,  $\mu$  is about  $3.5 \times 10^{-3}$  per meter. Therefore, we can simplify the above equation as

$$\Gamma = 3.65 \times 10^{-9} \sum_i f_i \times E_i \frac{(\text{C/kg}) \text{ m}^2}{\text{MBq} \cdot \text{h}}$$

# Specific Gamma-Ray Constant

- ☞ Specific Gamma Ray Constant ( $\Gamma$ ): The **exposure rate** from a gamma ray point source of unit activity and positioned at a unit distance. It is given in the unit of coulombs per kilogram per hour at a distance of 1 m from a 1 MBq point source, or (coulombs/kg/h/MBq at 1m).

**TABLE 6.3. Specific Gamma-ray Constant of Some Radioisotopes**

Isotope	$\Gamma$	
	$\frac{R \cdot m^2}{Ci \cdot h}^a$	$\frac{X \cdot m^2}{MBq \cdot h}^b$
Antimony 122	0.24	1.67E-09
Cesium 137	0.33	2.30E-09
Chromium 51	0.016	1.11E-10
Cobalt 60	1.32	9.19E-09
Gold 198	0.23	1.60E-09
Iodine 125	0.07	4.87E-10
Iodine 131	0.22	1.53E-09
Iridium 192	0.48	3.34E-09
Mercury 203	0.13	9.05E-10
Potassium 42	0.14	1.39E-09
Radium 226	0.825	5.75E-09
Sodium 22	1.20	8.36E-09
Sodium 24	1.84	12.80E-09
Zinc 65	0.27	1.88E-09

<sup>a</sup>From *Radiological Health Handbook*, rev. ed., U.S. Public Health Service, Bureau of Radiological Health, Rockville, MD, 1970.

<sup>b</sup>1 X unit = 1 C/kg.

# Specific Gamma-Ray Constant

Given the specific gamma ray constant,  $\Gamma$ , for an isotope, the exposure rate at a location at a distance,  $r$ , is simply

$$\dot{X} = \Gamma \frac{A}{r^2}$$

← activity  
← distance

## Example

(a) Estimate the specific gamma-ray constant for  $^{137}\text{Cs}$ . (b) Estimate the exposure rate at a distance of 1.7 m from a 100-mCi point source of  $^{137}\text{Cs}$ .

## Solution

(a) The isotope emits only a 0.662-MeV gamma ray in 85% of its transformations (Appendix D). The average energy per disintegration released as gamma radiation is therefore  $0.85 \times 0.662 = 0.563$  MeV. The estimated specific gamma-ray constant for  $^{137}\text{Cs}$  is therefore  $\Gamma = 0.28 \text{ R m}^2 \text{ Ci}^{-1} \text{ h}^{-1}$ .

(b) From Eq. (12.28), the exposure rate at a distance  $r = 1.7$  m from a point source of activity  $C = 100 \text{ mCi} = 0.1 \text{ Ci}$  is

$$\dot{X} = 0.28 \frac{\text{R m}^2}{\text{Ci h}} \times \frac{0.1 \text{ Ci}}{(1.7 \text{ m})^2} = 9.7 \times 10^{-3} \text{ R h}^{-1} = 9.7 \text{ mR h}^{-1}. \quad (12.29)$$

## Internally Deposited Radioisotope (IV) Gamma Ray Emitters

- ☞ For a uniformly distributed gamma-ray-emitting isotope, the dose rate from the isotope in an infinitesimal volume  $dV$  to a point  $p$  at a distance  $r$  away is

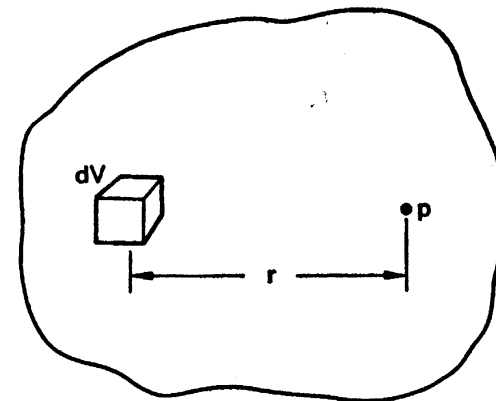
$$d\dot{D} = C(\text{MBq}/\text{m}^3) \cdot \Gamma\left(\frac{\text{C}/\text{kg}\cdot\text{m}^2}{\text{MBq}\cdot\text{hr}}\right) \cdot \frac{e^{-\mu(\text{m}^{-1})r(\text{m})}}{r^2(\text{m}^2)} \cdot dV(\text{m}^3) \cdot \left(34 \frac{\text{J}/\text{kg}}{\text{C}/\text{kg}}\right) \cdot \left(\frac{\mu_m/\rho_m}{\mu_a/\rho_a}\right)$$

Mass energy absorption coefficient  
of the dose-receiving media  
↓

↑  
Mass energy absorption  
coefficient of the air

where  $C$  is the concentration of the isotope,  $\Gamma$  is the specific gamma-ray emission, and  $\mu$  is the linear energy absorption coefficient.

**FIGURE 6.8.** Diagram for calculating dose at point  $p$  from the gamma rays emitted from the volume element  $dV$  in a tissue mass containing a uniformly distributed isotope.



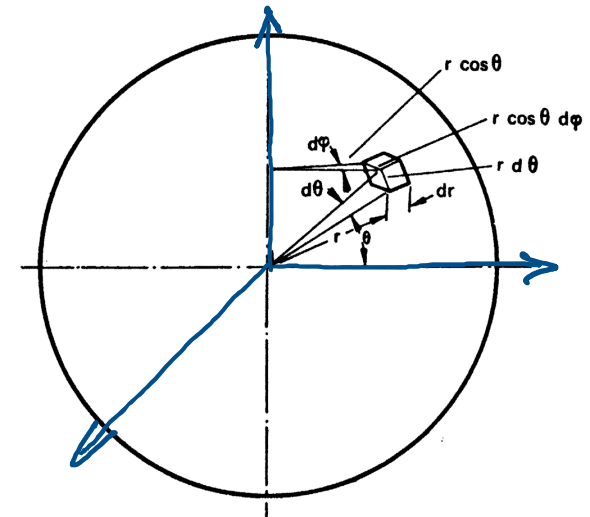
## Internally Deposited Radioisotope (IV) Gamma Ray Emitters

☞ For a uniform spherical source, the **dose rate at the center** is given by

$$\begin{aligned} \dot{D} &= 4C\Gamma \int_{r=0}^{r=R} \int_{\theta=0}^{\theta=\pi/2} \int_{\varphi=0}^{\varphi=\pi} \frac{e^{-\mu r}}{r^2} \cdot r \, d\theta \cdot r \cos\theta \, d\varphi \cdot dr \cdot \left(34 \frac{J/kg}{\text{Coulomb/kg}}\right) \cdot \left(\frac{\mu_m/\rho_m}{\mu_a/\rho_a}\right) \\ &= C\Gamma \cdot \frac{4\pi}{\mu} (1 - e^{-\mu R}) \cdot \left(34 \frac{J/kg}{\text{Coulomb/kg}}\right) \cdot \left(\frac{\mu_m/\rho_m}{\mu_a/\rho_a}\right) \end{aligned}$$

☞ And the **dose rate at the surface** of the spherical source volume is given by

$$\dot{D}_{\text{surface}} = 0.5 \dot{D}_{\text{center}}$$



$$dV = r d\theta \cdot r \cos\theta d\varphi \cdot dr = r^2 \cos\theta d\varphi d\theta dr$$

# The Buildup Factor

# The Buildup Factor

$$B \equiv 1 + \frac{R^s}{R^o} = 1 + \frac{1}{\mathcal{R}(E_o)\phi^o(r)} \int_0^\infty dE \mathcal{R}(E)\phi^s(r, E).$$

Extra response due to the scattered photons and characteristic X-rays

Distance traveled in shielding

The response due to the primary photons penetrating the shielding

Energy-dependent response function

Energy dependent response function

The photon flux reaching the object without interaction in the shielding

Flux of photons of energy E reaching the object (scattered photons, x-rays etc.)

- ☞ The buildup factor depends on
  - ☞ the source energy,
  - ☞ the distance traveled, and
  - ☞ the nature of the medium

# The Buildup Factor

**TABLE E.1** Gamma-Ray Buildup Factors for Air-Kerma Response to an Isotropic Point Source in an Infinite Air Medium. Also Given is the Total Mass Interaction Coefficient Used to Calculate the Mean-Free-Path Length.

MEAN FREE PATHS	ENERGY (MeV)										
	0.04	0.06	0.08	0.1	0.2	0.5	1	2	5	10	15
0	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
0.5	2.20	2.58	2.52	2.35	1.90	1.60	1.47	1.38	1.29	1.20	1.15
1	<u>3.38</u>	<u>4.76</u>	<u>4.83</u>	<u>4.46</u>	<u>3.28</u>	<u>2.44</u>	<u>2.08</u>	<u>1.83</u>	<u>1.57</u>	<u>1.37</u>	<u>1.28</u>
2	5.85	10.8	12.0	11.4	7.74	4.84	3.60	2.81	2.09	1.68	1.49
3	8.47	18.9	22.9	22.5	15.00	8.21	5.46	3.86	2.60	1.97	1.70
4	11.2	29.1	37.9	38.4	25.6	12.6	7.60	4.96	3.11	2.26	1.90
5	14.1	41.5	57.4	59.9	40.0	17.9	10.0	6.13	3.61	2.54	2.11
6	17.0	56.1	82	87.8	58.9	24.2	12.7	7.35	4.12	2.82	2.30
7	20.1	73.2	112	123	82.8	31.6	15.6	8.61	4.62	3.10	2.50
8	23.3	92.7	148	166	112	40.1	18.8	9.92	5.12	3.37	2.70
10	<u>30.0</u>	<u>140</u>	<u>242</u>	<u>282</u>	<u>192</u>	<u>60.6</u>	<u>25.8</u>	<u>12.6</u>	<u>6.13</u>	<u>3.92</u>	<u>3.08</u>
15	49.0	316	636	800	545	134	47.0	20.0	8.63	5.25	4.03
20	71.4	596	1350	1810	1220	241	72.8	27.9	11.1	6.55	4.96
25	97.2	1010	2540	3570	2360	385	103	36.2	13.6	7.84	5.87
30	126	1600	4390	6430	4150	567	136	45.0	16.1	9.11	6.75
35	159	2410	7140	10600	6770	788	173	54.0	18.5	10.4	7.58
40	195	3480	11100	15700	10500	1050	212	63.2	21.0	11.6	8.31
$\mu/\rho$ (cm <sup>2</sup> /g)	.2486	.1875	.1662	.1541	.1234	.08712	.06358	.04447	.02751	.02045	.01810

Source: Extracted from American National Standard ANSI/ANS-6.4.3-1991 with permission of the publisher, the American Nuclear Society.

## **Shielding Factor (or Attenuation Factor)**

# Broad Beam Consideration for Shielding Design

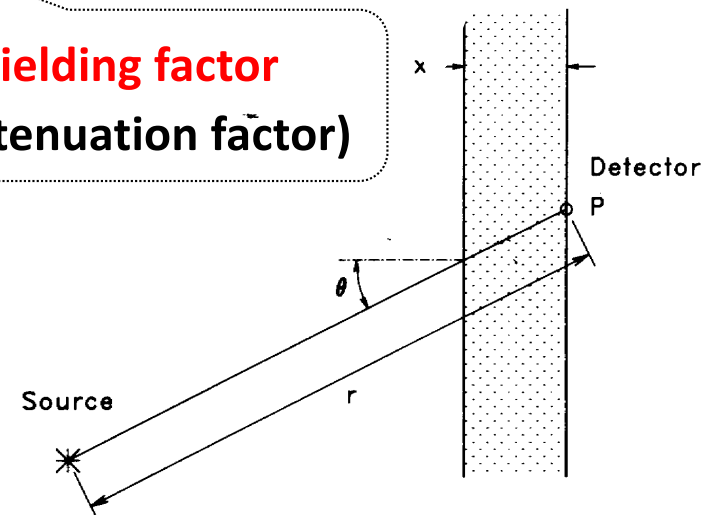
- ☞ For situations in which the radiation source is far away from the shielding slab, the photons of incident are traveling in almost parallel directions. Addressed by NCRP 49 (1976).
- ☞ For gamma rays from a point source

$$R(P) = R_o(P)A_f$$

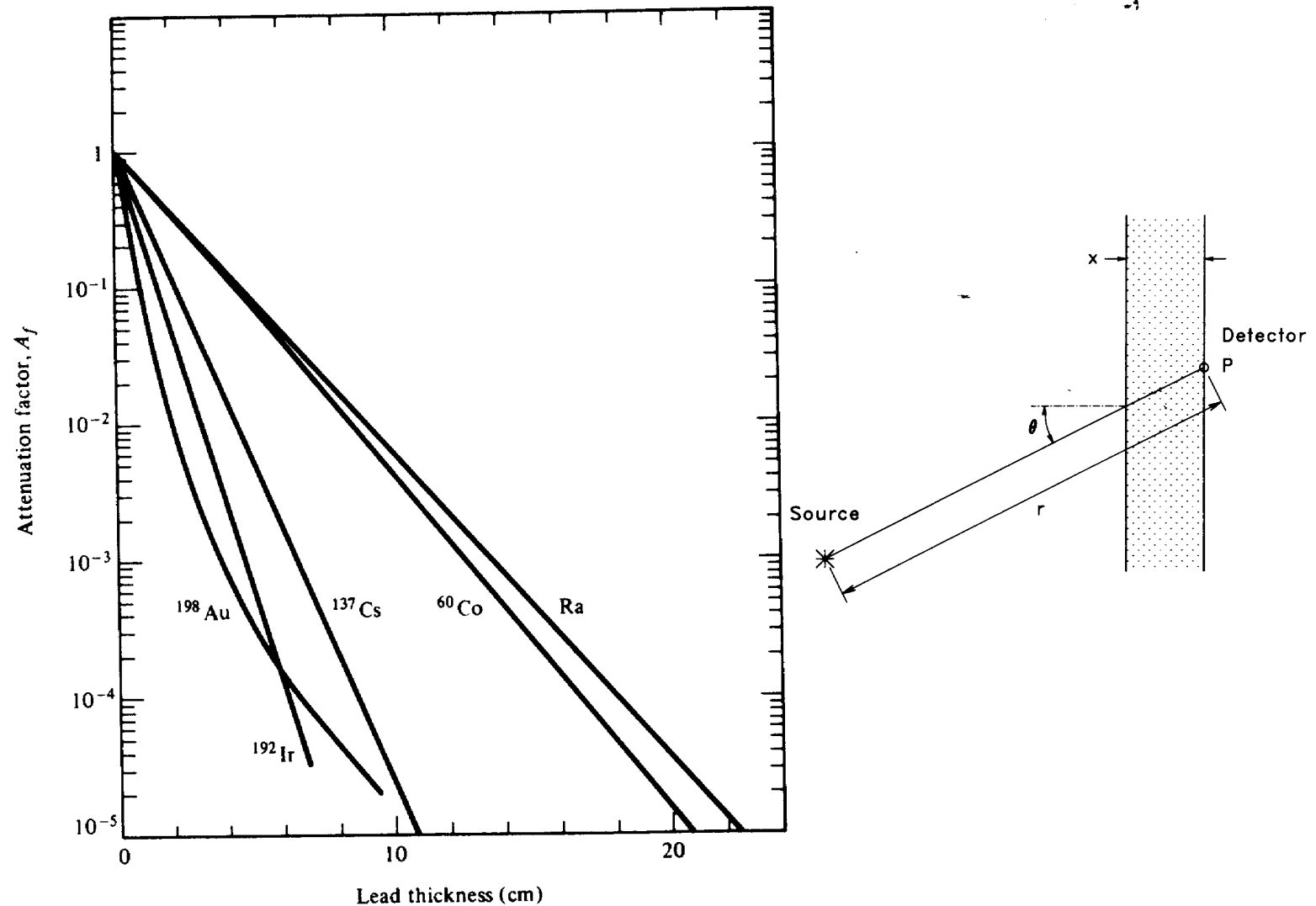
Dose or exposure rate at point P.

Dose or exposure rate without the shielding

**Shielding factor**  
(or attenuation factor)

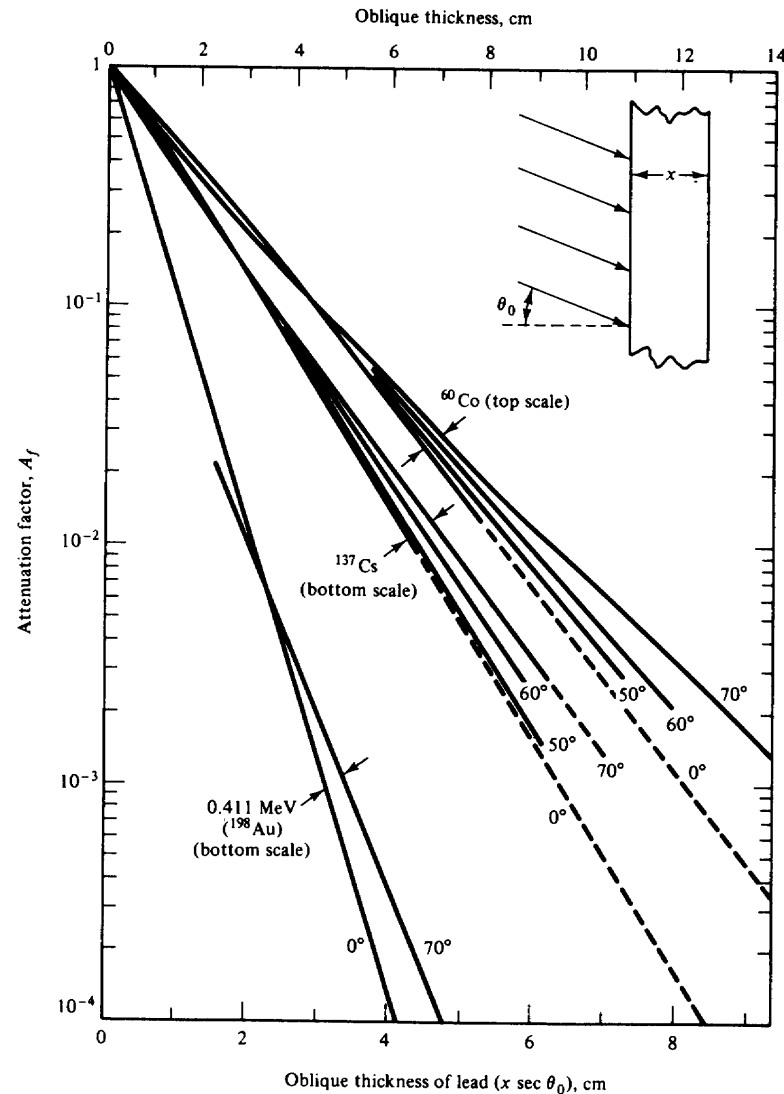


# Broad Beam Consideration for Shielding Design

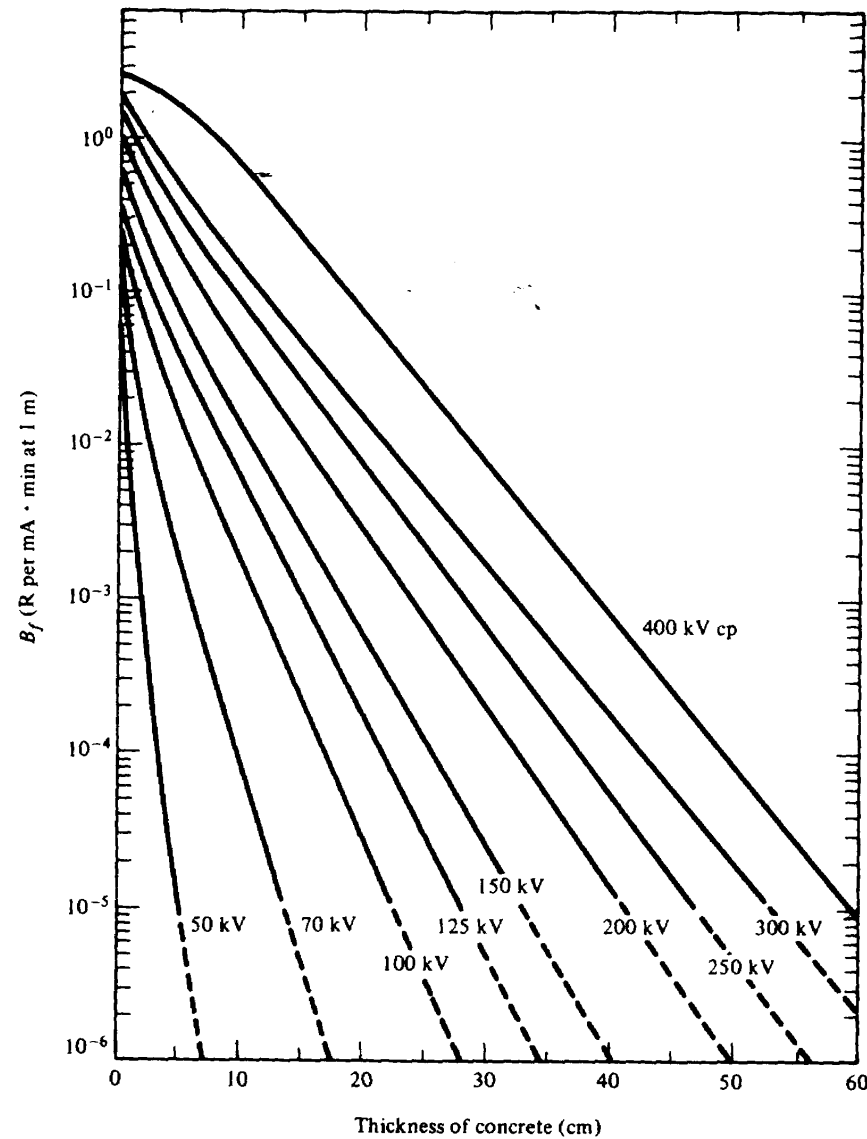


**Figure 6.19** Attenuation factor  $A_f$  for gamma photons from several radionuclides normally incident on lead slabs. [From NCRP (1976); by permission of the NCRP.]

# Broad Beam Consideration for Shielding Design



**Figure 6.21** Attenuation factors for the transmission of gamma photons from three radionuclide sources through a lead slab. The slab is uniformly illuminated at several angles of incidence with respect to the slab normal. Results for  $^{198}\text{Au}$  have been corrected so that only the transmission of the 0.411-MeV photon is shown. [After Kirn, Kennedy, and Wycoff (1954).]

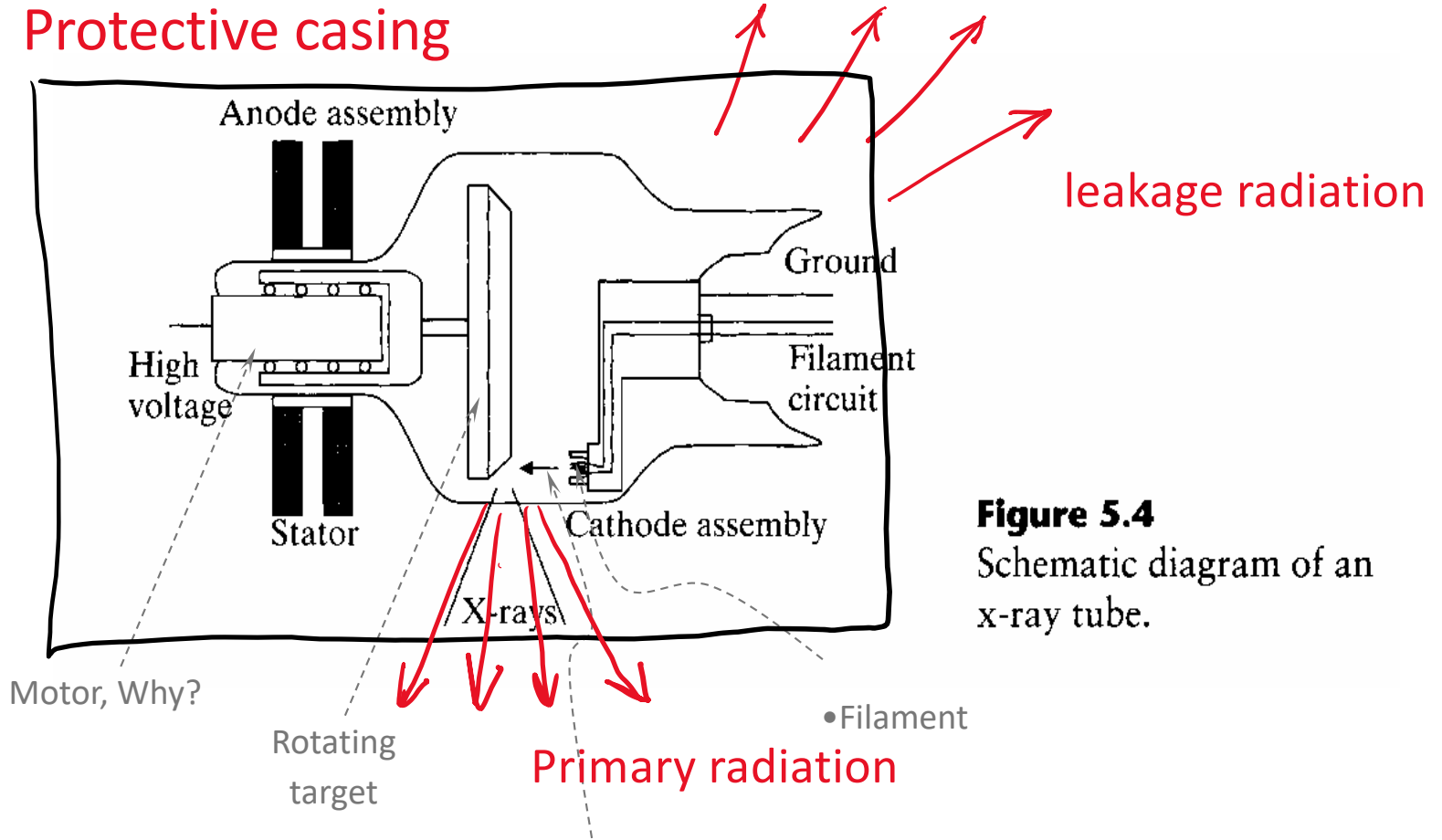


**Figure 6.23** Attenuation in concrete ( $2.35 \text{ g/cm}^3$  density) of x-rays produced by various peak potentials (kVp) with a  $90^\circ$  angle between the electron beam and the axis of the x-ray beam. Pulsed waveforms assumed except for the 400 kV cp curve which is for a constant potential generator. The x-ray beam filtrations are: 1 mm of Al for 50 kV, 1.5 for 70 kV, 2 mm for 100 kV, 3 mm for 100 to 300 kV, and 3 mm of Cu for 400 kV cp. [From NCRP (1976); by permission of the NCRP]

# X-Ray Shielding

- ☞ Basic terminologies related to the medical X-ray imaging setup
  - ☞ Tube voltage, workload, occupancy factor, and use factor.
- ☞ Analytical approaches for determining the shielding requirements for
  - ☞ the primary barrier
  - ☞ Secondary barrier

# A Typical X-Ray Sources



**Figure 5.4**  
Schematic diagram of an x-ray tube.

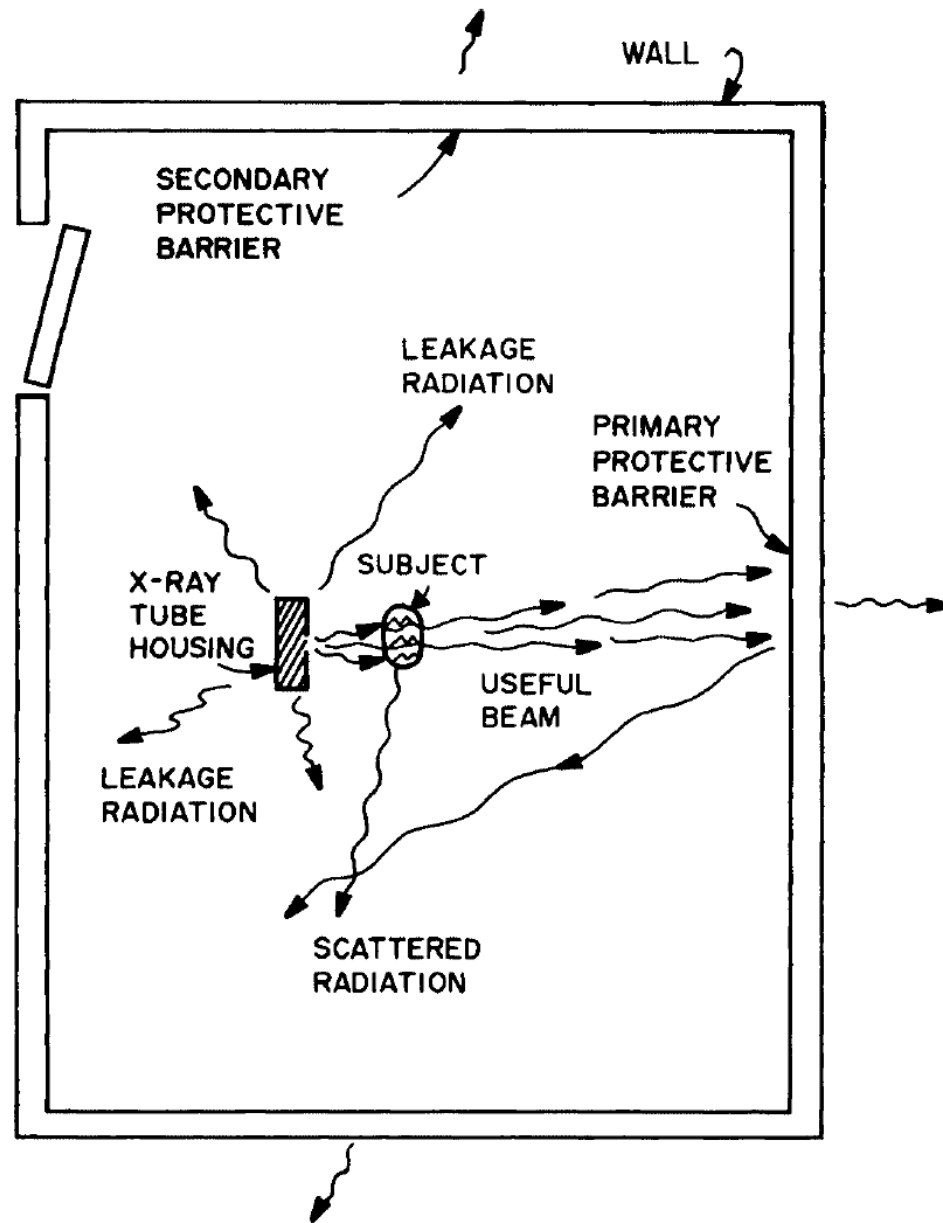


Fig. 15.8 Schematic plan view of X-ray room showing the different radiation components considered in the design of structural shielding to provide primary and secondary protective barriers.

## Key Characteristics of X-ray Source Operation

☞ Requirements for structural shielding is determined by considering

1. The maximum kilovoltage at which the X-ray tube is operated.
2. The maximum milliamperes of beam current.
3. The workload ( $W$ ), which is a measure, in suitable units, of the amount of use of an X-ray machine. For X-ray shielding design, workload is usually expressed in units of milliampere-minutes per week.
4. The use factor ( $U$ ), which is the fraction of the workload during which the useful beam is pointed in the direction under consideration.
5. The occupancy factor ( $T$ ), which is the factor by which the workload should be multiplied to correct for the degree or type of occupancy of the area in question. When adequate occupancy data are not available, the values for  $T$  given in Table 10.1 may be used as a guide in planning shielding.

## Considerations for X-ray Shielding Design

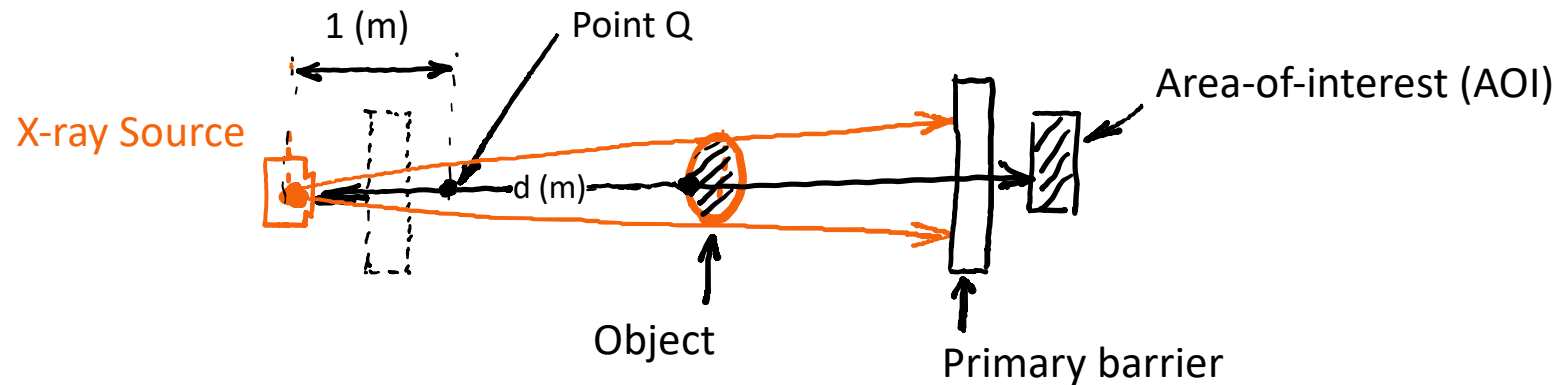
**TABLE 10.1. Occupancy Factors**

Full occupancy $T = 1$	Control space, wards, workrooms, darkrooms, corridors large enough to hold desks, waiting rooms, rest rooms used by occupationally exposed personnel, children's play areas, living quarters, occupied space in adjacent buildings
Partial occupancy $T = 1/4$	Corridors too narrow for desks, utility rooms, rest rooms not used routinely by occupationally exposed personnel, elevators run by operators, and uncontrolled parking lots
Occasional occupancy $T = 1/16$	Stairways, automatic elevators, outside areas used only for pedestrians or vehicular traffic, closets too small for future workrooms, toilets not used routinely by occupationally exposed personnel

☞ ICRP 60 Limits for X-ray exposure:

- ☞ 100mSv (100,000rems) over 5 years.
- ☞ Maximum dose in any single year <50mSv (50,000mrems).

## Design of the Primary Protection Barrier

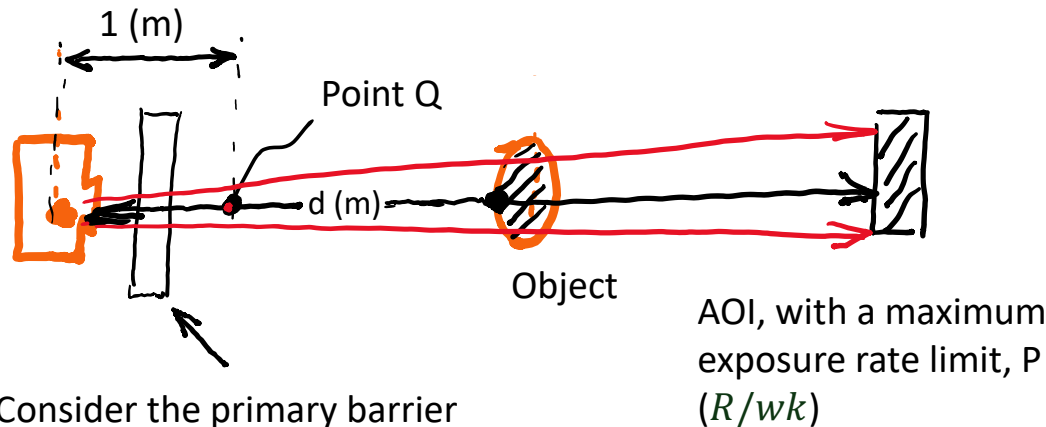


Consider that

- ✓ The maximum allowed exposure rate at the AOI is  $P$  (R/wk),
- ✓ The source is operated at a given workload,  $W$  (mA·min/wk),
- ✓ The AOI is occupied only for a fraction ( $T$ ) of time,
- ✓ The use factor of the source is  $U$  (the fraction of time when the source is pointing towards the AOI).

**Question:** What is the thickness of the primary barrier needed to ensure the exposure rate at the AOI stays below the limit  $P$ ?

# Design of the Primary Barrier (A Slightly Different Geometry )



Consider the primary barrier moved to here

AOI, with a maximum exposure rate limit,  $P$  ( $R/wk$ )

Consider that

- ✓ The maximum allowed exposure rate at the AOI is  $P$  ( $R/wk$ ),
- ✓ The source is operated at a given workload,  $W$  ( $mA \cdot min/wk$ ),
- ✓ The AOI is occupied only for a fraction ( $T$ ) of time,
- ✓ The use factor of the source is  $U$  (the fraction of time when the source is pointing towards the AOI).

Consider that the setup above delivers an exposure rate equal to  $P$  ( $R/wk$ ) at the AOI ...

**Question:** If the source (with the shielding in place) is only running at a unit workload of  $1 mA \cdot min/wk$ , what is the exposure rate at Point Q of 1 m away from the source?

$$K (R \cdot mA^{-1} \cdot min^{-1} \text{ at } 1m) = \frac{d^2 \cdot P (R/wk)}{W (mA \cdot min/wk) \cdot T \cdot U} \rightarrow \text{normalized shielded source output factor}$$

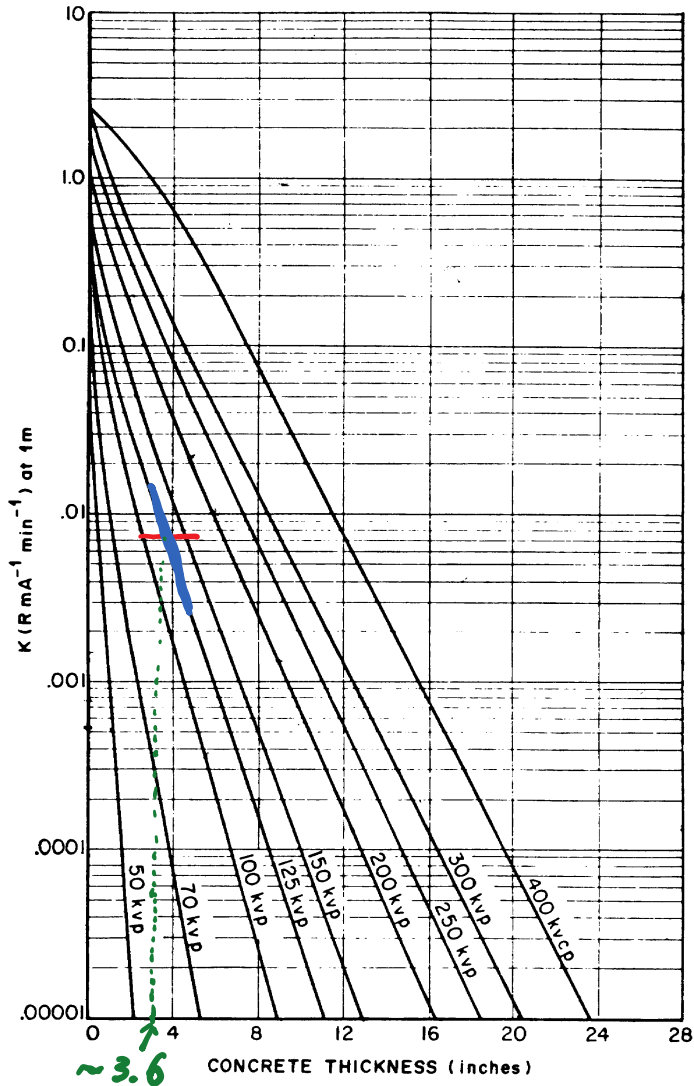


FIGURE 15.6. Attenuation in concrete of X rays produced with (peak) potential differences from 50 kVp to 400 kVp. (National Bureau of Standards Handbook 76, 1961, Washington, DC)

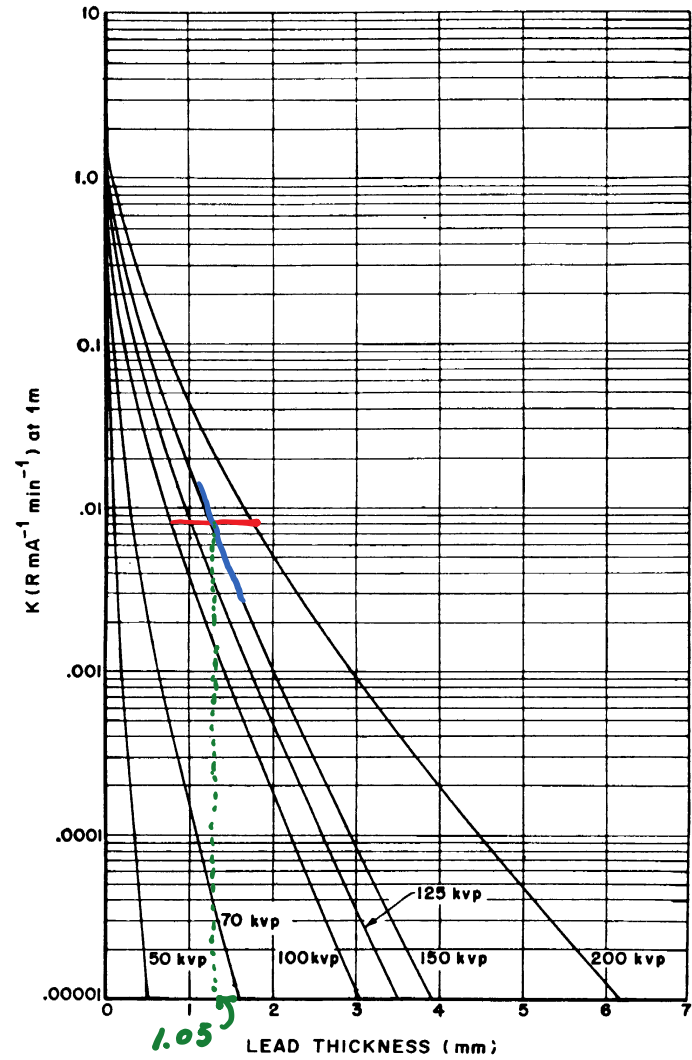


FIGURE 15.3. Attenuation in lead of X rays produced with (peak) potential differences from 50 kVp to 200 kVp. (National Bureau of Standards Handbook 76, 1961, Washington, DC)

1 Roentgen (R) =  $2.58 \times 10^{-4}$  (C/kg)

# Determining the Requirement for Primary Shielding

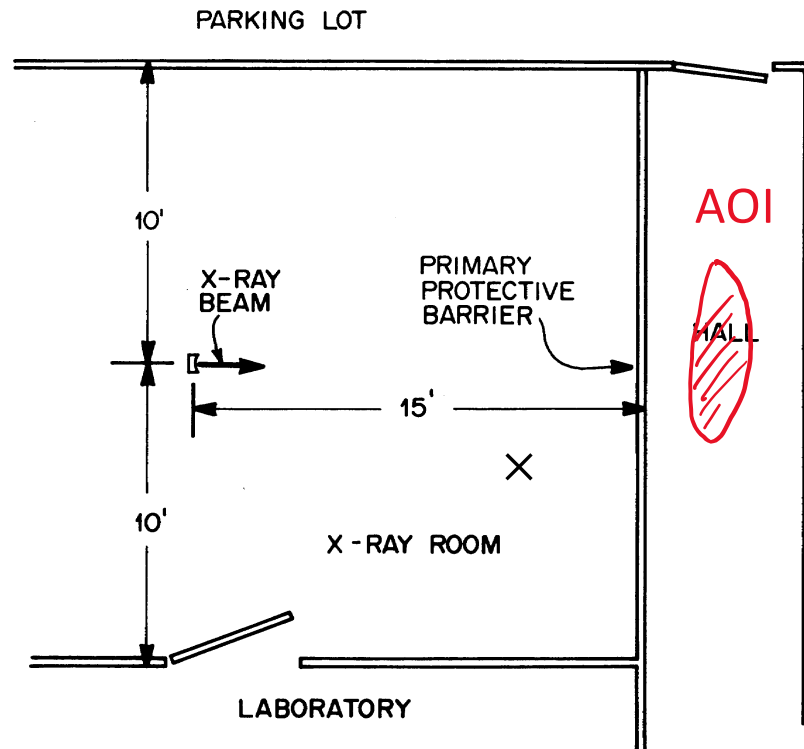


FIGURE 15.8. Schematic top view of an X-ray facility.

The exposure limit for the AOI is  
 $P=0.01 \text{ R/wk}$

Workload:  $W=220\text{mA} \times 1.5\text{min/wk}$

Use factor:  $U=1/3$

Occupancy factor:  $T=1/4$

## Example

A diagnostic X-ray machine is operated at 125 kVp and 220 mA for an average of  $90 \text{ s wk}^{-1}$ . Calculate the primary protective barrier thickness if lead or concrete alone were to be used to protect an uncontrolled hallway 15 ft from the tube target (Fig. 15.8). The useful beam is directed horizontally toward the barrier  $\frac{1}{3}$  of the time and vertically into the ground the rest of the time.

## Determine the Requirement for Primary Shielding

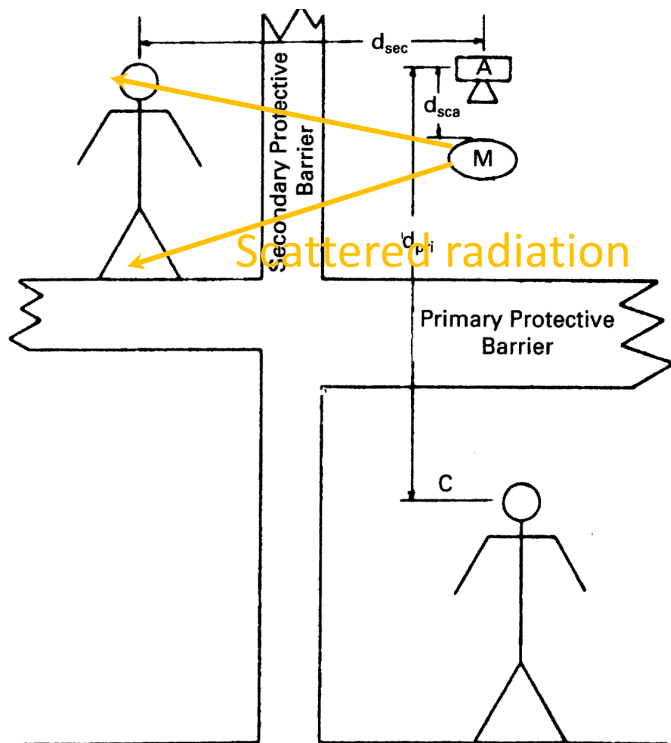
### *Solution*

For the uncontrolled hall,  $P = 0.01 \text{ R wk}^{-1}$ . The distance to the hall in meters is  $d = 15/3.28$ , and the workload is  $W = 220 \text{ mA} \times 1.5 \text{ min wk}^{-1} = 330 \text{ mA min wk}^{-1}$ . The use of factor is  $U = 1/3$  and the occupancy factor (Table 15.3) is  $T = 1/4$ . Equation (15.13) gives

$$K = \frac{d^2 P}{WUT} = \frac{0.01 \times (15/3.28)^2}{330 \times 1/3 \times 1/4} = 7.61 \times 10^{-3} \text{ R mA}^{-1} \text{ min}^{-1} \text{ at 1 m.}$$

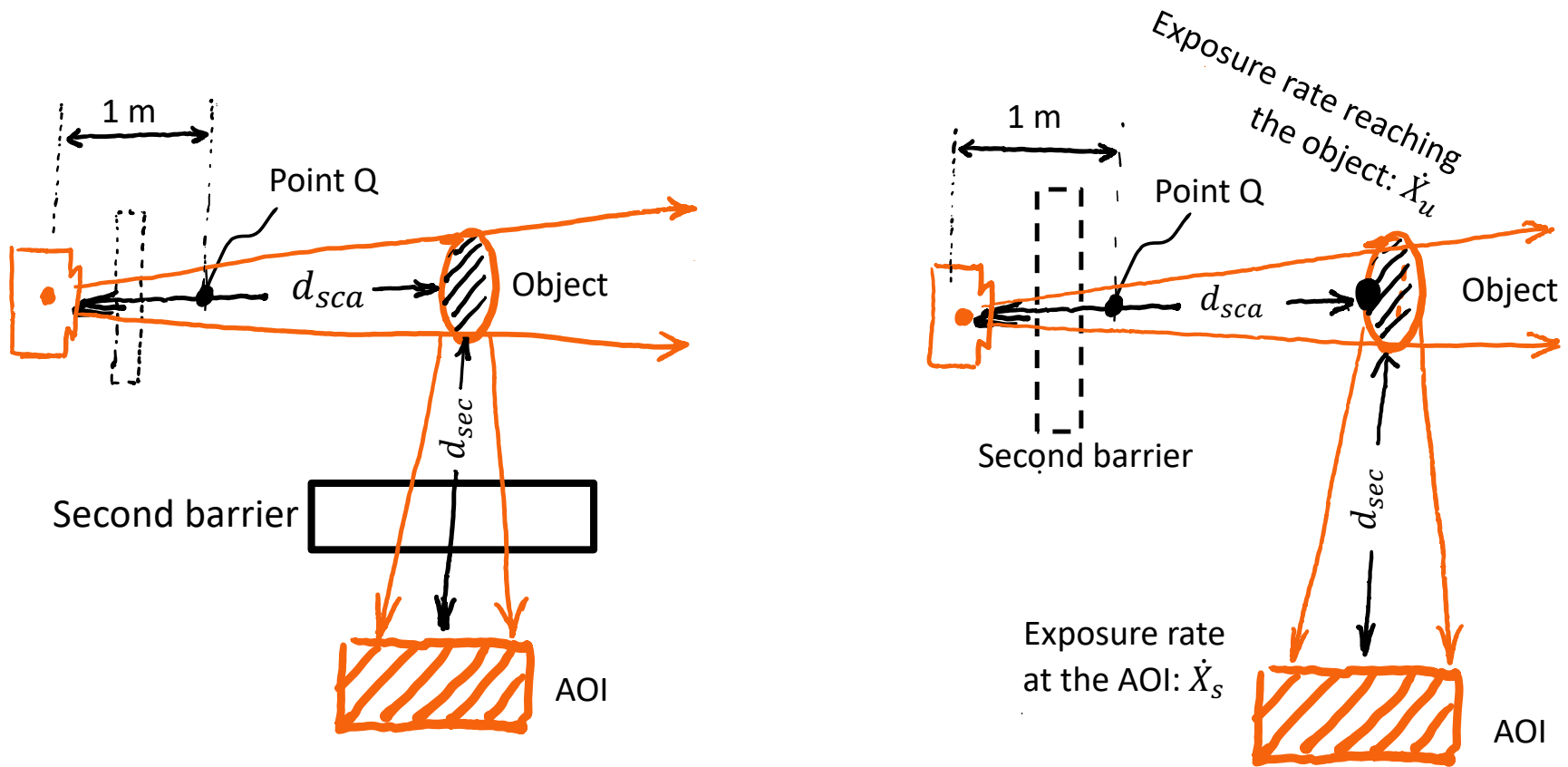
From Fig. 15.3 we find, for 125 kVp, that the required thickness of lead is about 1.05 mm; from Fig. 15.6, that of concrete is about 3.6 in.

# Design of the Secondary Protection Barrier



- ➡ Used against the **scattered** photons.
- ➡ Depends on
  - ➡ Scattering angle – assumed to be anisotropic.
  - ➡ The energy of the primary photons – a simple energy dependency is assumed to simplify the derivation.
  - ➡ Scattering area – assuming a perfect scattering plane.
- ➡ The required shielding needs to be determined separately for each of these components.

# Secondary Protection Barrier against Scattered Radiation



$$\dot{X}_S = \frac{a \times \dot{X}_u}{(d_{sec})^2} \cdot \frac{F}{400}$$

where  $a$  = ratio of scattered to incident radiation, Table 10.3;  
 $\dot{X}_u$  = exposure rate incident on scatterer;  
 $d_{sec}$  = distance from scatterer to point of interest;  
 $F$  = scattering field size, cm<sup>2</sup>; and  
 $t$  = exposure time.

## Secondary Protection Barrier against Scattered Radiation

- ☞ The amount of **scattered radiation** may be determined based on
  - ☞ primary beam energy,
  - ☞ scattering angle and
  - ☞ scattering area.
- ☞ If the exposure rate incident on the object is  $\dot{X}_u$ , then the exposure rate from scattered radiation **without shielding** may be given by

$$\dot{X}_S = \frac{a \times \dot{X}_u}{(d_{sec})^2} \cdot \frac{F}{400}$$

where  $a$  = ratio of scattered to incident radiation, Table 10.3;  
 $\dot{X}_u$  = exposure rate incident on scatterer;  
 $d_{sec}$  = distance from scatterer to point of interest;  
 $F$  = scattering field size, cm<sup>2</sup>; and

**TABLE 10.3. Ratio,  $a$ , of Scattered to Incident Exposure<sup>a</sup>**

Source	Scattering Angle (from Central Ray)					
	30	45	60	90	120	135
<b>X-rays</b>						
50 kV <sup>b</sup>	0.0005	0.0002	0.00025	0.00035	0.0008	0.0010
70 kV <sup>b</sup>	0.00065	0.00035	0.00035	0.0005	0.0010	0.0013
100 kV <sup>b</sup>	0.0015	0.0012	0.0012	0.0013	0.0020	0.0022
125 kV <sup>b</sup>	0.0018	0.0015	0.0015	0.0015	0.0023	0.0025
150 kV <sup>b</sup>	0.0020	0.0016	0.0016	0.0016	0.0024	0.0026
200 kV <sup>b</sup>	0.0024	0.0020	0.0019	0.0019	0.0027	0.0028
250 kV <sup>b</sup>	0.0025	0.0021	0.0019	0.0019	0.0027	0.0028
300 kV <sup>b</sup>	0.0026	0.0022	0.0020	0.0019	0.0026	0.0028
4 MV <sup>c</sup>	—	0.0027	—	—	—	—
6 MV <sup>d</sup>	0.007	0.0018	0.0011	0.0006	—	0.0004
<b>Gamma rays</b>						
<sup>137</sup> Cs <sup>e</sup>	0.0065	0.0050	0.0041	0.0028	—	0.0019
<sup>60</sup> Co <sup>f</sup>	0.0060	0.0036	0.0023	0.0009	—	0.0006

<sup>a</sup>Scattered radiation measured at 1 m from phantom when field area is 400 cm<sup>2</sup> at the phantom surface; incident exposure measured at center of field 1 m from the source but without phantom.

<sup>b</sup>From Trout and Kelley (*Radiology* 104:161, 1972. By permission). Average scatter for beam centered and beam at edge of typical patient cross-section phantom. Peak pulsating X-ray tube potential.

<sup>c</sup>From Greene and Massey (*Br J Radiol* 34:389, 1961. By permission.), cylindrical phantom.

<sup>d</sup>From Karzmark and Capone (*Br J Radiol* 41:222 1968. By permission.), cylindrical phantom.

<sup>e</sup>Interpolated from Frantz and Wyckoff (*Radiology* 73:263 1959. By permission). These data were obtained from a slab placed obliquely to the central ray. A cylindrical phantom should give smaller values.

<sup>f</sup>From Mooney and Braestrup (AEC Report NYO 2165, 1967. By permission.), modified for  $F = 400$  cm<sup>2</sup>.

Source: From NCRP 49. By permission.

## Secondary Protection Barrier against Scattered Radiation

In designing secondary protective barriers, we make the following simplifying and conservative assumptions:

1. The energy of the scattered radiation, when the X-rays are generated at 500 kV or less, is equal to the energy of the useful beam.
2. Primary X-ray beams generated at voltages greater than 500 kV are degraded in energy to that of a 500-kV beam after being scattered, and the exposure rate at 1 m from the scatterer is 0.1% of that in the useful beam at the point of scattering.

# Secondary Protection Barrier against Scattered Radiation

If the shielded source (shown below) running with the given (W, U, T) delivers an exposure rate  $P$  at the AOI, then the same shielded source, while running with 1 (mA · min), should deliver an exposure rate  $K$  to 1 m away,

$$K = \frac{\dot{X}_Q}{W \cdot U \cdot T} = \frac{P(R/wk) \cdot d_{sec}^2 \cdot d_{sca}^2}{a \cdot W(mA \cdot min/wk) \cdot T} \cdot \frac{400(cm^2)}{F(cm^2)}$$

to Point Q ...

Exposure rate at point Q :

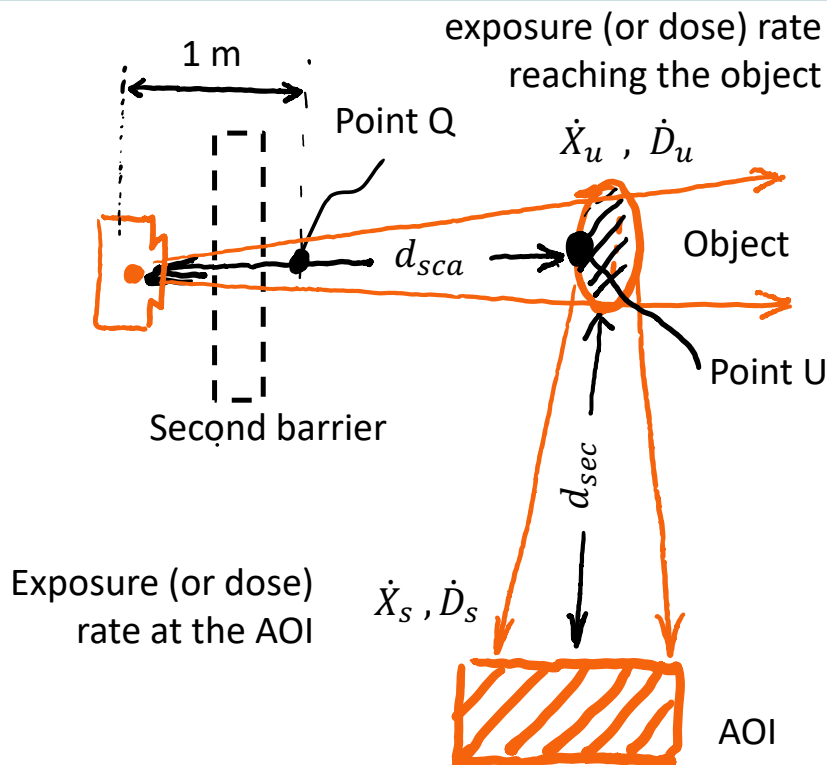
$$\dot{X}_Q = \dot{X}_u \cdot d_{sca}^2$$

$$= \frac{P(R/wk) \cdot d_{sec}^2 \cdot d_{sca}^2}{a} \cdot \frac{400(cm^2)}{F(cm^2)}$$

Exposure rate at point U:

$$\dot{X}_u = \frac{P(R/wk) \cdot d_{sec}^2}{a} \cdot \frac{400(cm^2)}{F(cm^2)}$$

where  $F$  (cm) is the actual target area of the object



Assuming the exposure rate at the AOI is

$$\dot{X}_s = P(R/wk)$$

# Secondary Protection Barrier against Scattered Radiation

Since  $It = WT$ , and since  $U = 1$  for scattered radiation, then,

**Normalized shielded source output factor:** the dose or exposure that the shielded source running at 1mA for 1 minute should deliver to a reference point at 1 m away

$$K = \frac{P}{aWT} \times (d_{sca})^2 \times (d_{sec})^2 \times \frac{400}{F}$$

Dist. from the source to the scatterer

Dist. from the scatterer to the AOI

Actual size of the scatterer/object

The occupancy of the AOI

Workload – the amount of usage of the x-ray tube (mA·min/wk)

a: ratio between incident exposure and scattered exposure measured at 1 m from the object/scatterer, whose scattering area is assumed to be 400 cm<sup>2</sup>

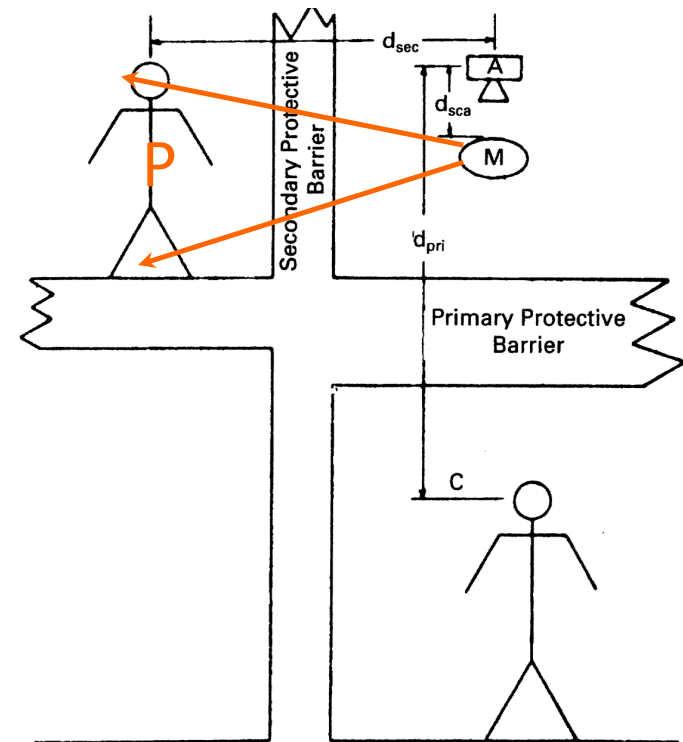
## Secondary Protection Barrier for Scattered Radiation

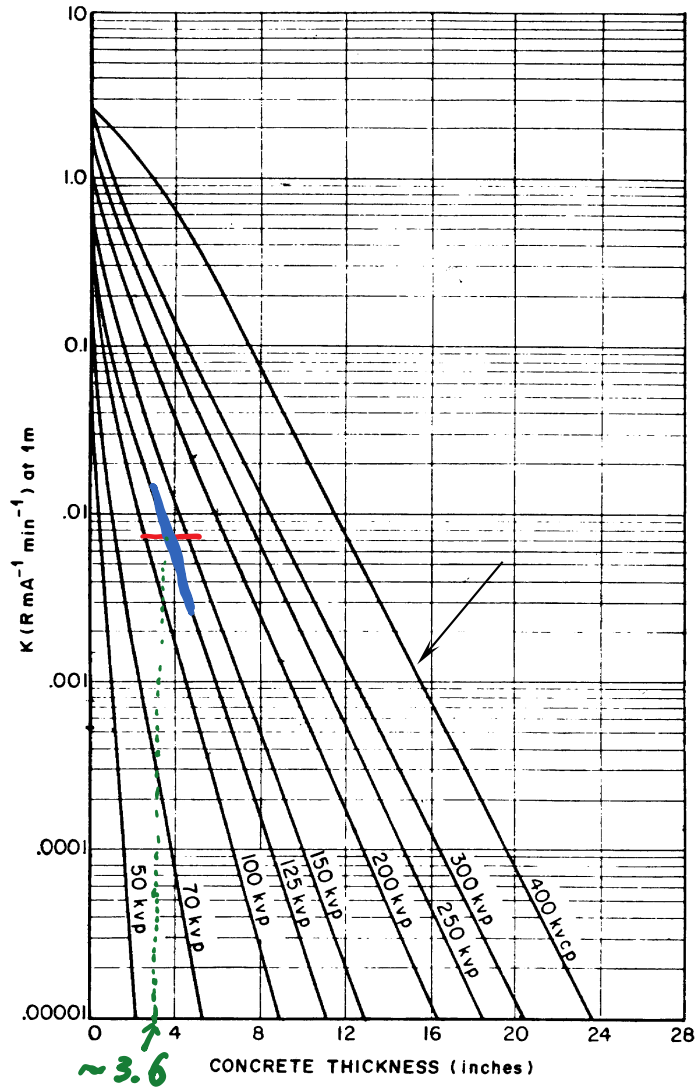
☞ The exposure rate that the same source would deliver at 1 m away (within the primary beam) by running at 1 mA for 1 min is given by

$$K = \frac{P \times (d_{sca})^2 \times (d_{sec})^2 \times 400}{a \times W \times T \times F \times f} \quad (R \cdot mA^{-1} \cdot min^{-1} \text{ at } 1m)$$

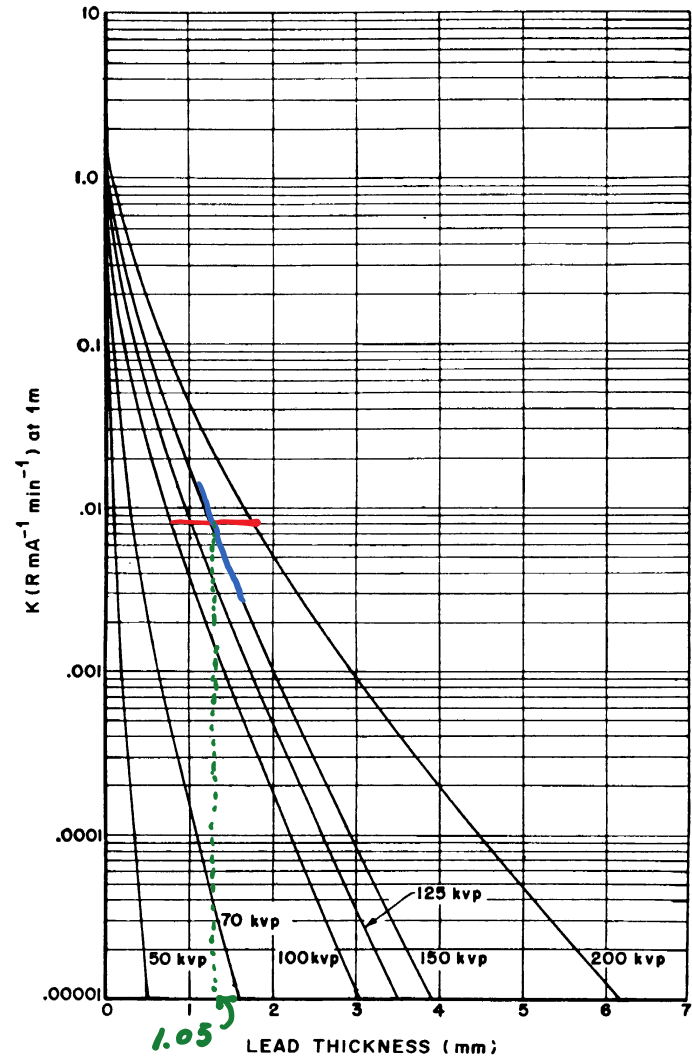
Correction factor, whose value increases with the increasing HV of the tube.

kV	$f$
500 or less	1
1000	20
2000	300
3000	700





**FIGURE 15.6.** Attenuation in concrete of X rays produced with (peak) potential differences from 50 kVp to 400 kVp. (*National Bureau of Standards Handbook 76, 1961, Washington, DC*)



**FIGURE 15.3.** Attenuation in lead of X rays produced with (peak) potential differences from 50 kVp to 200 kVp. (*National Bureau of Standards Handbook 76, 1961, Washington, DC*)

# Design of An X-ray Shielding Structure

## Example 10.9

The room shown in Fig. 10.17 will be used for diagnostic radiology. From the floor to the ceiling is 9 ft 7 in. (292 cm). The room above is another office that is not controlled by the radiologist. The X-ray room is on the ground floor of the building; there is no occupied space below. The floor and ceiling are made of concrete 5 in. (12.7 cm) thick; wall A is made of concrete 4 in. (10.2 cm) thick. Walls B, C, and D are made of hollow tile and thin plaster, as shown in section A-A. The maximum machine ratings are 125 kVp and 200 mA. As a fluoroscope, the machine will be operated 5 h a week at 3.5 mA, and for radiography the machine will be operated at 200 mA for 2 min in a week. The average target to skin distance is 0.5 m. Compute the required shielding for the ceiling and for wall B.

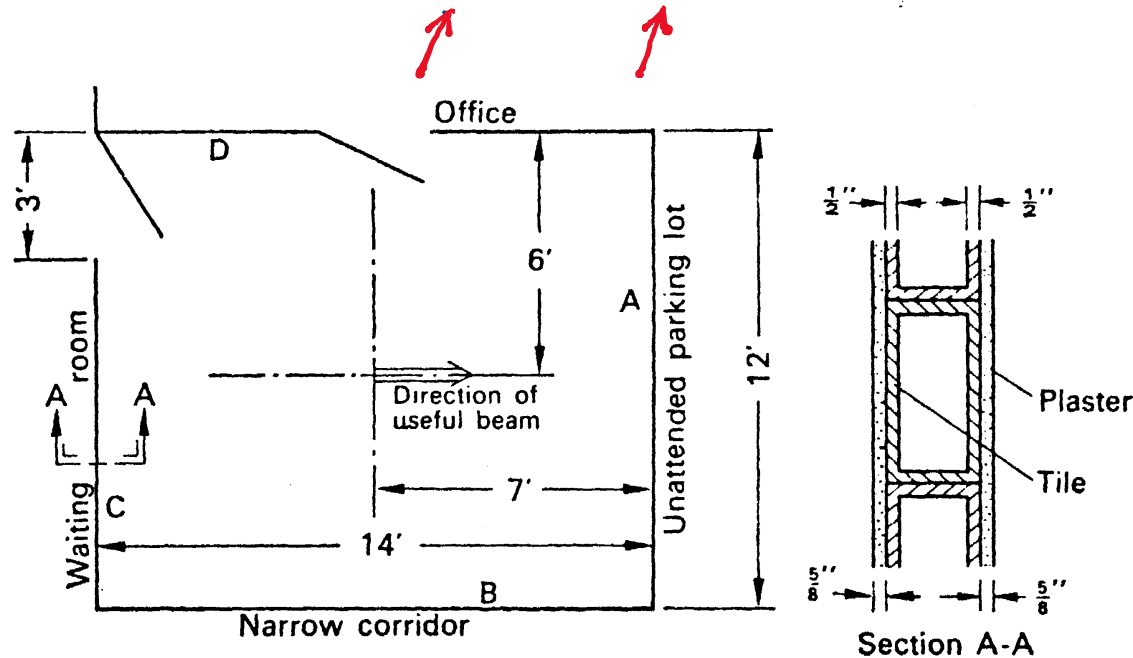
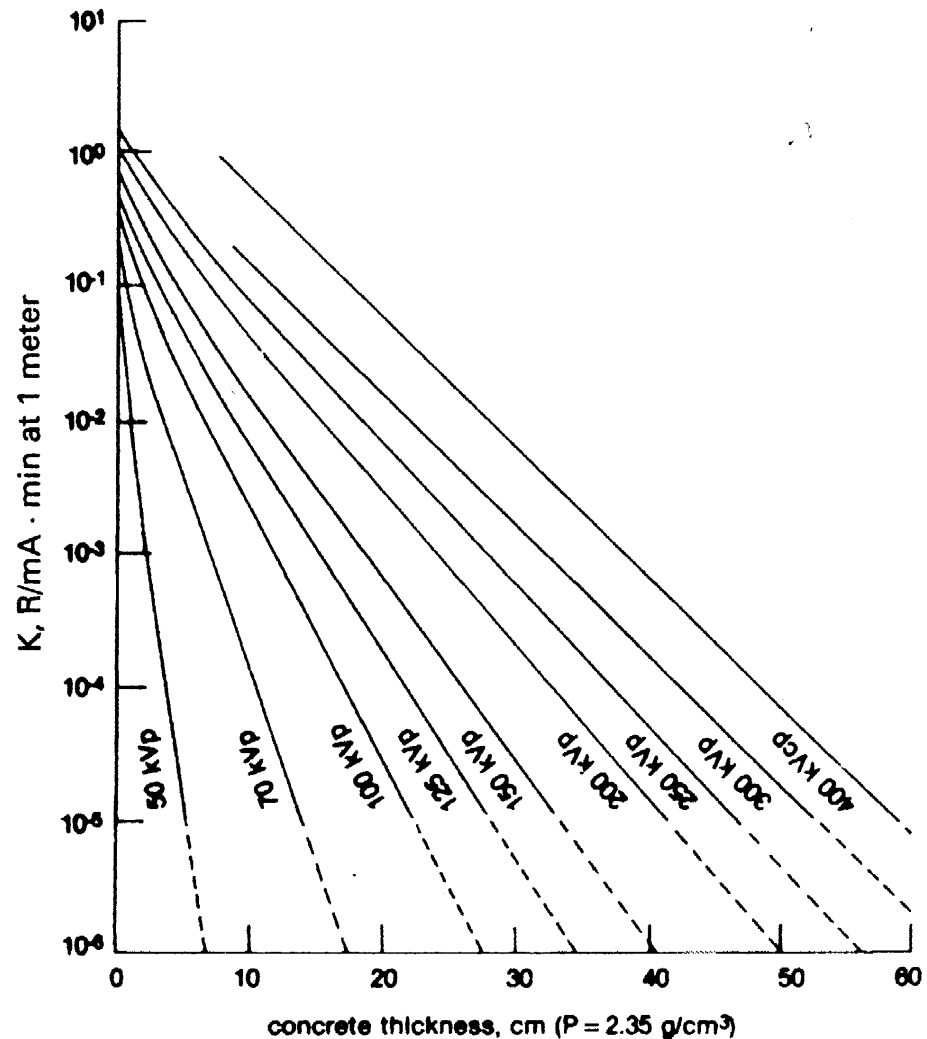


FIGURE 10.17. Layout of diagnostic X-ray room of Example 10.9.

# Design of An X-ray Shielding Structure

**FIGURE 10.13.** Attenuation in concrete of X-rays produced by potentials of 50 to 300 kVp; 400 kV constant potential. The measurements were made with a  $90^\circ$  angle between the electron beam and the axis of the X-ray beam. The curves for 50 to 300 kV are for a pulsed waveform. The filtrations were 1 mm Al for 50 kV, 1.5 mm Al for 70 kV, 2 mm Al for 100 kV, and 3 mm Al for 125, 150, 200, 250, and 300 kV. The 400-kV curve was interpolated from data obtained with a constant potential generator and inherent filtration of approximately 3 mm Cu. (From NCRP Report No. 49, *Structural Shielding Design and Evaluation for Medical Use of X-Rays and Gamma Rays of Energies up to 10 MeV*, 1976. Full-size reproductions of the figures giving barrier requirements are available from the NCRP as an adjunct to the report. By permission.)



## Step 1: Design of the primary barrier

Considering

- Directly irradiating Wall A at a distance  $d=7$  feet,
- Workload:  $W=2$  (mins/week)  $\times$  200 (mA),
- Use factor:  $U=1$ ,
- Occupancy factor:  $T=1/4$ ,
- Maximum allowed weekly dose to an uncontrolled area:  $P=0.02$  mSv/wk.

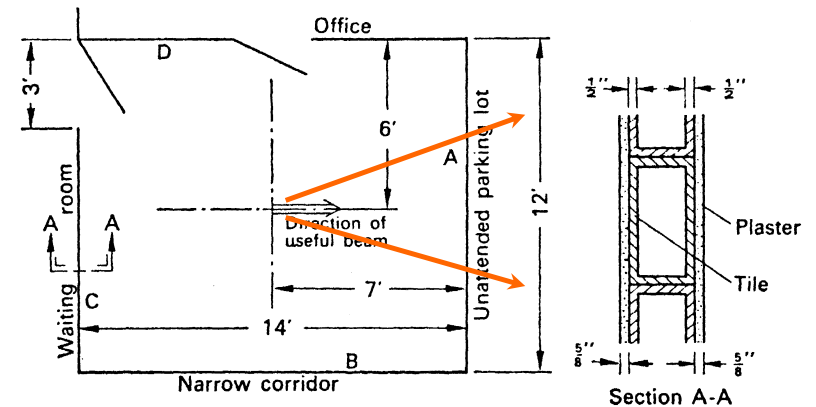


FIGURE 10.17. Layout of diagnostic X-ray room of Example 10.9.

Consider a shielded X-ray source that is running with given ( $W$ ,  $U$ ,  $T$ ) and delivers a dose rate (or exposure) rate  $P$  at the AOI. If we move the shielding in front of the same source, then the **dose (or exposure) that this shielded source would deliver to point Q at a distance of 1 m away by running the source with 1 mA tube current for 1 min** is given by

$$K_{\text{dose}} = \frac{d^2 \cdot P}{W \cdot U \cdot T} = 9.07 \times 10^{-4} \text{ mSv}/(\text{mA} \cdot \text{min}) \text{ at } 1\text{m}$$

Then we convert the maximum allowed dose  $K_{\text{dose}}$  to the maximum allowed exposure  $K_{\text{exposure}}$ , so we could use existing data to derive the shielding requirement.

For this purpose, we consider

- The quality factor of photons is 1, and
- 1 X Unit = 34Gy
- 1 X Unit = 3881 Roentgen

$$\begin{aligned} K_{\text{exposure}} &= K_{\text{dose}}(\text{Sv})/34 \times 3881 \\ &= 1.03 \times 10^{-4} \text{ R}/(\text{mA} \cdot \text{min}) \text{ at } 1\text{ m} \end{aligned}$$

## Step 1: Design of the primary barrier (continued)

The maximum exposure that is allowed at a unit distance and delivered by the source running at the "standard" operating condition ( $W=1, U=1, T=1$ )

$$K_{\text{exposure}} = K_{\text{dose}} / 34 \times 3881 = 1.03 \times 10^{-4} \text{ R}/(\text{mA} \cdot \text{min}) \text{ at } 1 \text{ m}$$

Given the source is 125 kVp, we can use the figure on the right to find the thickness of the concrete slab needed to achieve the maximum allowed exposure limit  $K_{\text{exposure}}$  is 21.5 cm.

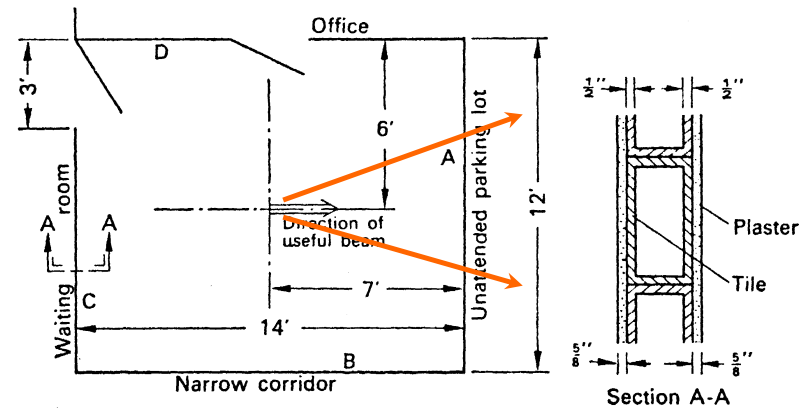
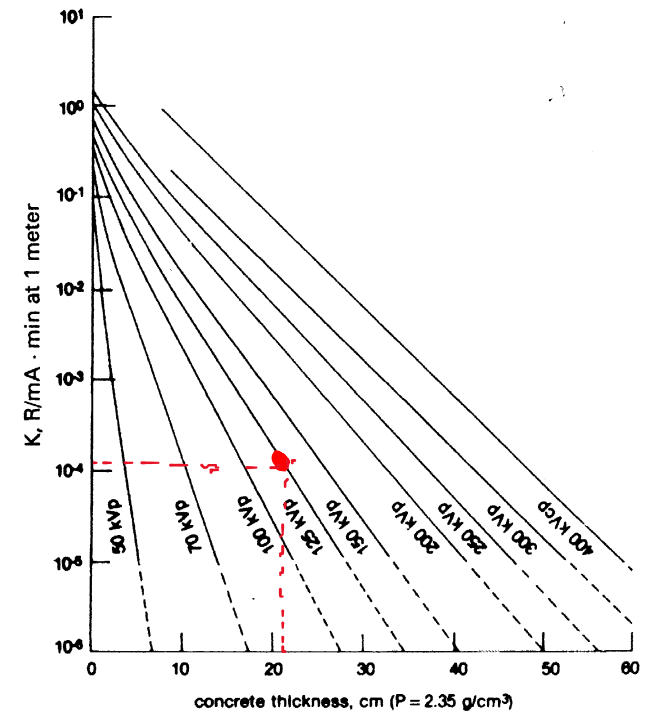


FIGURE 10.17. Layout of diagnostic X-ray room of Example 10.9.

**FIGURE 10.13.** Attenuation in concrete of X-rays produced by potentials of 50 to 300 kVp; 400 kV constant potential. The measurements were made with a  $90^\circ$  angle between the electron beam and the axis of the X-ray beam. The curves for 50 to 300 kV are for a pulsed waveform. The filtrations were 1 mm Al for 50 kV, 1.5 mm Al for 70 kV, 2 mm Al for 100 kV, and 3 mm Al for 125, 150, 200, 250, and 300 kV. The 400-kV curve was interpolated from data obtained with a constant potential generator and inherent filtration of approximately 3 mm Cu. (From NCRP Report No. 49, *Structural Shielding Design and Evaluation for Medical Use of X-Rays and Gamma Rays of Energies up to 10 MeV*, 1976. Full-size reproductions of the figures giving barrier requirements are available from the NCRP as an adjunct to the report. By permission.)



## Step 2: Design of the second barrier (ceiling) against scattered radiation

Considering

- Workload:  $W=2 \text{ (mins/week)} \times 200 \text{ (mA)}=1200 \text{ (mA} \cdot \text{min/week)}$ ,
- Use factor:  $U=1$ ,
- Occupancy factor:  $T=1$ ,
- $d_{sca}=0.5 \text{ m}$  (distance between the source and the patient),
- $d_{sec}=2.5 \text{ m}$  (distance between the patient and the area of interest, AOI),
- $a=0.0015$  from Table 10.17, Ratio of scattered to incident radiation
- Target area of the patient to the incident X-rays,  $F=400 \text{ cm}^2$ ,
- Maximum allowed weekly dose to an uncontrolled area:  $P=0.02 \text{ mSv/wk}$ ,

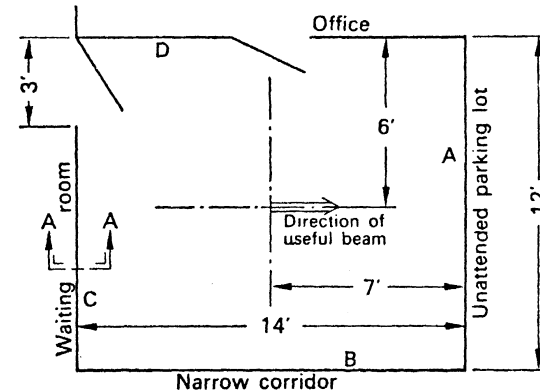


FIGURE 10.17. Layout of diagnostic X-ray room of Example 10.9.

The maximum dose that is allowed at a unit distance delivered by the source running at 1 mA for 1 min is given by

$$\begin{aligned}
 K &= \frac{P}{a \cdot W \cdot T} \cdot d_{sca}^2 \cdot d_{sec}^2 \cdot \frac{400 \text{ cm}^2}{F \text{ cm}^2} \\
 &= 1.4 \times 10^{-3} \text{ rem}/(\text{mA} \cdot \text{min}) \text{ at } 1\text{m} \\
 &= 1.4 \times 10^{-5} \text{ Sv}/(\text{mA} \cdot \text{min}) \text{ at } 1\text{m}
 \end{aligned}$$

TABLE 10.3. Ratio,  $a$ , of Scattered to Incident Exposure<sup>a</sup>

Source	Scattering Angle (from Central Ray)					
	30	45	60	90	120	135
X-rays						
50 kV <sup>b</sup>	0.0005	0.0002	0.00025	0.00035	0.0008	0.0010
70 kV <sup>b</sup>	0.00065	0.00035	0.00035	0.0005	0.0010	0.0013
100 kV <sup>b</sup>	0.0015	0.0012	0.0012	0.0013	0.0020	0.0022
125 kV <sup>b</sup>	0.0018	0.0015	0.0015	0.0015	0.0023	0.0025
150 kV <sup>b</sup>	0.0020	0.0016	0.0016	0.0016	0.0024	0.0026
200 kV <sup>b</sup>	0.0024	0.0020	0.0019	0.0019	0.0027	0.0028
250 kV <sup>b</sup>	0.0025	0.0021	0.0019	0.0019	0.0027	0.0028
300 kV <sup>b</sup>	0.0026	0.0022	0.0020	0.0019	0.0026	0.0028
4 MV <sup>c</sup>	—	0.0027	—	—	—	—
6 MV <sup>d</sup>	0.007	0.0018	0.0011	0.0006	—	0.0004
Gamma rays						
<sup>137</sup> Cs <sup>e</sup>	0.0065	0.0050	0.0041	0.0028	—	0.0019
<sup>60</sup> Co <sup>f</sup>	0.0060	0.0036	0.0023	0.0009	—	0.0006

<sup>a</sup>Scattered radiation measured at 1 m from phantom when field area is 400 cm<sup>2</sup> at the phantom surface; incident exposure measured at center of field 1 m from the source but without phantom.

<sup>b</sup>From Trout and Kelley (*Radiology* 104:161, 1972. By permission). Average scatter for beam centered and beam at edge of typical patient cross-section phantom. Peak pulsating X-ray tube potential.

<sup>c</sup>From Greene and Massey (*Br J Radiol* 34:389, 1961. By permission.), cylindrical phantom.

<sup>d</sup>From Karzmark and Capone (*Br J Radiol* 41:222, 1968. By permission.), cylindrical phantom.

<sup>e</sup>Interpolated from Frantz and Wyckoff (*Radiology* 73:263, 1959. By permission). These data were obtained from a slab placed obliquely to the central ray. A cylindrical phantom should give smaller values.

<sup>f</sup>From Mooney and Brastrup (AEC Report NYO 2165, 1967. By permission.), modified for  $F = 400 \text{ cm}^2$ .

Source: From NCRP 49. By permission.

## Step 2: Design of the second barrier (ceiling) against scattered radiation (continued)

The maximum dose that is allowed at a unit distance delivered by the source running at 1 mA for 1 min is given by

$$\begin{aligned}
 K_{\uparrow} &= \frac{P}{a \cdot W \cdot T} \cdot d_{sca}^2 \cdot d_{sec}^2 \cdot \frac{400 \text{ cm}^2}{F \text{ cm}^2} \\
 &= 1.4 \times 10^{-3} \text{ rem}/(\text{mA} \cdot \text{min}) \text{ at } 1 \text{ m} \\
 &= 1.4 \times 10^{-5} \text{ Sv}/(\text{mA} \cdot \text{min}) \text{ at } 1 \text{ m}
 \end{aligned}$$

Then we convert the maximum allowed dose  $K_{dose}$  to the maximum allowed exposure  $K_{exposure}$ , so we could use existing data to derive the shielding requirement.

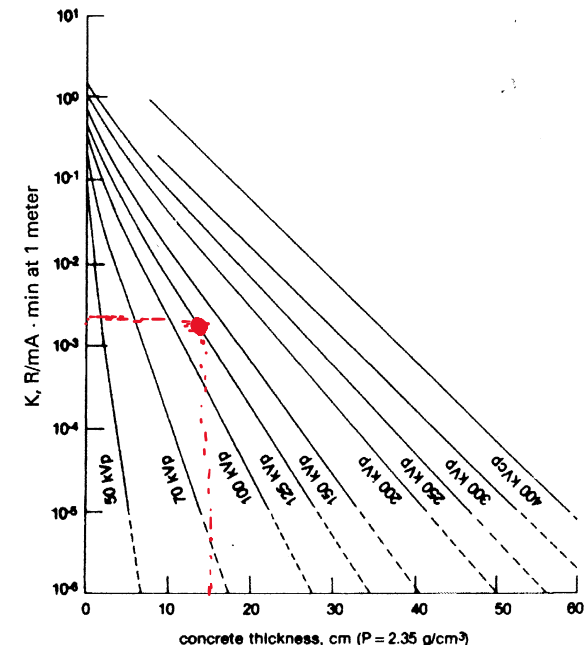
For this purpose, we consider

- The quality factor of photons is 1, and
- 1 X Unit = 34Gy
- 1 X Unit = 3881 Roentgen

$$\begin{aligned}
 K_{exposure} &= K_{dose}(\text{Sv})/34 \times 3881 \\
 &= 1.4 \times 10^{-5} \text{ Sv}/(\text{mA} \cdot \text{min})/34 * 3881 = \\
 &\quad 1.6 \times 10^{-3} \text{ R}/(\text{mA} \cdot \text{min}) \text{ at } 1 \text{ m}
 \end{aligned}$$

Using the figure on the right, we find that the thickness of concrete needed to bring the exposure down to the maximum allowed value  $K_{exposure}$  is 14 cm.

**FIGURE 10.13.** Attenuation in concrete of X-rays produced by potentials of 50 to 300 kVp; 400 kV constant potential. The measurements were made with a 90° angle between the electron beam and the axis of the X-ray beam. The curves for 50 to 300 kV are for a pulsed waveform. The filtrations were 1 mm Al for 50 kV, 1.5 mm Al for 70 kV, 2 mm Al for 100 kV, and 3 mm Al for 125, 150, 200, 250, and 300 kV. The 400-kV curve was interpolated from data obtained with a constant potential generator and inherent filtration of approximately 3 mm Cu. (From NCRP Report No. 49, *Structural Shielding Design and Evaluation for Medical Use of X-Rays and Gamma Rays of Energies up to 10 MeV*, 1976. Full-size reproductions of the figures giving barrier requirements are available from the NCRP as an adjunct to the report. By permission.)



# Shielding for Beta Sources

## Beta Ray Shielding – An Example

### *Example 10.10*

Fifty milliliters of aqueous solution containing  $37 \times 10^4$  MBq (10 Ci) carrier free  $^{90}\text{Sr}$  in equilibrium with  $^{90}\text{Y}$  is to be stored in a laboratory. The health physicist requires the dose-equivalent rate at a distance of 50 cm from the center of the solution to be no greater than 0.1 mSv (10 mrem) per hour. Design the necessary shielding to meet this requirement.

The maximum and mean beta-ray energies of  $^{90}\text{Sr}$  and  $^{90}\text{Y}$  are as follows:

	$E_{\text{max}}, \text{MeV}$	$E_{\text{mean}}, \text{MeV}$
$^{90}\text{Sr}$	0.54	0.19
$^{90}\text{Y}$	2.27	0.93
Sum		1.12

## Part 1: Shielding for the Beta Particles

The range of the beta particles with the maximum energy of 2.27 MeV is  $1.19\text{g/cm}^2$  in polyethylene (density:  $0.959\text{g/cm}^3$ ). If polyethylene is used as the shielding material, the wall thickness of the shielding is chosen to be equal to the range,

$$t_{\text{wall}} = \frac{1.19\text{ g/cm}^2}{0.959\text{ g/cm}^3} = 1.26\text{cm}$$

## Part 2: Extra Shielding for the Bremsstrahlung X-rays

The rate of beta energy emitted by the source of activity A is

$$\dot{E}_\beta = A \cdot \bar{E}_\beta = 3.7 \times 10^{11} \text{ Bq} \times 1.12 \text{ MeV/t}$$

The fraction of beta energy being converted to bremsstrahlung is

$$f = 3.5 \times 10^{-4} \cdot z_{\text{eff}} \cdot E_{\text{max}}(\text{MeV}).$$

$z_{\text{eff}}$  is the effective Z of the absorbing material

$$z_{\text{eff}} = \frac{\sum_i N_i z_i^2}{\sum_i N_i z_i} = 6.6 \text{ for polyethylene}$$

where


$N_i$ : number fraction of the  $i$ 'th element

$Z_i$ : atomic mass number of the  $i$ 'th element.

So

$$f = 3.5 \times 10^{-4} \cdot 6.6 \cdot 2.27 = 5.24 \times 10^{-3}$$

Maximum energy of the beta particles



## Part 2: Extra Shielding for Bremsstrahlung X-rays

Assuming all X-rays are emitted from a point, the dose rate at a distance  $d$  is

Linear energy absorption coefficient for 2.27 MeV photons in air

$$\dot{D} = \frac{f \cdot \dot{E}_\beta \left(\frac{\text{MeV}}{\text{s}}\right) \cdot 1.6 \times 10^{-13} \left(\frac{\text{J}}{\text{MeV}}\right) \cdot \mu_{en} \left(\frac{1}{\text{m}}\right) \cdot 3.6 \times 10^3 \left(\frac{\text{s}}{\text{h}}\right)}{\rho_{air} (\text{kg}/\text{m}^3) \cdot 4\pi \cdot d^2 (\text{m}^2) \cdot 10^{-3} (\text{Gy}/\text{mSv})} = 1.14 \text{ mSv/h}$$

Suppose all bremsstrahlung photons have the maximum beta particle energy, 2.27 MeV.

The linear attenuation coefficient for 2.27 MeV photons in lead is  $0.51 \text{ cm}^{-1}$ . The thickness of lead needed to bring the dose at 0.5 m away to 0.1 mSv is given by

$$0.1 = 1.14 \cdot e^{-0.51(\text{cm}^{-1}) \cdot t(\text{cm})}$$

$$t = -\frac{1}{0.51} \ln \frac{0.1}{1.14} = 4.8 \text{ cm}$$

# Neutron Shielding Design

- ☞ Review of dose calculation for fast neutrons and thermal neutrons.
- ☞ An example of shielding design for an alpha-neutron source.

## Radiation Dose from Fast Neutrons (Revisited)

### Example 6.16

What is the absorbed dose rate to soft tissue in a beam of 5-MeV neutrons whose intensity is 2000 neutrons per square centimeter per second?

Substituting the appropriate values into Eq. (6.103) yields

$$\begin{aligned}\dot{D}_n &= \frac{2 \times 10^3 \text{ n/cm}^2 \cdot \text{s} \times 5 \text{ MeV/n} \times 1.6 \times 10^{-13} \text{ J/MeV} \times 51.17 \text{ cm}^2/\text{kg}}{1 \text{ J/kg} \cdot \text{Gy}} \\ &= 8.19 \times 10^{-8} \text{ Gy/s} \quad (8.19 \times 10^{-6} \text{ rad/s}),\end{aligned}$$

$$\dot{D}_n(E) = \frac{\phi(E)E \sum_i N_i \sigma_i f}{1 \text{ J/kg} \cdot \text{Gy}}, \quad (6.103)$$

where  $\phi(E)$  = flux of neutrons whose energy is  $E$ , in neutrons/cm<sup>2</sup> · s,  
 $E$  = neutron energy, in joules,  
 $N_i$  = atoms per kilogram of the  $i$ th element,  
 $\sigma_i$  = scattering cross section of the  $i$ th element for neutrons of energy  $E$ , in barns  $\times 10^{-24}$  cm<sup>2</sup>,  
 $f$  = mean fractional energy transferred from neutron to scattered atom during collision with neutron.

## Radiation Dose from Thermal Neutrons (Revisited)

☞ Two reactions are normally considered, namely  $^{14}\text{N}(n,p)^{14}\text{C}$  and  $^1\text{H}(n,r)^2\text{H}$  reactions.

☞ For the  $^{14}\text{N}(n,p)^{14}\text{C}$  reaction, the dose is given by

$$\dot{D}_{np} = \frac{\phi N_N \sigma_N Q \times 1.6 \times 10^{-13} \text{ J/MeV}}{1 \text{ J/kg} \cdot \text{Gy}},$$

where  $\phi$  = thermal flux, neutrons per  $\text{cm}^2$  per second,  
 $N_N$  = number of nitrogen atoms per kg tissue,  $1.49 \times 10^{24}$ ,  
 $\sigma_N$  = absorption cross section for nitrogen,  $1.75 \times 10^{-24} \text{ cm}^2$ ,  
 $Q$  = energy released by the reaction = 0.63 MeV.

## Radiation Dose from Thermal Neutrons (Revisited)

- ☞ For the  ${}^1\text{H}(n, \gamma){}^2\text{H}$  reaction, the dose is deposited by the gamma rays emitted throughout the entire volume. The number of reaction per second per gram is governed by the neutron flux and is given by

$$A = \phi N_{\text{H}} \sigma_{\text{H}} \text{ "Bq"/kg,}$$

where  $\phi$  = thermal flux, neutrons per  $\text{cm}^2$  per second,  
 $N_{\text{H}}$  = number of hydrogen atoms per kg tissue =  $5.98 \times 10^{25}$ ,  
 $\sigma_{\text{H}}$  = absorption cross section for hydrogen =  $0.33 \times 10^{-24} \text{ cm}^2$ .

- ☞ The resulting gamma ray dose is illustrated with the following example.

### Example 6.17

What is the absorbed dose rate to a 70-kg person from a whole body exposure to a mean thermal flux of 10,000 neutrons per cm<sup>2</sup> per second?

The dose rate due to the n, p reaction is calculated from Eq. (6.105)

$$\begin{aligned}\dot{D}_{np} &= 1 \times 10^4 \times 1.49 \times 10^{24} \times 1.75 \times 10^{-24} \times 0.63 \times 1.6 \times 10^{-13} \\ &= 2.628 \times 10^{-9} \text{ Gy/s} \quad (2.628 \times 10^{-7} \text{ rad/s}),\end{aligned}$$

or

$$\dot{D}_{np} = 9.461 \mu\text{Gy/h} \quad (0.95 \text{ mrad/h}).$$

$$\dot{D}_{np} = \frac{\phi N_N \sigma_N Q \times 1.6 \times 10^{-13} \text{ J/MeV}}{1 \text{ J/kg} \cdot \text{Gy}},$$

where  $\phi$  = thermal flux, neutrons per cm<sup>2</sup> per second,  
 $N_N$  = number of nitrogen atoms per kg tissue,  $1.49 \times 10^{24}$ ,  
 $\sigma_N$  = absorption cross section for nitrogen,  $1.75 \times 10^{-24}$  cm<sup>2</sup>,  
 $Q$  = energy released by the reaction = 0.63 MeV.

The autointegral gamma-ray dose rate is calculated with Eq. (6.82). The gamma-ray “activity,” from Eq. (6.106) is

$$\begin{aligned}A &= 10^4 \text{ cm}^2 \text{ s}^{-1} \times 5.98 \times 10^{25} \text{ atoms/kg} \times 3.3 \times 10^{-25} \text{ cm}^2/\text{atom} \\ &= 1.973 \times 10^5 \text{ “Bq”/kg}.\end{aligned}$$

$$A = \phi N_H \sigma_H \text{ “Bq”/kg},$$

where  $\phi$  = thermal flux, neutrons per cm<sup>2</sup> per second,  
 $N_H$  = number of hydrogen atoms per kg tissue =  $5.98 \times 10^{25}$ ,  
 $\sigma_H$  = absorption cross section for hydrogen =  $0.33 \times 10^{-24}$  cm<sup>2</sup>.

The dose rate from this uniformly distributed gamma ray activity is calculated from Eq. (6.82):

$$\begin{aligned}\dot{D}_H &= A \cdot E_r \cdot \phi = 1.973 \times 10^5 \text{ Bq/kg} \cdot 2.23 \text{ MeV} \cdot 1.6 \times 10^{-16} \text{ J/MeV} \cdot 0.278 \\ &= 1.19 \times 10^{-11} \text{ Gy/sec} = 6.89 \times 10^{-2} \mu\text{Gy/h}\end{aligned}$$

The absorbed fraction,  $\phi$ , for the 2.23-MeV gamma ray is found, by interpolating in Table 6.8 between the 2.000- and 4.000-MeV values, to be 0.278

## Neutron Shielding – An Example

### *Example 10.11*

Design a shield for an  $18.5 \times 10^4$  MBq (5 Ci) Pu-Be neutron source that emits  $5 \times 10^6$  neutrons per second, such that the dose rate at the outside surface of the shield will not exceed  $15 \mu\text{Sv/h}$  (1.5 mrem/h). The mean energy of the neutrons produced in this source is 4 MeV.

## Radiation Dose from Fast Neutrons

### Example 6.16

What is the absorbed dose rate to soft tissue in a beam of 5-MeV neutrons whose intensity is 2000 neutrons per square centimeter per second?

Substituting the appropriate values into Eq. (6.103) yields

$$\begin{aligned}\dot{D}_n &= \frac{2 \times 10^3 \text{ n/cm}^2 \cdot \text{s} \times 5 \text{ MeV/n} \times 1.6 \times 10^{-13} \text{ J/MeV} \times 51.17 \text{ cm}^2/\text{kg}}{1 \text{ J/kg} \cdot \text{Gy}} \\ &= 8.19 \times 10^{-8} \text{ Gy/s} \quad (8.19 \times 10^{-6} \text{ rad/s}),\end{aligned}$$

$$\dot{D}_n(E) = \frac{\phi(E)E \sum_i N_i \sigma_i f}{1 \text{ J/kg} \cdot \text{Gy}}, \quad (6.103)$$

where  $\phi(E)$  = flux of neutrons whose energy is  $E$ , in neutrons/cm<sup>2</sup> · s,  
 $E$  = neutron energy, in joules,  
 $N_i$  = atoms per kilogram of the  $i$ th element,  
 $\sigma_i$  = scattering cross section of the  $i$ th element for neutrons of energy  $E$ , in barns  $\times 10^{-24}$  cm<sup>2</sup>,  
 $f$  = mean fractional energy transferred from neutron to scattered atom during collision with neutron.

**Example 6.17**

What is the absorbed dose rate to a 70-kg person from a whole body exposure to a mean thermal flux of 10,000 neutrons per cm<sup>2</sup> per second?

The dose rate due to the n, p reaction is calculated from Eq. (6.105)

$$\begin{aligned} \dot{D}_{np} &= 1 \times 10^4 \times 1.49 \times 10^{24} \times 1.75 \times 10^{-24} \times 0.63 \times 1.6 \times 10^{-13} \\ &= 2.628 \times 10^{-9} \text{ Gy/s} \quad (2.628 \times 10^{-7} \text{ rad/s}), \end{aligned}$$

or

$$\dot{D}_{np} = 9.461 \text{ } \mu\text{Gy/h} \quad (0.95 \text{ mrad/h}).$$

$$\dot{D}_{np} = \frac{\phi N_N \sigma_N Q \times 1.6 \times 10^{-13} \text{ J/MeV}}{1 \text{ J/kg} \cdot \text{Gy}},$$

where  $\phi$  = thermal flux, neutrons per cm<sup>2</sup> per second,  
 $N_N$  = number of nitrogen atoms per kg tissue,  $1.49 \times 10^{24}$ ,  
 $\sigma_N$  = absorption cross section for nitrogen,  $1.75 \times 10^{-24} \text{ cm}^2$ ,  
 $Q$  = energy released by the reaction = 0.63 MeV.

The autointegral gamma-ray dose rate is calculated with Eq. (6.82). The gamma-ray “activity,” from Eq. (6.106) is

$$\begin{aligned} A &= 10^4 \text{ cm}^2 \text{ s}^{-1} \times 5.98 \times 10^{25} \text{ atoms/kg} \times 3.3 \times 10^{-25} \text{ cm}^2/\text{atom} \\ &= 1.973 \times 10^5 \text{ “Bq”/kg}. \end{aligned}$$

$$A = \phi N_H \sigma_H \text{ “Bq”/kg},$$

where  $\phi$  = thermal flux, neutrons per cm<sup>2</sup> per second,  
 $N_H$  = number of hydrogen atoms per kg tissue =  $5.98 \times 10^{25}$ ,  
 $\sigma_H$  = absorption cross section for hydrogen =  $0.33 \times 10^{-24} \text{ cm}^2$ .

The dose rate from this uniformly distributed gamma ray activity is calculated from

$$\dot{D}_H = A \cdot E_r \cdot \varphi$$

The absorbed fraction,  $\varphi$ , for the 2.23-MeV gamma ray is found, by interpolating in Table 6.8 between the 2.000- and 4.000-MeV values, to be 0.278, and  $\Delta$ , the dose rate

## Radiation Dose from Fast Neutrons (Revisited)

- ☞ Neutron dose is deposited through scattering and neutron induced nuclear reactions.
- ☞ In cases of elastic scattering, the scattered nuclei dissipate their energy in the immediate vicinity of the primary neutron interaction. The radiation dose absorbed locally in this way is called the first collision dose. The scattered neutron is not considered after this primary interaction.
- ☞ For fast neutrons, the first collision dose rate is given by

$$\dot{D}_n(E) = \frac{\phi(E)E \sum_i N_i \sigma_i f}{1 \text{ J/kg} \cdot \text{Gy}},$$

- where
- $\phi(E)$  = flux of neutrons whose energy is  $E$ , in neutrons/cm<sup>2</sup> · s,
  - $E$  = neutron energy, in joules,
  - $N_i$  = atoms per kilogram of the  $i$ th element,
  - $\sigma_i$  = scattering cross section of the  $i$ th element for neutrons of energy  $E$ , in barns  $\times 10^{-24}$  cm<sup>2</sup>,
  - $f$  = mean fractional energy transferred from neutron to scattered atom during collision with neutron.

# Neutron Shielding – An Example

**TABLE 9.5. Values of Neutron Fluence Rates Which, in a Period of 40 H, Result in a Maximum Dose Equivalent of 1 mSv**

Neutron Energy, MeV	Neutron Fluence Rate, $\text{cm}^{-2}\text{s}^{-1}$	
	Adapted from NCRP Report No. 38 (NCRP. 1971) <sup>a</sup>	Adapted from Cross and Ing. 1985 <sup>a</sup>
$2.5 \times 10^{-8}$	270	280
$10^{-7}$	340	—
$10^{-6}$	280	280
$10^{-5}$	280	280
$10^{-4}$	290	290
$10^{-3}$	340	280
$5 \times 10^{-3}$	—	310
$10^{-2}$	350	300
$2 \times 10^{-2}$	—	250
$5 \times 10^{-2}$	—	110
$10^{-1}$	58	40
$3 \times 10^{-1}$	—	20
$3.8 \times 10^{-1}$	—	16
$4.4 \times 10^{-1}$	—	13
$5 \times 10^{-1}$	14	16
$6 \times 10^{-1}$	—	15
$8 \times 10^{-1}$	—	14
$9 \times 10^{-1}$	—	13
1.0	10	9.7
1.20	—	12
2.00	—	11
2.30	—	12
2.50	10	11
3.00	—	11
3.50	—	8.5
4.50	—	9.9
5.00	8.0	9.7
6.25	—	9.2
7.00	8.5	9.0
10.0	8.5	8.0
14.0	6.0	6.8
14.7	—	6.5
20	5.5	—
40	5.0	—
60	5.5	—
100	7.0	—
200	6.5	—
300	5.5	—
400	5.0	—

<sup>a</sup>The fluence rates presented here have been obtained from the cited references by dividing the respective reference values for thermal neutrons by 2.5 and the respective values for all other energies by 2.0. These adjustments have been made to reflect recommendations of the NCRP (1987) to increase the effective quality factors for thermal neutrons and more energetic neutrons by 2.5 and 2.0, respectively. SOURCE: From NCRP Report No. 112. By permission.

The 4MeV fast neutron flux that could introduce a dose equivalent rate of 1mSv/40h (25  $\mu\text{Sv/h}$ ).

We would need  $\sim 3.7\text{n}/(\text{s}\cdot\text{cm}^2)$  to deliver a dose equivalent of 10  $\mu\text{Sv/h}$  from fast neutrons.

## Step 1: Shielding for fast neutrons (continued)

Let's arbitrarily allow for a maximum dose from fast neutrons leaking from the shielding to be 10  $\mu\text{Sv/h}$ . We could use either Table 9.5 (given on the next page) or the equation for fast neutron dose to derive the fast neutron flux that leads to this dose rate, which turns out to be 3.7  $\text{n/s}\cdot\text{cm}^2$ .

Then **what is the thickness of the shielding needed to bring the fast neutron flux to this level?**

Considering a point source of fast neutrons, the neutron flux after passing through a thickness of  $nT$  (cm) could be approximately calculated as

$$\dot{\phi} = \frac{B \cdot S}{4\pi(nT)^2} \frac{1}{2^n} \left( \frac{\text{neutrons}}{\text{cm}^2 \cdot \text{s}} \right).$$

*B*: The build-up factor, 5 for this case.

*T*: Half-Valued-Layer (HVL), 3.71 cm for 4 MeV neutrons in water.

*S*: Source strength in neutrons/s,  $5 \times 10^6$  neutrons/s in this example.

## Step 1: Shielding for fast neutrons

Assuming that the shielding is made of water.

For the 4 MeV fast neutrons produced by the Pu-Be source, the cross sections of H and O atoms for elastic scattering are 1.9 barns and 1.7 barns, respectively.

So the linear scattering coefficient of water is given by

$$\begin{aligned}\Sigma &= 1.9 \times 10^{-24} \left( \frac{\text{cm}^2}{\text{atom}} \right) \times 6.7 \times 10^{22} \left( \frac{\text{atoms}}{\text{cm}^3} \right) + \\ &\quad 1.7 \times 10^{-24} \left( \frac{\text{cm}^2}{\text{atom}} \right) \times 3.35 \times 10^{22} \left( \frac{\text{atoms}}{\text{cm}^3} \right) \\ &= 0.186 \text{ cm}^{-1},\end{aligned}$$

which is corresponding to a **Half-Valued Layer (HVL)** of  $T=3.71$  cm.

## Step 1: Shielding for fast neutrons (continued)

Consider a point source of neutrons, the fast neutron flux after passing through a thickness of  $nT$  (cm) could be calculate approximately as

$$\dot{\phi} = \frac{B \cdot S}{4\pi(nT)^2} \frac{1}{2^n} \left( \frac{\text{neutrons}}{\text{cm}^2 \cdot \text{s}} \right).$$

Handwritten annotations in red:

- $3.7 \text{ n/s} \cdot \text{cm}^2$  points to  $\dot{\phi}$
- $5$  points to  $B$
- $5 \times 10^6 \text{ n/s}$  points to  $S$
- $3.71 \text{ cm}$  points to  $nT$

B: the build-up factor, 5 for this case.

T: Half-Valued-Layer, 3.71 cm.

S: Source strength in neutrons/s,  $5 \times 10^6$  neutrons/s.

Solving the above equation for  $n$  yields  $n=9$ .

Therefore, we would need  $9 \times 3.71$  cm of water to achieve a neutron flux of  $3.7 \text{ n/s/cm}^2$ , ensuring that the fast neutron dose stays below  $10 \mu \text{ Sv/h}$  at the surface of the shielding.

## Step 2: Shielding for thermal neutron radiation

Considering that the radius of the water-filled spherical shielding is 34 cm, which corresponds to 9 times the HVL. We could assume that most fast neutrons will be attenuated and thermalized close to the center.

We could therefore **assume the source-shielding volume as a point source of thermal neutrons located at the center of a spherical shielding volume with an approximate radius of 33 cm.**

The flux of the thermal neutrons emerging from the surface of the spherical shielding volume could be calculated as

$$\dot{\phi} = \frac{S}{4\pi RD} e^{-R/L} \left( \frac{\text{neutrons}}{\text{cm}^2 \cdot \text{s}} \right) = 0.55 \frac{n}{\text{s} \cdot \text{cm}^2} \Rightarrow \text{Dose negligible}$$

where

S: Strength of the thermal neutron source.  $\sim 5 \times 10^6$  neutrons/s.

R: Radius of the spherical shielding volume, 33 cm.

L: Thermal diffusion length, 2.88 cm.

D: Diffusion coefficient, 0.16cm.

## Fast- and Thermal-Diffusion Lengths

The **fast-diffusion length**: the average straight-line distance covered by fast neutrons traveling in a given medium.

The **thermal-diffusion length**: the average distance covered by thermalized neutrons before it is absorbed. It is measured by the thickness of a slowing-down medium that attenuates the beam of thermal neutrons by a factor of  $e$ . Thus, the attenuation of a beam of thermal neutrons by a substance of thickness  $t$  (cm), whose thermal diffusion length is  $L$  (cm) is given by

$$n = n_0 e^{-t/L}$$

**TABLE 5.6. Fast and Thermal Diffusion Lengths of Selected Materials**

Substance	Fast Diffusion Length, cm	Thermal Diffusion Length, cm	Thermal Diffusion Coefficient, cm
H <sub>2</sub> O	5.75	2.88	0.16
D <sub>2</sub> O	11	171	0.87
Be	9.9	24	0.50
C (graphite)	17.3	50	0.84

## Radiation Dose from Thermal Neutrons (Revisited)

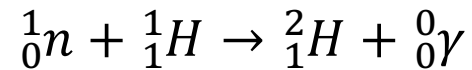
☞ Two reactions are normally considered, namely  $^{14}\text{N}(n,p)^{14}\text{C}$  and  $^1\text{H}(n, \gamma)^2\text{H}$  reactions.

☞ For the  $^{14}\text{N}(n,p)^{14}\text{C}$  reaction, the dose is given by

$$\dot{D}_{np} = \frac{\phi N_N \sigma_N Q \times 1.6 \times 10^{-13} \text{ J/MeV}}{1 \text{ J/kg} \cdot \text{Gy}},$$

where  $\phi$  = thermal flux, neutrons per  $\text{cm}^2$  per second,  
 $N_N$  = number of nitrogen atoms per kg tissue,  $1.49 \times 10^{24}$ ,  
 $\sigma_N$  = absorption cross section for nitrogen,  $1.75 \times 10^{-24} \text{ cm}^2$ ,  
 $Q$  = energy released by the reaction = 0.63 MeV.

## Neutron Induced Reactions (Revisited)



- ☞ Neutron absorption is followed by the immediate emission of a gamma ray photon.
- ☞ The gamma photon has the energy  $Q = 2.26$  MeV released by the reaction, which represents the binding energy of the deuteron.
- ☞ The capture cross-section per atom is 0.33 barn.
- ☞ When tissue is exposed to thermal neutrons, this reaction provides a source of gamma rays that delivers a finite dose to the tissue.

## Radiation Dose from Thermal Neutrons (Revisited)

- ☞ For the  ${}^1\text{H}(n, \gamma){}^2\text{H}$  reaction, the dose is deposited by the gamma rays emitted throughout the entire volume. The number of reaction per second per gram is governed by the neutron flux and is given by

$$A = \phi N_{\text{H}} \sigma_{\text{H}} \text{ "Bq"/kg,}$$

where  $\phi$  = thermal flux, neutrons per  $\text{cm}^2$  per second,  
 $N_{\text{H}}$  = number of hydrogen atoms per kg tissue =  $5.98 \times 10^{25}$ ,  
 $\sigma_{\text{H}}$  = absorption cross section for hydrogen =  $0.33 \times 10^{-24} \text{ cm}^2$ .

- ☞ The resulting gamma ray dose is illustrated with the following example.

### Step 3: Shielding for neutron-capture gamma rays

Consider that

1. the energy of each gamma-ray is 2.26 MeV, and
2. with the spherical water volume of 34 cm radius, there will be  $3.7 \text{ n/s/cm}^2$ , or  $5 \times 10^4 \text{ n/s}$  escaping from the surface, and
3. all fast neutrons that did not escape will eventually be absorbed by hydrogen atoms, giving rise to the 2.26 MeV photons,

then the apparent gamma-ray activity could be assumed to be uniformly distributed across the spherical volume, then the gamma-ray activity is

$$A = \frac{5 \times 10^6 (n/s) - 5.4 \times 10^4 (n/s)}{\frac{4}{3} \pi \cdot (34 \text{ cm})^3} = 30 \text{ (Bq/cm}^3\text{)}$$

## Internally Deposited Radioisotope (IV) Gamma Ray Emitters

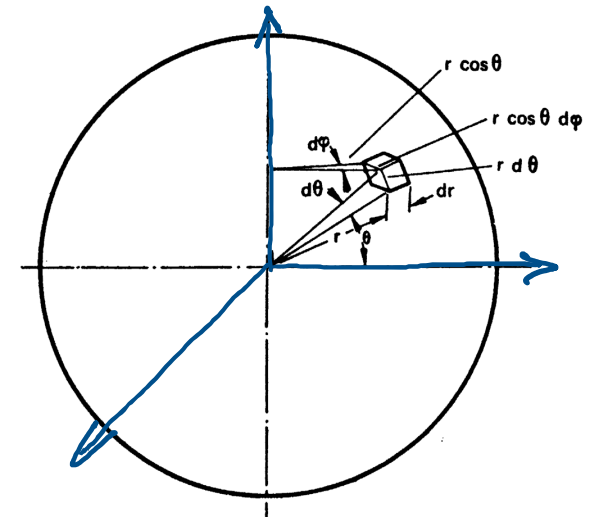
☞ For a uniform spherical source, the **dose rate at the center** is given by

$$\dot{D} = 4C\Gamma \int_{r=0}^{r=R} \int_{\theta=0}^{\theta=\pi/2} \int_{\varphi=0}^{\varphi=\pi} \frac{e^{-\mu r}}{r^2} \cdot r \, d\theta \cdot r \cos\theta \, d\varphi \cdot dr \cdot \left(34 \frac{J/kg}{\text{Coulomb/kg}}\right) \cdot \left(\frac{\mu_m/\rho_m}{\mu_a/\rho_a}\right)$$

$$= C\Gamma \cdot \frac{4\pi}{\mu} (1 - e^{-\mu R}) \cdot \left(34 \frac{J/kg}{\text{Coulomb/kg}}\right) \cdot \left(\frac{\mu_m/\rho_m}{\mu_a/\rho_a}\right)$$

☞ And the **dose rate at the surface** of the spherical source volume is given by

$$\dot{D}_{\text{surface}} = 0.5 \dot{D}_{\text{center}}$$



$$dV = r d\theta \cdot r \cos\theta d\varphi \cdot dr = r^2 \cos\theta d\varphi d\theta dr$$

## Step 3: Shielding for neutron-capture gamma rays (continued)

Remember that for a spherical volume filled with uniform radioactivity of A (Bq/cm<sup>3</sup>), the dose rate at the surface of the sphere is given by

$$\dot{D} = \frac{1}{2} \cdot C \cdot \Gamma \cdot \frac{4\pi}{\mu} \cdot (1 - e^{-\mu r}) \cdot \left(34 \frac{J/kg}{\text{Coulomb/kg}}\right) \cdot \underbrace{\left(\frac{\mu_{\text{water}}/\rho_{\text{water}}}{\mu_{\text{air}}/\rho_{\text{air}}}\right)}_{1.1}$$

Specific gamma-ray constant for 2.2 MeV gamma-rays:

$$7.2 \times 10^{-5} \left( \frac{C}{kg} \cdot cm^2 \cdot MBq^{-1} \cdot h^{-1} \right)$$

Radius of the shielding, 34 cm

Linear att. coef. for 2.2 MeV gamma-rays in water, 0.046 (cm<sup>-1</sup>)

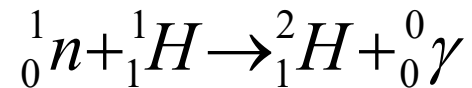
1.1

$$\dot{D} = 9 \times 10^{-3} (m Gy/h)$$

## Step 3: Shielding for neutron-capture gamma rays (continued)

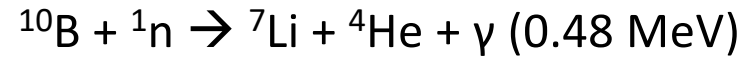
- ☞ A spherical shielding of 34 cm radius filled with water would lead to a fast neutron dose of  $10 \mu\text{Gy/h}$ , a negligible thermal neutron dose, and gamma-ray dose of  $9 \mu\text{Gy/h}$ .
- ☞ The total is  $19 \mu\text{Gy/h}$ , which is still greater than the  $15 \mu\text{Gy/h}$  target. What should we do now?

## Neutron Induced Reactions



- ☞ Neutron absorption is followed by the immediate emission of a gamma ray photon.
- ☞ Since the thermal neutron has negligible energy by comparison, the gamma photon has the energy  $Q=2.22\text{MeV}$  released by the reaction, which represents the binding energy of the deuteron.
- ☞ The capture cross-section per atom is 0.33 barn.
- ☞ When tissue is exposed to thermal neutrons, this reaction provides a source of gamma rays that delivers a finite dose to the tissue.

# Thermal Neutron Capture by Boron



- ☞ Capture cross section: 755 barns.
- ☞ The 0.48 MeV gamma ray is emitted in 93% of the capture.

## Step 4: Improve the shielding efficiency by adding boric acid ( $\text{H}_3\text{BO}_3$ ) in water

- ☞ Consider that we can **add boric acid to water**, whose molecular weight is 61.84, solubility is 63 g/L, and thermal absorption coefficient is 775 barns.
- ☞ If we add the maximum soluble concentration of boric acid in water, then the **concentration of boron atoms** in water is

$$\frac{63.2(\text{g/L}) \times 10^{-3}(\text{L/mL}) \cdot 6.02 \times 10^{23}(\text{molecules/mole})}{61.84(\text{g/mole})} = 6.17 \times 10^{20} \text{ (atoms/mL)}$$

- ☞ Compute the **ratio of linear thermal absorption coefficients due to boron and hydrogen atoms**,

$$\frac{\Sigma_H}{\Sigma_B} = \frac{1.9 \times 10^{-24}(\text{cm}^2/\text{atom}) \cdot 6.7 \times 10^{22}(\text{atoms}/\text{cm}^3)}{775 \times 10^{-24}(\text{cm}^2/\text{atom}) \cdot 6.17 \times 10^{20}(\text{atoms}/\text{cm}^3)} = 0.31$$

- ☞ Therefore, **for every 1 thermal neutron captured by a hydrogen atom**, there will be about **3.23 thermal neutrons each captured by boron atoms**.

## Step 4: Improve the shielding efficiency by adding boric acid ( $\text{H}_3\text{BO}_3$ ) in water

After adding  $\text{H}_3\text{BO}_3$  in water, the gamma-ray dose due to the 2.2 MeV gamma-rays from **thermal neutron capture by hydrogen atoms** in the water tank can be derived as follows:

- ☞ First, for every single thermal neutron captured by a hydrogen atom, there will be about 3.23 thermal neutrons captured by a boron atom.
- ☞ Second, if we assume that all thermal neutrons are captured by either hydrogen or boron atoms, then there will be  $[5 \times 10^6 (n/s) - 5.4 \times 10^4 (n/s)] \times \frac{1}{1+3.23}$  thermal neutrons being **captured by hydrogen atoms** per second.
- ☞ Therefore, the gamma ray dose due to thermal neutrons captured by hydrogen will be reduced by a factor of  $\frac{1}{1+3.23}$  to 2.1  $\mu\text{Gy/h}$ . (down from 9  $\mu\text{Gy/h}$  previously with pure water).

### Step 4: Improve the shielding efficiency by adding boric acid ( $\text{H}_3\text{BO}_3$ ) in water

Gamma-ray dose due to gamma rays from **thermal neutron capture by boron atoms** in the water tank can be derived as the following:

- ☞ First, for every thermal neutron captured by a hydrogen atom, there will be about 3.23 thermal neutrons each captured by a boron atom.
- ☞ Second, if we assume that all thermal neutrons are captured by either hydrogen or boron atoms, then there will be  $[5 \times 10^6 (n/s) - 5.4 \times 10^4 (n/s)] \times \frac{3.23}{1+3.23}$  thermal neutrons being captured by boron atoms per second, leading to an apparent gamma-ray activity of

$$\frac{[5 \times 10^6 (n/s) - 5.4 \times 10^4 (n/s)] \times \frac{3.23}{1+3.23}}{\frac{4}{3} \cdot \pi \cdot (34 \text{ cm})^3} = 22 \text{ (Bq/cm}^3\text{)}$$

## Step 4: Improve the shielding efficiency by adding boric acid ( $H_3BO_3$ ) in water

Therefore, the gamma-ray dose due to **thermal neutron capture by boron atoms** in the water tank can be derived as follows:

$$\dot{D} = \frac{1}{2} \cdot C \cdot \Gamma \cdot \frac{4\pi}{\mu} \cdot (1 - e^{-\mu r}) \cdot \left(34 \frac{J/kg}{Coulomb/kg}\right) \cdot \underbrace{\left(\frac{\mu_{water}/\rho_{water}}{\mu_{air}/\rho_{air}}\right)}_{1.1}$$

$22(Bq \cdot cm^{-3})$                       Radius of the shielding, 34 cm

Specific gamma-ray constant for 0.48 MeV gamma-rays:

$$1.92 \times 10^{-5} \left( \frac{C}{kg} \cdot cm^2 \cdot MBq^{-1} \cdot h^{-1} \right)$$

Linear att. coef. for 0.48 MeV gamma-rays in water,  
 $0.09 (cm^{-1})$

$$\dot{D} = 1 \times 10^{-3} (m Gy/h)$$

## Step 4: Improve the shielding efficiency by adding boric acid ( $\text{H}_3\text{BO}_3$ ) in water

- ☞ A spherical shielding of 34 cm radius filled with water would lead to a fast neutron dose of  $10 \mu\text{Gy/h}$ , a negligible thermal neutron dose, a gamma ray dose of  $2.1 \mu\text{Gy/h}$  from thermal neutron capture by hydrogen, and a gamma ray dose of  $1 \mu\text{Gy/h}$  from thermal neutron capture by boron.
- ☞ The total is  $13.1 \mu\text{Gy/h}$ , which is smaller than the  $15 \mu\text{Gy/h}$  target.