Neutron Shielding Design

Considerations for Neutron Shielding Design

- ☞ Radiological assessment of radiation hazards associated with neutron sources is normally a complex task.
  - ☞ Involves both **neutron** radiation and **secondary gamma-ray radiation**.
  - ☞ Few problems can be solved with elementary techniques.

- ☞ General treatment involves considerations of the dose delivered by the **elastic scattering of fast neutrons** and the **thermal neutron capture gamma-ray radiations**.
- ☞ **Neutron removal cross-section**: Under special circumstances, the fast neutron dose after the shielding can be derived with an exponential attenuation factor.

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

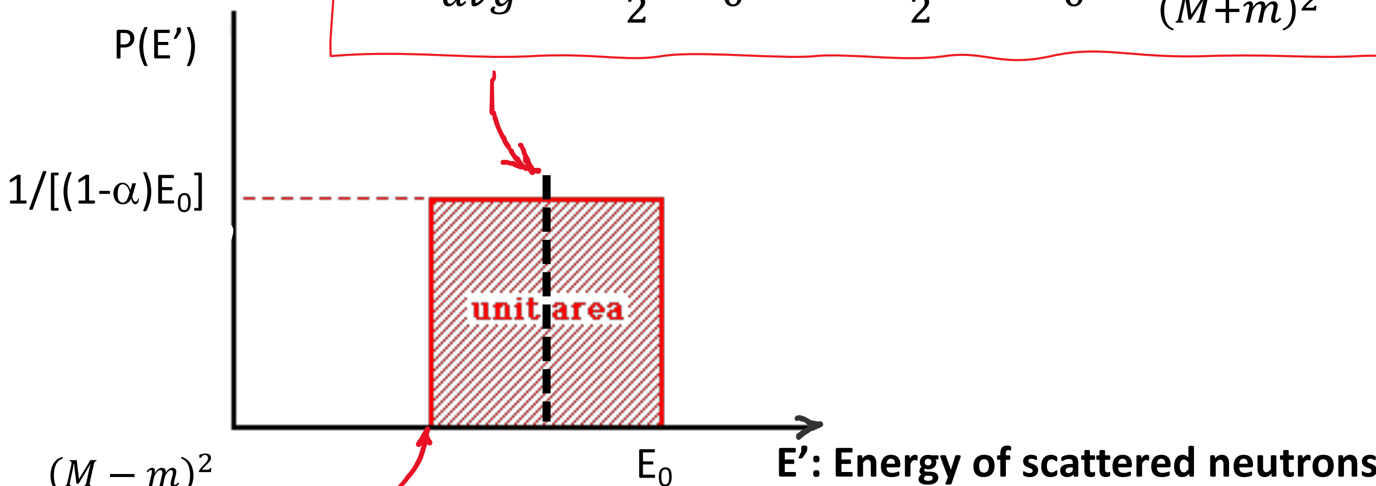
$$\dot{D}_n(E) = \frac{\phi(E)E\sum_i N_i\sigma_i f}{1\ \text{J/kg}\cdot\text{Gy}},$$

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

Spectrum of Energy-Loss by Neutron Scattering (Revisited)

Average energy carried by the scattered neutron:

$$E'_{avg} = \frac{1+\alpha}{2}E_0 = \frac{1+\frac{(M-m)^2}{(M+m)^2}}{2}E_0 = \frac{M^2+m^2}{(M+m)^2}\cdot E_0$$



$$\frac{(M-m)^2}{(M+m)^2}E_0 \equiv \alpha E_0$$

Average energy transferred to the recoil nucleus:

$$E_{avg\_energy\_loss} = E_0 - E'_{avg} = \frac{2Mm}{(M+m)^2}\cdot E_0$$

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

$$\dot{D}_n = \frac{2 \times 10^3\ n/\text{cm}^2 \cdot \text{s} \times 5\ \text{MeV}/n \times 1.6 \times 10^{-13}\ \text{J/MeV} \times 51.17\ \text{cm}^2/\text{kg}}{1\ \text{J/kg} \cdot \text{Gy}}$$

$$= 8.19 \times 10^{-8}\ \text{Gy/s}\left(8.19 \times 10^{-6}\ \text{rad/s}\right),$$

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

where $\phi(E)$ = flux of neutrons whose energy is E, in neutrons/cm^2. s,
E = neutron energy, in joules,
N_i = atoms per kilogram of the ith element,
σ_i = scattering across section of the ith element for neutrons of energy E, in barns $\times 10^{-24}$ cm^2,
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 $^{14}N(n,p)^{14}C$ and $^{1}H(n,r)^{2}H$ reactions.

☞ For the $^{14}N(n,p)^{14}C$ 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 ϕ = thermal flux, neutrons per cm^2 per second,
N_N = number of nitrogen atoms per kg tissue, 1.49×10^{24},
σ_N = absorption cross section for nitrogen, 1.75×10^{-24} cm^2,
Q = energy released by the reaction = 0.63 MeV.

Radiation Dose from Thermal Neutrons (Revisited)

☞ For the $^{1}H(n, \gamma)^{2}H$ 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

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

where ϕ = thermal flux, neutrons per cm^2 per second,
N_H = number of hydrogen atoms per kg tissue = 5.98×10^{25},
σ_H = absorption cross section for hydrogen = 0.33×10^{-24} 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 cm² per second?

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

$$\dot{D}_{np} = 1 \times 10^4 \times 1.49 \times 10^{24} \times 1.75 \times 10^{-24} \times 0.63 \times 1.6 \times 10^{-13}$$

$$= 2.628 \times 10^{-9}\ \text{Gy/s} \quad (2.628 \times 10^{-7}\ \text{rad/s}),$$

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

where ϕ = thermal flux, neutrons per cm² per second,
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σ_N = absorption cross section for nitrogen, 1.75×10^{-24} cm²,
Q = energy released by the reaction = 0.63 MeV.

or

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

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

$$A = 10^4\ \text{cm}^2\ \text{s}^{-1} \times 5.98 \times 10^{25}\ \text{atoms/kg} \times 3.3 \times 10^{-25}\ \text{cm}^2/\text{atom}$$

$$= 1.973 \times 10^5\ \text{"Bq"/kg}.$$

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

where ϕ = thermal flux, neutrons per cm² per second,
N_H = number of hydrogen atoms per kg tissue = 5.98×10^{25},
σ_H = absorption cross section for hydrogen = 0.33×10^{-24} cm².

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

$$\dot{D}_H = A \cdot E_r \cdot \varphi = 1.973\times10^5\, Bq/kg \cdot 2.23\text{MeV} \cdot 1.6\times10^{-16}\, J/MeV \cdot 0.278$$
$$= 1.19\times10^{-11}\ Gy/sec = 6.89\times10^{-2}\ \mu Gy/\text{h}$$

The absorbed fraction, φ, 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 × 10^4 MBq (5 Ci) Pu-Be neutron source that emits 5 × 10^6 neutrons per second, such that the dose rate at the outside surface of the shield will not exceed 15 μSv/h (1.5 mrems/h). The mean energy of the neutrons produced in this source is 4 MeV.

☞ Cember, P452.

Radiation Dose from Fast Neutrons

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$$= 8.19 \times 10^{-8}\ \text{Gy/s}\left(8.19 \times 10^{-6}\ \text{rad/s}\right),$$

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where $\phi(E)$ = flux of neutrons whose energy is E, in neutrons/cm^2. s,
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$$= 2.628 \times 10^{-9}\ \text{Gy/s} \quad (2.628 \times 10^{-7}\ \text{rad/s}),$$

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

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

or

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$$= 1.973 \times 10^5\ \text{"Bq"/kg}.$$

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

where ϕ = thermal flux, neutrons per cm^2 per second,
N_H = number of hydrogen atoms per kg tissue = 5.98×10^{25},
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The dose rate from this uniformly distributed gamma ray activity is calculated from

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The absorbed fraction, φ, 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 Δ, 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.
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E = neutron energy, in joules,
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σ_i = scattering across section of the ith element for neutrons of energy E, in barns $\times 10^{-24}$ cm^2,
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, $cm^{-2}s^{-1}$	
	Adapted from NCRP Report No. 38 (NCRP. 1971)[a]	Adapted from Cross and Ing. 1985[a]
2.5×10^{-8}	270	280
10^{-7}	340	—
10^{-6}	280	280
10^{-5}	280	280
10^{-4}	290	290
10^{-3}	340	280
5×10^{-3}	—	310
10^{-2}	350	300
2×10^{-2}	—	250
5×10^{-2}	—	110
10^{-1}	58	40
3×10^{-1}	—	20
3.8×10^{-1}	—	16
4.4×10^{-1}	—	13
5×10^{-1}	14	16
6×10^{-1}	—	15
8×10^{-1}	—	14
9×10^{-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	

[a]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 μSv/h).

We would need ~ 3.7n/(s·cm²) to deliver a dose equivalent of 10 μSv/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 μSv/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 n/s·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

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

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, 5×10^6 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

.

$$\Sigma = 1.9\times10^{-24}\left(\frac{\text{cm}^2}{atom}\right)\times6.7\times10^{22}\left(\frac{atoms}{cm^3}\right)+$$
$$1.7\times10^{-24}\left(\frac{\text{cm}^2}{atom}\right)\times3.35\times10^{22}\left(\frac{atoms}{cm^3}\right)$$
$$=0.186\ \text{cm}^{-1},$$

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

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

$3.7\ n/s\cdot cm^2$; 5; $5\times10^6\ n/s$; $3.71\ cm$

B: the build-up factor, 5 for this case.
T: Half-Valued-Layer, 3.71 cm.
S: Source strength in neutrons/s, 5×10^{-6} neutrons/s.

Solving the above equation for n yields n=9.

Therefore, we would need 9 × 3.71 cm of water to achieve a neutron flux of 3.7 n/s/cm^2, ensuring that the fast neutron dose stays below 10 μ 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

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

where
S: Strength of the thermal neutron source. $\sim 5\times10^6$ 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 Lenghts of Selected Materials

Substance	Fast Diffusion Length, cm	Thermal Diffusion Length, cm	**Thermal Diffusion Coefficient, cm**
H_2O	5.75	2.88	**0.16**
D_2O	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 $^{14}N(n,p)^{14}C$ and $^{1}H(n, \gamma)^{2}H$ reactions.

- For the $^{14}N(n,p)^{14}C$ 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 ϕ = thermal flux, neutrons per cm^2 per second,
N_N = number of nitrogen atoms per kg tissue, 1.49×10^{24},
σ_N = absorption cross section for nitrogen, 1.75×10^{-24} cm^2,
Q = energy released by the reaction = 0.63 MeV.

Neutron Induced Reactions (Revisited)

$$^{1}_{0}n + ^{1}_{1}H \rightarrow ^{2}_{1}H + ^{0}_{0}\gamma$$

- ☞ 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 $^{1}H(n, \gamma)^{2}H$ 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

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

where ϕ = thermal flux, neutrons per cm^2 per second,
N_H = number of hydrogen atoms per kg tissue = 5.98×10^{25},
σ_H = absorption cross section for hydrogen = 0.33×10^{-24} 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 n/s/cm^2, or $5\times10^4\, {}^{n}/_{s}$ 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

$$A = \frac{5\times10^6({}^{n}/_{s})-5.4\times10^4({}^{n}/_{s})}{\frac{4}{3}\pi\cdot(34cm)^3}=30\ (Bq/cm^3)$$

Internally Deposited Radioisotope (IV)
Gamma Ray Emitters

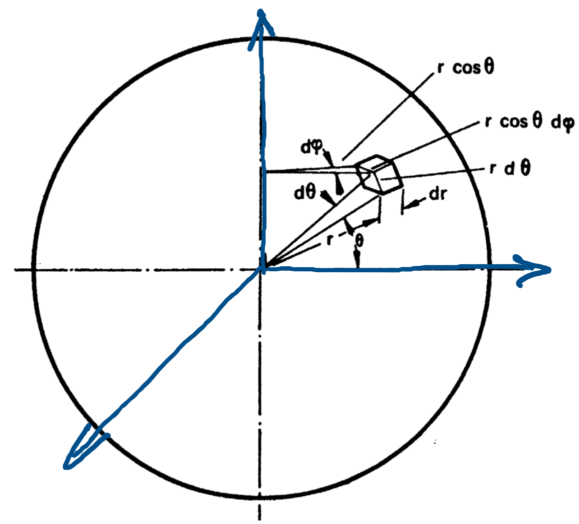
☞ For a uniform spherical source, the dose rate at the center is given by

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

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

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

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



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

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

Remember that for a spherical volume filled with uniform radioactivity of A (Bq/cm³), the dose rate at the surface of the sphere is given by

$30(Bq \cdot cm^{-3})$ (C)

Radius of the shielding, 34 cm (r)

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

Specific gamma-ray constant for 2.2 MeV gamma-rays (Γ): $7.2\times10^{-5}\left(\frac{C}{kg} \cdot cm^2 \cdot MBq^{-1} \cdot h^{-1}\right)$

Linear att. coef. for 2.2MeV gamma-rays in water (μ), $0.046(cm^{-1})$

$\frac{\mu_{water}/\rho_{water}}{\mu_{air}/\rho_{air}}$: 1.1

$$\dot{D} = 9\times10^{-3}(\,m\,Gy/h)$$

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 μGy/h, a negligible thermal neutron dose, and gamma-ray dose of 9 μGy/h.
- The total is 19 μGy/h, which is still greater than the 15 μGy/h target. What should we do now?

Neutron Induced Reactions

$$ {}_{0}^{1}n + {}_{1}^{1}H \rightarrow {}_{1}^{2}H + {}_{0}^{0}\gamma $$

- 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=2.22MeV 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

$$^{10}B + {}^{1}n \rightarrow {}^{7}Li + {}^{4}He + \gamma\ (0.48\ \text{MeV})$$

- ☞ 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 ($\mathbf{H_3BO_3}$) 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

$$\frac{63.2(g/L)\times10^{-3}(L/mL)\cdot 6.02\times10^{23}(molecules/mole)}{61.84(g/mole)}$$

$=6.17\times10^{20}$ (atoms/mL)

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

$$\frac{\Sigma_H}{\Sigma_B}=\frac{1.9\times10^{-24}(cm^2/_{atom})\cdot 6.7\times10^{22}(^{atoms}/_{cm^3})}{775\times10^{-24}(cm^2/_{atom})\cdot 6.17\times10^{20}(^{atoms}/_{cm^3})}=0.31$$

☞ 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 ($\mathbf{H_3BO_3}$) in water

After adding H_3BO_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 $[5\times10^6(^n/_s) - 5.4\times10^4(^n/_s)]\times\frac{1}{1+3.23}$ 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 μGy/h. (down from 9 μ Gy/h previously with pure water).

Step 4: Improve the shielding efficiency by adding boric acid (H_3BO_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 $[5\times10^6(^n/_s) - 5.4\times10^4(^n/_s)]\times\frac{3.23}{1+3.23}$ thermal neutrons being captured by boron atoms per second, leading to an apparent gamma-ray activity of

$$\frac{[5\times10^6(^n/_s)-5.4\times10^4(^n/_s)]\times\frac{3.23}{1+3.23}}{\frac{4}{3}\cdot\pi\cdot(34cm)^3} = 22\ \left(^{Bq}/_{cm^3}\right)$$

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:

$22 (Bq \cdot cm^{-3})$

Radius of the shielding, 34 cm

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

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

$1.92\times10^{-5} \left(\frac{C}{kg} \cdot cm^2 \cdot MBq^{-1} \cdot h^{-1}\right)$

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

1.1

$$\dot{D} = 1\times10^{-3} (m\,Gy/h)$$

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

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