Lecture 2
Macroscopic Interactions

22.106 Neutron Interactions and Applications

Spring 2010

Objectives

- Macroscopic Interactions
- Atom Density
- Mean Free Path
- Moderation in Bulk Matter
- Neutron Shielding
- Effective Dose Equivalent

Macroscopic interactions

- Suppose a target of thickness X is placed in a monodirectional beam of intensity I_0 and that a neutron detector is placed at some distance behind the target. It is also assumed that the target and the detector are small so that the detector only detects the neutrons that do not interact.

- Let $I(x)$ be the intensity of the non interacted neutrons after penetrating the distance x into the target.
- In traversing an additional distance dx, the intensity of the beam will be decreased by the number of neutron that have interacted in dx. This decrease in intensity will be given by:

$$- dI(x) = N\sigma_t I(x)dx$$

where N is the atom density in the target (at/cc)

- Integrating the previous equation from 0 to *x*, we get

$$I(x) = I_0 \, exp(-N\sigma_t x)$$

- We define the macroscopic x.s. has

$$\Sigma = N\sigma$$

units of Σ are cm^{-1}

- Thus

$$-dI(x) = \Sigma_t I(x)dx \quad or \quad -dI(x)/I(x) = \Sigma_t dx$$

- $-dI(x)/I(x)$ in the previous equation is equal to the fraction of the neutrons that have penetrated the distance x into the target without interacting, which subsequently interact in the distance *dx*
  - Equivalent to the probability that a neutron which survives up to *x* and interacts in the next *dx*
  - Thus $\Sigma_t dx$ is the <u>probability that a neutron interacts in *dx*</u>
  - It follows that Σ_t is the probability per unit path length that a neutron will undergo an interaction
- $I(x)/I_0 = exp(-\Sigma_t x)$ is equal to the probability that a neutron can move through this distance without interacting.

- Let the quantity p(x)dx be the probability that a neutron will have its first interaction in *dx* in the neighborhood of *x*.

$$p(x)dx = exp(-\Sigma_t x)\ .\Sigma_t dx$$

First interaction probability distribution function

- Examples
  - Calculate first interaction probability between $x = a$ and $x = b$
  - Calculate probability of not interacting in slab

- Macroscopic interactions are the simplest way to evaluate gamma ray attenuation
  - Low energy gamma’s are more likely to be absorbed
  - Neutrons on the other hand are more likely to scatter, which interferes with the measurements.
- If the target is a compound, the total macroscopic x.s. is the sum of the individual elements macroscopic x.s.

$$\Sigma_t = \Sigma_1 + \Sigma_2 + \ldots$$

Atom Density

- The atom density of each element *i* is given by:

$$N_i = \rho \, N_a \, n_i \, / \, M_m$$

where ρ = density of compound (g/cc)

M_m = molecular mass (g/mol)

N_a = Avogadro number

n_i = number of atoms of element *i* in one molecule

- $\Sigma = \rho N_a (n_1\sigma_1 + n_2\sigma_2 + \ldots) / M_m$
- Example of Natural Uranium

Mean Free Path

- The distance that a neutron moves between interactions is called a free path, and the average distance is known as the mean free path.

- Integral of *xp(x)dx* between *0* and *infinity*

Significance

- If the mean free path of neutrons emitted by a sample in a passive assay instrument is long compared to the dimensions of the sample, it is likely that most neutrons will escape the sample and enter the detection region.
- If we know the number of collisions to thermalize, we can estimate the needed moderator thickness (must also consider the scattering angle)
- If the thickness of a shield is many times the mean free path of a neutron trying to penetrate the shield, then the shield fulfills its purpose (complicated by the energy dependance of the x.s.)

Objectives

- Neutron Moderation
- Shielding
- Dose

Moderation in Bulk Matter

- Moderation is often times needed for
  - Neutron detection
  - Thermal reactors
  - Neutron diffraction
  - Small angle neutron scattering (SANS)
- Purpose:
  - Increase the probability of interaction in “1/v” region
  - Or, produce neutrons with low energy for condensed matter studies

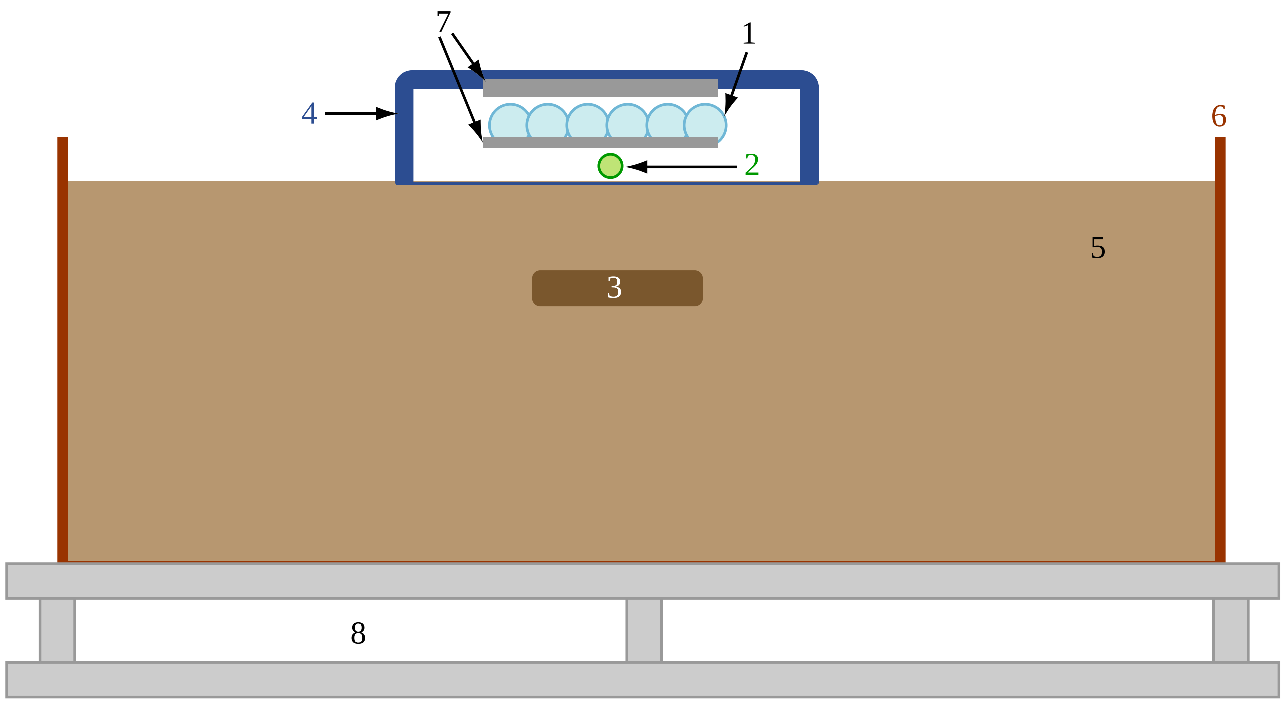
Slowing Down

- Neutrons are slowed down by collisions in the manner of a random walk
- Since fast neutrons have much more energy than the molecules they are colliding with, the neutrons will lose energy
- Extent of slowing down depends on
  - Temperature of medium
  - Size, shape and nature of moderation medium

Mine Detection

- Neutron sources and neutron moderation are used to detect land mines
- Cf252 source
  - Spontaneous fission with spectrum in slides of Lecture 1
- Explosives
  - Organic compounds with lots of Carbon and Hydrogen

Mine Detection



Experimental arrangement. Detector consisting of six 14-inch, 4-atmosphere ^{3}He tubes (1) held between two Al plates (7). Time-tagged neutron source (2) is located directly below detector. Simulated mine (3) is buried in sand (5); sand is held in 4-foot stock tank (6) supported on a wooden pallet (8). Cd sheet (4) forms thermal neutron shield. For wet-sand tests, a stainless steel pan is buried in sand and filled with dry sand. Water is added to give the desired water content.

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Mine Detection

- Fast neutrons enter soil
- If there is no explosive they scatter in sand, soil, …
  - Not really good moderators
- If a plastic explosive is found
  - Neutrons scatter off hydrogen and carbon
  - Lower energy neutrons are reflected and detected on the surface
- Potential issues
  - Wet soil, requires very fine calibration

Moderating Power

- Compares ability of materials to moderate neutrons
- Two important factors to consider
  - Probability of scattering interaction
  - Average change in kinetic energy of the neutron after scattering
- Define as

$\xi\Sigma_s$ ξ: average logarithmic energy decrement

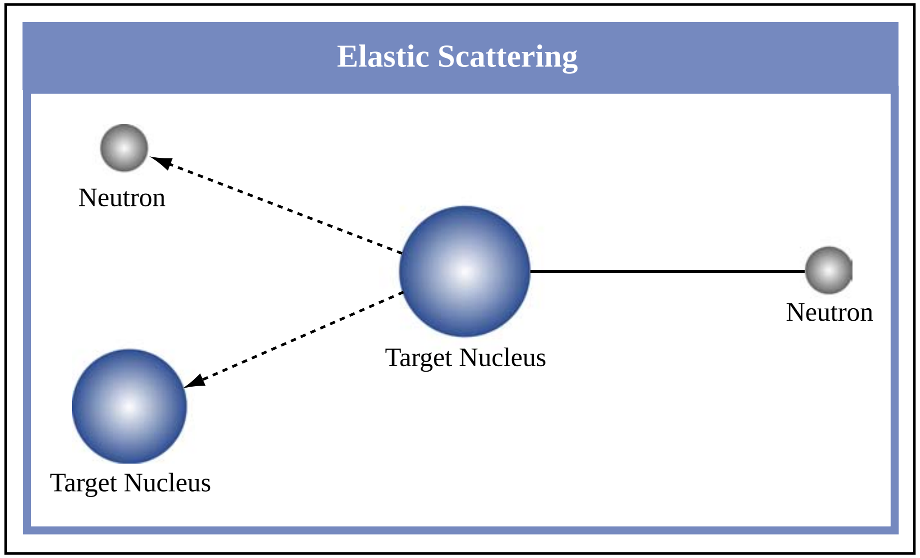
Σ_s: scattering macroscopic x.s.

Average Log Energy Decrement

- $\xi = \ln (E_0/E_1)$
- Assuming isotropic elastic scattering only, we can find the following expression

$$\xi = 1 - \frac{(A-1)^2}{2A} \ln \frac{A+1}{A-1}$$

$$\xi = 1 + \frac{\alpha}{1-\alpha} \ln \alpha, \text{ where } \alpha = \left(\frac{A-1}{A+1}\right)^2$$



Number of collisions

- If we lose ξ at every collision, we can approximate the average number of collisions it takes to go from E_0 to E_n

$$\overline{\ln E_n} = \ln E_0 - n\xi.$$

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which yields

$$n = \frac{1}{\xi} \ln \frac{E_0}{E_n^1}.$$

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Derivation: See PDF on Neutron Moderation on stellar

Moderating Ratio

- A material with a large moderating power might still not be practical if it has a large absorption x.s.
- Moderating ratio helps in selecting a good moderator: $\xi\Sigma_s / \Sigma_a$

	Moderating Power	Moderating Ratio
Water	1.28	58
Heavy Water	0.18	21000
He @ STP	0.00001	45
Be	0.16	130
Graphite	0.064	200
Polyethylene	3.26	122

Shielding Objectives

- Limit radiation exposure of staff, patients, visitors and the public to acceptable levels
- Optimize protection of patients, staff and the public
- Protect electronics (i.e. computer chips, transistors, …) in satellites or other instruments
- Reduce background counts in detectors

Considerations

- Source characterization
  - Particle type
  - Energy
  - Rate of emission
  - Direction of the source
  - Utilization factor
- Geometry of the room
  - Shielding is a 3-D problem
- Location
  - 10^{th} floor vs basement
- Requirements
  - Size, cost, weight, …

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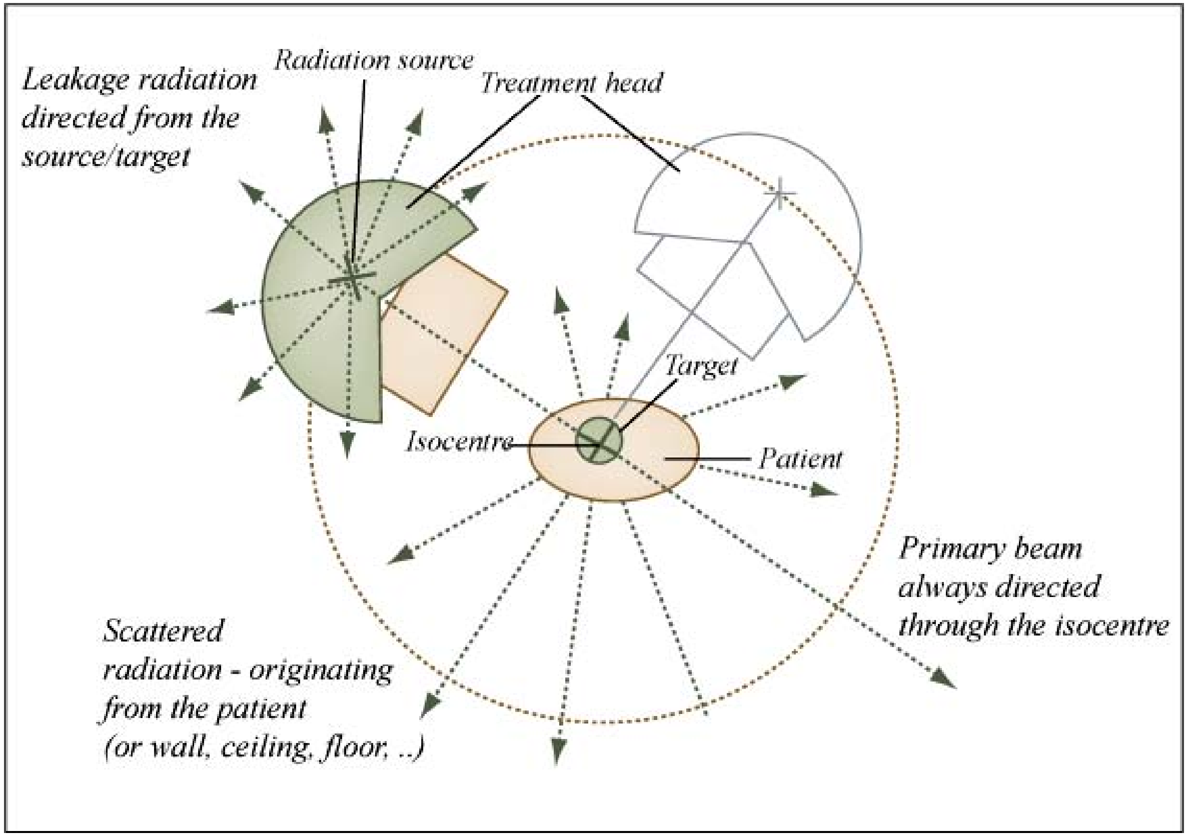
- Must consider
  - Primary beam
  - Scattered beams
  - Leakage from source



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Gamma Shielding

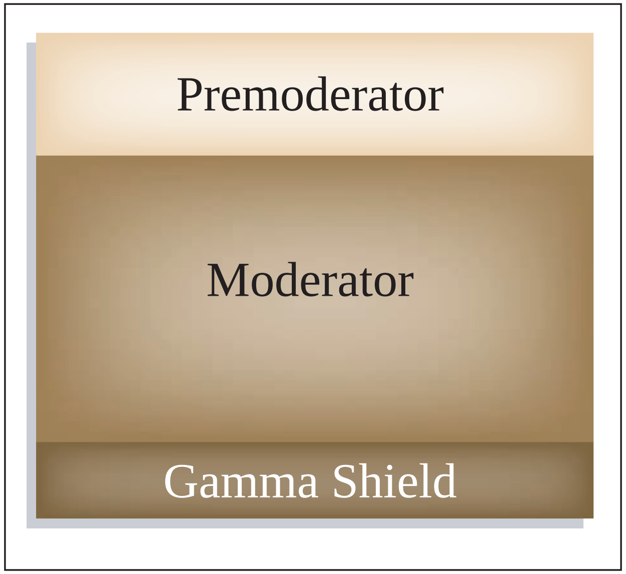
- Low energy gamma
  - Lead
- High energy gamma
  - Concrete (high density)

- Cost plays a major role in the decision
  - Lead is much more expensive than concrete!
- Very little scattering present

Neutron shielding

- Much more scattering than neutrons
- Attenuation depends very strongly on neutron energy

- Shielding designs require multiple layers
  - – Premoderator
  - – Moderator
  - – Gamma shield



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Typical Materials – Premoderator

- Scatter the high energy neutrons to facilitate efficient of other two layers
  - Iron
    - Low activation, high Z number
    - Corrodes
  - Tungsten
    - Dense, high Z number
    - Expensive
  - Lead
    - Low activation, high Z number
    - Toxic

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Typical Materials – Moderator

- Absorbs and slows neutrons down
- Concrete
  - Cheap
  - Loses water (i.e. hydrogen) at high temperatures, H creates 2.2 MeV gamma
- Borated concrete
  - Increase absorption
  - Boron depletes over time
- Water
  - Good absorption
  - No structural integrity, H creates 2.2 MeV gamma
- Polyethylene
  - Lightweight, good absorption
  - H creates 2.2 MeV gamma, C creates 4.4 MeV gamma

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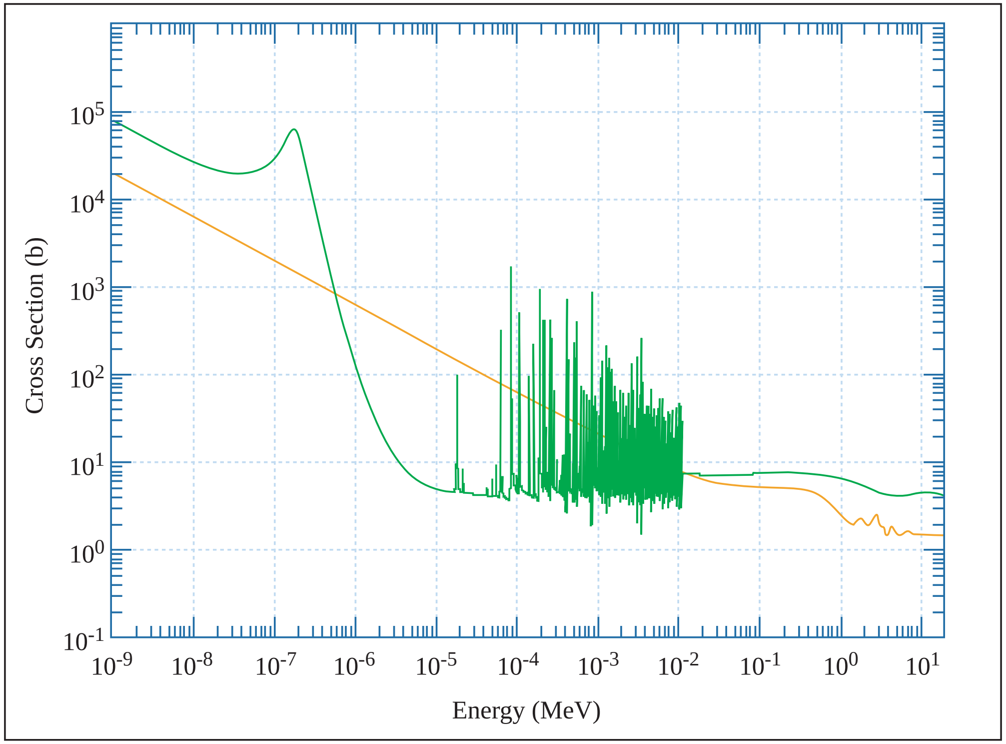
Typical Materials – Gamma Shield

- Lead
  - Excellent gamma absorption
  - Toxic
- Depleted uranium
  - Very high Z number
  - expensive

- Thermal neutrons can be easily shielded by Boron or Cadmium as they have large neutron cross sections in the thermal region
  - Cd has the disadvantage of emitting high energy gamma's after neutron capture which requires additional gamma shielding
- High speed neutrons are more difficult to shield against, because absorption x.s. are smaller in that range.
  - Thus it is first necessary to moderate the neutrons



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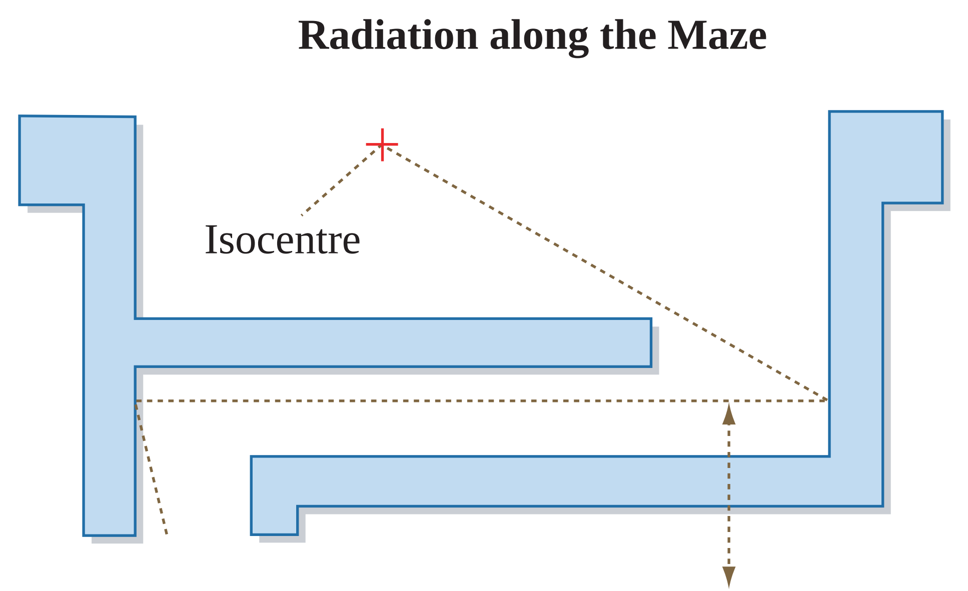
Geometry - Maze



Calculate dose rate at reflection point by ISL, then reduce this by 0.1% to give the dose rate at 1 metre from reflection point.

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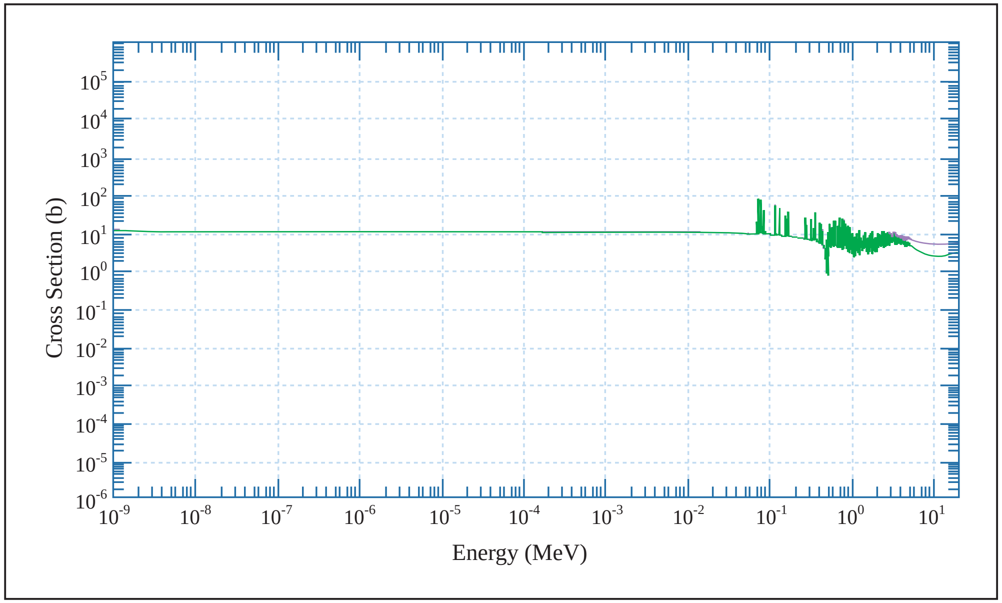
Activation

- Neutrons tend to activate shield material which can create disposition problems and can also contribute to dose to environment
  - Must carefully select materials with low activation and short-lived decay

- Effective radiation shields consist of combinations of materials
  - Low A materials to moderate
  - Thermal neutron absorbers
  - High Z materials to absorb gamma’s
- Examples
  - Polyethylene – lead
  - Concrete – Iron
  - Lithium hydride
- Choice of material depends on sample to shield, cost, space, weight restrictions, …

Pb-208



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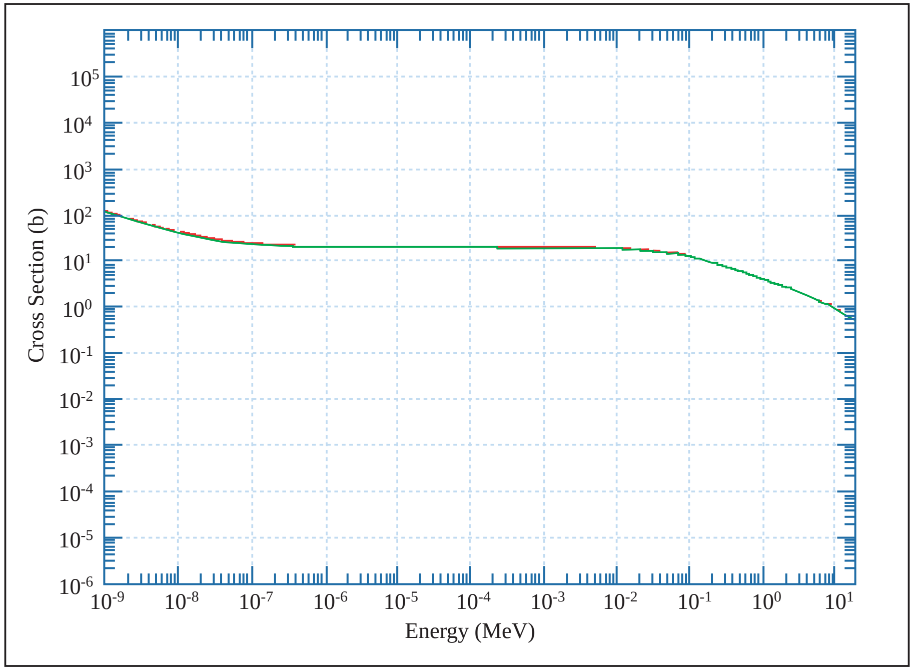
H-1



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Effective Dose Equivalent

- The energy absorbed from any type of radiation per unit mass of the absorber is defined as the absorbed dose
  - Historical unit is the rad = 100 ergs/gram (1 erg = 10^{-7} J)
  - SI unit is the Gray (Gy) = 1J/gram (1 Gy = 100 rad)

- Since neutron and gamma's are relatively penetrating radiations, the dose equivalent is easier to estimate from fluence
  - Dose equivalent (H) = Dose (D) x Quality (Q)
    - Quality factor is a function of the type of radiation
    - If D is in rad, H is in rem
    - If D is in Gy, H is in Sievert (Sv)
- Fluence is given by Φ
  - Integral of the flux over time of exposure

- In air, since there is almost no attenuation, we can estimate fluence to be

$$\Phi = N / 4\pi d^2$$

where d is the distance from the source and N the total integrated number of neutrons

- Conversion from fluence-to-dose must take into account
  - Secondary particle production
  - Kinetic energy of these particles
  - Quality factor of particle type
    - Fast electrons Q = 1
    - Gamma’s Q = 1
    - Alpha Q = 20
    - Neutrons Q = 5-20 (depends on energy level)

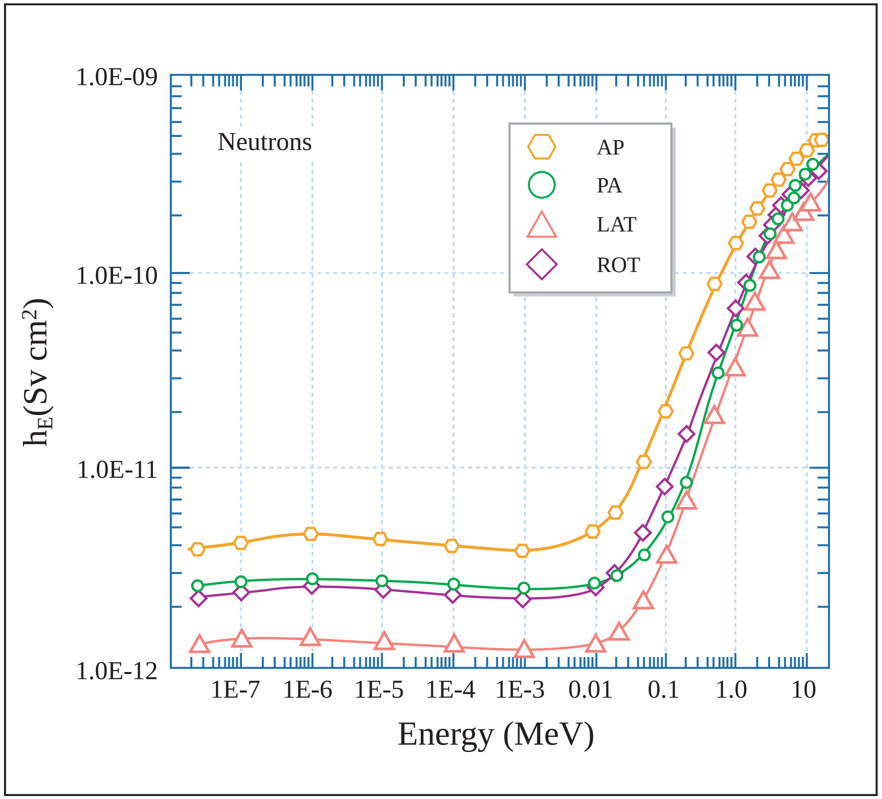
- Conversion will also depend on specific target
  - Tissues and organs
  - Orientation
  - Self-shielding
  - Attenuation
- Effective dose equivalent is correlated to the flux

$$H_E = h_E \Phi$$

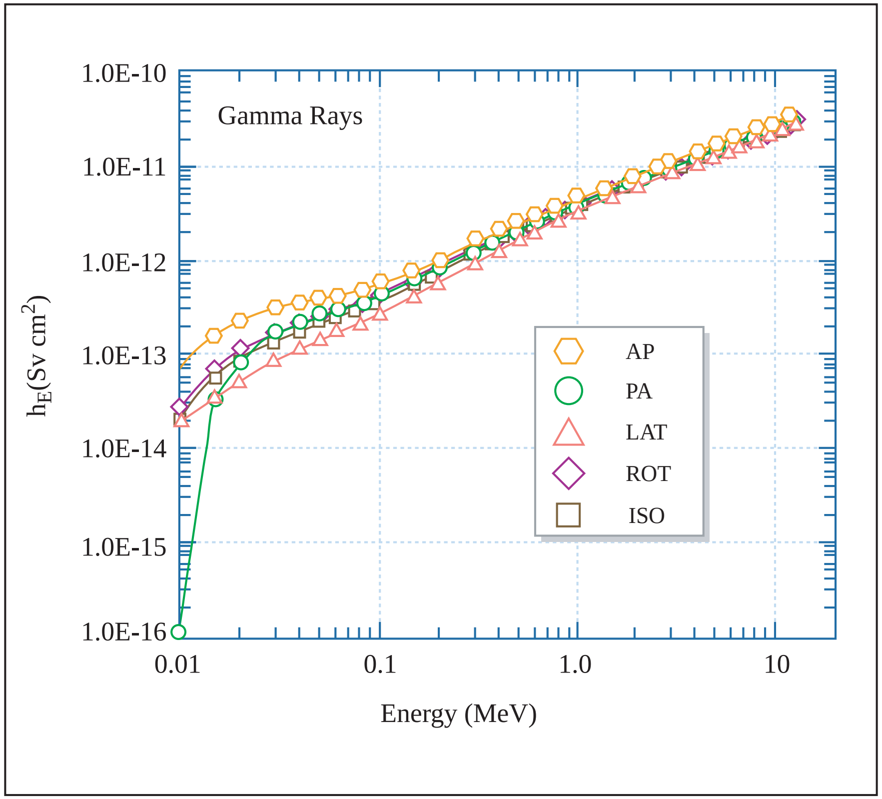
where h_E is the fluence-to-dose factor and is evaluated for different models using photon-neutron-electron transport calculations

Conversion factor

Neutrons



Photons



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- Estimation of the effects of a given exposure to ionizing radiation is by its nature an inexact science.
- Biological effects are not absolute physical quantities that can be measured with high precision.
- Effective dose calculations provide guidance in approximating the potential effects of a given exposure to radiation.

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