NE585
NUCLEAR FUEL CYCLES
Interaction of radiation with matter
2

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Learning objectives

Explain the different ways radiation interacts with matter

Apply the concept of cross sections to neutron interactions with matter

Derive neutron scattering relationships

Book – Chapter 3

Learning nodes

Gamma ray interactions
Photoelectric effect
Pair production
Compton effect

Fission

Neutron interactions
Elastic scattering
Lethargy
Inelastic scattering
Radiative capture
Charged particle emission
Bremsstrahlung radiation

Cross sections

Neutron density

Neutron fluence

Neutron flux

Gamma rays interact with matter in very complicated ways

Fortunately for us we only need to know three

Photoelectric effect

The photoelectric effect occurs when a high energy photon knocks out an electron

Interacts with entire atom

Electron ejected

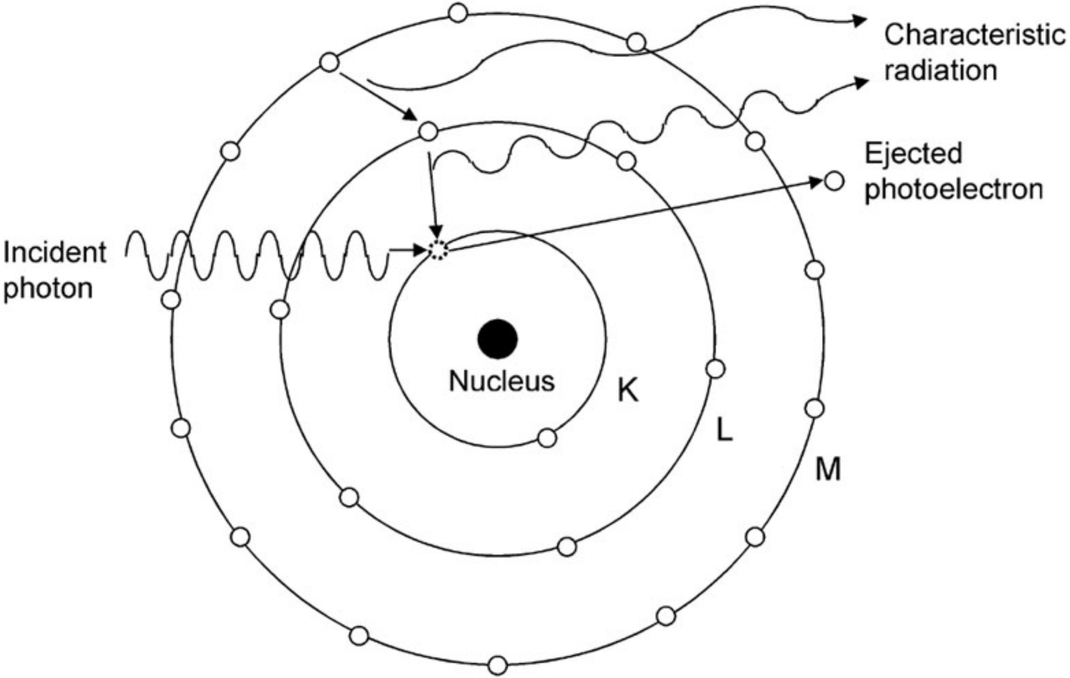
Leaves a positively charged ion

Usually an x ray emission

Atom recoil carries little Kinetic Energy (KE)

Highly probable with heavier atoms

Why?


Characteristic
radiation
Ejected
photoelectron
Incident
photon
Nucleus
K
L
M

Pair production

For pair production, a high energy photon creates a negatron and positron

Only occurs with photon above 1.022 *MeV*

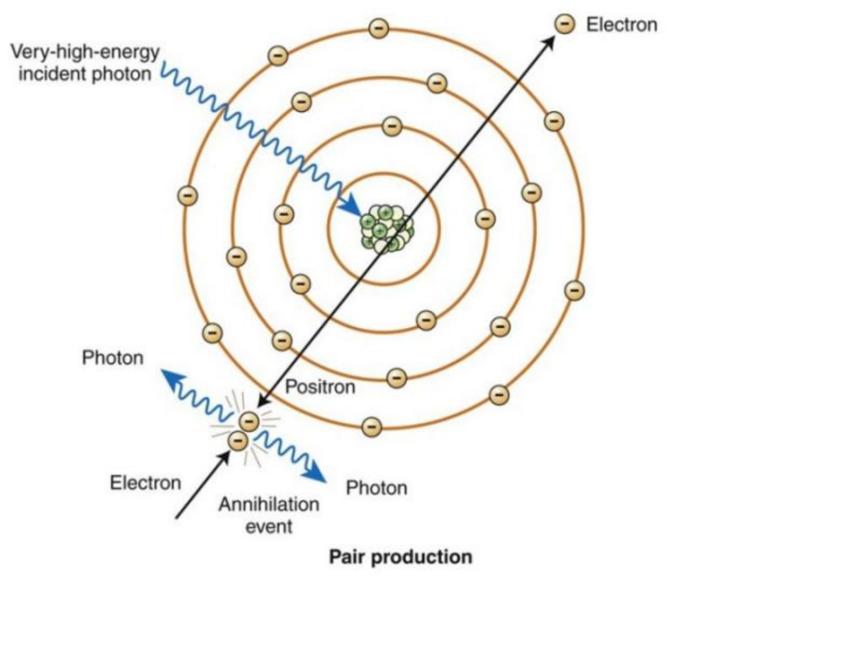
And near the nucleus

Why?

Then they react with other negatrons/positrons

Give off characteristic 0.511 *MeV* photons

Only in vicinity of Coulomb field


Electron
Very-high-energy incident photon
Photon
Positron
Electron
Photon
Annihilation event
Pair production

Compton effect

The Compton effect is elastic scattering of a photon by an electron

Energy and momentum are conserved

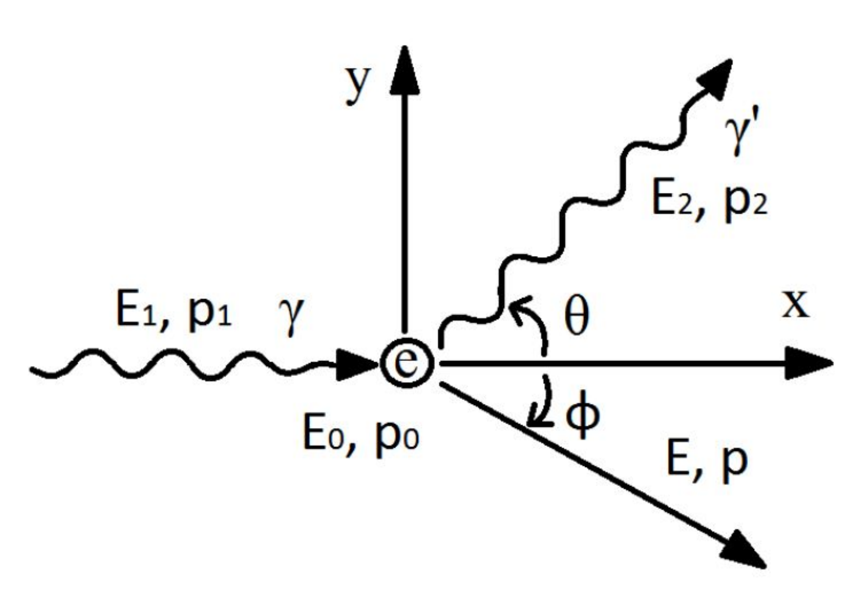
All angles of scattering possible

$$E_2 = \frac{E_1 E_\epsilon}{E_1(1 - cos\theta) + E_\epsilon} \tag{1}$$

Should be able to derive maximum energy from this equation

Why would the Compton effect cause problems with shielding? [1]

[1] (Talk about NE class experiment at WPI)


y
γ'
E2, p2
E1, p1
γ
θ
x
e
ϕ
E0, p0
E, p

Fission

What is the energy of fission from thorium and uranium fuel cycles?

$$ {}_0^1n + {}_{92}^{235}U \rightarrow {}_{36}^{92}Kr + {}_{56}^{141}Ba + {}_0^1n + {}_0^1n + {}_0^1n \tag{2} $$

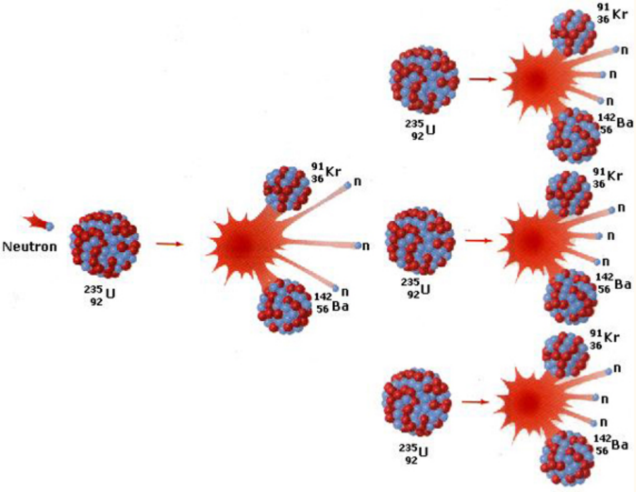
$$ {}_0^1n + {}_{92}^{233}U \rightarrow {}_{36}^{92}Kr + {}_{56}^{141}Ba + {}_0^1n \tag{3} $$

$$ Q = 173.3\ MeV,\ 185.4\ MeV \tag{4} $$

For comparison, burning a carbon atom releases 4 eV

Some super heavy nuclei fission spontaneously

Fission is the splitting of atoms, which releases a lot of energy



Neutron interactions

What is the energy of fission from thorium and uranium fuel cycles?

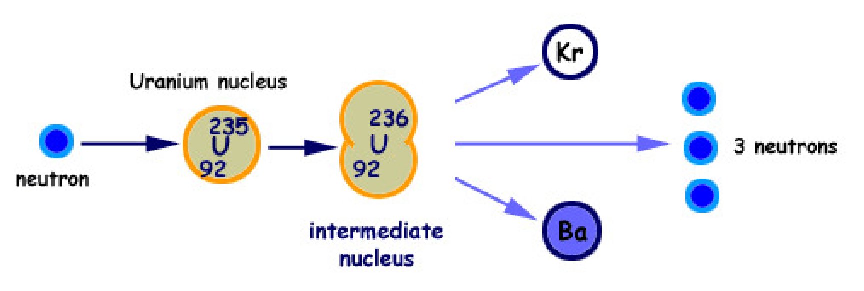
Why?

$$ {}_{0}^{1}n + {}_{92}^{233}U \rightarrow {}_{92}^{236}U^{*} \rightarrow \tag{5} $$

Sometimes the compound nucleus is left in an excited state, before the reaction proceeds, but this can be a very short time

What happens is highly dependent on neutron energy/temp

So you do not always write the compound nucleus in the reaction expression


Uranium nucleus
235
U
92
neutron
236
U
92
intermediate
nucleus
Kr
Ba
3 neutrons

There are several ways neutrons interact with nuclei

(1) Elastic scattering (probably most important)

(2) Inelastic scattering

(3) Radiative capture/neutron absorption

(4) Charged-particle reactions

(5) Neutron producing reactions

What is the energy of fission from thorium and uranium fuel cycles?

$$ {}_{0}^{1}n + {}_{92}^{233}U \rightarrow {}_{92}^{236}U^{*} \rightarrow {}_{92}^{233}U + {}_{0}^{1}n \tag{6} $$

$$ {}_{0}^{1}n + {}_{92}^{233}U \rightarrow {}_{92}^{236}U^{*} \rightarrow {}_{92}^{233}U + {}_{0}^{1}n' \tag{7} $$

$$ {}_{0}^{1}n + {}_{92}^{233}U \rightarrow {}_{92}^{236}U^{*} \rightarrow {}_{92}^{233}U + \gamma \tag{8} $$

$$ {}_{0}^{1}n + {}_{92}^{233}U \rightarrow {}_{92}^{236}U^{*} \rightarrow {}_{92}^{233}U + {}_{2}^{4}\alpha \tag{9} $$

$$ {}_{0}^{1}n + {}_{92}^{233}U \rightarrow {}_{92}^{236}U^{*} \rightarrow {}_{92}^{233}U + 2{}_{0}^{1}n \tag{10} $$

Elastic scattering

With elastic scattering, the neutron strikes the nucleus, and they scatter

If you’ve ever played pool (billiards), it’s that sort of

But the neutron is a lot smaller than the nucleus

Kinetic energy and momentum is conserved with nucleus and neutron

Nucleus can be at rest or moving, but neutron velocity is really faster

So basically the nucleus is ‘at rest’

With elastic scattering, the neutron strikes the nucleus, and they scatter

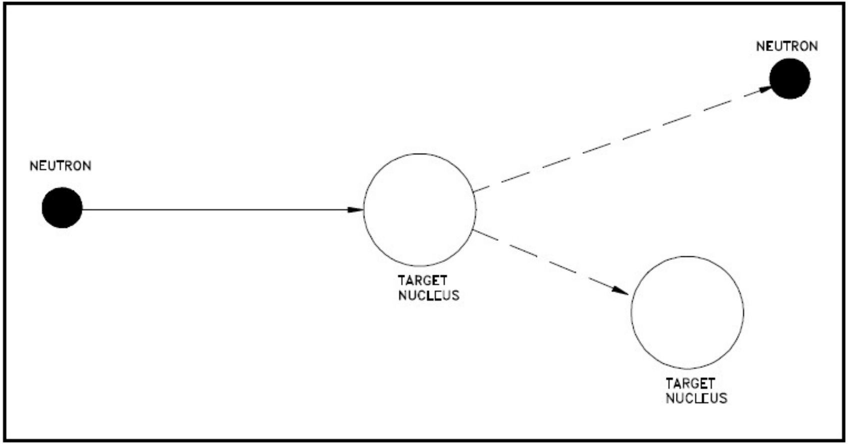
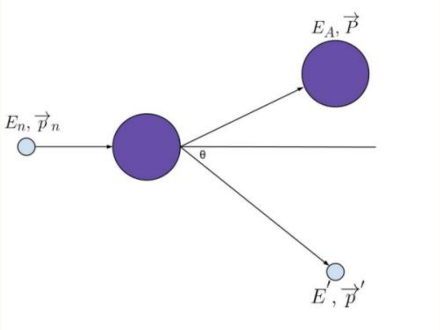


Figure 16 Elastic Scattering

The neutron energy after the collision can be obtained by conservation principles



$$E = E' + E_A \tag{11}$$

$$\vec{p} = \vec{p}' + \vec{P} \tag{12}$$

$$E' = \frac{E_n}{(A+1)^2} \cdot [cos\theta\sqrt{A^2 - sin^2\theta}]^2 \tag{13}$$

$$\alpha \equiv (\frac{A-1}{A+1})^2 \tag{14}$$

(p58-9) - but in any edition

Lethargy

The concept of lethargy was invented for neutron moderation

They aren't good at naming things

$$u \equiv ln\frac{E_M}{E} \tag{15}$$

E_M is the highest neutron energy in the system

What is happening mathematically?

The concept of lethargy was invented for neutron moderation

So, lethargy increases when neutron slows down

Log scale is used because neutrons have huge energy range

What we really want is average log energy loss per elastic scatter/collision

$$\xi \equiv 1 + \frac{\alpha}{1-\alpha} ln\alpha \tag{16}$$

$$\alpha \equiv (\frac{A-1}{A+1})^2 \tag{17}$$

Average change in lethargy

What does this tell us?

In terms of thermal reactor design?

Because elastic scattering can be applied to find out how a neutron will slow down

$$\overline{n}_{COL} \equiv \frac{1}{\xi} ln \frac{E_1}{E_2} \tag{18}$$

What is the physical interpretation?

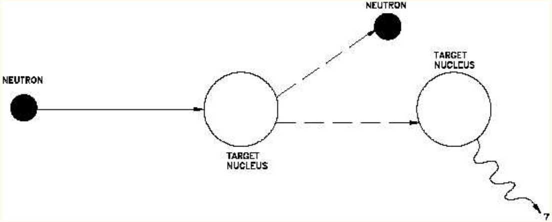
Moderation = getting high energy neutrons to low energy

This means from fast moving to slow

Why? This is a very fundamental reactor design constraint

Inelastic scattering

With inelastic scattering, it is the same as elastic, but the nucleus is left in excited state



Energy is lower for heavier nuclei

Kinetic energy and momentum not conserved

Compound nucleus emits lower energy neutron

Then the nucleus emits gamma rays

Occurs above a certain neutron energy for each nucleus

A neutron with same energy may not scatter with different nuclei

Radiative capture

Because elastic scattering can be applied to find out how a neutron will slow down

This is important with reactor design (fission and fusion)

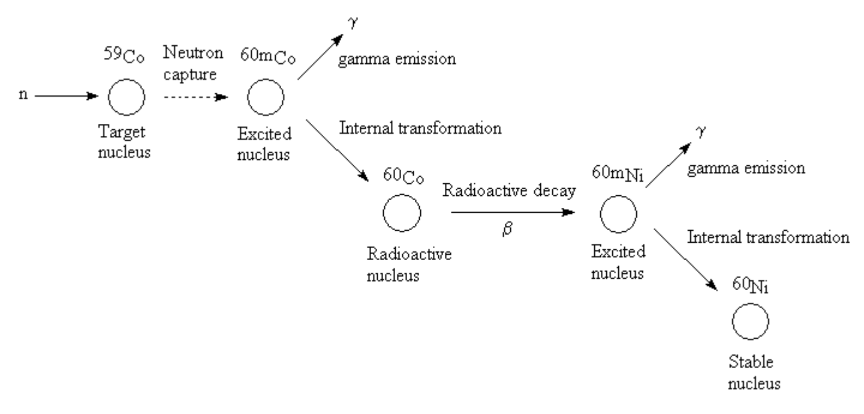
Take a guess why?

We also call this neutron absorption in this context

Usually a low energy neutron reaction (why?)

Depends on neutron energy and material

Other applications include NAA, isotope production


59Co
Neutron capture
60mCo
γ
gamma emission
n
Target nucleus
Excited nucleus
Internal transformation
60Co
Radioactive decay
β
Radioactive nucleus
60mNi
γ
gamma emission
Excited nucleus
Internal transformation
60Ni
Stable nucleus

Charged particle emission

The nucleus can also absorb a neutron and emit a charged particle

This can be used to design neutron detectors

Part of fusion reactions

Boron is used as neutron ‘poison’ and can absorb low energy

Materials testing for fuel cladding (fusion and fission)

Charged particle production could be adverse

Neutron depth profiling; surface of semiconductors, etc.

$$ {}_0^1n +_5^{10} B \rightarrow 11_5B^* \rightarrow_3^7 Li + \alpha + \gamma \tag{19} $$

Bremsstrahlung radiation

A consequence of charged particle reactions is the Bremsstrahlung radiation

Deceleration of the charged particle through matter

A photon is emitted from interaction with the electron cloud or nucleus with the charged particle

The moving particle loses energy and this is in the form of a photon

aka ‘braking radiation’

Continuous radiation distribution

Can be used to identify galaxy clusters

Cross sections

A cross section tells us basically the probability of an interaction (with something)

Neutrons for us

Cross sections for other events like Compton, etc.

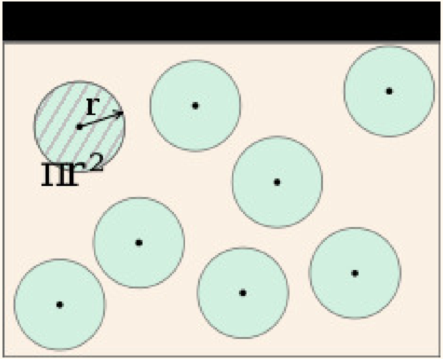
People dedicate careers to establishing precise cross sections (and good for them)

Cross sections are given as a hypothetical area around the target nucleus (big area, high probability)

$$1\ b = 10^{-24}\ cm^2 \tag{20}$$

Dependent on speed/energy/temperature

Try to think of this as the likelihood of interaction



Since there is a lot of space in between nuclei, some neutrons just fly by

Radius of a nucleus $\sim 10^{-12}$ cm^2, so cross section is $\sim 10^{-24}$ cm^2

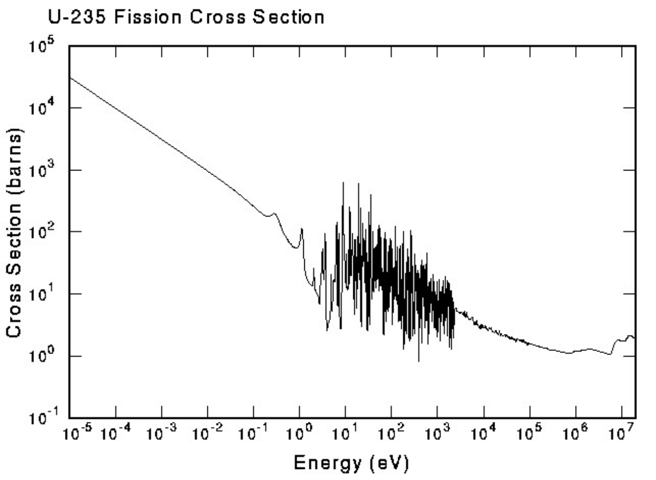
Which is where they derived barns

Probability of interaction is the ratio of the total surface area of the atoms to the total area of the medium

How does cross section vary with temperature?

What is physically happening?

Fission cross sections have this general trend



If fission neutrons start at 2 MeV, where would you want them for a reactor?

Neutron density

Neutron density in the thermal region follows Maxwellian distribution

As fast neutrons slow down, they eventually come into thermal equilibrium with the thermal motion of the atoms in the medium through which they are moving

Also call the $\frac{1}{v}$ region

Resonances occur if the compound nucleus is produced in one of its characteristic excited states

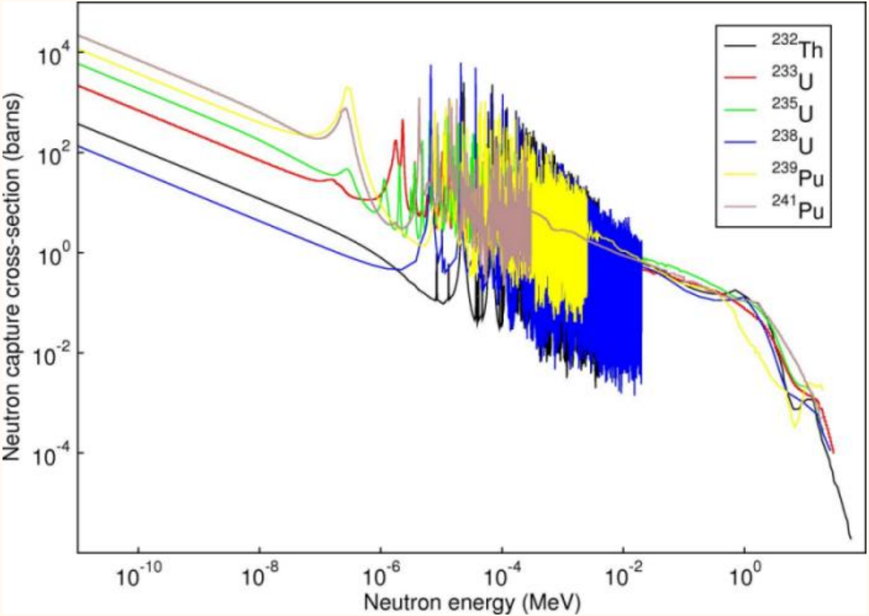
And the fast end is a smooth region

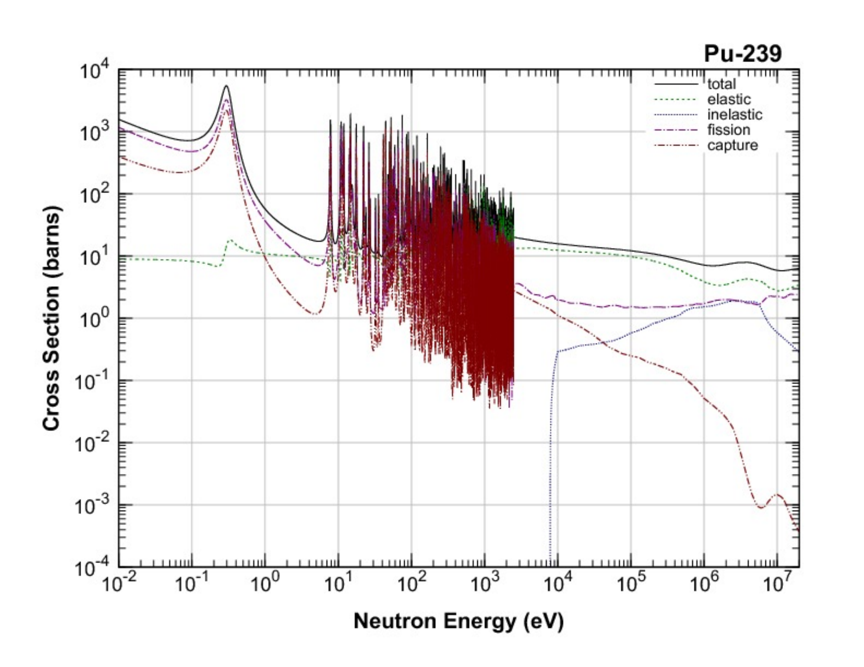
What we really want is average log energy loss per elastic scatter/collision

$$n(E) = n_0 \frac{2\pi\sqrt{E}}{(\pi k T)^{\frac{3}{2}}} \cdot e^{-\frac{E}{kT}} \tag{21}$$

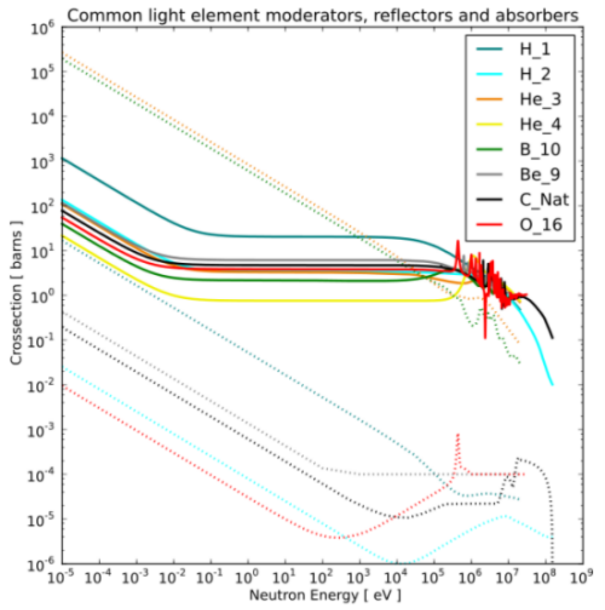
$$n(v) = n_0 \frac{4\pi v^2}{\left(\frac{2\pi k T}{m}\right)^{\frac{3}{2}}} \cdot e^{-\frac{mv^2}{2kT}} \tag{22}$$

How does this affect reactor operation?




Pu-239
total
elastic
inelastic
fission
capture
Cross Section (barns)
Neutron Energy (eV)
10^{4}
10^{3}
10^{2}
10^{1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-2}
10^{-1}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}
10^{7}

How do you choose materials?

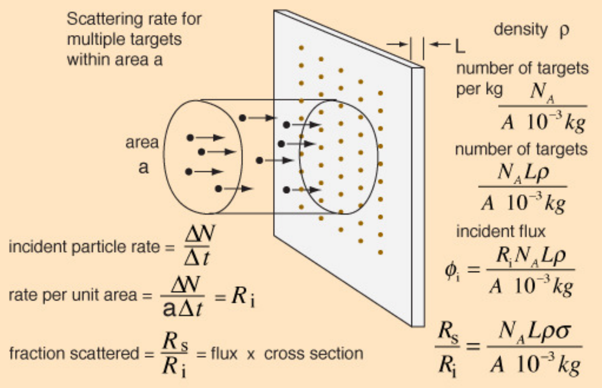


solid = scatter

dotted = absorption

Neutron fluence

How many neutrons/s that strike a target actually interact with target nuclei?



$$I[\frac{n}{cm^2 \cdot s}] = [\frac{n}{cm^3}] \cdot v[\frac{cm}{s}] \tag{23}$$

$$[\frac{n}{cm^3}] \cdot v[\frac{cm}{s}] \cdot a[cm^2] = \frac{n}{s} \tag{24}$$

$$nuclei_{tar} \equiv N[\frac{nuclei}{cm^3}] \cdot a[cm^2] \cdot L[cm] \tag{25}$$

$$\frac{interactions}{s} \equiv (\sigma I)(NaL) \tag{26}$$

The cross section is the probability the neutron interacts with the nucleus

Section 3.2, example 3.1

Then we can define the mean free path

First, macroscopic cross section is –

$$\Sigma[cm^{-1}]N\sigma = \frac{\rho N_A}{A} \tag{27}$$

Then mean free path is –

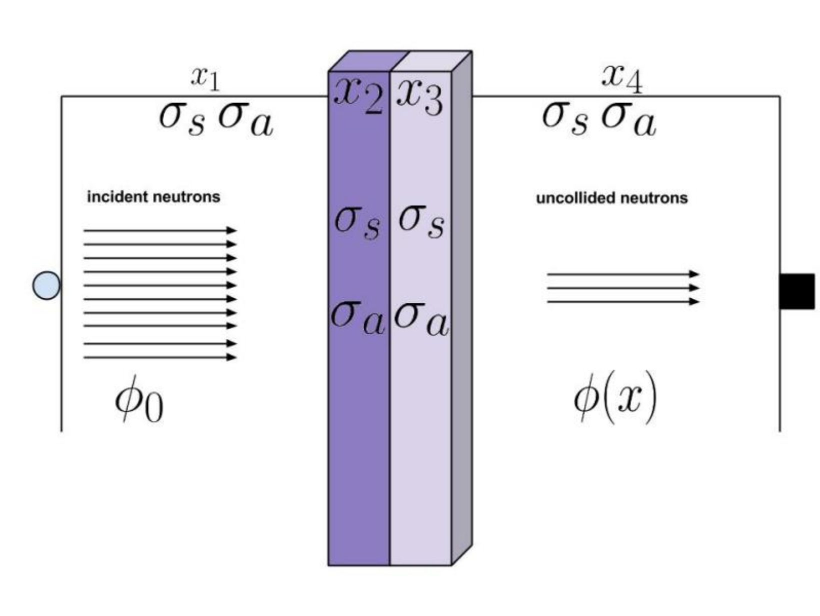
$$\lambda \equiv \Sigma^{-1}\ [cm] \tag{28}$$

Expected value of distance between interactions

Then the interaction rate is –

$$R_X = \phi\Sigma_X \tag{29}$$

Neutron flux



Which flux is larger?

Point source

Isotropic

Uncollided flux decreases exponentially with distance

$$-d\phi = \Sigma_t \phi dx \tag{30}$$

$$-\frac{d\phi}{\phi} = \Sigma_t dx \tag{31}$$

Probability a neutron interacts in dx

$$\phi(x) = \phi_0 e^{-\Sigma_t x} \tag{32}$$

Probability a neutron does not interact in x

It's the same for gamma rays

Shielding is material density dependent

$$\phi(x) = \phi_0 e^{-\mu x} \tag{33}$$

Look up mass attenuation coefficient

$$\frac{\mu}{\rho}\,[\frac{cm^2}{g}] \tag{34}$$

Then just multiply mass attentuation by material density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平成十九年元日