# Nuclear Physics and Astrophysics

PHY-302

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# Lecture 16 - Uranium

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## Energy In Fission

Consider $^{235}U + n \rightarrow {}^{236}U^*$

Excitation energy, $E_{ex} = [m(^{236}U^*) - m(^{236}U)]c^2$

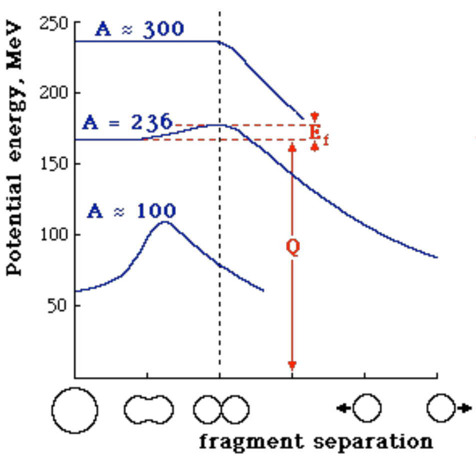
(neglecting binding energy of this neutron!)

Assuming negligible neutron kinetic energy: $m(^{236}U^*) = m(^{235}U) + m_n$

Then, $E_{ex}$ = 6.5 MeV

Activation energy for $^{236}U$ is 6.2 MeV

Thus $^{235}U$ can be fissioned with neutrons of negligible energy!




Similarly for $^{238}U$ activation energy = 6.6 MeV, but $E_{ex}$= 4.8 MeV

Thus for $^{238}U$ neutrons must have ~MeV energy for fission to occur

## Prompt Neutron Energy Spectrum From $^{235}U$ Fission

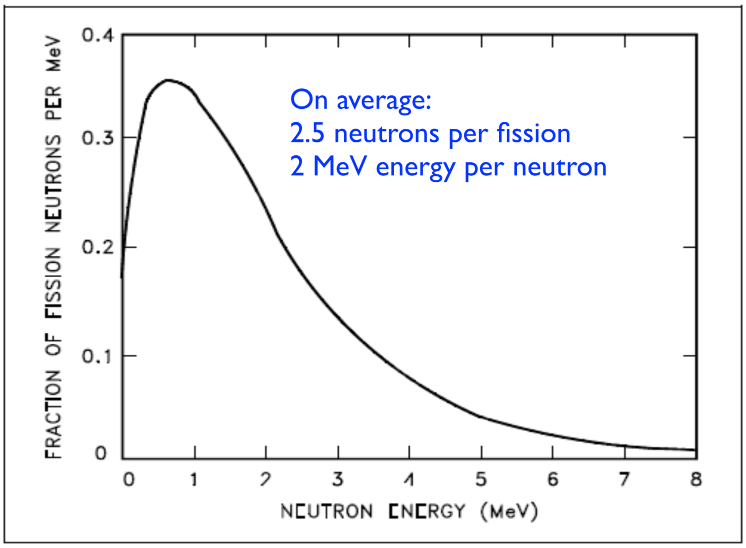



Figure 2 Prompt Fission Neutron Energy Spectrum for Thermal Fission of Uranium-235

Thus $^{235}U$ produces more neutrons
Each can induce fission again
This is a chain reaction!

Problem:
Natural abundances of uranium are: 0.72% $^{235}U$ and 99.28% $^{238}U$



## How To Control Fission

To most people "nuclear physics" means only nuclear power and atomic bombs

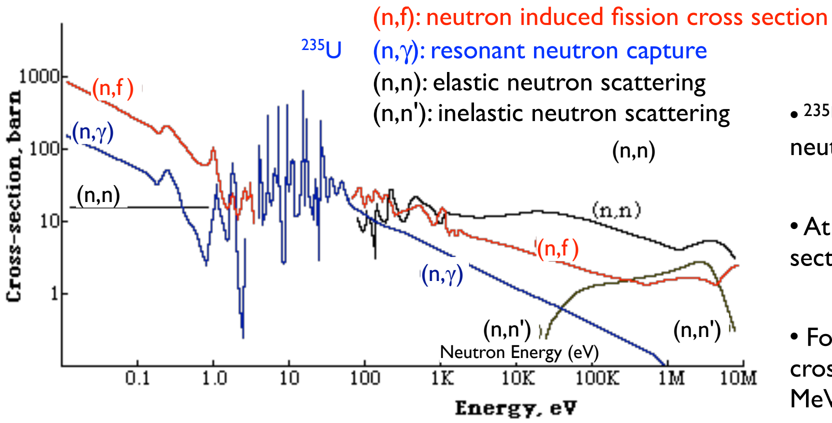
Chain reaction of fission for power generation was patented in 1939 Leo Szilard

With the start of WWII race for harnessing fission began

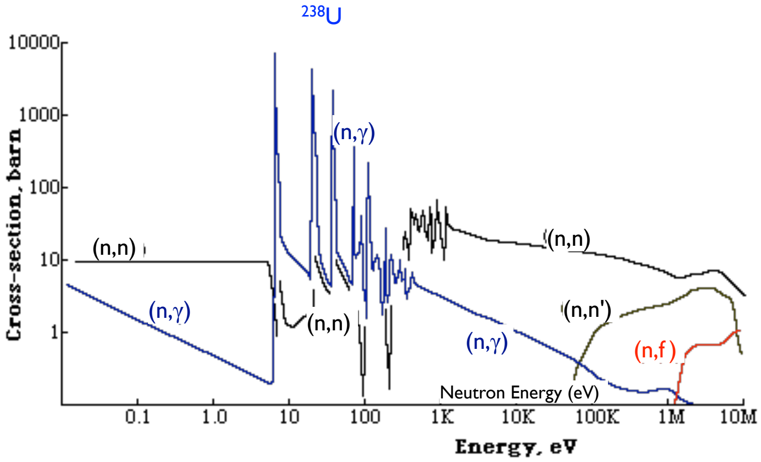
Famous Manhattan Project developed technology resulting in atomic bomb in 1945

We will now look at the factors for controlled & uncontrolled energy release

Biggest problem: ensure enough neutrons at right energy survive to continue chain reaction




- $^{235}U$ is fissile at all energies of incident neutron
- At low neutron energy (n,f) is largest cross section for $^{235}U$
- For $^{235}U$ thermal neutron (<0.1 eV) fission cross section is 1000 times greater than for MeV neutrons




- Prompt neutron energy spectrum peaks ~1MeV (see page 4)
- At 1MeV scattering cross section is larger than fission cross section, specially for $^{238}U$
- At 10-100 eV resonant neutron capture (n,γ) cross section very large! Specially for $^{238}U$



Key to controlling chain reaction: neutrons that feed next generation of fission

Define **neutron reproduction factor** $k_\infty$:

**net change in neutron number between generations**

Defined for infinite medium - losses through surfaces ignored

Thus each thermal neutron produces $k_\infty$ new thermal neutrons

Require $k_\infty$>1 for chain reaction to occur

Note: $^{235}U$ produces on average 2.5 fast (~1 MeV) neutrons per fission

Need to slow neutrons down to thermal energies (0.025 eV)

Process known as **moderating**

Achieved by forcing elastic collisions between neutrons & nuclei

Light nuclei preferred - more energy transferred in collisons

Carbon (graphite) is a good choice

See h/w 4

$$T_\alpha = \frac{Q}{1 + (m_\alpha / m_Y)}$$

Create a chain reacting **pile** - lattice structure of U blocks & alternating carbon

If reproduction factor for finite pile is 1.0 pile is critical

subcritical pile: k<1

supercritical pile: k>1

## Calculating $k_\infty$

Consider N thermal neutrons in first generation

Each fission produces ν neutrons

We will not get νN fast neutrons immediately

Some thermal $n^0$s will be absorbed eg. (n,γ) reactions

Define η: mean number of fission (i.e fast) neutrons per original thermal neutron

Not all thermal neutrons will cause fission: η<ν

Fission cross section (thermal) = $\sigma_f$

Absorption cross section (thermal) = $\sigma_a$

Relative probability for <u>thermal</u> neutron to cause fission = $\dfrac{\sigma_f}{\sigma_f + \sigma_a}$

Thus

$$\eta = \nu \frac{\sigma_f}{\sigma_f + \sigma_a}$$ at thermal neutron energies



## Calculating $k_\infty$

For $^{235}$U: $\sigma_f$ = 584 b
$\sigma_a$ = 97 b
η = 2.08 no. fast $n^0$ per thermal $n^0$

For $^{238}$U: $\sigma_f$ = 0 b
$\sigma_a$ = 2.75 b
η = 0 no. fast $n^0$ per thermal $n^0$

Natural abundances of uranium are: 0.72% $^{235}$U and 99.28% $^{238}$U

$$\sigma_f = 0.72\%\ \sigma_f^{235} + 99.28\%\ \sigma_f^{238}$$
$$\sigma_a = 0.72\%\ \sigma_a^{235} + 99.28\%\ \sigma_a^{238}$$

This has η = 1.33 : mean no. fast neutrons per thermal neutron
Already very close to 1.0 - need to increase this to allow for other $n^0$ losses
Increase amount of $^{235}$U
Enriched uranium has 3-5% $^{235}$U (>20% $^{235}$U used in nuclear weapons)
η = 1.84

Enrichment process is technologically challenging
...see next lecture

## Calculating $k_\infty$

We start with **N** thermal neutrons

Number of fast neutrons is $\eta.N$

**fast fission factor:** $\varepsilon$ $^{238}U$ can fission from fast neutrons producing more fast $n^0$

Increased number of neutrons $\varepsilon$ typically ~ 1.03

Number fast neutrons = $\eta.\varepsilon.N$

**Resonance escape probability: p** prob of $n^0$ **not** being resonantly absorbed via

$^{235}U(n,\gamma)^{236}U$ or $^{238}U(n,\gamma)^{239}U$ reaction

p ~ 0.9 depends on reactor geometry

- Typically ~100 collisions with carbon needed for thermalisation
- in 10-100 eV region $^{238}U$ has very large capture cross section: (n,f) > (n,n)
- If U and C mixed as fine powder: much larger chance of $n^0$ being captured by $^{238}U$
- If C in larger lumps: $n^0$ undergo more collisions before emerging from C: thermalised
- Crucial part of reactor design

Number of neutrons = $\eta.\varepsilon.p.N$

**thermal utilisation factor: f** fraction of thermal $n^0$ actually available to uranium

$$\mathbf{k_\infty} = \eta \varepsilon p f N/N$$

Moderator can also absorb thermal neutrons



## Calculating k For Realistic Reactor

For realistic reactor, neutrons are also lost through surfaces

Finally complete reproduction factor is

$$k = \eta\epsilon p f(1 - l_f)(1 - l_t)$$

$l_f$ and $l_t$ are fraction of lost thermal and fast neutrons

Expect larger pile to have smaller surface area:volume ratio - i.e smaller losses

| | | |
|---|---|---|
| Thermal Neutrons | 1000 | Initial number of thermal neutrons |
| $^{235}$U fissions produces fast neutrons | 1330 | $\eta$ ~1.33 some thermal neutrons absorbed before fission occurs, and fission produces fast neutrons |
| $^{238}$U fissions produces more fast neutrons | 1370 | $\varepsilon$ ~1.03 fast fission factor |
| | 1300 | $(1-l_f)$ ~ 0.95 leakage of fast neutrons |
| Uranium has large "resonance" absorbtion cross section | 1170 | p ~ 0.9 resonance escape probability |
| moderator thermalises neutrons some are absorbed by moderator | 1050 | f ~ 0.9 thermal utilisation factor |
| | | $(1-l_t)$~0.95 leakage of thermal neutrons |
| 1000 | | |