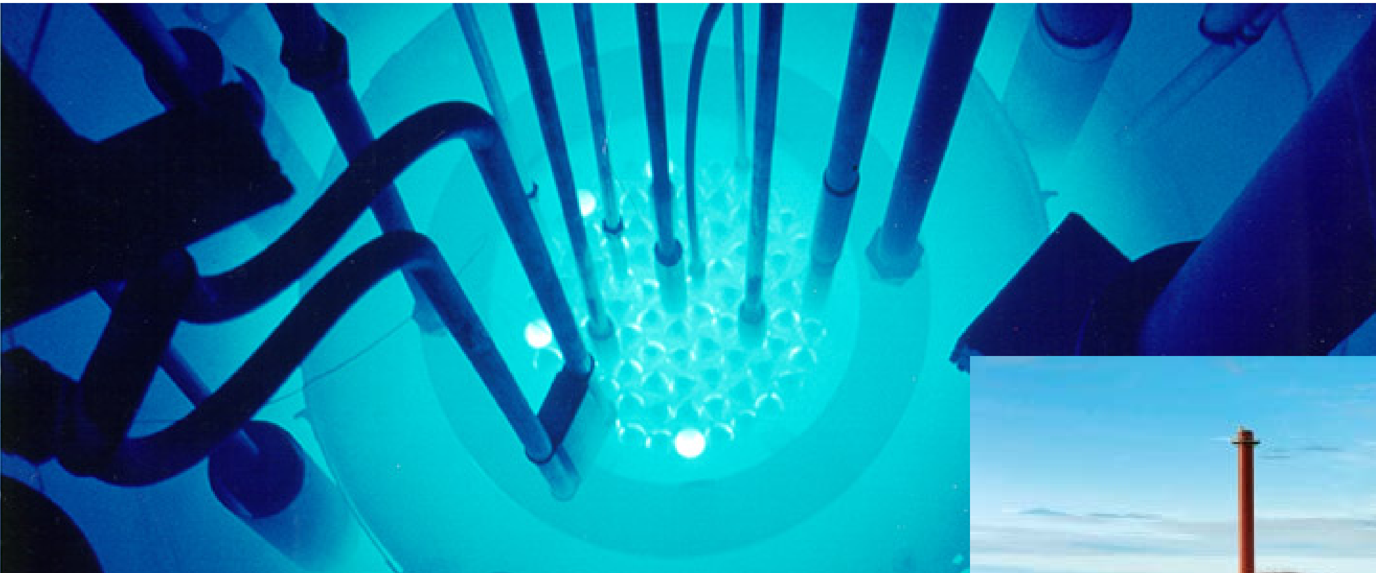
Energy from nuclear fission



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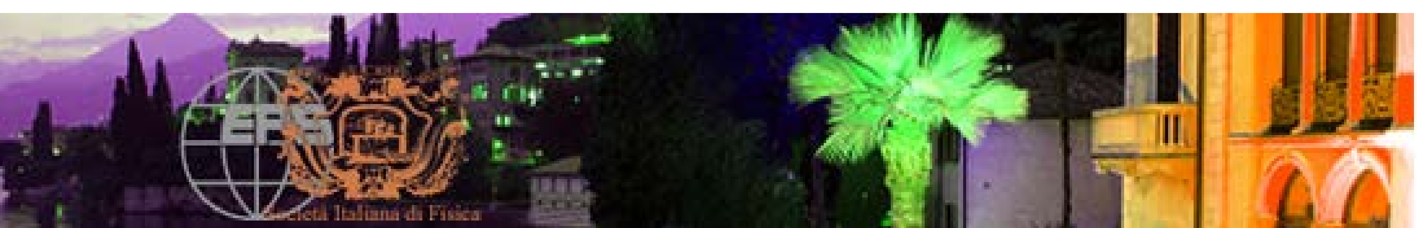


Plan

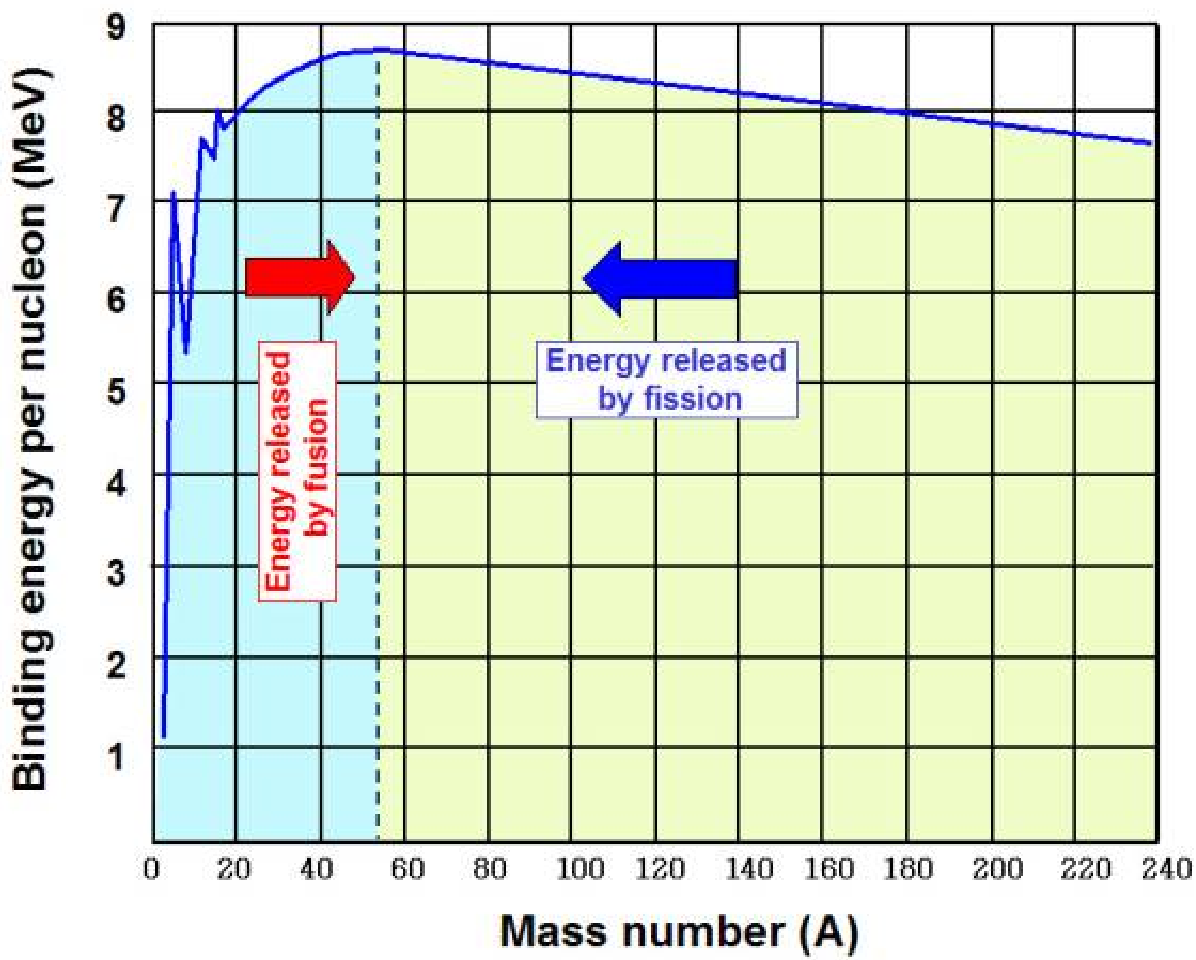
- ✓ Physics of fission
- ✓ Energy balance
- ✓ Reaction products
- ✓ Cross sections and flux
- ✓ Fuel
- ✓ Fast and slow neutrons
- ✓ Neutron “economics” and reactor kinetics
- ✓ Reactor types
- ✓ Decay heat
- ✓ Sketch of past and future

Why can fission produce energy ?

Nuclear mass and nuclear binding energy $M(Z,A) = ZM_p + (A - Z)M_n + B(Z,A)$

B(Z,A) < 0 !!! i.e. a nucleus weighs less than the sum of proton and neutron masses

$$\varepsilon \equiv \frac{|B(Z,A)|}{A}$$



How can fission produce energy ?

M(Z,A) → M(Z_1, A_1) + M(Z_2, A_2); Z = Z_1+ Z_2; A = A_1+ A_2

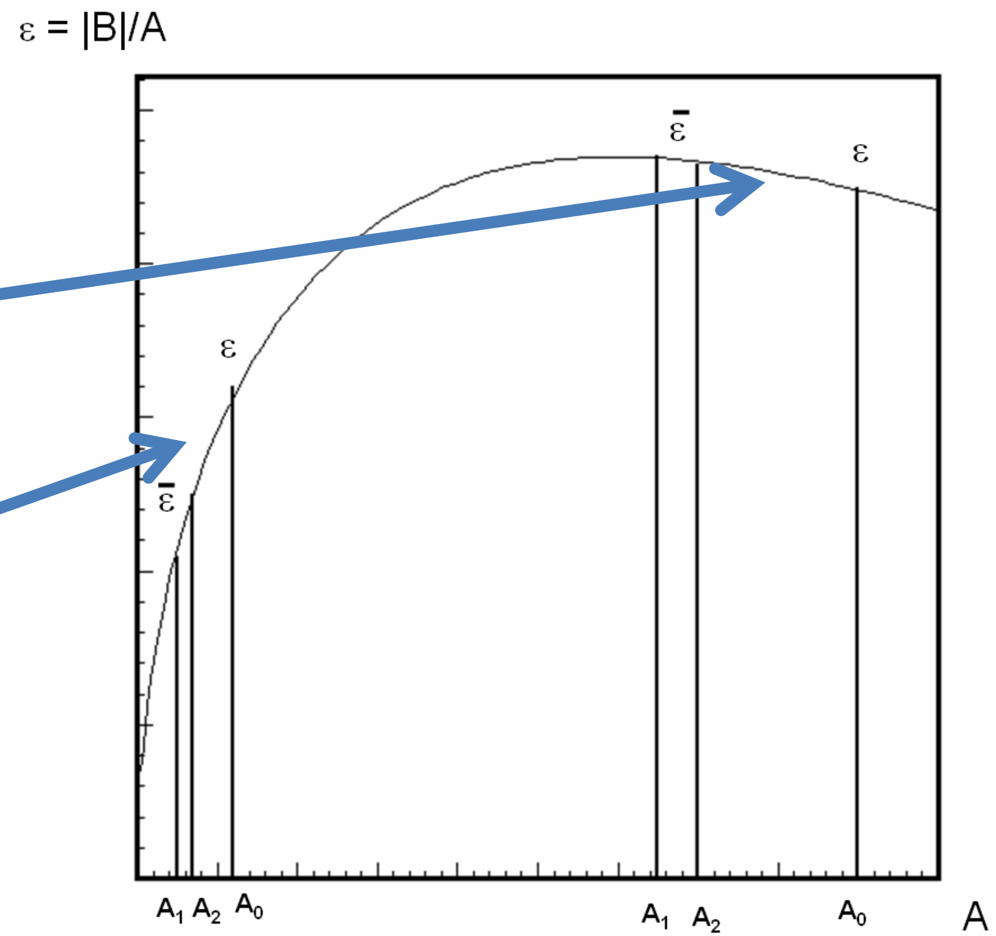
Energy balance (Q-value) $Q_{fiss} = M(Z,A) - M(Z_1,A_1) - M(Z_2,A_2)$

$$= B(Z,A) - B(Z_1,A_1) - B(Z_2,A_2) = -\varepsilon A + \varepsilon_1 A_1 + \varepsilon_2 A_2 = -\varepsilon A + \bar{\varepsilon} A = (\bar{\varepsilon} - \varepsilon) A$$

$$\bar{\varepsilon} = \frac{\varepsilon_1 A_1 + \varepsilon_2 A_2}{A_1 + A_2}$$

Qfiss > 0

Qfiss < 0

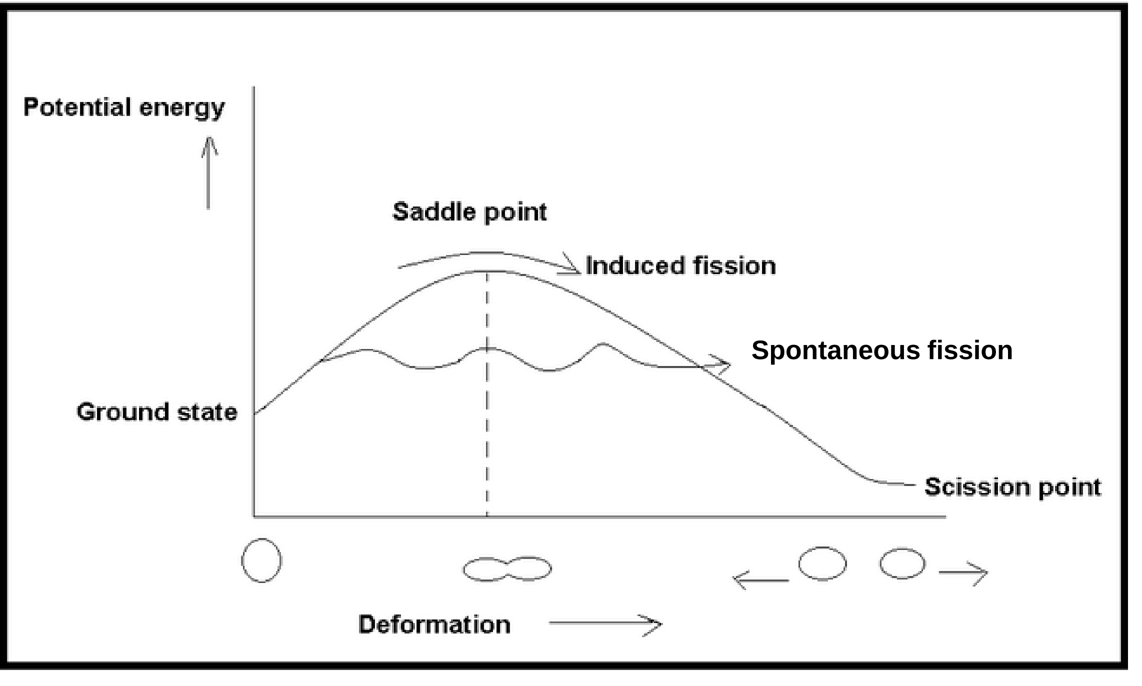


How does fission happen ?

Imagine bringing the two daughter nuclei close to one another
→ they will feel Coulomb repulsion

But in the parent nucleus they are bound together….

Therefore, their potential energy looks like this



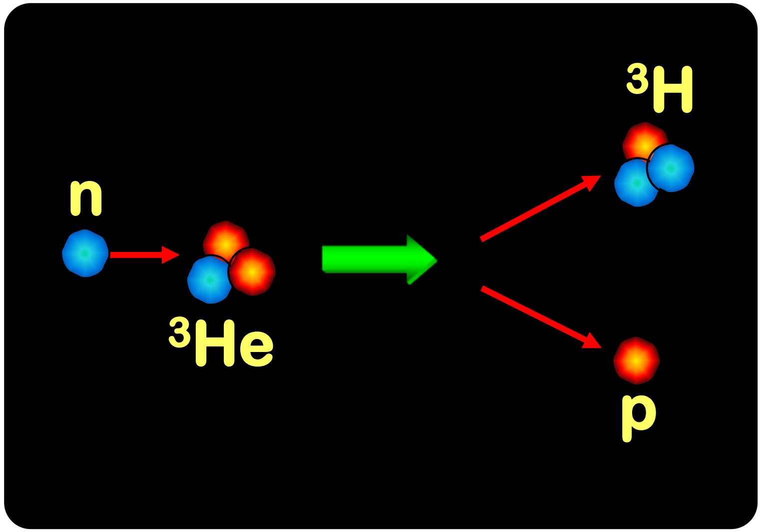
Even when they split, they have to overcome the Coulomb barrier…

To accomplish this, some extra energy will help….

Where can they get it from ?

For instance capturing a neutron…

Other neutron absorption processes yielding energy

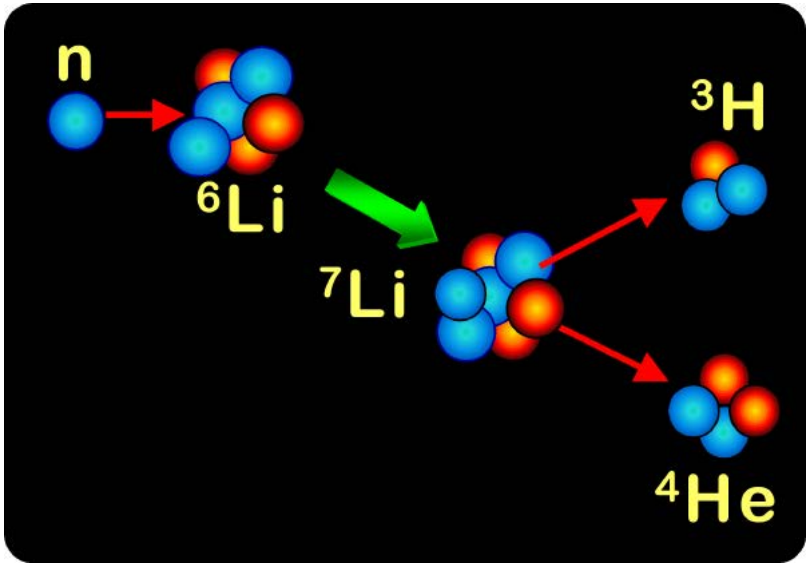


σ(thermal neutrons) ≈ 5330 b

available energy
0.76 MeV

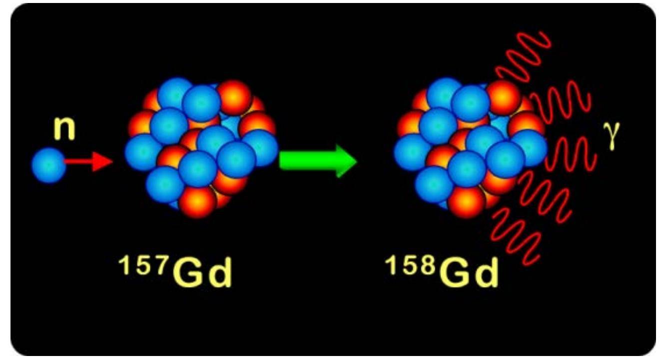
σ(thermal neutrons) ≈ 940 b

available E
4.78 MeV

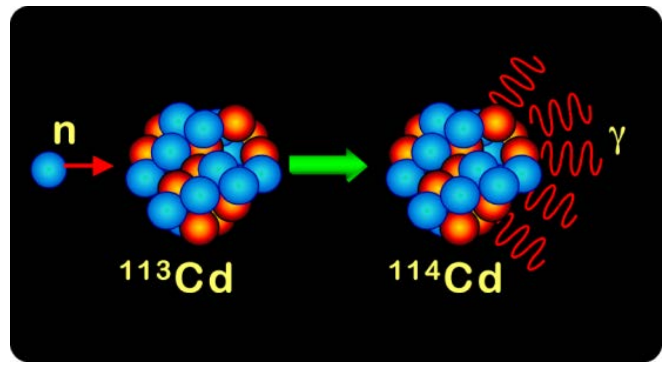


Other neutron absorption processes yielding energy

σ(thermal neutrons) ≈ 240 kb



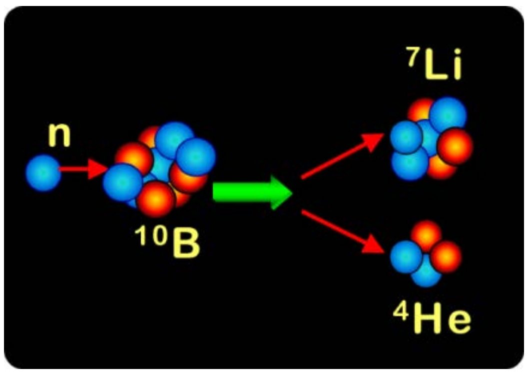
^{157}Gd

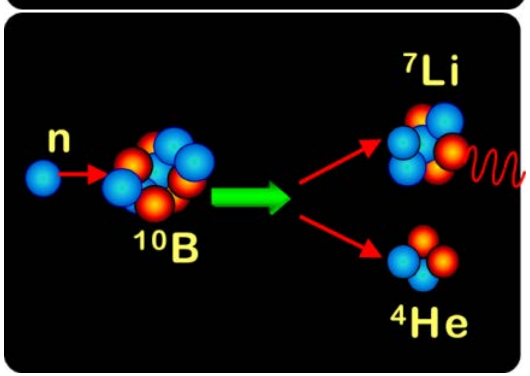


^{113}Cd

σ(thermal neutrons) ≈ 20 kb

large energy release in form of gamma rays



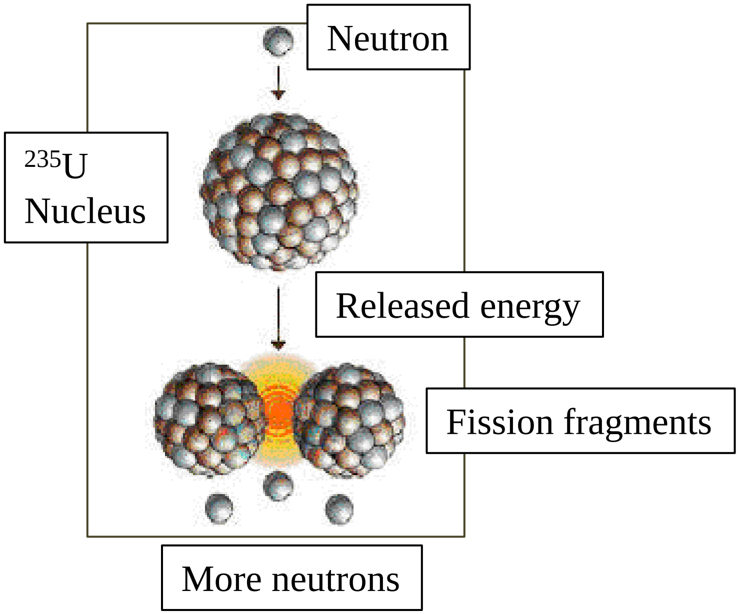


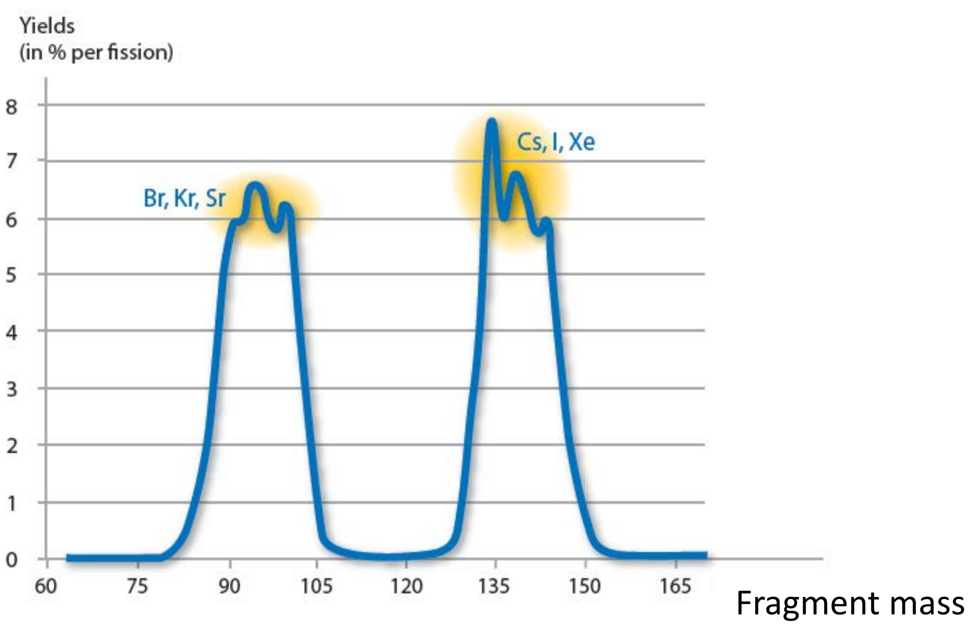
^{10}B

σ(thermal neutrons) ≈ 3840 b

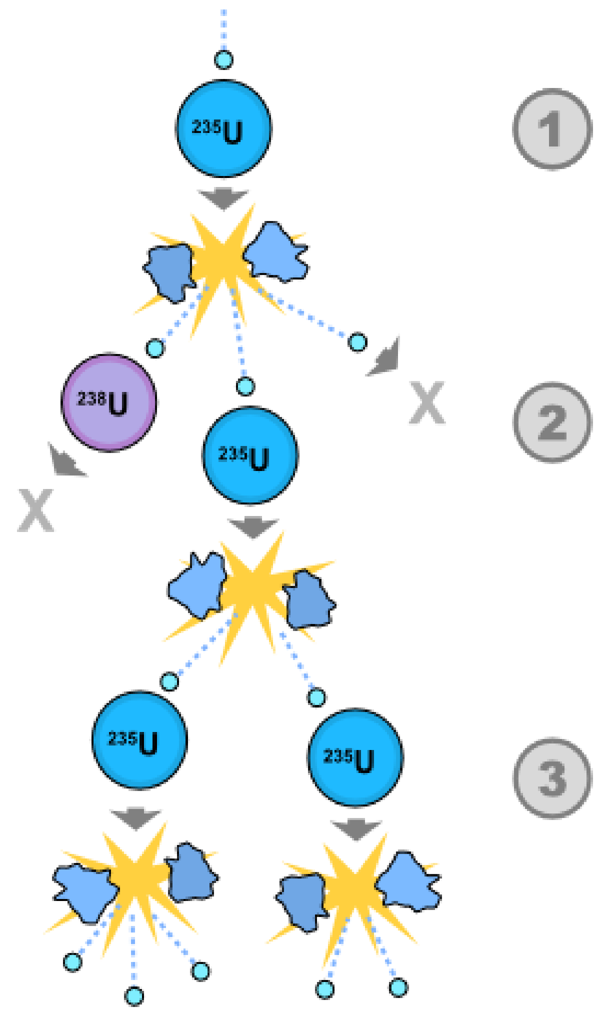
available E=2.79 MeV (and gamma rays)

But then, why is fission of heavy elements so special ?





The additional neutrons can give rise to a sustained **chain reaction**



Amount of energy and reaction products

When a uranium nucleus fissions into two daughter nuclei fragments, about **0.1 % of uranium mass appears as fission energy of ~200 MeV**

→ **much bigger than any other exoenergetic nuclear reaction**

For **^{235}U** (total mean fission energy 202.5 MeV), typically

- ~**169 MeV** appears as the **kinetic energy of the daughter nuclei**,
-which fly apart at about 3% of the speed of light

- an **average of 2.5 <u>prompt</u> neutrons are emitted**, with a **mean kinetic energy per neutron of ~2 MeV** (total of 4.8 MeV); the **average number of neutrons emitted is called ν (order of 2-3)**

- **~7 MeV** are released in form of **prompt gamma ray photons**

Chemical reactions vs nuclear fission

$C+O_2 = CO_2 + Q (Q = 3.6\ eV)$,
$CH_4 + 2O_2 = CO_2 + 2H_2O + Q (Q = 9.22\ eV)$,
$^{235}U+n = F_1+F_2+ \nu n+Q (Q = 211.5 MeV)$.

Most of the kinetic energy released in the fission process **is converted to thermal energy**

Amount of energy and reaction products

Reaction product	Energy (%)	Range (cm)	Delay
Fission fragments	80	< 0.01	prompt
Fast neutrons	3	10-100	prompt
Gammas	4	100	prompt
Fission product β decay	4	few	delayed
Neutrinos	5	"∞"	delayed
Non fission reactions due to neutron capture	4	100	delayed

Physics: nuclear cross sections

Cross section: quantity that characterizes a nuclear reaction (elastic, inelastic scattering, etc.) connected to the range of the involved forces; **effective area of a nuclear target**

Here we will consider the **total cross section**, defined as follows:

Given a **flux** $$\frac{dN_{in}}{dSdt}$$

number of incident particles per unit surface and unit time on a single nucleus (target)

and given an **interaction rate** $$\frac{dN_{reac}}{dt}$$

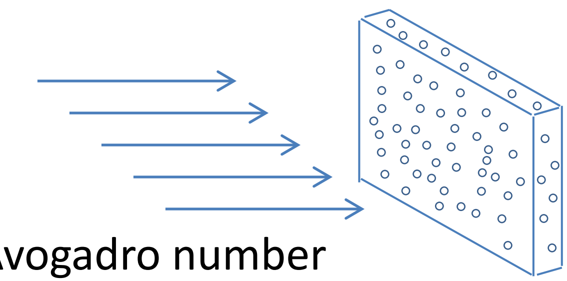
number of interacting particles (scattered or absorbed projectiles) per unit time, then

$$\sigma = \frac{\frac{dN_{reac}}{dt}}{\frac{dN_{in}}{dSdt}}$$

σ → physical dimensions of a surface

Nuclear cross sections

Macroscopic target comprising several nuclei with **density ρ (es. gr/cm3) and thickness x**, struck by a particle beam of intensity I (particles/sec) →



$$R = \frac{dN_{reac}}{dt} = I\,\frac{\rho\, x}{A} N_A \sigma$$

where A is the target atomic weight (es. in gr.) e N_A is the Avogadro number

$\frac{\rho}{A} N_A$ is the **number density of nuclei** in the target (i.e. number of nuclei per unit volume)

This is all valid for a <u>small thickness</u> x

For a target of <u>arbitrary thickness</u>, first divide it in thin slices of thickness dx →

$$dR = \frac{dN_{reac}}{dt} = I(x)\frac{\rho}{A} N_A \sigma \, dx$$

$$dI = -I(x)\frac{\rho}{A} N_A \sigma \, dx$$

$$I(x) = I(0)\exp(-\frac{\rho}{A} N_A \sigma \, x)$$

$\Rightarrow$ $\Sigma \equiv \frac{\rho}{A} N_A \sigma$ **Macroscopic cross section = prob.ty of interaction per unit length**

$1/\Sigma$ **= Mean free path** $\Sigma\, \mathrm{v} =$ **Frequency with which reactions occur, v= projectile speed**

Types of nuclear reactions

Nuclear scattering can be
- Elastic A+B→A+B
- Inelastic A+B→A+B*, A+B→A+C+D, A+B→C+D, etc.

Simplest type of nuclear reaction occurring in a nuclear reactor → **potential scattering**

neutrons scatter **elastically** off nuclear potential without ever penetrating the nucleus itself (similar to billiard balls collision)

By quantum mechanical arguments, it is possible to show that at low energies
the cross section for such a reaction essentially just geometrical cross section of nucleus

→ **rather flat energy dependence from about 1 eV up to the MeV range**

Another very relevant reaction mechanism is **neutron capture**
→ for heavy nuclei, addition of one more neutron can provide several MeV from binding energy
→ capture is **followed by gamma emission**, radiative capture, or **fission**

By quantum mechanical arguments, it is possible to show that at low energies, if the energy gained from the neutron capture is sufficient to produce the phenomenon of interest

→ **cross section follows a 1/v law**

Capture resonances

Capture process → neutron first absorbed by nucleus ${}_Z^{A}X$ → **compound nucleus** $\mathbf{{}_Z^{A+1}X}$

This **compound nucleus subsequently decays** by emitting an energetic particle

Compound nucleus formation occurs in many neutron-nuclear reactions of interest for reactor physics, including fission, radiative capture, and certain types of scattering.

Formation of a compound nucleus can proceed through a so-called **resonance reaction**

→CM energy of neutron+nucleus system + binding energy of the captured neutron match one of the energy levels in the compound nucleus

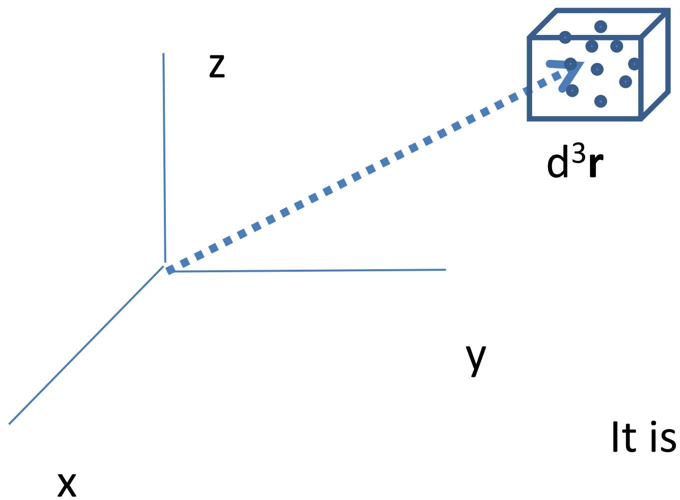
→This phenomenon is **indicated by sharp peaks in the capture cross section**

→ as a consequence, neutrons that "cross" a resonance when they scatter around and lose energy, can be more strongly absorbed by elements other than fuel

Fission itself can produce fission fragments with very strong radiative capture cross sections → they are called **neutron poisons**, e.g. ^{135}Xe ($\sigma \sim 2\text{x}10^6$ barns)

Neutron density and flux

Neutron density $\equiv n(\mathbf{r},E,t)$ [cm^{-3}] $\equiv$
expected number of neutrons with energy between E and E+dE, in the volume $d^3\mathbf{r}$ about $\mathbf{r}$, at a time t



Reaction density $\equiv R(\mathbf{r},E,t) \equiv$
Number of reactions in the volume $d^3\mathbf{r}$ about $\mathbf{r}$, at a time t, initiated by neutrons with energy between E and E+dE = $n(\mathbf{r},E,t)\,\Sigma v$

We give a special name to the quantity $n(\mathbf{r},E,t)v$
It is called the **neutron "flux" $\phi(r,E,t) \equiv n(r,E,t)v$ [$cm^{-2}\ s^{-1}$]**

Reaction density $\equiv$ number of reactions per unit volume $\equiv R(r,E,t) = \Sigma\,\phi$

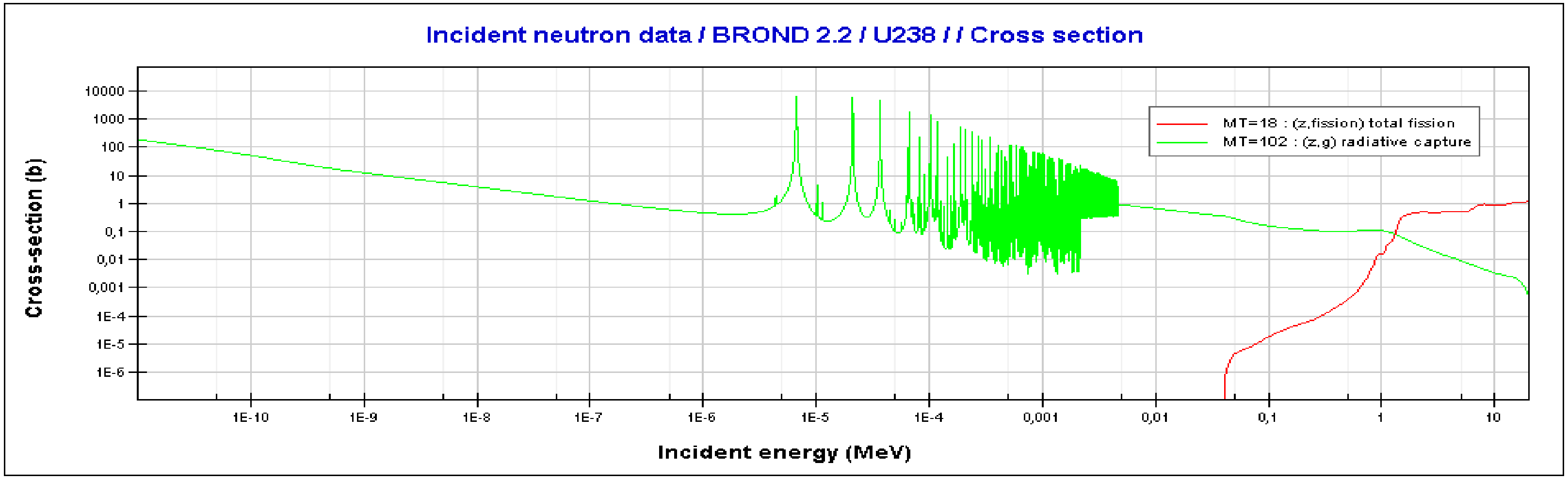
Suppose you've got a reactor with 1 GW thermal power = 10^9 Joule/sec
Assume each fission releases order of 200 MeV energy = 3.2×10^{-11} Joule
→ In the reactor the fission rate is about 3×10^{19} fissions/sec
→Almost 10^{20} neutrons/sec emitted, about 2×10^{20} neutrinos/sec
→ $\phi \sim 10^{14}$ neutrons $cm^{-2}\ s^{-1}$

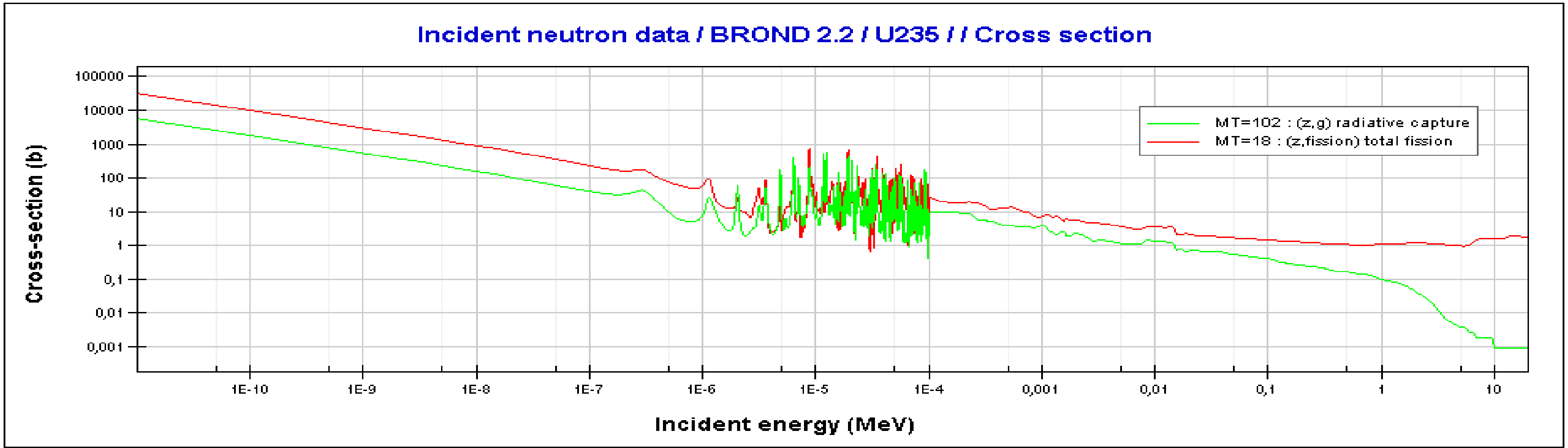
Nuclear cross sections

Since the nuclear radius is roughly 10^{-12} cm, the geometrical cross sectional area of the nucleus is roughly **10^{-24} cm^2 = 1 barn**

Hence we might expect that nuclear cross sections are of the order of 10^{-24} cm^2 ≡ 1 barn

However, quantum mechanical effects can make nuclear cross sections a lot bigger…





Fissile, fissionable, fertile isotopes

- Heavy nuclei with a high fission cross section at low (thermal) neutron energies are called **fissile** (e.g. ^{233}U, ^{235}U, ^{239}Pu,…)

- Those with a non-zero fission cross section only at higher neutron energies are called **fissionable** (e.g. ^{238}U,…)

- Those that can produce a fissile isotope via neutron radiative capture and β decay are called **fertile**, i.e. they can be used to **produce fuel** (e.g. ^{238}U,…)

- $n + {}^{238}U \rightarrow {}^{239}U + \gamma \rightarrow {}^{239}Np + \beta + \text{anti-}\nu \rightarrow \boxed{{}^{239}Pu} + \beta + \text{anti-}\nu$ fissile

- $n + {}^{232}Th \rightarrow {}^{233}Th + \gamma \rightarrow {}^{233}Pa + \beta + \text{anti-}\nu \rightarrow \boxed{{}^{233}U} + \beta + \text{anti-}\nu$

Natural Uranium → 0.7 % ^{235}U + 99.3 % ^{238}U

How much fuel ?

Suppose you've got a **reactor with 1 GW thermal power (1 GW_{th} → ~ 300 Mw_e) = 10^9 Joule/sec**
Assume each fission releases order of 200 MeV energy = 3.2×10^{-11} Joule
→In the reactor the fission rate is about 3×10^{19} fissions/sec

→ which means that e.g. 3×10^{19} (nuclei of ^{235}U)/sec disappear (actually a bit more because of radiative capture)
→ this is roughly **12 mg/sec of ^{235}U are "burnt" in the reactor**
→ for **1 year of operation at 80 % load factor** → consumption of about **300 Kg of ^{235}U**
→ in volume of pure metallic ^{235}U, this would be a cube of about 25 cm side

→ Just for comparison, the same amount of thermal power can be obtained by burning about 27 m^3/sec of methane gas (i.e. about 700 million m^3 per year), or by burning 27 l/sec of oil (i.e. about 700 million liters per year), or by burning 42 Kg/sec of coal (i.e. about 1 million metric tons per year).

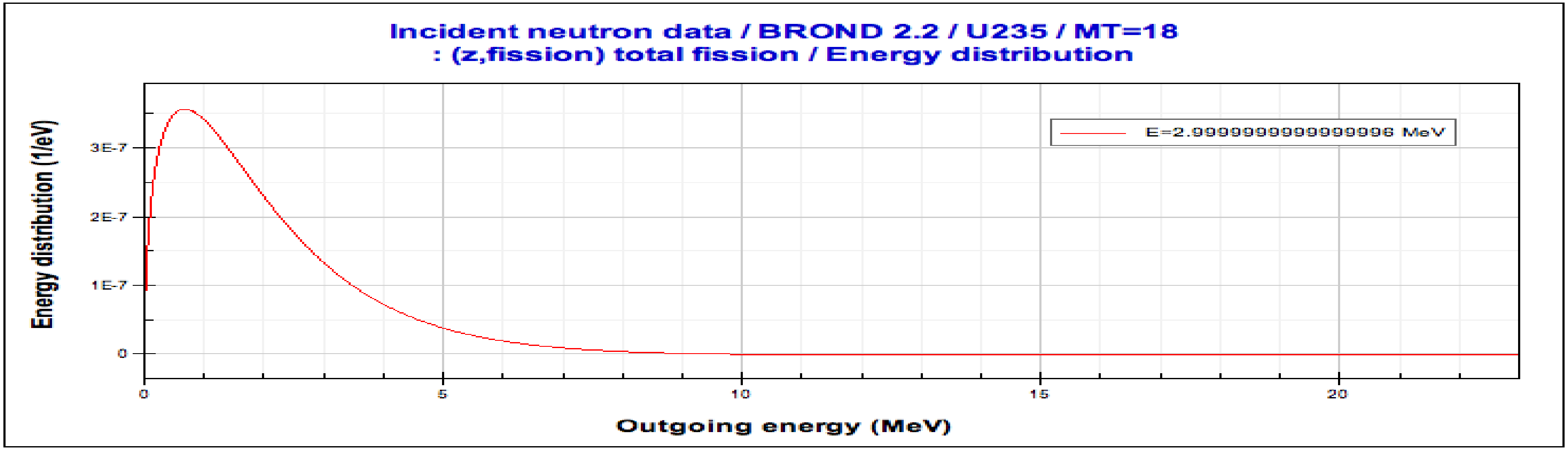
For a thermal reactor (see later) loaded with mixed UO_2 fuel (density about 11 gr/cm^3) comprising 4 % ^{235}U and 96 % ^{238}U, this corresponds to 8500 Kg of fuel → 0.8 m^3

In practice, there has to be much more as the chain reaction needs the presence of fissile nuclei at all times → the reactor has to be critical at all times
However, ^{235}U consumption is partly compensated by Plutonium (^{239}Pu) breeding

Inventories at loading and discharge of a 1GWe PWR

Nuclides	Initial Load(kg)	Discharge inventory(kg)
^{235}U	954	280
^{236}U		111
^{238}U	26328	25655
U total	**27282**	**26047**
^{239}Pu		156
Pu total		**266**
Minor Actinides		**20**
^{90}Sr		13
^{137}Cs		30
Long Lived FP		**63**
FP total		946
Total mass	**27282**	**27279**

Fission spectrum, fast and slow neutrons



It is customary to adopt the following classification:

- **slow neutrons**: those with kinetic energy $T_n < 1$ eV
- in particular **thermal neutrons** have T_n around 0.025 eV or 25 meV (the value of kT, where k is the Boltzmann constant and T is the temperature
- **epithermal neutrons**: 1 eV < T_n < 100 keV (0.1 MeV)
- **fast neutrons**: 0.1 MeV < T_n < 20 MeV

Obviously neutrons in general can have energies above 20 MeV but this is an extreme limit in reactor physics (e.g. neutrons from D+T fusion have 14 MeV fixed energy)

Slowing down neutrons (moderation)

It is easy to show in non-relativistic kinematics that **after a scattering off a nucleus with mass number A**, the kinetic energy of the neutron changes according to the ratio

$$\frac{T_n'}{T_n} = \frac{m_n^2 + m_A^2 + 2m_n m_A cos\theta_{CM}}{(m_n + m_A)^2}$$

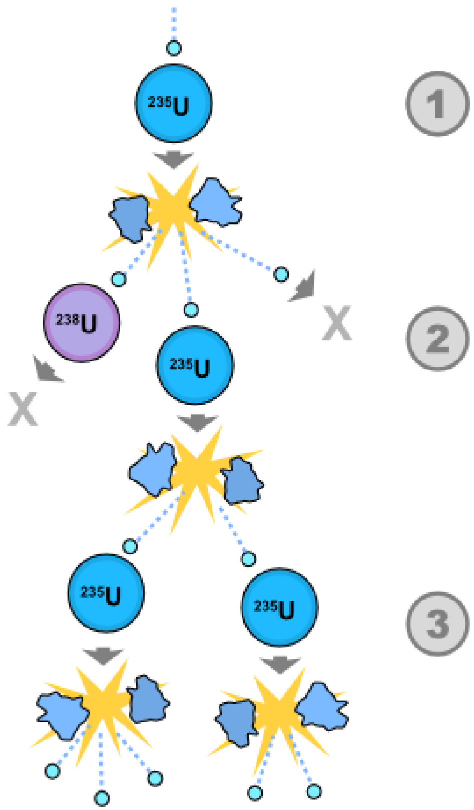
Assuming an isotropic CM cross section that does not depend on $\cos\theta_{CM}$, the corresponding term averages out to zero, so that we can write <u>on average</u>

$$\frac{T_n'}{T_n} = \frac{m_n^2 + m_A^2}{(m_n + m_A)^2}$$ ➔ Assuming $M_A \cong Am_n$ ➔ $$\frac{T_n'}{T_n} = \frac{1 + A^2}{(1 + A)^2}$$

For a **heavy nucleus A>>1** → $\mathbf{T_n' \cong T_n}$ or in other words, the neutron has to undergo many collisions in order to significantly lose energy.
Consider instead the case **A=1** → (target containing hydrogen, i.e. protons as nuclei) $\mathbf{T_n' = T_n/2}$ i.e. on average a neutron will lose half of its energy at each collision and therefore few collisions are sufficient to rapidly decrease its energy

➔ **Moderators = light materials containing hydrogen = water, paraffine or graphite**

The chain reaction and the critical reactor



The chain reaction:
- must not diverge (more and more fissions at each "generation")
- must not die away (less and less fissions at each generation)

→<u>precisely one neutron from each fission has to induce another fission event</u>

The <u>remaining fission neutrons</u> will then either be
- <u>absorbed by radiative</u> capture or
- <u>will leak out</u> from the system

Suppose we can count the number of neutrons in one generation and in the next one
Then

$$k \equiv \frac{\text{number of neutrons in one generation}}{\text{number of neutrons in the preceding generation}}$$

- The condition **k=1** corresponds to a **critical reactor**
- **k>1** is a **supercritical reactor** (fission reactions tend to diverge)
- **k<1** is a **subcritical reactor** (fission reactions tend to die away)

“Simple-minded” reactor kinetics

$$\frac{dn(t)}{dt} = P(t) - L(t)$$

n(t)=neutron population at time t
P(t)= neutron production at time t (mainly as fission products)
L(t)=neutron loss (fission+capture+leakage) at time t

All are functions of time as reactor evolves over time

Alternative definition $k \equiv \frac{P(t)}{L(t)}$ $\quad$ $\text{Neutron lifetime} \equiv \tau \equiv \frac{n(t)}{L(t)}$

$$\frac{dn(t)}{dt} = \frac{k-1}{\tau} n(t)$$

Let’s assume k and l are time independent (not true…)

$$n(t) = n_0(t) \exp\left(\frac{k-1}{\tau} t\right)$$

- **k=1 → steady state →** critical reactor
- **k>1 → increase →** supercritical
- **k<1 → decrease →** subcritical

$$\text{Time constant} \equiv \text{T} \equiv \text{Reactor period} \equiv \frac{l}{k-1}$$

Delayed neutrons: crucial for reactor control

Typical neutron lifetime in a thermal power reactor ~ 10^{-4} sec

If k=1.001 ➡ **T=0.1 sec** ➡ **power will increase by 2.7 in 0.1 sec !!**

Actually, **we neglected** the very small amount (< 1 %) of **delayed neutrons**

Emitted by fragments after fission on **time scale from ms to sec**

The trick is to **make the reactor critical thanks to that small fraction of neutrons**

→ Delayed neutrons **dominate the reactor response time** making it much longer

➡ Reactor control manageable by control rods

The 4-factor formula

Multiplication can be written as

$$k = \frac{N_2}{N_1} = \eta \; f \; P_{NL}$$

N_1, N_2 = number of neutrons in two subsequent generations

η = average number of neutrons produced per neutron absorbed in the fuel

where

$$\eta = \nu \frac{\sigma_f^F}{\sigma_a^F}$$

$\sigma_f^F =$ Fission cross section in the fuel

$\sigma_a^F =$ Absorption cross section in the fuel

$\nu =$ Average number of emitted neutrons

f = conditional probability that, if neutron will be absorbed, it will be absorbed in fuel

P_{NL}= probability of non-leakage

Infinite reactor → P_{NL}= 1 ⟹ $k_\infty = \eta \; f$

This is a <u>property of the material</u>, not of the geometry

The 4- and 6-factor formula

We take into account the energy dependence of the cross section via additional factors

$$\varepsilon = \frac{\text{Total number of fission neutrons (from both fast and thermal fissions)}}{\text{Total number of fission neutrons from thermal fissions}} > 1$$

p = fraction of fission neutrons that survive moderation without being absorbed

Infinite reactor → P_{NL}= 1 ⟹ $$k_\infty = \eta\, f\, p\, \varepsilon$$ 4-factor formula

$$P_{NL} = P_{FNL}\, P_{TNL}$$

Finite reactor and energy dependence: probability of non-leakage for fast and thermal neutrons, separately

$$k_{eff} = \eta\, f\, p\, \varepsilon\, P_{FNL}\, P_{TNL}$$

6-factor formula

"effective" → we are not considering an infinite, homogeneous medium

Clearly $k_\infty > k_{eff}$

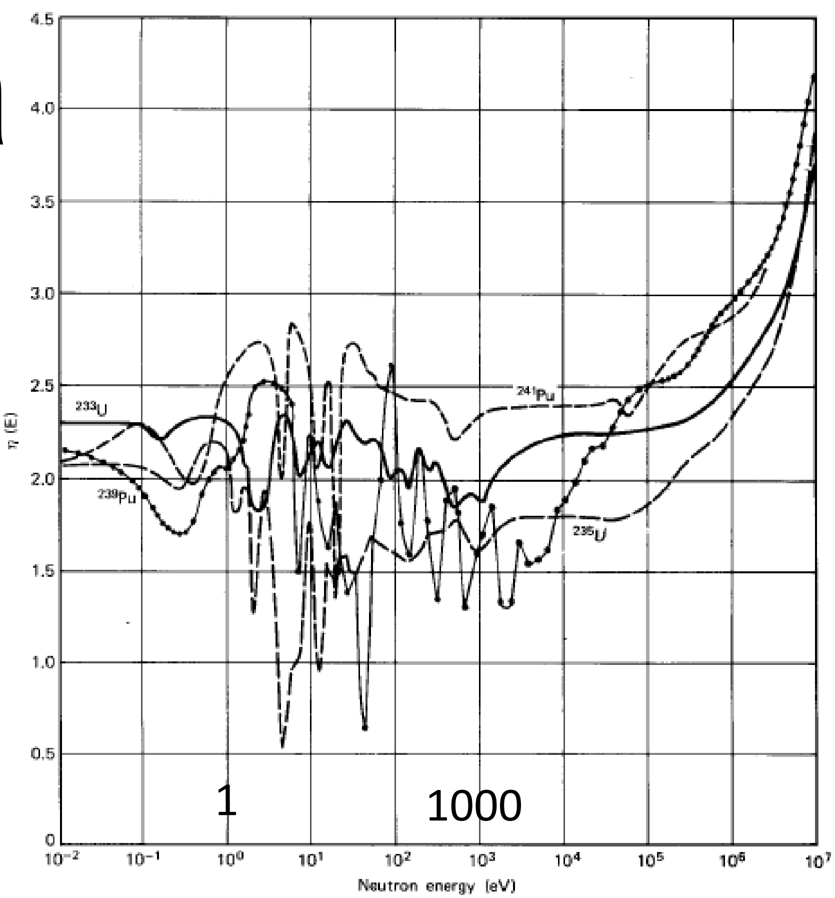
P_{FNL}, P_{TNL} must not be too < 1→ a <u>reflector</u> (e.g. graphite, Cu, Pb) surrounds the core

Simple considerations

$$k = \frac{N_2}{N_1} = \eta \; f \; P_{NL}$$

$f < 1$, $P_{NL} < 1$ → We must have η significantly > 1

η



Neutron energy (eV)

Which is indeed the case (on average):
variation of η with energy for ^{233}U, ^{235}U, ^{239}Pu, ^{241}Pu

Neutron propagation in more detail

The exact description of neutron transport in a reactor is given by the **Boltzmann equation**

$$\frac{d}{dt}\iiint n(\vec{r},v,t)\,d^3r =$$

=[(entering-exiting)+(created-absorbed)+(inscattered-outscattered)]

Very complicated equation
→ assume that all neutrons have the same velocity (**one group** approximation)
→ make assumption on form of vector flux (neutron vector current)
→ simplified to a **diffusion equation**

$$\frac{\partial\phi(\vec{r},t)}{\mathrm{v}\partial t} = D\nabla^2\phi(\vec{r},t) + \phi(\vec{r},t)\left[\nu\Sigma_f(\vec{r}) - \Sigma_a(\vec{r})\right] + S(\vec{r},t)$$

S = external neutron source
(very important for subcritical systems, see later)

$$D = \frac{\Sigma_s}{3\Sigma_T^2}$$

Σ_S=scattering macroscopic cross section
Σ_T=total macroscopic cross section

More elaborate reactor kinetics

For an infinite reactor containing only fuel

$$\frac{\partial \phi(\vec{r},t)}{\mathrm{v}\partial t} = D\nabla^2\phi(\vec{r},t) + \phi(\vec{r},t)\Sigma_a(\vec{r})(k_\infty - 1) + S(\vec{r},t)$$

Homogeneous medium → $\nabla^2\phi(\vec{r},t) = 0$

$$\Rightarrow \quad \phi(t) = \phi(0)\exp\left[v(k_\infty - 1)\Sigma_a(\vec{r})t\right]$$

Neutron lifetime → $\tau = \dfrac{\text{absorption mean free path}}{\text{velocity}} = \dfrac{\Lambda_a}{\mathrm{v}} = \dfrac{1}{\Sigma_a \mathrm{v}}$

$$\Rightarrow \quad \phi(t) = \phi(0)\exp\left[\frac{(k_\infty - 1)}{\tau}t\right]$$

which was our “simple-minded” reactor kinetics…

How to keep the chain reaction going in a subcritical system

$$k_\infty < 1, S(t) = S_0 > 0$$

→ The system is subcritical, but an external source supplies neutrons

Stationary state → $$\frac{\partial \phi(t)}{\partial t} = 0$$

$$\Rightarrow \quad \phi = \frac{S_0}{(1-k_\infty)\Sigma_a}$$

Reaction rate → $$R = \Sigma_a \phi = \frac{S_0}{(1-k_\infty)}$$

Critical reactor control: delayed neutrons

Reactivity → $$\rho = \frac{k_{eff} - 1}{k_{eff}}$$

A fraction β of the neutrons are emitted much later by the fission fragments, following β decay to a highly excited state of the final nucleus
→**delayed neutrons**

For instance, for ^{235}U, β = 0.64 %, **mean decay time** T_d=8.8 sec

Therefore in practice, a reactor is designed such to have $k_{eff} \cong 1-\beta$ without considering delayed neutrons, while it becomes $k_{eff} \cong 1$ when adding their contribution

A reactivity variation equal to β is called **a 1 $ insertion**

Time constant in the exponential increase/decrease of the flux or power → βT_d~20-60 ms
→ manageable with in-out motion of absorptive control rods

(Not the neutron lifetime which ranges from 10^{-7} to 10^{-4} sec from fast to thermal reactors)

Neutron population and reactor classes

Neutron energy range in a reactor
Neutrons slow down through collisions with nuclei (in particular with light nuclei)
→ **Energies** go from 10 MeV (usually max energy of fission neutrons) down to as low as 10^{-3} eV

Neutron cross sections have a strong dependence on neutron energy → generally, they decrease with increasing energies, in particular absorption cross sections such as capture or fission

➔ it is easiest to maintain a fission chain reaction using slow neutrons
➔ Hence **most nuclear reactors until now (Gen. I to III+) use low mass number materials such as water or graphite to slow down or moderate the fast fission neutrons**
➔ neutrons slow down to energies comparable to the thermal energies of the nuclei in the reactor core
➔ **Thermal reactor:** average neutron energy comparable to thermal energies
➔ They require the **minimum amount of fissile material** for fueling

As an example, a **Light Water Reactor (LWR)** can start with 3 % ^{235}U + 97 % ^{238}U
Burn-up of ^{235}U is compensated by **breeding** of ^{239}Pu
After 1 year, the core may contain 1 % ^{235}U + 1 % ^{239}Pu

Neutron population and reactor classes

However
the number of neutrons emitted per neutron absorbed in the fuel is largest for fast neutrons

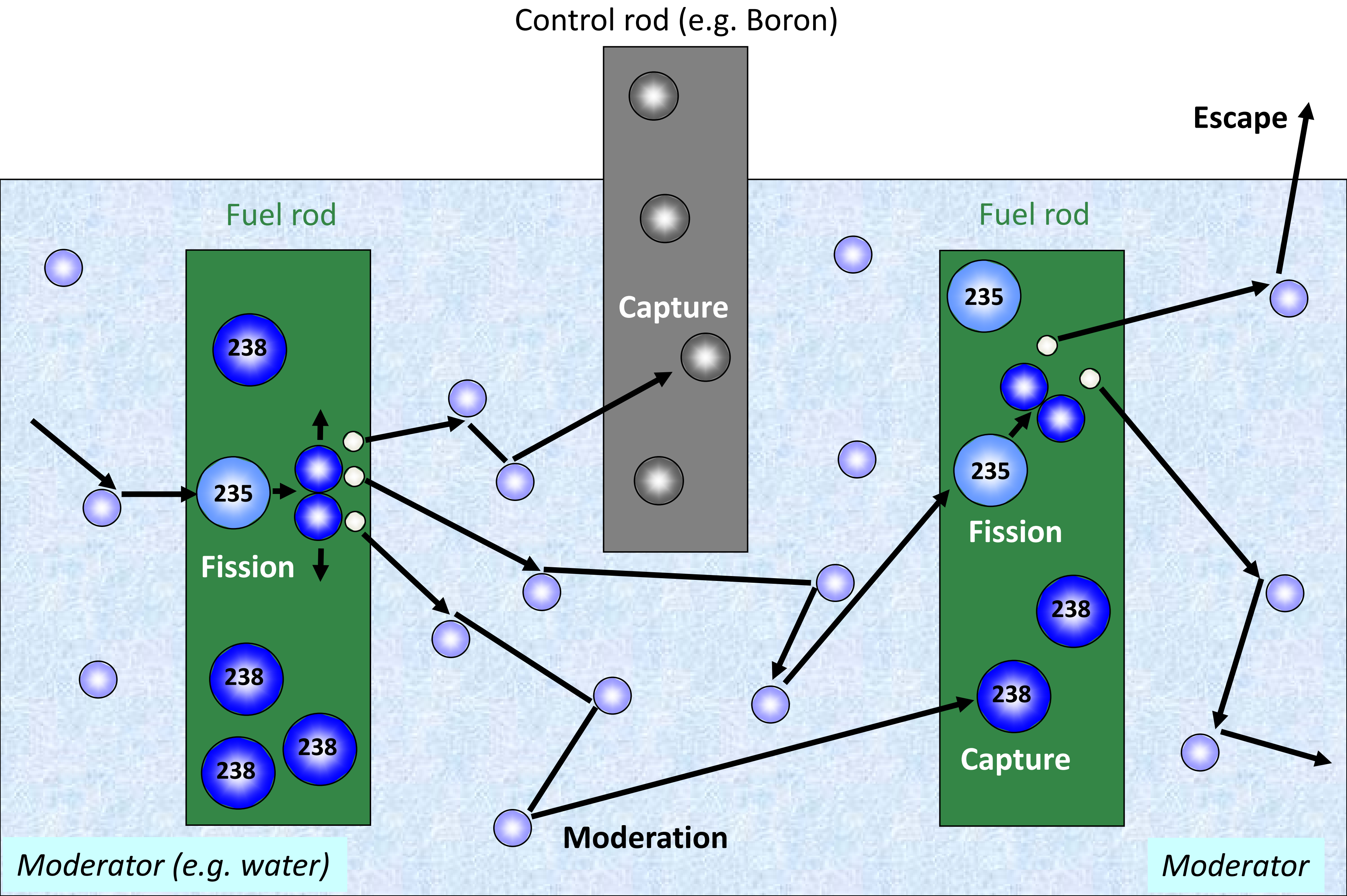
→ one can use the "extra" neutrons to **convert or breed new fuel**.
→ but σ_f **is smaller**
→ **need much more fuel** to sustain the chain reaction
→ to keep the neutron energy high, **only high mass-number materials in the core**
→ **Fast reactor:** average neutron energies above 100 keV

$$\text{Conversion ratio CR} = \frac{\text{Average rate of fissile atom production}}{\text{Average rate of fissile atom consumption}}$$

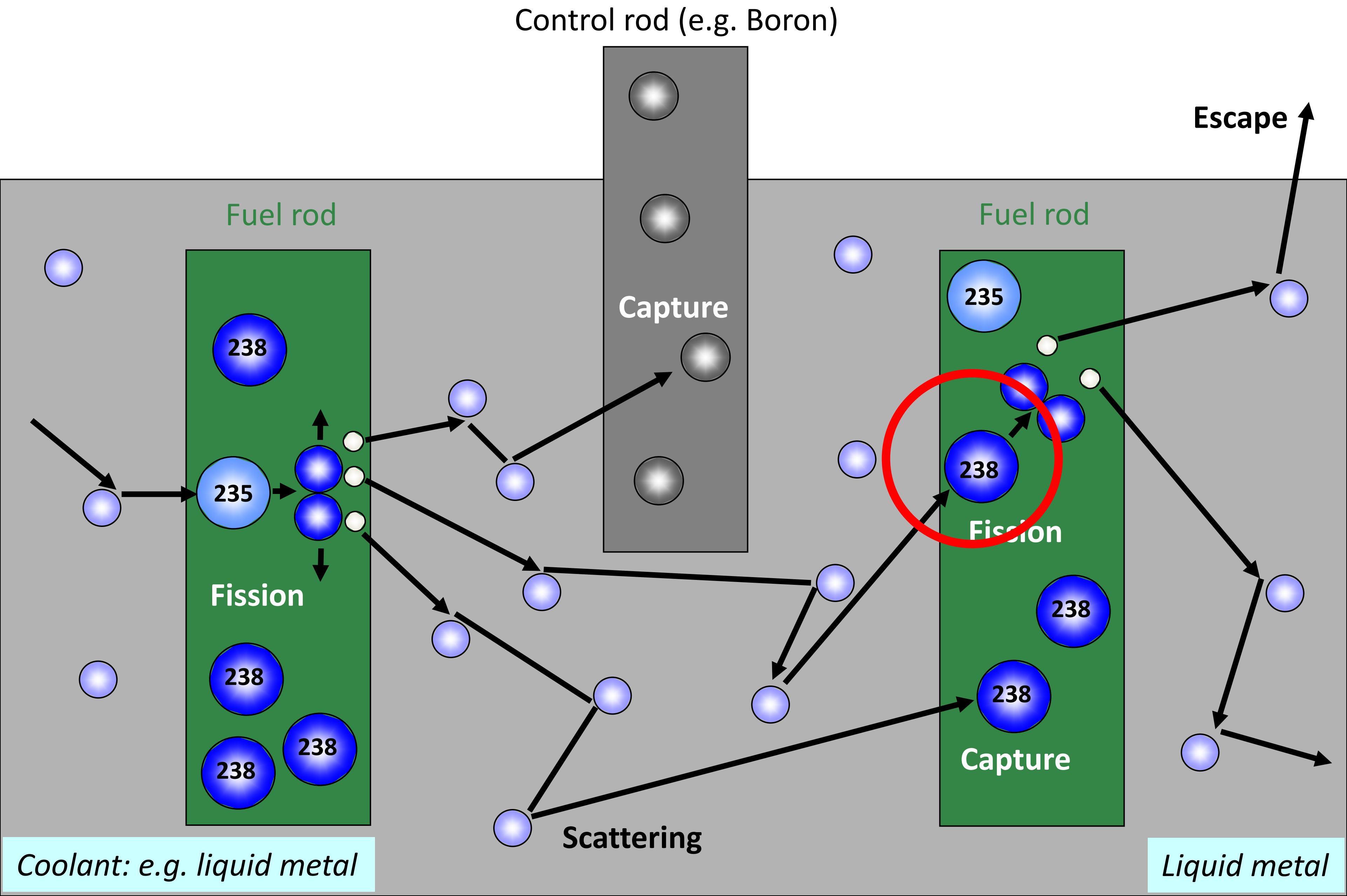
If CR>1 it is called “breeding ratio” BR
For CR/BR > 1 we must have η>2 as > 1 neutron is needed to keep k=1
and the other is needed for the production of new fissile nuclei

η is definitely greater than 2 for T_n > about 100 keV → “**Fast Breeder” concept**

The thermal reactor



The fast reactor



Nuclear reactor zoo

Most current reactors
→ ordinary water serves as both coolant and moderating material in the reactor

There are two major types of Light Water Reactors (LWR):
1) pressurized water reactors (PWR)
2) boiling water reactors (BWR)

In a **PWR** the primary coolant is water maintained under very high pressure (~150 bar)
→ high coolant temperatures without steam formation within the reactor

Heat transported out of the reactor core by the primary coolant is then transferred to a secondary loop containing the "working fluid" by a steam generator
Such systems typically contain from two to four primary coolant loops and associated steam generators.

Nuclear reactor zoo

In a **BWR**, the primary coolant water is maintained at lower pressure (~ 70 bar)
→ appreciable boiling and steam within the reactor core itself
→ the reactor itself serves as the steam generator → no secondary loop and heat exchanger

In both PWR and BWR, the nuclear reactor itself and the primary coolant are contained in a **large steel pressure vessel** designed to accomodate the high pressures and temperatures

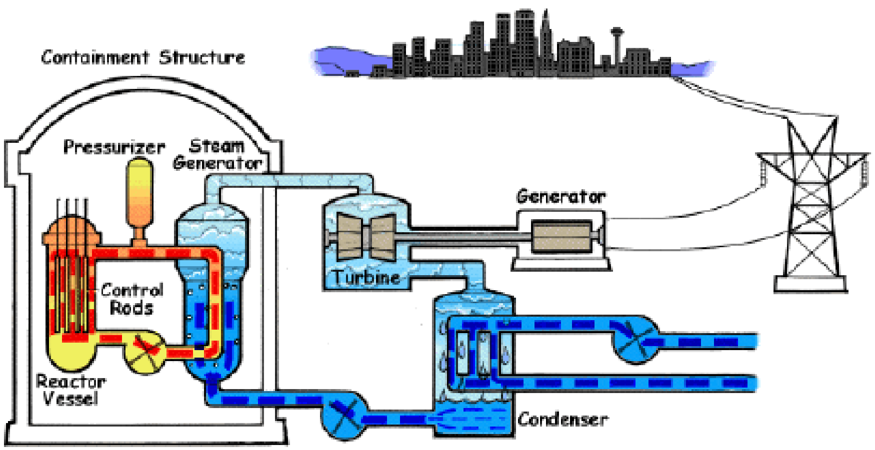
In a PWR → vessel has thick steel walls due to the higher pressure
In a BWR → pressure vessel not so thick, but larger → contains both nuclear reactor and steam moisture-separating equipment

Heavy water (D_2O) reactor
→ deuteron has lower neutron capture capture cross section with respect to hydrogen
→ low-enrichment uranium fuels (including natural uranium)
→ Developed in Canada in the CANDU (CANadian Deuterium Uranium) series of power reactors and in the UK as Steam Generating Heavy Water Reactors (SGHWR).

Gas-based reactors
→ the early MAGNOX reactors developed in the UK: low-pressure CO_2 as coolant
→ High-Temperature Gas-cooled Reactor (HTGR, USA): high-pressure helium as coolant
→ Pebble-bed concept
→ Advanced Gas Reactors (AGR, Germany and UK)

Reactor classification

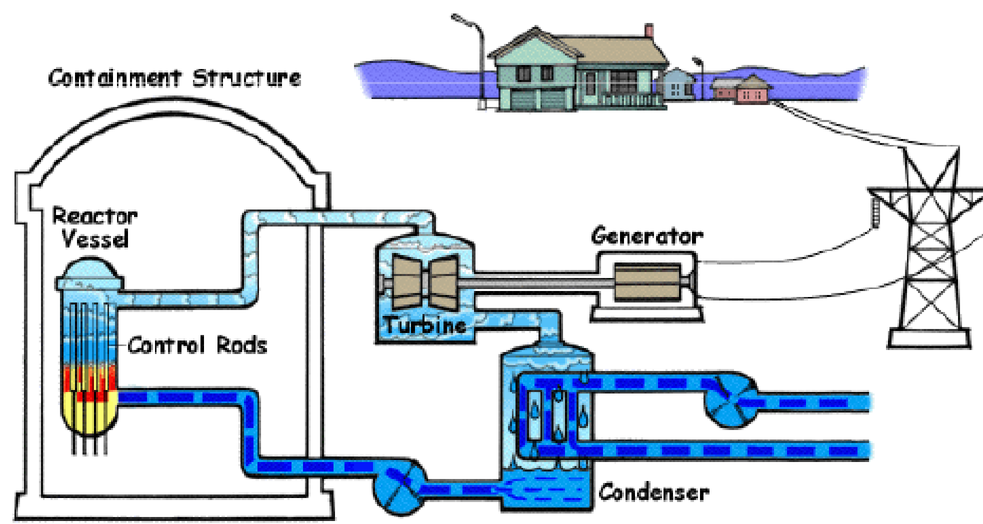
INDIRECT CYCLE REACTOR: thermal fluid → takes away the heat from the core → transfers heat through a heat exchanger/steam generator to a secondary thermal fluid that drives a turbine



PWR

DIRECT CYCLE REACTOR : thermal fluid → takes away the heat from the core and directly drives a turbine

BWR



Moderator/coolant classification

THERMAL REACTORS

Generally **classified based on the moderator:**

Graphite reactors - Magnox, AGR, HTGR, RBMK
LWR (Light Water Reactor) – PWR, BWR, VVER
HWR (Heavy Water Reactor) – CANDU, PHWR

or **based on the thermal fluid:**

Gas-cooled reactors - Magnox, AGR, HTGR
Water-cooled reactors (light/heavy) – LWR, HWR, RBMK
Based on the cycle:
Pressurized (indirect cycle) – PWR, PHWR
Boiling (direct cycle) - BWR

Decay heat

Decay heat is the heat released as a result of radioactive decay: the energy of the alpha, beta or gamma radiation is converted into atomic motion

In nuclear reactors **decay of the short-lived radioisotopes created in fission continues at high power**, for a time after shut down

Heat production comes **mostly from β decay** of fission products

A practical approximation is given by the formula

$$\frac{P}{P_0} = 6.6 \cdot 10^{-2} \left[\frac{1}{(\tau - \tau_s)^{0.2}} - \frac{1}{\tau^{0.2}} \right]$$

Where P is the decay power, P_0 is the reactor power before shutdown, τ is the time since reactor startup and τ_s is the time of reactor shutdown measured from the time of startup (in seconds)

→ **At shutdown, the heat power is about 6.5 % (~200 MW for a 1 GWe reactor)**
Sufficient to melt the core….

About 1 hour after shutdown, the decay heat will be about 1.5% of the previous core power. After a day, the decay heat falls to 0.4%, and after a week it will be only 0.2%
Spent fuel rods are kept for long time in a spent fuel pool of water, before being further processed.

Removal of decay heat very important → Fukushima…

Past and future