Nuclear physics for nuclear reactors

Class objective

- In this class, we will explain, starting from the microscopic nuclear reactions and processes, how a nuclear reactor operates.
- The aspects of safety and life-cycle of the fuel will be examined and we will highlight the limitations of the current nuclear power plant designs.
- This will lead us to look at the designs of next generation reactors and how they can meet the nuclear energy challenges.
- Obviously, we will only glance over the topics as we have very little time compared to the complexity of the topic.
- I will present you here a short version of a 10 hours class I give in the M2 here.

Central equation for nuclear power

Today, the central nuclear reaction used in nuclear power generation is:

$$\mathbf{n + {}^{235}U \rightarrow \mathit{x}n + heat + Fission\ Fragments}$$

Today, the central nuclear reaction used in nuclear power generation is:

$$\boxed{\mathbf{n} + {}^{235}\mathbf{U}} \;\mathbf{\text{-->}}\; \boxed{x\mathbf{n}} + \boxed{\mathbf{heat}} + \boxed{\mathbf{Fission\ Fragments}}$$

This class will follow the different terms of the equation.

n + ^{235}U

The entrance channel for the reaction is a neutron (n) and a nucleus of Uranium 235.

The neutron

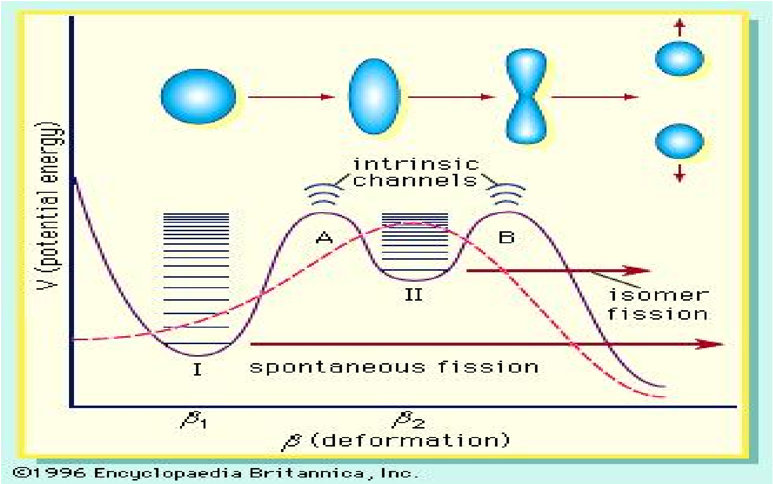
- Has a mass slightly above the one of the proton: $m_{neutron}=1.675\times10^{-27}$ kg=939.57 MeV=1.0087 u
- Unstable when free with a life time of τ=881.5 seconds
- or approximately 15 minutes.
- Has no electric charge (q=0) and a spin of 1/2 (i.e. it's a fermion).
- It's magnetic moment is $\mu=-1.9\mu_N$, and it's electric moment is expected to be zero, verified to be $<2.9\times10^{-26}$ e.cm.

Uranium 235

- Uranium is a naturally occurring element (Z=92),
- 5 isotopes found in nature (233, 234, 235, 236, 238),
- all of them unstable with lifetime ranging from 10^5 to 10^9 years.
- The most abundant are 238 (99.274 %) and 235 (0.720 %).

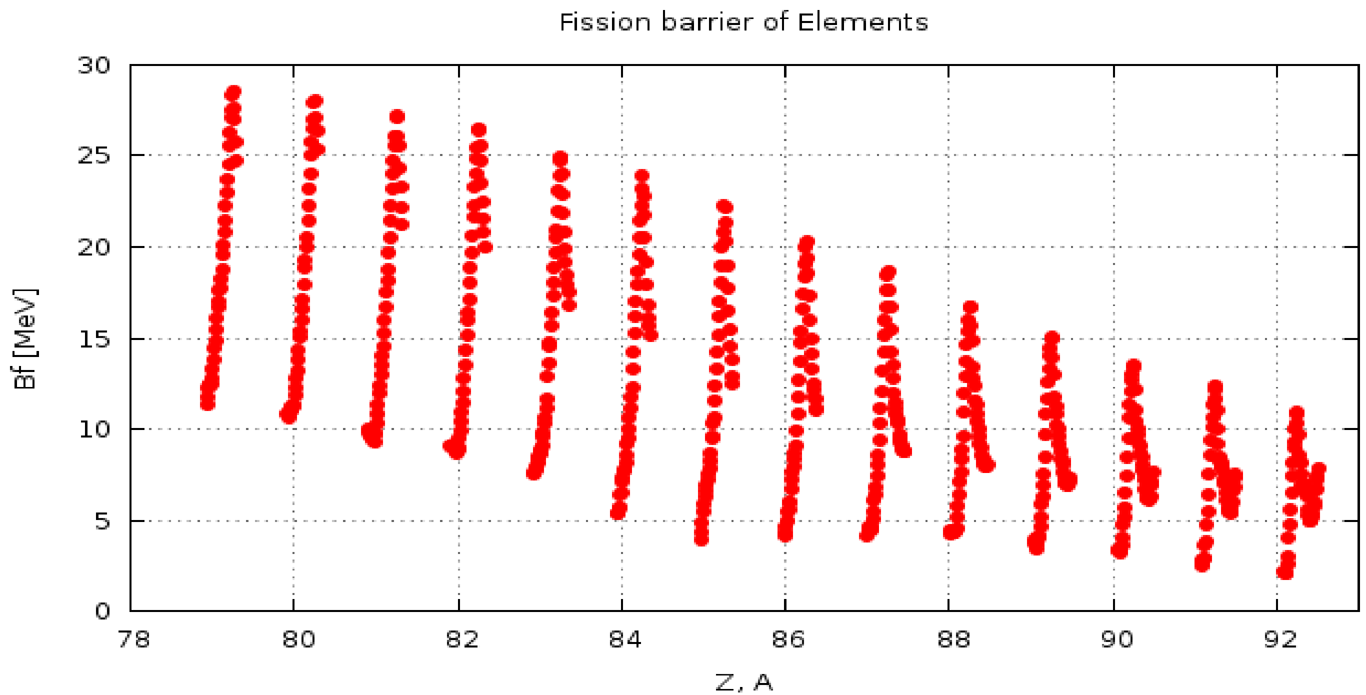
Neutron induced fission

- Fission is governed by a parameter called the fission barrier (B_f).
- The lower B_f, the easier it is to make a nucleus fission.
- The fission process is similar to have fragment go thru a potential energy barrier:



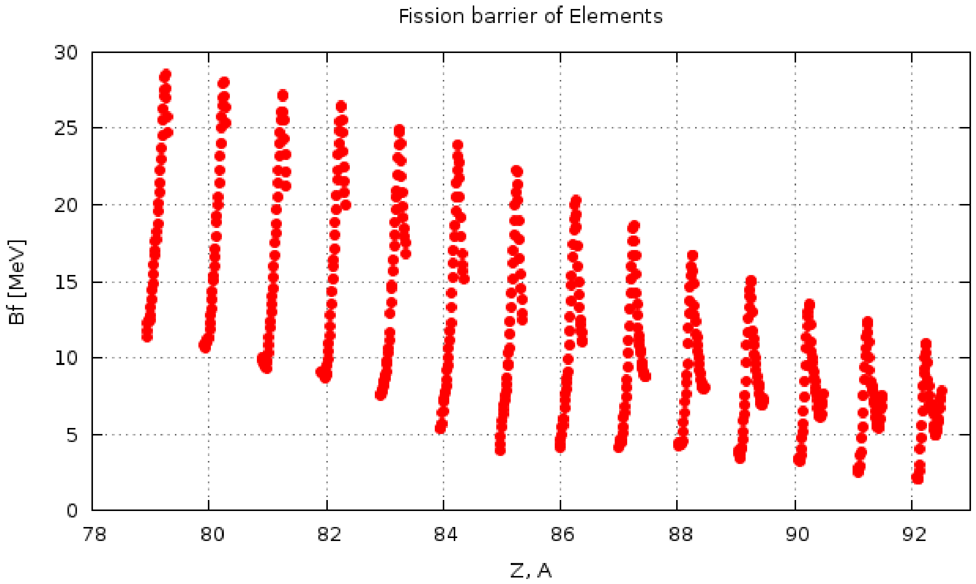
Neutron induced fission

- To get the most out of fission, we are looking to find the element that is easier to split with this process. So we investigate the fission barrier of elements:



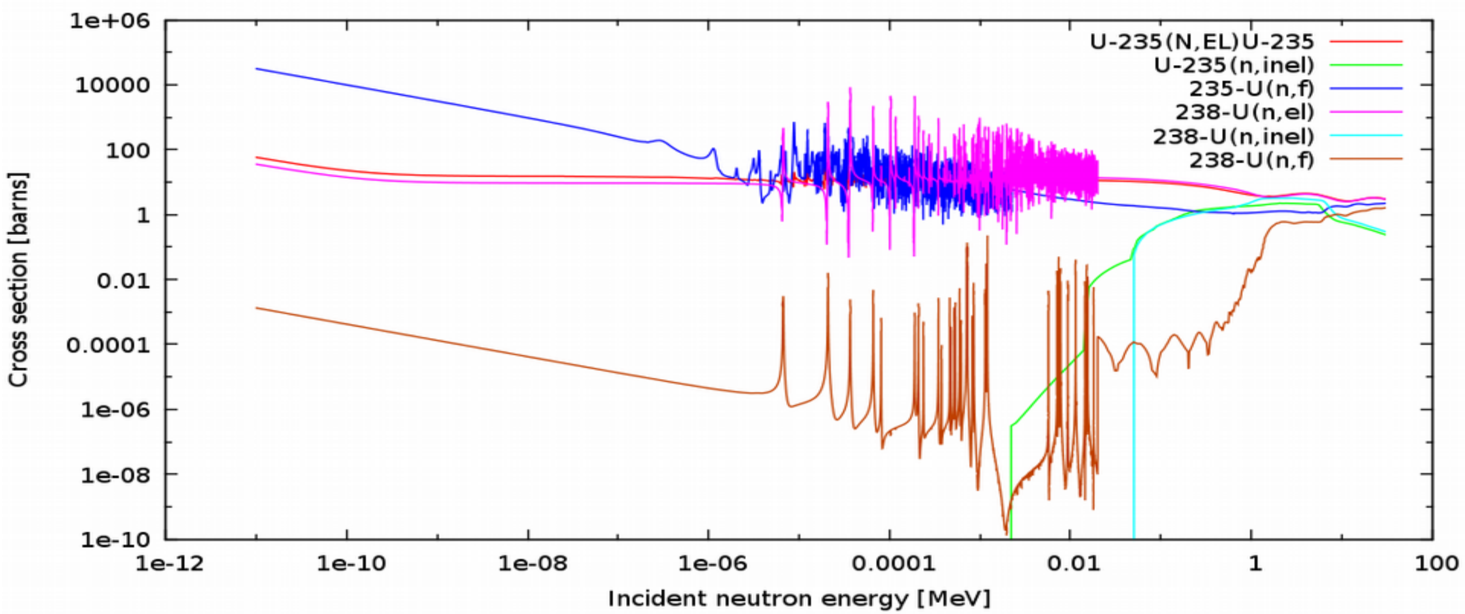
Neutron induced fission

- We notice that Uranium is the best element.
- Heavier elements have even smaller fission barrier, but are not found in nature).
- In particular, for the two isotopes we can found in nature:
- $B_f(^{235}U) = 4.87$ MeV
- $B_f(^{238}U) = 5.63$ MeV



Neutron induced fission

- Different isotopes react differently with an incoming neutron and several outcomes are possible
- We want to maximize the fission process because that is the one that release energy.



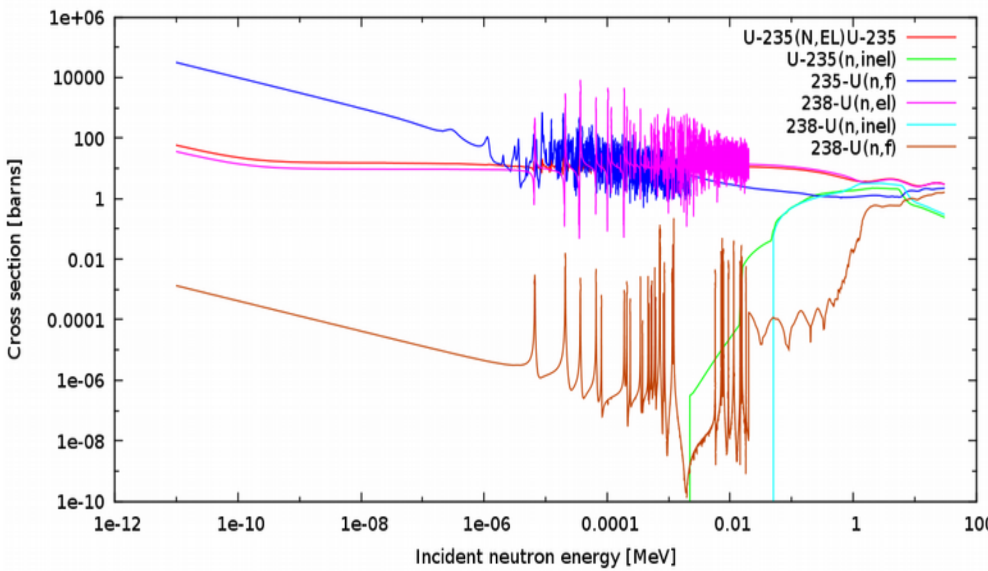
- The low energy neutrons are great: they fission ^{235}U without affecting other channels (or much less).
- This is because the capture of a low energy neutron by ^{235}U is very favorable and leads to an internal excitation energy for the resulting ^{236}U above the fission barrier:

- $Q_{neutroncapture}(^{235}U) = 6.546$ MeV,

  $B_f{}^{236}U = 5.03$ MeV

- $Q_{neutroncapture}(^{238}U) = 4.806$ MeV,

  $B_f{}^{239}U = 6.21$ MeV



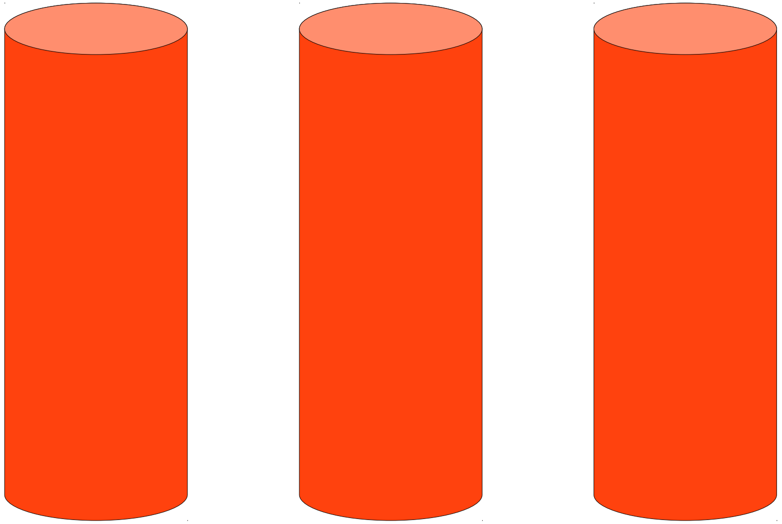
^{235}U enrichement

- Remember, the natural abundance of ^{235}U is 0.7 %.
- To really use this isotope in fuel, one will need to enrich the Uranium in 235.
- Typical enrichment for power generating fuel is 3-6 %.

- (Higher enrichment is possible, for research or weapon. But it has to stay limited: too much ^{235}U and the material will get critical out of control)

Reactor building blocks

- First building block of the reactor: 235 Uranium enriched material
- Usually as Uranium-Oxyde (UOx)

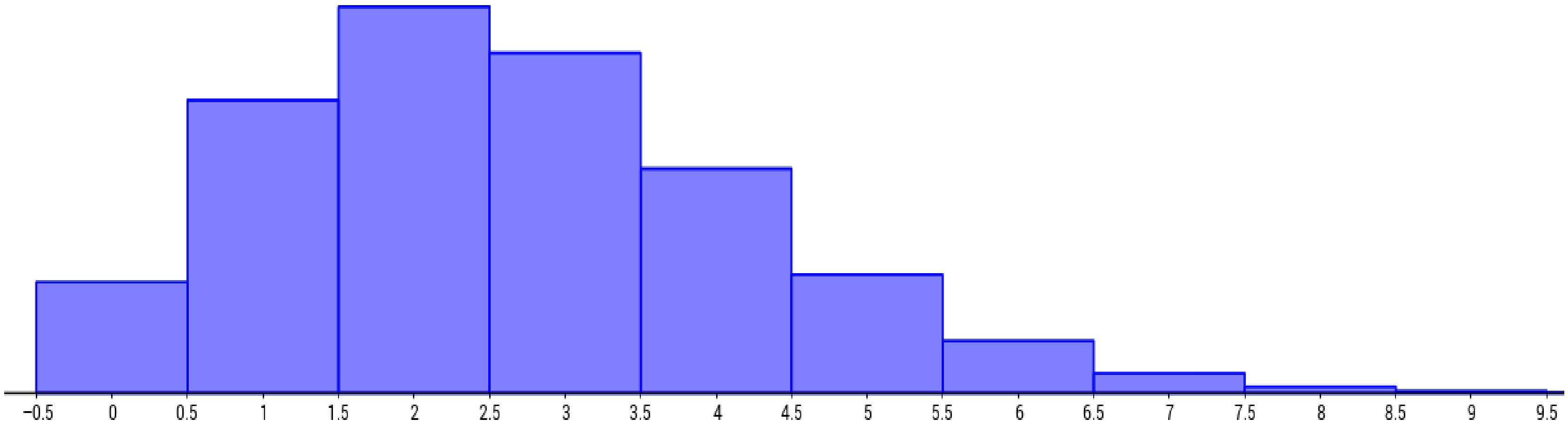


$$\boxed{\mathbf{n + {}^{235}U}} \mathbf{->} \boxed{\boldsymbol{x}\mathbf{n}} \mathbf{+} \boxed{\mathbf{heat}} \mathbf{+} \boxed{\mathbf{Fission\ Fragments}}$$

- The neutrons released in the fission process are key for the chain reaction used in nuclear power.
- Indeed, the goal is to maintain a constant rate of fission reaction, to release energy in a constant manner.
- As **1** neutron is used to induce the reaction, at least 1 neutron has to be produced in the out channel.

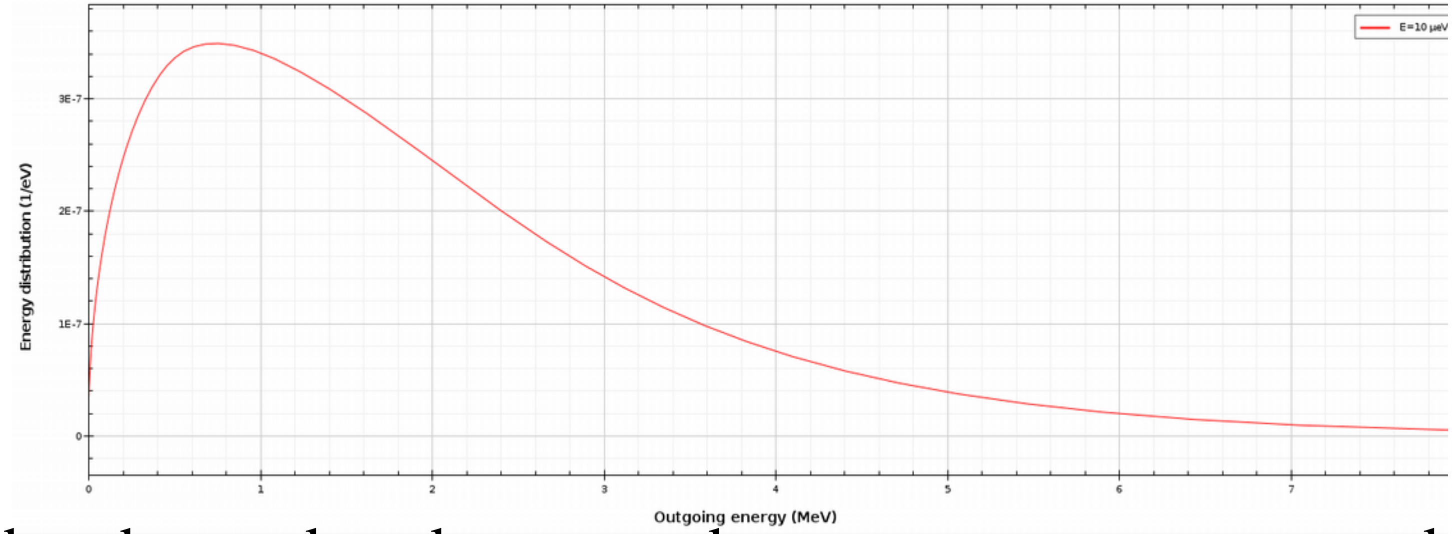
x n

- The neutron multiplicity *x* is not fixed.
- Because fission is a statistical process, the number of neutrons produced varies around an average value noted $\bar{\nu} = 2.64$



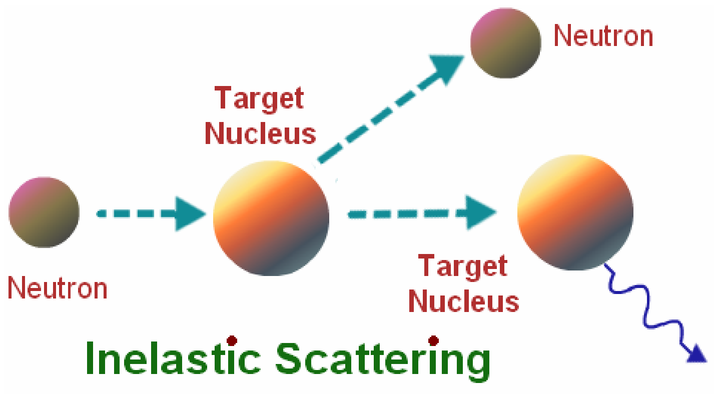
Fission neutrons energy spectrum

- The produced neutron also have a distribution of energy, that is close in shape to a Maxwellian.



We see that the produced neutrons have a mean energy around 1 MeV. But the neutron used to induce fission are slow (E_n of the order of meV and below). We need to ***slow*** down the neutrons, other wise the chain reaction can't sustain itself.

- By inelastic scattering on heavy elements:



- By elastic scattering on light elements

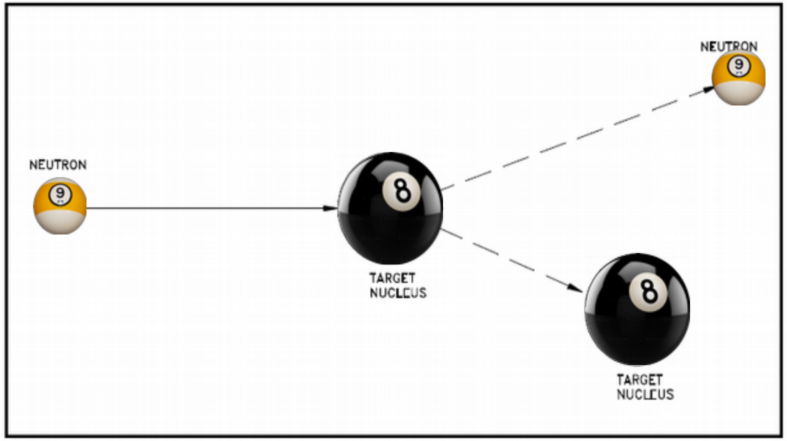
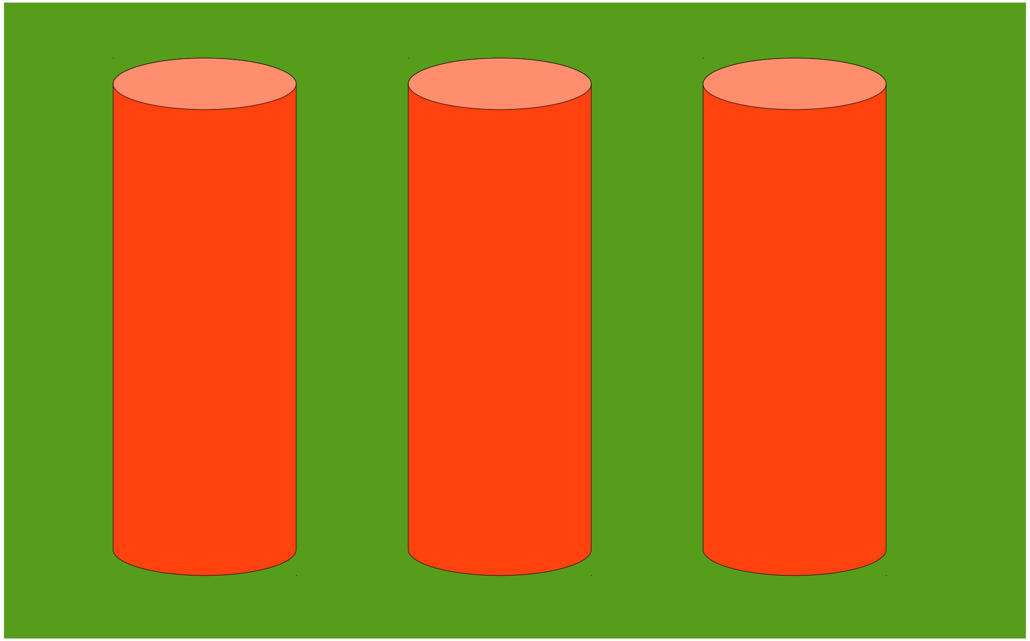


Figure 16 Elastic Scattering

Reactor building blocks

- Second building block of the reactor: **Moderator**
- ^{238}U in fuel (inelastic)
- Water (elastic)



Criticality

- The number of neutrons in the reactor should be constant to maintain a chain reaction.
- The evolution of the number of neutrons is characterized by the neutron multiplication number k
- $k = N_{i+1}/N_i$

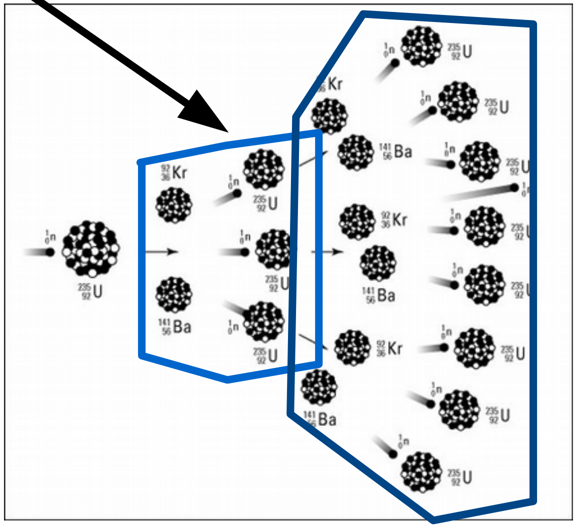
Criticality

- k needs to be one to sustain a chain reaction
- $k < 1$: the reaction will die down
- $k > 1$: the reaction rate (hence power output) will increase exponentially with time.

Evolution of number of neutrons

N(t+a) = k N(t)

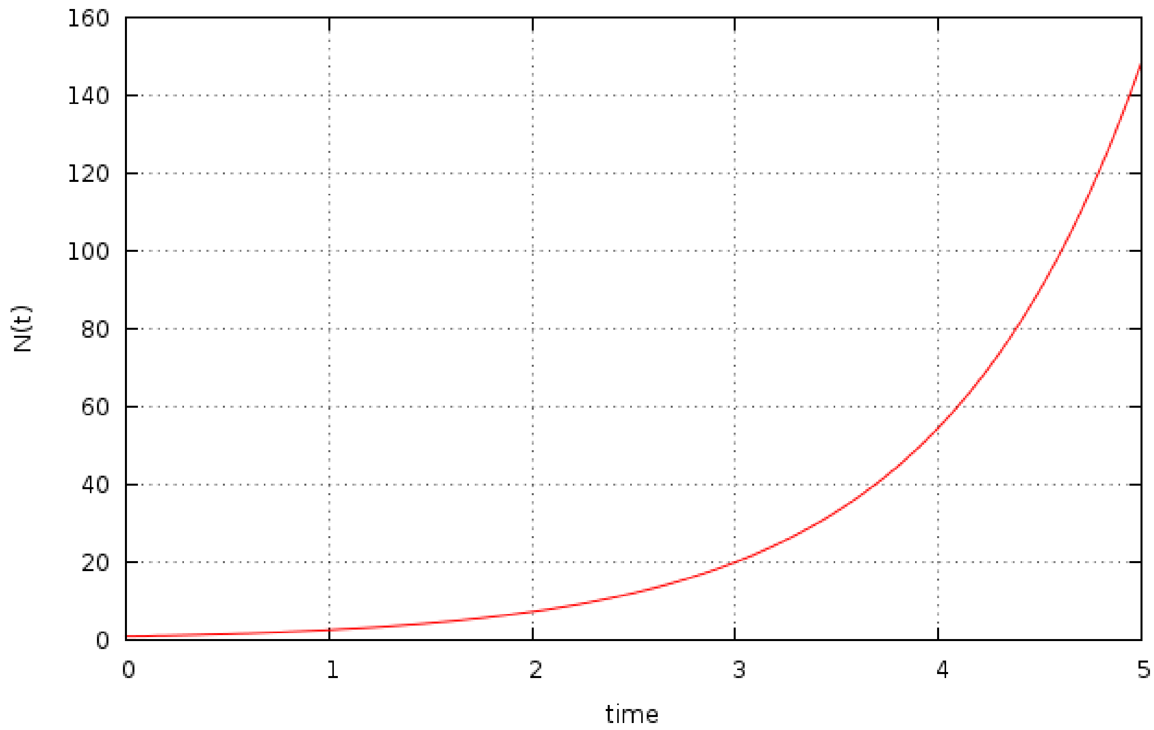
a = Unit of time
  = lifetime of a generation of neutron in the fuel
  Around 0.1 second



Evolution of number of neutrons

$N(t+a) = k\,N(t)$
$N(t) + dN/dt\ a = k\,N(t)$
$N(t)\ (k-1)/a = dN/dt$

$(k-1)/a = 1/\tau$
$N(t) = N_0 \exp(t/\tau)$



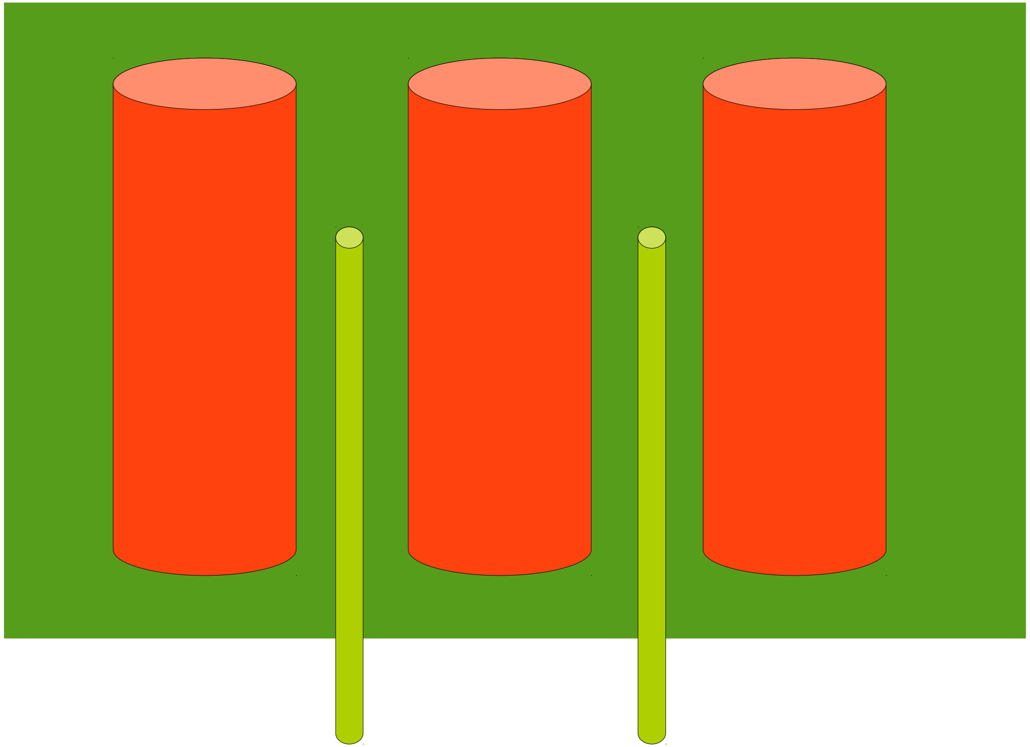
Criticality

- k is influenced by neutron productions ($\bar{\nu}$),
- Neutron losses due to geometry
- And neutron absorption by reactions other than fission

- k_∞ characterized the infinite fuel (no geometry effect)
- $k_\infty = \bar{\nu} / 1 + \alpha$, $\alpha = \sigma_{capture}/\sigma_{fission}$
- k_{eff} takes into account the geometry:
- $k_{eff} = k_\infty \times P$, P is a geometric factor, depends on the volume to surface ratio (more surface = more neutron loss)

Reactor building blocks

- Control rod
- Neutron absorbing material (B, Cd) inserted for control



Heat

$$\boxed{\mathbf{n} + {}^{235}\mathbf{U}} \mathbf{->} \boxed{\boldsymbol{x}\mathbf{n}} + \boxed{\mathbf{heat}} + \boxed{\mathbf{Fission\ Fragments}}$$

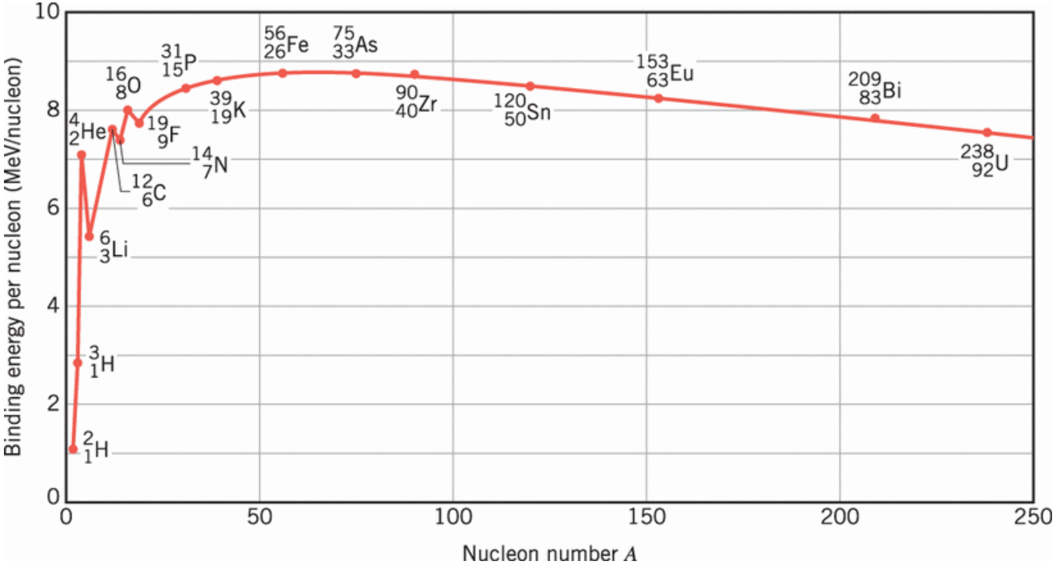
- The fission process produces energy because it leads to elements of greater stability.
- This energy is collected in the reactor to produce power.

Fission of ^{235}U

Fission : $^{235}U \rightarrow 2x\ ^{117}FF$

Initial binding energy ~ 7,5 MeV/A
Final binding energy ~ 8,5 MeV/A

Energy gain : ~ 200 MeV



Fission of one $^{235}U \rightarrow$ 200 MeV
Molar density: 238 g/mol, 1 eV = 1,6 10^{-19} J

1g of ^{235}U produces 82GJ = 22.8 MWh

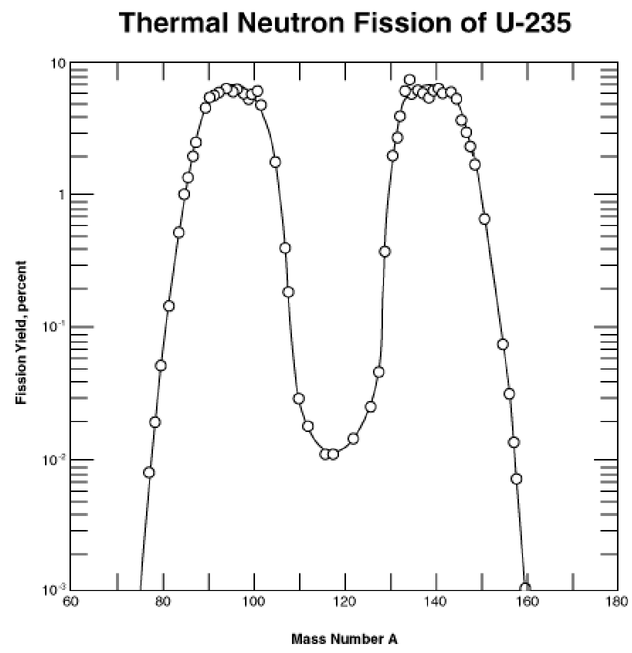
- ~ 3 tons of Coal
- ~ 1.7 tons of oil
- ~ 0.08 g of D, 0.12 g of T and 0.26 g of 6-Li

Fission of ^{235}U

Fission releases in average 200 MeV as...

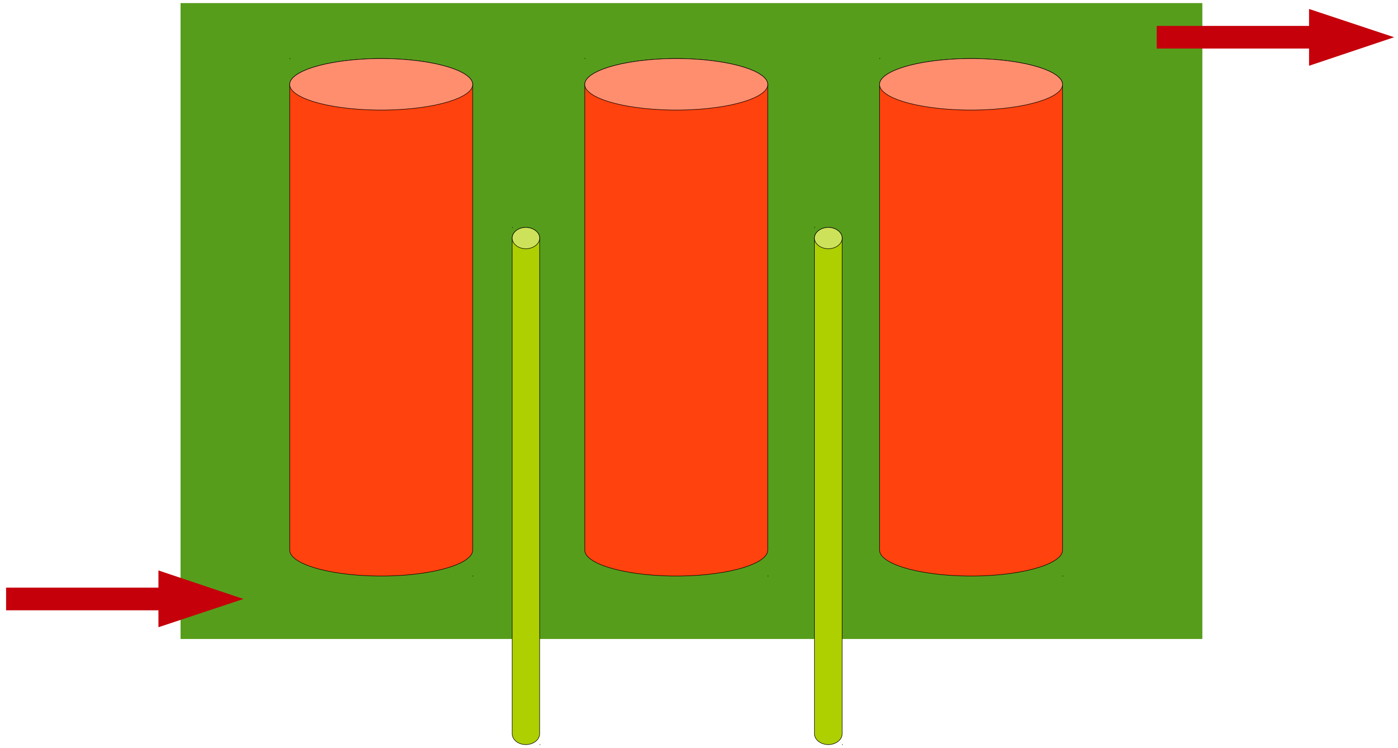
- Kinetic energy of fragments (170 MeV)
- Neutrons $\bar{\nu}\approx 2.5$ (2 MeV each)
- Gamma rays (7 MeV)

- Beta decay (8 MeV)
- (anti)neutrino 12 MeV
- Delayed gamma (7MeV)

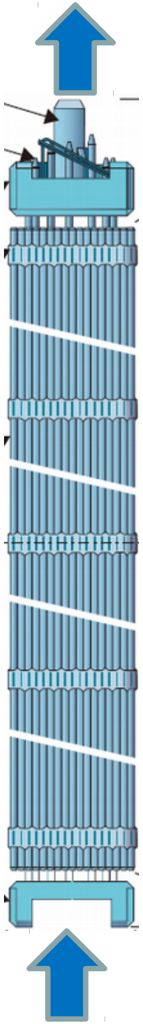


Reactor building blocks

- Building block of the reactor: **Heat exchange system**
- Water is a good heat exchange fluid (high Pressure)



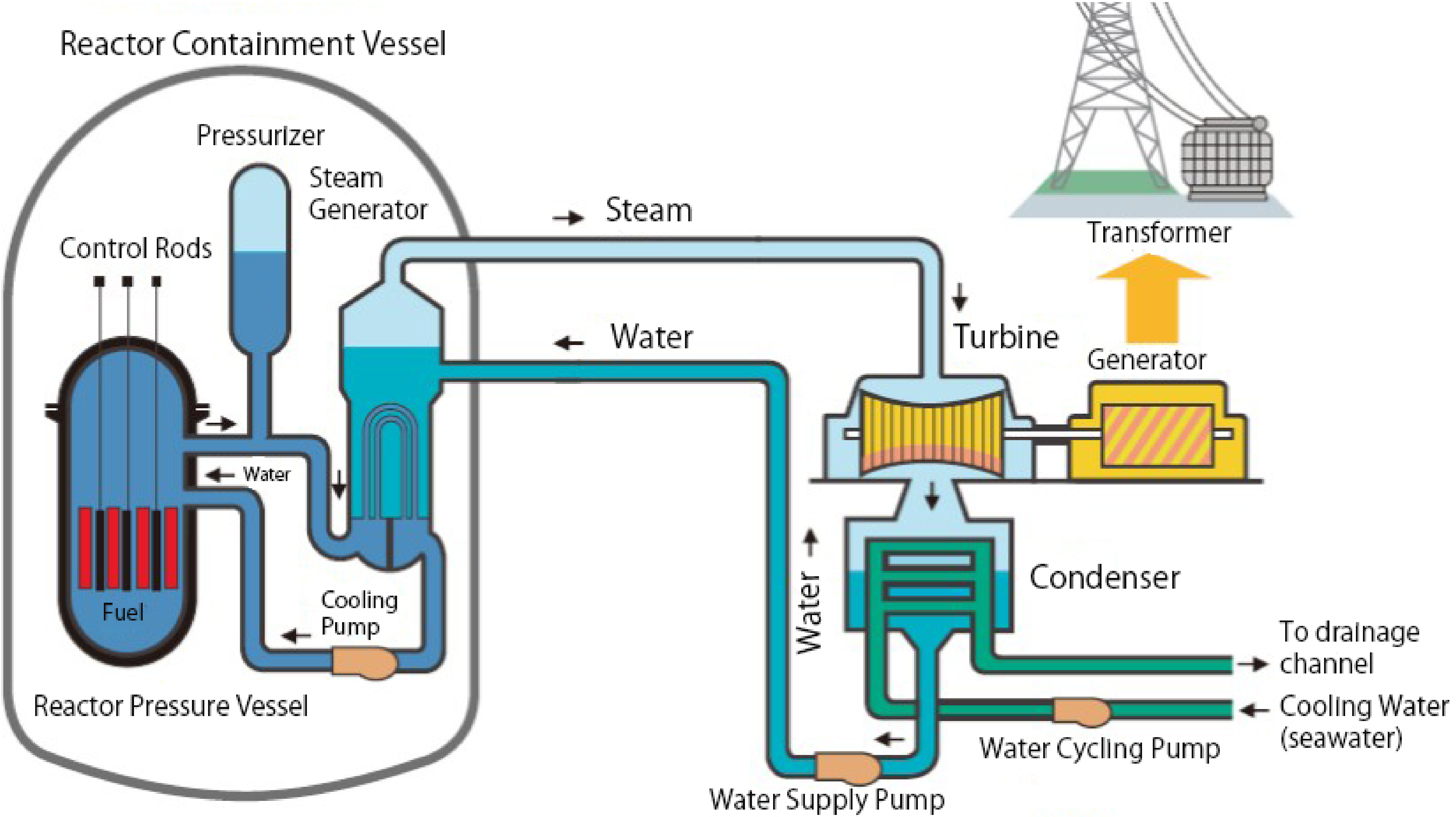
Heat exchange



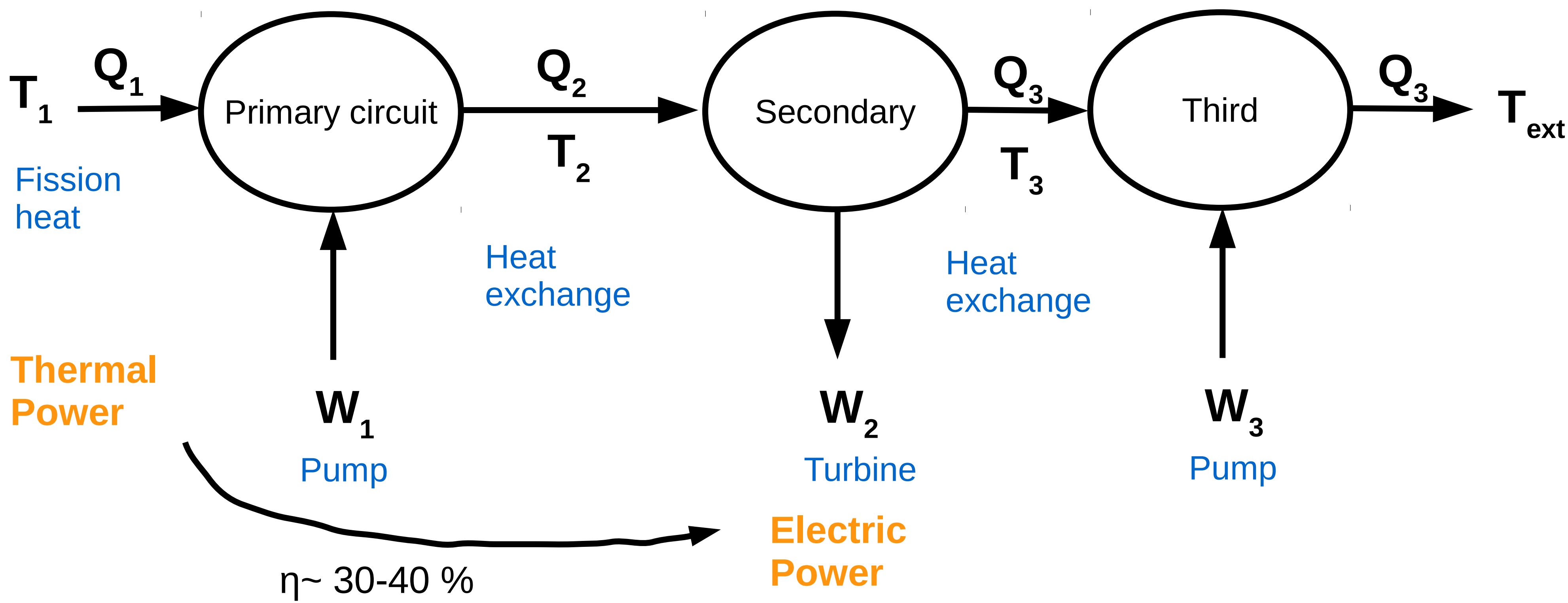
Typically:

- 1800°C in fuel
- 600°C on fuel material surface
- 400°C around fuel cladding
- 300°C for heat exchange fluid.

From heat to electric power

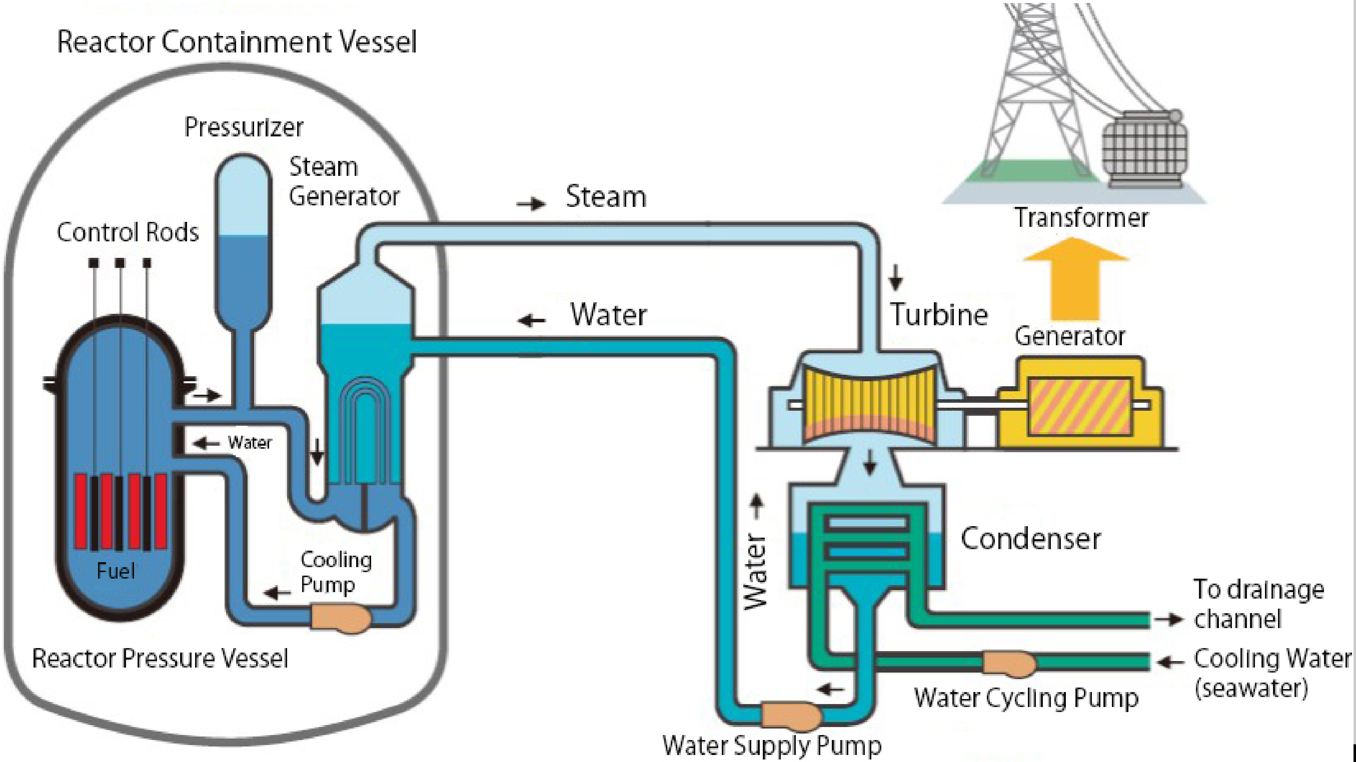


Thermal machine



Pressurized Water Reactor

- Thermal neutrons reactor.
- Fuel: Uranium Oxide (2.3% enrich. at start, 3% when refueling)
- Moderator and heat exchange fluid : water at 150 bars, 325°C.
- Heat exchange fluid stays liquid.
- Control: Graphite + Borated water
- 60 % of world's reactors.
- 100 % of French reactors
- Auto-stability
- 80 to 100 tons of U.
- 2700 to 4300 MW_{th}
- 900 to 1600 MW_{e}
- 32 % efficiency

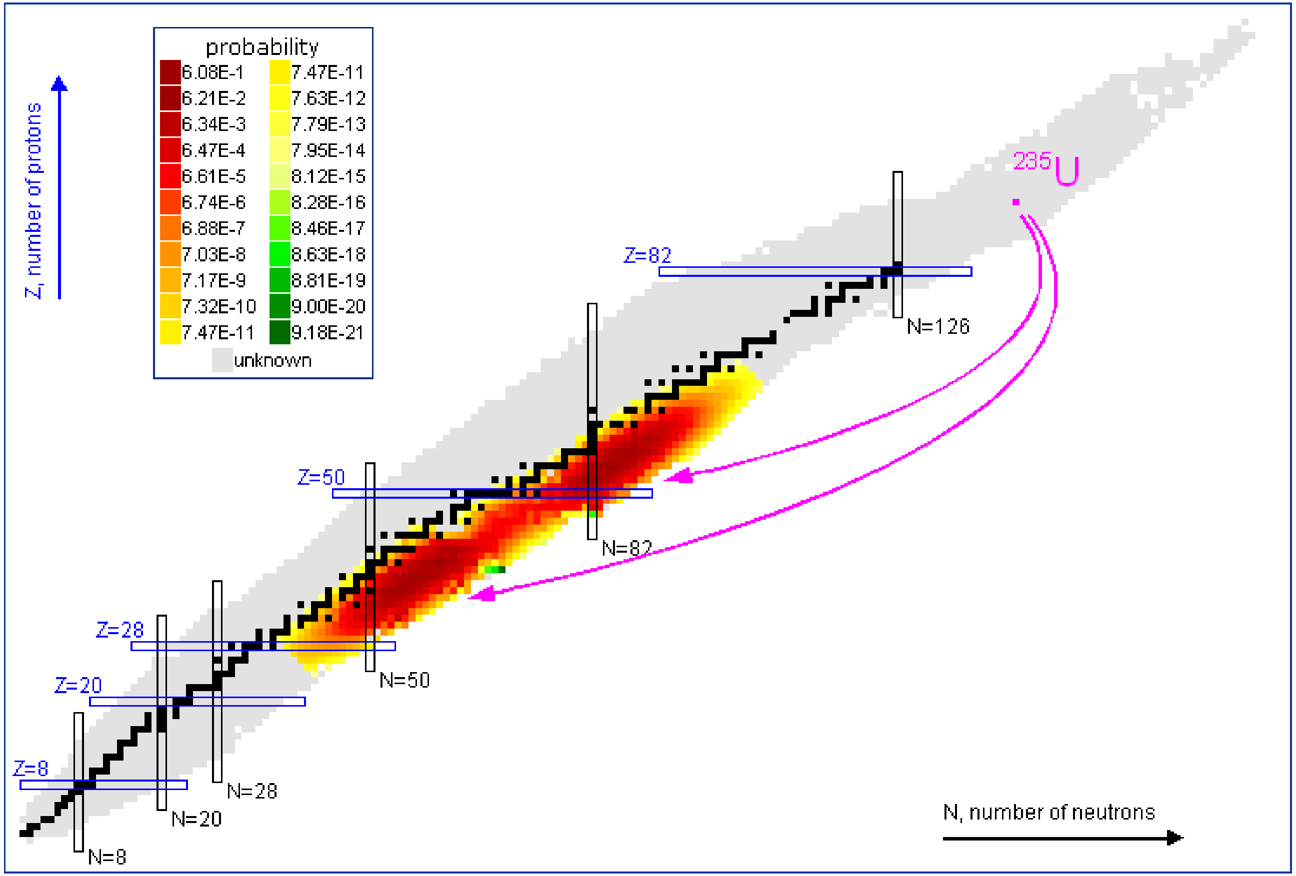


$$\boxed{\mathbf{n} + {}^{235}\mathbf{U}} \mathbf{\text{-->}} \boxed{x\mathbf{n}} + \boxed{\textbf{heat}} + \boxed{\textbf{Fission Fragments}}$$

- The fission fragments are usually unstable and can perturb the reactor running.
- Fuel waste has to be processed

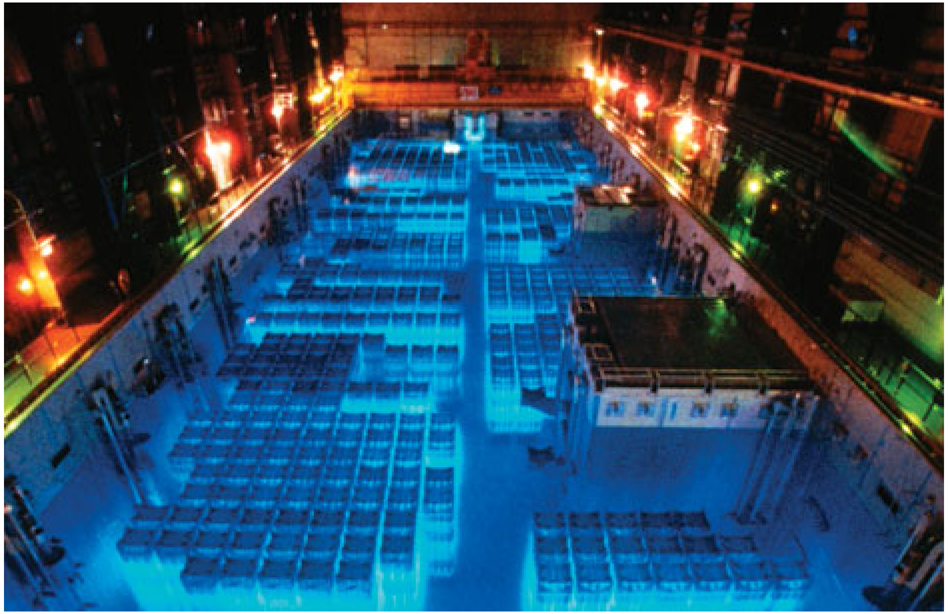
Fission Fragments

- Most of the fission products are below the stability line → β^- decay
- Life time : 10^{-5} to 10^{5} seconds.
- Delayed energy released.



Residual heat

- Spent fuel storage in pools to cool down.
- 1 to 20 years.
- After one year, the residual power is about 10 kW per ton.
- Down to 1 kW/t after 10 years
- No reprocesssing possible while the spent fuel is *hot*

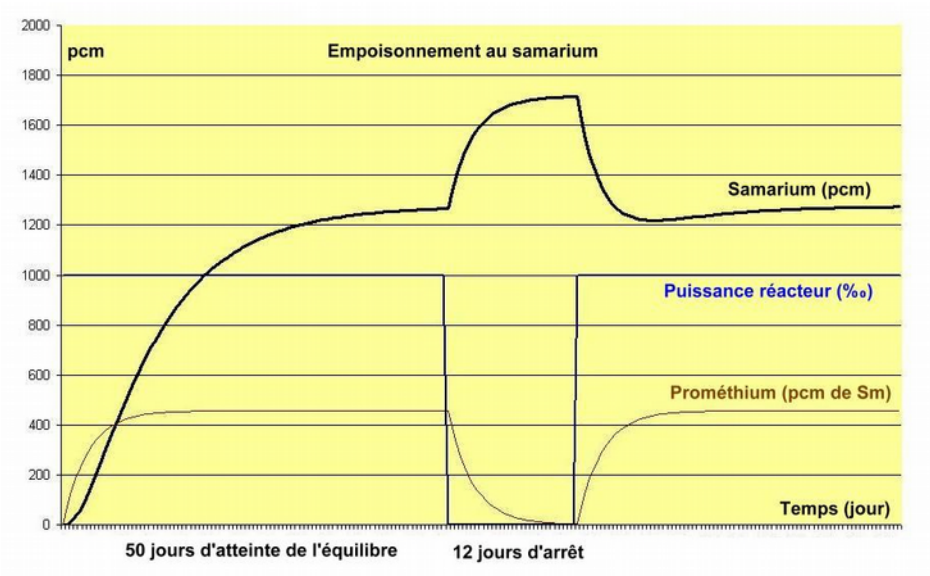


Poisons in the fuel

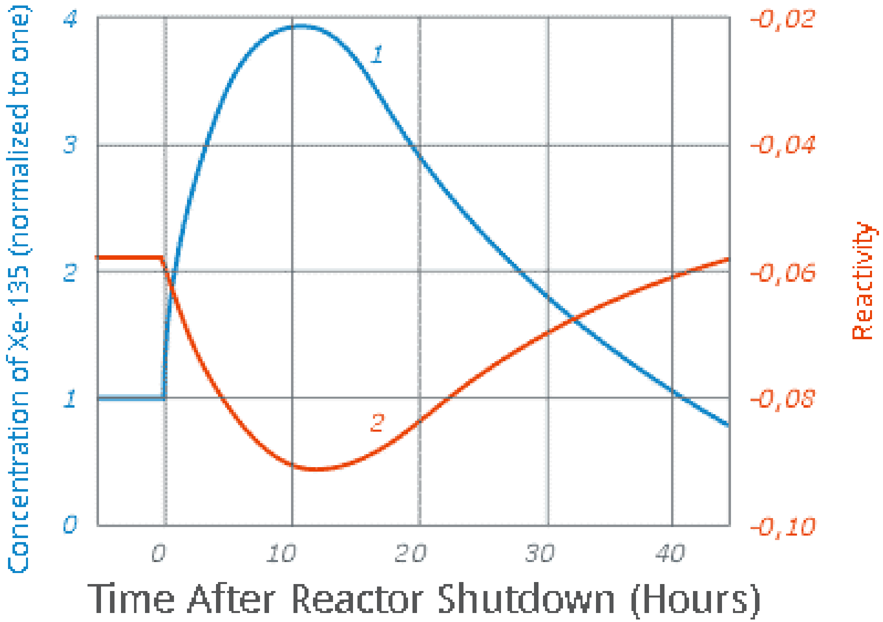
- Some fission Fragments have a very large neutron capture cross section
- Reduces the reactivity (consume neutrons, k down)
- These isotopes have specific decay, appearance times

Samarium 149

- $\sigma_{(n,\gamma)}$ = 41140 barn
- ^{149}Sm stable
- From : fission → ^{149}Nd → ^{149}Pm → ^{149}Sm, T ~55 h
- During reactor operation, concentration is stable (flux independent). Anti reactivity : -0,65 %
- After shutdown, concentration increase and get to a limit in ~10 days. Antireactivity ~ -2 %

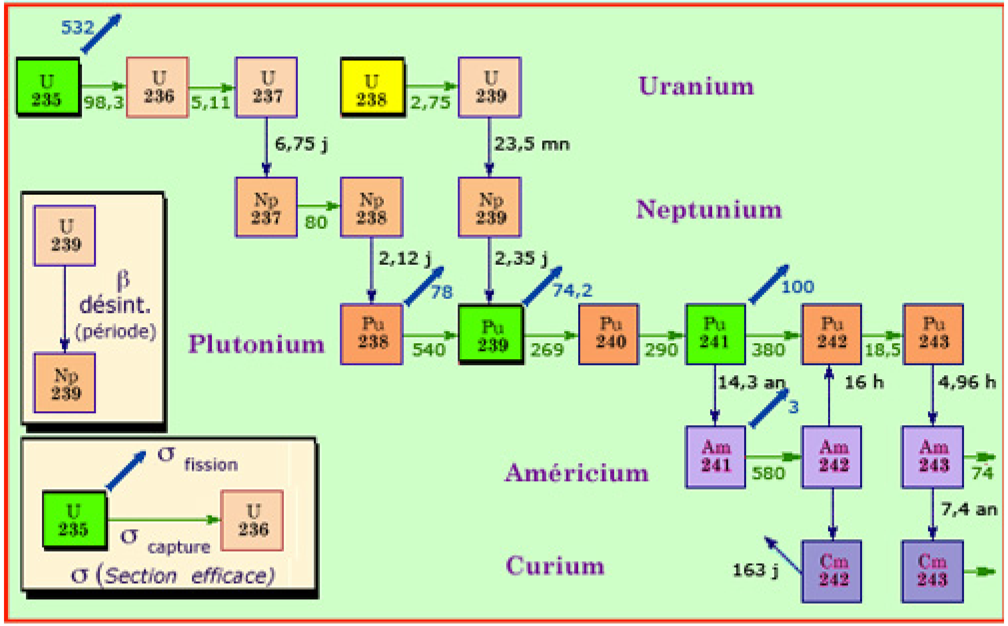


- $\sigma_{(n,\gamma)}$ = 2,65 10^6 barn
- ^{135}Xe is unstable, $t_{1/2}$=9,2h
- From : fission → ^{135}Te → ^{135}I → ^{135}Xe, T ~7 h
- During operations, concentration is stable (flux dependance). Anti reactivity ~ -4 %
- After shutdown,concentration increases and gets to a maximum in ~12 h. Antireactivity up to -20 %
- After shutdown, restart possible during ~30 minutes. After that, a few days need to wait for the decay.



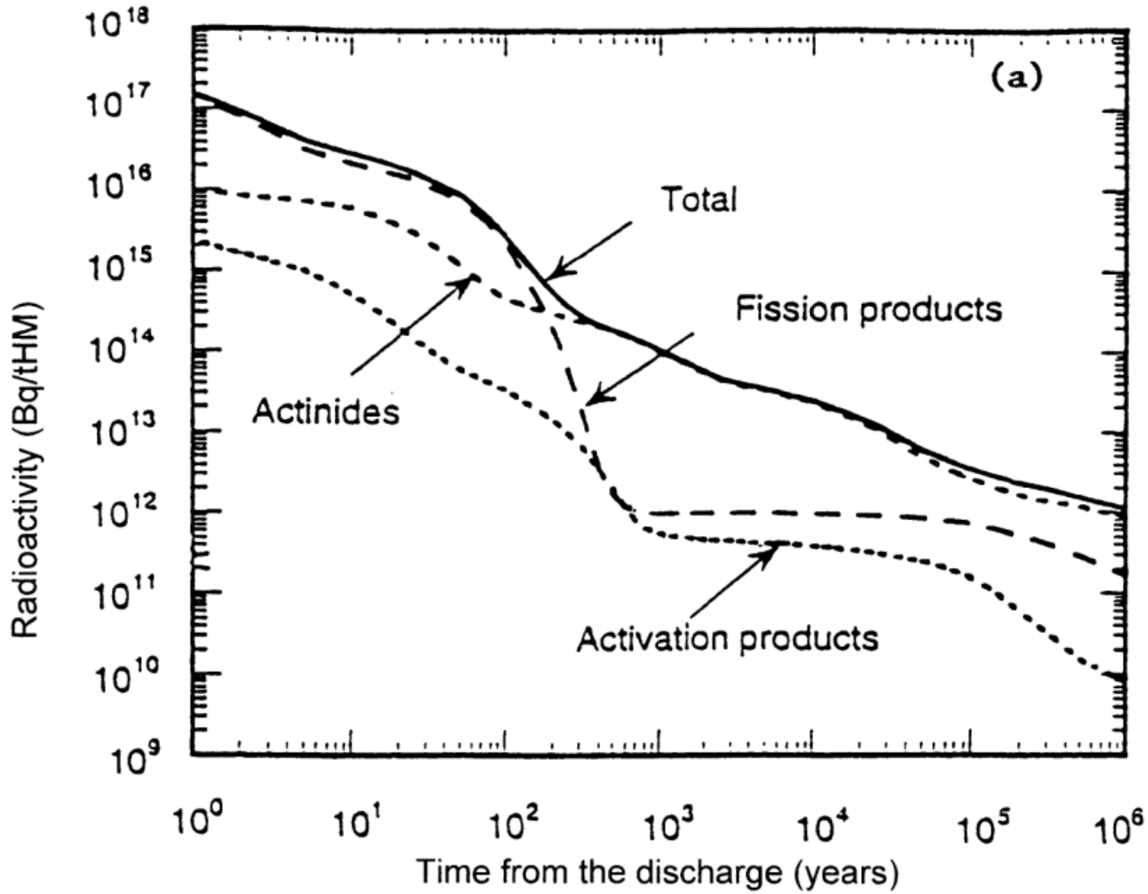
Minor Actinides

- From neutron captures on $^{235,238}U$
- $T_{1/2}$ several hundred, thousands of years
- Mostly ^{237}Np, $^{241,243}Am$, $^{243,244,245}Cm$
- Source of ^{239}Pu



Minor Actinides

Minor actinides are responsible for most of the radioactivity and residual power in spent fuel at mid and long term (300-20000 years).



Challenges of nuclear power

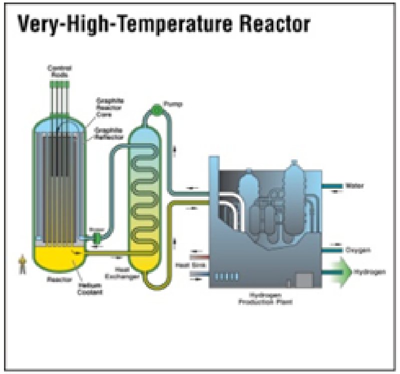
- ✔ Sustainable fuel source
- ✔ Reduction of radioactive waste
- ✔ Safety conditions

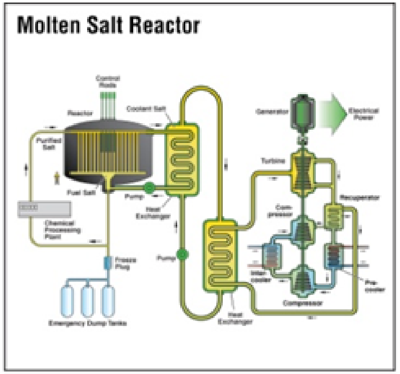
Not all objectives can be met at the same time and in the same reactor type.

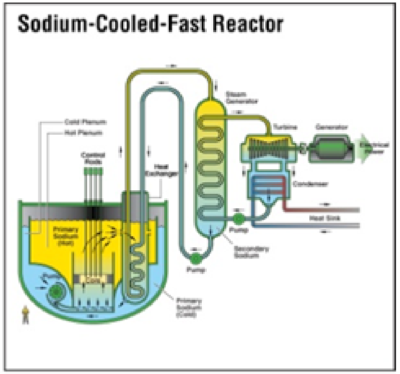
New reactor designs are needed.

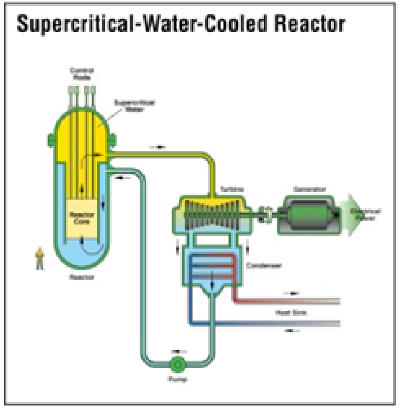
Gen-4 reactors

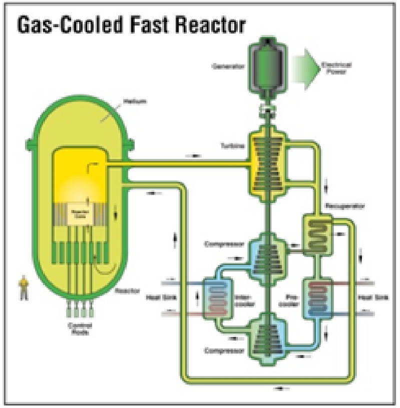
6 designs

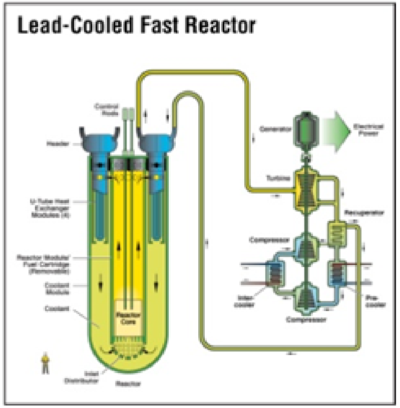












How to create new generation designs ?

Mainly done thru simulations

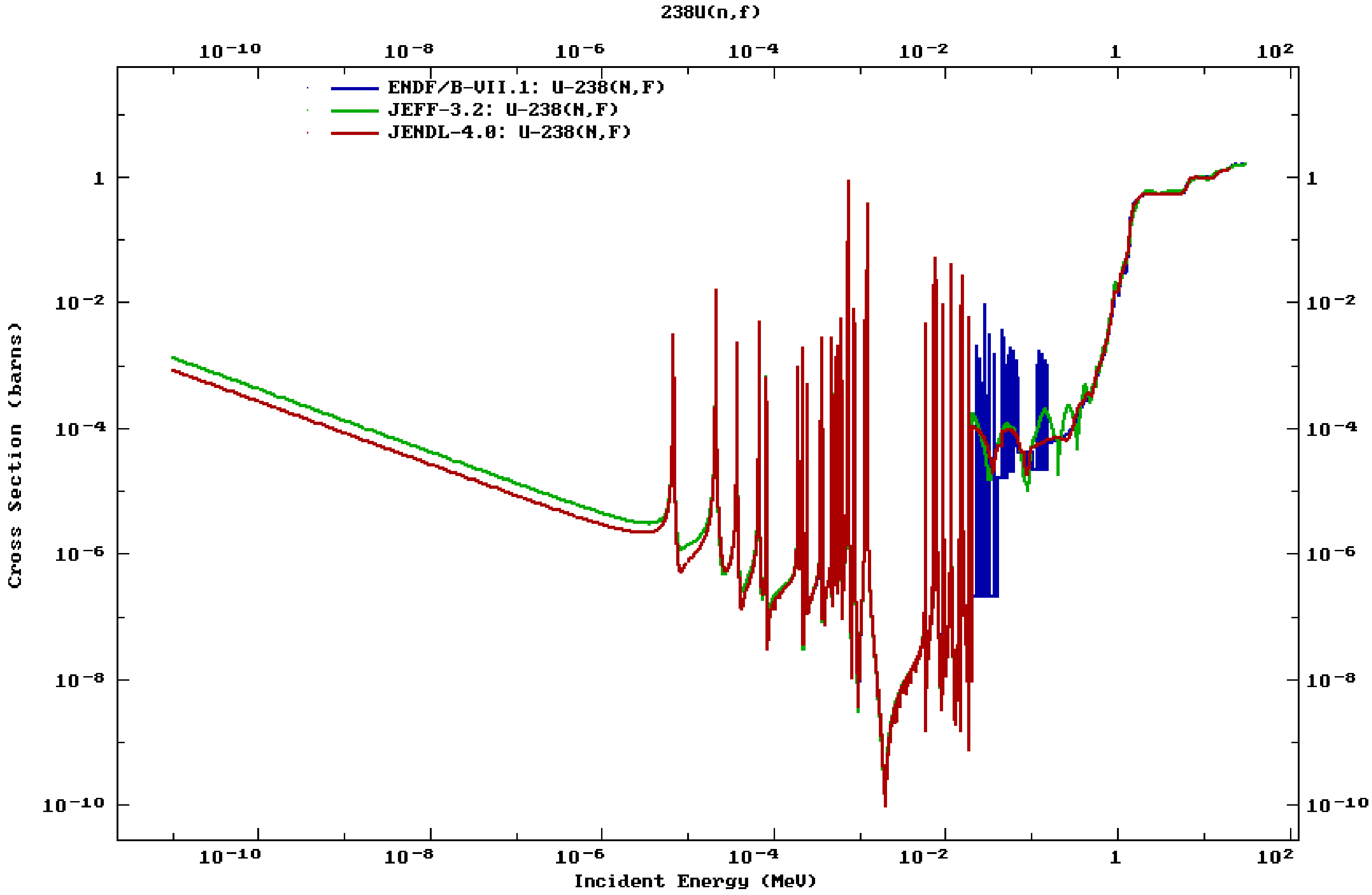
Analytical computations : Bateman equation, neutron diffusion, ...

Monte-Carlo (random) simulation : Computer softwares (MCNP, Geant, ...)

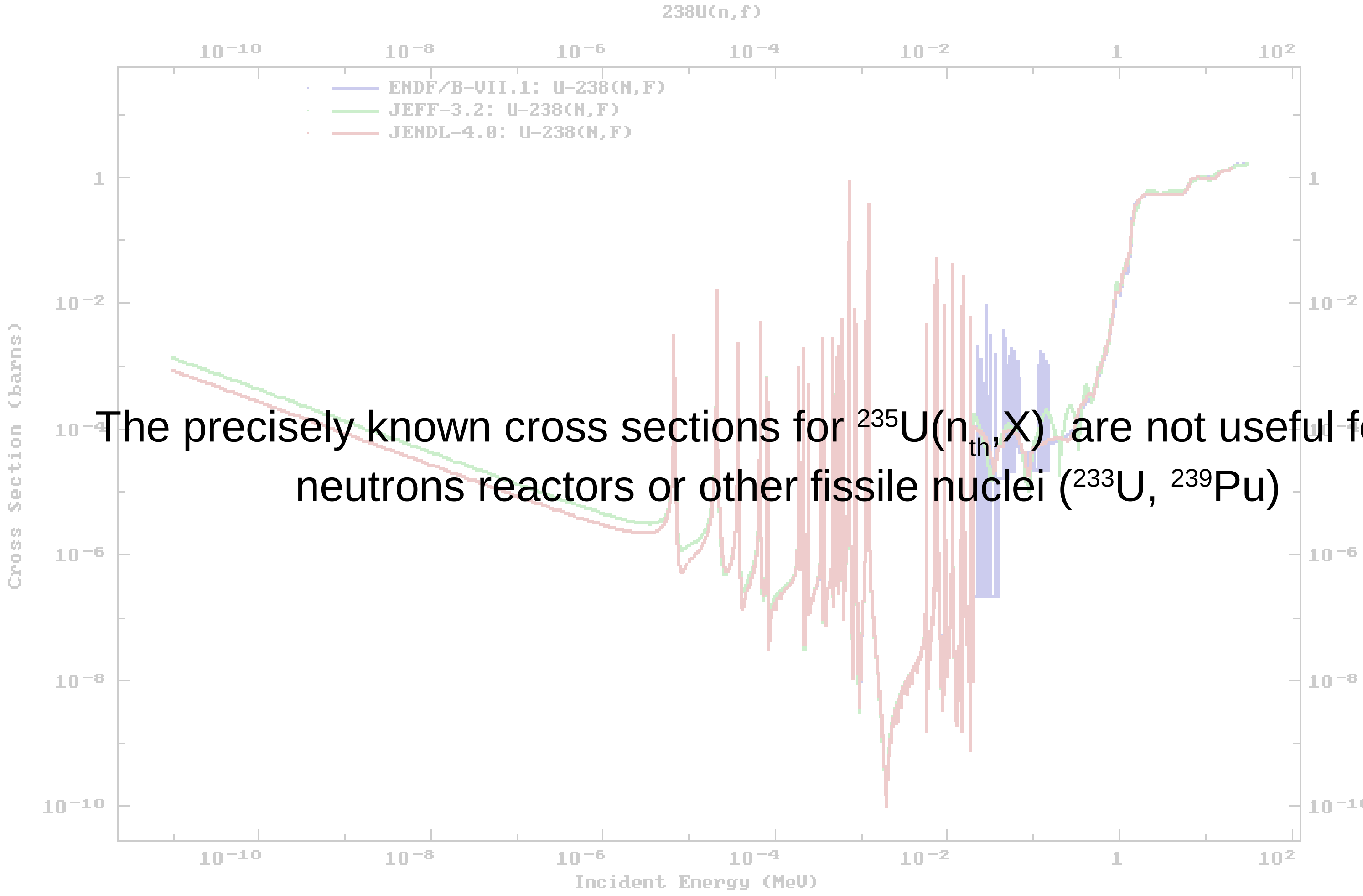
Coupling to hydrodynamics, heat diffusion, magnetic fields, ...

Criticity calculations, power map, heat, neutron budget, fuel usage and waste production ...

Evaluated data bases

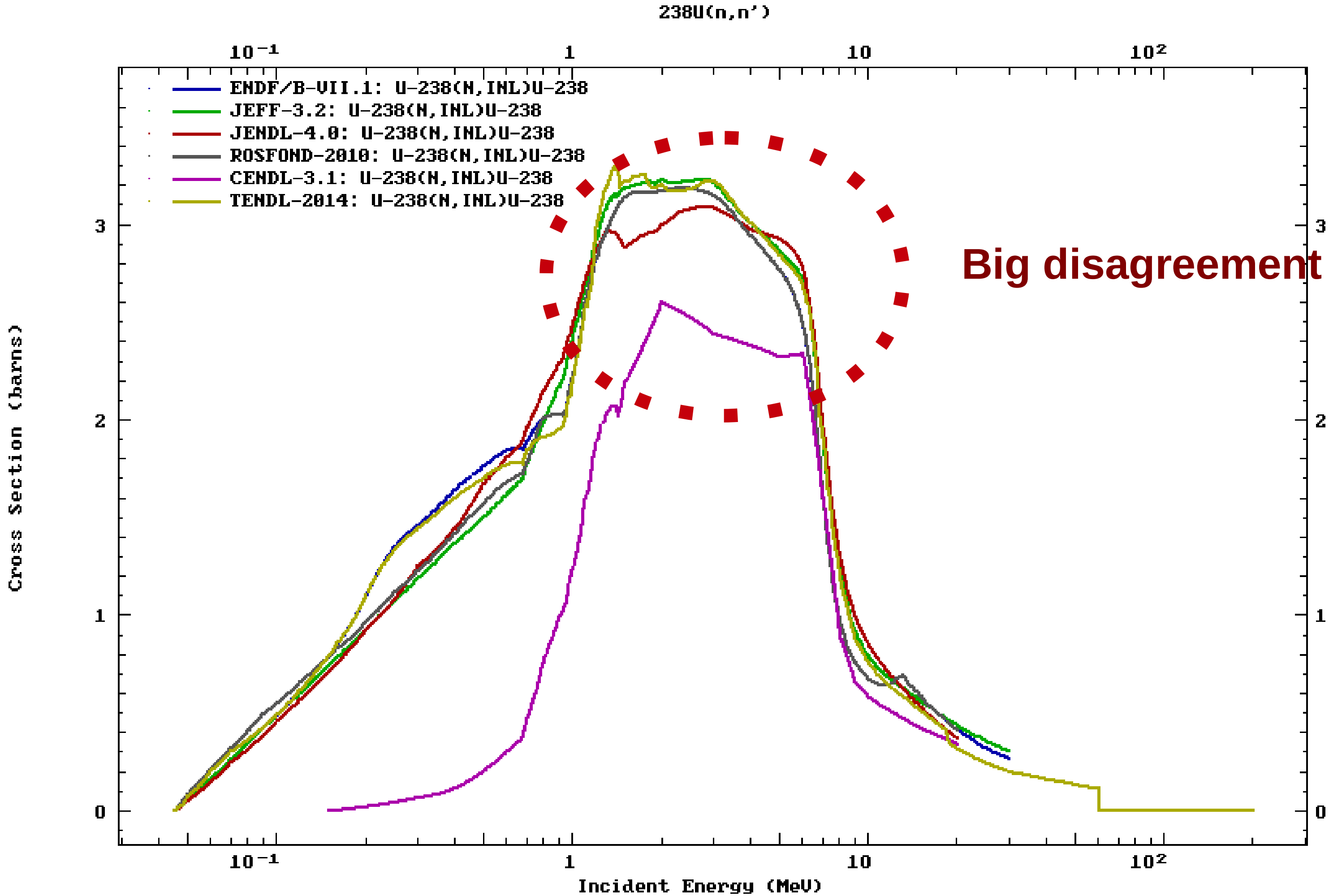


Evaluated data bases



The precisely known cross sections for $^{235}U(n_{th},X)$ are not useful for fast neutrons reactors or other fissile nuclei (^{233}U, ^{239}Pu)

Evaluated data bases



What can nuclear physics do for new reactor designs ?

Improve evaluations for better and more precise simulations of gen4 reactors, in particular for fast neutrons reactors.

Measures for

- fissiles, fertiles isotopes (^{232}Th, ^{233}U, ^{238}U, ^{239}Pu)
- Structure materials (Fe, W, Zr)
- Moderator, heat exchange fluid
  (O, Na, F, …)

Req.ID		View	Target	Reaction	Quantity	Energy range	Sec.E/Angle	Accuracy	Cov	Field
G	1		14-SI-28	(n,np)	SIG	Threshold-20 MeV	4 pi	20	Y	Fusion
H	2		8-O-16	(n,a),(n,abs)	SIG	2 MeV-20 MeV		See details	Y	Fission
H	3		94-PU-239	(n,f)	Prompt g-prod	Thermal-Fast	Eg=0-10MeV	7.5	Y	Fission
H	4		92-U-235	(n,f)	prompt g-prod	Thermal-Fast	Eg=0-10MeV	7.5	Y	Fission
H	5		72-HF-0	(n,g)	SIG	0.5-5.0 keV		4	Y	Fission
G	6		92-U-233	(n,g)	SIG	10 keV-1.0 MeV		9	Y	Fission
G	7		26-FE-56	(n,xn)	SIG,DDX	7 MeV-20 MeV	1MeV-20MeV	30		Fission,ADS
H	8		1-H-2	(n,ela)	dA/dE	0.1 MeV-1 MeV	0-180 Deg	5	Y	Fission
G	9		92-U-233	(n,g)	nubar, SIG	Thermal-10 keV		.5	Y	Fission
G	10		79-AU-197	(n,tot)	SIG	5 keV-200 keV		5		Science,Fusion
G	11		94-PU-239	(n,f), (n,g)	SIG,eta, alpha	1 meV-1 eV		1	Y	Fission
H	12		92-U-235	(n,g)	SIG, RP	100 eV-1 MeV		3	Y	Fission
G	13		24-CR-52	(n,xd), (n,xt)	SIG	Threshold-65 MeV		20	Y	Fusion
G	14		94-PU-242	(n,g), (n,tot)	SIG	0.5 eV-2.0 keV		8	Y	Fission
H	15		95-AM-241	(n,g), (n,tot)	SIG	Thermal		See details		Fission
G	16		95-AM-243	(n,f)	n spectrum	Eth-10 MeV		10		ADS
G	17		96-CM-244	(n,f)	n spectrum	Eth-10 MeV		10		ADS
H	18		92-U-238	(n,inl)	SIG	65 keV-20 MeV	Emis spec.	See details	Y	Fission
H	19		94-PU-238	(n,f)	SIG	9 keV-6 MeV		See details	Y	Fission
H	21		95-AM-241	(n,f)	SIG	180 keV-20 MeV		See details	Y	Fission
H	22		95-AM-242	(n,f)	SIG	0.5 keV-6 MeV		See details	Y	Fission
H	25		96-CM-244	(n,f)	SIG	65 keV-6 MeV		See details	Y	Fission
H	27		96-CM-245	(n,f)	SIG	0.5 keV-6 MeV		See details	Y	Fission
H	29		11-NA-23	(n,inl)	SIG	0.5 MeV-1.3 MeV	Emis spec.	See details	Y	Fission
H	32		94-PU-239	(n,g)	SIG	0.1 eV-1.35 MeV		See details	Y	Fission
H	33		94-PU-241	(n,g)	SIG	0.1 eV-1.35 MeV		See details	Y	Fission
H	34		26-FE-56	(n,n')	SIG	0.5 MeV-20 MeV	Emis spec.	See details	Y	Fission
H	35		94-PU-241	(n,f)	SIG	0.5 eV-1.35 MeV		See details	Y	Fission
H	36		92-U-238	(n,g)	SIG	20 eV-25 keV		See details	Y	Fission
H	37		94-PU-240	(n,f)	SIG	0.5 keV-5 MeV		See details	Y	Fission
H	38		94-PU-240	(n,f)	nubar	200 keV-2 MeV		See details	Y	Fission
H	39		94-PU-242	(n,f)	SIG	200 keV-20 MeV		See details	Y	Fission
H	40		14-SI-28	(n,inl)	SIG	1.4 MeV-6 MeV		See details	Y	Fission
H	41		82-PB-206	(n,inl)	SIG	0.5 MeV-6 MeV		See details	Y	Fission
H	42		82-PB-207	(n,inl)	SIG	0.5 MeV-6 MeV		See details	Y	Fission
H	43		1-H-1	(n,n)	SIG,DA	10 MeV-20 MeV	4 pi	1-2	Y	Standard

Requests listed in *HPRL*

Coordination :

UE,
OCDE/NEA,
IAEA

Conclusion

$$\boxed{n + {}^{235}U} \rightarrow \boxed{x\mathrm{n}} + \boxed{\mathrm{heat}} + \boxed{\text{Fission Fragments}}$$

Fuel enrichment

Neutron moderators
Control of criticity (k)

Heat exchange system
Power Generation

+ Minor actinides
Spent fuel handling and reprocessing

Safety and sustainability challenges could be met with new reactor designs, new fuel cycles.

→ **Need for new nuclear physics data for simulation and development.**