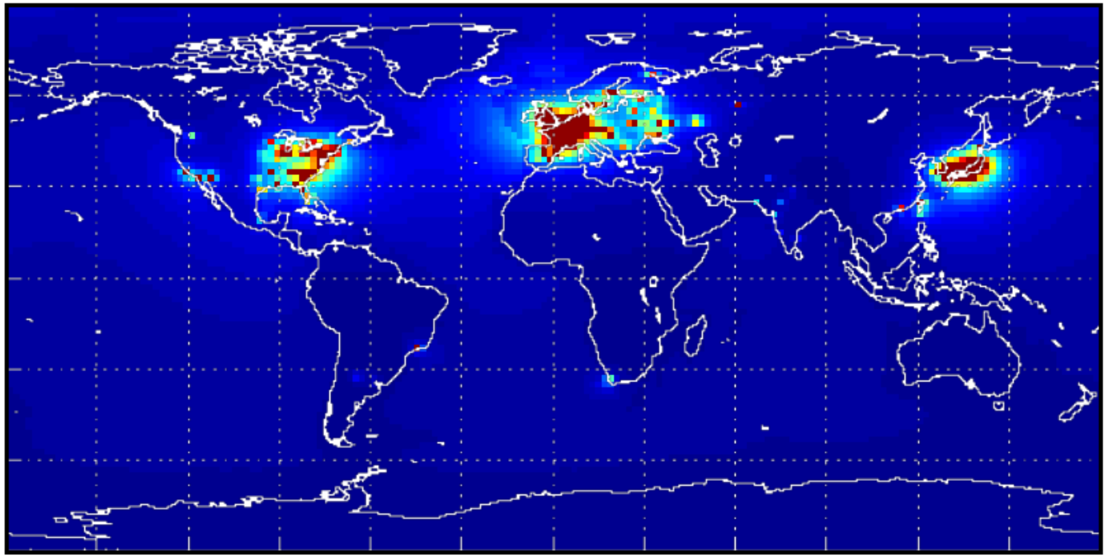
Nuclear power plants



- Fission for power generation: fast and thermal reactors
- Criticality
  - Critical mass and radius
  - Criticality and delayed neutrons
- The moderator
  - LWR and EPR
  - HWR and CANDU
  - Graphite: from Fermi to Chernobil
- What is being burnt in a thermal reactor?
- Plutonium
- Fertile material and breeders
- Neutrinos ad reactors
  - Antineutrinos from reactors
  - Reactor monitoring with neutrinos?

Remind: Neutron Induced Fission

- Neutron induced fission,

$n + (Z,A) \rightarrow (Z,A+1)^* \rightarrow$

$(Z_1,A_1) + (Z1,A1) +$ n's

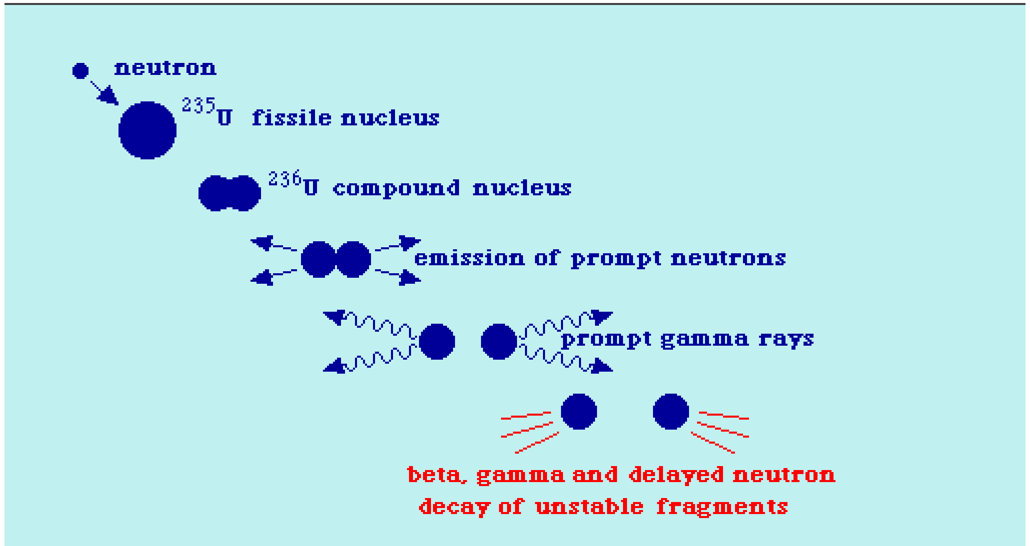
is particularly effective for the following reasons:

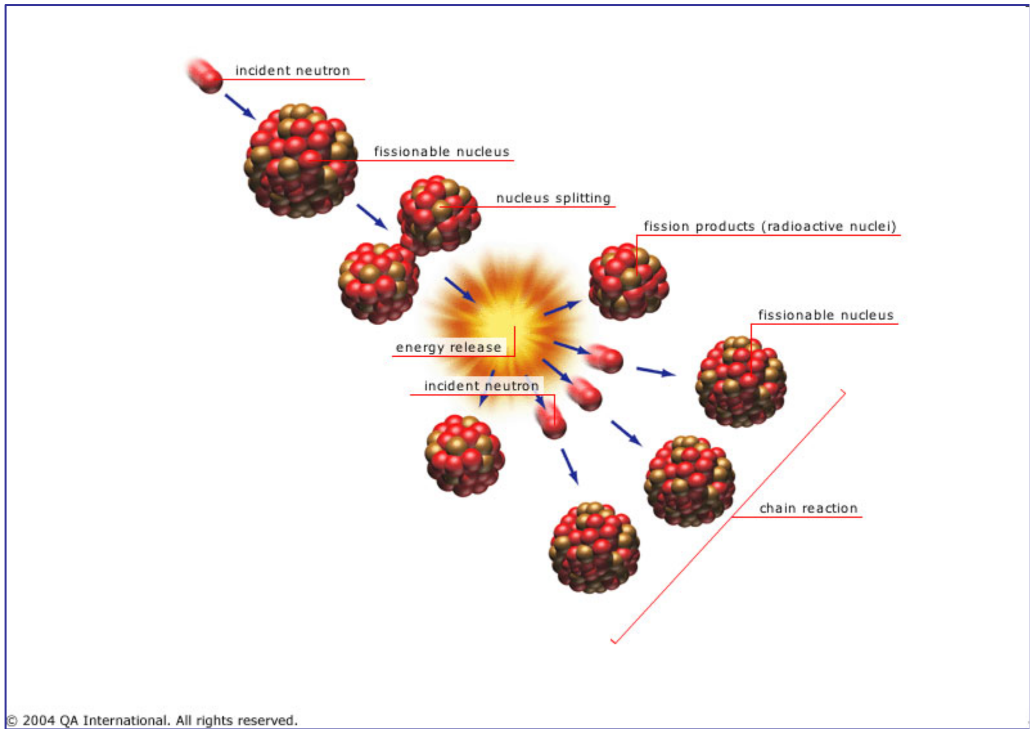
1)neutrons are penetrating,

2)slow neutrons cross sections are large, order 100-1000 barn at thermal energies (see later)

3) Neutrons are regenerated during fission

The induction of fission by neutron absorption and the susequent emission of neutrons in the process of fission leads to the possibility of a **self sustaining** or **chain reaction**.



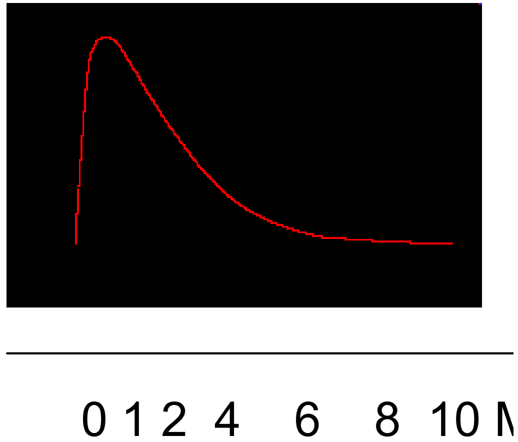


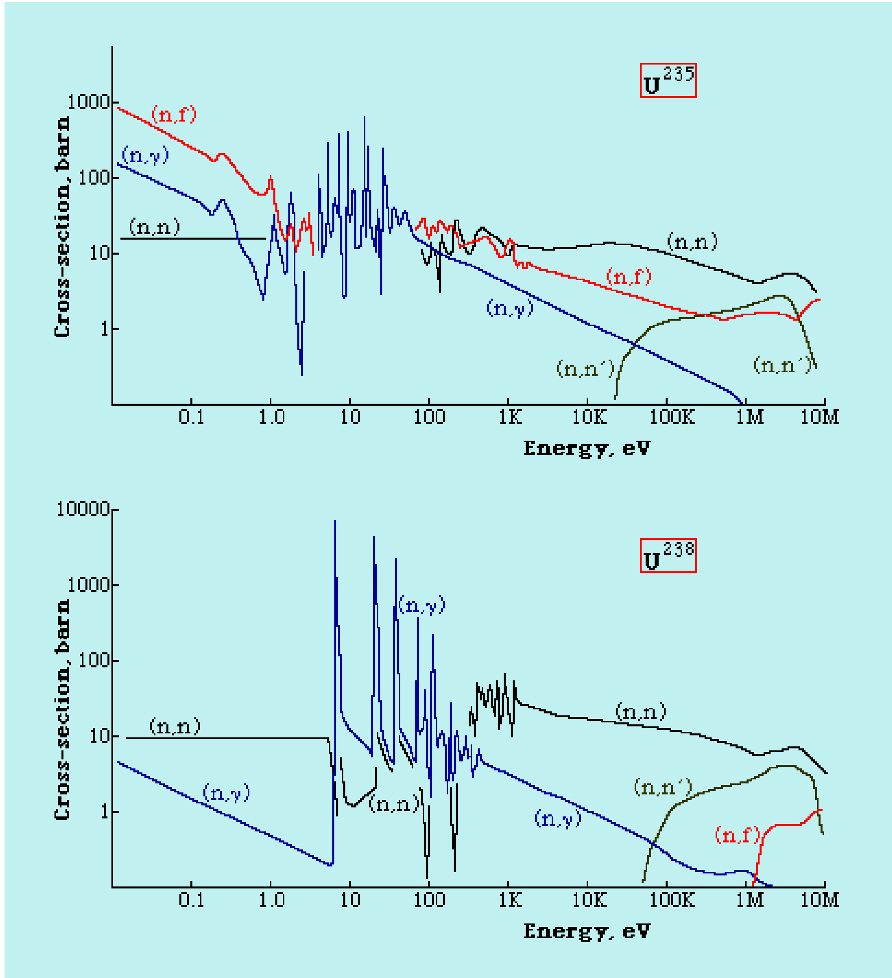
Remind: The fate of neutrons in natural Uranium today

-The induction of fission by neutron absorption and the susequent emission of neutrons in the process of fission leads to the possibility of a **self sustaining** or **chain reaction**. However such a reaction cannot occur in natural uranium (**0.7%** ^{235}**U, 99.3%** ^{238}**U**) *

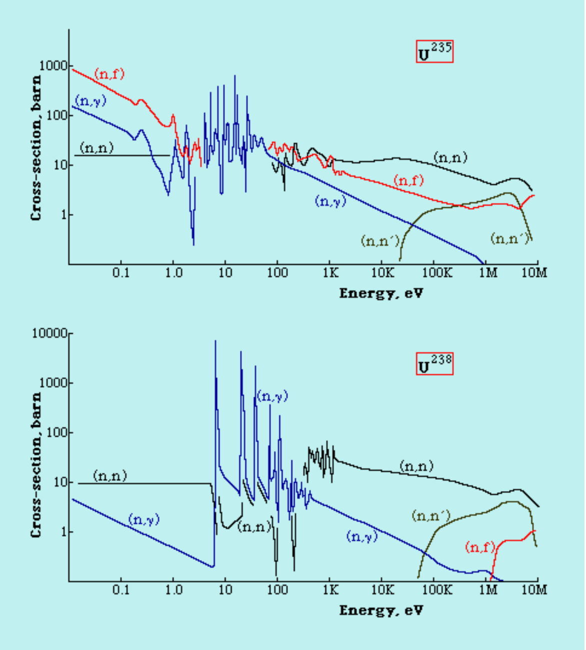
-Starting with **100** fission neutrons about **98** are **abosrbed** in ^{238}**U** and only **8** of these captures **result in fission**. The remaining **2** neutrons cause **fission** of ^{235}**U**. Since each fission produces **2-3 neutrons** there will be only about **25** neutrons in the second generation - clearly insufficient to sustain a chain reaction.

*Things were different long ago, see the Ohlo natural reactor…





Fission for power generation



- For using fission for power generation there are basically two ways to avoid the above limitations.

  - By enriching the ^{235}U content then sufficient fission of this isotope will occur. For example a 50-50 mix of the two isotopes will sustain a chain reaction with most of the fission events occuring in ^{235}U at neutron energies in the range 0.3 - 2.0 keV. A reactor working in this way is called a **FAST REACTOR.**

- By mixing natural (or few per cent 235U enriched) uranium with a material which slows down the neutrons without absorption, sufficient of them will get to low energies where the fission cross-section for 235U is extremely large. Such a slowing down medium is called a moderator. In this case most of the fissions are induced by neutrons with thermal energy (0.025 eV) - hence the name **THERMAL REACTOR**.

Criticality

- A **critical mass** is the smallest amount of fissile material needed for a sustained nuclear chain reaction.
- The critical mass of a fissionable material depends upon its nuclear properties (e.g. the nuclear fission cross-section), its density, its shape, its enrichment, its purity, its temperature, and its surroundings.
- A numerical measure of a critical mass is dependent on the neutron multiplication factor

  $$\mathbf{k = f - l}$$

  where **f** is the average number of neutrons released per fission event and **l** is the average number of neutrons lost by either leaving the system or being captured in a non-fission event.
- When **k = 1** the mass is **critical**.
- A **subcritical** mass is a mass of fissile material that does not have the ability to sustain a fission reaction (**k < 1**).
- A **supercritical** mass is one where there is an increasing rate of fission (**k >1**).

Criticality and size

- Consider an infinite amount of fissile material. Its multiplication factor k_{oo} will depend on the properties of the material

  (For example, for a pile of Uranium and graphite $\mathbf{k_{oo}}$ =1.1)
- For a finite amount, the true multiplication factor k will depend on how many neutrons are lost though the walls, i.e. it will depend on the extension of surface.
- Assume one has a sphere of radius R. In this case

  $\mathbf{K = k_{oo} - \pi D^2/R^2}$

This can be understood observing that:

-The factor $1/R^2$ expresses the surface effect

-D is the migration distance of the neutron (mainly due by the path of thermalize neutrons before absorption)

- Criticality is achieved for k=1, i.e.

  $R_c = \pi D/(k_{oo}-1)^{1/2}$
- This critical radius is some 50 cm for the U-C pile
- For a pure ^{235}U it is 17 cm, correponding to some 50 kg of critical mass
- The critical mass for lower-grade uranium depends strongly on the grade: with 20% U-235 it is over 400 kg; with 15% U-235, it is well over 600 kg.

Criticality and delayed neutrons

- Note that the neutron population (and the fission rate) will vary as

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

  where τ e' is the time between two generations, of order ms, (mainly due to thermal neutron migration time).
- In a reactor k=1 is mantained by acting on the delayed neutrons, through insertion/removal of rods made of Cadmium (a material with large n-absorption cross section and good mechanical properties at high temperatures).
- The cadium control rods are under mechanical control and can be gradually removed from the pile or rapidly inserted.
- The reactor is designed in such a way that k<1 when delayed neutrons are absorbed (i.e. Cadmium rods are inserted).
- Reactor is switched on until criticality by removing the rods; it is switched off by inserting them back.
- Note that this can only work with the delayed neutrons, which have time constants of order seconds to minutes
- Since criticality depends on physical conditions, reactor are to be designed so as to have suitable feedback on k (see later)

The moderator

- Thermal reactors need a "moderator", a component where neutrons are slowed down thus avoiding (or minimizing) lossed due to capture in ^{238}U.
- The ideal moder has to be:
- light (atomic mass comparable to neutron mass, for efficient slowing down)
- Not neutron thirsty (i.e. does not capture neutrons)
- cheap
- not inflammable
- In practice, a compromise has to be obtained.
- Most common used materials are

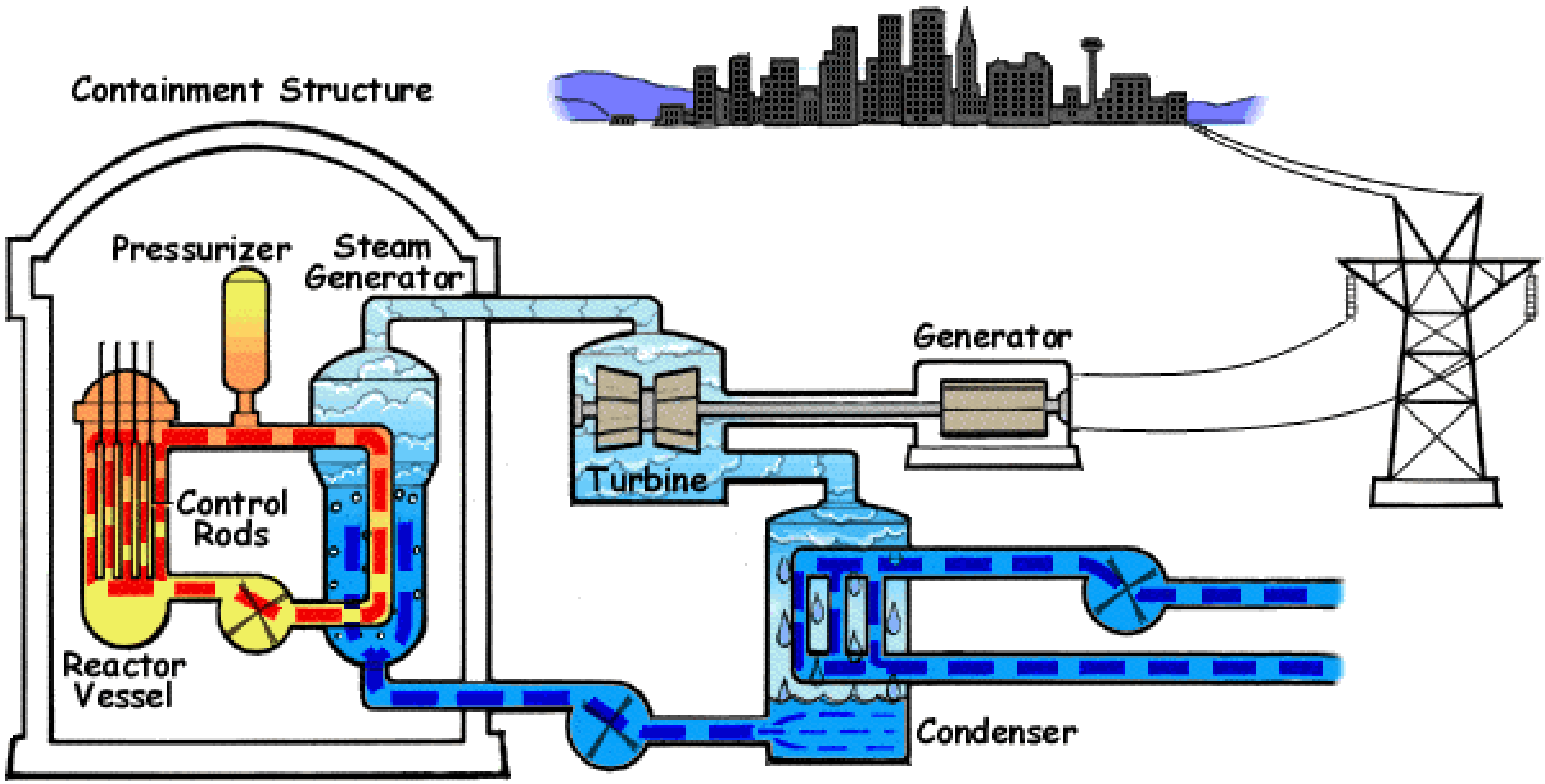
-light water (H_20)
-heavy water (D_2O)
-graphite

	H_20	D_2O	graphite
M_a/M_n	Ok	Ok	Accept.
σ_n	Accept.	small	small
Fire danger	NO	NO	YES
cost	cheap	Expens.	cheap

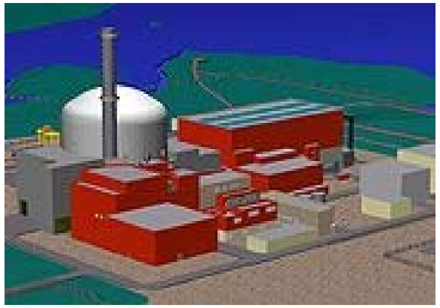
Light Water Reactors (LWR)

- Light water reactors use ordinary water to moderate and cool the reactors.
- Water is light, cheap, and does not burn.
- However it eats neutrons ($p+n \rightarrow d + \gamma$) and Uranium has to be enriched, to about 3 per cent for reaching criticality.
- When at operating temperature, if the temperature of the water increases, its density drops, and fewer neutrons passing through it are slowed enough to trigger further reactions. **That negative feedback** stabilizes the reaction rate.
- Also, if the moderator/coolant is lost, neutrons are not slowed down. Rather they are captured by ^{238}U and the reactor stops.
- For this reason, LWR are the most diffuse world wide, both in the Boling Water and Pressurized Water form.
- There are some 360 LWR in in 27 countries, with a global generating capacity of some 330 GW.

Pressurized water reactors



EPR = European/Evolutionary Pressurized/Power Reactor



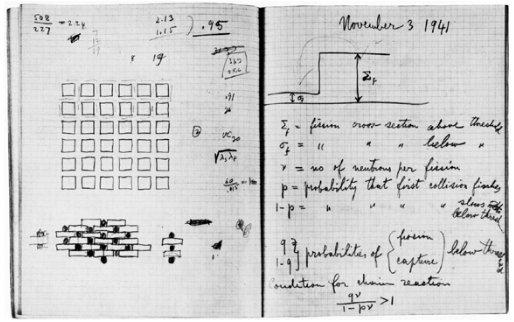
- It is the generation **3+** reactor, developped in Europe by AREVA, EDF and Siemens AG* Light water reactors use ordinary water to moderate and cool the reactors.
- Two such reactors are being built, in Finland (Okiluoto) and in France (Flamanville 3)
- Several agreements for EPR have been signed with United Arab Emirates, China, India and recently Italy
- The main design objectives of the generation 3+ EPR design are increased safety while providing enhanced economic competitiveness
- The EPR has a design maximum core damage frequency of 6.1×10^{-7} per plant per year.
- Economic competitiveness is improved through scale economies, up to an electrical power output of 1650 MWe.

*Similar design in US is called US-EPR

CANDU = CANada Deuterium Uranium

- Light water reactors need enriched Uranium, and thus the (expensive) technology for Uranium enrichment .
- Canada has developped a system which can burn natural Uranium, by employing as moderator and coolant (expensive) heavy water.
- The use of heavy water moderator is the key to the CANDU system, enabling the use of natural uranium as fuel (in the form of ceramic UO_2), which means that it can be operated without expensive uranium enrichment facilities.
- Compared with light water reactors, a heavy water design is "neutron rich". This makes the CANDU design suitable for "burning" a number of alternative nuclear fuels. To date, the fuel to gain the most attention is mixed oxide fuel (MOX). MOX is a mixture of natural uranium and plutonium, such as that extracted from former nuclear weapons.
- Today there are 17 CANDU reactors in use in Canada, 12 in the rest of the world, and a further 13 "CANDU-derivatives" in use in India (these reactors were developed from the CANDU design after India detonated a nuclear bomb in 1974 and Canada stopped nuclear dealings with India).

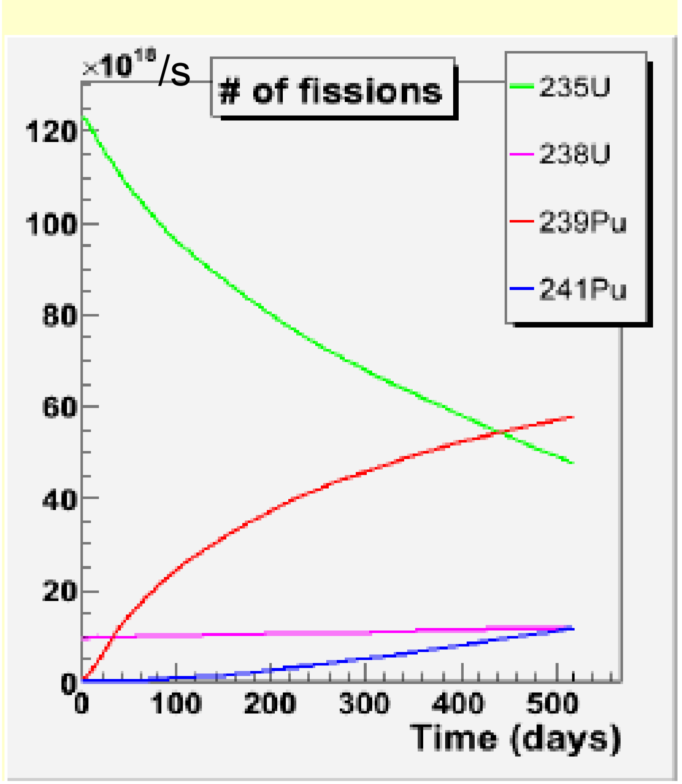
Graphite moderated reactor: from Fermi pile to Chernobil



- A graphite moderaed reactor can also burn natural Uranium
- The first nuclear reactor, Chicago Pile-1, a graphite-moderated device that produced a microscopic amount of heat, was constructed by a team led by Enrico Fermi in 1942. The construction and testing of this reactor (an "atomic pile") was part of the Manhattan Project.
- The former USSR has devloped a series of graphite reactors (RBMK), the type involved in the Chernobyl accident.
- In 2008, there were at least 12 RBMK reactors still operating in Russia and Lithuania, but there are no plans to build new RBMK type reactors (the RBMK technology was developed in 1950s and is now considered obsolete) and there is international pressure to close those that remain.
- Water cooled graphite moderated reactor have a **positive k-coefficient**. Water abosrbs neutrons but if temperature raises and it moves to vapour density decreases and thus there is in afixed volume less water to absorb neutrons (positive feedback !)
- Control of the reactor relies on temperature sensors, which command the insertion of (boron) control rods.

What is being burnt in a nuclear power plant ?

- Consider a reactor which at burn up contains a critical mixture of 235U and 238U.
- Of course, the main ingredient is the fissile ^{235}U, where fission is indiced by thermalized neutrons
- A non negligible fraction arises from ^{238}U, due to the fast neutrons, before they are slowed down
- Note also the role of Plutonium, which at later times even passes the role of 235U.
- Plutonium was not there at burn up. It has been produced and burnt by the reactor itself, see next slide…



(Modern reactors are built so that they can burn since the beginning admixtures of U and Pu (MOX) since there is “abundance” of Pu, from nuclear weapons which are being dismantled as the consequence of international treaties…)

Natural and man made fissile materials: Plutonium

- Plutonium is a radioactive actinide metal whose isotope, (Pu-239), is one of the three primary fissile isotopes (uranium-233 and uranium-235 are the other two).
- It is produced by n-capture on ^{238}U, followed by 2 beta decays

$$^{238}_{92}\mathrm{U} + {}^{1}_{0}\mathrm{n} \longrightarrow {}^{239}_{92}\mathrm{U} \xrightarrow[23.5\ \mathrm{min}]{\beta^-} {}^{239}_{93}\mathrm{Np} \xrightarrow[2.3565\ \mathrm{d}]{\beta^-} {}^{239}_{94}\mathrm{Pu}$$

- In a thermal reactor Pu is produced and it is also burnt (in part). Towards the end of its life, a uranium (not MOX, just uranium) PWR fuel element is producing more power from the fissioning of plutonium than from the remaining uranium-235
- Pu-239 is a key fissile component in nuclear weapons, due to its ease of fission and availability. Encasing the bomb's sphere of plutonium in a tamper (an optional layer of dense material) decreases the amount of plutonium needed to reach critical mass by reflecting escaping neutrons back into the plutonium core.
- This reduces the amount of plutonium needed to reach criticality from 16 kg to 10 kg, which is a sphere with a diameter of 10 cm. This critical mass is about a third of that for U-235…

Fissile and Fertile materials

- Note also that 238U is not a fissile material, but it can be transformed into a fissile one by neutron induced processes

$$ {}^{238}_{92}\mathrm{U} + {}^{1}_{0}\mathrm{n} \longrightarrow {}^{239}_{92}\mathrm{U} \xrightarrow[23.5\ \mathrm{min}]{\beta^-} {}^{239}_{93}\mathrm{Np} \xrightarrow[2.3565\ \mathrm{d}]{\beta^-} {}^{239}_{94}\mathrm{Pu} $$

  Thus it is termed “fertile” material.

- Same is true for 232 Th

$$ {}^{232}\mathrm{Th} + \mathrm{n} \rightarrow {}^{233}\mathrm{Th} \xrightarrow{\beta^-} {}^{233}\mathrm{Pa} \xrightarrow{\beta^-} {}^{233}\mathrm{U} $$

- Note that the fissile ^{235}U is just 0.7% of Uranium in the Earth and Earth contains 4 times Th than U. If we could burn ^{238}U and Th we would have much more fuel…
- If one has a device where (more) than two neutrons are availble after each fission, than one can transform fertile material into fissile material and burn it ….

Breeders

- A **breeder reactor** is a nuclear reactor that generates new fissile or fissionable material at a greater rate than it consumes such material.
- Two types of traditional breeder reactor have been proposed:

- **fast breeder reactor** or FBR.

The superior neutron economy of a fast neutron reactor makes it possible to build a reactor that, after its initial fuel charge of plutonium, requires only natural (or even depleted) uranium feedstock as input to its fuel cycle. This fuel cycle has been termed the plutonium economy

- **thermal breeder reactor**.

The excellent neutron capture characteristics of fissile uranium-233 make it possible to build a moderated reactor that, after its initial fuel charge of enriched uranium, plutonium or MOX, requires only thorium as input to its fuel cycle. thorium-232 produces uranium-233 after neutron capture and beta decay.

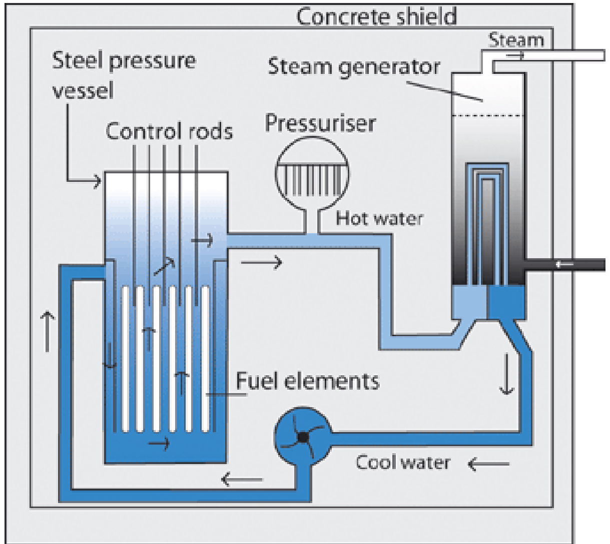
β^-

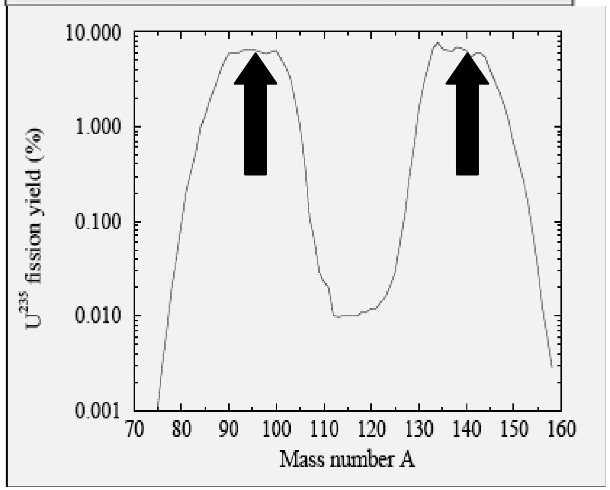
I reattori nucleari come sorgenti di neutrini

- I tipici reattori commerciali hanno potenze termiche di 3GW e utilizzano Uranio arricchito ^{235}U al 3%
- In media, ciascuna fissione nucleare produce Δ=200MeV quindi un tipico reattore produce 10^{20} fis/s
- E' facile comprendere che il numero medio di (anti) neutrini e' 6 per fissione. Nella fissione indotta da neutroni si ha

$$_{92}U^{235} + n \rightarrow X1 + X2 + 2n$$

- La distribuzione dei prodotti di fissione e' piccata intorno a A= 94 e A =140; per questi numeri di massa i nuclei stabili sono $_{40}Zr^{94}$ e $_{58}Ce^{140}$. Per raggiungere questi nuclei, in cui la carica totale e' 98, partendo da 92 protoni, e' necessario che 6 protoni si trasformino in neutroni, e dunque si hanno 6 decadimenti beta, ossia sei antineutrini.
- Dunque un reattore con potenza termica di 3GW produce $L_\nu \approx 6 \times 10^{20}$ neutrini/s, in maniera isotropa.
- A una distanza di 10 m il flusso e' Φ= 5 10^{13} anti ν /.cm^2/s



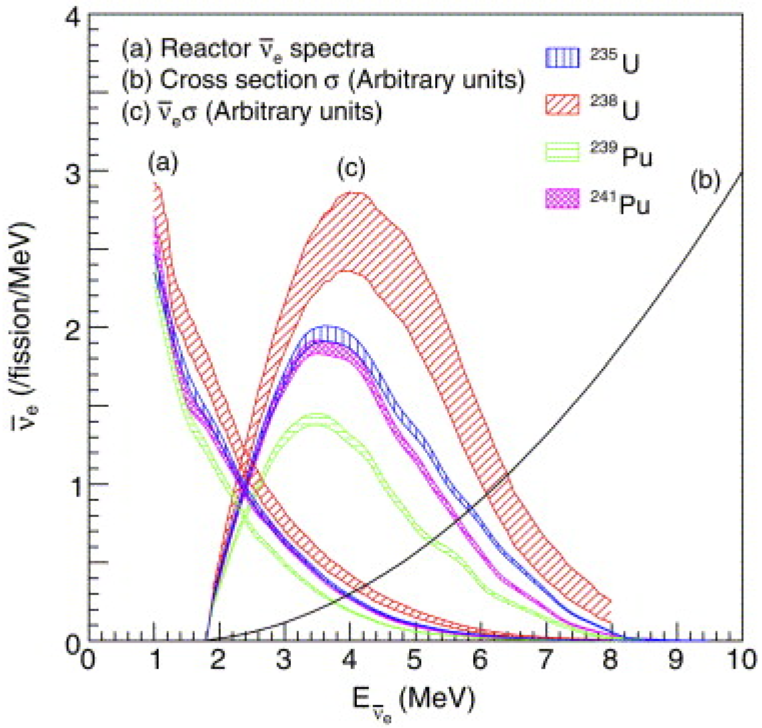


Lo spettro degli anti neutrini da reattore

- I neutrini della fissione portano via mediamente 1.6 MeV, il che vuol dire che dei 200 MeV di ciascuna fissione il 6% non va in calore, ma in neutrini
- I neutrini piu’ abbondanti sono quelli provenienti dalla fissione con neutroni termici dell’235U, ma sono importanti anche quelli provenienti da ^{238}U, nonche’ da due isotopi del Plutonio, ^{239} Pu e ^{241}Pu, prodotti attraverso lo schema

$$^{238}\mathrm{U}(\mathrm{n},\gamma)^{239}\mathrm{U} \underset{E_{\max}=1.265\ \mathrm{MeV}}{\overset{T_{1/2}=23.5\ \min}{\longrightarrow}} {}^{239}\mathrm{Np}$$

$$\underset{E_{\max}=0.722\ \mathrm{MeV}}{\overset{T_{1/2}=2.357\ \mathrm{d}}{\longrightarrow}} {}^{239}\mathrm{Pu}(\mathrm{n},\gamma)^{240}\mathrm{Pu}(\mathrm{n},\gamma)^{241}\mathrm{Pu}.$$



•La figura mostra lo spettro delle varie componenti, con le relative incertezze, e il prodotto di questo con la sezione d’urto per la reazione tipica usata per la rivelazione (con soglia a 1.8 MeV

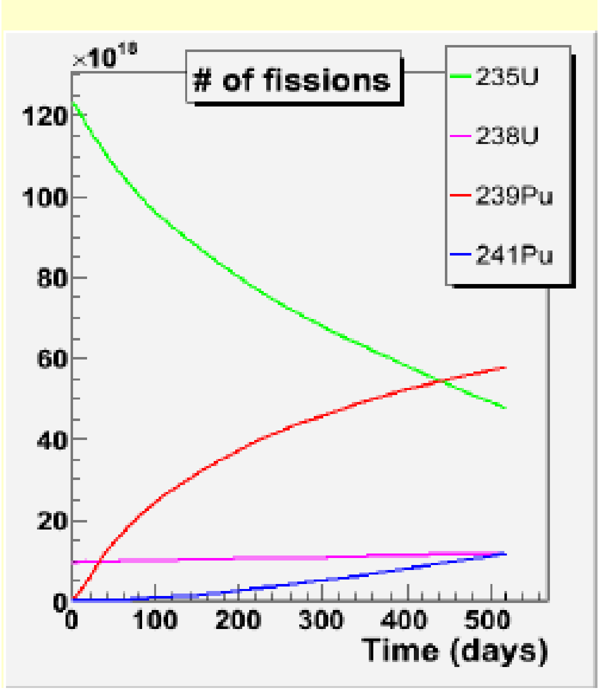
.

$$\bar{\nu}_e + p \rightarrow e^- + n.$$

•Da notare il picco intorno a 4 MeV, cioe’ l’energia alla quale si trovano il maggior numero dei neutrini rivelati

Reactor neutrino monitoring ?

- Reactors have been used for decades in order to study neutrino properties, including oscillations.
- Now that we have learnt about neutrinos it is time to exploit them.
- People is considering the possibility of monitoring reactors (in particular Pu removal) by using neutrino detectors placed around it
- Several projects in US, France and also in Italy
- IAEA controls the reactor power to monitor **Pu** extraction from reactor core...
- …but by inserting the right content of U one can extract Pu without changing the power...
- …On the other hand, nobody can stop neutrinos and a change in composition corresponds to a change in neutrino flux



- 1.1 GW_{el} → ~10^{21} ν/s
- 3800 int. expected per day in $1m^3$ liq. scint. target with ε_{det} = *100%*

Se son rose fioriranno

Grazie dell’attenzione

s
t

s
t
a
b
l
e

i
s
o
t
o
p
e
s

Uranium Isotopes

iso	**NA**	**half-life**	**DM**	**DE (MeV)**	**DP**
^{232}U	syn	68.9 y	α & SF	5.414	^{228}Th
^{233}U	syn	159,200 y	SF & α	4.909	^{229}Th
^{234}U	0.0054%	245,500 y	SF & α	4.859	^{230}Th
^{235}U	0.7204%	7.038×10^{8} y	SF & α	4.679	^{231}Th
^{236}U	syn	2.342×10^{7} y	SF & α	4.572	^{232}Th
^{238}U	99.2742%	4.468×10^{9} y	SF & α	4.270	^{234}Th

Plutonium isotopes

iso	NA	half-life	DM	DE (MeV)	DP
^{238}Pu	syn	88 y	SF	—	—
			α	5.5	^{234}U
^{239}Pu	syn	2.41×10^4 y	SF	—	—
			α	5.245	^{235}U
^{240}Pu	syn	6.5×10^3 y	SF	—	—
			α	5.256	^{236}U
^{241}Pu	syn	14 y	β^-	0.02078	^{241}Am
			SF	—	—
^{242}Pu	syn	3.73×10^5 y	SF	—	—
			α	4.984	^{238}U
^{244}Pu	trace	8.08×10^7 y	α	4.666	^{240}U
			SF	—	—

Thorium Isotopes

iso	NA	half-life	DM	DE (MeV)	DP
^{228}Th	syn	1.9116 years	α	5.520	^{224}Ra
^{229}Th	syn	7340 years	α	5.168	^{225}Ra
^{230}Th	syn	75380 years	α	4.770	^{226}Ra
^{231}Th	trace	25.5 hours	β	0.39	^{231}Pa
^{232}Th	100%	1.405×10^{10} years	α	4.083	^{228}Ra
^{234}Th	trace	24.1 days	β	0.27	^{234}Pa