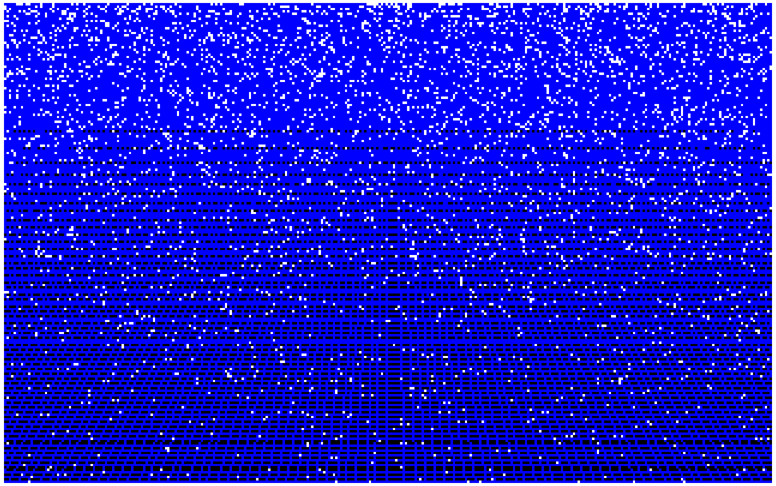
Energy Science

FEW Cursus



Jo van den Brand

April 20, 2010

Overzicht

Algemene ontwikkeling
Tentamenstof
Ter informatie

- **Docent informatie**
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- **Rooster informatie**
  - Dinsdag 13:30 – 15:15 in S655 (totaal 8 keer); HC vdB
  - Donderdag 15:30 – 17:15 in S345 (totaal 7 keer); WC Roel Aaij
- **Boek en dictaat**
  - Andrews & Jelley, Hoofdstukken 8 en 9
  - Zie website voor pdf van dictaat
- **Cijfer**
  - Huiswerk 20%, tentamen 80%

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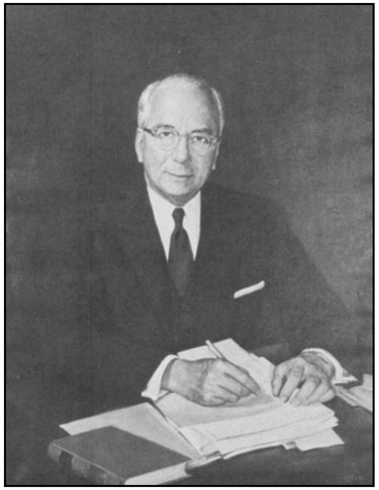
Introduction

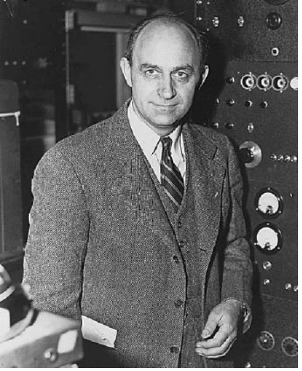
Nuclear power today:

- Provides 16% of world's electricity
  - 19% in U.S.
  - 77% in France
  - 54% Belgium
  - 26% Germany
  - 46% Sweden
  - 4% Netherlands
- 69% of U.S. non-carbon electricity generation
- About 439 plants in the world
  - 147 in European Union (200+ in Europe)
  - 104 in U.S.
    - ❑ None built in US since the 1970s
    - ❑ Small budgets for nuclear R&D

Lewis Strauss, Chairman of the U.S. Atomic Energy Commission (1954

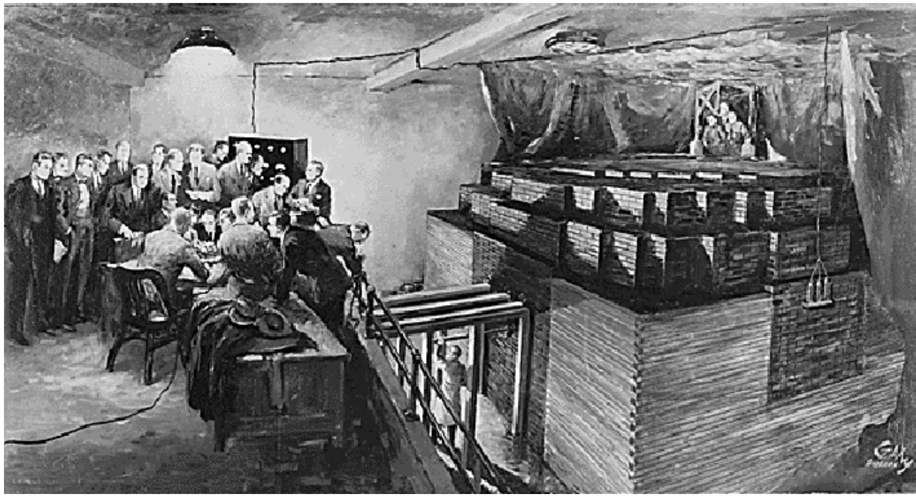


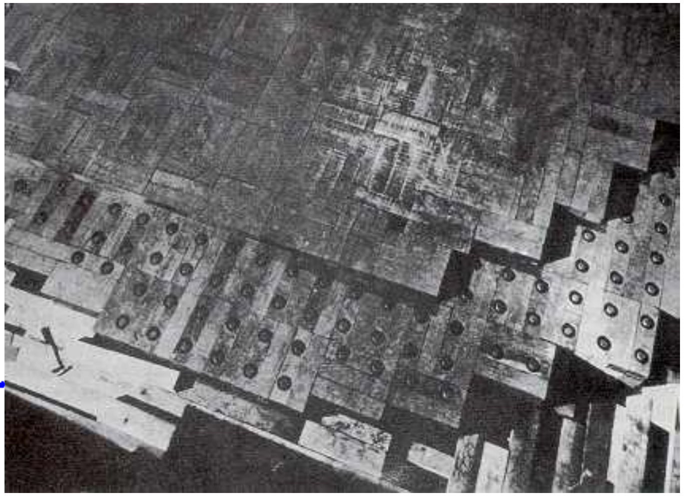
"It is not too much to expect that our children will enjoy in their homes [nuclear generated] electrical energy too cheap to meter."

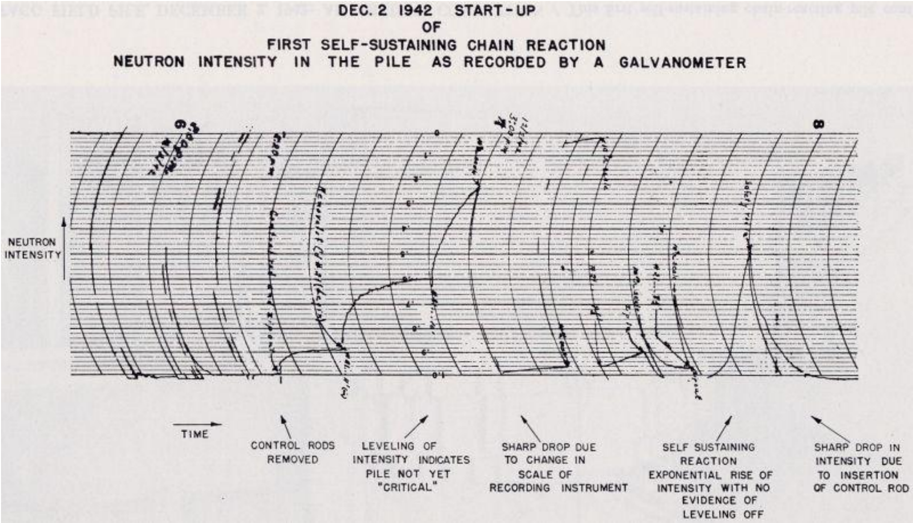


Early beginnings

- **Enrico Fermi**
- **Chicago, Dec. 2, 1942**
- **Criticality reached**







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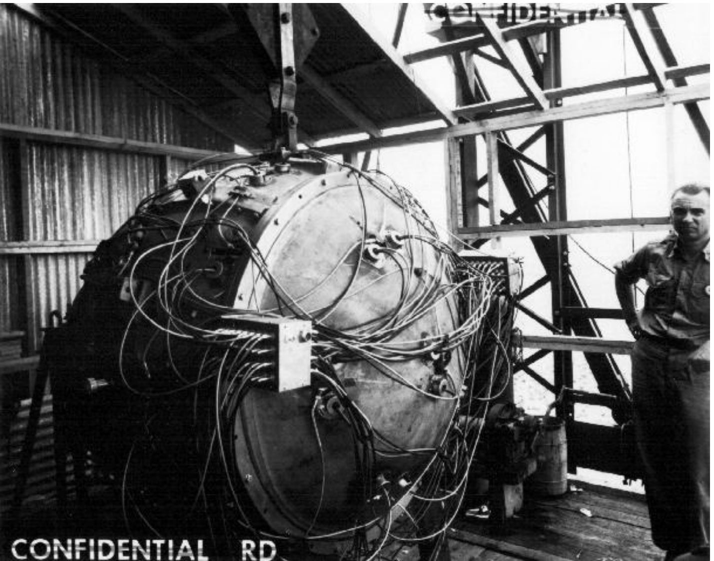
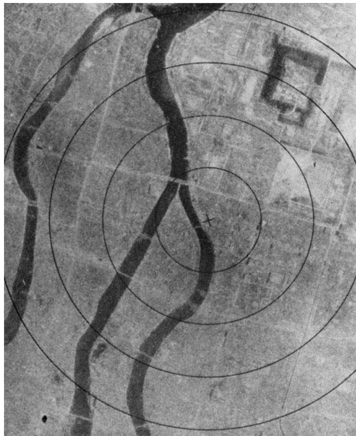
Early beginnings

- **Manhattan project**
- **Plutonium production**
- **Reactor B in Hanford**
- **Trinity: the gadget**
- **Nagasaki bomb**





Early beginnings

- Atomic Energy Commission (AEC) established by Congress in 1946 as part of the Atomic Energy Act
- AEC authorized the construction of Experimental Breeder Reactor I (EBR-1) at a site in Idaho in 1949
- In August of 1951, criticality (a controlled, self-sustained, chain reaction) was reached using uranium
- A football sized core was created and kept at low power for four months until December 20, 1951
- Power was gradually increased until the first usable amount of electricity was generated, lighting four light bulbs and introducing nuclear generated power for the first time
- In 1953, the EBR-1 was creating one new atom of nuclear fuel for every atom burned, thus the reactor could sustain its own operation
- With this creation of new cores, enough energy was created to fuel additional reactors
- A few years later, the town of Arco, Idaho became the world's first community to get its entire power supply from a nuclear reactor
- This was achieved by temporarily attaching the town's power grid to the reactor's turbines

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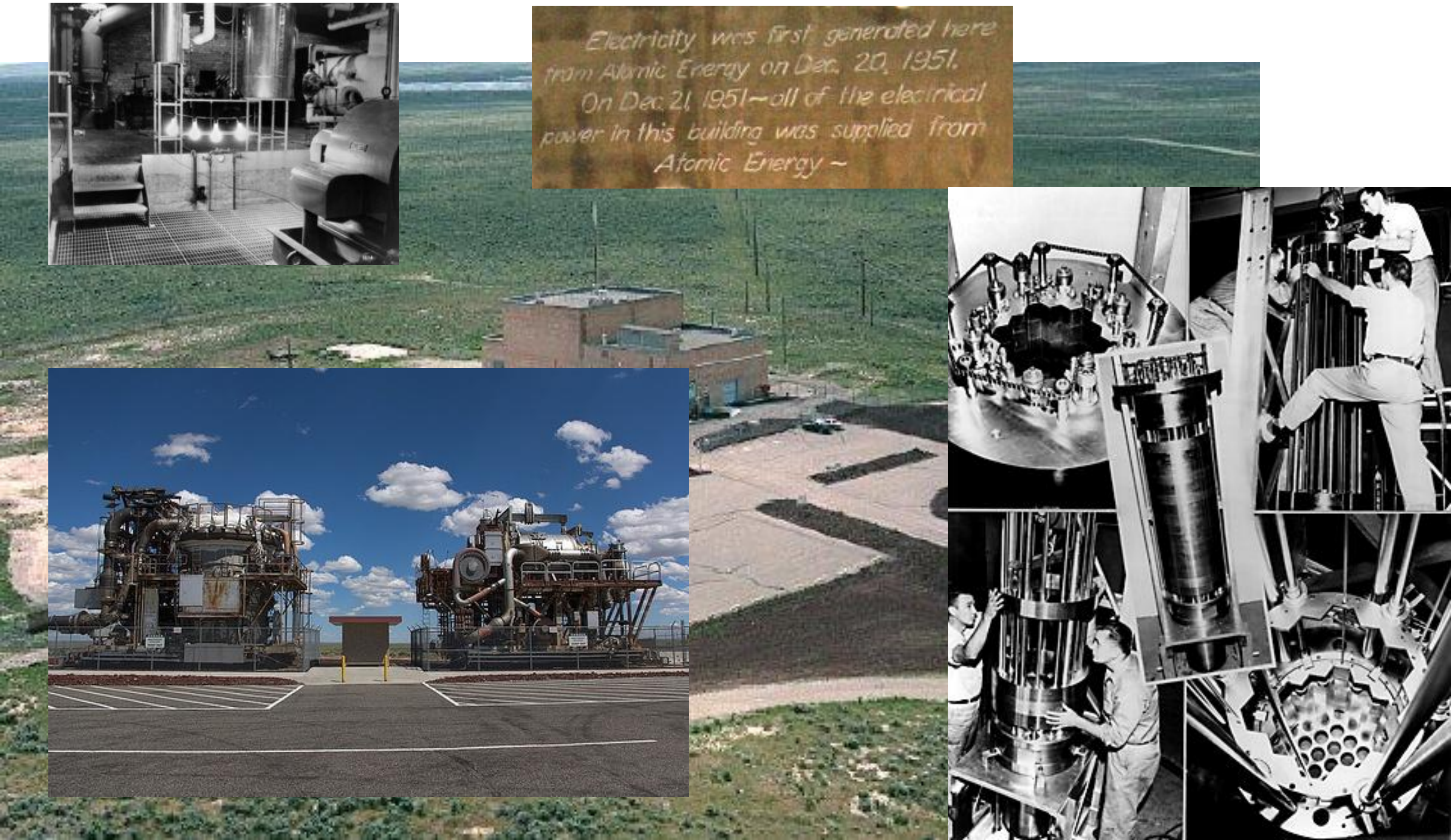
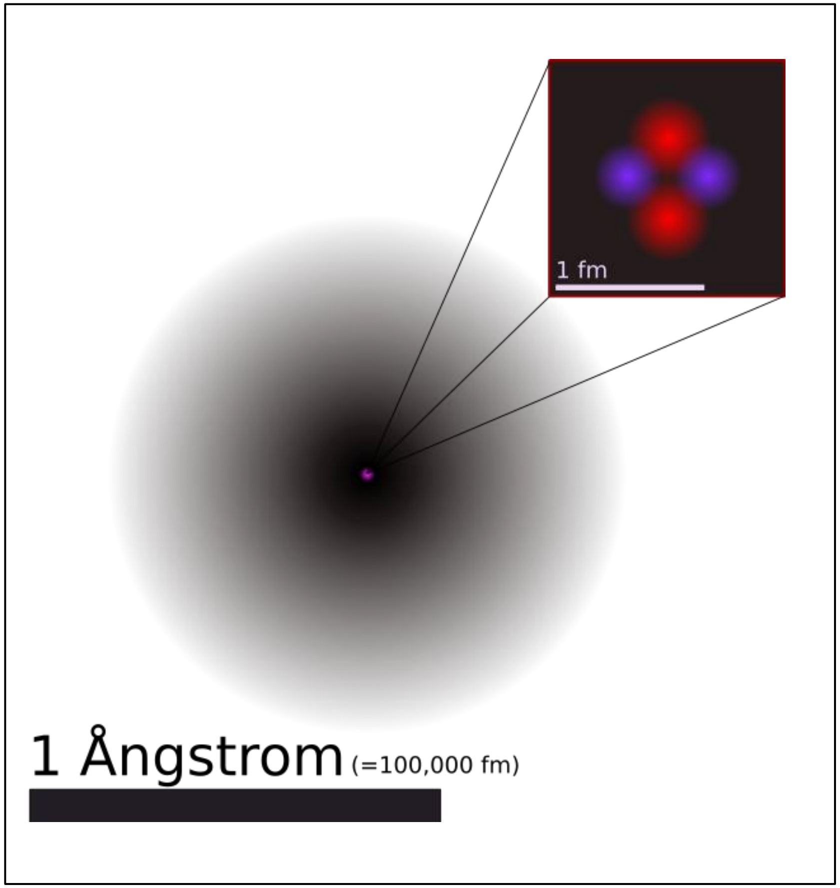
EBR – 1 at Idaho (1951)



Basic knowledge about atom



Atoms

- The smallest particle (10^{-10} m or 0.1 nm) of an element that still retains the characteristics of the element.
- An atom has a very small size (10^{-15} m) nucleus surrounded by negatively charged electrons.
- Nucleus consists of protons (positively charged) and neutrons.
- Proton and neutron have about the same mass that is very larger than electron's. So an atom's mass is essentially the mass of its nucleus.

Binding energy and nuclear stability

The number of protons (the atomic number) determines which element it belongs to.

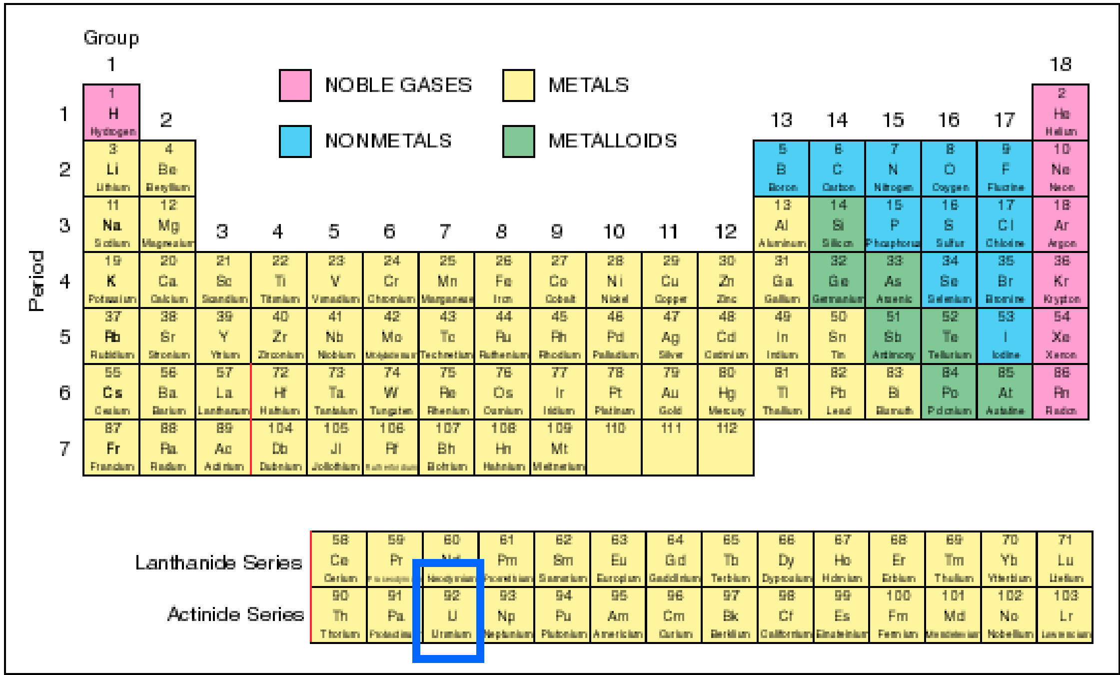
Nuclide: a specific nuclear configuration $^{A}_{Z}X$

Isotopes: atoms with the same atomic number but different number of neutrons, e.g., ^{235}U and ^{238}U.

$^{232}_{92}U$	Uranium – 232
$^{233}_{92}U$	Uranium – 233
$^{234}_{92}U$	Uranium – 234
$^{235}_{92}U$	Uranium – 235
$^{236}_{92}U$	Uranium – 236
$^{237}_{92}U$	Uranium – 237
$^{238}_{92}U$	Uranium – 238

Electron configuration
Use diagonal rule

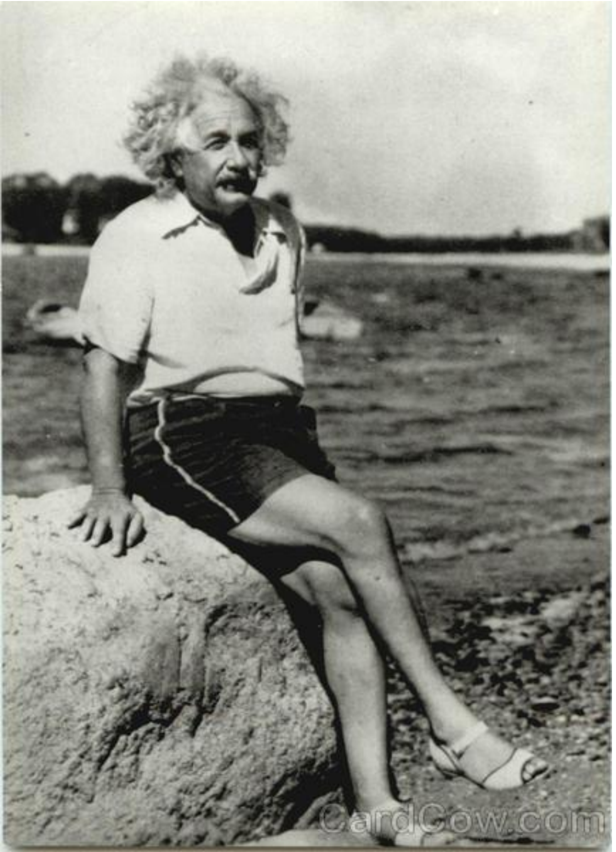
1s			
2s	2p		
3s	3p	3d	
4s	4p	4d	4f
5s	5p	5d	5f
6s	6p	6d	
7s			



Binding energy

Nuclear force needed to overcome repulsive force between protons

It is strong and short-ranged



- MASS DEFECT

  $M_{Atom} < \sum M_{Constituents}$

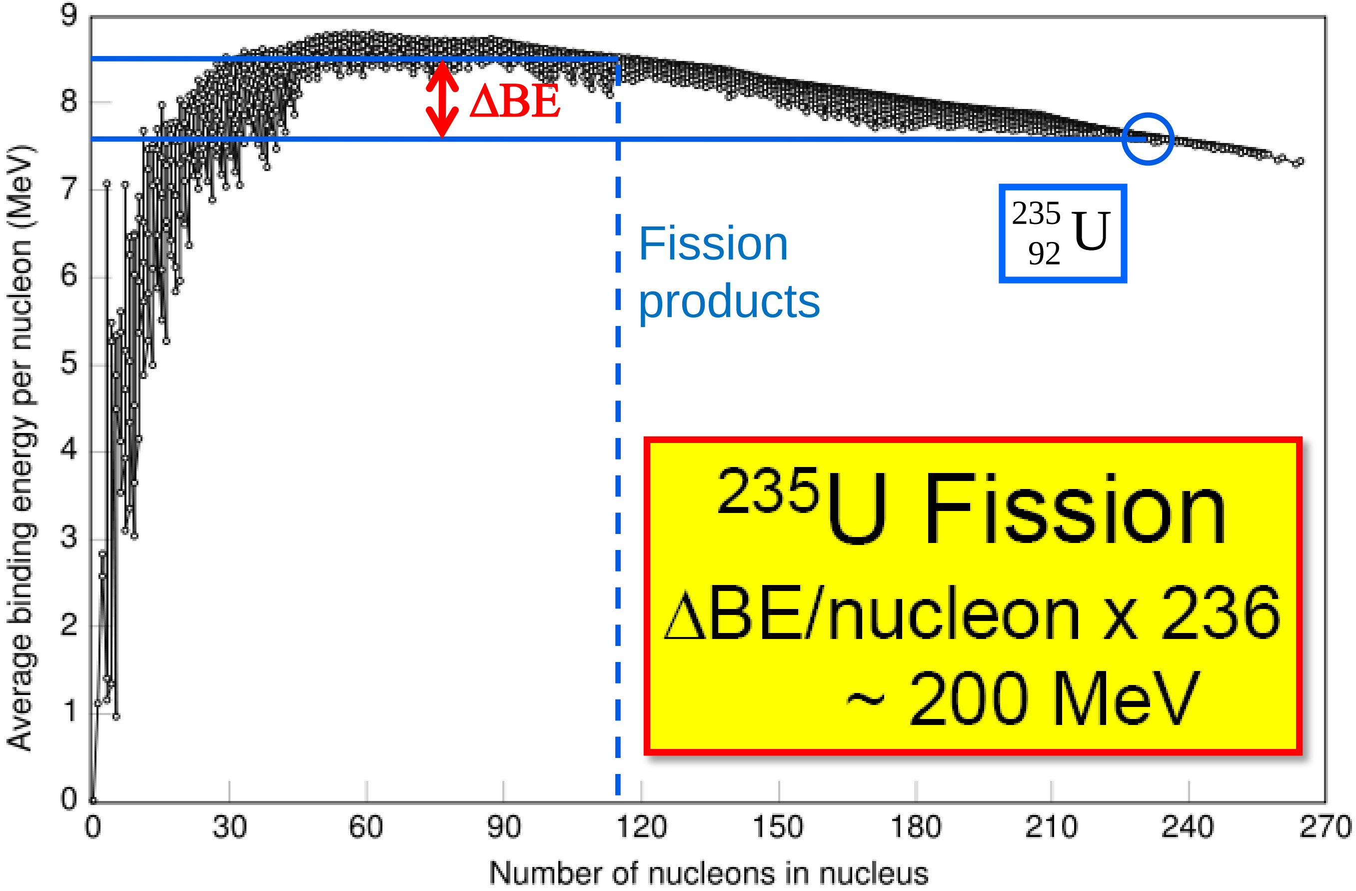
- MASS-ENERGY EQUIVALENCE

  $E = mc^2$

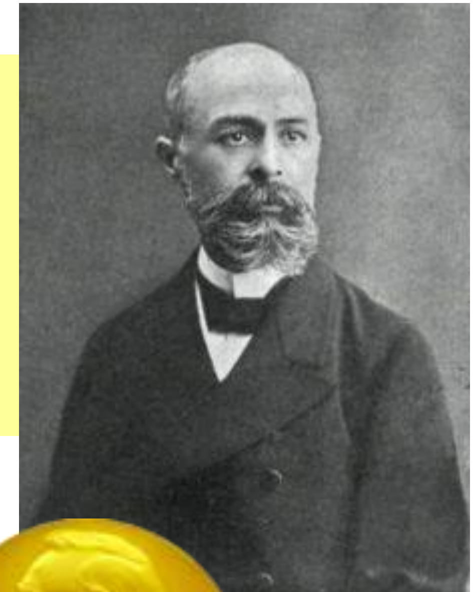
- BINDING ENERGY

  $BE = [\, M_{Atom} - \sum M_{Constituents} \,]c^2$

Binding energy per nucleon



Radioactive decay



Some elements will spontaneously turn into other elements. This is called *radioactivity* and was discovered in 1896 by Henri Becquerel.

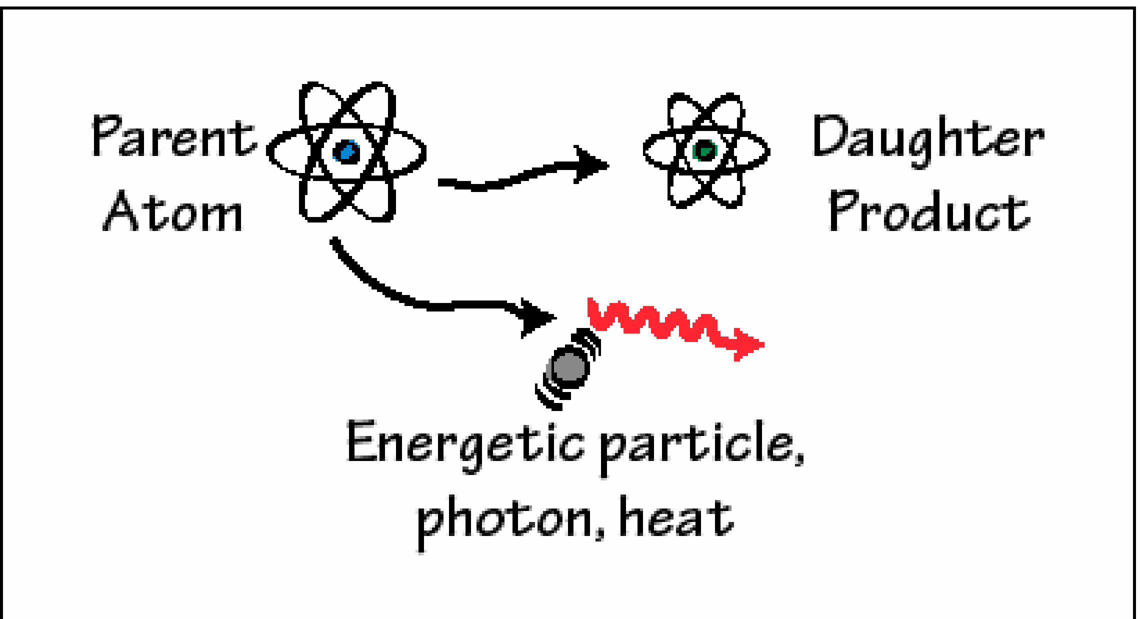
It happens randomly and the *probability* only depends on the structure of the nucleus (isotope).

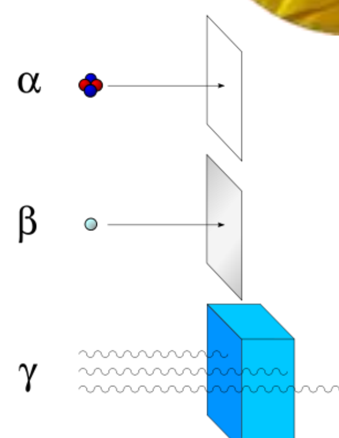
Scientists use *half-life* to describe the probability of decay.

$$dN = -\lambda N(t)dt \rightarrow N(t) = N_0 e^{-\lambda t}$$

$$\tau = 1/\lambda \text{ en } \tau_{1/2} = \tau \ln 2$$

1903





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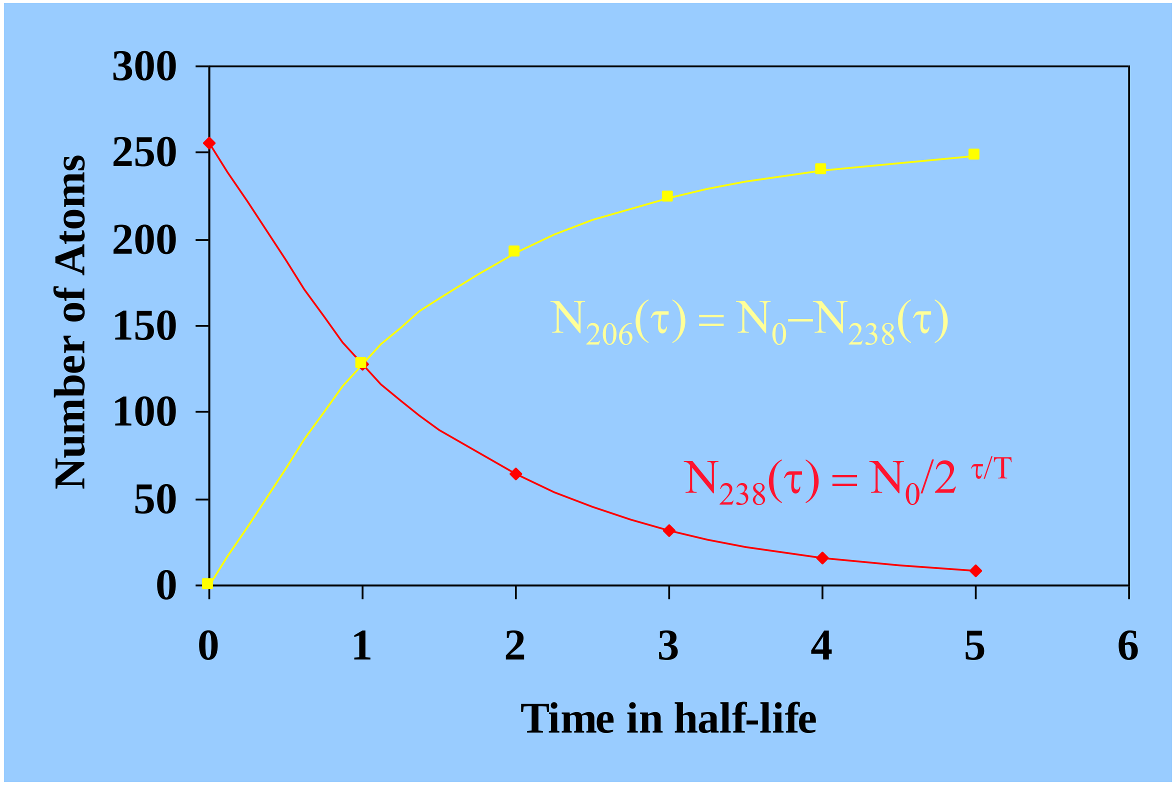
Radioactive decay

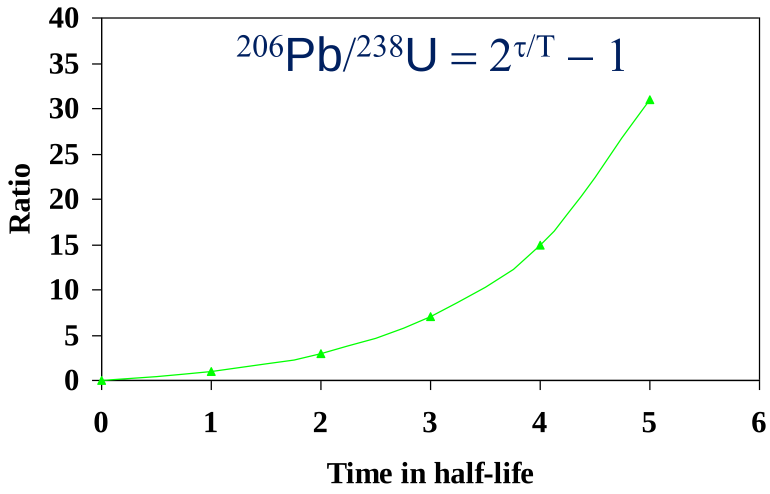
Example: $^{238}U \longrightarrow {}^{206}Pb + 8\ {}^{4}He + 6ß$

The half-life $T_{1/2}$ of ^{238}U is about 4.5 billion years.

The graph shows the number of ^{238}U and ^{206}Pb at time t



Radioactive dating



This can be used in the opposite way: if we can count how much daughter isotope in the sample as compared to the parent isotope we can get the age of the sample

$$t = T \log_2 (^{206}Pb/^{238}U + 1)$$

This method is called radioactive dating.

By using it, we find that Earth is ~4.5 Ga old.

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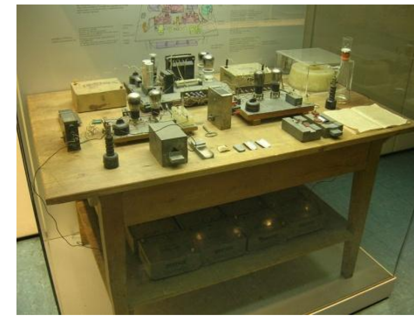
Neutron induced decay: fission of ^{235}U

Neutron induced nuclear fission of heavy nuclei was predicted by Enrico Fermi and discovered by Otto Hahn, Lise Meitner and Fritz Strassmann (Dec.1938)

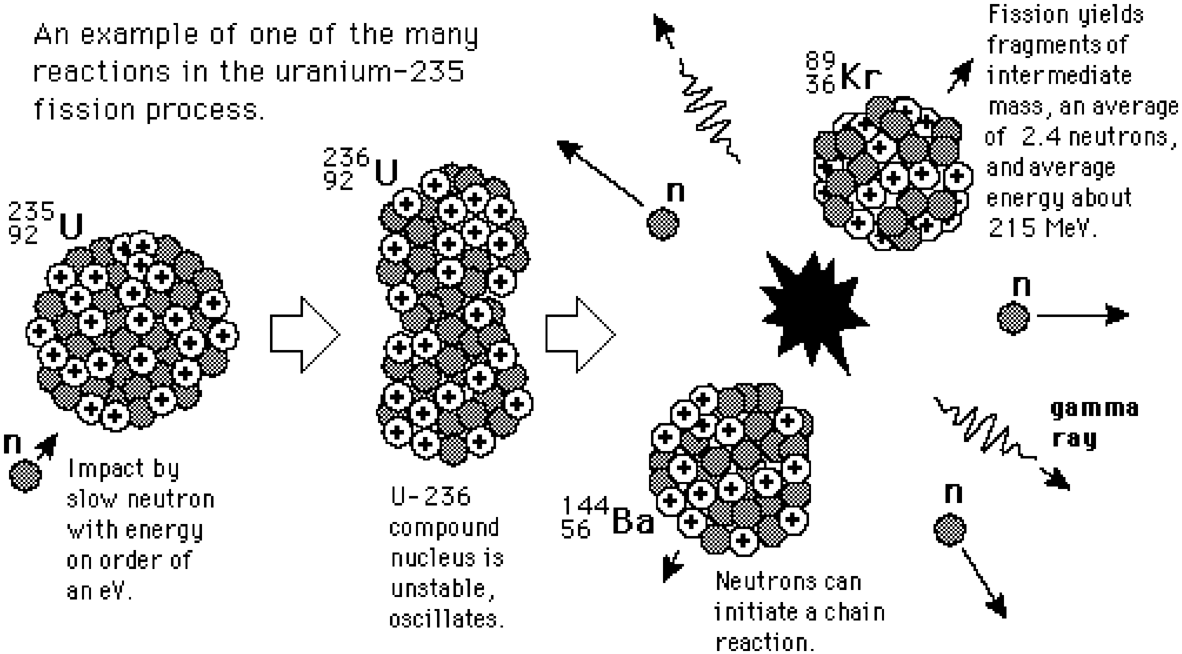
Change of elements: create gold from lead ...

Traces of Barium found

Energy release in fission of uranium



$$^{235}U + \text{neutron} \longrightarrow \text{fission fragments} + 2.4 \text{ neutrons} + 192.9 \text{ MeV}$$

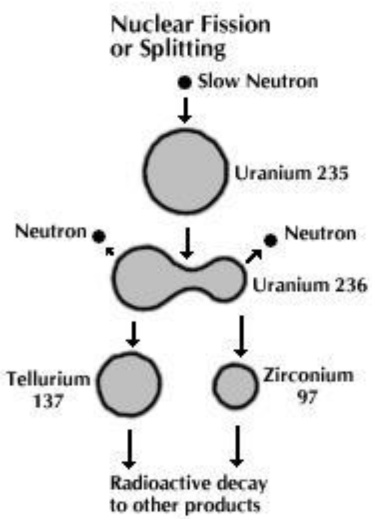


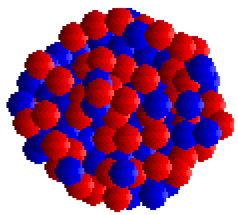
Uranium Ore (0.7%)

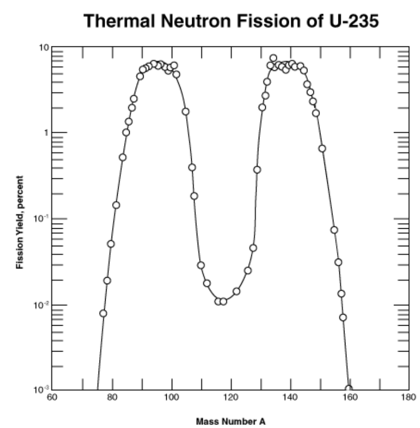
Fuel Pellet (3.5%)

Neutron induced decay: fission of ^{235}U

Fission of heavy nuclei be understood in terms of liquid drop model







Energy of excited state = BE of initial nucleus + kinetic energy of neutron

When $E_{exc} > E_{crit}$ then nucleus fissions

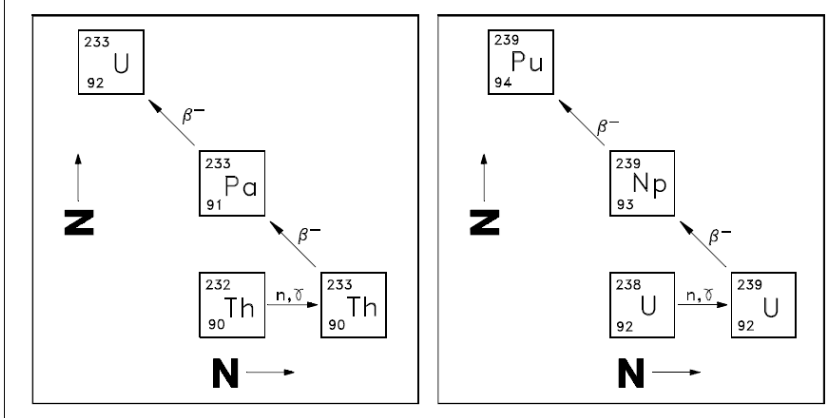
Excite oscillations by neutron capture

Critical Energies Compared to Binding Energy of Last Neutron

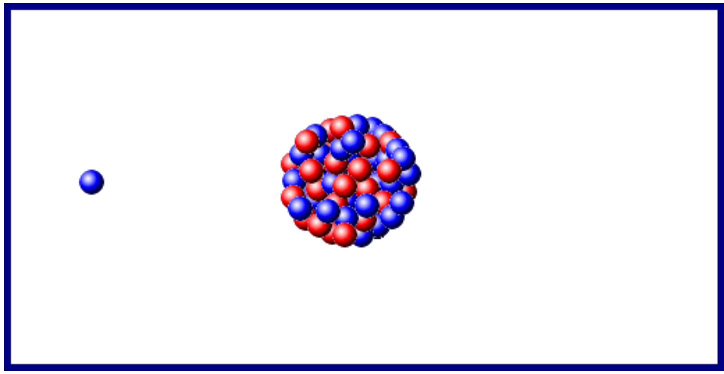
Target Nucleus	Critical Energy E_{crit}	Binding Energy of Last Neutron BE_n	$BE_n - E_{crit}$
$^{232}_{90}Th$			
$^{238}_{92}U$	7.0 MeV	5.5 MeV	-1.5 MeV
$^{235}_{92}U$	6.5 MeV	6.8 MeV	+0.3 MeV
$^{233}_{92}U$	6.0 MeV	7.0 MeV	+1.0 MeV
$^{239}_{94}Pu$	5.0 MeV	6.6 MeV	+1.6 MeV

Fission by thermal neutrons possible

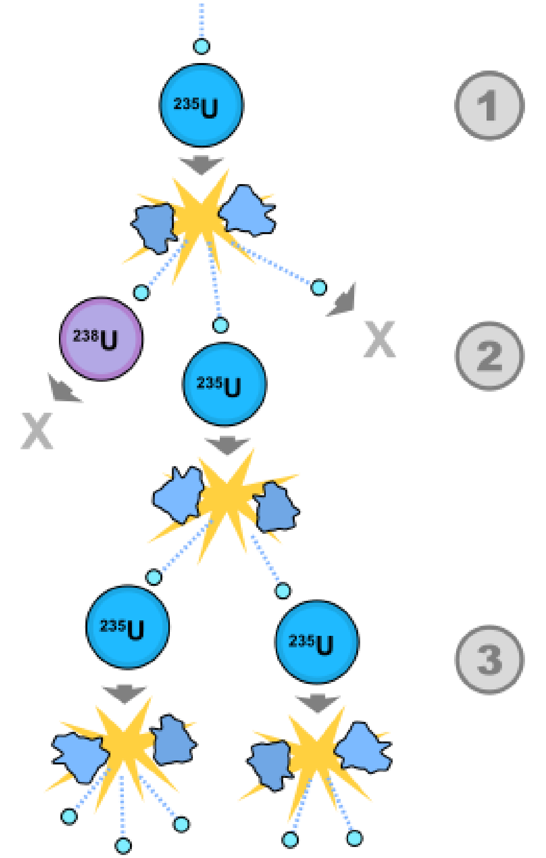
Breeding: Fertile material becomes fissile through transmutation



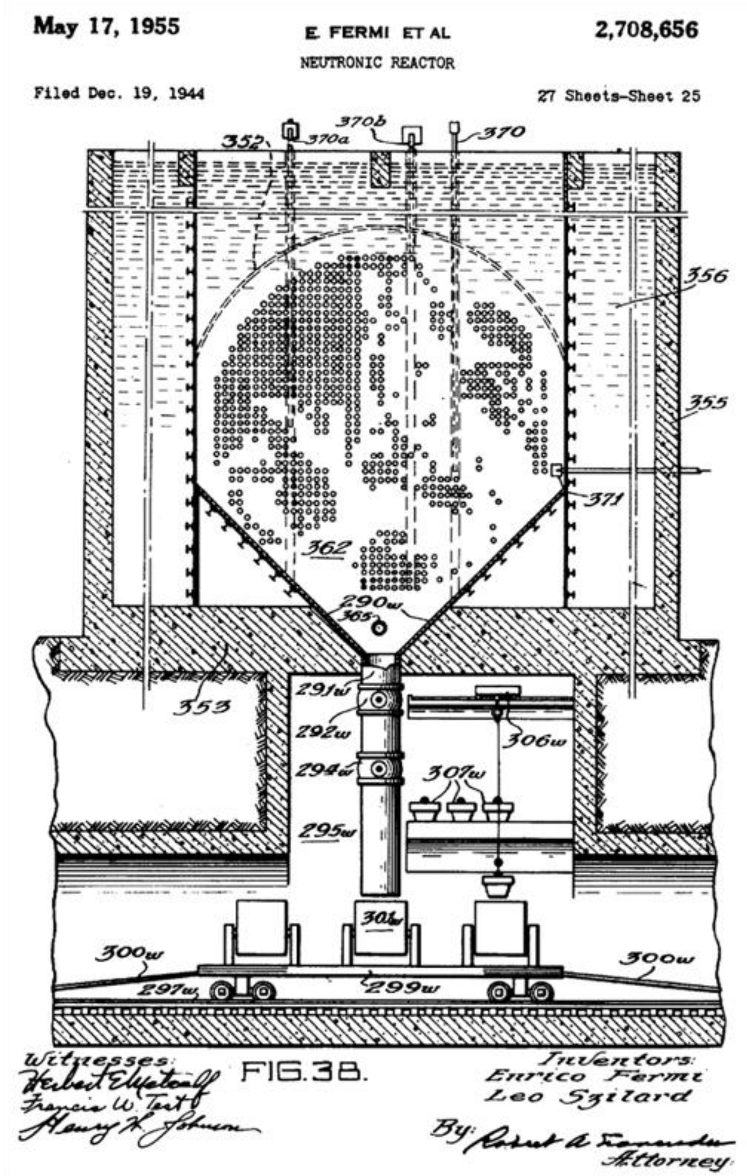
Nuclear chain reaction

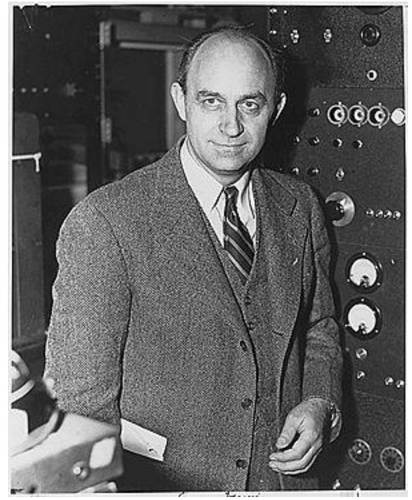


- For ^{235}U and ^{239}Pu, one fission takes one neutron and produces 2.4 neutrons on average. The produced neutrons can be used to split more ^{235}U and start a chain reaction
- To sustain the chain reaction, it is necessary to have many fissionable atoms around to catch neutrons.
- The smallest amount of fissile material is called *critical* mass.
- The critical mass depends on the density of the material and how easy for ^{235}U to capture a neutron.
- Inside a nuclear power plant, the reaction is controlled by absorbing neutrons and moderating their speed.
- Slow neutrons can be captured more easily so nuclear power plants can use low-grade uranium (a few per cent in ^{235}U) as fuel.
- Fission of one ^{235}U atom releases energy of 235×0.85 MeV/nucleon = $235 \times 0.85 \times 1\ 10^{6} \times 1.6\ 10^{-19} = \boxed{3.2\ 10^{-11}\ J}$.
- 235 gram of ^{235}U has $6.0\ 10^{23}$ atoms (Avogadro's number). Thus 1 kg of uranium enriched to 3% in ^{235}U has a yield of
  $0.03 \times 1{,}000/235 \times 6.0\ 10^{23} \times 3.2\ 10^{-11} = 2.4\ 10^{12}$ J =2,450 GJ.
- Coal: $C + CO \longrightarrow CO_2 + 4.2\ eV = \boxed{7\ 10^{-19}\ J}$
  50 million times less per atom than nuclear fission.

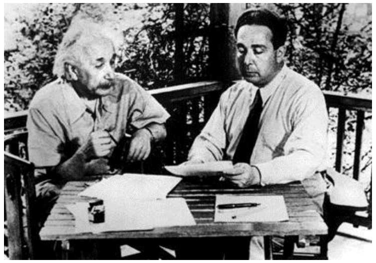


Patents





Enrico Fermi & Leo Szilard (1955)
US PATENT 2,708,656



Patented Nov. 11, 1930 1,781,541

UNITED ST

ALBERT EINSTEIN, OF BERLIN
MANY, ASSIGNORS TO ELECT
CORPORATION OF DELAWAR

Application filed December 16, 1

Our invention relates to the ar eration and particularly to an ap method for producing refrigerati the refrigerant evaporates in the an inert gas and more particularly disclosed in Patent No. 1,685,764 g tember 25th, 1928, to Von Platen a and our British Patent No. 282,42

Patented Oct. 27, 1936 2,058,562

UNITED STATES PATENT OFFICE

2,058,562

LIGHT INTENSITY SELF-ADJUSTING CAMERA

Gustav Bucky, New York, N. Y., and Albert Einstein, Princeton, N. J.

Application December 11, 1935, Serial No. 53,884

9 Claims. (Cl. 95—10)

This invention relates to a camera with a device for automatically adjusting the light intensity, and an object of the invention is to provide means for automatically adapting the light impinging the photographic plate or film of the camera to the light intensity of the surroundings and particularly of the object to be photographed.

A further object of the invention is to provide means for an automatic adjustment of the light path of the light rays passing through objective 12, and it is furthermore so arranged that the screen portion with maximum transparency is in the path of the rays when the photo-electric cell is not or only very little energized, that means when the light intensity is weakest. The plane in which the screen intersects these light rays is immaterial. We, however, prefer to position the screen between the lenses 13 and 14 for which

Reactor theory: neutron cross sections

- Atom density (N) is the number of atoms of a given type per unit volume of material.
- Microscopic cross section (σ) is the probability of a given reaction occurring between a neutron and a nucleus.
- Microscopic cross sections are measured in units of barns, where 1 barn = 10^{-24} cm^2.
- Macroscopic cross section (Σ) is the probability of a given reaction occurring per unit length of travel of the neutron. The units for macroscopic cross section are cm^{-1}.

$$\mathbf{N} = \frac{\rho\,\mathbf{N_A}}{\mathbf{M}}$$

where:

N = atom density (atoms/cm^3)
ρ = density (g/cm^3)
N_A = Avogadro's number (6.022×10^{23} atoms/mole)
M = gram atomic weight

$$\Sigma = N\,\sigma$$

where:

Σ = macroscopic cross section (cm^{-1})
N = atom density of material (atoms/cm^3)
σ = microscopic cross-section (cm^2)

$$\sigma_T = \sigma_a + \sigma_s$$

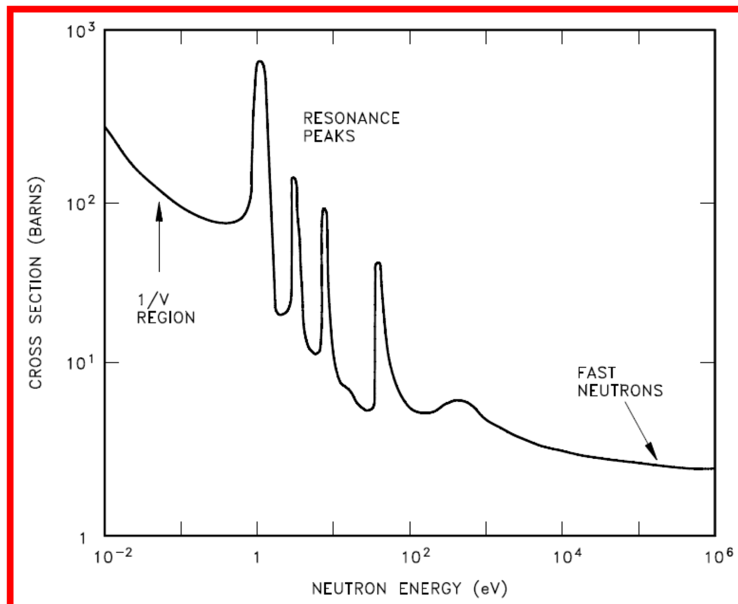
Total: absorption + scattering

$$\sigma_s = \sigma_{se} + \sigma_{si}$$

Scattering: elastic + inelastic

$$\sigma_a = \sigma_f + \sigma_c$$

Absorption: fission + capture



Reactor theory: neutron flux and mfp

- The *mean free path* (λ) is the average distance that a neutron travels in a material between interactions.
- Neutron *flux* ($\phi = nv$) is the total path length traveled by all neutrons in one cubic centimeter of material during one second.
- The mean free path equals $1/\Sigma$.
- The macroscopic cross section for a mixture of materials can be calculated using the equation

$$\Sigma = N_1\sigma_1 + N_2\sigma_2 + N_3\sigma_3 + \ldots\ldots\ N_n\sigma_n$$

- Self-shielding is where the local neutron flux is depressed within a material due to neutron absorption near the surface of the material.

$$\phi = n\,v$$

where:

ϕ = neutron flux (neutrons/cm^2-sec)
n = neutron density (neutrons/cm^3)
v = neutron velocity (cm/sec)

Example

An alloy is composed of 95% aluminum and 5% silicon (by weight). The density of the alloy is 2.66 g/cm^3. Properties of aluminum and silicon are shown below.

Element	Gram Atomic Weight	$\sigma_{a\,(\text{barns})}$	$\sigma_{s\,(\text{barns})}$
Aluminum	26.9815	0.23	1.49
Silicon	28.0855	0.16	2.20

Atom densities

$$N_{Al} = \frac{\rho_{Al}\,N_A}{M_{Al}} = \frac{0.95\left(2.66\ \frac{g}{cm^3}\right)\left(6.022 \times 10^{23}\ \frac{atoms}{mole}\right)}{26.9815\ \frac{g}{mole}} = 5.64 \times 10^{22}\ \frac{atoms}{cm^3}$$

$$N_{Si} = \frac{\rho_{Si}\,N_A}{M_{Si}} = \frac{0.05\left(2.66\ \frac{g}{cm^3}\right)\left(6.022 \times 10^{23}\ \frac{atoms}{mole}\right)}{28.0855\ \frac{g}{mole}} = 2.85 \times 10^{21}\ \frac{atoms}{cm^3}$$

Macr. cross sections

$$\Sigma_a = N_{Al}\,\sigma_{a,Al} + N_{Si}\,\sigma_{a,Si} = \left(5.64 \times 10^{22}\ \frac{atoms}{cm^3}\right)\left(0.23 \times 10^{-24}\ cm^2\right) + \left(2.85 \times 10^{21}\ \frac{atoms}{cm^3}\right)\left(0.16 \times 10^{-24}\ cm^2\right) = 0.0134\ cm^{-1}$$

$$\Sigma_s = N_{Al}\,\sigma_{s,Al} + N_{Si}\,\sigma_{s,Si} = \left(5.64 \times 10^{22}\ \frac{atoms}{cm^3}\right)\left(1.49 \times 10^{-24}\ cm^2\right) + \left(2.85 \times 10^{21}\ \frac{atoms}{cm^3}\right)\left(2.20 \times 10^{-24}\ cm^2\right) = 0.0903\ cm^{-1}$$

Mfp

$$\lambda_a = \frac{1}{\Sigma_a} = \frac{1}{0.01345\ cm^{-1}} = 74.3\ cm$$

$$\lambda_s = \frac{1}{\Sigma_s} = \frac{1}{0.0903\ cm^{-1}} = 11.1\ cm$$

Reactor theory: reaction rates

- The *reaction rate* is the number of interactions of a particular type occurring in a cubic centimeter of material in a second.
- The reaction rate can be calculated by the equation R = $\phi\Sigma$.
- Over a period of several days, while the atom density of the fuel can be considered constant, the neutron flux is directly proportional to *reactor power*.

$$R = \phi \Sigma$$

where:

R = reaction rate (reactions/sec)
ϕ = neutron flux (neutrons/cm^2-sec)
Σ = macroscopic cross section (cm^{-1})

Substituting the fact that $\Sigma = N\sigma$

$$\mathbf{R} = \phi \mathbf{N} \sigma$$

where:

σ = microscopic cross section (cm^2)
N = atom density (atoms/cm^3)

Example:

If a one cubic centimeter section of a reactor has a macroscopic fission cross section of 0.1 cm^{-1}, and if the thermal neutron flux is 10^{13} neutrons/cm^2-sec, what is the fission rate in that cubic centimeter?

Solution:

$$R_f = \phi \Sigma_f$$

$$= \left(1 \times 10^{13} \frac{\text{neutrons}}{\text{cm}^2\text{-sec}}\right)\left(0.1 \text{ cm}^{-1}\right)$$

$$= 1 \times 10^{12} \frac{\text{fissions}}{\text{cm}^3\text{-sec}}$$

In a reactor where the average energy per fission is 200 MeV, it is possible to determine the number of fissions per second that are necessary to produce one watt of power using the following conversion factors.

Fission of one ^{235}U atom releases energy of 235 × 0.85 MeV/nucleon = 235 × 0.85 × 1 10^6 × 1.6 10^{-19} = 3.2 10^{-11} J.
Thus 1 Watt requires 1/ 3.2 10^{-11} = 3.12 10^{10} fissions per second.

Power released in a reactor

$$P = \frac{\phi_{th} \Sigma_f V}{3.12 \times 10^{10} \frac{\text{fissions}}{\text{watt-sec}}}$$

where:

P = power (watts)
ϕ_{th} = thermal neutron flux (neutrons/cm^2-sec)
Σ_f = macroscopic cross section for fission (cm^{-1})
V = volume of core (cm^3)

Reactor theory: moderators

- *Thermalization* is the process of reducing the energy level of a neutron from the energy level at which it is produced (~ 2 MeV) to an energy level in the thermal range (0.025 eV).
- The *moderator* is the reactor material that is present for the purpose of thermalizing neutrons (in as few collisions as possible, without absorbing them). Low A materials are used.
- Moderating (MR) ratio is the ratio of the macroscopic slowing down power to the macroscopic cross section for absorption.
- The average logarithmic energy decrement (ξ) is the average change in the logarithm of neutron energy per collision.
- Macroscopic slowing down power is the product of the average logarithmic energy decrement and the macroscopic cross section for scattering.
- There are three desirable characteristics of a moderator.
  - large scattering cross section
  - small absorption cross section
  - large energy loss per collision
- The energy loss after a specified number of collisions can be calculated using the equation $E_N = E_0 (1 – x)^N$ where x is the fractional energy loss per collision.

$$\xi = \Delta E/E$$

$$\xi = \ln E_i - \ln E_f$$

$$\xi = \ln\left(\frac{E_i}{E_f}\right)$$

where:

ξ = average logarithmic energy decrement
E_i = average initial neutron energy
E_f = average final neutron energy

$$\boxed{\xi = \frac{2}{A + \frac{2}{3}}}$$

$$\boxed{MR = \frac{\xi\, \Sigma_s}{\Sigma_a}}$$

Moderating Properties of Materials

Material	ξ	Number of Collisions to Thermalize	Macroscopic Slowing Down Power	Moderating Ratio
H_2O	0.927	19	1.425	62
D_2O	0.510	35	0.177	4830
Helium	0.427	42	9×10^{-6}	51
Beryllium	0.207	86	0.154	126
Boron	0.171	105	0.092	0.00086
Carbon	0.158	114	0.083	216

Reactor theory: prompt and delayed neutrons

- *Prompt neutrons* are released directly from fission within 10^{-13} seconds of the fission event.
- *Delayed neutrons* are released from the decay of fission products that are called *delayed neutron precursors*. Delayed neutron precursors are grouped according to half-life. Half-lives vary from fractions of a second to almost a minute.
- The fraction of neutrons born as delayed neutrons is different for different fuel materials. For ^{235}U: 0.0065 and ^{239}Pu: 0.0021.
- The delayed neutron generation time is the total time from the birth of the fast neutron to the emission of the delayed neutron in the next generation. Delayed neutron generation times are dominated by the half-life of the delayed neutron precursor. The average delayed neutron generation time is about 12.5 seconds.
- Delayed neutrons are responsible for the ability to control the rate at which power can rise in a reactor. If only prompt neutrons existed, reactor control would not be possible due to the rapid power changes

$$^{87}_{35}\mathrm{Br} \xrightarrow[55.9\ \mathrm{sec}]{\beta^-} {}^{87}_{36}\mathrm{Kr} \xrightarrow[\mathrm{instantaneous}]{\mathrm{n}} {}^{86}_{36}\mathrm{Kr}_{\mathrm{stable}}$$

Delayed Neutron Precursor Groups for Thermal Fission in Uranium-235

Group	Half-Life (sec)	Delayed Neutron Fraction	Average Energy (MeV)
1	55.7	0.00021	0.25
2	22.7	0.00142	0.46
3	6.2	0.00127	0.41
4	2.3	0.00257	0.45
5	0.61	0.00075	0.41
6	0.23	0.00027	-
Total	-	0.0065	-

Fraction: 0.65 %
Average half life: 12.5 s

Average generation time

$$\boxed{\mathrm{Time}_{\mathrm{average}} = \mathrm{Time}_{\mathrm{prompt}}\ (1 - \beta) + \mathrm{Time}_{\mathrm{delayed}}\ (\beta)}$$

Reactor can be controlled

Example:

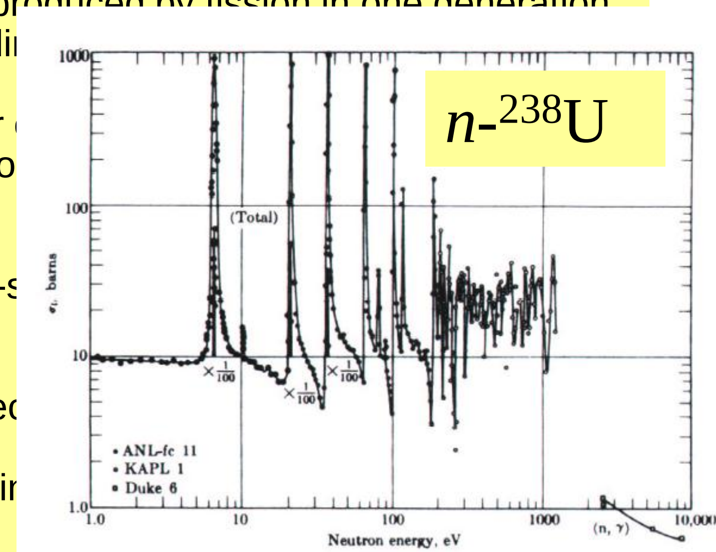
Assume a prompt neutron generation time for a particular reactor of 5×10^{-5} seconds and a delayed neutron generation time of 12.5 seconds. If β is 0.0065, calculate the average generation time.

Solution:

$$\begin{aligned}\mathrm{Time}_{\mathrm{average}} &= \mathrm{Time}_{\mathrm{prompt}}\ (1 - \beta) + \mathrm{Time}_{\mathrm{delayed}}\ (\beta)\\ &= (5\ \mathrm{x}\ 10^{-5}\ \mathrm{seconds})\ (0.9935) + (12.5\ \mathrm{seconds})\ (0.0065)\\ &= 0.0813\ \mathrm{seconds}\end{aligned}$$

Reactor theory: neutron life cycle

- The *infinite multiplication factor,* k_∞, is the ratio of the neutrons produced by fission in one generation to the number of neutrons lost through absorption in the precedi
- The *effective multiplication factor*, k_{eff}, is the ratio of the number o one generation to the number of neutrons lost through absorptio generation.
- *Critical* is the condition where the neutron chain reaction is self-s is neither increasing nor decreasing.
- *Subcritical* is the condition in which the neutron population is dec
- *Supercritical* is the condition in which the neutron population is in



Infinite reactor core (no leakage)

$$k_\infty = \frac{\text{neutron production from fission in one generation}}{\text{neutron absorption in the preceding generation}}$$

Four-factor formula

$$k_\infty = \varepsilon\, p\, f\, \eta$$

where:

ε = fast fission factor
p = resonance escape probability
f = thermal utilization factor
η = reproduction factor

$$\varepsilon = \frac{\text{number of fast neutrons produced by all fissions}}{\text{number of fast neutrons produced by thermal fissions}} \qquad 1.03$$

$$p = \frac{\text{number of neutrons that reach thermal energy}}{\text{number of fast neutrons that start to slow down}} \qquad 0.95 \text{ to } 0.99$$

$$f = \frac{\text{number of thermal neutrons absorbed in the fuel}}{\text{number of thermal neutrons absorbed in all reactor materials}} \qquad 0.932$$

$$\eta = \frac{\text{number of fast neutrons produced by thermal fission}}{\text{number of thermal neutrons absorbed in the fuel}} \qquad 2.07$$

Reactor theory: thermal utilization factor

Thermal utilization factor

$$f = \frac{\text{number of thermal neutrons absorbed in the fuel}}{\text{number of thermal neutrons absorbed in all reactor materials}}$$

$$f = \frac{\text{rate of absorption of thermal neutrons by the fuel}}{\text{rate of absorption of thermal neutrons by all reactor materials}}$$

$$f = \frac{\Sigma_a^U \phi^U V^U}{\Sigma_a^U \phi^U V^U + \Sigma_a^m \phi^m V^m + \Sigma_a^p \phi^p V^p}$$

U, m and p for uranium, moderator and poison

$$f = \frac{\Sigma_a^U}{\Sigma_a^U + \Sigma_a^m + \Sigma_a^p}$$

Homogeneous reactor (same flux and volume everywhere)

Example:

Calculate the thermal utilization factor for a homogeneous reactor. The macroscopic absorption cross section of the fuel is 0.3020 cm^{-1}, the macroscopic absorption cross section of the moderator is 0.0104 cm^{-1}, and the macroscopic absorption cross section of the poison is 0.0118 cm^{-1}.

Solution:

$$f = \frac{\Sigma_a^U}{\Sigma_a^U + \Sigma_a^m + \Sigma_a^p}$$

$$= \frac{0.3020 \ cm^{-1}}{0.3020 \ cm^{-1} + 0.0104 cm^{-1} + 0.0118 cm^{-1}}$$

$$= 0.932$$

An increase in moderator temperature will have the following effects.

Increase the thermal utilization factor
Decrease resonance escape probability
Decrease fast non-leakage probability
Decrease thermal non-leakage probability

Reactor theory: reproduction factor

Reproduction factor

$$\eta = \frac{\text{rate of production of fast neutrons by thermal fission}}{\text{rate of absorption of thermal neutrons by the fuel}}$$

$$\eta = \frac{\Sigma_f^U \phi^U \nu}{\Sigma_a^U \phi^U}$$

fission reaction rate ($\Sigma_f^u \phi^u$) average number of neutrons produced per fission (ν)

$$\eta = \frac{N^{U-235}\,\sigma_f^{U-235}\,\nu^{U-235}}{N^{U-235}\,\sigma_a^{U-235} + N^{U-238}\,\sigma_a^{U-238}}$$

When core contains ^{235}U and ^{238}U

TABLE 1 Average Number of Neutrons Liberated in Fission				
Fissile Nucleus	Thermal Neutrons		Fast Neutrons	
	ν	η	ν	η
Uranium-233	2.49	2.29	2.58	2.40
Uranium-235	2.42	2.07	2.51	2.35
Plutonium-239	2.93	2.15	3.04	2.90

Example:

Calculate the reproduction factor for a reactor that uses 10% enriched uranium fuel. The microscopic absorption cross section for uranium-235 is 694 barns. The cross section for uranium-238 is 2.71 barns. The microscopic fission cross section for uranium-235 is 582 barns. The atom density of uranium-235 is 4.83×10^{21} atoms/cm^3. The atom density of uranium-238 is 4.35×10^{22} atoms/cm^3. ν is 2.42.

Solution:

Use Equation (3-2) to calculate the reproduction factor.

$$\eta = \frac{N^{U-235}\,\sigma_f^{U-235}\,\nu^{U-235}}{N^{U-235}\,\sigma_a^{U-235} + N^{U-238}\,\sigma_a^{U-238}}$$

$$= \frac{\left(4.83 \times 10^{21}\ \frac{\text{atoms}}{\text{cm}^3}\right)\left(582 \times 10^{-24}\ \text{cm}^2\right)(2.42)}{\left(4.83 \times 10^{21}\ \frac{\text{atoms}}{\text{cm}^3}\right)\left(694 \times 10^{-24}\ \text{cm}^2\right) + \left(4.35 \times 10^{22}\ \frac{\text{atoms}}{\text{cm}^3}\right)\left(2.71 \times 10^{-24}\ \text{cm}^2\right)}$$

$$= 1.96$$

Reactor theory: effective multiplication factor

Effective multiplication factor

$$k_{eff} = \frac{\text{neutron production from fission in one generation}}{\text{neutron absorption in the preceding generation} + \text{neutron leakage in the preceding generation}}$$

$$k_{eff} = k_{\infty}\, \mathfrak{L}_f\, \mathfrak{L}_t$$

Fast non-leakage probability

$$\mathfrak{L}_f = \frac{\text{number of fast neutrons that do not leak from reactor}}{\text{number of fast neutrons produced by all fissions}}$$

Thermal non-leakage probability

$$\mathfrak{L}_t = \frac{\text{number of thermal neutrons that do not leak from reactor}}{\text{number of neutrons that reach thermal energies}}$$

Total non-leakage probability depends on coolant temperature with negative temperature coefficient.

As coolant temperature rises, the coolant expands. Density of the moderator is lower; there neutrons travel farther while slowing down.

Six factor formula

$$k_{eff} = \varepsilon\, \mathfrak{L}_f\, p\, \mathfrak{L}_t\, f\, \eta$$

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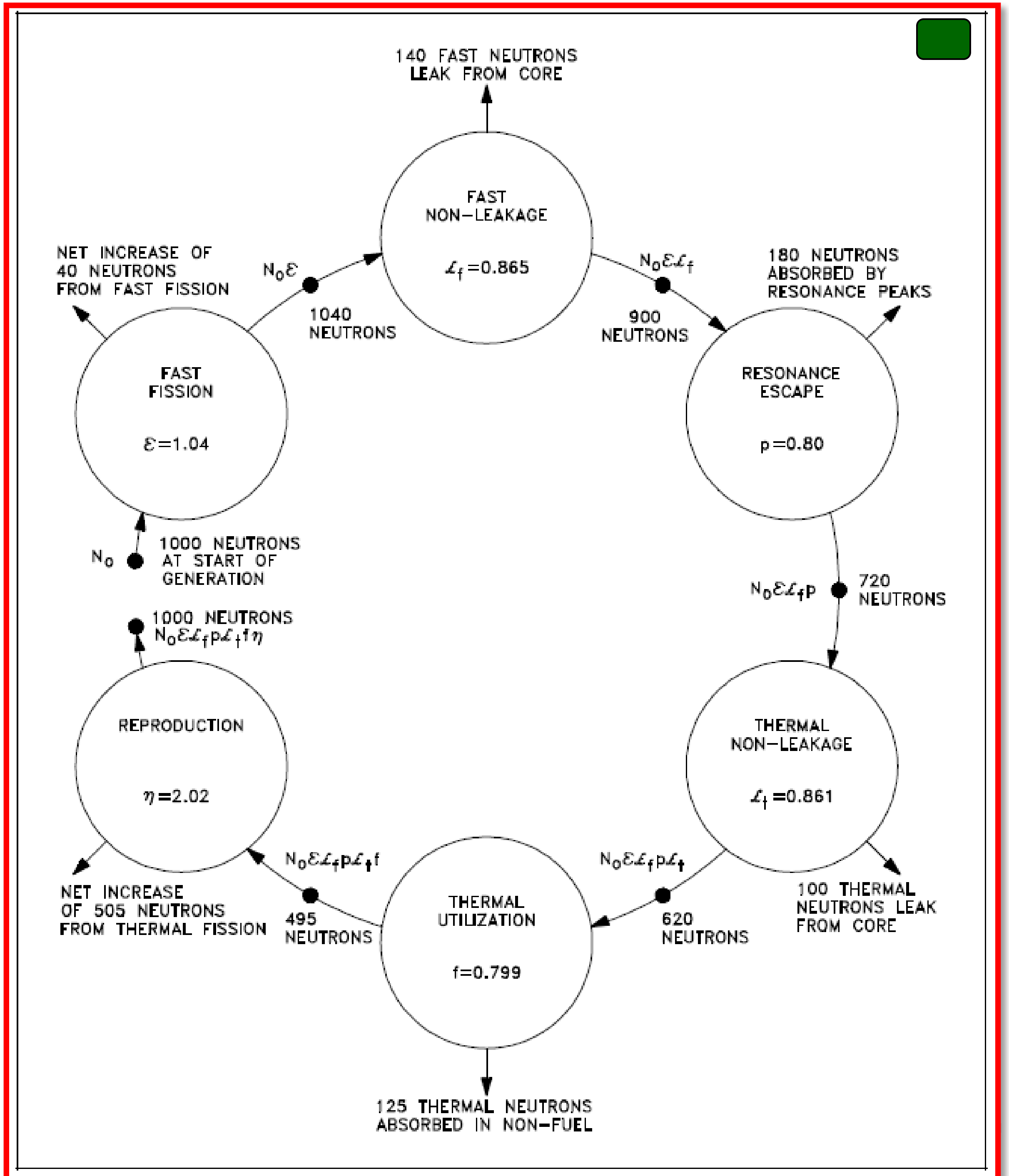
Neutron life cycle for k_{eff}=1 in a thermal reactor

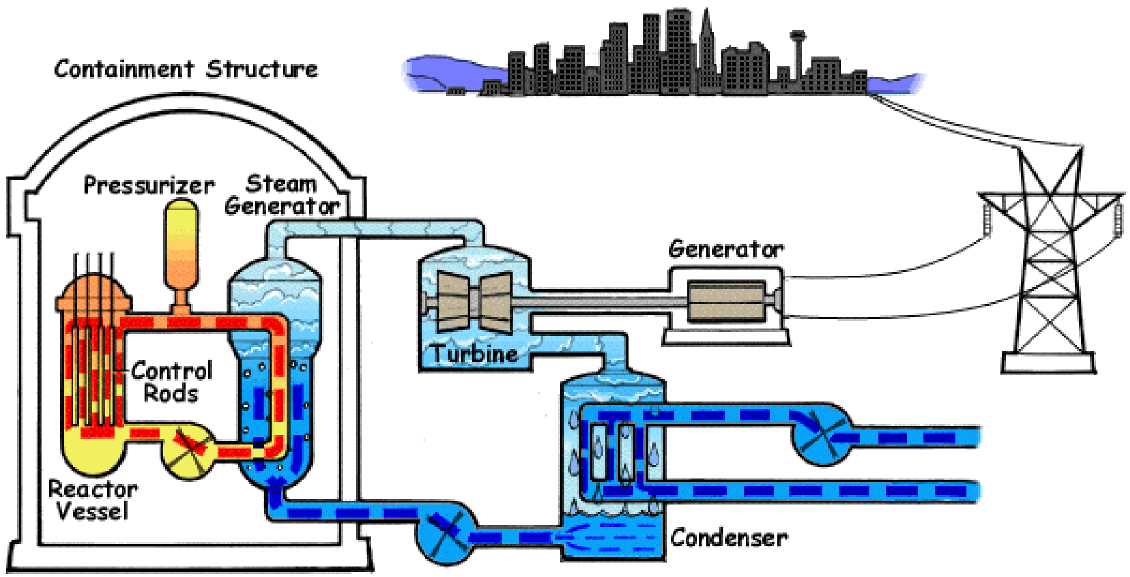


Enrichment affects thermal utilization and reproduction factor, and resonance escape probability

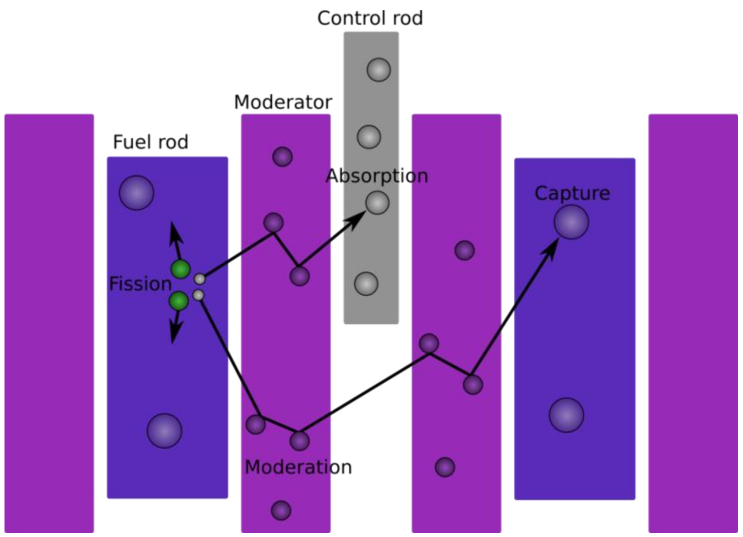
Life cycle in a fast breeder reactor is different.

Thermalization is minimized and almost all fissions take place by fast neutrons.

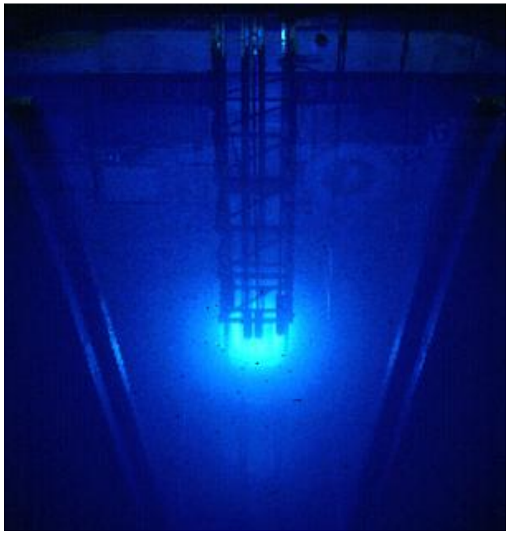
PWR – Pressurized water reactor

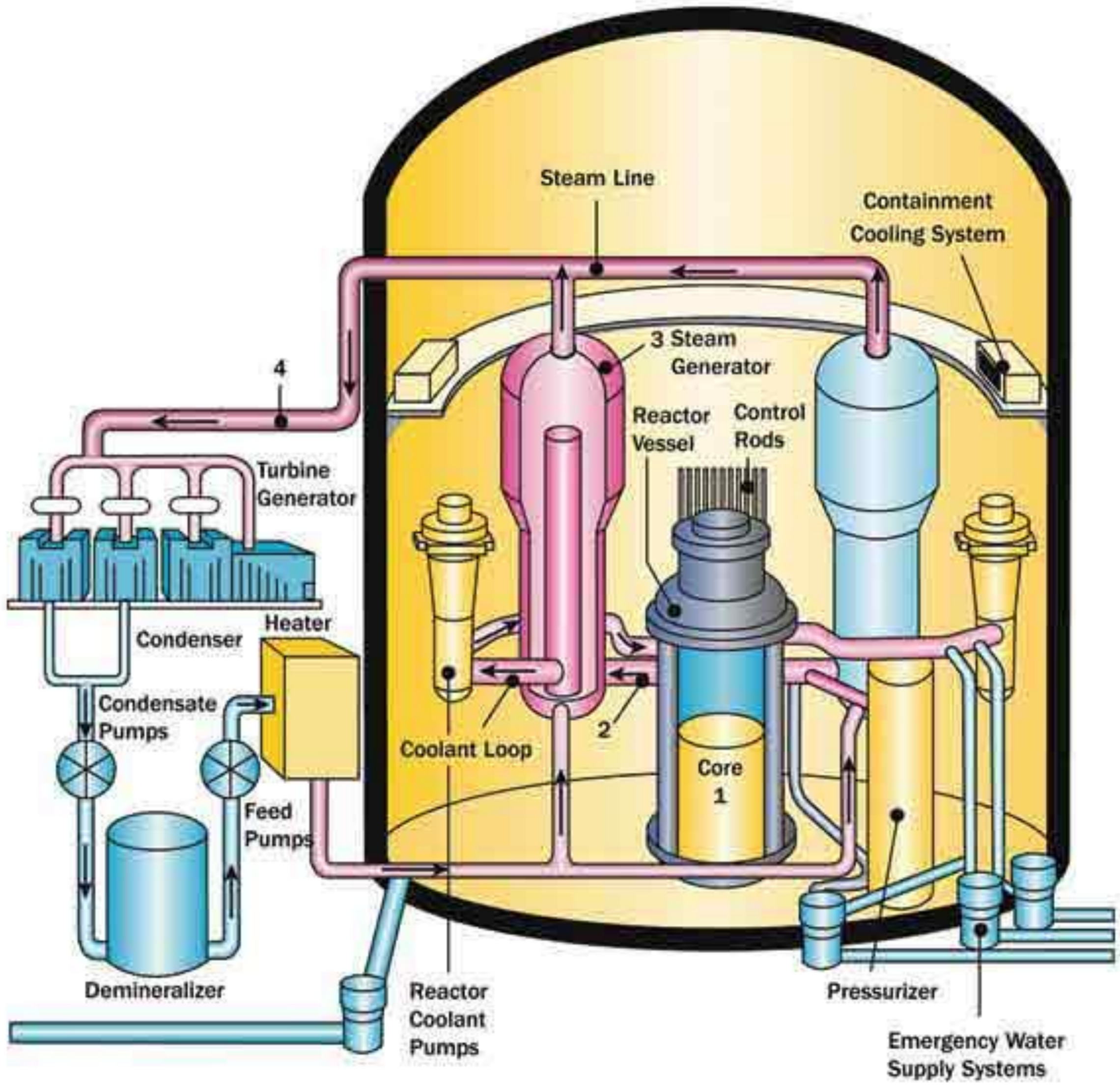


- *PWR* most common reactor type (~1 GW) with thermal efficiency ~30 %.
- Keep water under pressure (~15 MPa) so that it heats (~315 ^{o}C), but does not boil.
- Water from the reactor and the water in the steam generator (~5 Mpa) never mix. In this way, most of the radioactivity in the reactor area.
- Use enriched uranium as fuel.
- Fuel in rods increases resonance escape probability p and fast fission factor ε.



Pressurized water reactor





PWR – Reactor control

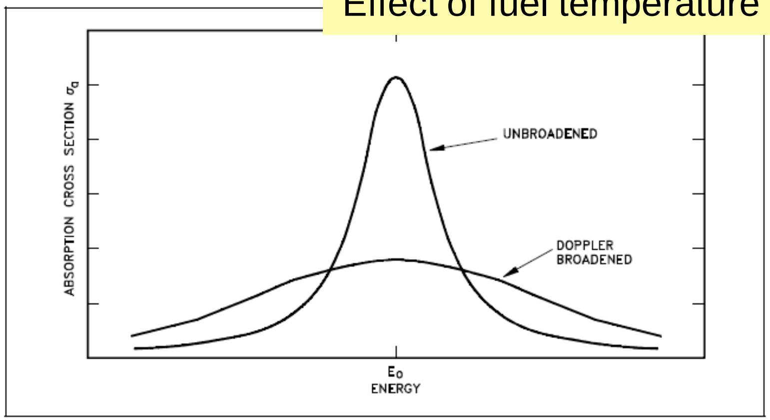
Reactivity: If there are N_0 neutrons in the preceding generation, then there are $N_0(k_{eff})$ neutrons in the present generation. The numerical change in neutron population is $(N_0k_{eff} – N_0)$. The relative change, reactivity is

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

The number of neutrons present in the core after a given number of generations is

$$N_n = N_o (k_{eff})^n$$

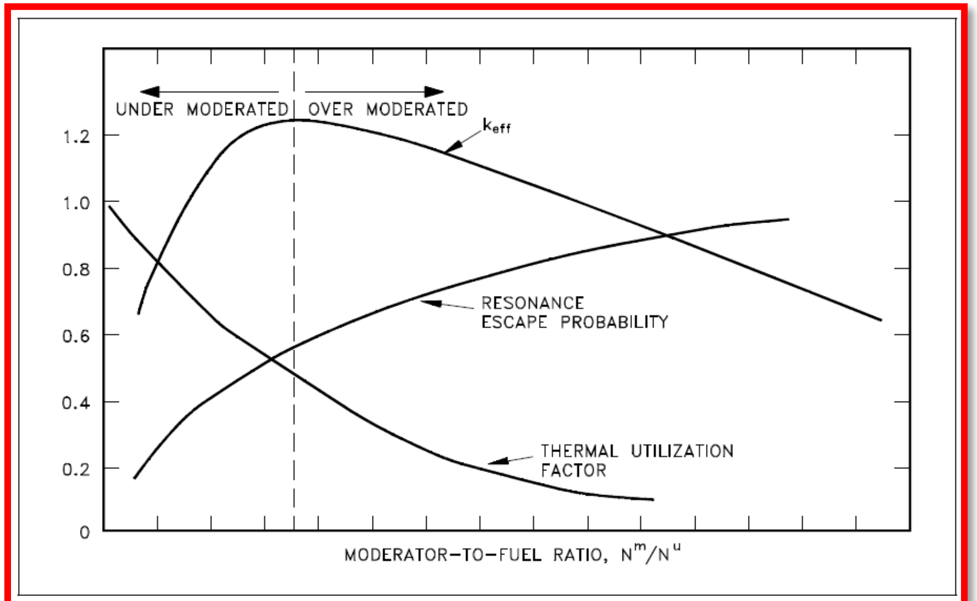
Effect of fuel temperature



Amount of moderator influences product *pf*

Thermal utilization factor: fraction *f* of thermal neutrons absorbed on U rather than moderator.

Resonance escape probability: The fraction *p* of neutrons that escape resonance capture on ^{238}U and manage to slow down to thermal energies.



PWR – Reactor control

The reactor period is defined as the time required for reactor power to change by a factor of "e," where "e" is the base of the natural logarithm and is equal to about 2.718.

The reactor period is given by

$$\tau = \underbrace{\frac{\ell^*}{\rho}}_{\text{prompt term}} + \underbrace{\frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho + \dot{\rho}}}_{\text{delayed term}}$$

where:

ℓ^*	=	prompt generation lifetime
$\bar{\beta}_{eff}$	=	effective delayed neutron fraction
ρ	=	reactivity
λ_{eff}	=	effective delayed neutron precursor decay constant
$\dot{\rho}$	=	rate of change of reactivity

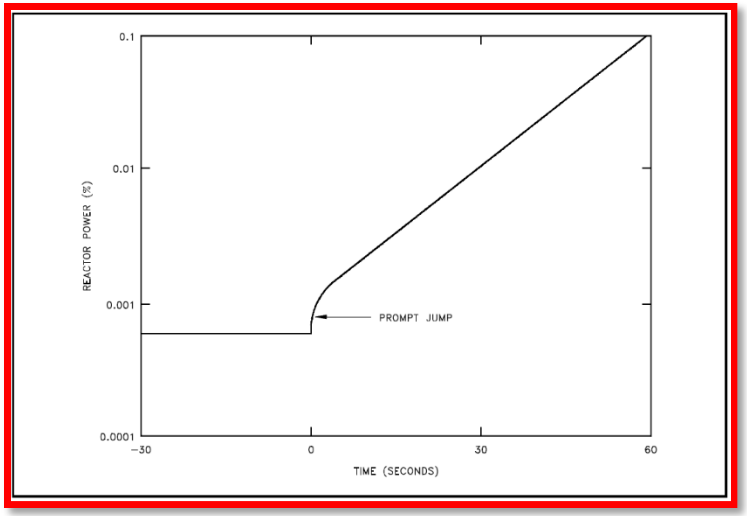
$$P = P_o\, e^{t/\tau}$$

where:

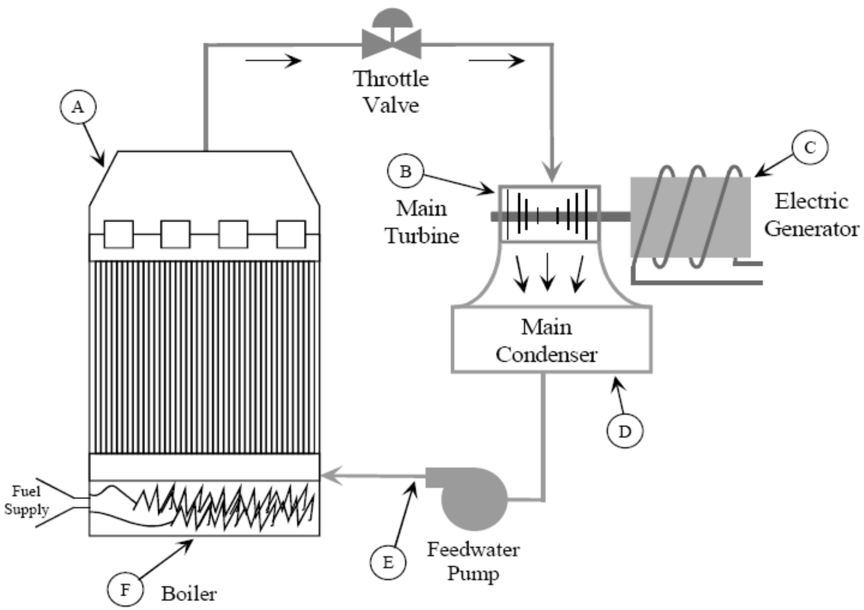
P	=	transient reactor power
P_o	=	initial reactor power
τ	=	reactor period (seconds)
t	=	time during the reactor transient (seconds)

Stable operation (no prompt term and stable reactivity in core)

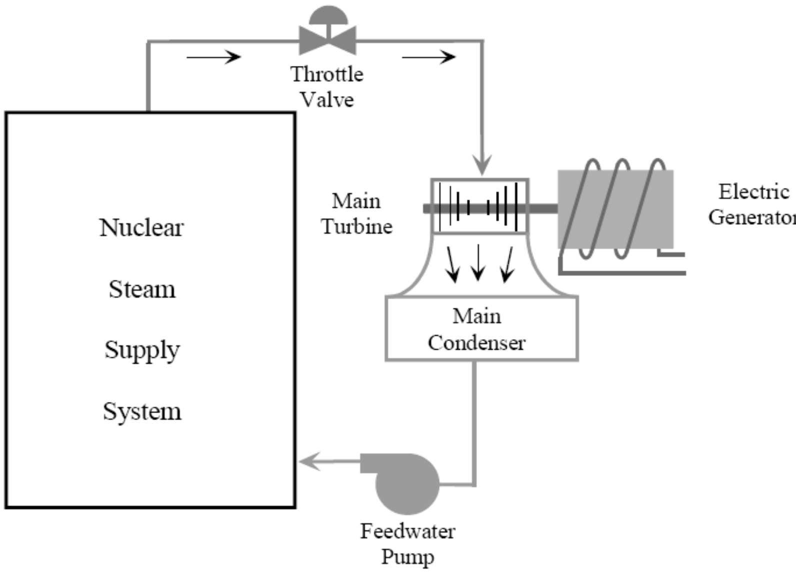
$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\,\rho}$$



Power plant

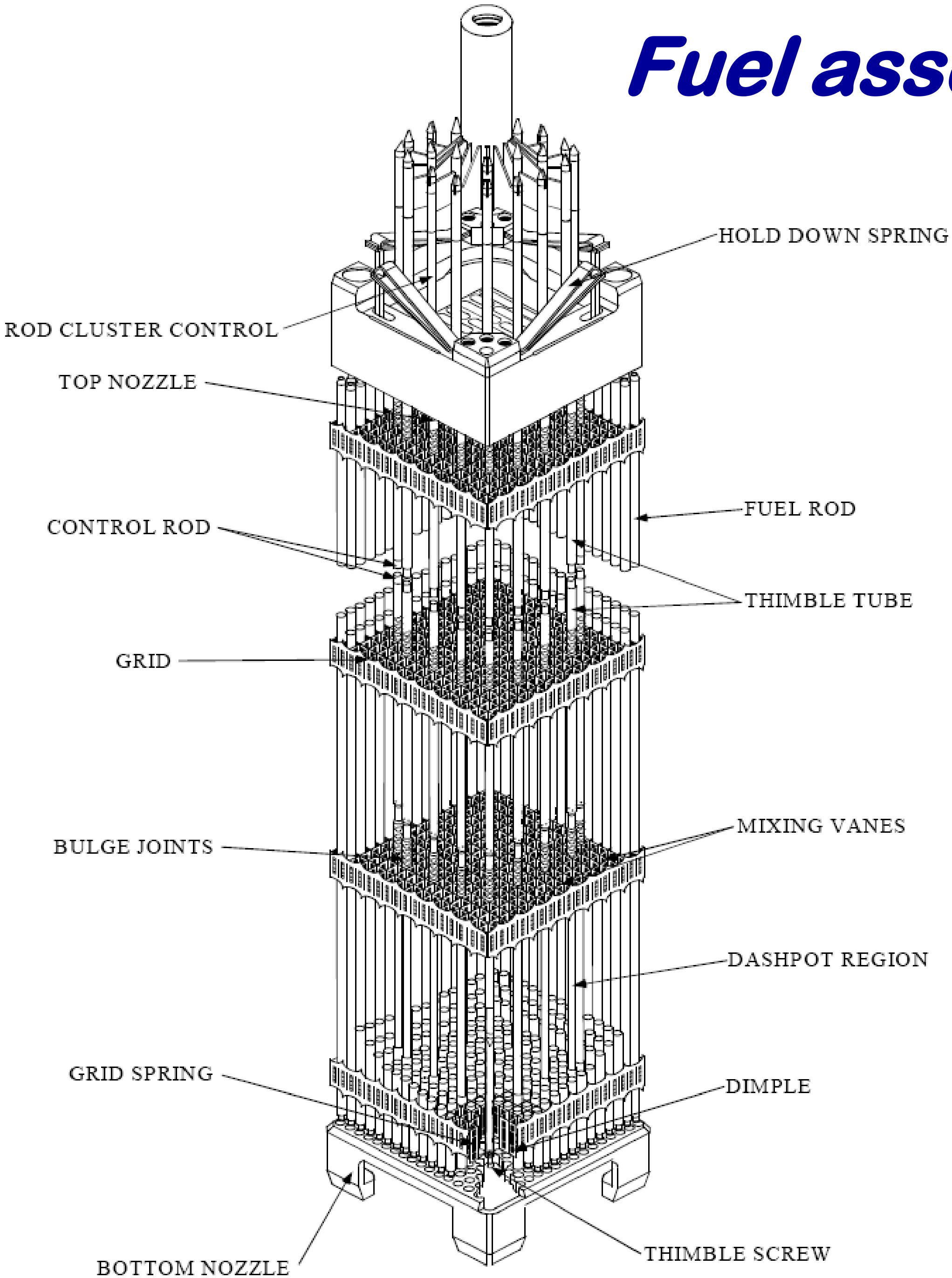


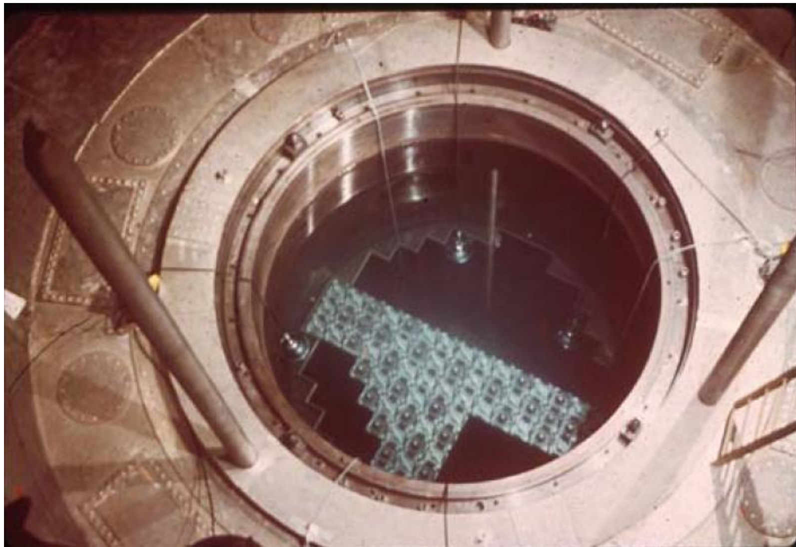
FOSSIL FUEL STEAM PLANT



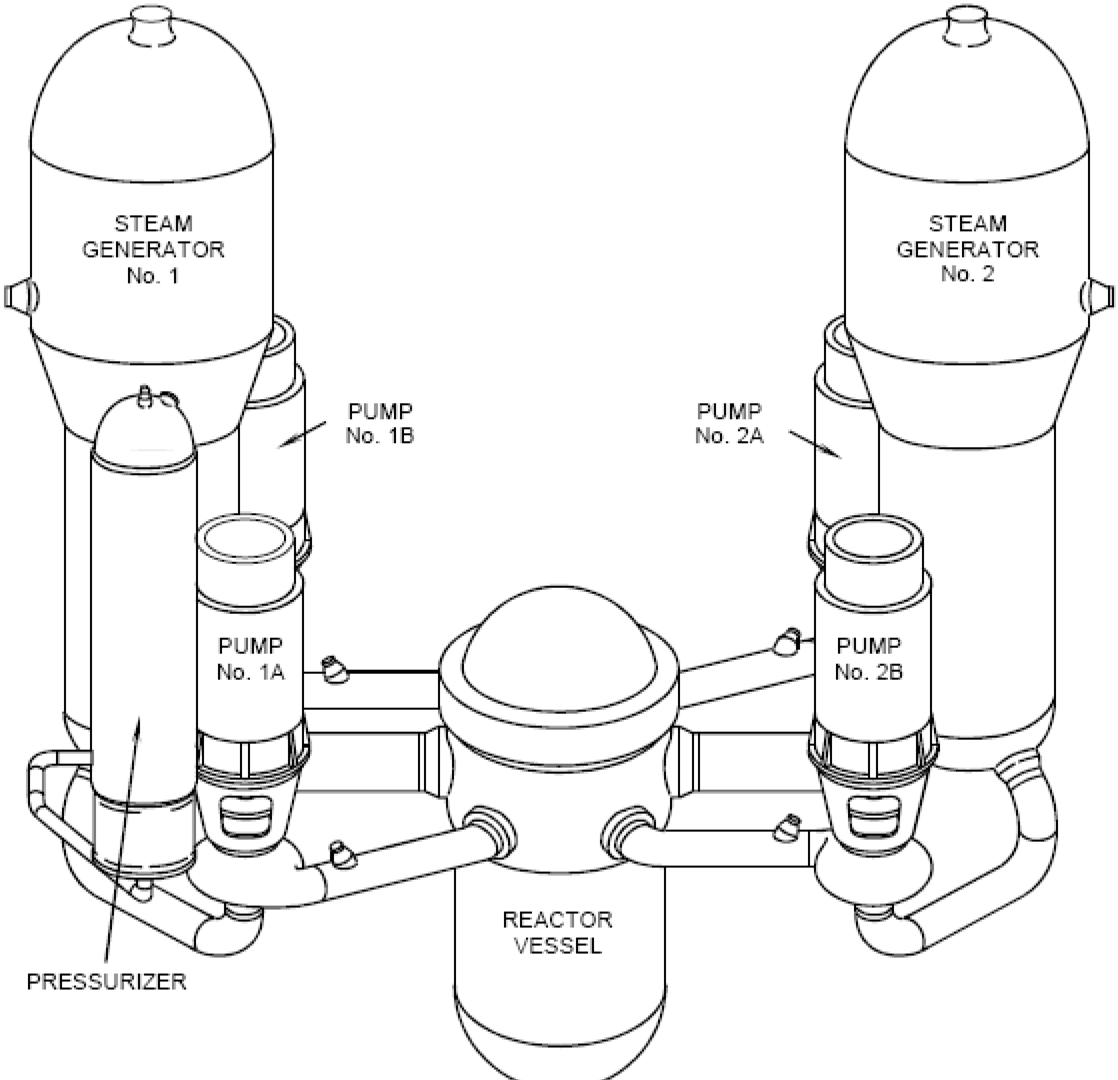
Nuclear Fuel Steam Plant

Fuel assembly

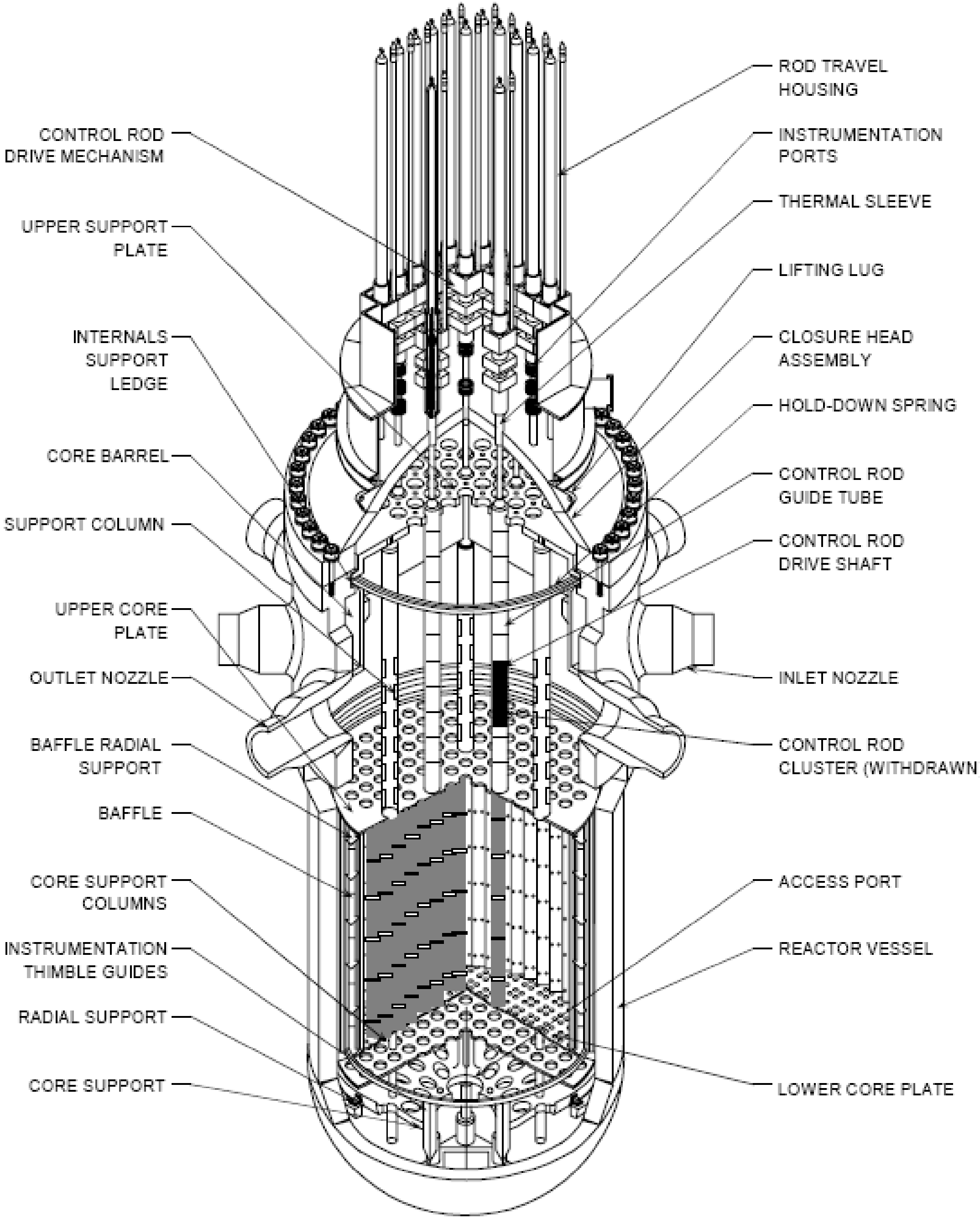




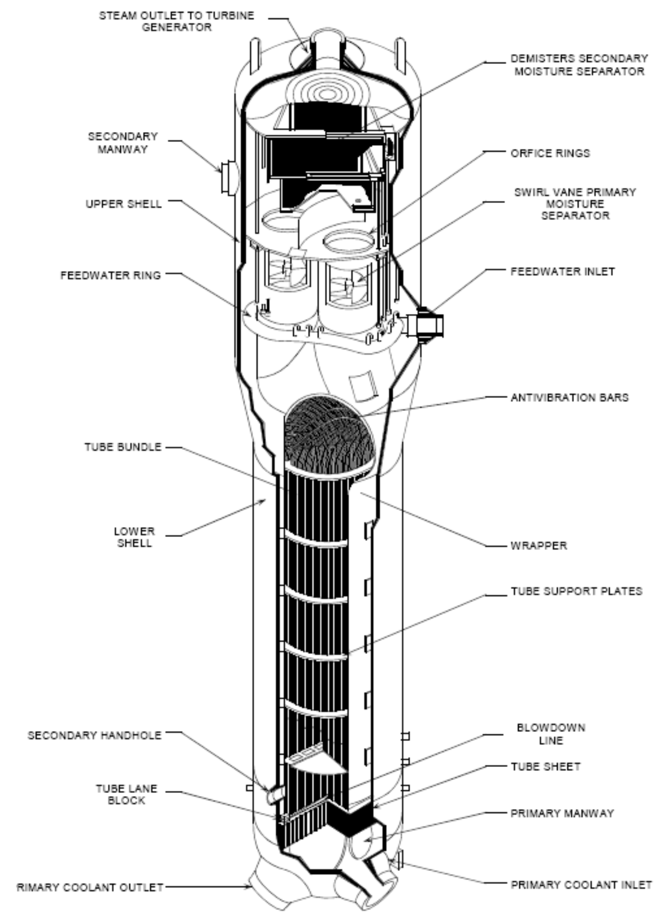
PWR core


STEAM
GENERATOR
No. 1
STEAM
GENERATOR
No. 2
PUMP
No. 1B
PUMP
No. 2A
PUMP
No. 1A
PUMP
No. 2B
REACTOR
VESSEL
PRESSURIZER

Reactor vessel

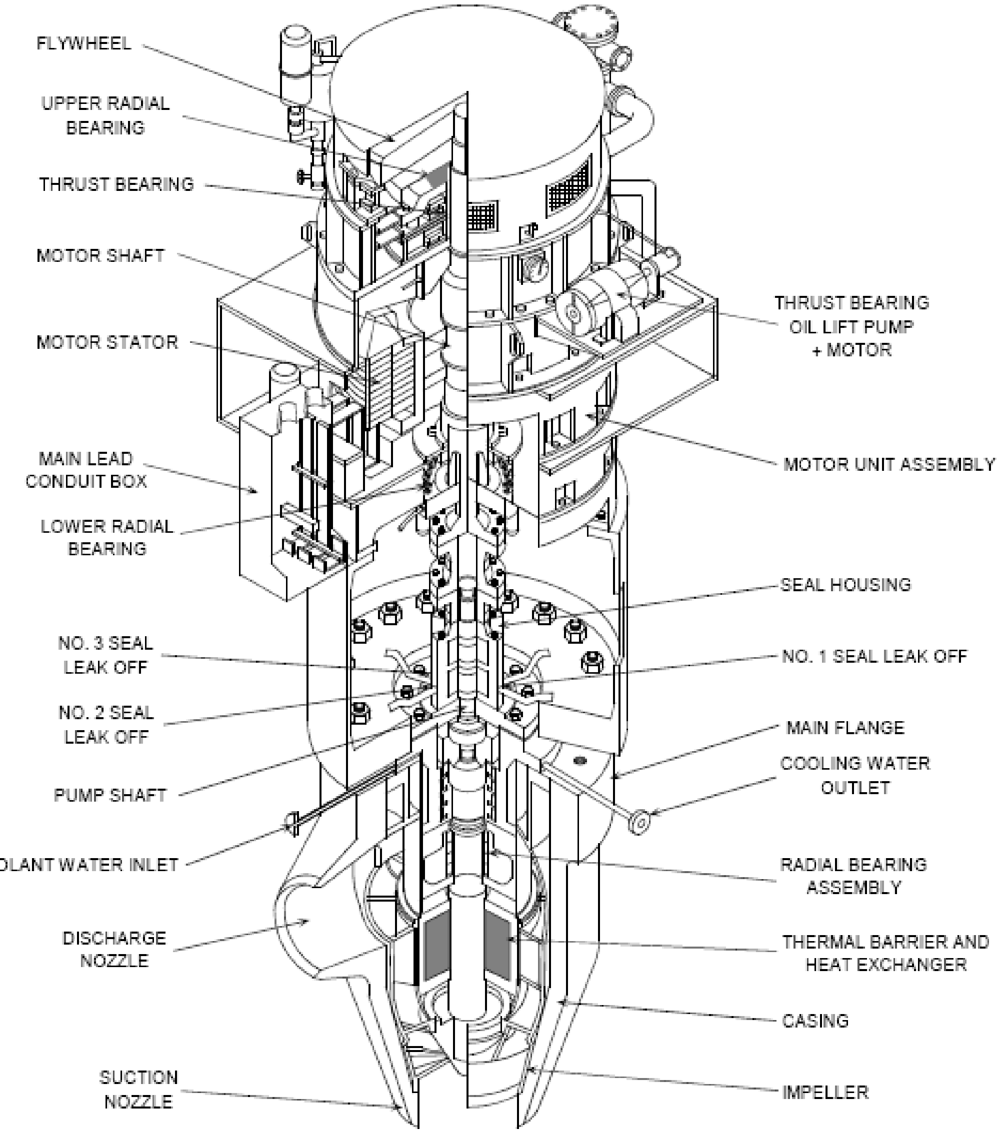


Cutaway View of Reactor Vessel

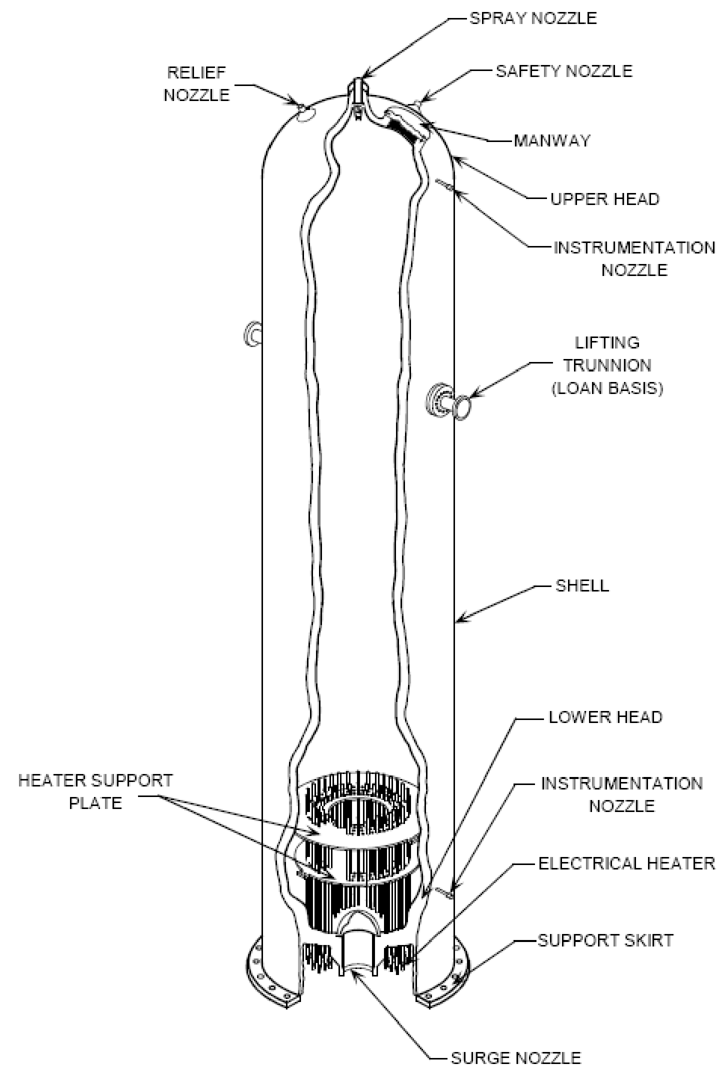


Cutaway View of A Westinghouse Steam Generator

Reactor vessel

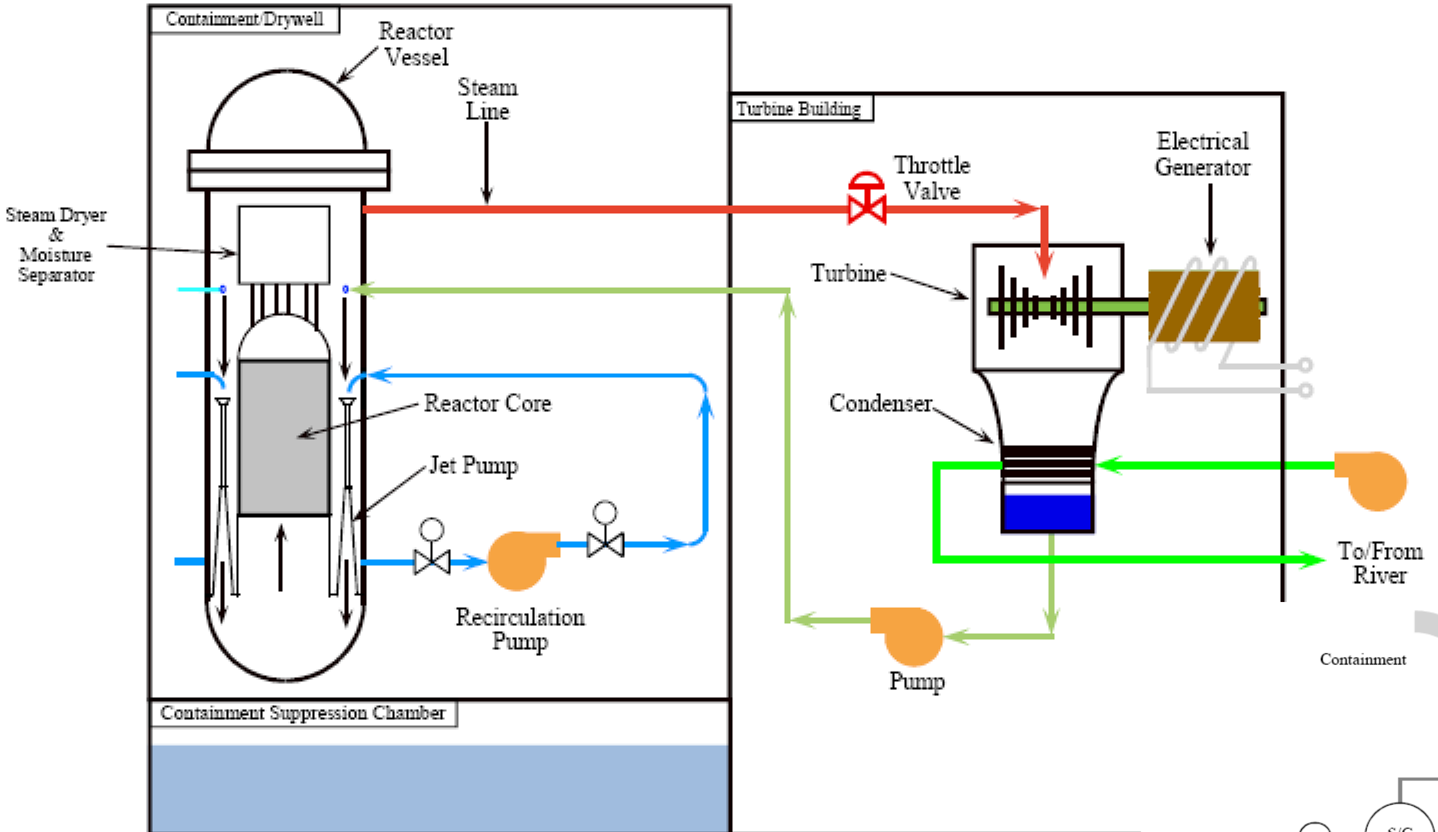


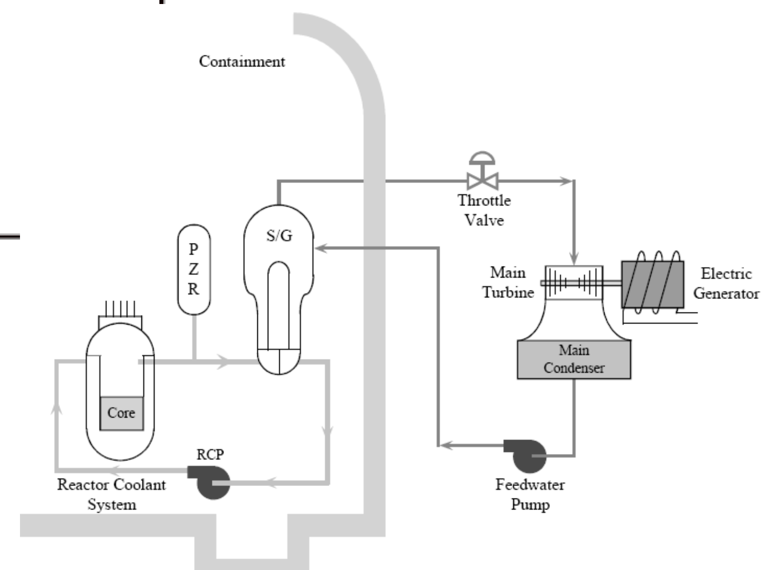
Cutaway View of a Reactor Coolant Pump



Cutaway View of a Pressurizer

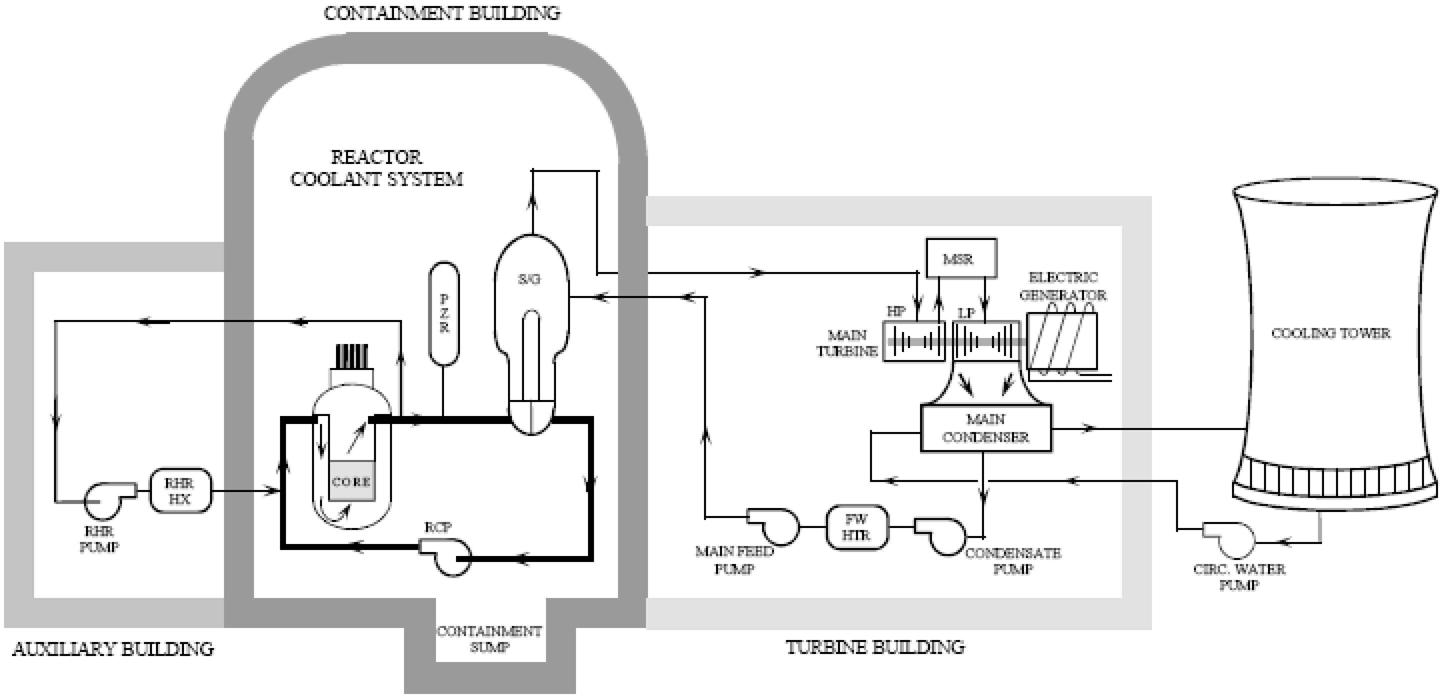
Reactor core containment



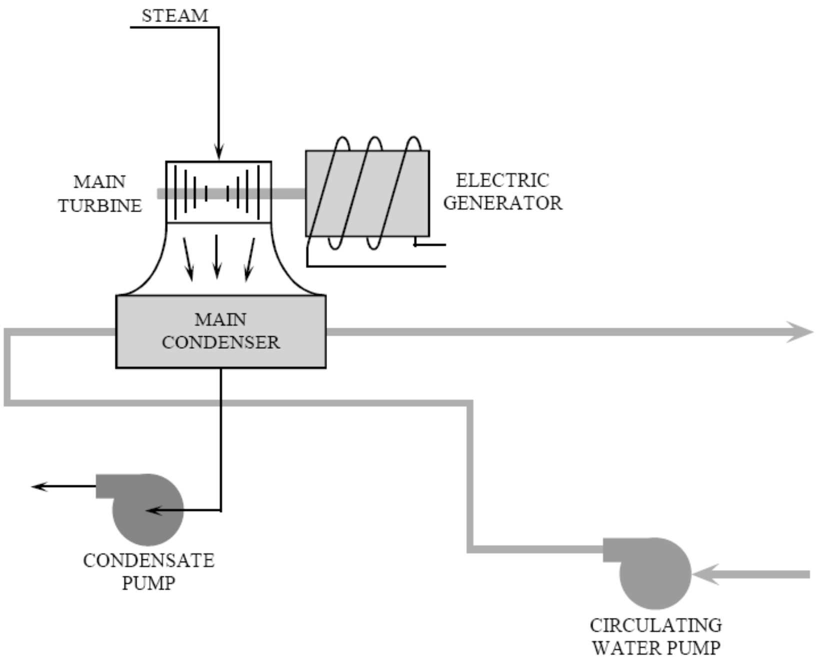


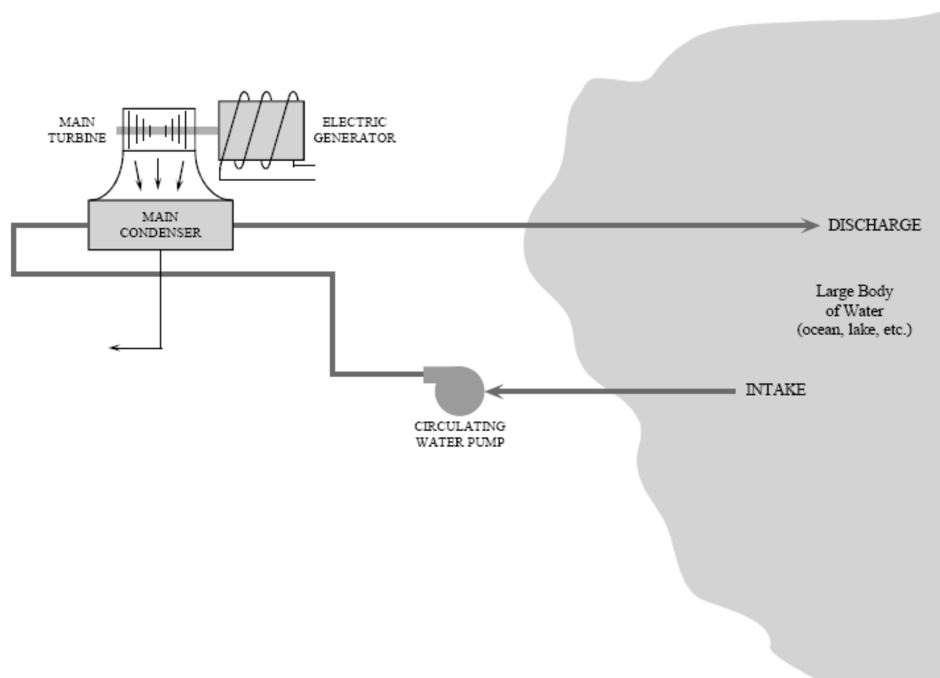
Reactor core containment

PRESSURIZED WATER REACTOR PLANT LAYOUT

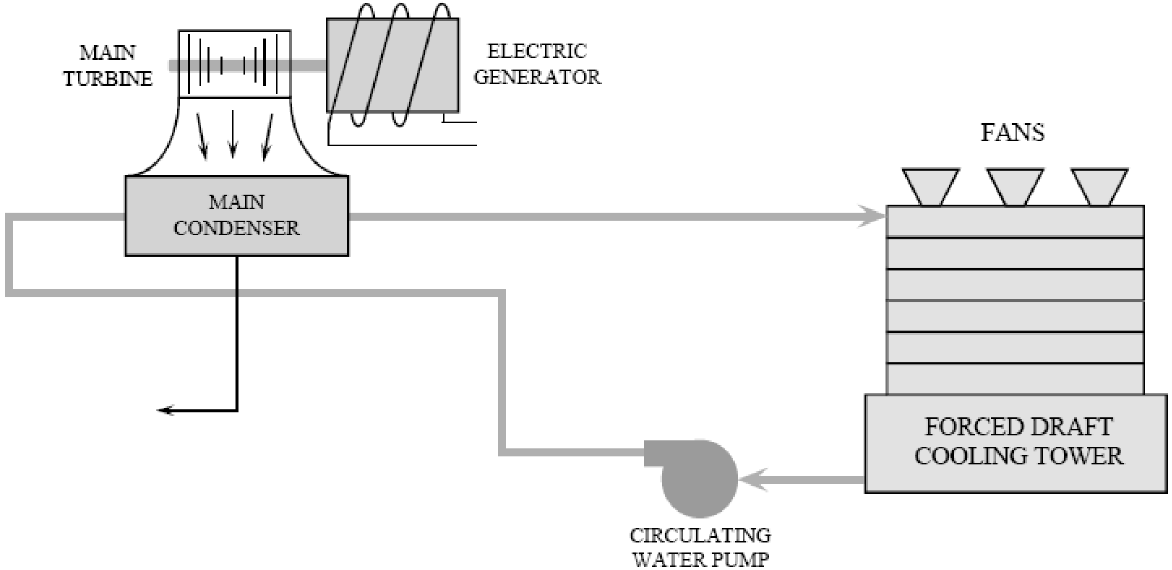


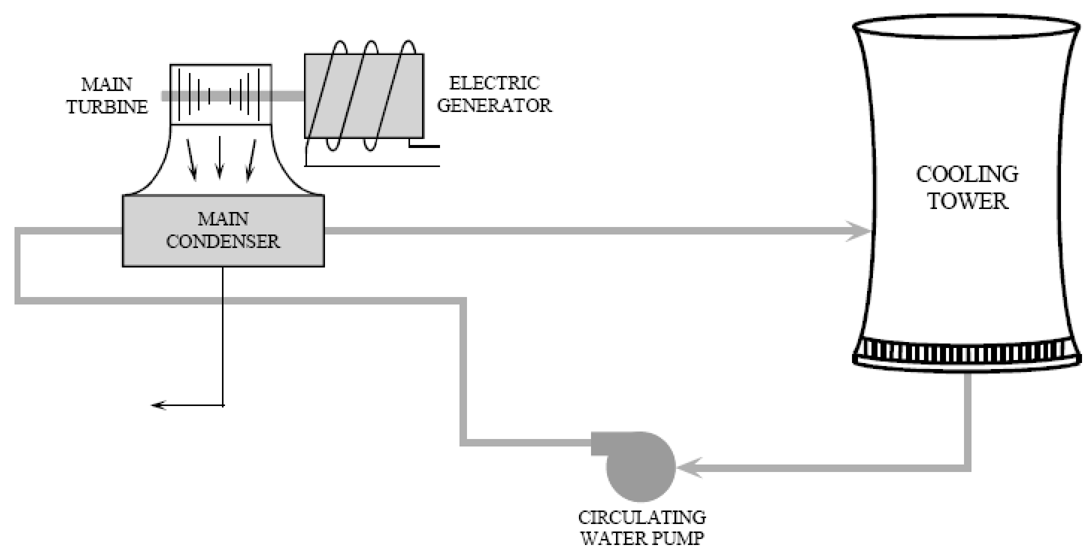
Cooling water systems



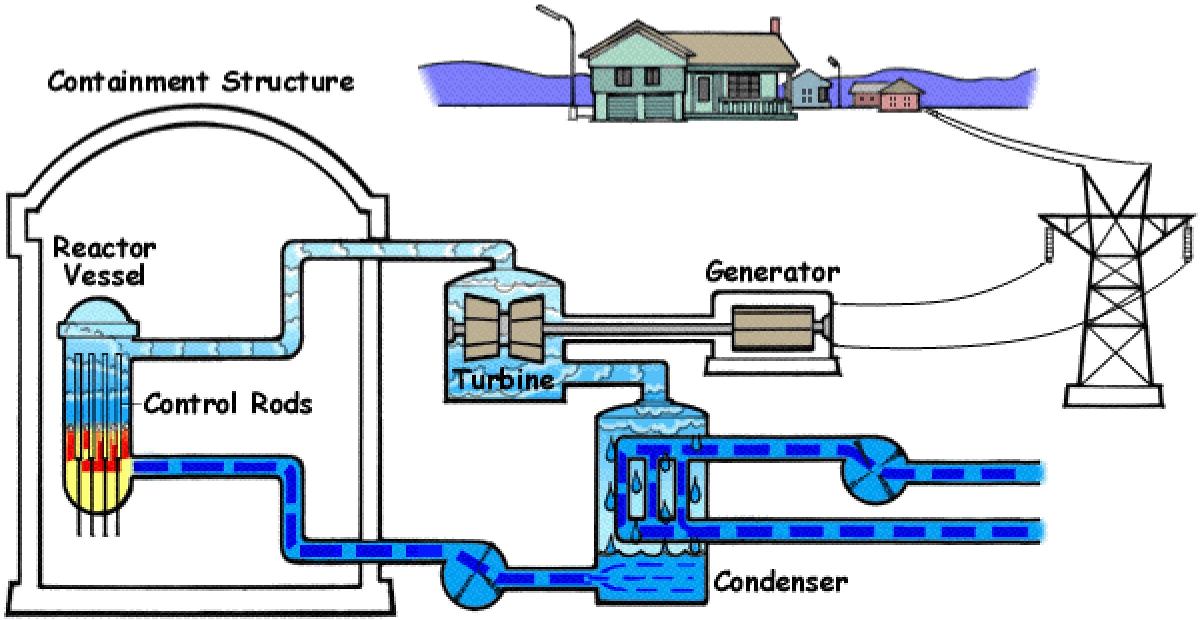


Cooling water systems



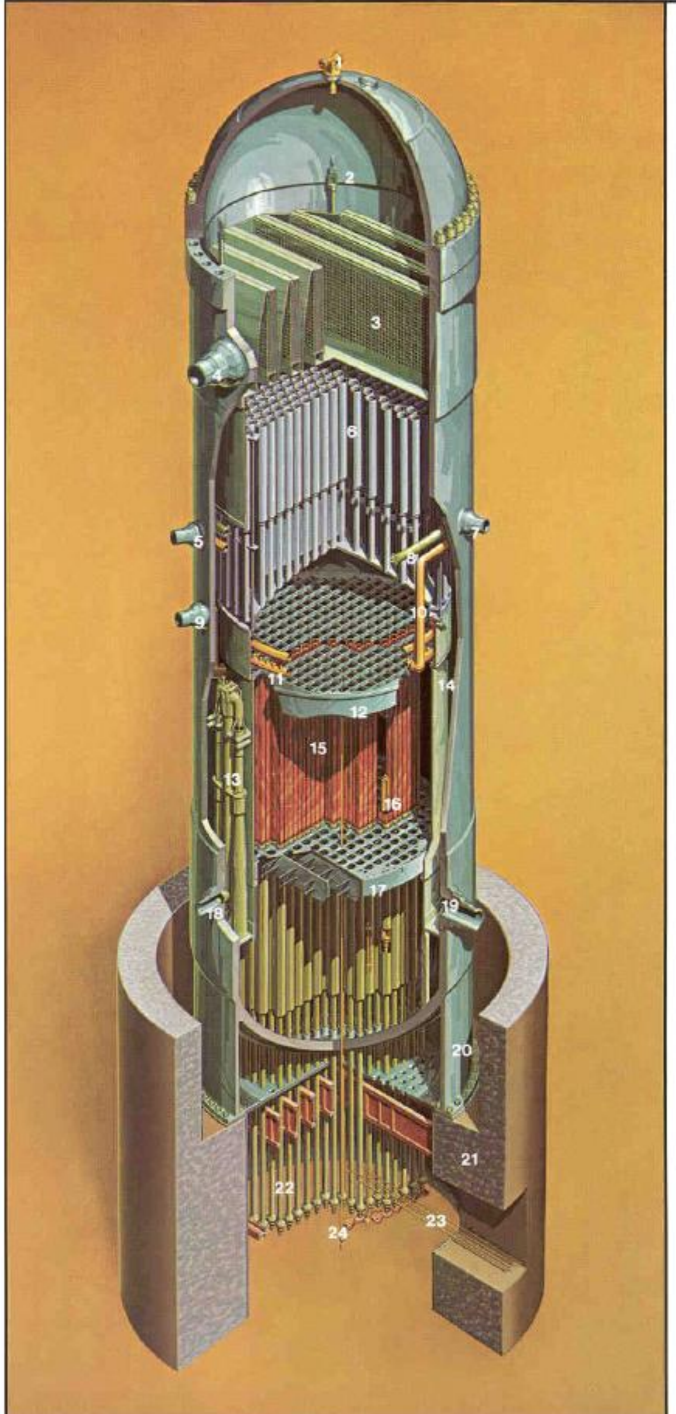


BWR – Boiling water reactor



- In BWRs, the water heated by fission actually boils and turns into steam to turn the generator.
- Simpler design and lower operating pressure (7.5 Mpa and 285 ^{o}C in core), thus more commercially attractive.
- Natural water circulation is used.
- Lower radiation load on reactor vessel.
- Much larger pressure vessel than PWR at same power.

BWR



BWR/6
REACTOR ASSEMBLY

1. VENT AND HEAD SPRAY
2. STEAM DRYER LIFTING LUG
3. STEAM DRYER ASSEMBLY
4. STEAM OUTLET
5. CORE SPRAY INLET
6. STEAM SEPARATOR ASSEMBLY
7. FEEDWATER INLET
8. FEEDWATER SPARGER
9. LOW PRESSURE COOLANT INJECTION INLET
10. CORE SPRAY LINE
11. CORE SPRAY SPARGER
12. TOP GUIDE
13. JET PUMP ASSEMBLY
14. CORE SHROUD
15. FUEL ASSEMBLIES
16. CONTROL BLADE
17. CORE PLATE
18. JET PUMP / RECIRCULATION WATER INLET
19. RECIRCULATION WATER OUTLET
20. VESSEL SUPPORT SKIRT
21. SHIELD WALL
22. CONTROL ROD DRIVES
23. CONTROL ROD DRIVE HYDRAULIC LINES
24. IN-CORE FLUX MONITOR

GENERAL ELECTRIC

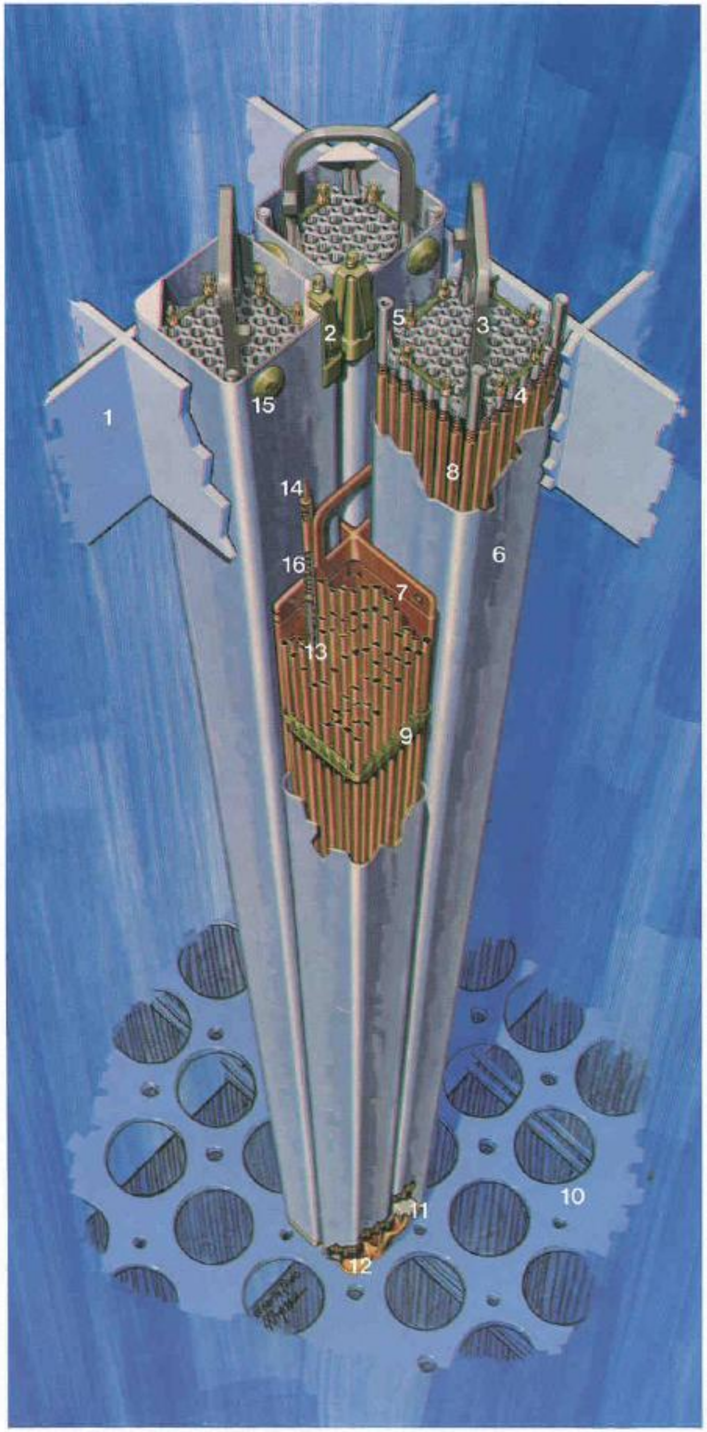
BWR fuel

BWR/6 FUEL ASSEMBLIES & CONTROL ROD MODULE

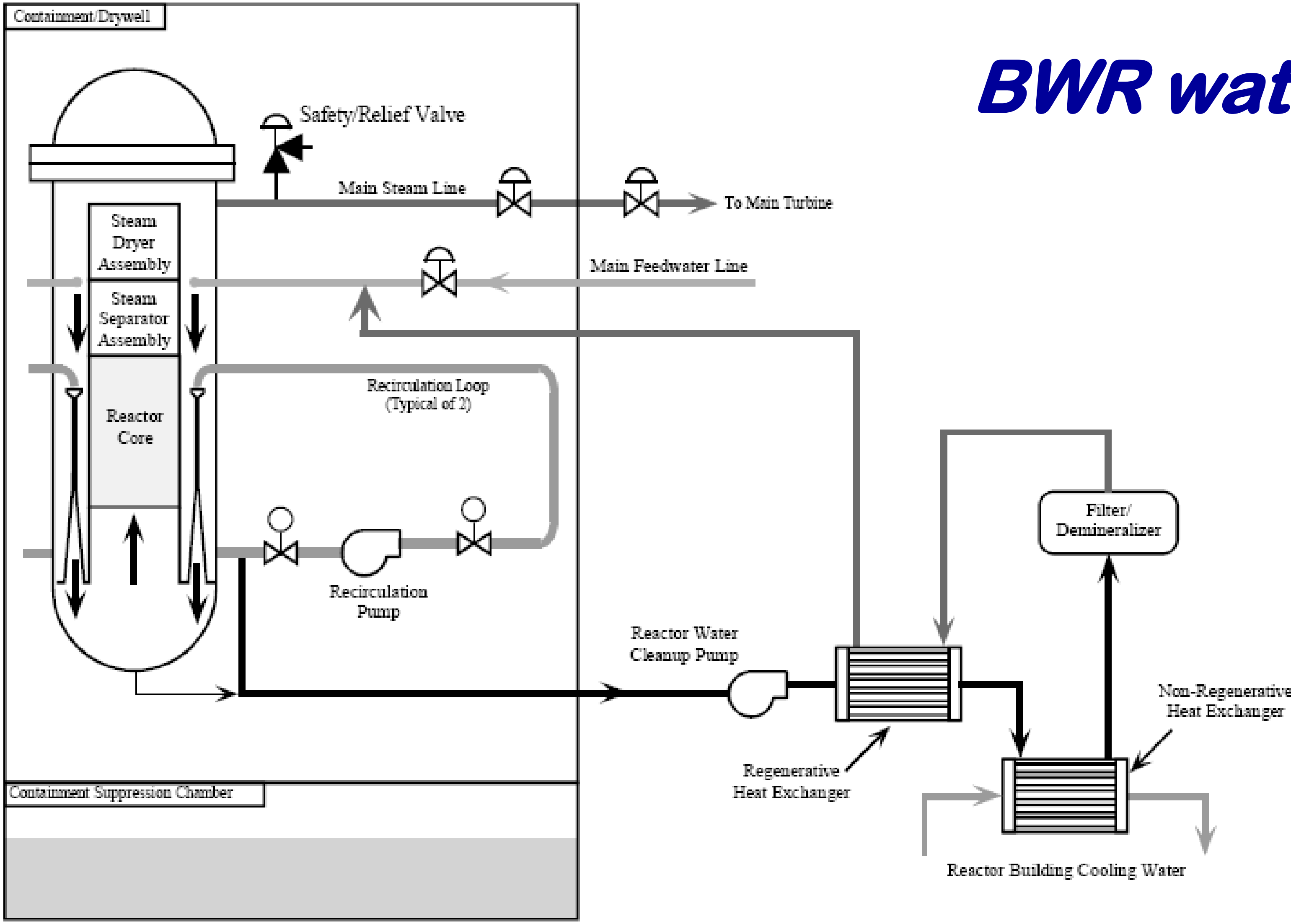
1. TOP FUEL GUIDE
2. CHANNEL FASTENER
3. UPPER TIE PLATE
4. EXPANSION SPRING
5. LOCKING TAB
6. CHANNEL
7. CONTROL ROD
8. FUEL ROD
9. SPACER
10. CORE PLATE ASSEMBLY
11. LOWER TIE PLATE
12. FUEL SUPPORT PIECE
13. FUEL PELLETS
14. END PLUG
15. CHANNEL SPACER
16. PLENUM SPRING

GENERAL ELECTRIC

GEZ-4383

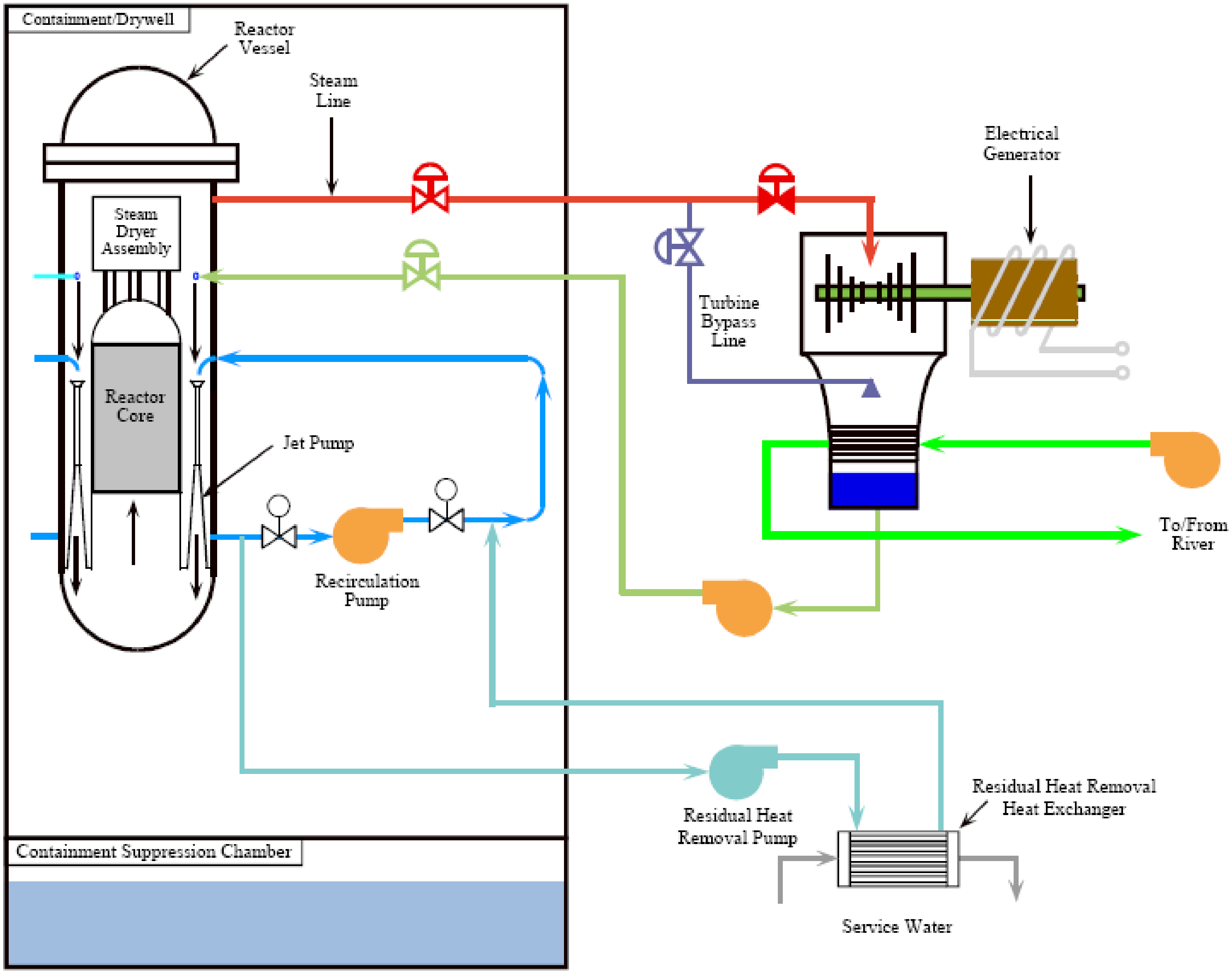


BWR water

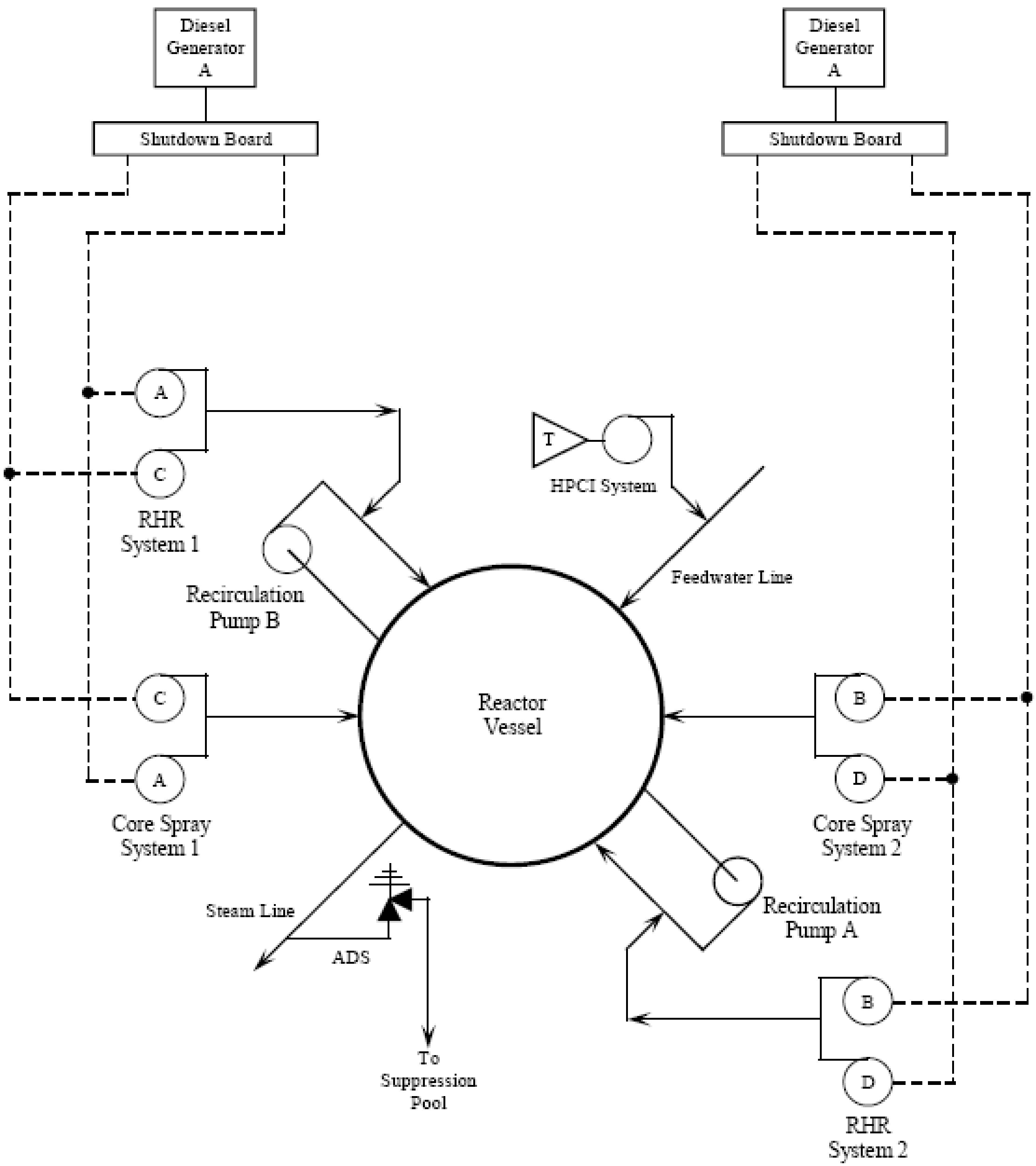


Reactor Water Cleanup System

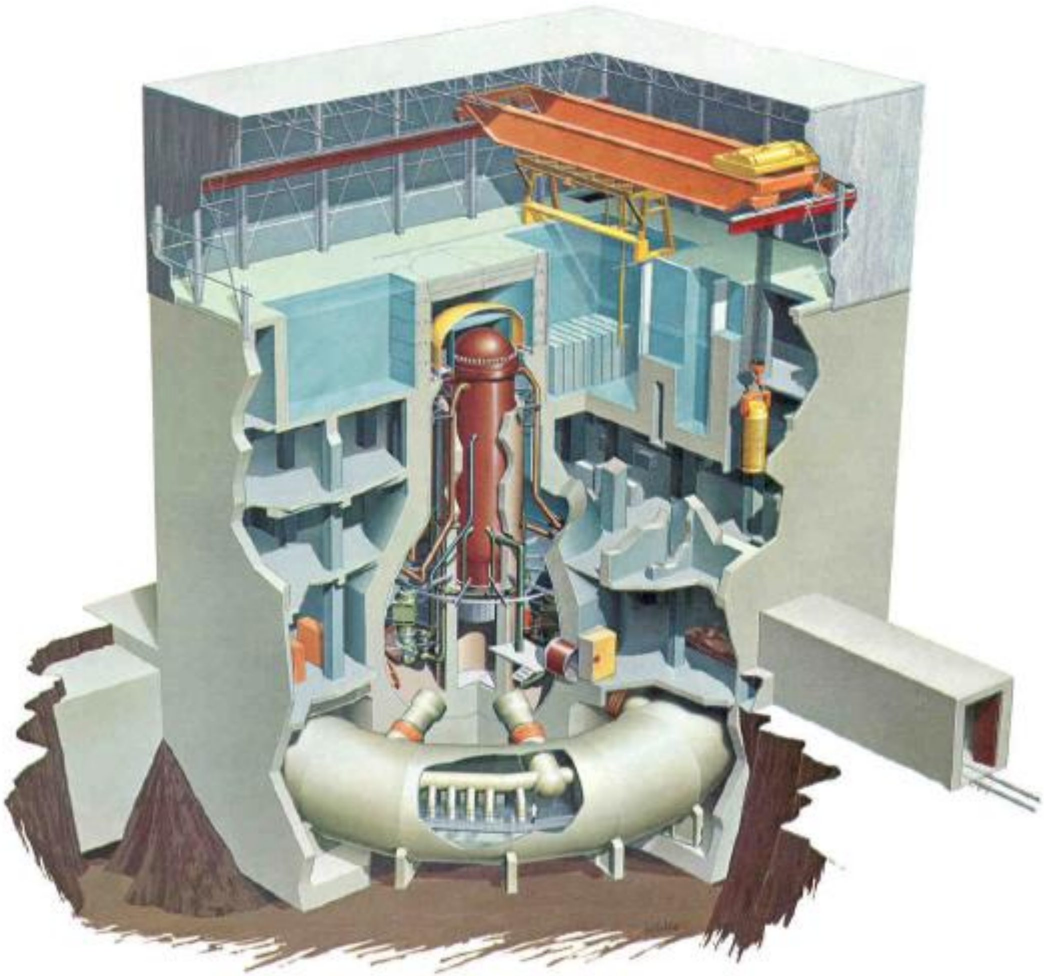
BWR heat removal



BWR emergency core cooling

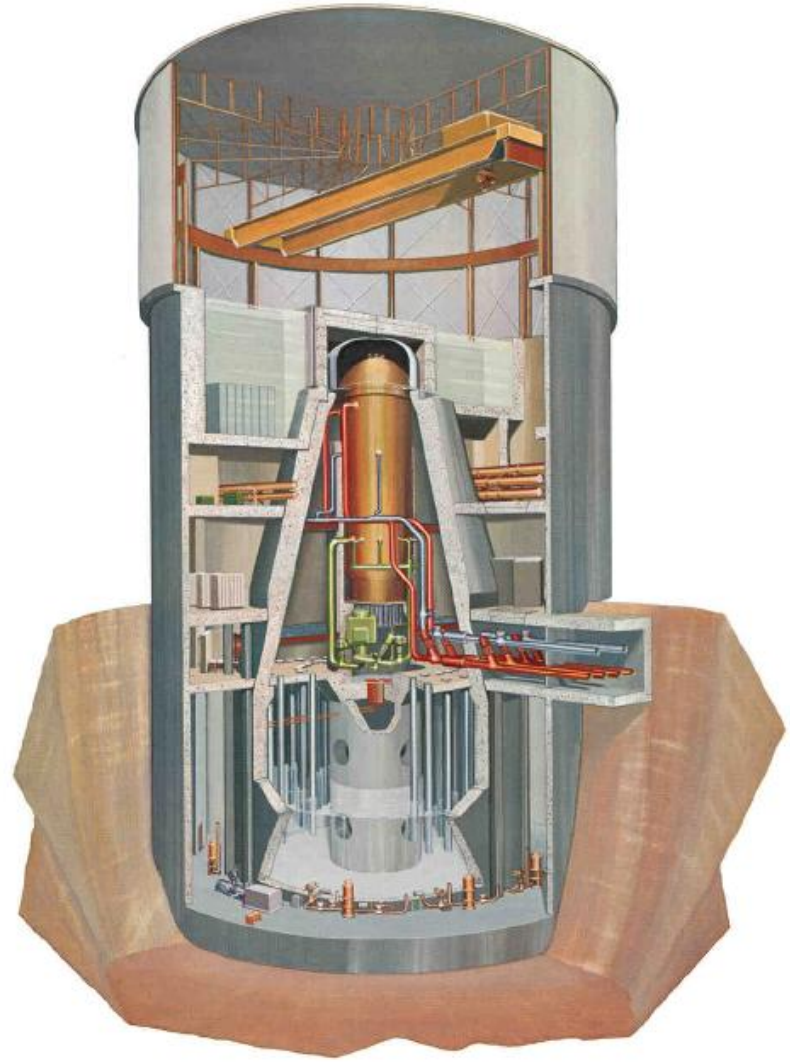


BWR buildings



DRYWELL TORUS

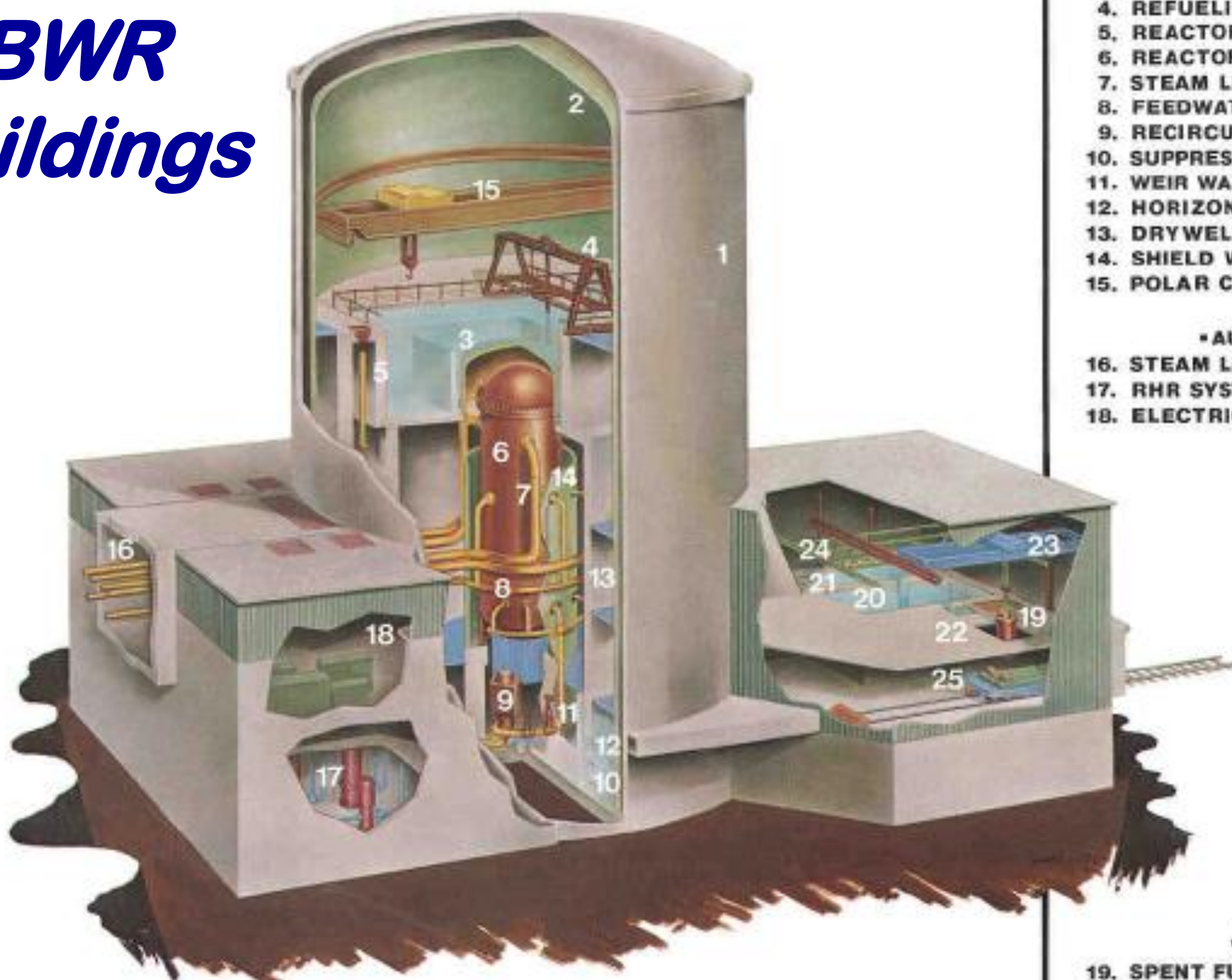
GENERAL ELECTRIC



GENERAL ELECTRIC

BWR buildings

MARK III CONTAINMENT



•REACTOR BUILDING•

1. SHIELD BUILDING
2. FREESTANDING STEEL CONTAINMENT
3. UPPER POOL
4. REFUELING PLATFORM
5. REACTOR WATER CLEANUP
6. REACTOR VESSEL
7. STEAM LINE
8. FEEDWATER LINE
9. RECIRCULATION LOOP
10. SUPPRESSION POOL
11. WEIR WALL
12. HORIZONTAL VENT
13. DRYWELL
14. SHIELD WALL
15. POLAR CRANE

•AUXILIARY BUILDING•

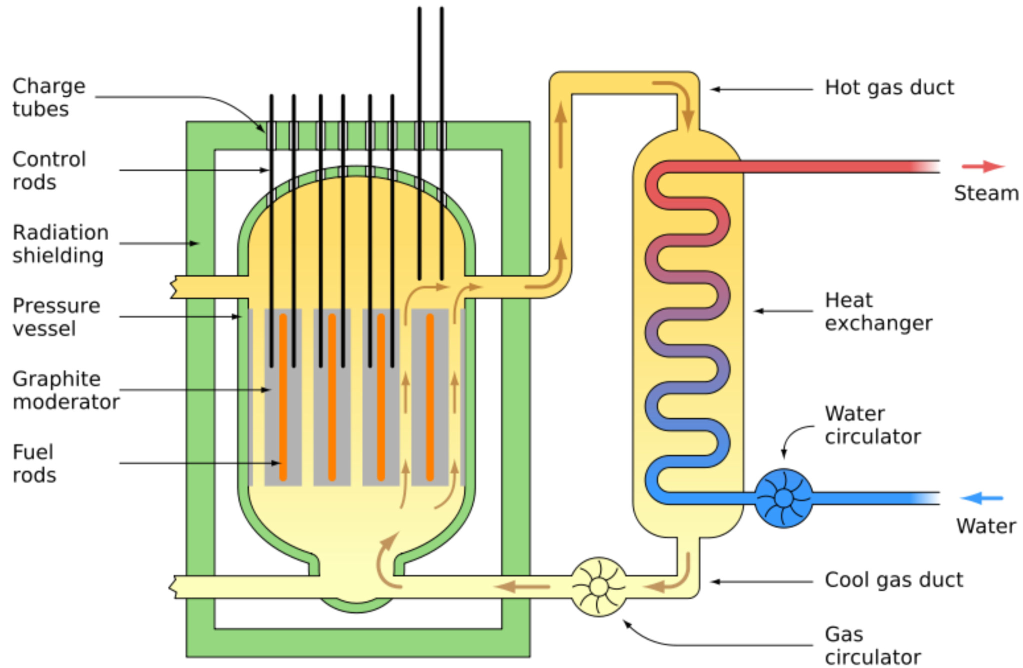
16. STEAM LINE TUNNEL
17. RHR SYSTEM
18. ELECTRICAL EQUIPMENT ROOM

•FUEL BUILDING•

19. SPENT FUEL SHIPPING CASK
20. FUEL STORAGE POOL
21. FUEL TRANSFER POOL
22. CASK LOADING POOL
23. CASK HANDLING CRANE
24. FUEL TRANSFER BRIDGE
25. FUEL CASK SKID ON RAILROAD CAR

GENERAL ELECTRIC

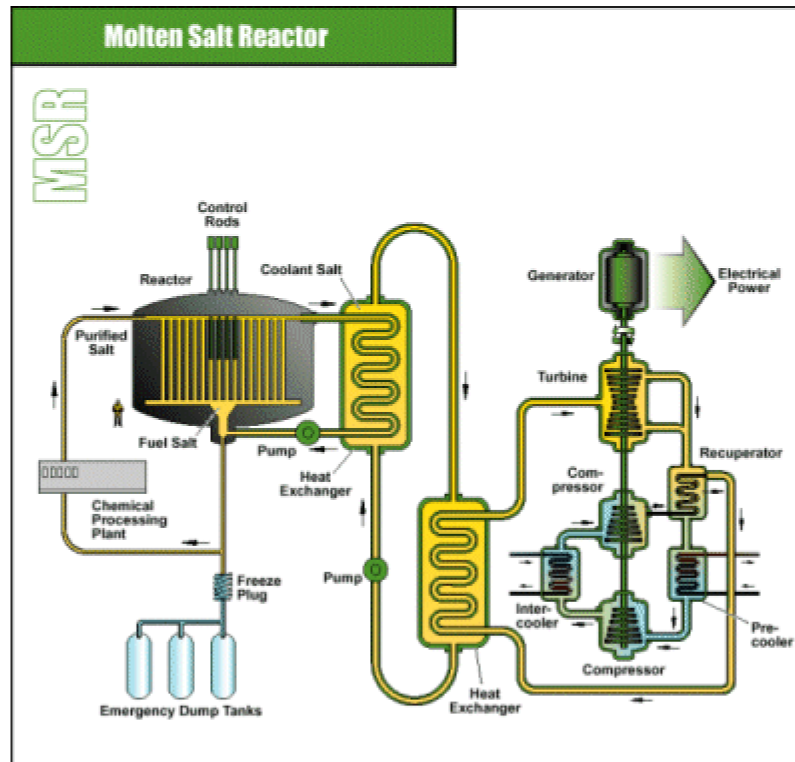
Magnox and UNGG reactors



- Used in UK (26 units). Now obsolete type, but 2 in operation. Used for power and plutonium production. Magnox is now realized in N. Korea.
- Pressurized, CO_2 gas cooled, graphite moderated, natural uranium as fuel. Similar to France UNGG reactor: Uranium Naturel Graphite Gaz
- Coolant is a gas, so explosive pressure buildup from boiling (Chernobyl) is not possible.
- Magnesium non-oxidizing.

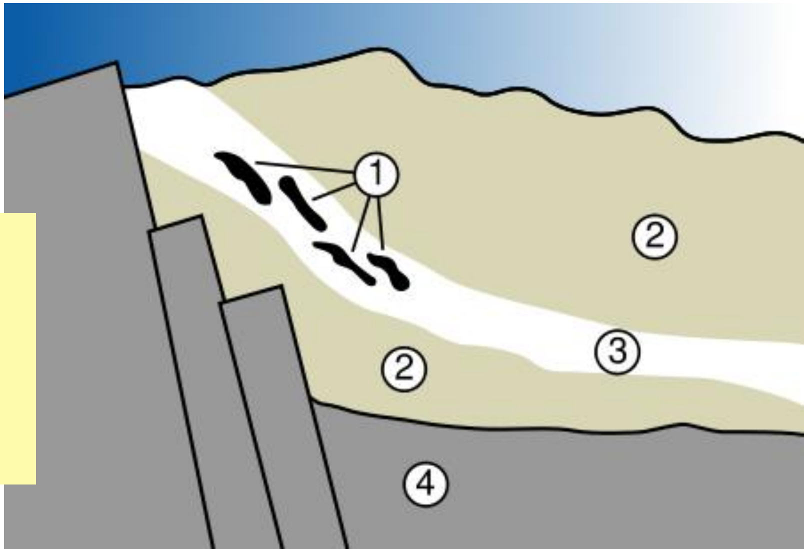
MSR – Molten salt reactor

- Generation IV reactor: primary coolant is a molten salt.
- Nuclear fuel dissolved in the molten fluoride salt coolant (LiF and BeF_2) as uranium tetrafluoride UF_4. Graphite core serves as the moderator.
- Low pressure: makes design simpler and safer, high temperature cooling: makes turbines more efficient.
- Compact: MSRE study to power aircraft.
- Inherently safe, but immature technology. Pressure explosion impossible, meltdown proof.
- Molten salt thorium breeders possible (thorium is abundant and cheap). Can operate decades without refueling.
- Co-locate with reprocessing facility.



Gabon natural fission reactors

- Predicted by Paul Kuroda (Univ. of Arkansas) (1956).
- Fifteen natural reactors found (in 1972) at the Oklo mine in Gabon.
- Nuclear fission reactions took place 1.5 billion years ago, and ran for a few hundred thousand years (100 kW).
- Uranium-rich mineral deposit became inundated with groundwater that acted as a neutron moderator.
- Extensively studied by scientists interested in geologic radioactive waste disposal.



Geological situation in Gabon leading to natural nuclear fission reactors

1. Nuclear reactor zones
2. Sandstone
3. Uranium ore layer
4. Granite