*Neutron absorption cross-sections of important nuclides*

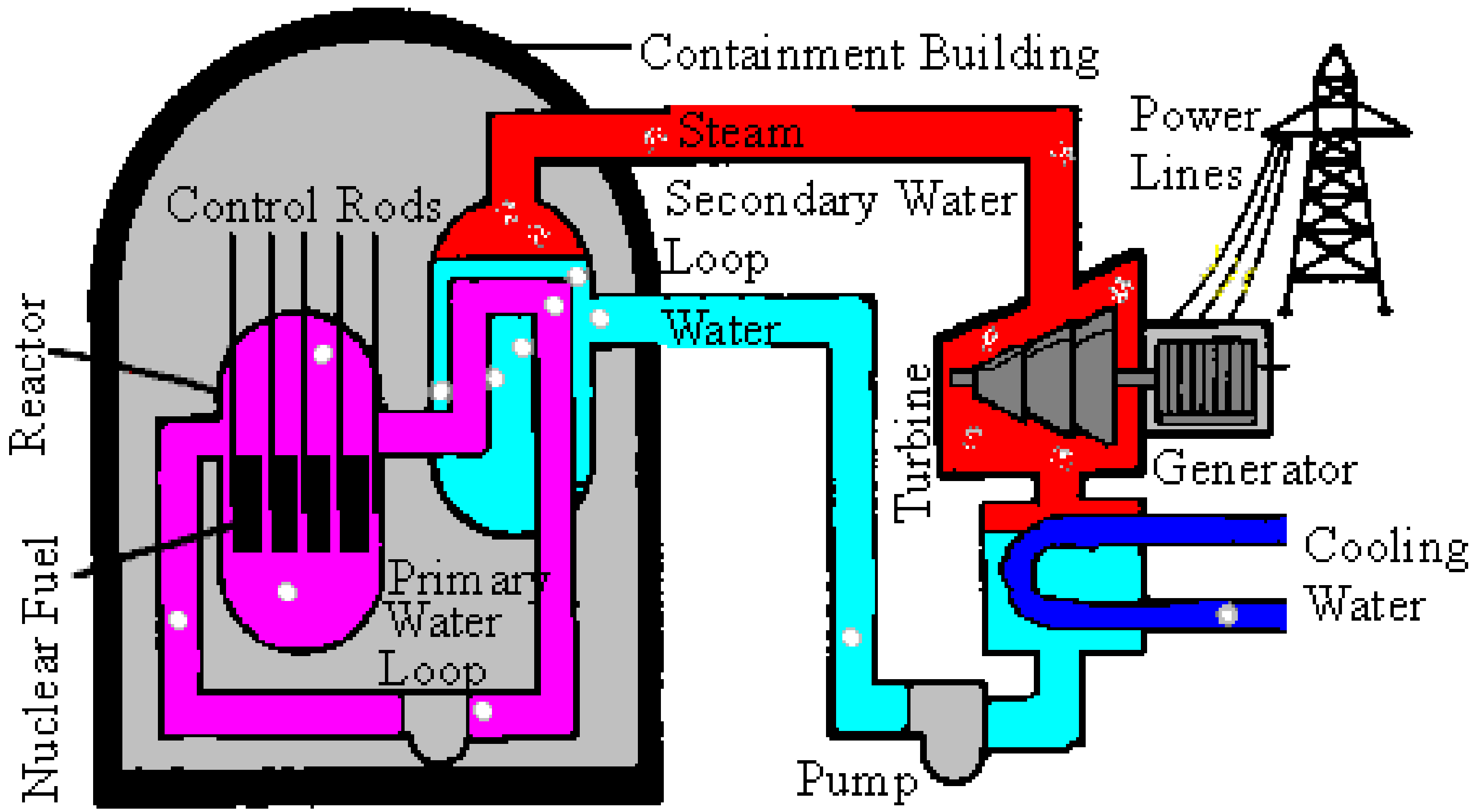
| Nuclide | $\sigma$(barns) neutron absorption | Comments |
|---|---|---|
| ${}^{4}_{2}\mathrm{He}$ | $\sim 0$ | gas cooled reactor coolant, "filled shell", nucleus is an alpha particle |
| ${}^{16}_{8}\mathrm{O}$ | 0.00019 | in light & heavy water, "filled shell", is like 4 alphas |
| ${}^{2}_{1}\mathrm{H}$ | 0.00052 | deuterium, in $D_2O$, good collisional energy transfer with modest absorption |
| ${}^{12}_{6}\mathrm{C}$ | 0.0035 | graphite, good collisional transfer with modest absorption, "3 alphas" |
| Zr | 0.18 | zirconium, non-corrosive, high temp metal used for cladding fuel |
| ${}^{1}_{1}\mathrm{H}$ | 0.332 | hydrogen, in $H_2O$, optimum collisional energy transfer, medium absorption |
| ${}^{238}_{92}\mathrm{U}$ | 2.7 | 99.7% of natural uranium, low fission cross-section, A is even |
| ${}^{115}_{49}\mathrm{In}$ | $3 \times 10^4$ | at 1.46 eV resonance peak |
| ${}^{235}_{92}\mathrm{U}$ | 681 | 0.7% of natural uranium, fission cross-section is similar |
| ${}^{239}_{94}\mathrm{Pu}$ | 1022 | fission cross-section is 750 barns |
| ${}^{10}_{5}\mathrm{B}$ | 3840 | boron, graphite contaminant, boronated water used for control |
| ${}^{135}_{54}\mathrm{Xe}$ | $2.6 \times 10^6$ | fission product, "poison", can cause restart problems (slow creation) |
| ${}^{113}\mathrm{Cd}$ | 7800 | at 0.175 eV resonance peak, used in control rods |

- Reactor designs PWR, BWR, CANDU, Graphite, Fast Breeder
- Void coefficient
- Economics, price of uranium, enrichment, breeding, reprocessing
- Biological safety, activation, half-life
- Reactor Safety
- Three Mile Island, Chernobyl

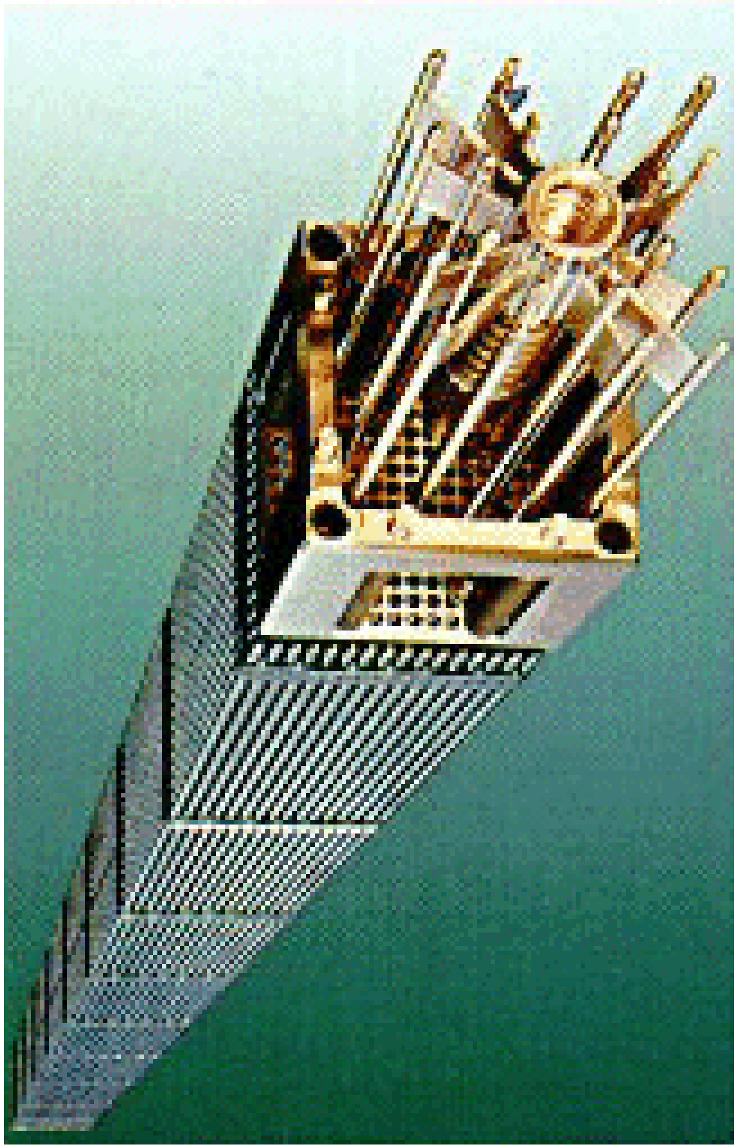
# Reactor types

- Heavy water moderator (natural uranium)
  - CANDU
- Light water moderator (2-4% enriched uranium)
  1. Pressurizing water allows operating at higher temperatures than atmospheric boiling point (PWR)
  2. Boiling water reactor (BWR)
- Graphite moderator (RMBK-Chernobyl,enriched)
- Gas-cooled reactors
- Sodium-cooled breeder reactors (highly enriched, fast neutrons)

# Pressurized Water Reactor



# Fuel and control rod assembly for PWR



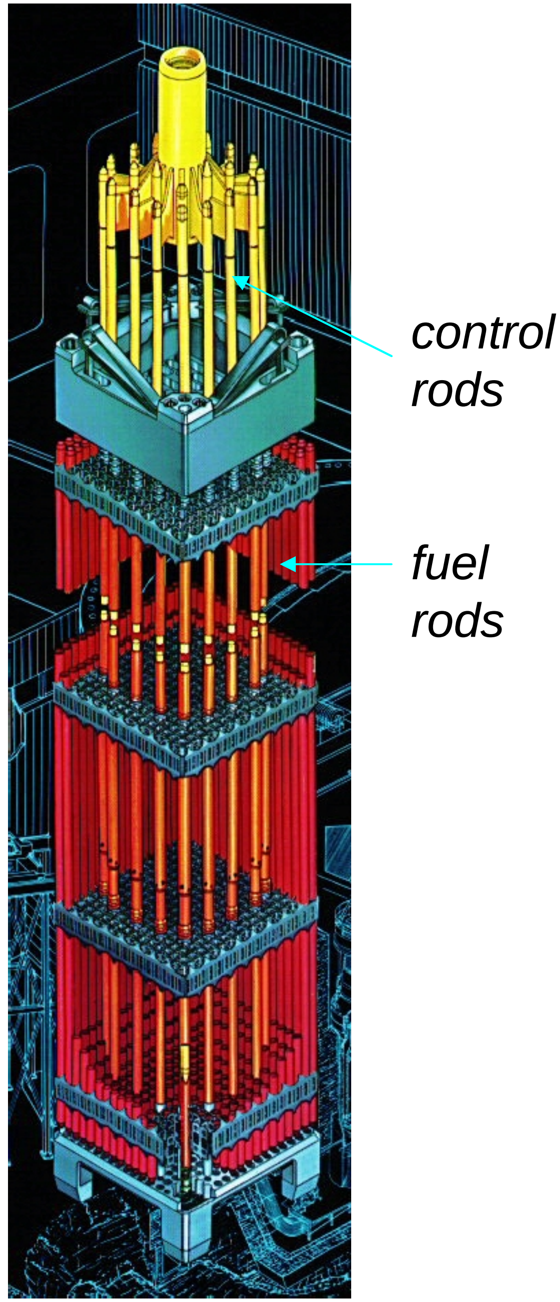
PWR fuel/control assembly
Length is 4.4 m
Cross-section= 21 cm x21cm

17x 17 array of which
264 are fuel rods (red),
25 are control rods (yellow)
Rods are 1 cm diameter
Fuel in form of $UO_2$ pellets
clad by zircaloy (thin, has
low neutron absorption)

Coolant flows between rods
Soluble boric acid in coolant
acts as adjustable poison to
compensate for long term
reactivity changes

Uranium mass is 461 kg

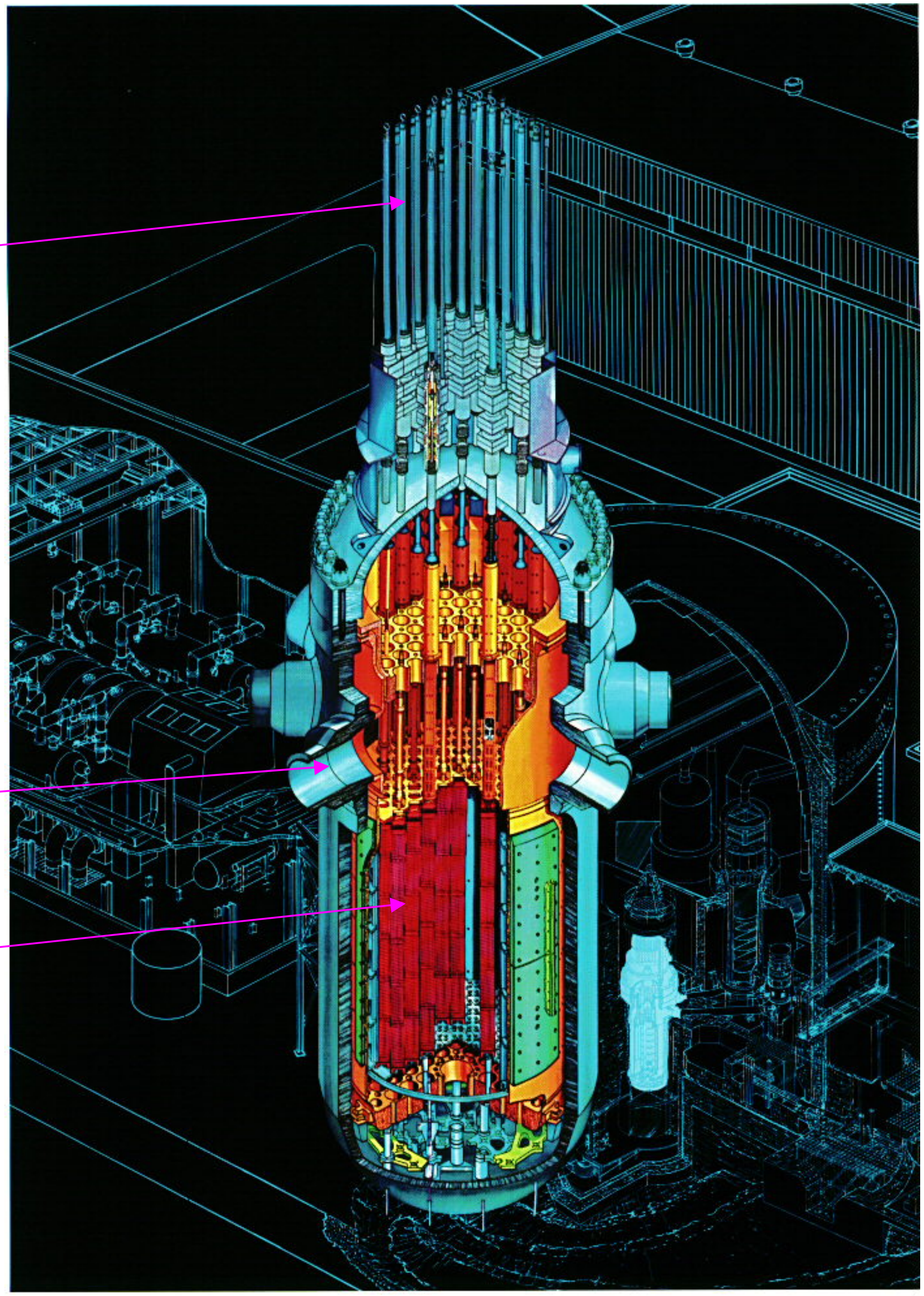
Reactor has 190-240 assemblies
Total uranium in reactor
is 90-125 tons




Control rod drive

Water inlet/outlets

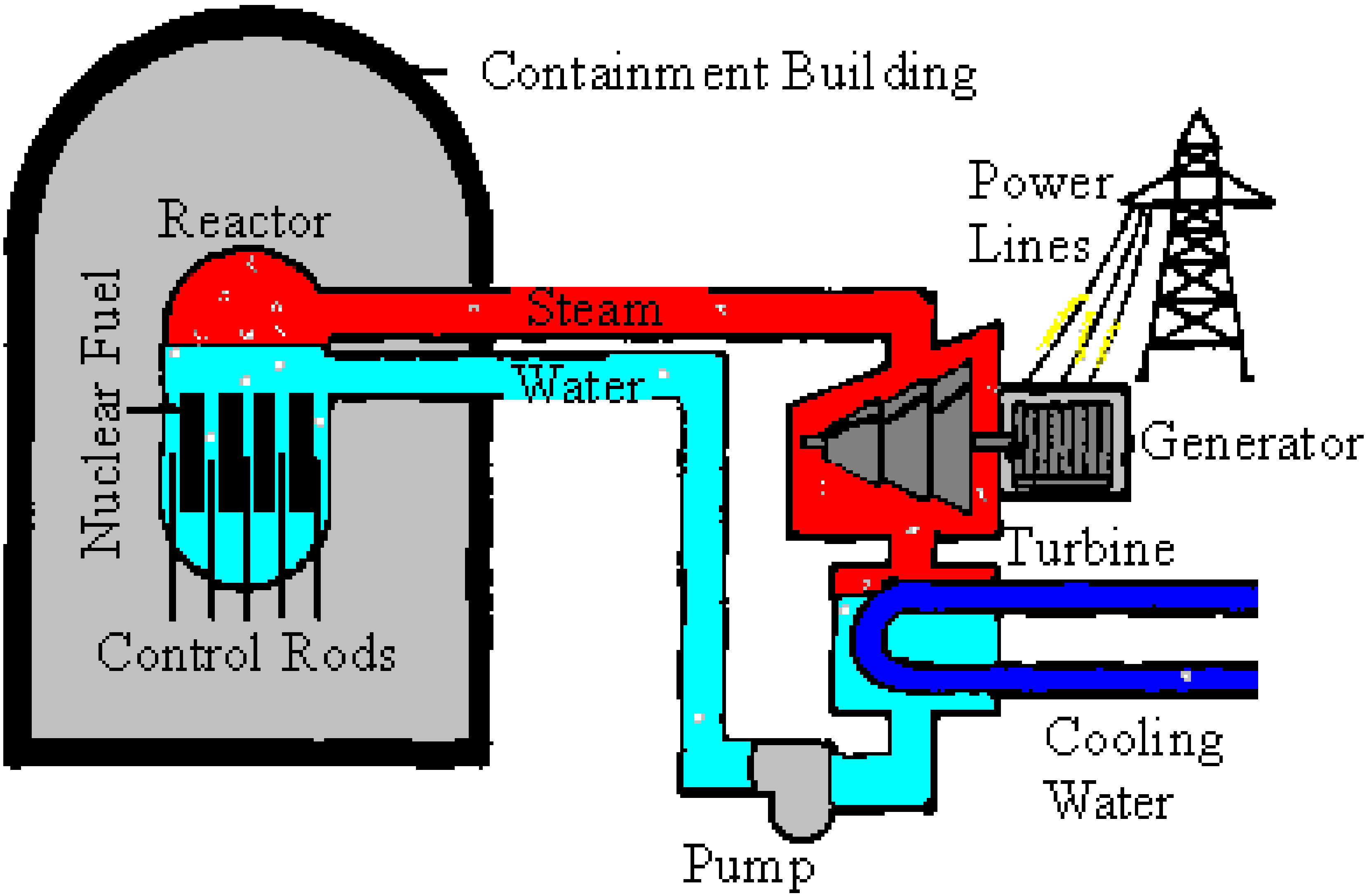
Fuel assemblies



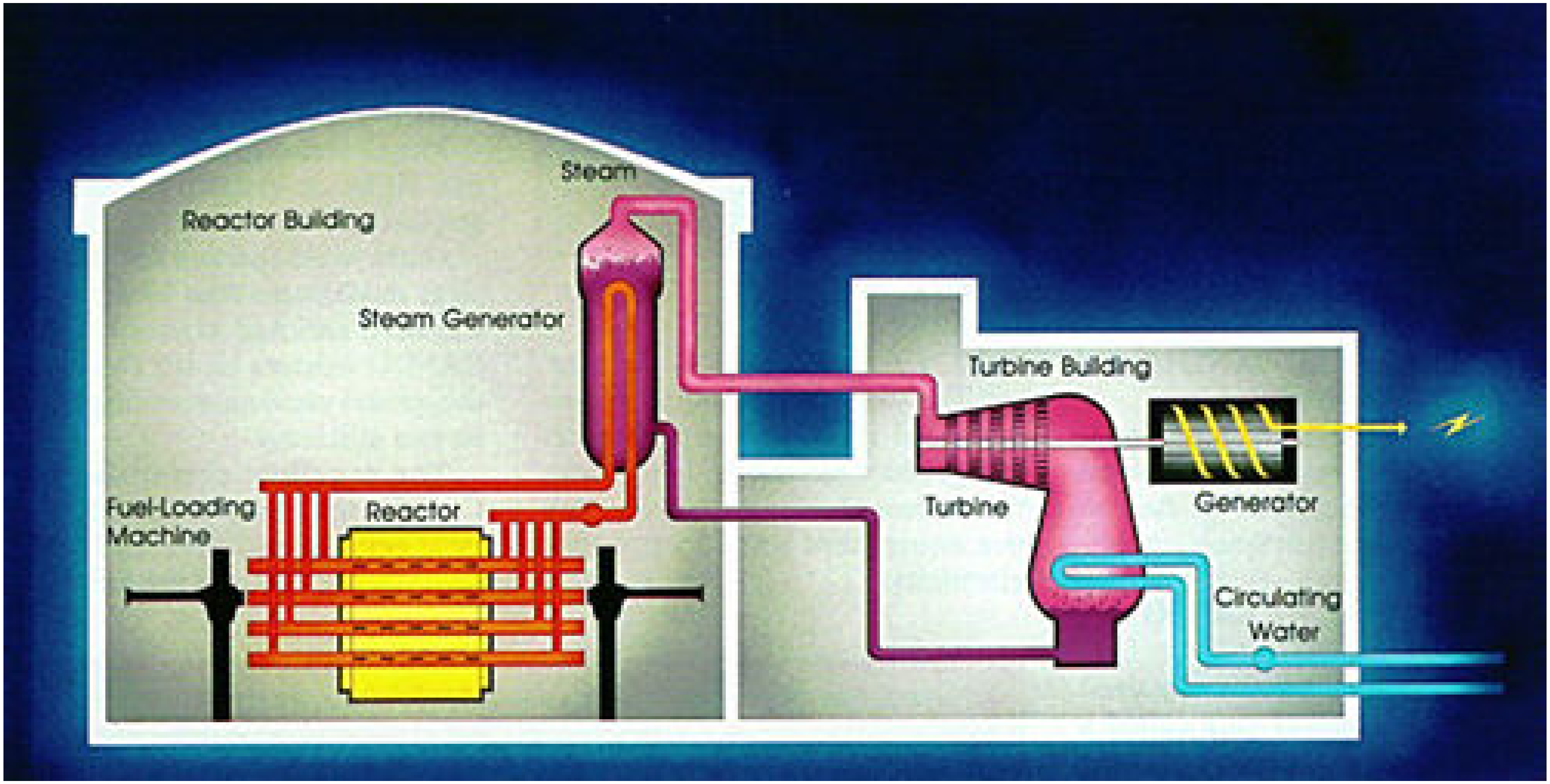
Westinghouse NUCLEAR REACTOR

Pressurized
Water
Reactor

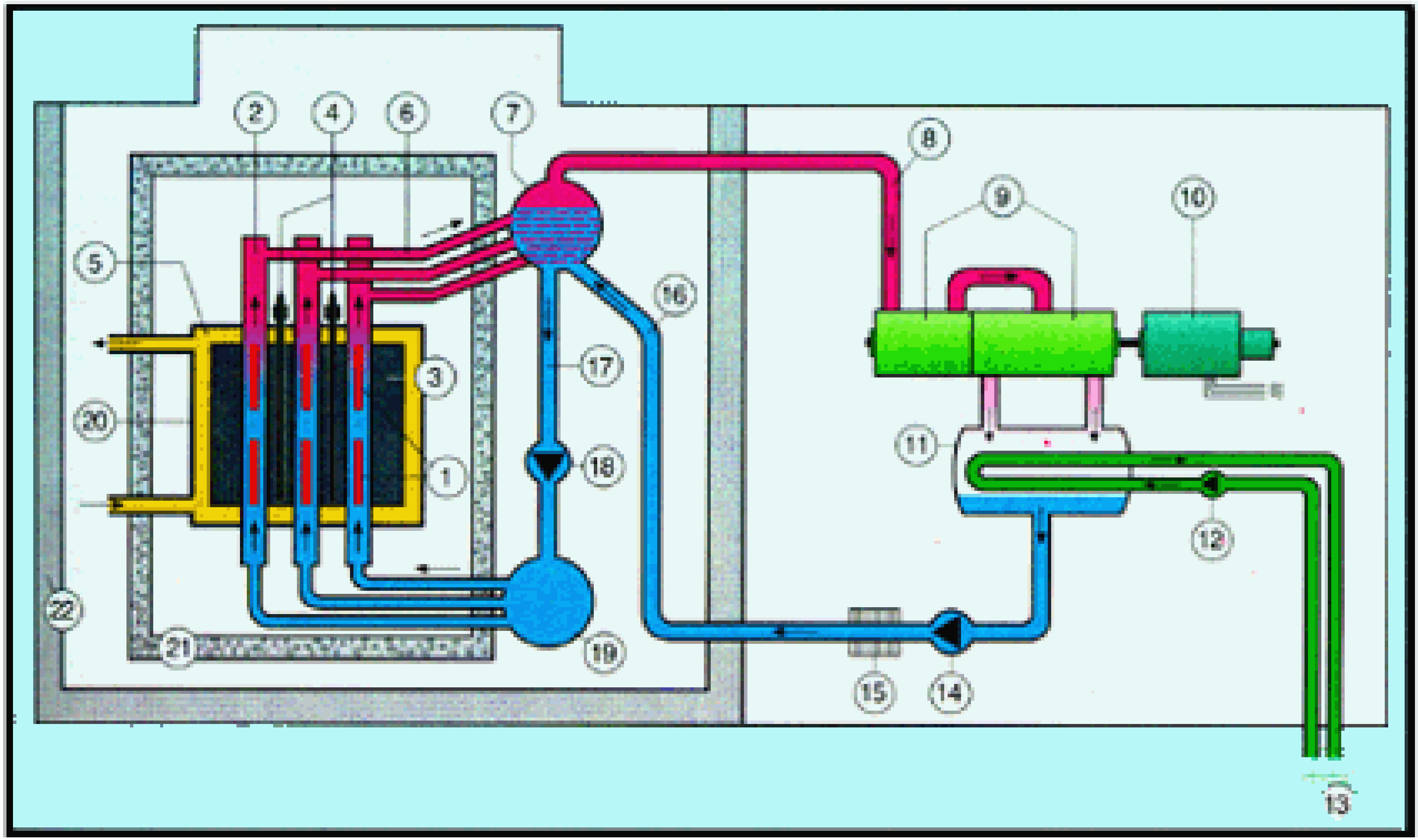
*Boiling water reactor*



*CANDU reactor*



*RMBK reactor (Russian graphite-moderated, Chernobyl-type)*



# Reactor design

- Neutron economy
  - Fission produces neutrons which can be used to stimulate more fission
  - Fission products also produce some neutrons after a delay
  - Some neutrons are lost (fly out or absorbed by neutron absorbers)
  - Neutrons can be reflected by some materials
  - Fission cross section much higher with slow neutrons (thermal)

## *Number of neutrons produced per thermal neutron absorbed (from Lilley)*

| Nucleus | Density ($g\ cm^{-3}$) | Fission (barns) $\sigma_f$ | Capture (barns) $\sigma_c$ | Absorption (barns) $\sigma_a = \sigma_f + \sigma_c$ | Scatter (barns) $\sigma_s$ | Neutrons per fission $\nu$ |
|---|---|---|---|---|---|---|
| $^{235}U$ | 18.7 | 579 | 101 | 680 | 10 | 2.42 |
| $^{238}U$ | 18.9 | 0 | 2.72 | 2.72 | 8.3 | 0 |
| $^{nat}U$ | 18.9 | 4.17 | 3.43 | 7.60 | 8.3 | |

Let $\eta$ be number of neutrons produced per thermal neutron *absorbed* (fission + capture)

$$\eta = \nu \frac{\sigma_f}{\sigma_f + \sigma_c} = 2.42 \times \frac{579}{579 + 101} = 2.06 \qquad \text{for } ^{235}U$$

$$\eta = \nu \frac{\sigma_f}{\sigma_f + \sigma_c} = 2.42 \times \frac{4.17}{4.17 + 3.43} = 1.328 \qquad \text{for natural uranium}$$

$\eta$ can be varied between these values by adjusting enrichment

# Loss of neutrons due to capture

- Fast neutrons produced by fission must be slowed down to be thermal
- During this process, some neutrons will be captured by $^{238}U$

- Probability that neutrons are <u>not</u> captured is denoted by *p*

# Thermal utilization factor

- Not all thermal neutrons are absorbed by fuel
- Some are absorbed by moderator and structure
- Let fraction absorbed by fuel be denoted by $f$

# Leakage

- Let $l_f$ denote fraction of fast neutrons which leak out (i.e., before moderation)
- Let $l_s$ denote fraction of slow neutrons which leak out (i.e., after moderation)

- These provide another multiplier

$$(1-l_f)(1-l_s)$$

Neutron multiplication factor

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

Define

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

as the multiplication factor

if there is no neutron leakage

Example for graphite-moderated reactor (from Lilley)

Using 1.6% enrichment, $\eta = 1.6454$

Graphite moderator:

$p = 0.749$

$f = 0.834$

These give

$$\begin{aligned} k_\infty &= \eta p f \\ &= 1.6454 \times 0.749 \times 0.834 \\ &= 1.028 \end{aligned}$$

This exceeds unity.

Could make a reactor if leakage kept below 2.8%

Reactor should be designed so that $k = 1$

only when delayed neutrons included.

# Economics

- Energy content of 1 kG of natural uranium
- Take into account that only 0.72% is $^{235}U$
- Consider each fission reaction gives 200 MeV
- Count up number of $^{235}U$ atoms in 1 kG
- Convert energy to kilowatt-hours
- Use wholesale price for electricity,
  - say 1 cent per kW-hr
- This will give maximum possible value of energy

# Other costs

- Must then subtract other costs
- E.g. plant costs $2 billion, lasts 30 years, produces 1000 MW
- Like buying a house so with interest rates, plant cost is two or three times as much, say $6 billion
- Divide total kW-hr produced by cost to get plant costs per kW-hr
- Do same for labor (operating costs)

# Control

- Reactor designed to rely on delayed neutrons
- Use control rods (neutron absorbers) to adjust *k*
- *k* kept hovering around unity in normal operation
  - Like maintaining constant speed in an automobile
- Reactor turned off by inserting control rods completely
- several sets of control rods (incl. emergency)
- catastrophe if *k* exceeds unity *using only prompt neutrons (prompt critical, avoid at all costs)*
  - neutron flux grows exponentially, cannot halt
- Moderator (e.g. water) absorbs neutrons

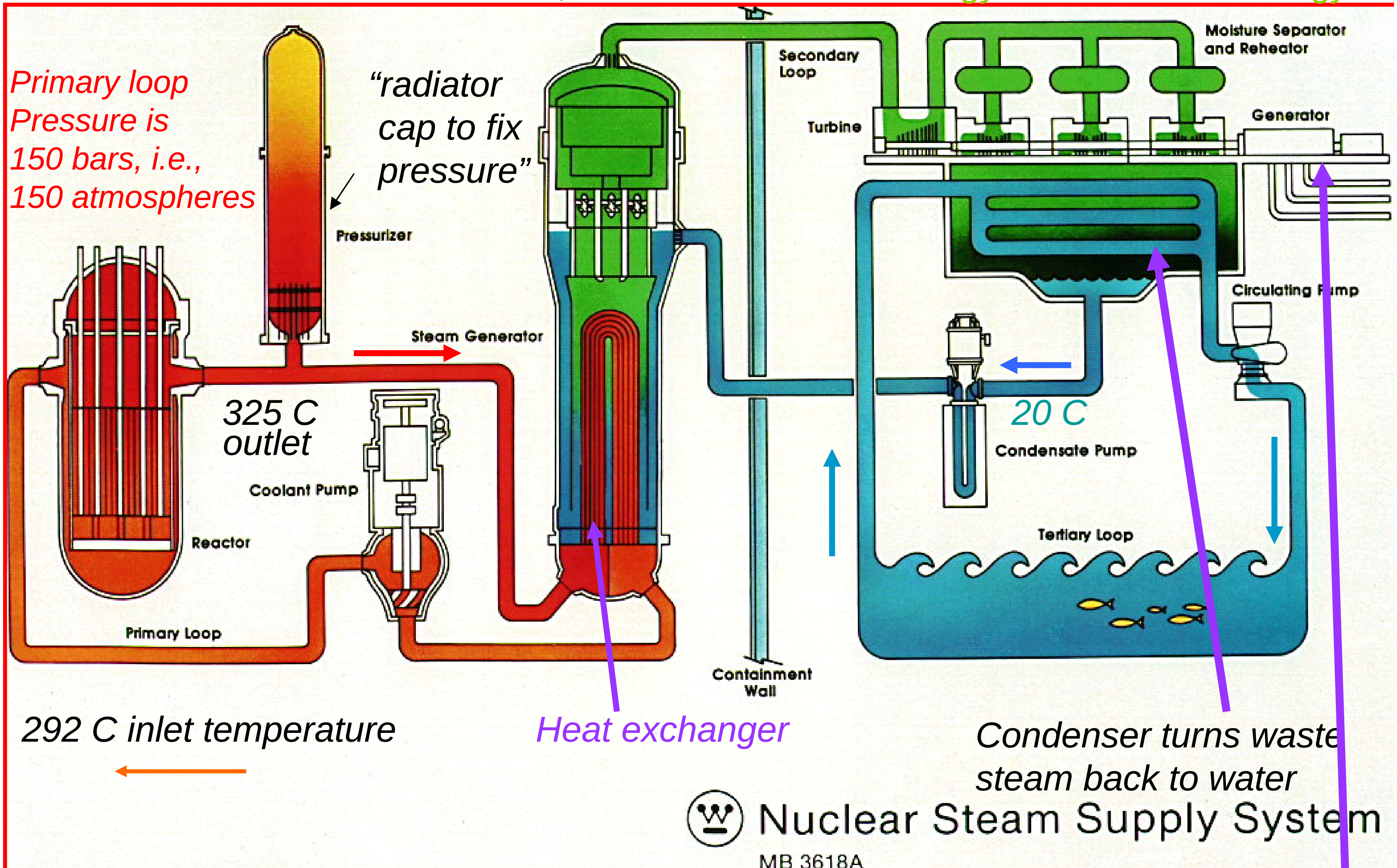
# Power extraction
# (PWR as example)

Three water circuits used in PWR

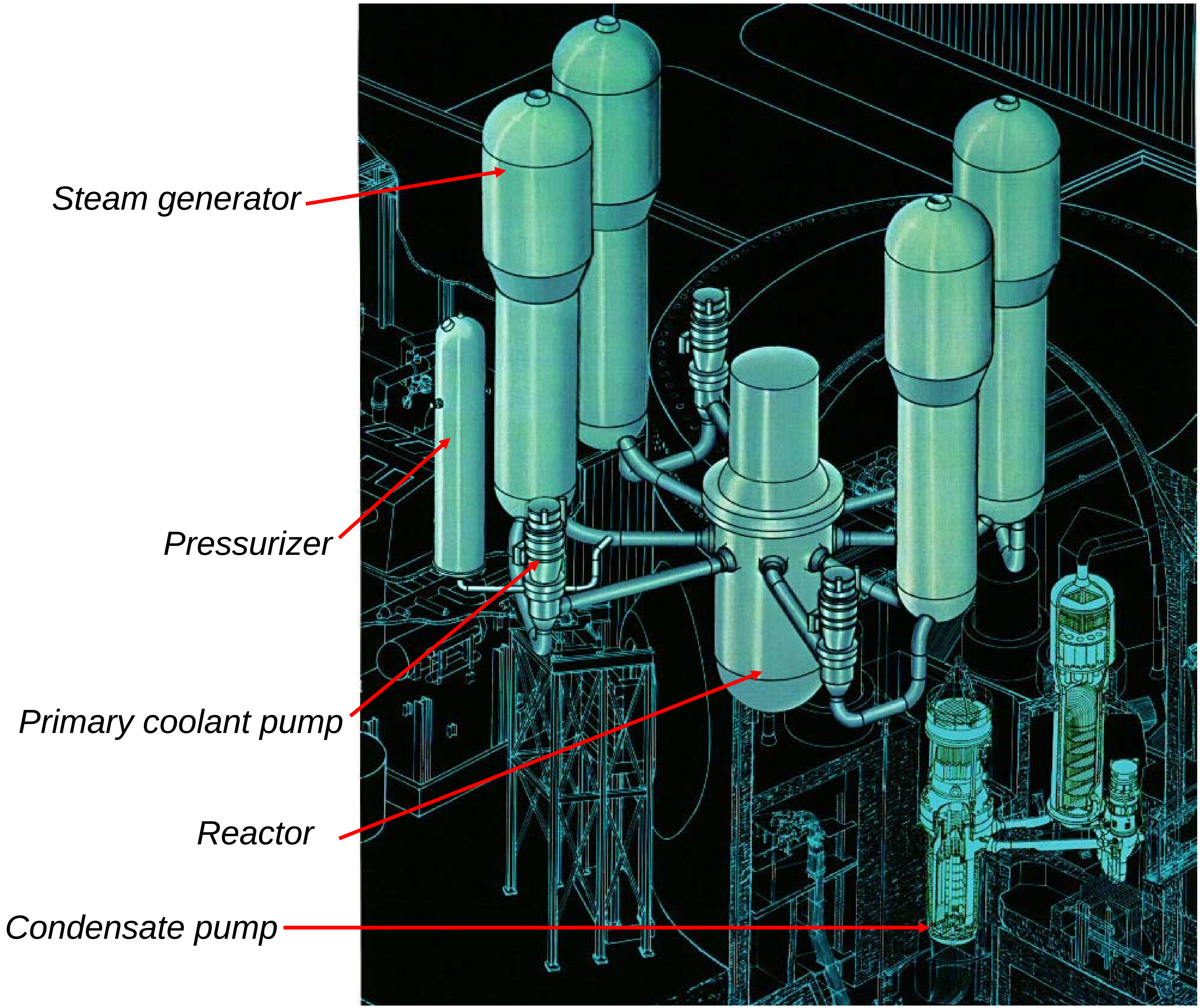
1) Primary circuit flows through reactor core
   - Inlet temperature is 292 C, outlet temp is 325
   - Reactor provides this 33 C temp rise
   - Reactor circuit is pressurized so no steam
   - High temp operation improves Carnot efficiency
2) Heat exchanger transfers heat to secondary circuit where steam is produced (at lower pressure)
   - Steam runs turbines to make electricity
3) Spent steam is condensed using heat exchanger with cold water in tertiary circuit

Steam output,
70 bars, 280 C

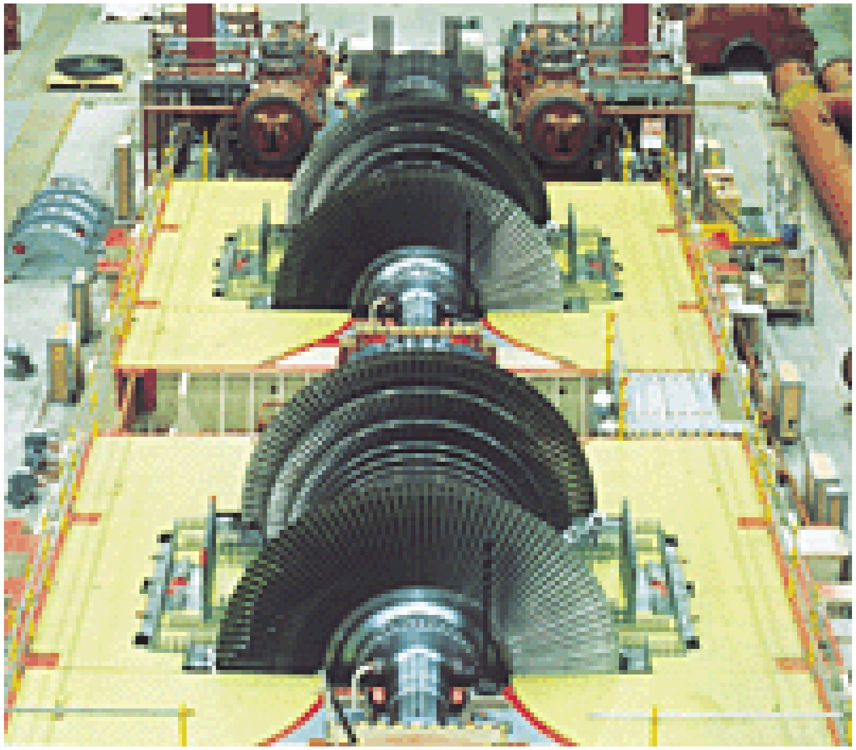
Steam spins turbines, convert
heat energy to mechanical energy




Generator converts mechanical energy
to electrical energy


Steam generator
Pressurizer
Primary coolant pump
Reactor
Condensate pump

# Turbines turn heat into mechanical energy



*Uncovered 52" turbines (Mitsubishi Heavy Industries, Japan)*

*Turbines at Seabrook Nuclear Plant, Seabrook, New Hampshire*

# Turbine Carnot cycle efficiency

- Basic result from thermodynamics (Oxtaby, p.242)
- Efficiency for turning heat into mechanical energy

Carnot efficiency

$$\eta = \frac{T_h - T_l}{T_h} = \frac{\Delta T}{T_h} \text{ in Kelvin}$$

$$= \frac{280 - 20}{(280 + 273)} = 0.47$$

- Governs maximum turbine efficiency
- Actual plant efficiency is about 32%

# Reactor design must take into account various *Operating States*

(ref: Collier/Hewitt)

1. Normal operation & operational transients: Steady-state normal operation, Starting up/Shutting down

2. Upset condition: Not normal, but can reasonably be expected to occur several times during lifetime of reactor (e.g., power lines struck by lightning, minor earthquake in California), typically externally induced and no damage to plant

3. Emergency event: Unlikely in lifetime of a single plant (e.g., 1 in 10 chance in lifetime of single plant), so if USA has 100 reactors with 30 year lifetime would expect one US reactor emergency every 3 years, some damage to plant (e.g., small broken pipe, leakage, small electrical fire, stuck valve)

4. Limiting fault condition: Very rare event (say 1 in 10,000 chance) which is not supposed to happen ever, e.g., major earthquake, severance of important pipe, failure of fuel canning, reactor sustains major damage

Reactor must be designed so that no radiation harmful to public is released for any of these situations

# PWR potential accident analysis

- System is designed to transfer substantial energy from reactor to load
- Typical thermal output in normal operation is 3400 MW
- When reactor turned off there is still 200 MW thermal produced by fission products
- If heat is not removed, this 200 MW overheats fuel rods

# Fuel rod heat output

- Single fuel rod generates about 40 kW per meter in normal operation
- Generates about 2 kW per meter after shutdown due to decay of fission products
- In normal operation, primary water circuit removes this heat
- Need to maintain some flow after shutdown to remove 2 kW/meter

# Consequences of fuel rod overheating

If heat not removed on shutdown, then get overheating

- Loss of system pressurization causes steam to be produced
  (like automobile radiator boiling over when pressure cap removed)
- At ~800 C Zircaloy cladding swells, can block cooling channels
- At ~1400 C Zircaloy breaks, fission products no longer clad, mix with water
- At ~1400 C steam interacts with Zircaloy to make $H_2$
- At 2700 C fuel melts

- With no cooling, fuel melting occurs in ~ 8 minutes
- Thus, essential to maintain cooling
  - Hence, many redundant systems to avoid loss of cooling
- Accidents can interrupt coolant flow (leaks, pump failure)

# Containment Vessel

- A “last-ditch” safety measure
- Enclose reactor and all of primary circuit by containment vessel
- Typically reinforced concrete or steel dome 50 m diameter
- Designed to withstand pressure differential of 4-5 bar
  (1 bar =1 atmosphere)
- Designed to handle overpressure associated with loss of coolant, fuel melting, gas production, fire
- Volume is many times reactor volume
- Worked fine at Three Mile Island (had insignificant release of radioactivity despite partial meltdown)
- Chernobyl and other RMBK reactors do not have containment vessel (reason why there was such a large release of radioactive material)

# Chernobyl accident: prompt criticality

- Loss of water (leak, boiling) replaces water by air, can cause increase of $k$
- At Chernobyl, design allowed $k$ to exceed unity if water removed (positive void coefficient)
  - Moderation provided by graphite, so neutrons still moderated when water lost
  - Control rod design allowed void to develop when control rod moved abruptly