Nuclear Reactor Technology, SH2702

Lecture No8

Title:
Gen-IV Goals:
Sustainability - Fuel Conversion and Breeding

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Outline of the Lecture

- Generation IV concepts and goals: Sustainability

- Fuel Conversion and breeding
  - Conversion and breeding ratio
  - Reactor doubling time

- Principles of design of FBRs (Fast Breeder Reactors)

Gen IV Concepts

- Sustainability is difficult to achieve due to complexity
  - Individual elements of the system are not sustainable.
  - The whole system might be sustainable.

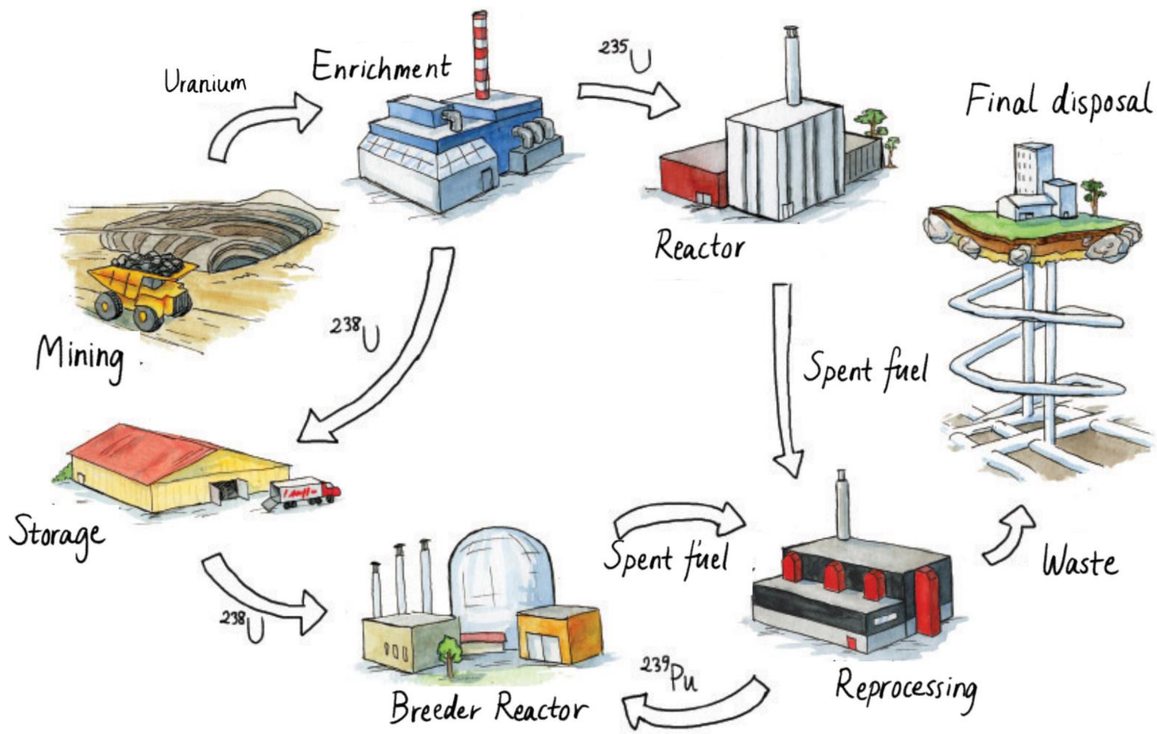
- We have found ~300 years fuel supply
  - Extraction of uranium from seawater (currently too expensive) would make available 4.5 billion metric tons of uranium—a 60,000-year supply at present rates.
  - Breeding fuel can further expand amount of available fuel.

- What is needed for sustainability of nuclear?
  - Fuel
  - Safety
  - Waste management

Gen IV Concepts

- Fourth generation nuclear power, or Generation IV, implies a system of
  - reactors and nuclear fuel cycle facilities
    - fuel fabrication plants and reprocessing facilities
- that together may make the nuclear power sustainable in a very long term.

- More fuel-efficient and economically competitive.
- Small amount of long lived radioactive wastes.
- Safe.
- The fuel cycle is designed so that diversion of fissile material for weapons production is unattractive.
  - Uranium and plutonium are never separated but only ever present mixed together and with other elements.
  - The quality of the nuclear material thus becomes too poor to serve as weapons material, but good enough for fuelling a reactor.



Gen IV Concepts

- Uranium in nature consists of ^{235}U (fissile) and ^{238}U, which constitutes 99.3 % of natural uranium.
- Today, there are huge stores of ^{238}U isotope available, as a by-product of the process where ^{235}U was enriched over the years to the concentration required for the current reactors.
- There is no need for mining uranium for the fourth generation reactors until after a very long time even if the nuclear power production is significantly expanded.
- Would nuclear power production remain at the current level, the already mined uranium would be sufficient to operate the reactors for several thousand years.

Types of FSRs

- The interest in fast spectrum reactors (FSR) was for developing
  - Breeding capability.
  - Transmute minor actinides (more recently).

- FSRs are divided into three categories:
  - Experimental and test reactors
    - Power level up to 100 MWth
    - Built to test the concepts and obtain experimental data

- Demonstration or prototype reactors
  - Power in general in range 250 – 350 MWth
  - Built to demonstrate the technology viability

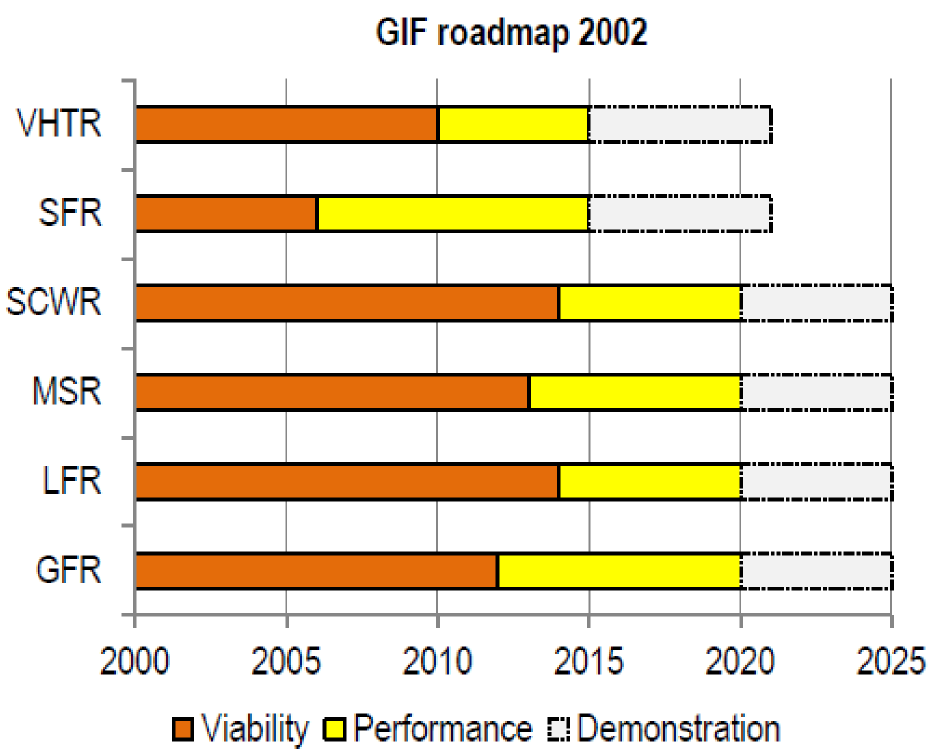
- Commercial-size reactors

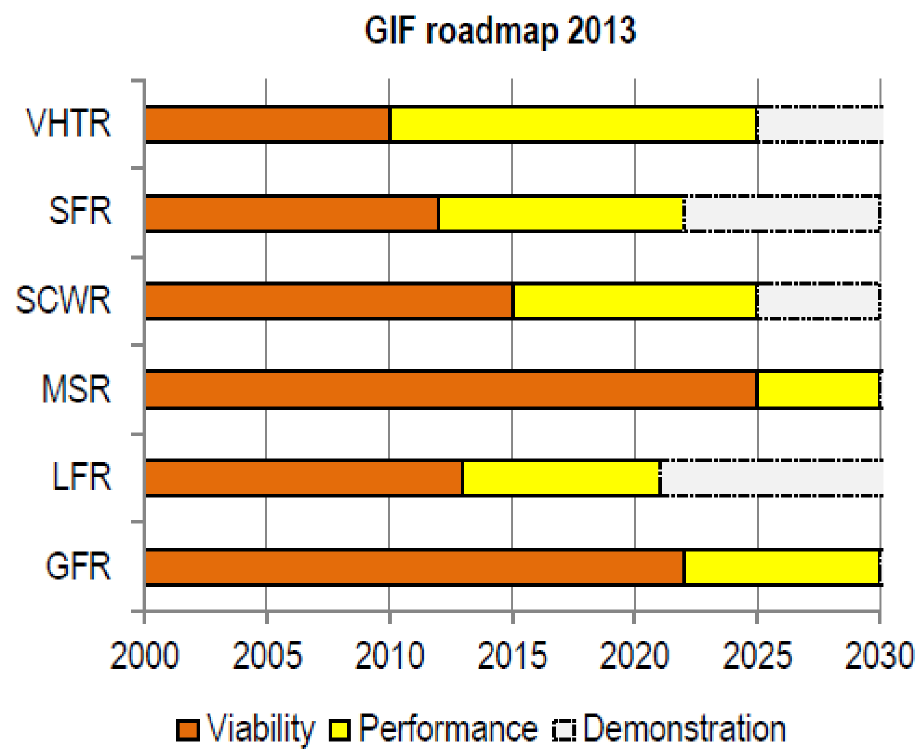
Types of FSRs

- The following technologies are considered:
  - – Sodium-cooled Fast Reactor (SFR)
  - – Lead-cooled Fast Reactor (LFR)
  - – Gas-cooled Fast Reactor (GFR)
  - – Molten Salt Fast Reactor (MSFR)
  - – Supercritical-water-cooled reactor (SCWR) with fast neutron spectrum
    - The only one that has not been built yet

Gen-IV Roadmap

- System development timeline
  - Compared Roadmap 2002 and GIF, January 2014
    - Fukushima caused a delay
  - Delay for LFRs are the smallest





Three European Candidate FSR

- Gas Fast Reactor – GFR
  - ALLEGRO - demonstration reactor with potential sites:
    - Jaslovské Bohunice, Slovakia
    - Dukovany, Czech Republic
    - Paks, Hungary
- Lead Fast Reactor – LFR
  - ALFRED – Advanced Lead Fast Reactor European Demonstrator, with potential site:
    - Cerna Voda, Romania
- Sodium Fast Reactor – SFR (currently on hold)
  - ASTRID – Advanced Sodium Technical Reactor for Industrial Demonstration with site:
    - Marcoule, France

Operation & Design

- Basic characteristics of fast reactors
  - Conversion chains/ratio
  - Breeding ratio and breeding gain
  - Doubling time

- Design principles of fast breeder reactors
  - Core and blanket arrangements
  - Fuel lattice, fuel assembly, vessel internals
  - Pool vs loop design

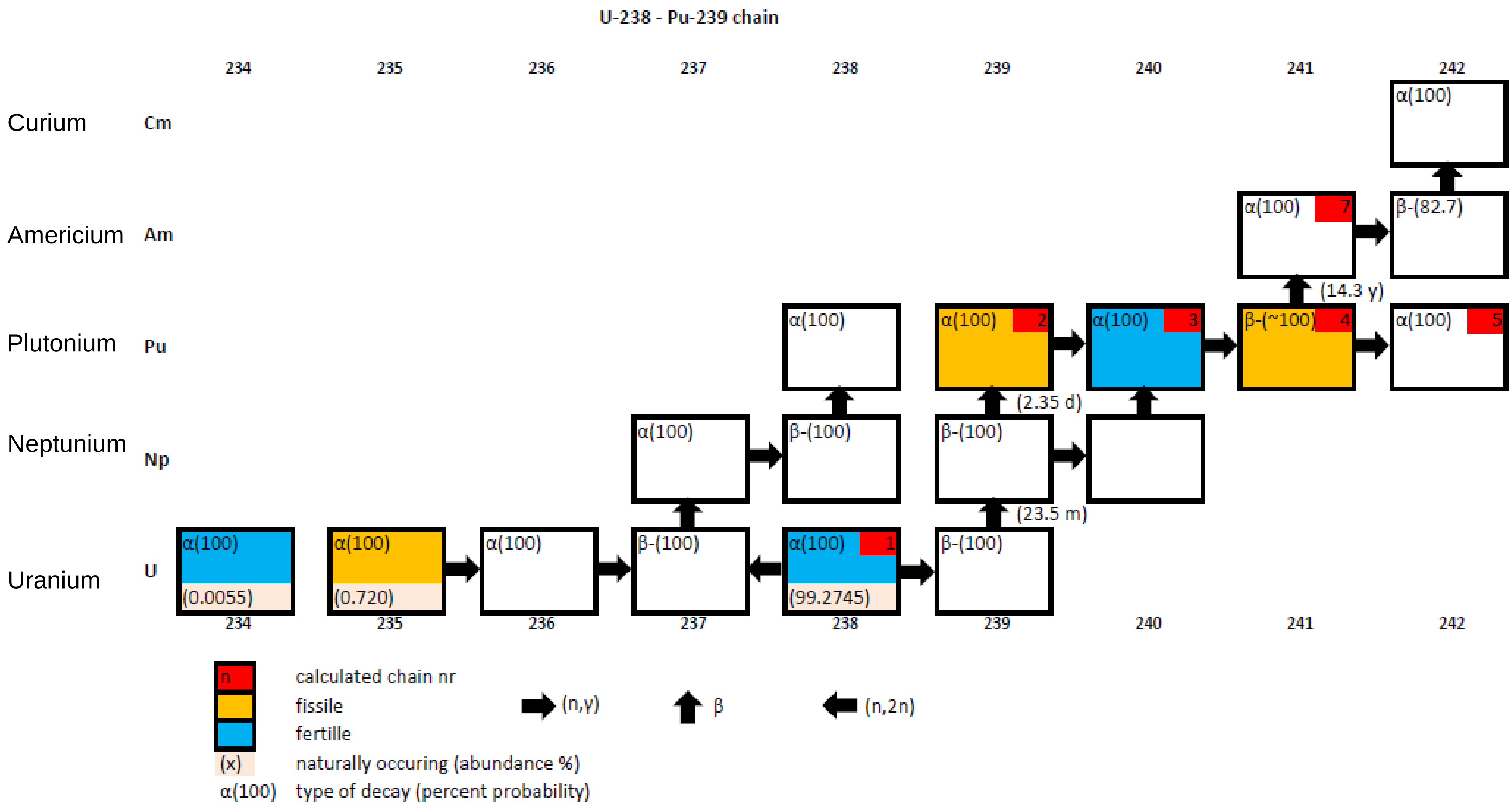
Basic Characteristics of Fast Reactors

- Fuel **breeding** is resulting from fuel conversion chains
  - U-238 → Pu-239 and
  - Th-232 → U-233

- In order to achieve breeding, a fertile isotope
  - (U-238, Pu-240, Th-232, U-234)
- must be converted via neutron capture (n,γ) into a fissile isotope
  - (Pu-239, Pu-241, U-233, U-235)

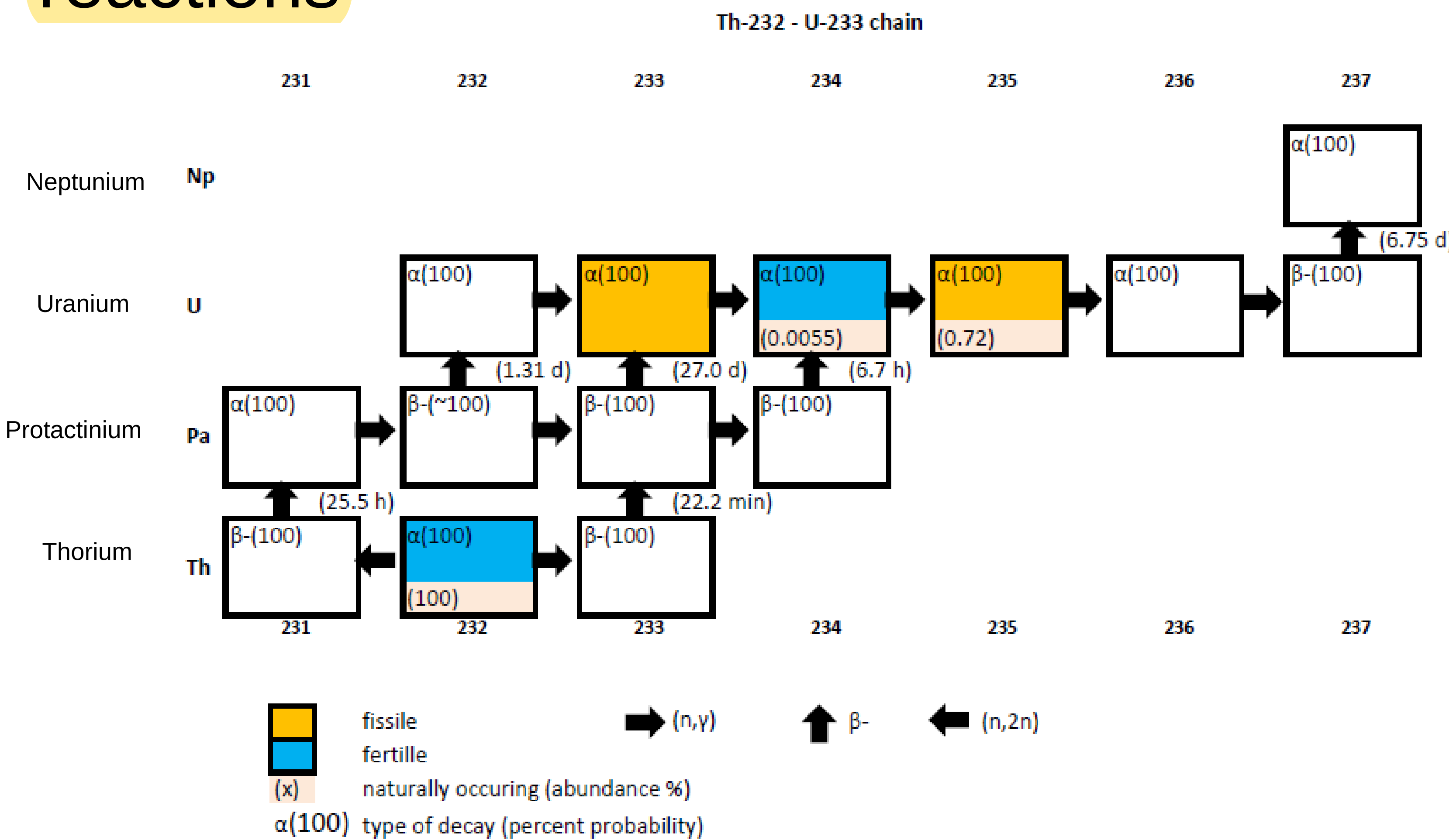
Conversion Chain U-238→Pu-239

- The nonfissile *uranium-238* can be *converted* to fissile *plutonium-239* by the following nuclear reactions



Conversion Chain Th-232→U-233

- The nonfissile *thorium-232* can be *converted* to fissile *uranium-233* by the following nuclear reactions



Conversion Ratio

- The degree of conversion that occurs in a reactor is denoted as conversion ratio $\overline{CR}$, defined as:

$$\overline{CR} = \frac{\text{Fissile material produced}}{\text{Fissile material destroyed}} = \frac{FP}{FD}$$

$$\overline{CR} = \frac{\int_0^{T_f} dt \int_{V_c} RR_c^{(FP)}(\vec{r},t)dv}{\int_0^{T_f} dt \int_{V_c} RR_a^{(FD)}(\vec{r},t)dv}$$

- where
  - $RR_c^{(FP)}$ capture reaction rate (produces isotopes)
  - $RR_a^{(FD)}$ absorption reaction rate (causes fission)
  - T_f fuel cycle length
  - V_c core volume

- Reality is a bit more complicated due to the loses
  - But those can be considered as minor and for our purposes this approximation is sufficient.

Conversion Ratio

- The conversion ratio is applicable to thermal reactors with natural or slightly enriched uranium.
- In a thermal reactor ^{239}Pu is produced due to capture of thermal and resonance neutrons by ^{238}U.
  - Let's make an estimate of the reaction rates.

- The rate of production of fission neutrons is

$$R_{FNP} = \varphi N \sigma_a \eta \varepsilon$$

where

φ – thermal neutron flux.

N – concentration of fissile nuclei.

σ_a – thermal-neutron absorption cross section.

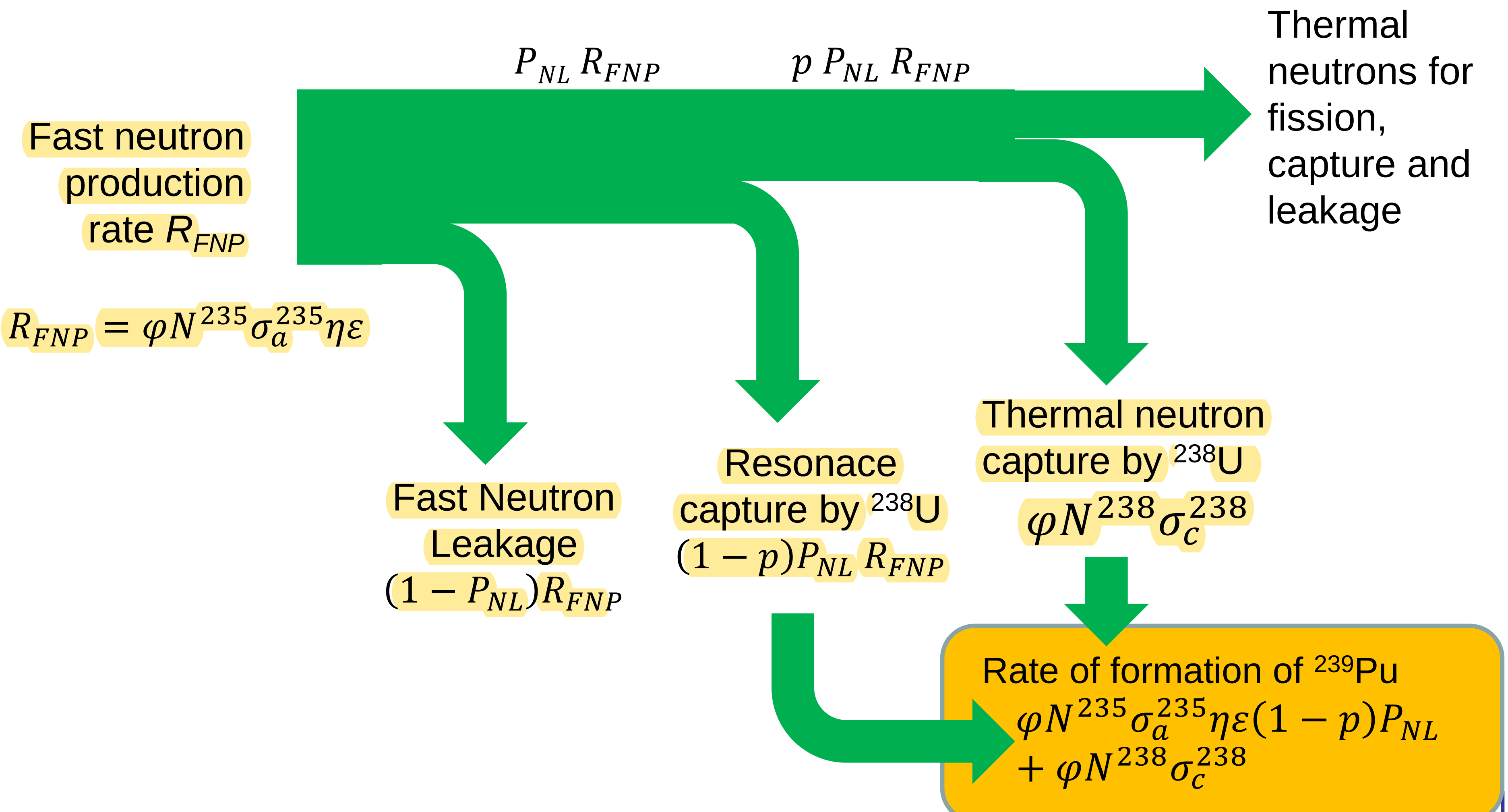
η – number of fast neutrons produced per absorbed neutron.

ε – fast-fission factor

- ratio of the fast neutrons produced by fissions at all energies to the number of fast neutrons produced in thermal fission.

Conversion Ratio

- This represents the initial formation of ^{239}Pu, when the only fissile material is ^{235}U
- There are two ways for plutonium formation:
  - (1) resonance capture
  - (2) thermal neutron capture in U-238
- P_{NL} - probability of non-leakage in slowing down of fast neutrons into the resonance region
- p is the resonance escape probability (thus $(1-p)$ is the fraction of the neutrons in the resonance region that is captured by ^{238}U to form ^{239}Pu.



Conversion Ratio

- Thus, the initial (when there is no ^{239}Pu and ^{235}U is the only fissile material) conversion ratio is

$$CR(initial) = \frac{\varphi N^{235}\sigma_a^{235}\eta\varepsilon(1-p)P_{NL} + \varphi N^{238}\sigma_c^{238}}{\varphi N^{235}\sigma_a^{235}} =$$

$$= \frac{N^{238}\sigma_c^{238}}{N^{235}\sigma_a^{235}} + \eta\varepsilon(1-p)P_{NL}$$

- After some operation period fissions and captures occur in ^{239}Pu and ^{235}U;
  - this tends to decrease CR
    - (^{239}Pu has larger cross-section for capture of thermal neutrons)
- Also fission products and heavy nuclides capture the resonance neutrons, decreasing CR as well.
- A large initial conversion ratio is desirable, since it extends fuel burnup.
  - In commercial water-cooled reactors its value is approximately 0.6.
  - However, only slightly above 50% of the generated plutonium-239 is fissioned, one-sixth is lost by neutron capture and the rest remains in the spent fuel.

Conversion Ratio (Example)

- EXAMPLE: In a critical reactor with natural uranium as fuel
  - for each 1000 neutrons absorbed in ^{235}U,
  - 250 neutrons are absorbed in resonances of ^{238}U and
  - 645 neutrons are absorbed by ^{238}U at thermal energies.

- Assume no neutron leakage from the reactor.
- Calculate the conversion ratio in the reactor.

Conversion Ratio (Example)

- SOLUTION: Each absorption of a neutron (irrespective if thermal or resonance) produces an atom of ^{239}Pu via the following reaction:

$$^{238}\mathrm{U} \xrightarrow{(n,\gamma)} {}^{239}\mathrm{U} \xrightarrow{\beta^-(23.5\mathrm{m})} {}^{239}\mathrm{Np} \xrightarrow{\beta^-(2.35\mathrm{d})} {}^{239}\mathrm{Pu}$$

- So fissile material produced (FP) is
  250 (^{238}U) + 645 (^{238}U) = 895 ^{239}Pu.

- Since the fissile material (^{235}U) destroyed (FD) is 1000, from the definition, the conversion ratio is:

- CR = FP/FD = 895/1000 = 0.895

Breeding Ratio

- If conversion ratio is **greater than 1**, it is called a **breeding ratio** BR

- A reactor with the conversion ratio CR less than 1 is called a **converter**
- A reactor with the conversion ratio CR greater than 1 is called a **breeder**

- Present LWRs are converters

- Fast reactors have potential to be breeders

Breeding Ratio

- Breeding ratio has the same definition as the conversion ratio

$$BR(t) = \frac{\text{Fissile material produced}}{\text{Fissile material destroyed}} = \frac{FP}{FD}$$

- However, breeding can occur in core and in the blanket, thus, at any time t, we have

$$BR(t) = \frac{\int_{core} \varphi(t)\Sigma_c^{fertile}(t)dV + \int_{blanket} \varphi(t)\Sigma_c^{fertile}(t)dV}{\int_{core} \varphi(t)\left(\Sigma_f(t) + \Sigma_c(t)\right)^{fissile} dV}$$

- $\Sigma_f(t), \Sigma_c(t)$ the macroscopic fission and capture cross sections, respectively

Breeding Ratio

- It can be assumed that all the neutrons leaking from the core lead to breeding.

- This is a reasonable assumption, since some fissions occur in the blanket compensating for the leakage

- We divide BR into two parts:
  - external (blanket)
  - internal (core)

$$Ext.\,breeding\;ratio = \frac{\text{Core leakage(= breeding in blanket)}}{\text{Destruction of fissile mat. in core}}$$

Breeding Ratio

- The core leakage can be found from the following neutron balance in the core:

Core leakage = neutrons produced in core (=$\int_{core} \varphi \nu \Sigma_f^{core} dV$)

- loss of neutrons for fission (=$\int_{core} \varphi \Sigma_f^{core} dV$)

- loss of neutrons by capture (=$\int_{core} \varphi \Sigma_c^{core} dV$)

- $$\text{Core leakage} = \int_{core} \varphi\left(\nu\Sigma_f - \Sigma_f - \Sigma_c\right)^{core} dV$$

- here
  ν - the number of neutrons produced per fission (yield)
  $\Sigma_f,\ \Sigma_c$ - include contributions of all materials in the core to the cross section.

Breeding Ratio

- Omitting time t in notation and neglecting any spatial variations in the core leakage and the destruction rate, we get

$$BR_{external} \approx \frac{\left(\nu\Sigma_f - \Sigma_f - \Sigma_c\right)^{core}}{\left(\Sigma_f + \Sigma_c\right)^{fissile}} = \frac{\Sigma_f^{core}}{\Sigma_f^{fissile}} \cdot \frac{\nu - (1 + \alpha^*)}{1 + \alpha}$$

Where

$\alpha^* = \Sigma_c/\Sigma_f$ - for the core

$\alpha = \Sigma_c/\Sigma_f$ - for the fissile species

if $\alpha^* = \alpha$

$$BR_{external} \approx \frac{\Sigma_f^{core}}{\Sigma_f^{fissile}}(\eta - 1)$$

$$\eta = \nu/(1 + \alpha)$$

- η – **reproduction factor** is defined as the ratio of
  - the number of fast neutrons produced by thermal fission to
  - the number of thermal neutrons absorbed in fissile nuclei.

Breeding Ratio

- The **internal breeding ratio** can be found as:

$$BR_{internal} = \frac{\int_{core} \varphi \Sigma_c^{fertile} dV}{\int_{core} \varphi \left(\Sigma_f + \Sigma_c\right)^{fissile} dV} \approx \frac{\Sigma_c^{fertile}}{\Sigma_f^{fissile}} \cdot \frac{1}{1+\alpha} = \frac{\Sigma_c^{fertile}}{\Sigma_f^{fissile}} \cdot \frac{\eta}{\nu}$$

Typical cross section and fission neutron data

	Thermal neutrons						Fast neutrons	
Material	σ_f	σ_c	σ_a	α	ν	η	ν	η
^{233}U	531	48.0	579	0.090	2.49	2.29	2.58	2.40
^{235}U	582	99.0	681	0.170	2.42	2.07	2.51	2.35
^{239}Pu	747	269	1016	0.360	2.93	2.15	3.04	2.90
^{238}U	-	2.70	2.70	-				
U-nat	4.20	3.40	7.60	0.81				

Breeding Ratio

- The sodium cooled fast reactors, with $^{239}PuO_2$ and $^{238}UO_2$ are expected to have the total (external+internal) breeding ratio of about 1.2.

- With carbide (UC and PuC) fuel and blanket material the breeding ratio should be even larger due to higher heavy-metal atom densities than oxides.

Breeding Gain

- In a breeder reactor it is possible that the conversion ratio for the core only, is less than 1, while the breeding ratio for the entire reactor, core and blanket, is greater than 1

- Breeding gain is defined as:

$$G = BR - 1$$

- This is equivalent to:

$$G = \frac{F_{EOC} - F_{BOC}}{FD} = \frac{FG}{FD} = \frac{\text{Fissile material gained}}{\text{Fissile material destroyed}}$$

- here: F_{EOC}, F_{BOC} – fuel inventory at End of Cycle and Beginning of Cycle, respectively.

Breeder Reactor

- It is possible for a nuclear reactor to breed over a wide neutron energy spectrum.
- Adequate breeding ratios can be achieved for a given energy spectrum
  - only by carefully selecting the appropriate fissile isotopes for that spectrum.
- It can be shown that a **high breeding gain** can be obtained only with a **fast neutron spectrum**.
- **Thus fast-spectrum reactors can serve as breeder reactors**.

Breeding Potential

- The highest possible breeding ratio (breeding potential) is

$$\overline{BR}_{\max} = \bar{\eta} - 1$$

- where

$$\bar{\eta} = \frac{\bar{\nu}_f \bar{\sigma}_f}{\bar{\sigma}_a} = \frac{\bar{\nu}_f}{1 + \frac{\bar{\sigma}_c}{\bar{\sigma}_f}} = \frac{\bar{\nu}_f}{1 + \bar{\alpha}}$$

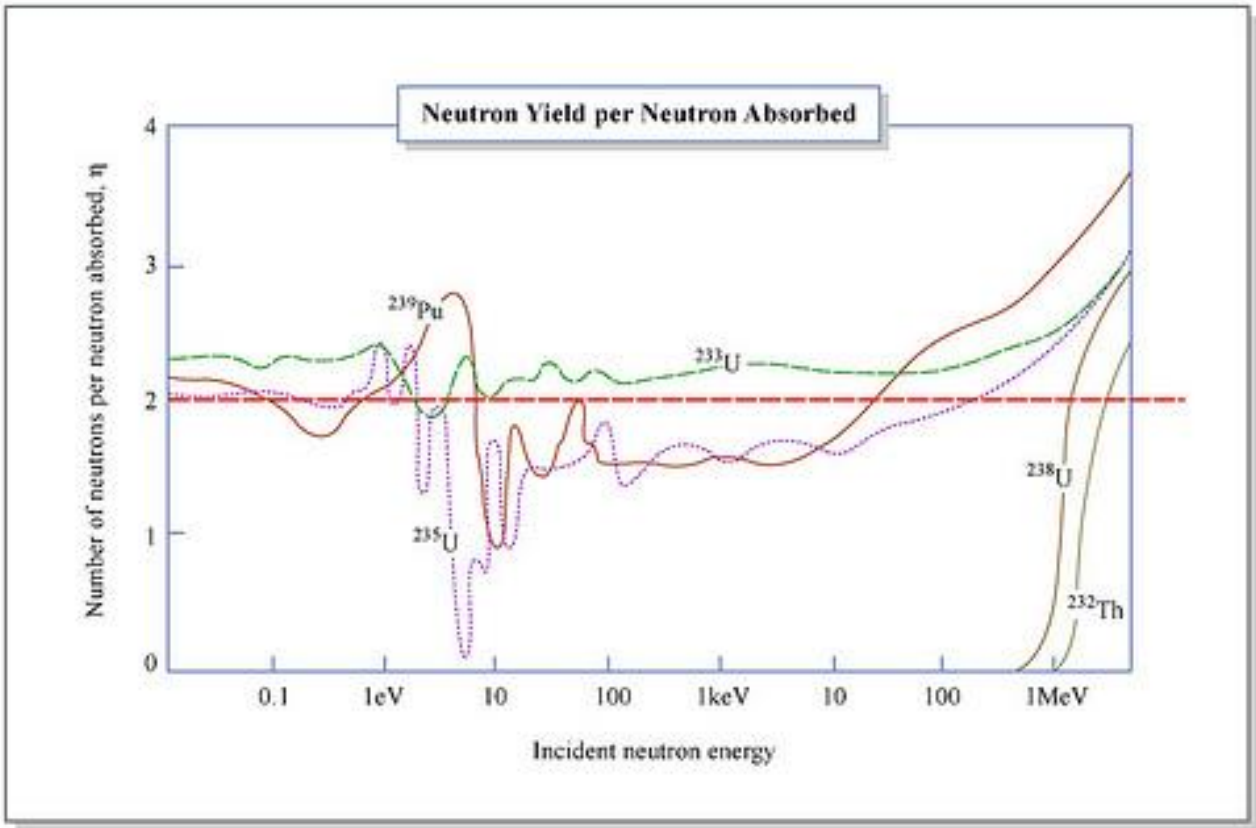
$$\bar{\sigma}_a = \bar{\sigma}_c + \bar{\sigma}_f$$

absorption = capture + fission

  - $\bar{\eta}$ Number of neutrons produced per neutron absorbed
  - $\bar{\nu}_f$ Number of neutrons per fission
  - $\bar{\alpha}$ Capture-to-fission ratio

Neutron Yield per Absorption

- For breeding to be possible, the number of neutrons produced per neutron absorbed has to be greater than 2.0
  - ^{235}U is not suitable
  - ^{233}U is possible but very small margin
    - Similarly good for LWR and FSR



Minimum η for Breeder

- Balance of neutrons for breeding:
  - one neutron must be absorbed in fissile material to continue the chain reaction
  - L neutrons are lost unproductively (absorption + leakage)
  - $\bar{\eta}$ – (1 + L) neutrons are captured in fertile material

- Since for breeding at least one neutron must be captured in the fertile material, we have

$$\bar{\eta} \geq 2 + L$$

Averaged Values of η

$\bar{\eta}$ averaging spectrum type	Pu-239	U-235	U-233
Averaged over LWR spectrum	2.04	2.06	2.26
Averaged over oxide-fueled FSR	2.45	2.10	2.31

Doubling Time

- The Reactor Doubling Time (RDT) can be expressed as:

$$RDT = M_0/Mg$$

- M_0 – initial fissile inventory in a reactor (kg)
- Mg – the fissile material gained during a year (kg/y)

- Accurate computation of M_g is complex and requires computer codes
- Doubling time has significant economic meaning
  – How long will it take to clone our reactor?

Doubling Time

- It is instructive to consider an approximate calculation of Mg – the fission material gained during a year:

- It can be expressed in terms of
  - breeding gain, G,
  - rated power P,
  - fraction of time at rated power f
  - and $\bar{\alpha}$ - capture-to-fission ratio.

- First, we express Mg as:

$$Mg = G * (\text{fissile mass destroyed/y}) \cong$$
$$\cong G(1+\bar{\alpha})(\text{fissile mass fissioned/y})$$

- Next we expressed "fissile mass fissioned" in terms of P

Doubling Time

- "fissile mass fissioned/y" = (P[MWth]x10^6) x power in W
  (2.93 x 10^{10} fissions/W·s) x number of fissions per J
  (3.15 x 10^7s/y) x number of seconds in a year
  (f) x fraction of time at rated power
  (239 kg/kmol) / molar mass of fuel M
  (6.02 x 10^{26} atoms/kmol) = Avogadro's number N_A
  = P x f / 2.73

- M/N_A = mass of one fuel (^{239}Pu) atom (kg/atom)

- Thus:

$$M_g = \frac{G \cdot P \cdot f \cdot (1 + \bar{\alpha})}{2.73}$$

Doubling Time

- Finally, the doubling time is found as:

$$RDT \cong \frac{2.73M_0}{G \cdot P \cdot f \cdot (1 + \bar{\alpha})}$$

- Here, P is in MWth, M_0 in kg and RDT in years

- We see that the RDT is proportional to
  – the fissile specific inventory, M_0/P,
- and inversely proportional to
  – breeding gain G
- for FSR with oxide fuel M_0/P ~= 1 ÷ 2 kg/MWth

Doubling Time (Exercise)

- **EXERCISE:** After a core re-design, the doubling time of a breeder reactor was reduced by factor 2.
  - Q1) What was BR after re-design, if before re-design, the breeding ratio was equal to 1.2? Assume that M_0/P, f and capture-to-fission ratio are the same in both cases.
  - Q2) What was RDT before and after re-design of the reactor? Assume that $M_0/P=1.5$ *kg/MWth*, $f = 0.75$, $\sigma_c = 0.11$ *b and* $\sigma_f = 1.6$ *b*.
  - Q3) Calculate M_g and M_0 before and after re-design if reactor power in both cases was the same and equal to $P = 3$ GWth

$$RDT \cong \frac{2.73 M_0}{G \cdot P \cdot f \cdot (1 + \bar{\alpha})}$$

$$M_g = \frac{G \cdot P \cdot f \cdot (1 + \bar{\alpha})}{2.73}$$

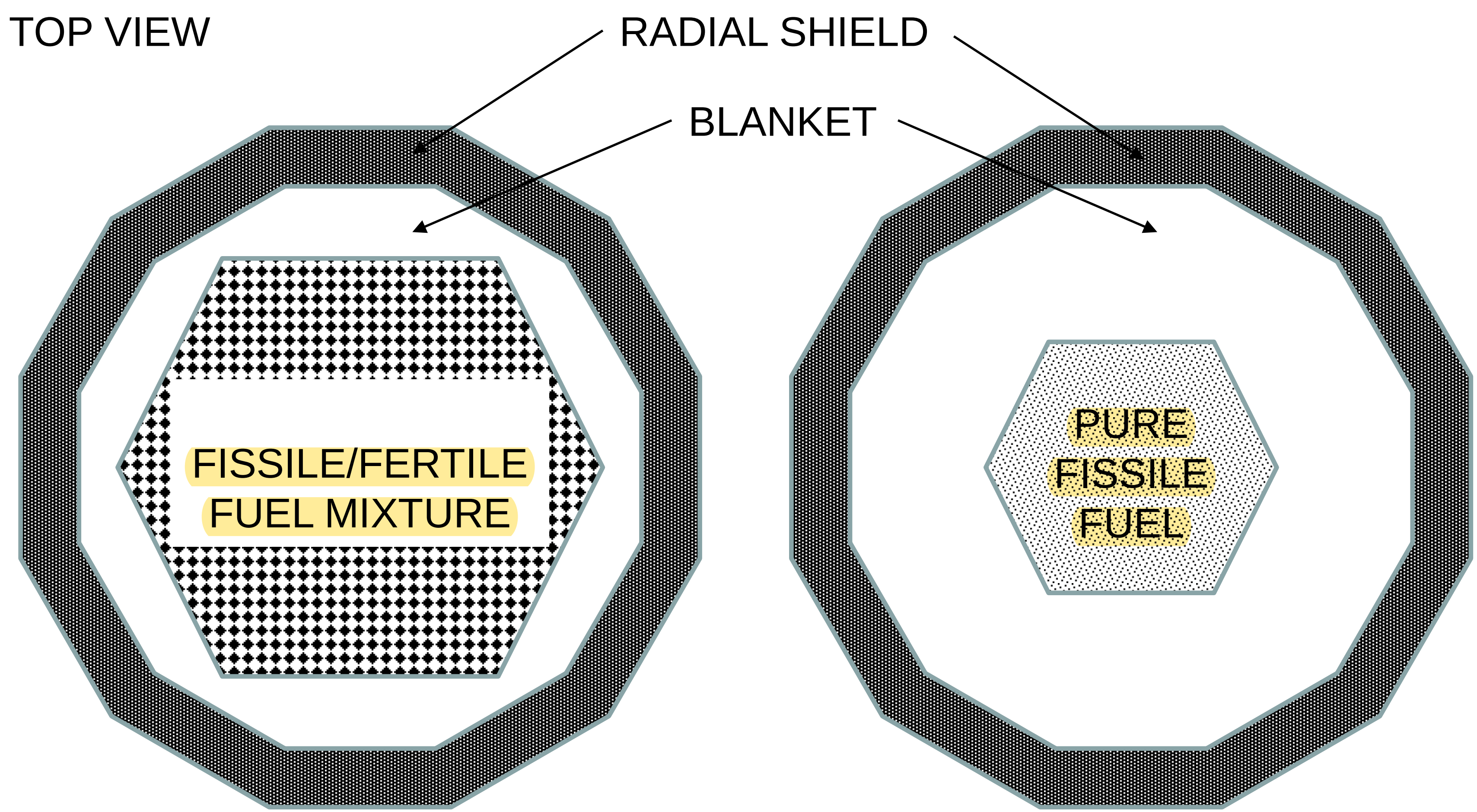
Design Principles of FBR

- Main design objectives of FBR are
  - Safe operation
  - High breeding ratio
  - Low doubling time
  - Low cost

- To achieve these objectives, certain design approaches must be adopted.
- Main concerns are:
  - core and blanket arrangements
  - fuel lattice and fuel assembly
  - vessel internals and pool versus loop design
  - materials selection for fuel, coolants and construction

Core and Blanket Arrangements

- Typical arrangements of core and blanket

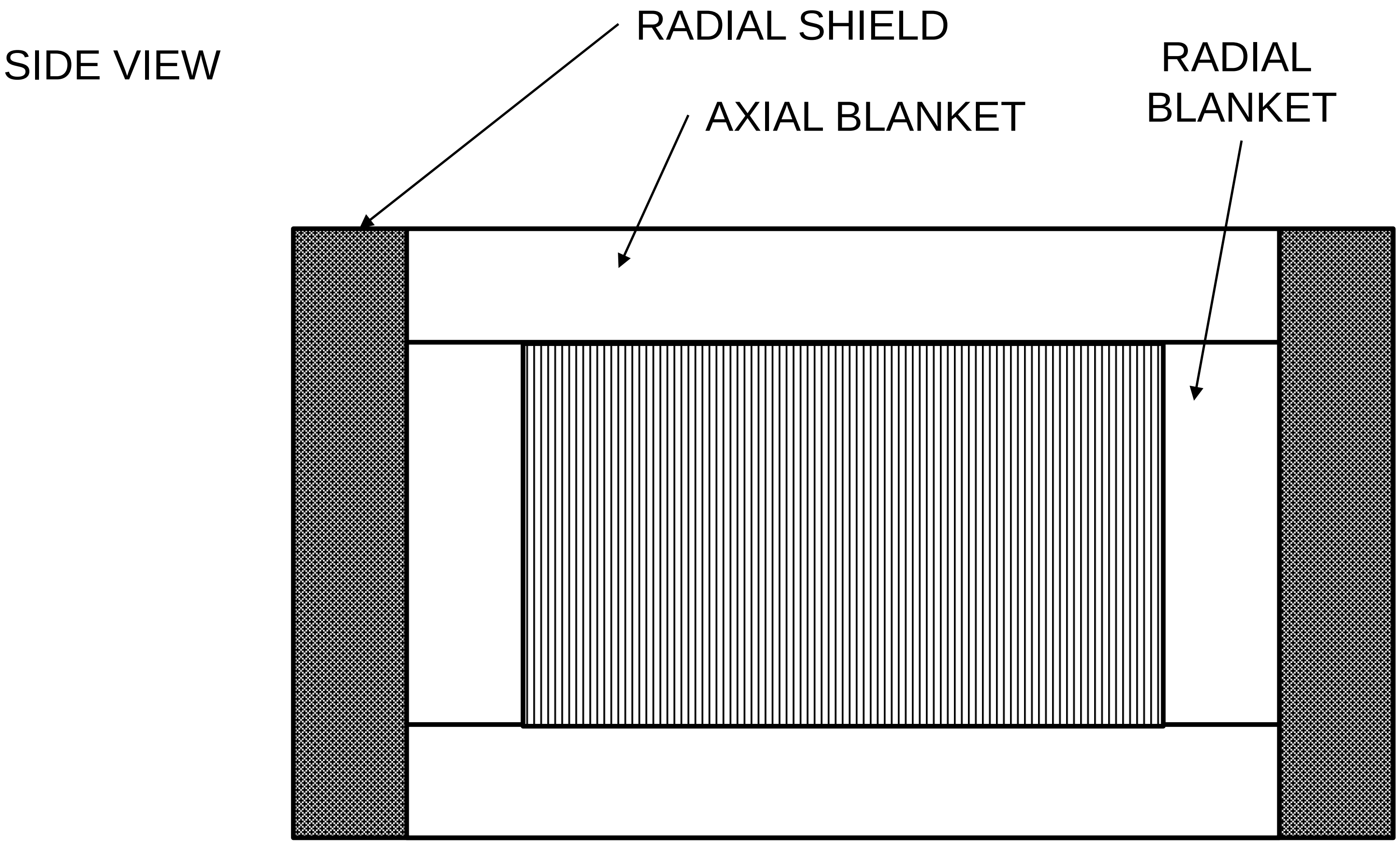


In-core breeding concept

External breeding concept

Core and Blanket Arrangements

- Typical core arrangement

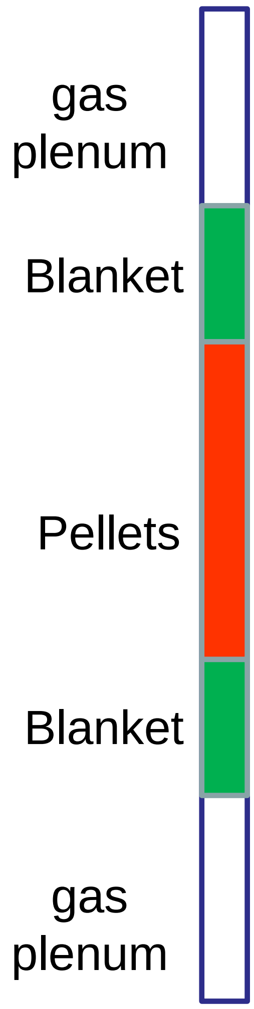


Fuel Lattice and Fuel Assembly

- In order to minimize the specific fuel inventory in the breeder reactor, a “tight” fuel lattice that maximizes fuel volume fraction is desired.
- A triangular lattice arrangement intrinsically allows a higher fuel volume fraction.
- Thus triangular lattice is usually selected (except Fermi).

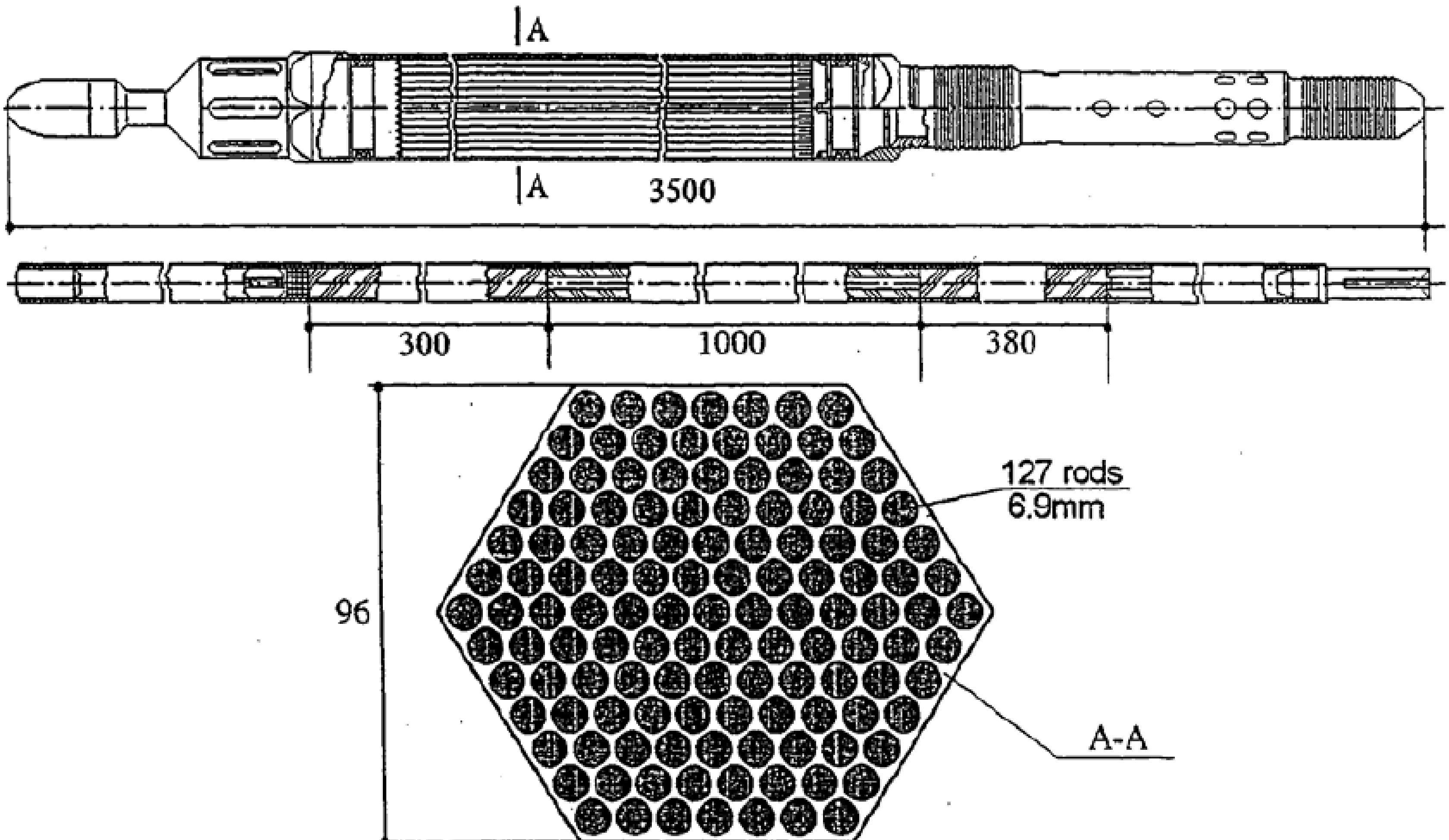
Fuel Lattice and Fuel Assembly

- In tight lattice the rods are separated by wire wrap.
- Grid spacers are used for less tight lattice.
- Fuel pellets make up the active core region and the blanket pellets provide the axial boundaries.
- The fission gas plenum is located above the upper blanket region or below the lower blanket region.



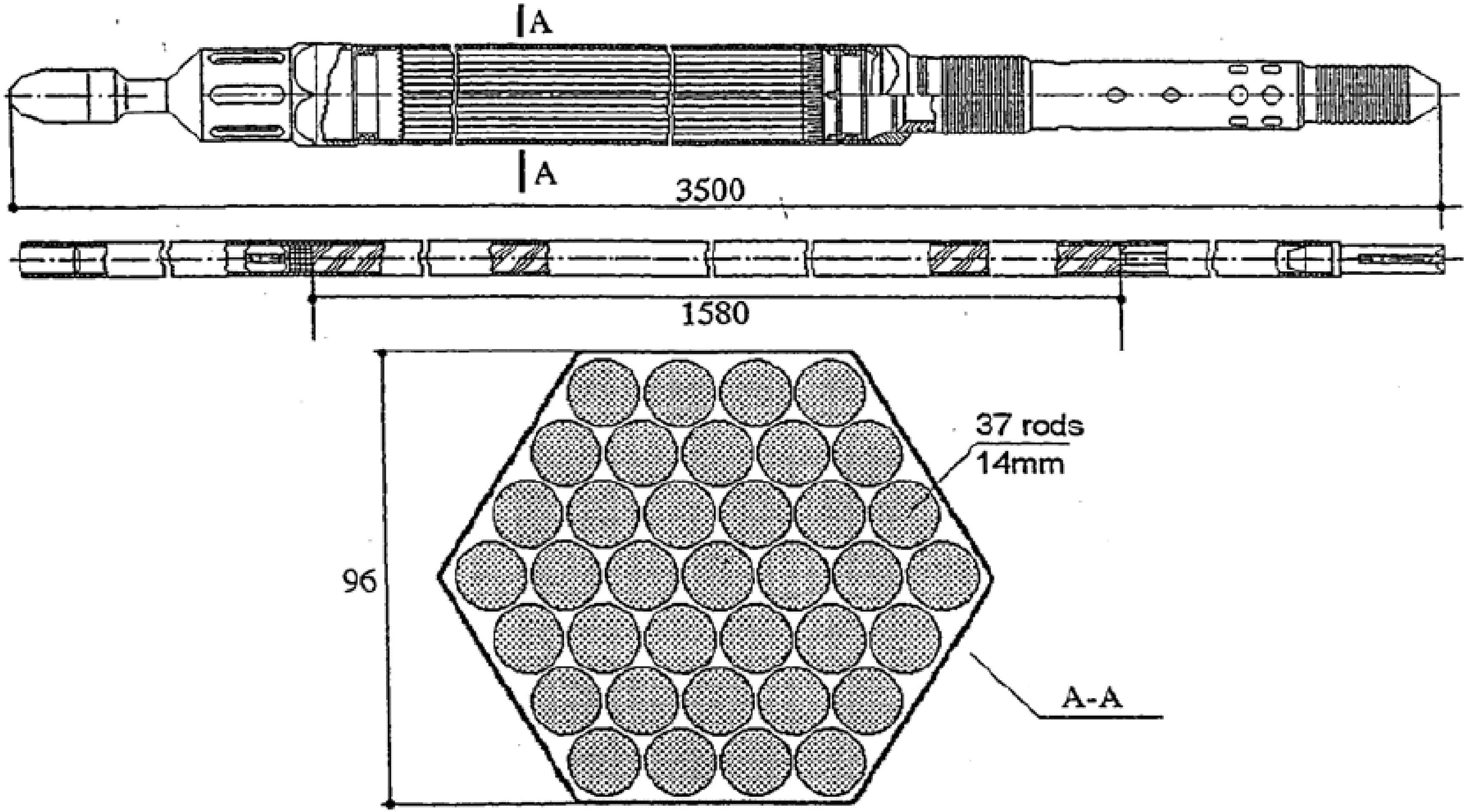
Fuel Lattice and Fuel Assembly

- Example fuel assembly in BN-350
  - Fuel rods in the core region are arranged in triangular lattice



Fuel Lattice and Fuel Assembly

- Example assembly in BN-350 blanket region

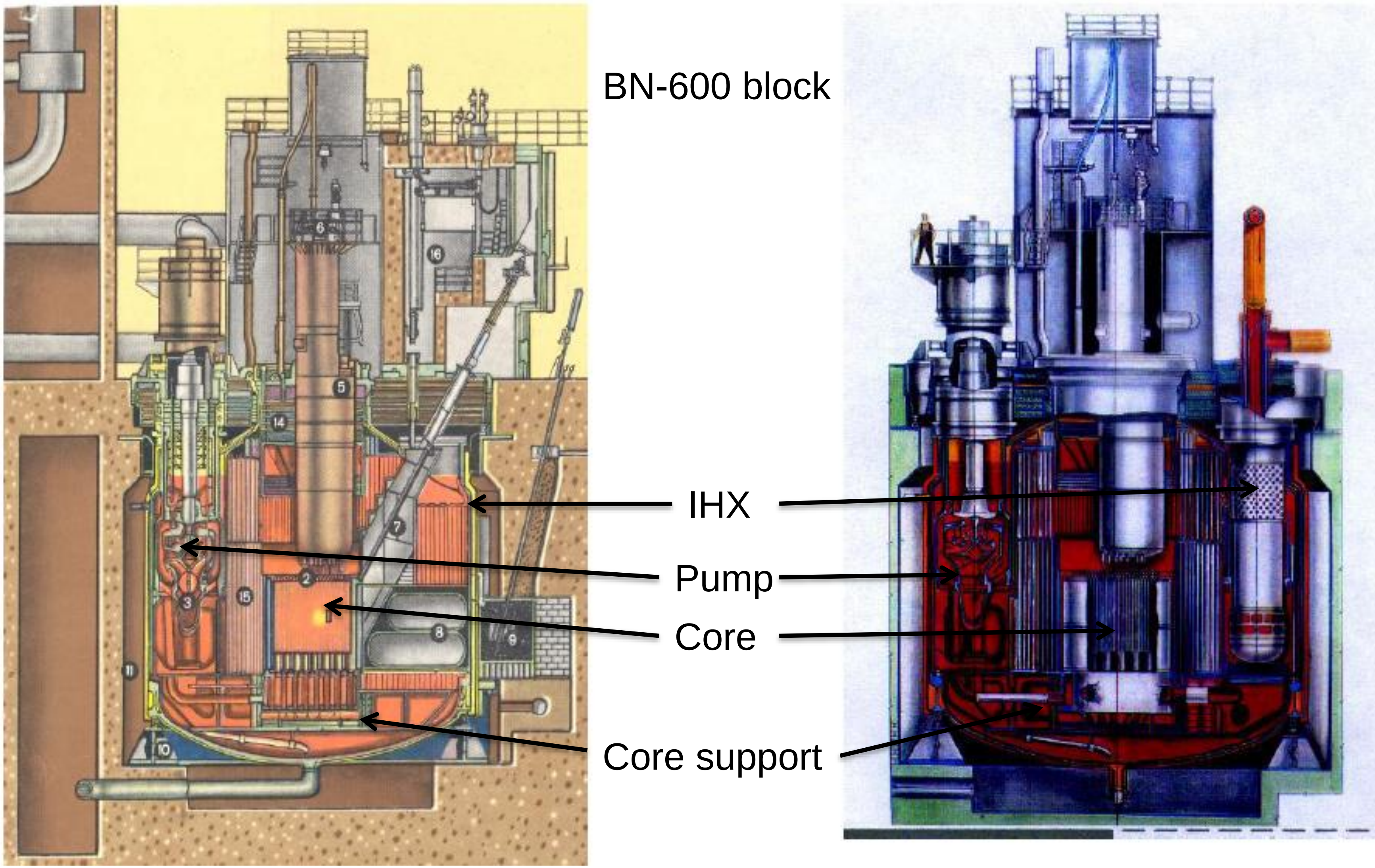


Vessel Internals

- The core in SFR is located on top of core support structure.
- Because SFR does not require pressurization the vessel walls need only be thick enough to satisfy standard weigh loads (typical thickness can be 30 mm = 10% of LWR).
- Control rods enter from the top of the core.

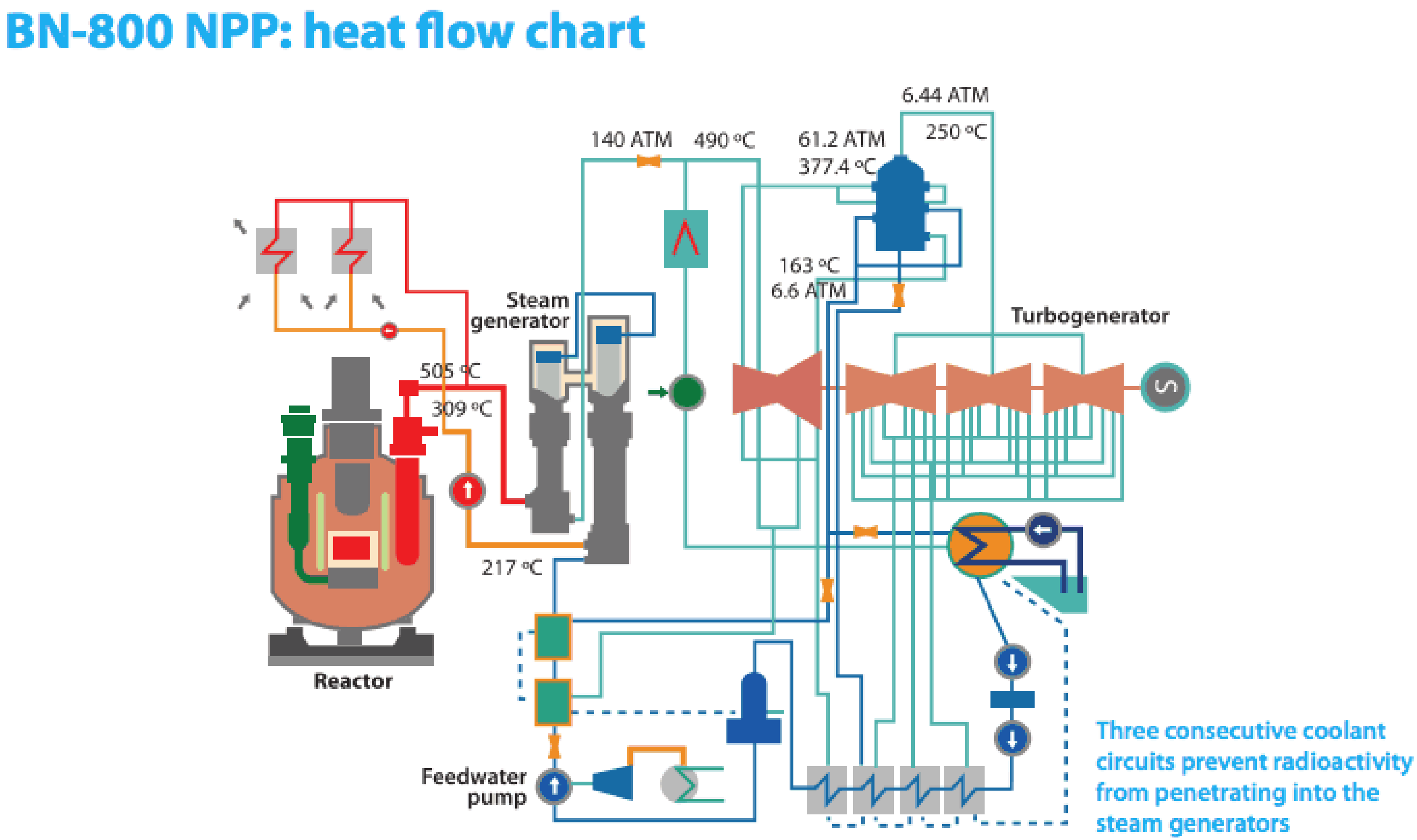
Vessel Internals

- BN-600



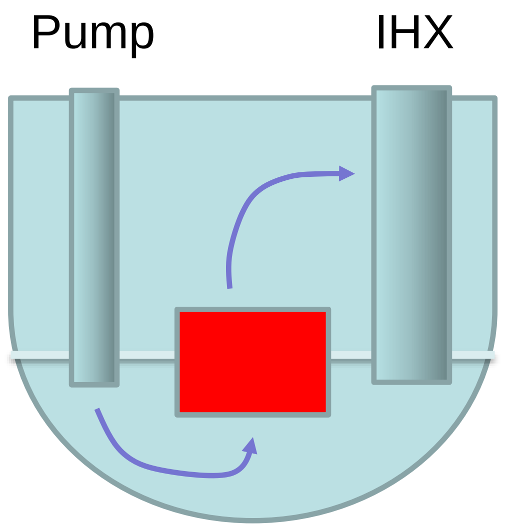
Vessel Internals

- BN-600

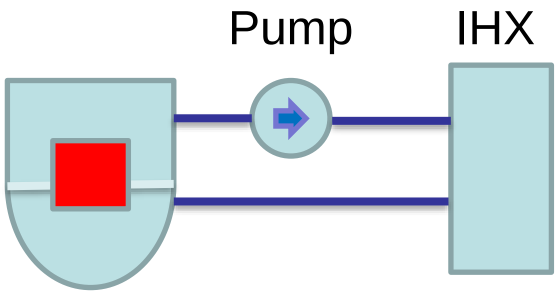


Pool vs Loop Design

- Two dominating approaches to the design:



- pool type
  - primary pumps and intermediate heat exchanger (IHX) are located in a pool



- loop type
  - primary pumps and IHXs are located in separate cells outside of the reactor vessel

Pool vs Loop Design

- The term loop applies to a sequential series of connected components in the heat transport system between reactor and the turbine.

- Each loop contains a pump, heat source and heat sink.

Pool vs Loop Design

- The only test reactor to use the pool design was EBR-II.
- The rest (FFTF, BOR-60, Rapsodie, JOYO, KNK-II, Fermi, Dounreay, FBTR and PEC) employed the loop system.
- Demonstration or prototype pool reactors
  - Phénix, PFR, BN-600, PFBR, Super-Phénix, ALMR.
- Demonstration or prototype loop reactors
  - BN-350, SNR-300, MONJU, CRBRP.

Pool vs Loop Design

- Pool systems have the following advantages:
  - leakage in the primary system components does not result in the leakage from the primary system.
  - large mass of sodium in the primary system (~3 x that of loop system) providing large heat capacity, resulting in lower temperature rise in off-normal transients.
  - large thermal inertia that dampen transient thermal effects.

Pool vs Loop Design

- Loop systems have the following advantages:
  - Maintenance is simple since components are isolated in cells.
  - Great flexibility in making system modifications.
  - Less shielding required to protect secondary system.
  - Structure of the vessel head is simpler.
  - Greater difference in IHX level against the core enhances natural circulation (safety aspects).
  - Greater potential to optimize system components (economic gain).

Selection of Materials

- Fuel
  - – high burnup.
  - – high linear power.
- Coolant
  - – low neutron moderation.
  - – high heat removal efficiency.
  - – low parasitic neutron absorption.
- Structure
  - – resistance to high temperature.
  - – resistance to high fluence.
  - – compatibility with coolant.
  - – low parasitic neutron absorption.

Selection of Fuel - Oxide

- Oxide fuels (mixture of UO_2 and PuO_2)
  - – high burnup, perhaps in excess of 100 MWd/kg (positive).
  - – high linear power thanks to high melting temperature (positive).
  - – low thermal conductivity (negative).
  - – softer neutron spectrum (some moderation by oxygen atoms).
    - strong negative Doppler coefficient (positive)
    - lower breeding ratio (negative)

Selection of Fuel – Metal, C & N

- Metal (U-Pu), carbide (UC-PuC), nitride (UN-PuN)
  - – higher thermal conductivity (positive).
  - – harder neutron spectrum (no or a single light atom per fuel atom)
    - higher breeding ratio (positive).
    - difficulty in achieving high burnup (negative).
    - small Doppler coefficient (negative for safety reasons).

Selection of Fuel

- Required high enrichment in fast reactors due to low fission cross section, as compared with LWRs

- Pu fraction usually in range 12 – 30%, where 75% of fissile Pu (that is Pu-239 and Pu-241)
  - thus fissile fractions are in range 9 – 23 %

Selection of Fuel

- For **ASTRID** reactor (CEA in France), the reference fuel in core is $(U,Pu)O_2$ (mixed oxide)

- This is mainly thanks to significant experience in France with this type of fuel.

- Very good performance both in Phenix and Superphenix.

Fuel-Pin Diameter

- In LWRs fuel-pin diameter (that is clad outer diameter) is governed by critical heat flux (CHF) limitations
  - when the linear power is selected to not allow fuel melting at the center, pin diameter is selected to not allow CHF

- In FSR there is no CHF limitation and pin diameter is not governed by heat transfer considerations

- Pin diameter in FSR does however have significant influence on economics
  - influences the breeding ratio and the specific fuel inventory
  - typical values in FSR: 6 – 9 mm (2/3 of LWRs pin diameter)

Fuel-Pin Diameter

- In **ASTRID reactor**, the fuel pin diameter is between 9 and 10 mm (8.5 mm in Superphenix).

- The fuel pins separated with a helical wire of 1 mm in diameter.

Fuel Burnup

- Burnup for an economical breeder is considerably more demanding than for a LWR
  - 45 – 55 MWd/kg is normally acceptable for LWR
  - 100 MWd/kg or higher is required for economic operation of FSR
  - 144 MWd/kg was achieved in Phenix with an experimental sub-assembly

- This high burnup requirement is a reason for selecting oxide fuel in present FSRs.

Coolant

- Principal coolant candidates
  - liquid metals
    - sodium,
    - sodium/potassium eutectic,
    - mercury,
    - Lead,
    - lead/bismuth eutectic
  - helium
  - steam

- The ASTRID design has sodium coolant and intermediate sodium loop.

Structure Materials

- Principal structure material candidates
  - stainless steel (zircaloy is not selected due to low resistance at high temperatures)
  - austenitic stainless steel is used for main vessel, inner vessel, safety vessel, core support structure, grid plate, etc.

- In the ASTRID design the target clad material is a ferritic or martensitic oxide dispersion strengthened (OSD) steel.
- However, at present OSD steels are not tested enough and qualified as cladding material.
- Thus the first cores in ASTRID were planning to use 15-15 Ti work-hardened austenitic steel AIM1.