



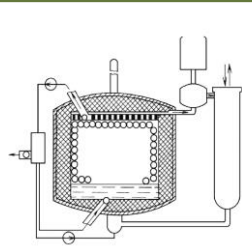
Fuel cycle potential of MSR

J. Krepel, MSR pSSC

15 April 2026

Example of MSR designs

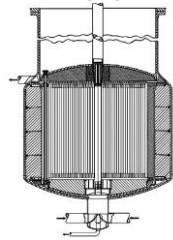
Fluoride salt cooled reactors



SINAP TMSR-LF1

Graphite moderated

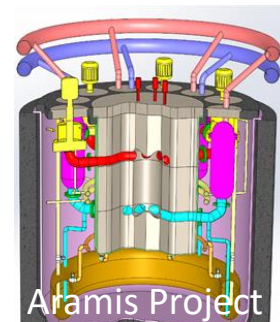
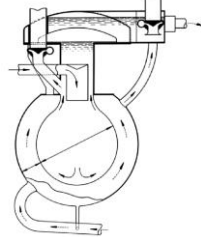
MSRs



MSFR

Homogeneous fluoride fast

MSRs



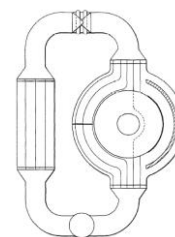
Aramis Project



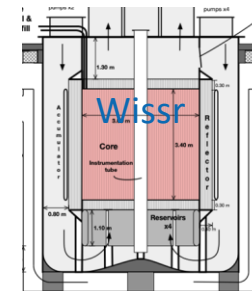
Curio

Homogeneous chloride fast

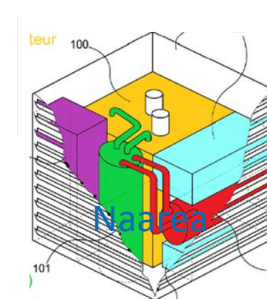
MSRs



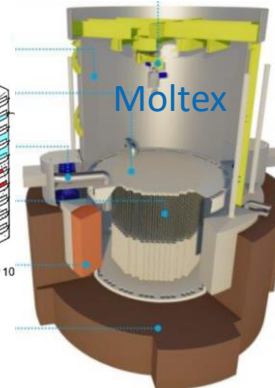
Thorizon



WISSR



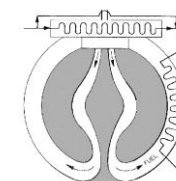
Naaree



Moltex

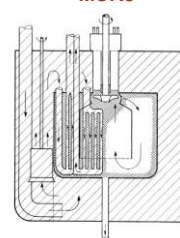
Non-graphite moderated

MSRs



Heterogeneous chloride fast

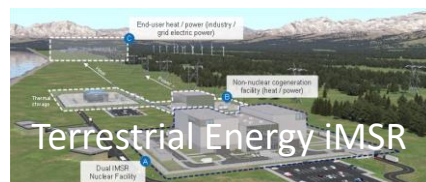
MSRs



Kairos Power



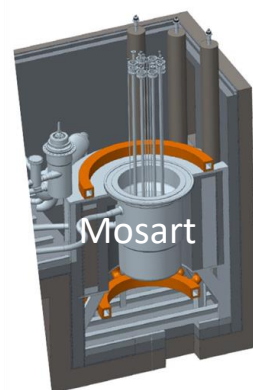
Natura Resources



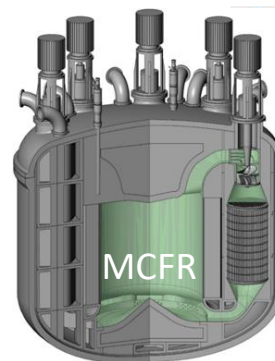
Terrestrial Energy iMSR



SALT-FOSS



Mosart



MCFR

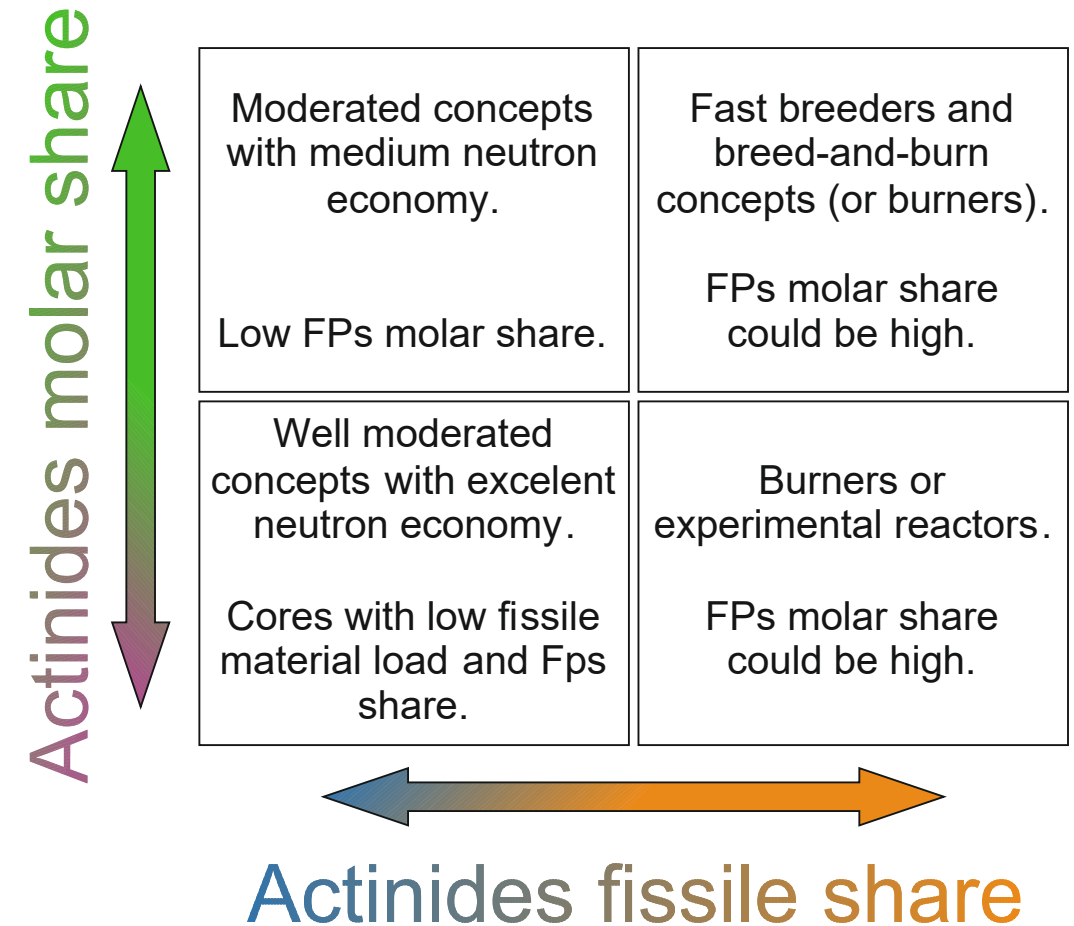
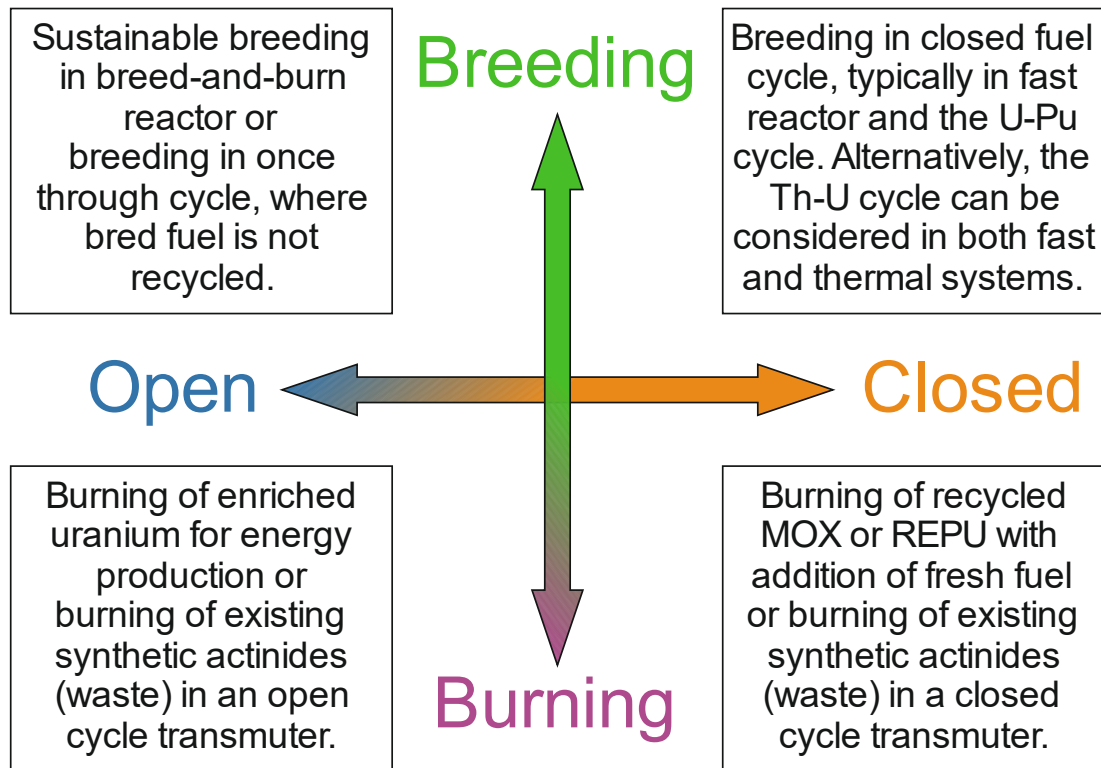


Stellaris



Copenhagen Atomics

Fuel cycle and salt types

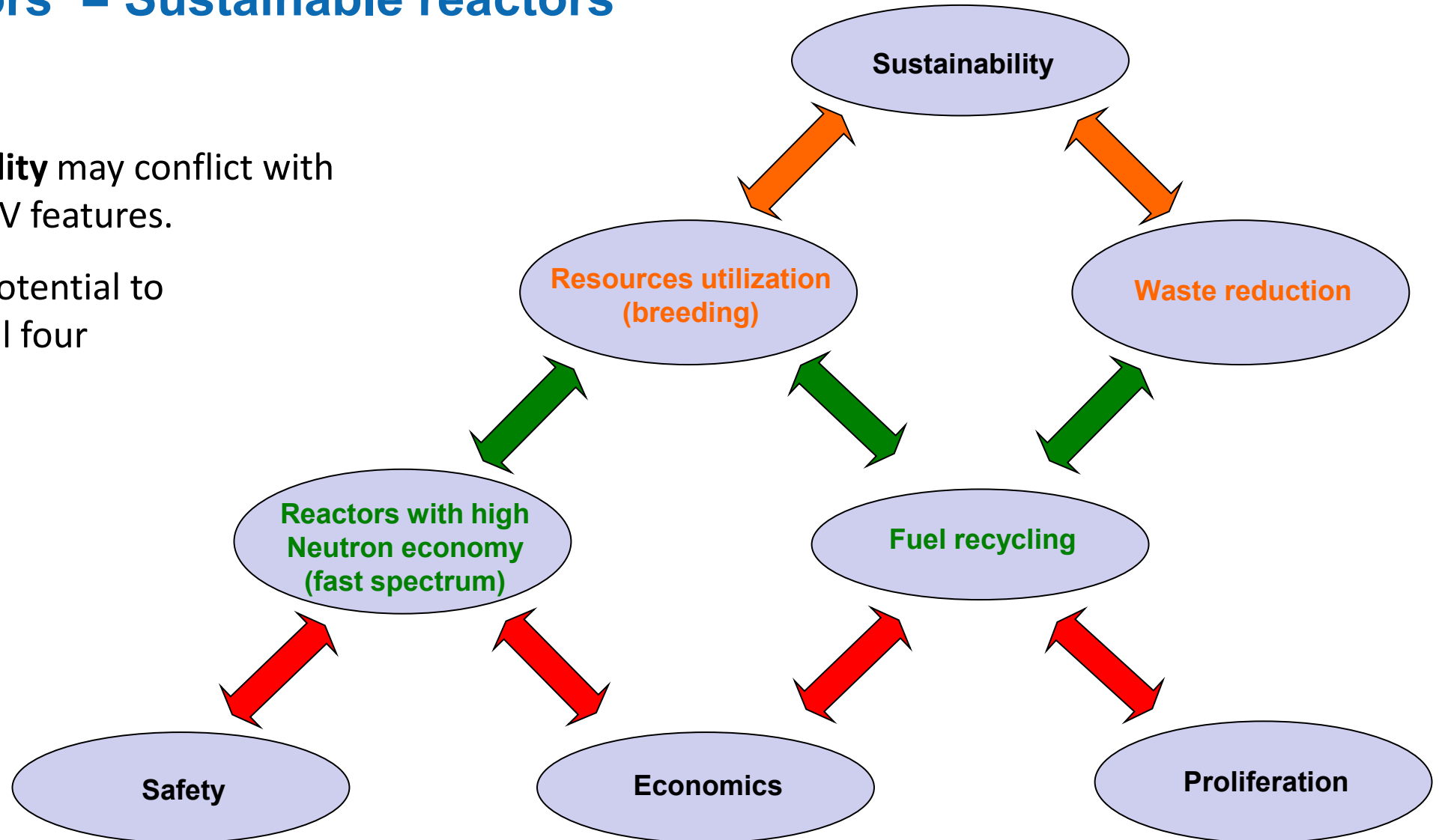


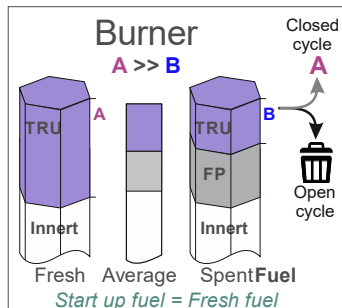
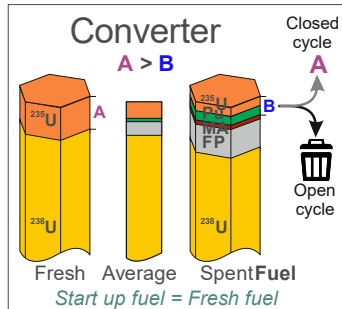
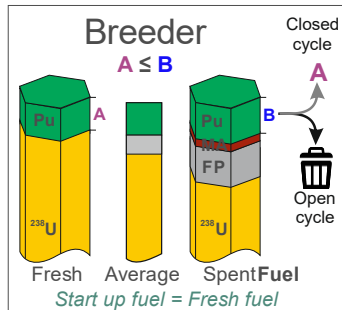
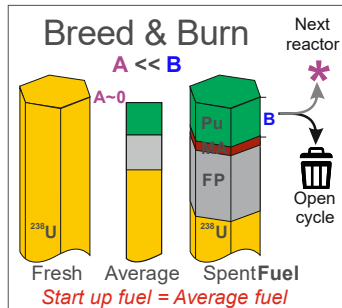
5 major fuel cycle types (reactor physics perspective)

- I. **Enriched uranium burning** (^{235}U burning) Resources utilization: <1% of nat. U
 - Enrichment level can range from 0.7% to 20%.
 - The cycle is generally open and “waste” intensive.
 - However, irradiated U and generated Pu can be recycled as MOX fuel.
- II. **Closed Th-U cycle** (^{232}Th burning in closed cycle) <95% of ^{232}Th
 - Actinides recycling in a breeder reactor fueled by ^{232}Th .
- III. **Closed U-Pu cycle** (^{238}U burning in closed cycle) <95% of nat. U
 - Actinides recycling in a breeder reactor fueled by ^{238}U .
- IV. **Breed-and-burn U-Pu cycle** (^{238}U burning in open cycle) cca 20-30% nat. U
 - Open cycle, Ac. are not recycled, but the reactor acts as a breeder.
 - “Waste” intensive cycle, however fuel can be reused by another reactor.
- V. **Synthetic actinides burning** not relevant
 - Cycle dedicated to minimization of existing synthetic actinides.
- **Combination or transition between above cycles**
 - e.g. actinides from I. or V. can acts as an initial or add on fuel for II. - IV. or vice versa.

GenIV reactors = Sustainable reactors

- **Sustainability** may conflict with other GenIV features.
- **MSR** has potential to combine all four features.





MSR fuel cycle choices and options

- The carrier salt (liquid fuel solvent) is an ionic liquid and can be irradiated without a limit.
- No matter the fuel cycle, it is always good to keep actinides as long as possible in the reactor (high burnup).
- At best only Fission Products (FPs) should leave the core.
- Unfortunately, FPs (Lanthanides) tends to leave the fuel salt as last.
- In any fuel cycle, core with higher actinides load could be designed as smaller.
- Actinides density collides with salt melting temperature and potentially also with solubility limits (solidus – liquidus temperature gap?).
- Concept capable to operate in Breed-and-burn cycle can be also designed as breeder, converter or burner.
- Concept designed as burner or converter are not necessarily capable of breeding or breed-and-burn operation mode.

Five fuel cycle performance parameters

I. Breeding capability

- How many neutrons can be captured by ^{232}Th or ^{238}U so that the reactor is still critical.
- BTW: Uranium enrichment reduces ^{238}U capture, hence also the breeding capability.
- It is about neutron economy.

II. Achievable burnup

- Is limited by FPs neutron capture and by fuel irradiation stability.
- Depends on initial reserve of fissile material and its renewal (breeding capability).

III. Initial fissile mass

- It is determined by neutron economy and spectrum type of the reactor.
- Higher burnup may impose higher initial fissile mass reserve.

IV. Means of criticality maintenance

- Ac. irradiation and FPs creation results in reactivity oscillations / swing.
- Compensation option for reactivity swing differ between reactor types.

V. Transmutation capability

- “Neutron costs” and “speed” of synthetic actinides fission.
- Synthetic Ac. compatibility with the fuel and fabrication process.

MSR: possible absence of structural materials

Radiation stability of the salt

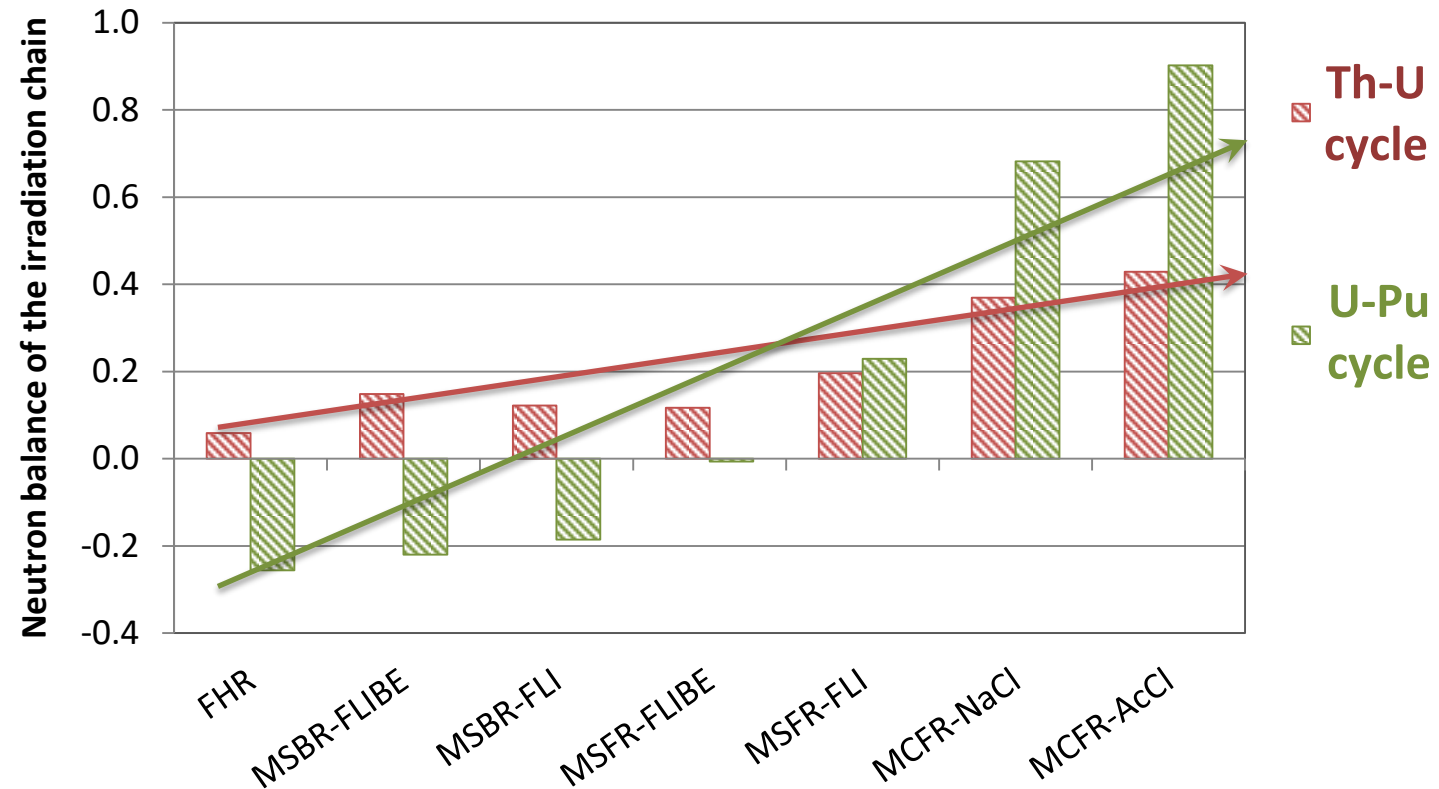
Online refuelling and removal of some FPs

Possible liquid fuel reshaping / draining

Absence of fabrication
Solubility of actinides?

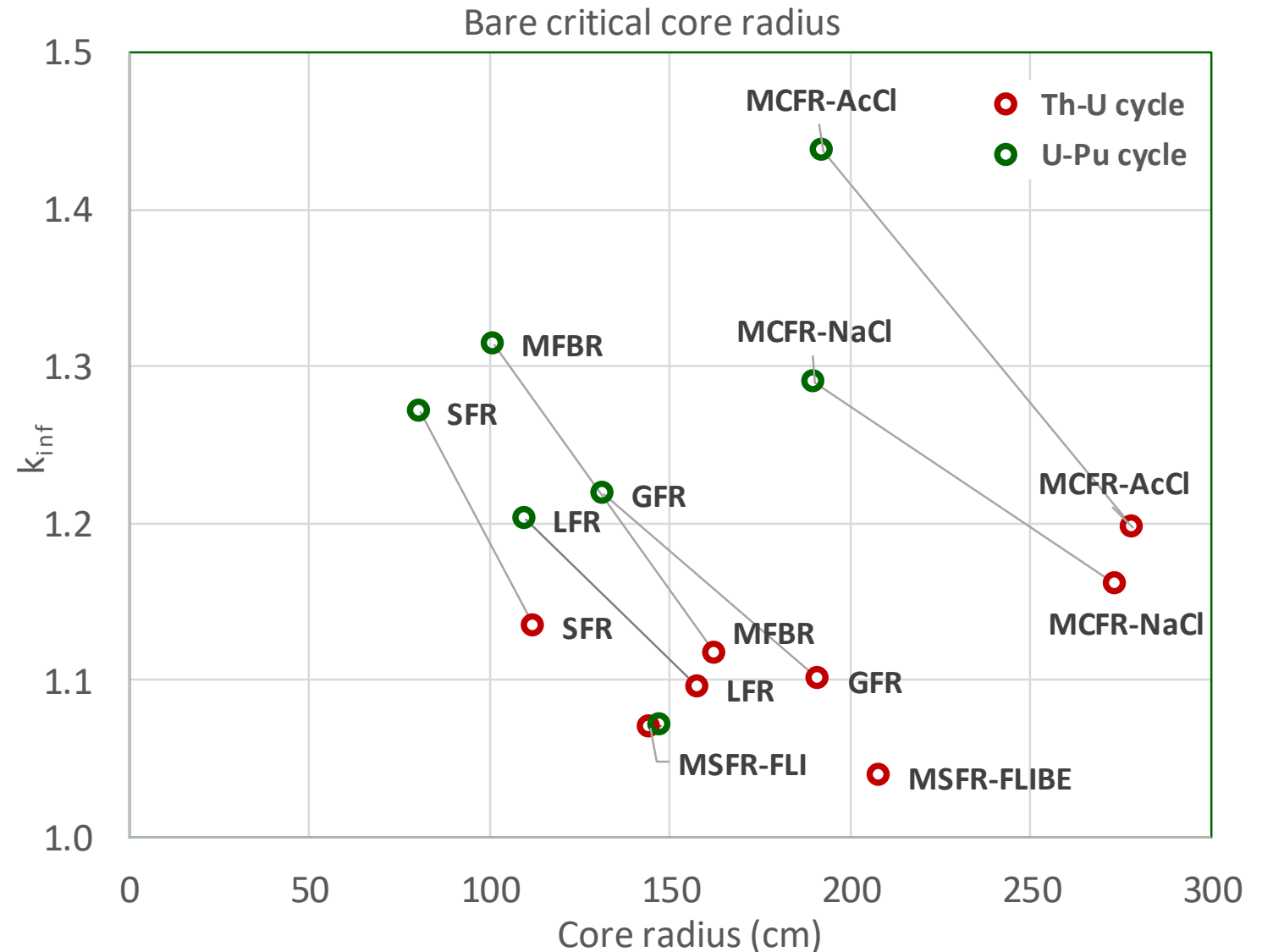
Breeding capability: comparison between ^{232}Th and ^{238}U actinides chain

- ^{238}U actinides chain (U-Pu cycle) profits more from spectrum hardening.
- Better performance: Th-U in thermal and U-Pu in fast spectra.
- Graphite mod. MSR only in Th-U.
- Fluorides fast MSFR possible in both cycles (almost epithermal).
- Chloride fast MCFR possible in both cycles (bulky core for Th-U).
- B&B possible only for chlorides and U-Pu cycle.



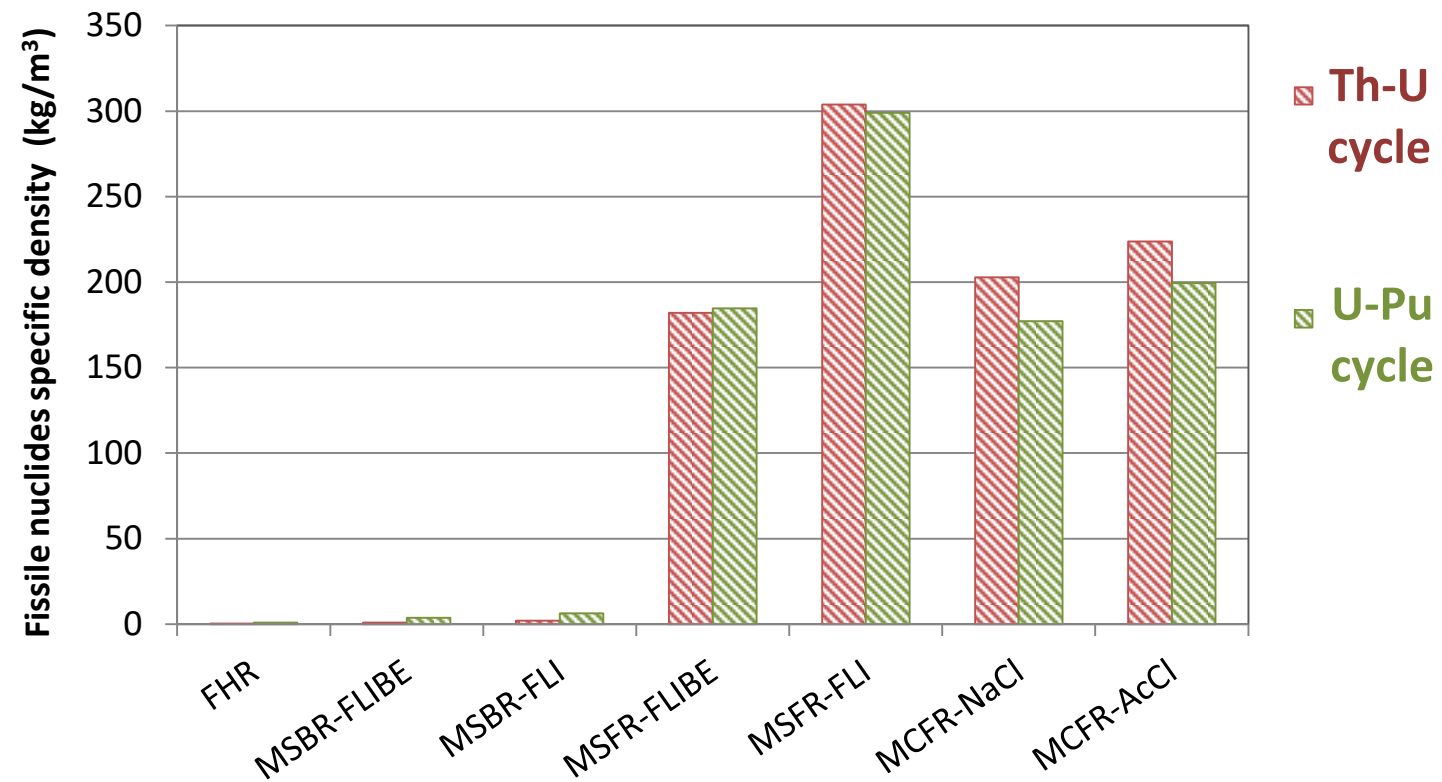
Core radius estimate: Th-U cycle X U-Pu cycle

- MSFR with Li^7F is the smallest MSR core and it has the same core size for both cycles. (very soft fast spectrum)
- By all other fast reactors U-Pu cycle provides smaller cores.
- SFR is the most compact bare iso-breeding core in both cycles.
- MCFR is the biggest bare iso-breeding core in both cycles.
- MSFR with $\text{BeF}_2\text{-Li}^7\text{F}$ is subcritical for U-Pu cycle.



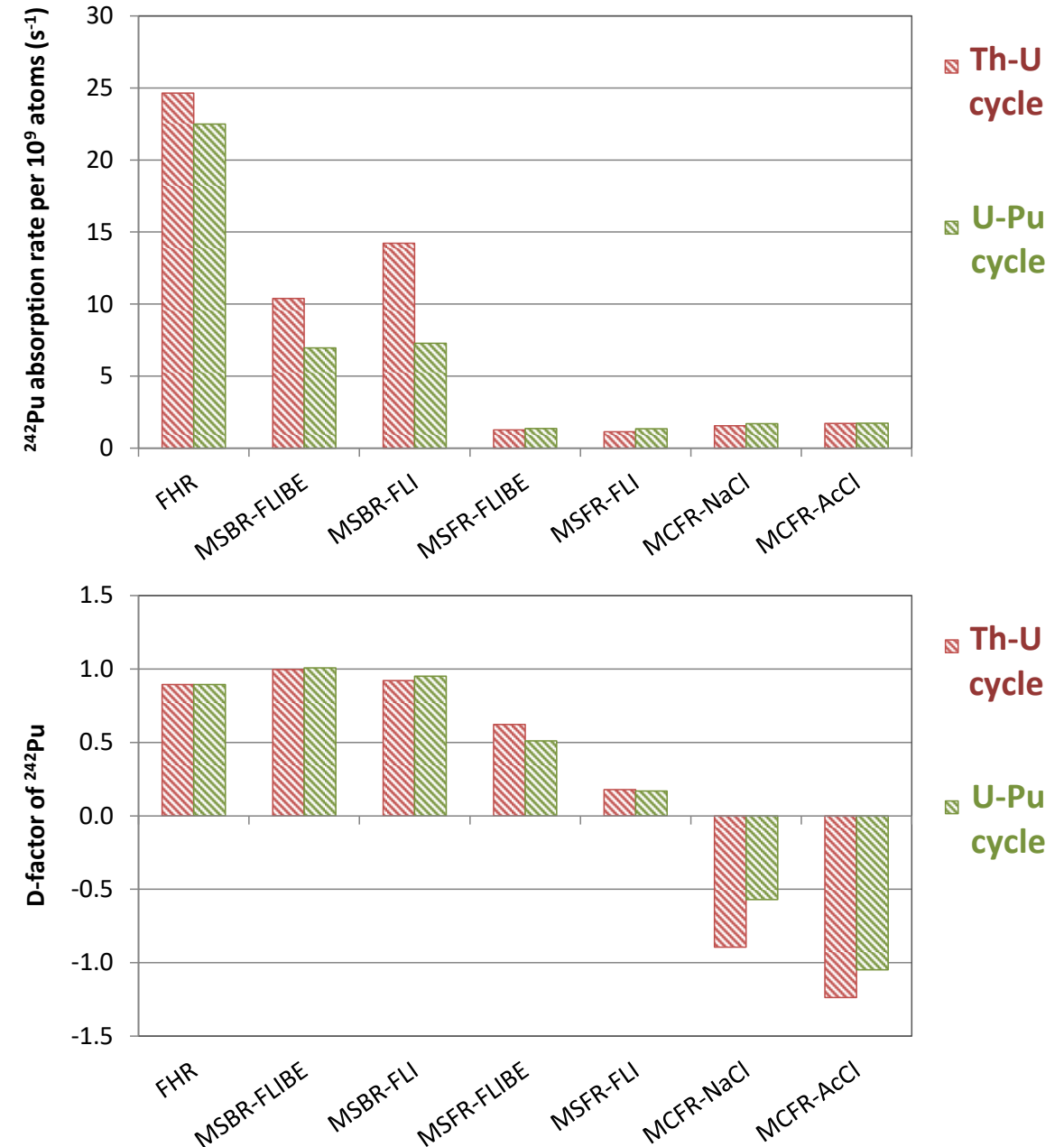
Initial fissile mass

- Initial fissile mass can be defined as a product of core size and fissile actinides specific density.
- In general, initial fissile mass is lower in thermal reactors.
- Especially when moderated by heavy water.



Transmutation capability

- In open cycle the increase or decrease of radiotoxicity per produced unit of energy should be considered. (reprocessing losses in closed cycle)
- The “pace” of transmutation per atom is proportional to the respective cross-section.
- The neutron cost of transmutation in closed cycle can be expressed by D-factor. (total neutron cost of given nuclide transmutation together with its daughters)



Salt compatibility with reprocessing method

Reprocessing as “à la carte” choice

Fuel salts components:

1. Carrier salt (LiF, NaCl,...)
2. Fertile actinides (^{232}Th and ^{238}U).
3. Fissile actinides (^{233}U and ^{239}Pu).
4. Minor actinides (MA).
5. FPs.

Salt treatment / reprocessing techniques:

- Gaseous and volatile FPs removal (off-gas system).
- Metallic FPs removal (sponge filter or by off-gas sys.).
- Molten salt / liquid metal reductive extraction.
- Electro-separation processes.
- Compound evaporation or possibly precipitation.
- Fluoride volatilization techniques, fluorination of the molten salt mixture.

Salt removal from the core	Removed salt share	Fissile fuel recycling	Fissile fuel return after reprocessing	Carrier salt cleaning	Carrier salt return after reprocessing	Reprocessing waste immobilization
Continuous or Batch-wise	From 0.1% to whole salt volume	In-situ or Ex-situ	ASAP or with months or years of delay	In-situ or Ex-situ	ASAP or with months or years of delay	In-situ or Ex-situ

Burnup definition as well as nuclear material accountancy is fluid

- For solid fuel burnup is defined as:

$$B_{GWd/tHM}(t) = \frac{\int_0^t P(t) dt}{M_{Ac}(0)}$$

$$B_{FIMA\%}(t) = \frac{\int_0^t F(t) dt}{N_{Ac}(0)}$$

$$FPS_{share} = \frac{M_{FPS}(t)}{M_{Ac}(0)} = \frac{M_{FPS}(t)}{M_{Ac}(t) + M_{FPS}(t)} = B_{FIMA\%}(t)$$

- For liquid fuel two definitions are possible:
- Differential

$$B_{GWd/tHM}(t) = \frac{P(t)}{\dot{M}_{Ac,in}(t)}$$

$$B_{FIMA\%}(t) = \frac{F(t)}{\dot{N}_{Ac,in}(t)} \cong \frac{\dot{N}_{Ac,in}(t) - \dot{N}_{Ac,out}(t)}{\dot{N}_{Ac,in}(t)} \cong \frac{\dot{N}_{FPS,off-gas}(t) + \dot{N}_{FPS,out}(t)}{\dot{N}_{Ac,in}(t)}$$

- Integral

$$B_{GWd/tHM}(t) = \frac{\int_0^t P(t) dt}{M_{Ac,core}(0) + \int_0^t \dot{M}_{Ac,in}(t) dt}$$

$$B_{FIMA\%}(t) = \frac{\int_0^t F(t) dt}{N_{Ac,core}(0) + \int_0^t \dot{N}_{Ac,in}(t) dt} = \frac{\int_0^t F(t) dt}{N_{Ac,core}(t) + \int_0^t \dot{N}_{Ac,out}(t) dt + \int_0^t F(t) dt}$$

Thank you for your attention