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Jiri Krepel :: Advanced Nuclear System Group :: Paul Scherrer Institut

MSRs: potential for modularity and sizing

中国原子能科学研究院
CHINA INSTITUTE OF ATOMIC ENERGY
中核集团
CNNC

jiri.krepel@psi.ch

Interregional Workshop on Advances in Design of Generation-IV Small and Medium Sized or Modular Reactors (SMRs)

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Outline

Safety consideration:

Basic safety features

Source term mobility

Criticality safety

Fuel cycle considerations

MSR breeding capability assessment

Core size and transition to Th-U cycle

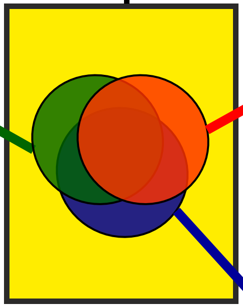
Breed and burn cycle

Safety considerations: source term and driving forces

Why is nuclear safety so particular?

Potential energy:
fission chain reaction can release enormous amount of energy in very short time.

Fuel pellet



Source term:
fission products are highly radiotoxic and cumulate in the fuel several years

Decay heat:
even without chain reaction, the fuel produces heat.

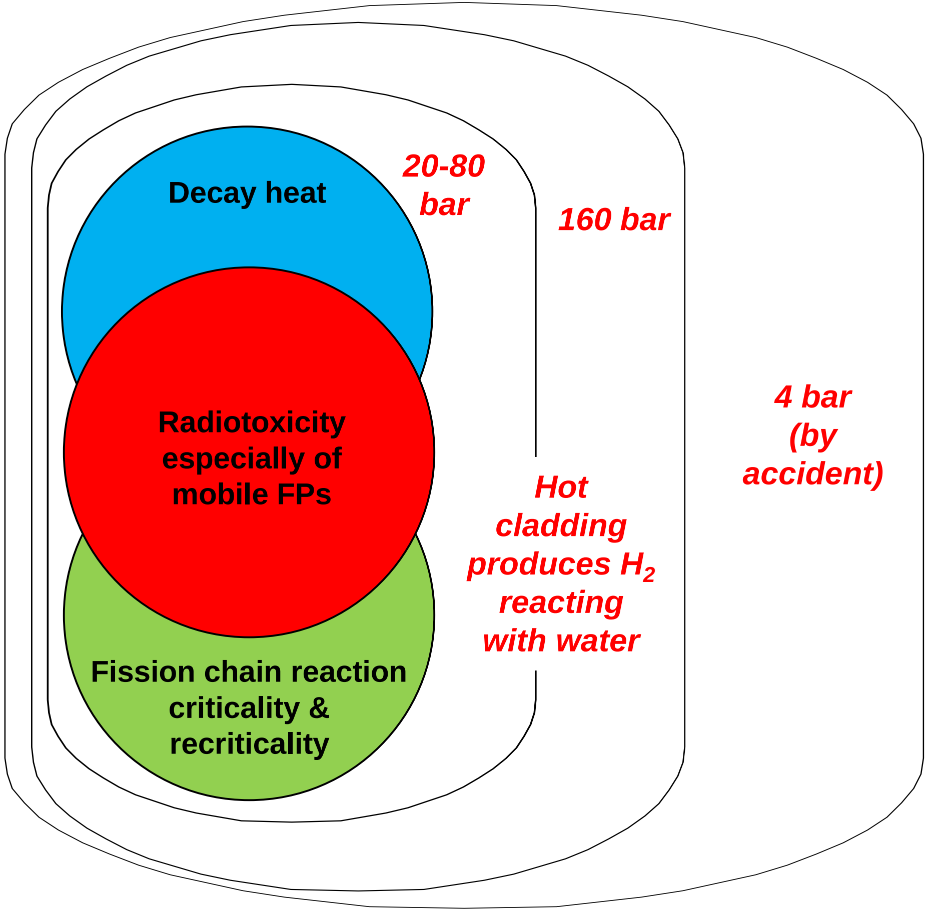
***Dangerous substance** in **dangerous environment**.*

To keep it safe:
***1) criticality accidents** must be excluded,*
***2)** and **continuous cooling** assured.*

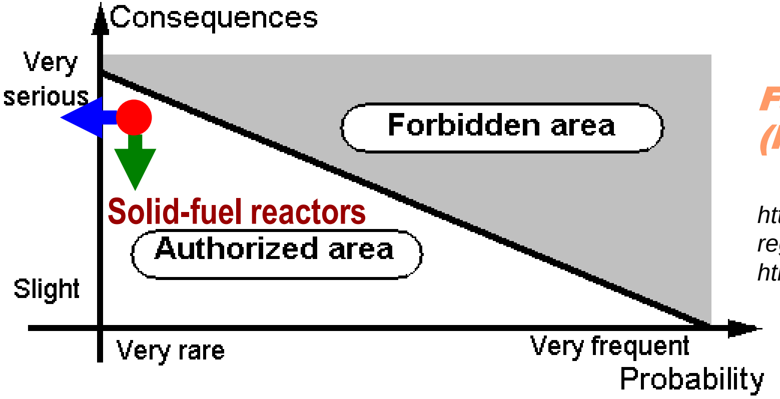
PWR overview: confined hazard

PWR

- *Low criticality & recriticality risks.*
- *Pressurized pin.*
- *Pressurized coolant.*
- *Reactive cladding.*
- *Containment should withstand the steam pressure (& non-condensables).*



Risk = probability X consequences (Farmers curve)



Farmer curve illustration (log-log scale)

http://www.iaea.org/ns/tutorials/regcontrol/assess/assess3212.htm

2 basic directions to improve the position of solid-fuel reactors:

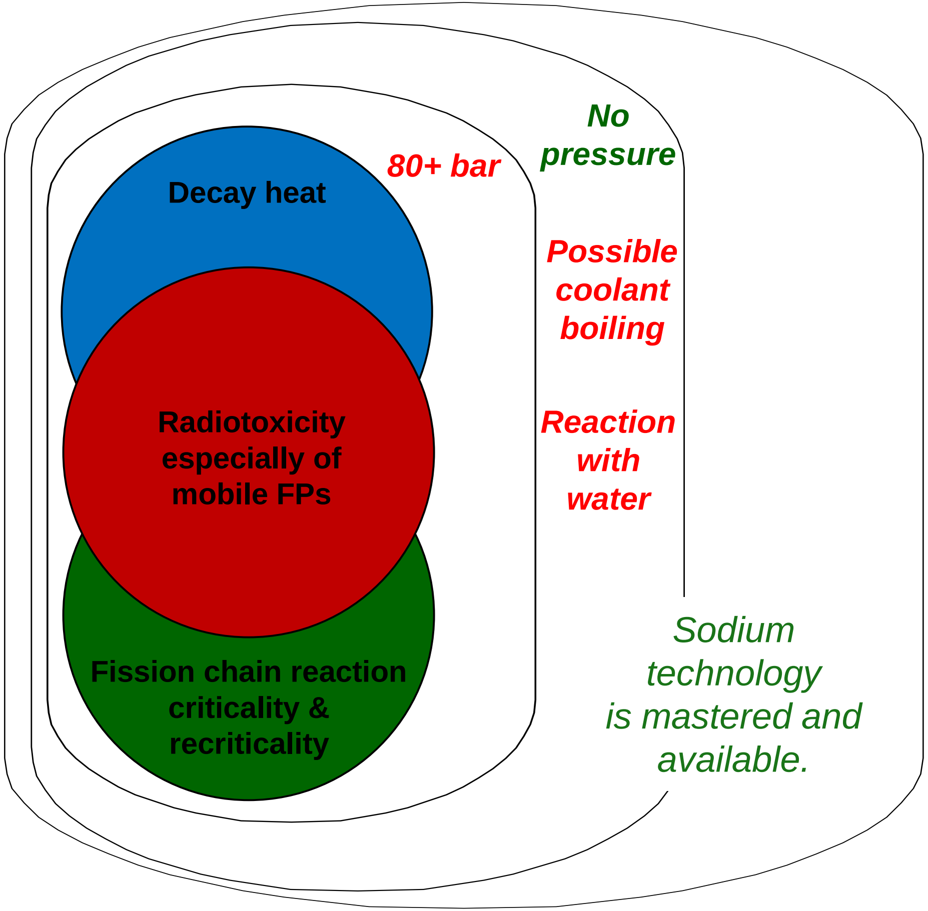
Reducing consequences = reducing source term:
Decreasing installed power or fuel burn-up
(small modular reactors, shorter fuel cycle)

Adding barriers or decreasing the frequency of their failure:
Adding second or third containment…?
Passive systems, redundancy, diversity, underground reactors, etc.
Filtered venting of containment = clever chemical barrier.

Sodium cooled fast reactor safety

SFR

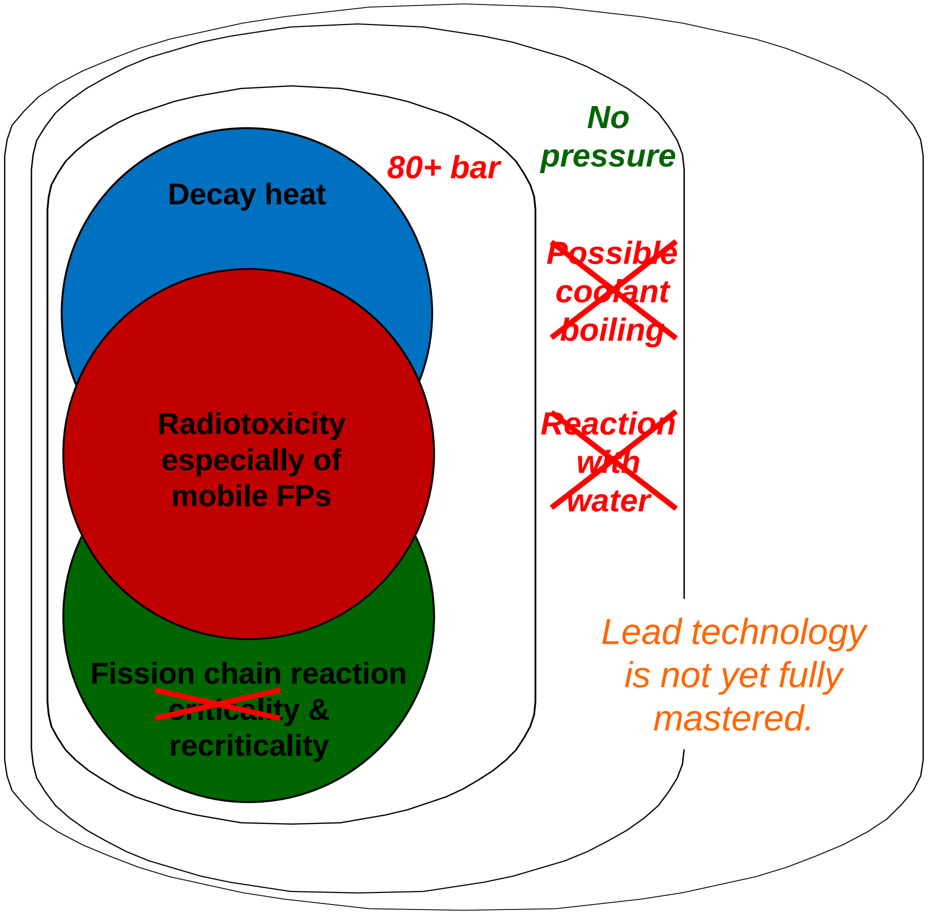
- **Higher burn-up =**
  1) higher source term of radiotoxicity
  2) higher decay heat
  3) higher pin pressure
- **High criticality & recriticality risks**. (enriched fuel, positive void effect)
- Coolant not pressurized but reacting with water**.**
- **Pool type design** may prevent LOCA accidents.



Lead cooled reactor safety (comparison to SFR)

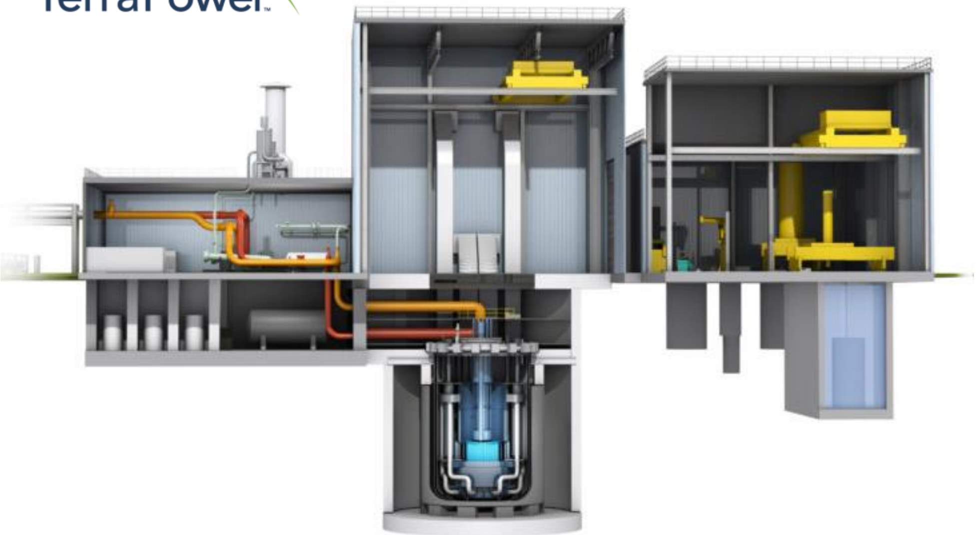
LFR

- **Higher burn-up =**
  1) higher source term of radiotoxicity
  2) higher decay heat
  3) higher pin pressure
- **High criticality & recriticality risks**. (enriched fuel~~, positive void effect~~)
- Coolant not pressurized ~~but reacting with water~~.
- **Pool type design** may prevent LOCA accidents.

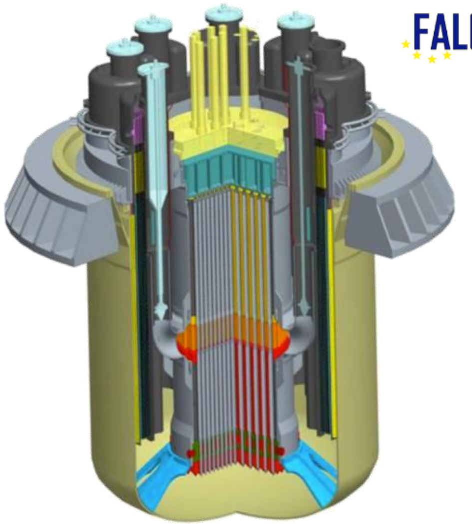


Two examples of LFR and SFR SMRs

Natrium (US)
Small modular SFR
345 MWe
Integrated thermal storage

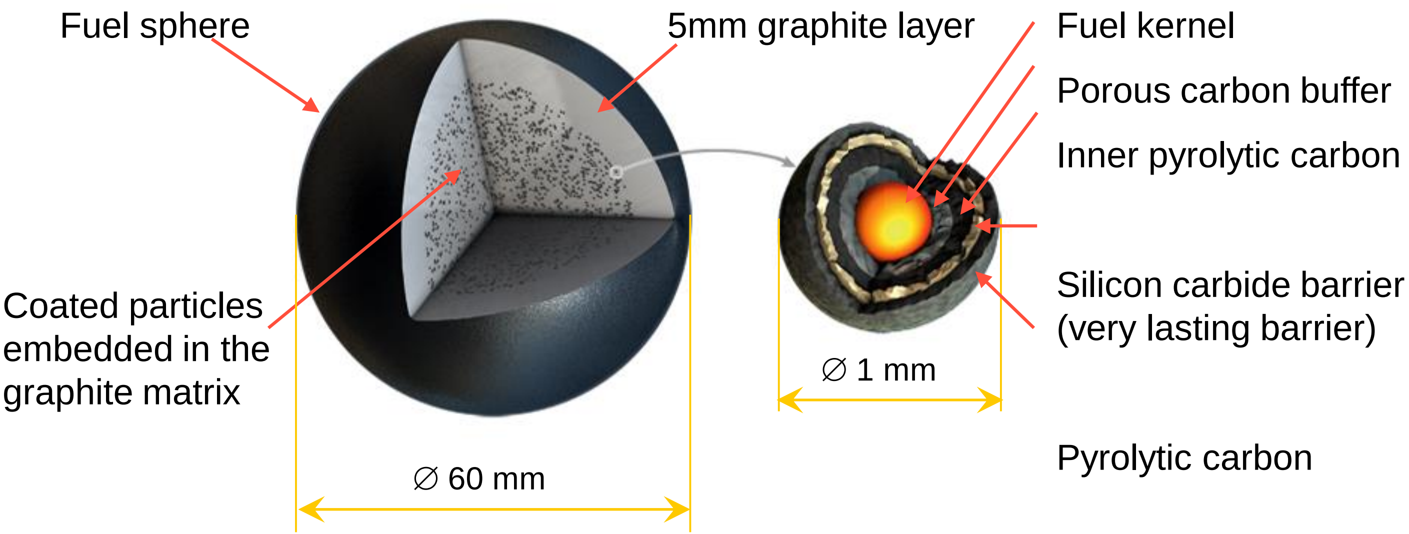


ALFRED (EU)
Small modular LFR
125 MWe
Oxygen control to limit corrosion



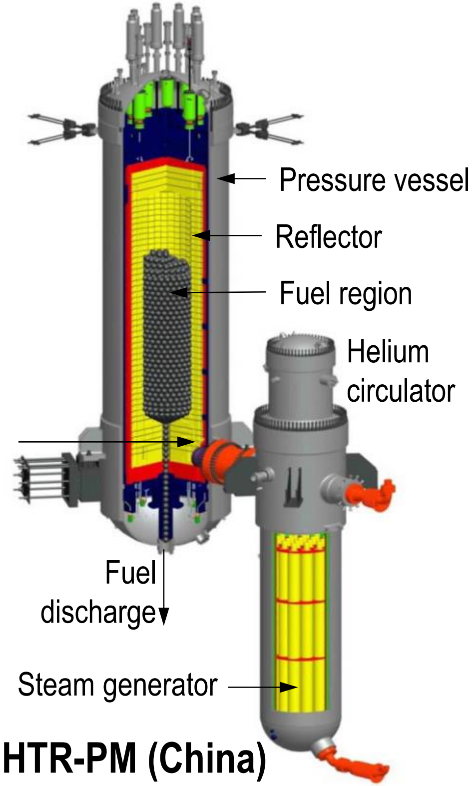
HTR with TRISO particles – different confinement

- ***TRISO*** *(TRIstructural-ISOtropic)* ***coated particles***



- ***Tight up to 1400-1600 ºC.***
- *Even above, no wild changes, melting, or degradation, just FPs release.*
- *If cooled by salt with low pressure, no driving force for FPs release.*
- *Very complicated reprocessing, but suitable for deep burnup.*

Two examples of HTR



HTR-PM (China)

X-Energy (USA)

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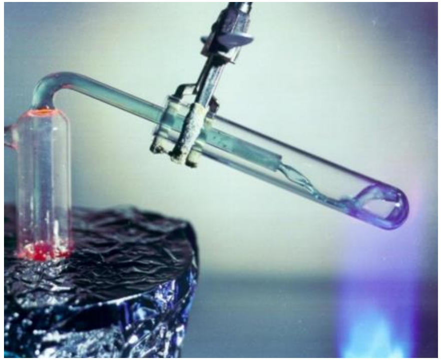
MSR with liquid fuel

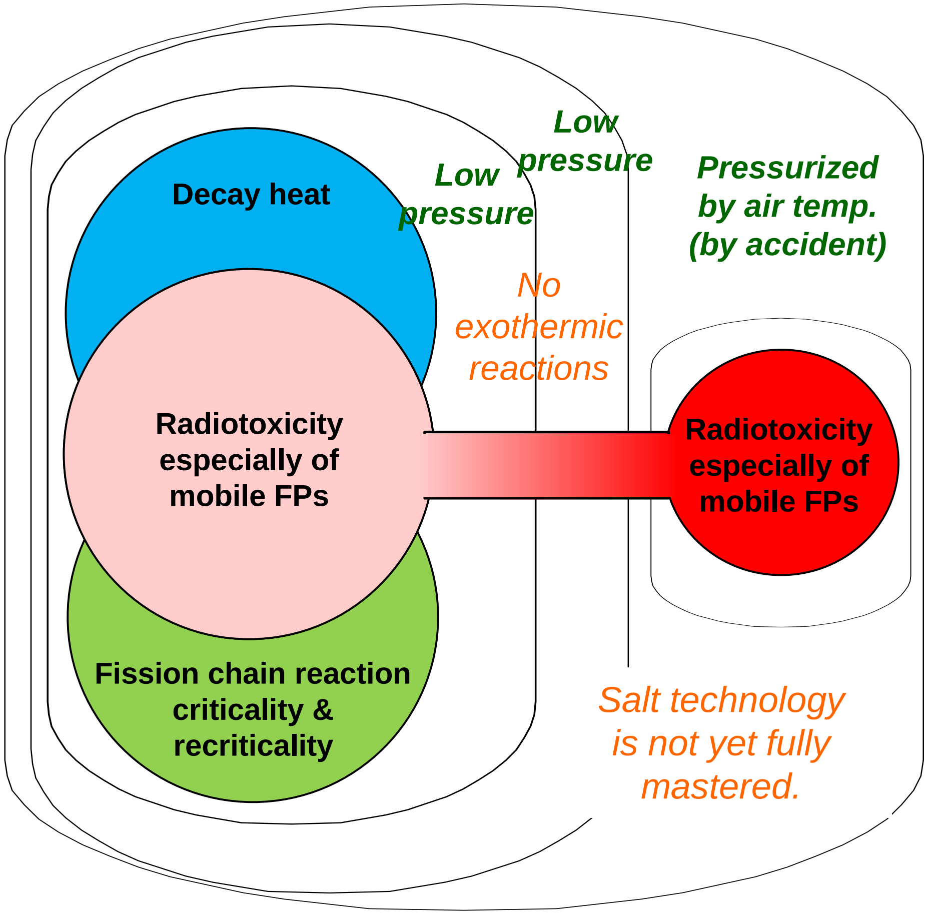


- ***Liquid fuel****, similarly as the solid fuel, retains: uranium, TRU, and soluble FPs.*
- *On the other hand, it provides* ***zero confinement for mobile FPs!***
- *Not only that the* ***mobile FPs*** *are not confined, they* ***must be removed*** *from the fuel.*
- ***Decoupling of mobile FPs from major heat sources*** *may strongly change* ***safety philosophy.***
- *Such a decoupling is possible only in systems with* ***liquid fuel****.*
- *Or in reactor with vented assemblies (engineering challenge).*

Change of safety philosophy: decoupling from heat sources

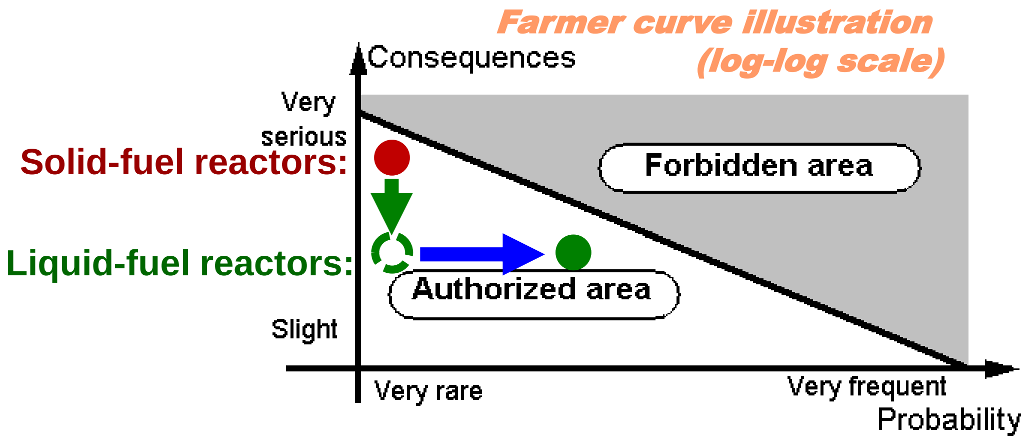
MSR

- *If the mobile FPs would be decoupled from the major heat sources, the risk would be strongly reduced.*
- *Deterministic reduction of serious hazard and consequences.*
- *Regulation for reprocessing plants may be applicable to the removed FPs.*



Risk = probability X consequences (MSR case)

- ***Technological requirement** of volatile FPs **continuous removal** may, together with the negative temperature feedbacks (criticality safety) and possible passive fuel drainage, provide “**deterministic” safety**.*

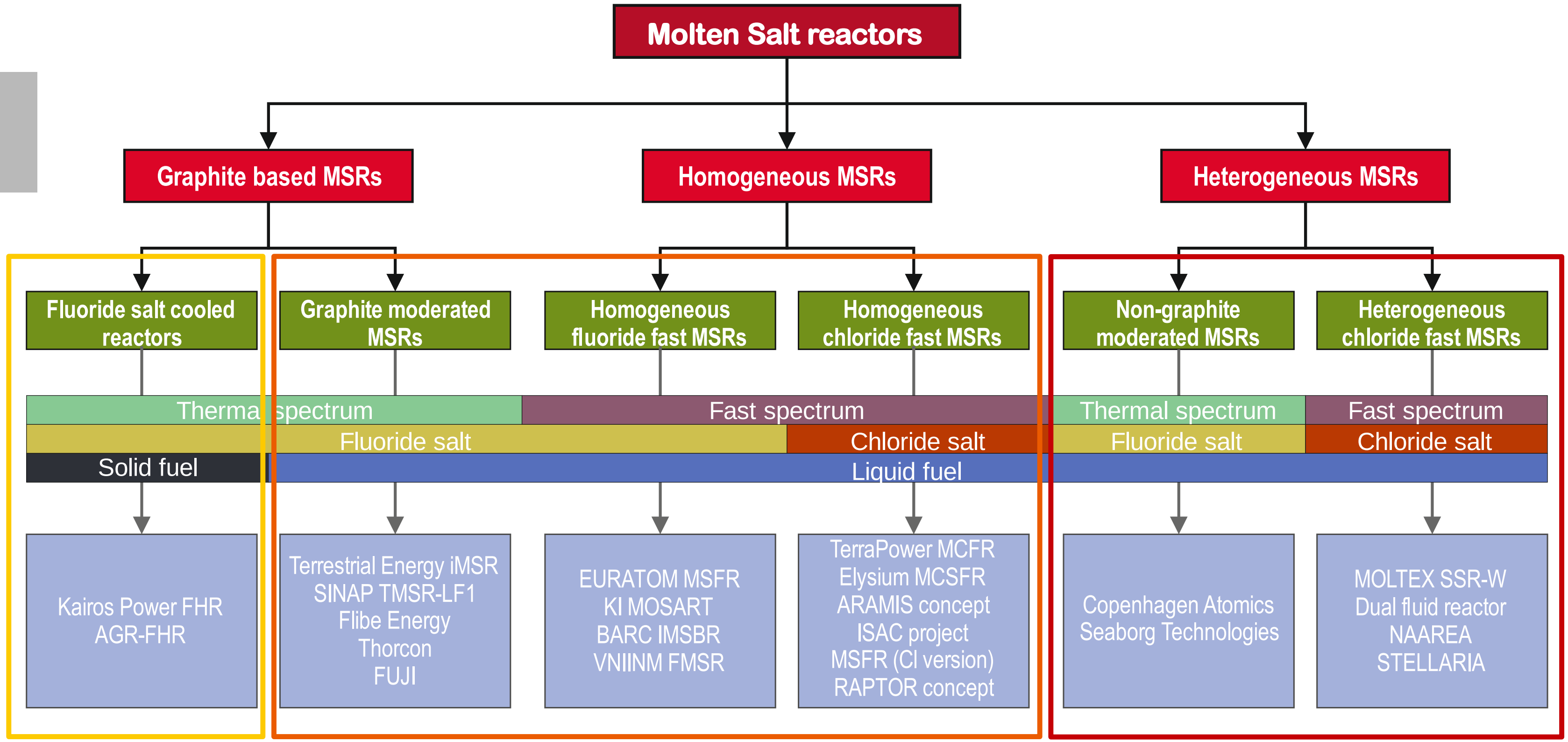


http://www.iaea.org/ns/tutorials/regcontrol/assess/assess3212.htm

Online removal of highly mobile gaseous and volatile FP
(Ac and remaining FP are embedded in the salt)

Daily removal = possible reduction of classical barriers
(safety standards for reactor X reprocessing plant)

MSR as a reactor category rather than a single system



HTR-like safety (salt as a coolant)

Circulating fuel possibly with online cleanup / offline reprocessing

Circulating or solid like fuel (possibly sealed in pins)

IAEA TRS-489 taxonomy, graphical form as in : GenIV International Forum – Annual Report 2022 .

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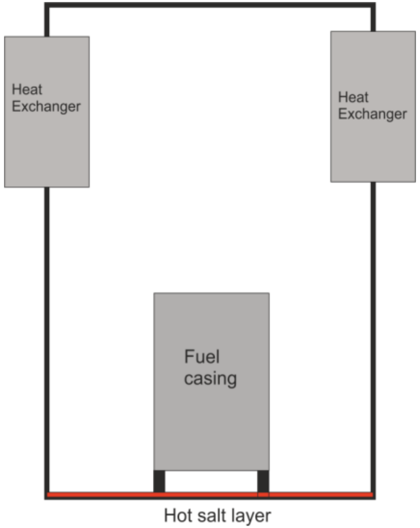
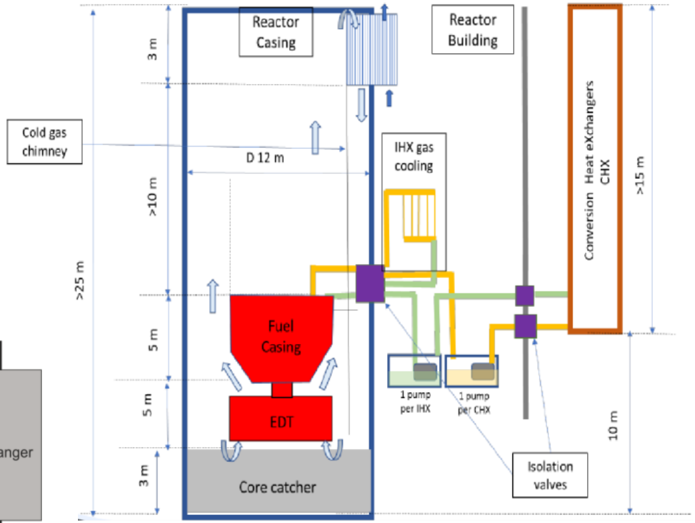
Safety considerations: source term mobility

Simulation of severe accident in MSFR

- In the frame of SAMOSAFER project the deliverable **Aerosols formation and filtration in accidental conditions** was resealed. It is not public, but the results have been also published e.g. in Journal of Nuclear Materials: J. Kalilainen, S. Nichenko, J. Krepel: "Evaporation of materials from the molten salt reactor fuel under elevated temperatures" *https://doi.org/10.1016/j.jnucmat.2020.152134*

- The simulated scenario assumes **salt spill at the bottom of the containment.**

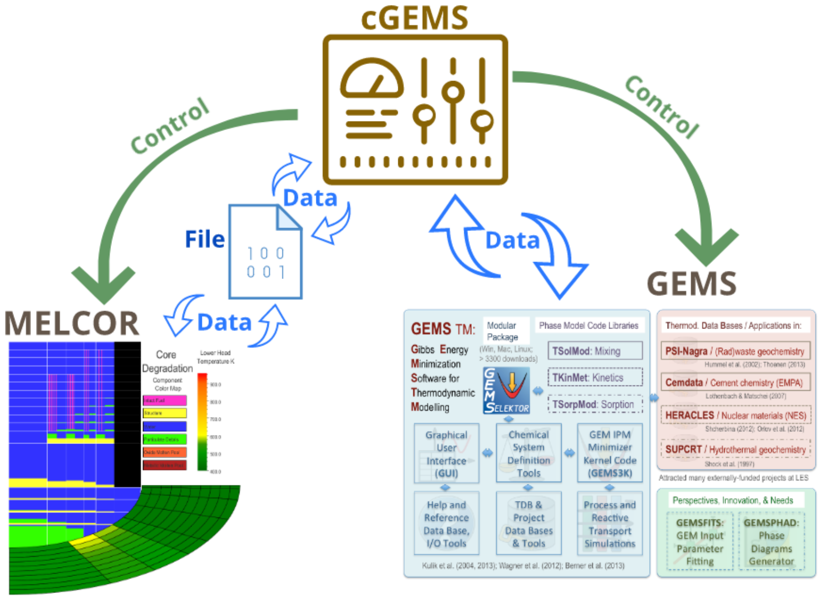


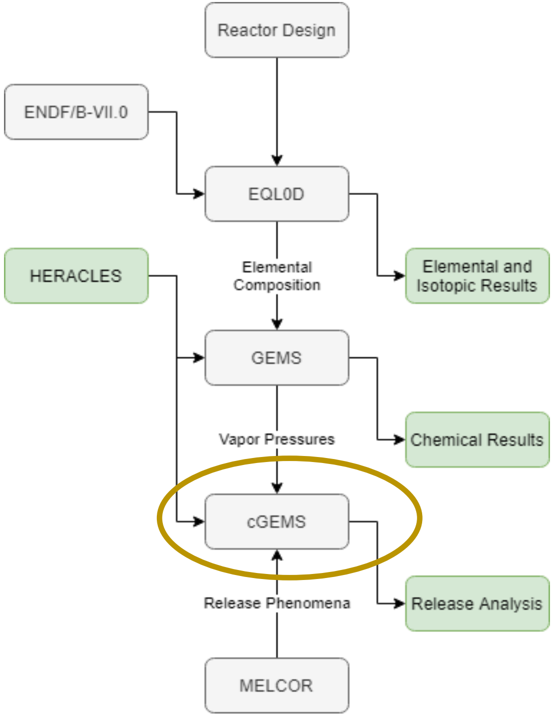


From the SAMOFAR Final meeting, E. Merle et al.

cGEMS code

- cGEMS code characteristics:
  - Composition of the salt from the EQL0D simulations
  - Uses the updated HERACLES database in GEMS software
  - Li, F, U, Th, Cs, Ba, Pu, Sr, La, Zr, Ce, Np, Nd extended system
  - The data exchange between MELCOR and GEMS: cGEMS





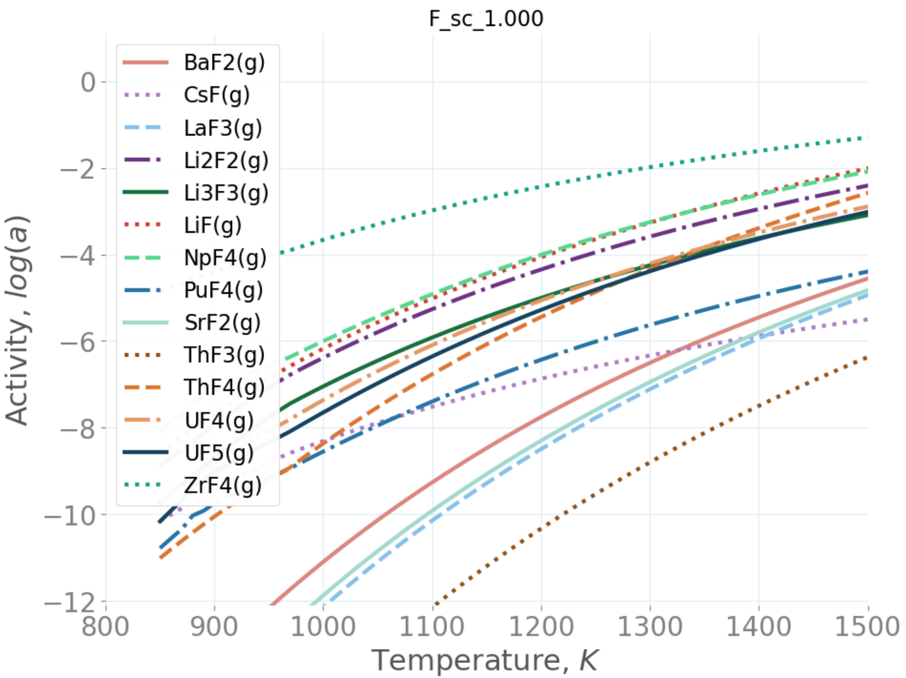
Activity obtained by GEMS/Heracles

- For the simulation Heracles database of the GEMS code was extended.

Species	Changes Made
$ThCl_4$	Imported as is from literature
Np	Imported as is from literature
$PuCl_3$	Adjusted previously existing data entry to conform with literature melting point
UCl_3	Missing liquid phase data manually matched based on literature values
NpF_3	Missing liquid phase constructed from melting-/boiling points and similarity to UF_3
AmF_3	Solid adjusted and liquid designed from assumed similarity to UF_3
ZrF_4	Imported as is from literature
$NdCl_3$	Imported as is from literature
$PrCl_3$	Imported as is from literature
PrF_3	Imported as is from literature
Na_2ThCl_6	Created in GEMS function ReacDC
Pr	Imported as is from literature

Additional Changes were made to:
NpF_4, NdF_3, SrF_2, LaF_3, CeF_3, BaF_2, CsF

- GEMS code was applied to obtain the vapor pressures.



Compounds activity (proportional to vapor pressure) as a function of temperature

Released mass sensitivity to redox

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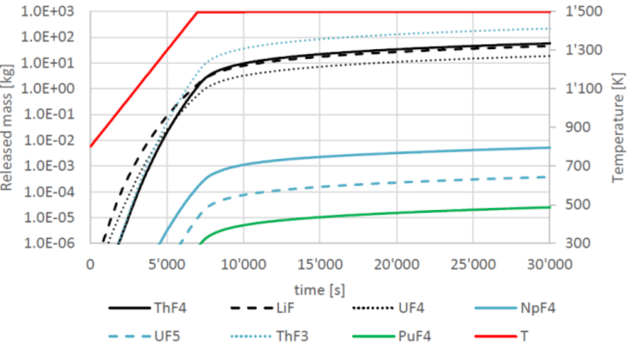
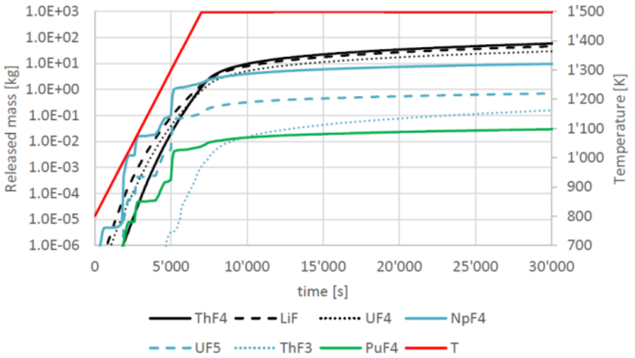
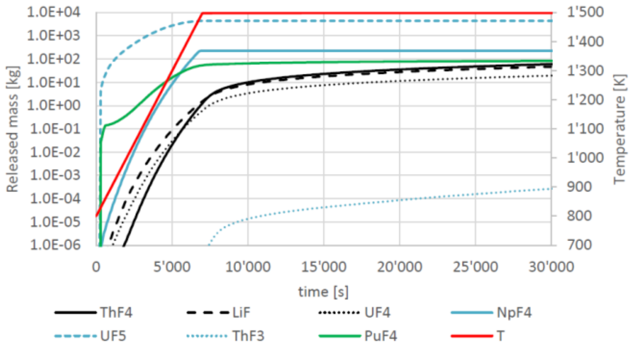
Fluorine: -1% mol.

Stoichiometric

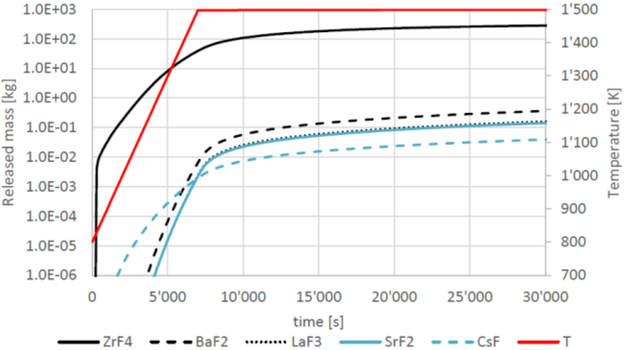
+1% mol.

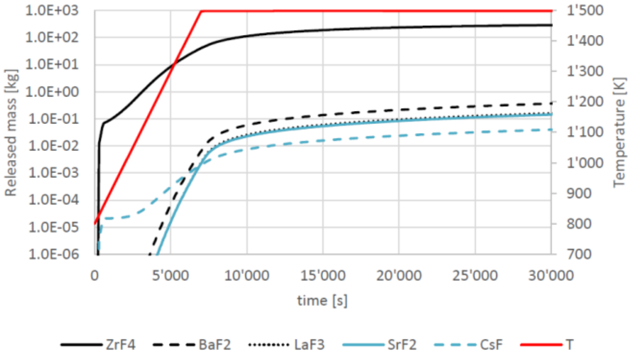


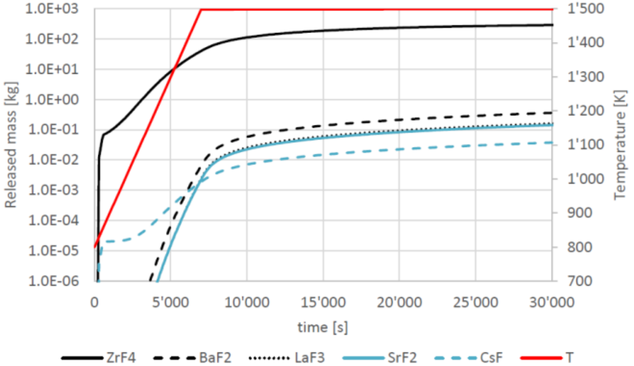




Actinides and salt species





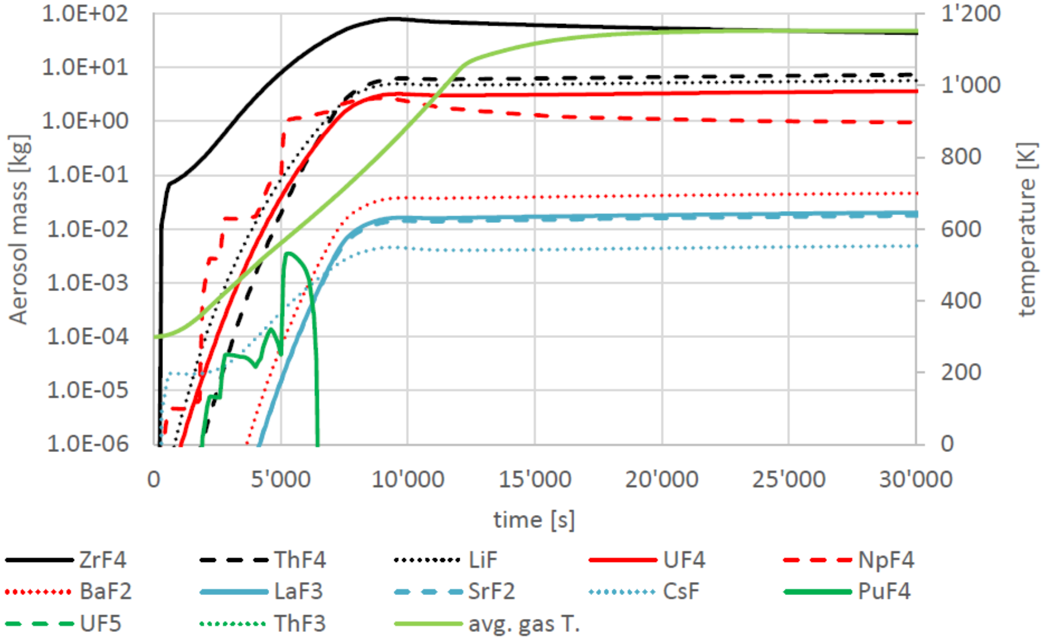


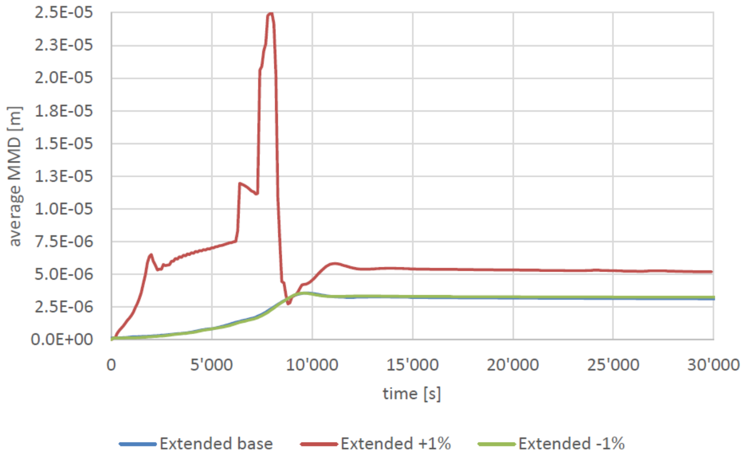
Fission products species

Total released mass during the accident (salt heat up from 800°C to 1500°C)

Released aerosols mass (cGEMS)

- Characterization of released aerosols



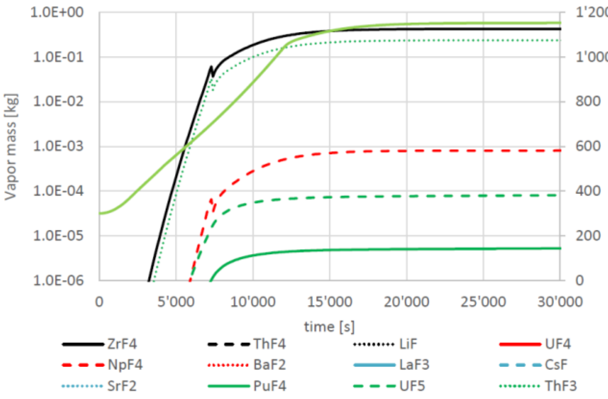


Total released mass of aerosols (left) and their mass median diameter (MMD) during the accident (salt heat up from 800°C to 1500°C)

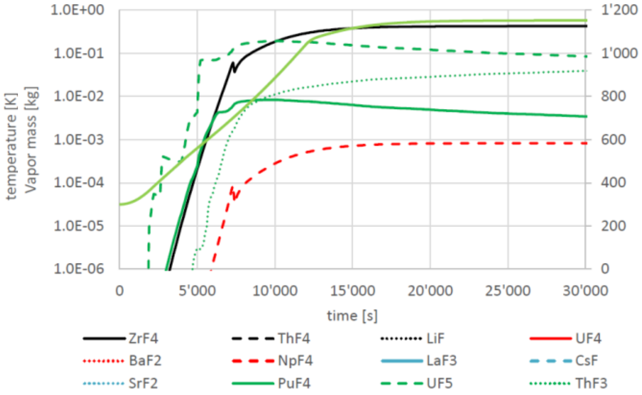
Released vapors mass (cGEMS)

- Characterization of released vapors

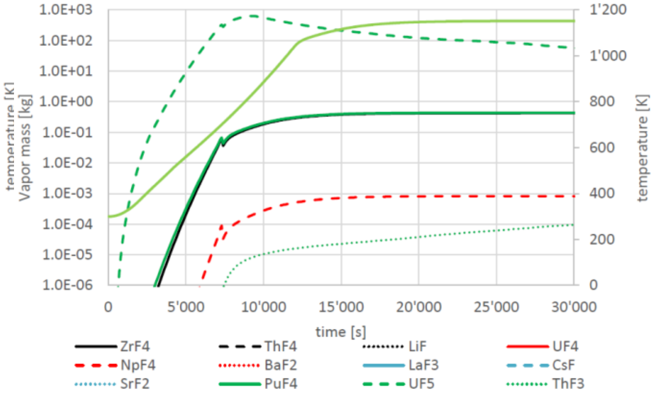
Fluorine: -1% mol.



Stoichiometric



+1% mol.

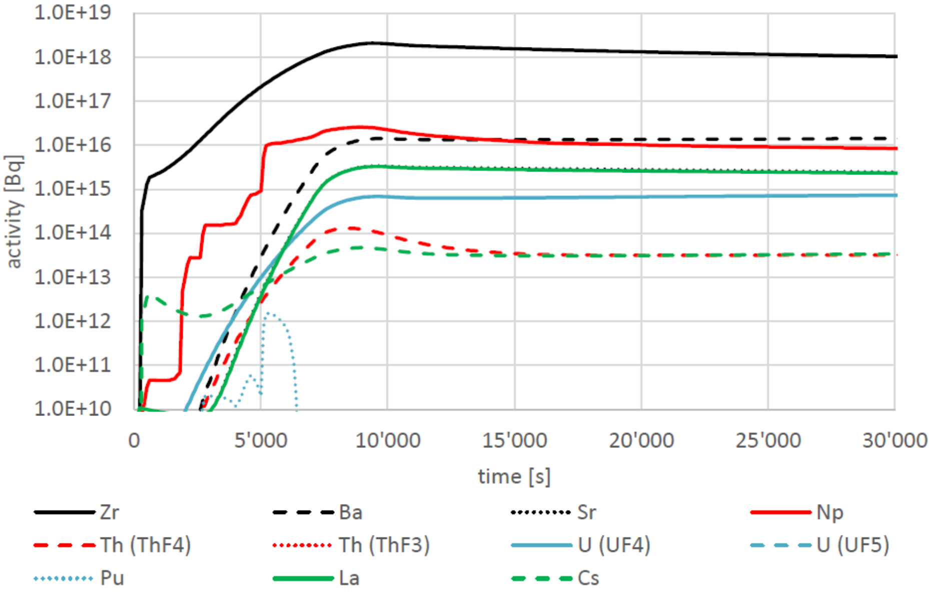


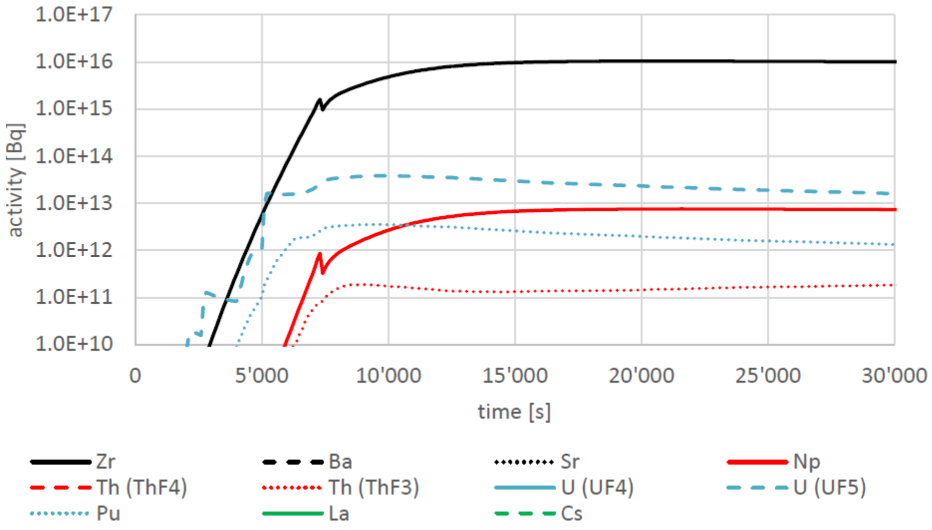
avg. gas T.

Total released mass of vapors during the accident (salt heat up from 800°C to 1500°C)

Released activity (cGEMS)

- Characterization of released activity in form of aerosols and vapors

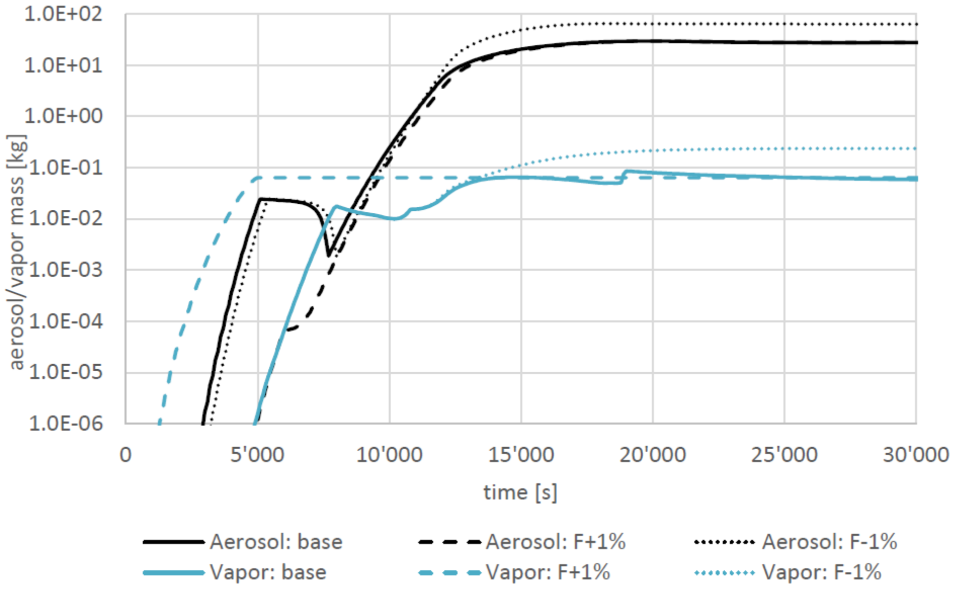




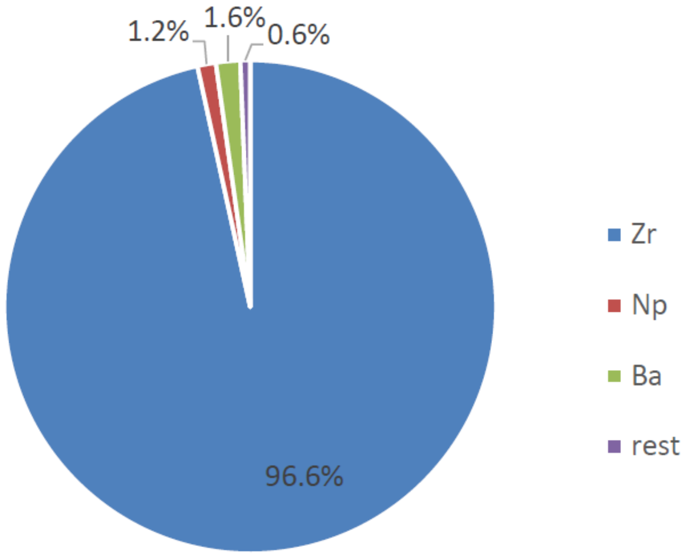
Total released activity in form of aerosols (left) and vapors (right) during the accident (salt heat up from 800°C to 1500°C)

ZrF_4 as the major component?

- Based on the applied reprocessing scheme, ZrF_4 in form of aerosols seems to be the major activity carrier during the postulated accident.



Total released activity in form of aerosols and vapors during the accident (salt heat up from 800°C to 1500°C)



Activity break-down at the end of simulation (t=30’000s) of the accident (salt heat up from 800°C to 1500°C)

Summary – source term

- Localization and mobility of source term in MSR strongly depends on the in-situ salt treatment.
- Off-gas system may represent its major component.
- Source term released from the salt after vessel damage is determined by the salt treatment during nominal operation.
- In presented study ZrF_4 was not removed from the fuel salt during nominal operation, hence it dominates in the source term.
- The distributed source term might increase both capital and operation costs and bigger units might be more profitable.

Safety considerations: criticality safety

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Criticality safety: thermal feedback coefficients

Thermal-hydraulics interaction with neutronics

Type of:

Impact

Impacted property / medium

Impact mechanism

Materials temperature

Atomic motion velocity

1. Changed microscopic cross-sections

Atomic density

2. Changed density and geometry of solids

3. Changed density of fluids

Fluids velocity

Moving of coolant atoms

Negligible impact
(particle motion velocity >> fluids velocity)

Moving of fuel atoms

Drift of delayed neutron precursors (issue for MSR)

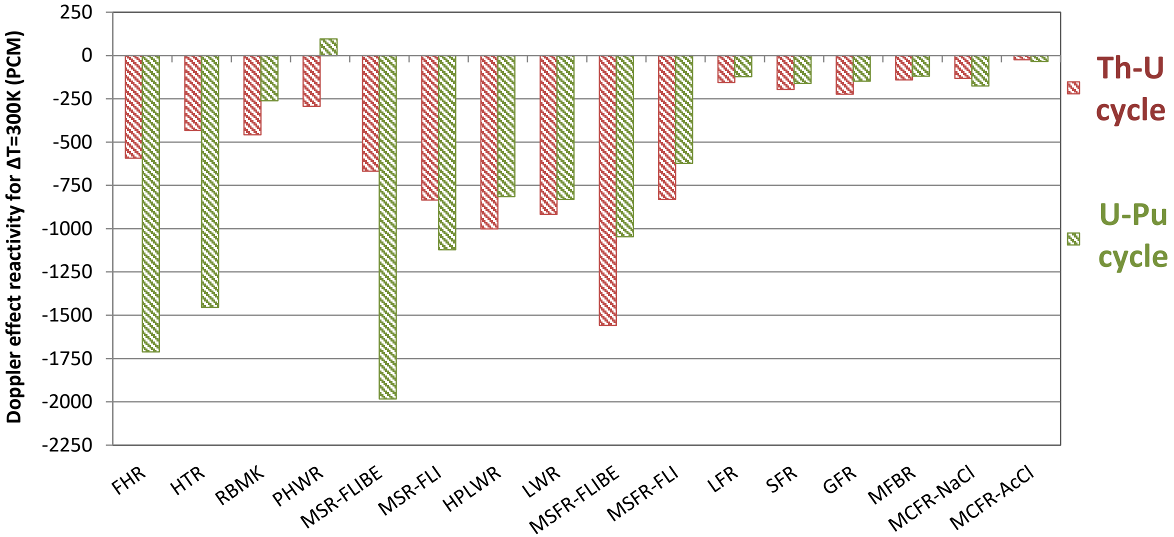
1. Changed microscopic Cross-sections (XS)

- **Doppler broadening of XS** result in **negative reactivity** practically in all reactor with substantial share of heavy nuclei with capture XS resonances (**^{238}U and ^{232}Th**).

- As an example reactivity introduced by **300K** fuel temperature increase is shown.

- *The figure shows results for **iso-breeding fuel composition**. It is not a value for standard fuel composition.*

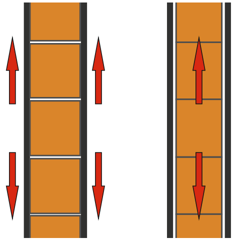
- *No time to discuss PHWR. Have a look on the refered paper.*



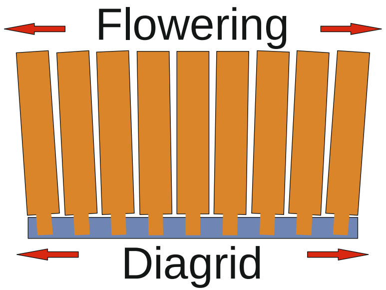
Krepel, J., Losa, E., 2021, Self-sustaining breeding in advanced reactors: Characterization of selected reactors, Chapter 00123, Nuclear encyclopedia.

2. Changed density and geometry of solids

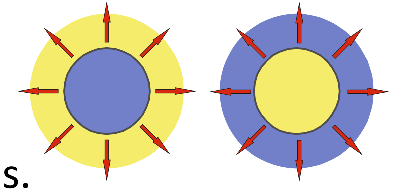
- **Temperature increase** of solid materials results in their **thermal expansion**.
- **Active core** is delimited by **fuel presence**.
- Active core **axial** expansion is driven by **fuel or cladding temperature**.
  (cladding temperature drives fuel position in case of closed gap between them)



- Active core **radial** expansion is driven by **diagrid expansion** and assemblies **flowering effect**.



- **Non-fuel** solid **materials** can expand:
  – **outwards** from the active core, e.g., **cladding** with open gap
  – **inwards** to the active core, e.g., control rods and **control rods drivers expansion**
- Expansion of **solid materials can expel fluids** from the core.
  – e. g., cladding radial expansion expels sodium from the core.
  – in rare cases it can be the opposite, e. g., fluid in expanding tubes.



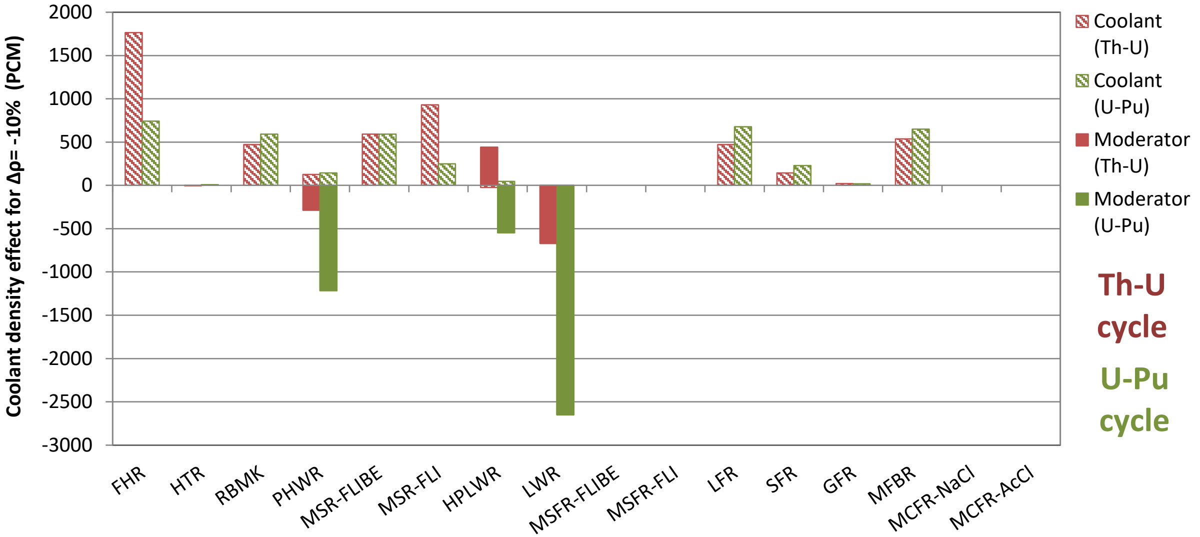
Yellow: fluid, Blue: solid

3. Changed density of fluids

- **Decreased density** result typically in **positive reactivity.** Only for moderating fluid it is negative (reduced scattering XS dominate).

- As an example **10% density reduction** is shown. *(infinite lattice simulation = 0 for hom. reactors)*

- *The figure shows results for **iso-breeding** fuel **composition**. It is not a value for standard fuel composition.*



- *In MSR-FLI and MSR-FLIBE two effects take a part: reduced salt capture and changed fuel-to-moderator ratio, Doppler effect is not included.*

Krepel, J., Losa, E., 2021, Self-sustaining breeding in advanced reactors: Characterization of selected reactors, Chapter 00123, Nuclear encyclopedia.

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4. Fuel circulation

- In many MSR concepts fuel circulate through the reactor and primary circuit.
- Obviously, actinides and fission product circulate too.
- As a consequence, decay heat and delayed neutrons release is distributed.
- Using simple 1D approximation, the equation for precursors writes as:

$$\frac{\partial}{\partial t}C(z,t)+\frac{\partial}{\partial z}\big(vC(z,t)\big)=Q(z,t)-\lambda C(z,t)$$

$$\cancel{\frac{dC(t)}{dt}=Q(t)-\lambda C(t)}$$

- Assuming constant velocity v and characteristic line z=vt we can integrate it:

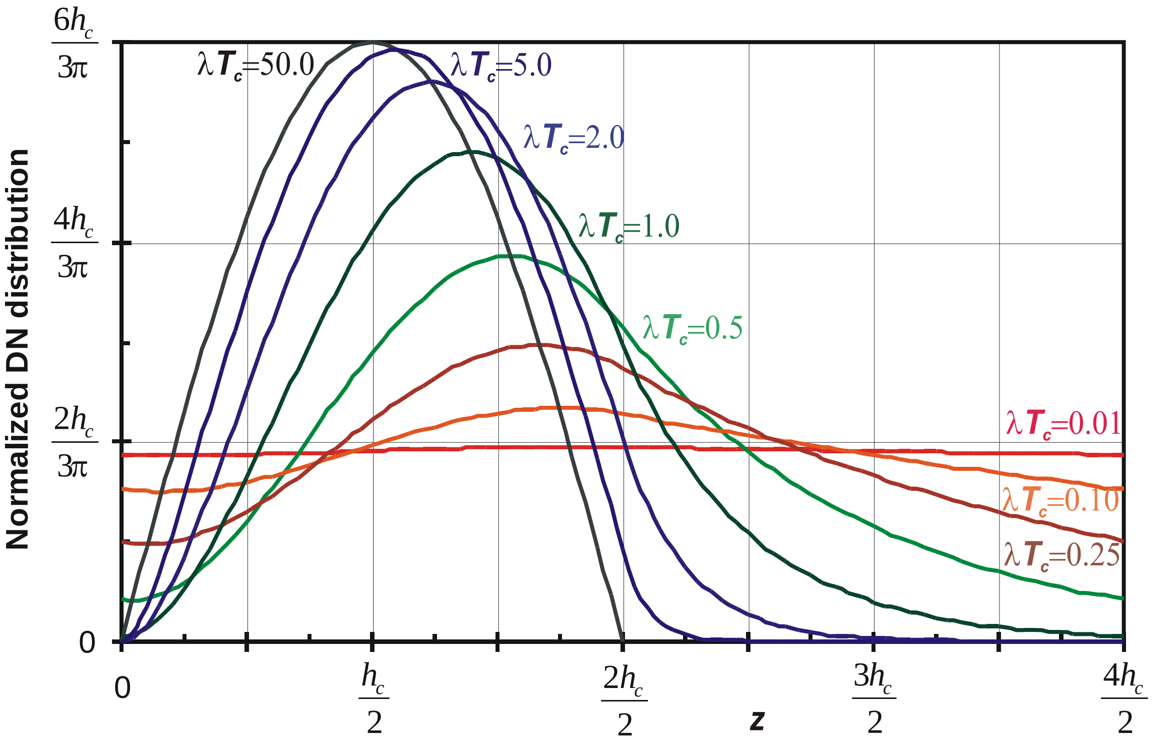
$$C(z,z/v)=e^{-\frac{\lambda}{v}(z-z_0)}\left(C(z_0,z_0/v)+\int_{z_0}^{z}\frac{Q(z',z'/v)}{v}e^{\frac{\lambda}{v}(z'-z_0)}dz'\right)$$

- Lets assume also sinus shape of the neutron flux:

$$Q(z)=\frac{\pi}{2h_c}\sin(z\pi/h_c)\quad for\quad z=(0,h_c)$$

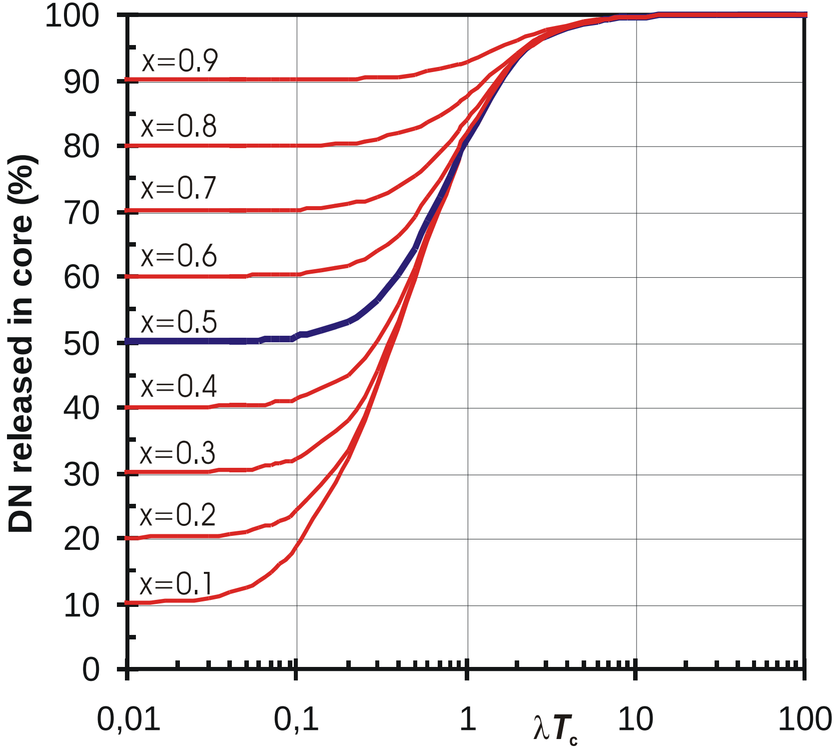
4. Fuel circulation

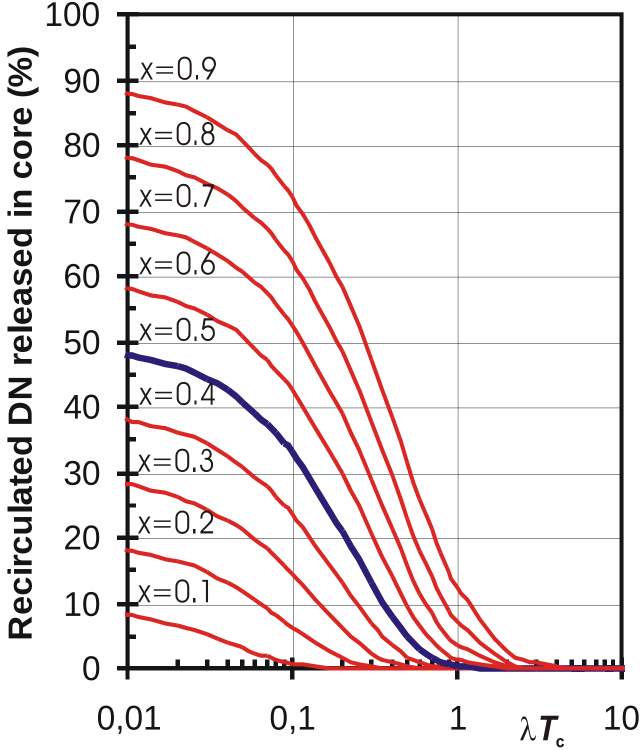
- Distribution of delayed neutrons.



4. Fuel circulation

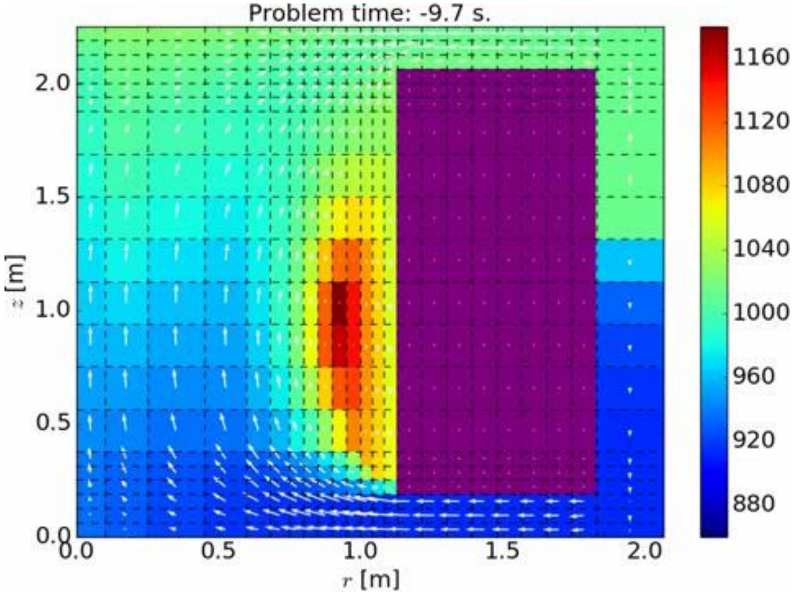
- Share of delayed neutrons released in the core.



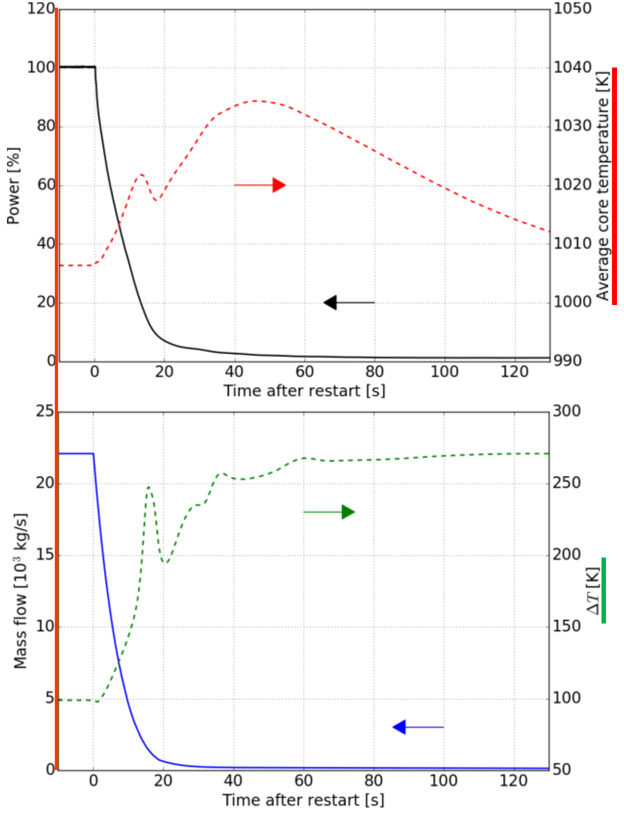


TRACE application to MSFR

- **TRACE-PARCS** system code application to **MSFR**.
- Applying TRACE vessel component.
- Applicable for 3D transients (2D symmetric transients).
- Capability of system analysis with acceptable accuracy and CPU time.



E. Pettersen, LRS, MSc thesis 2016



ULOF- unprotected loss of flow

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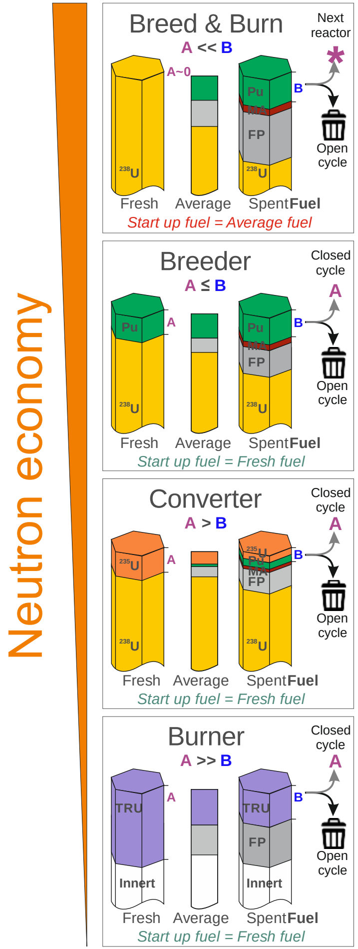
Summary – criticality safety

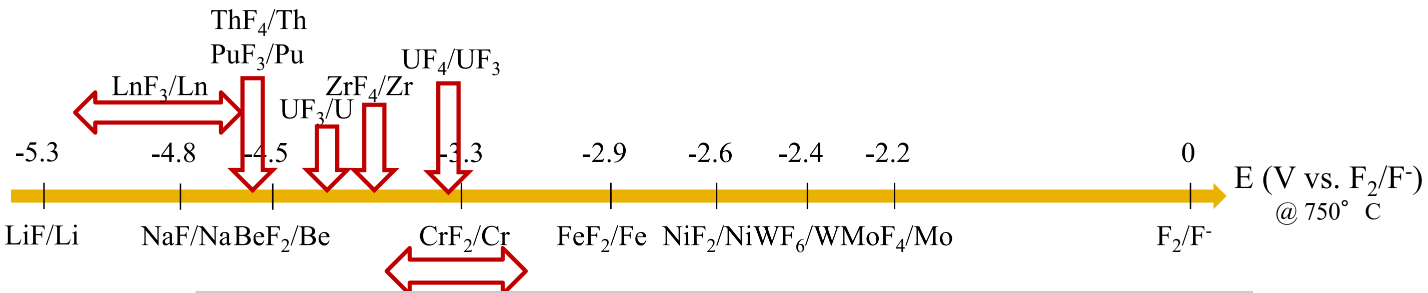
- MSR can be designed with negative Doppler effect.
- In case of waste burner fueled by minor actinides, the weaker or even positive, Doppler effect can be compensated by negative fuel salt expansion coefficient.
- MSR can be designed as a breeder in Th-U or U-Pu cycle with all thermal feedback coefficients being negative.
- Small size of the core or low power in not necessary for that.
- Fuel circulation redistributes delay neutron and decreases their weight.
- The only sizing consideration related to safety is decay heat removal.

Fuel cycle considerations

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.

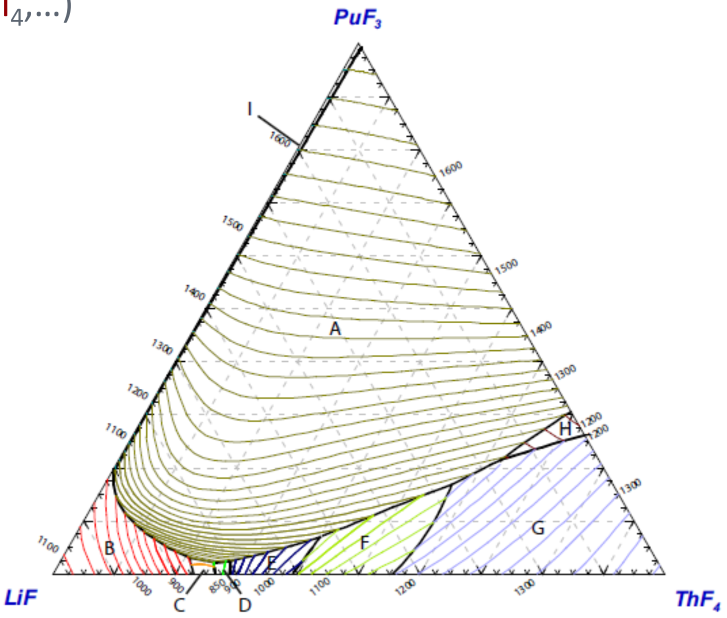


Relative stability of redox couples, Gibbs energy : $\Delta E\ (V) = -\Delta_r G(J/mol)/nF$, $E°(F_2/F^-) = 0$ V

Adopted from: Laurent Cassayre, LGC.CNRS.FR, Corrosion in molten salt reactors, EVOL Winter School, 4-6 November 2013, IPNO, France

Reasonable temperature window for operation:
interval between salt melting and structural materials failure

- Typically eutectic mixture of carrier salts (LiF, BeF_2, NaF, LiCl, NaCl,...) and actinides salts (ThF_4, UF_4, PuF_3, $PuCl_3$, UCl_3, $ThCl_4$,...)
- **MSRE** salt, **$T_{melt.}$=432°C**
  65%LiF - 29.1BeF_2 - 5%ZrF_4 - 0.9%UF_4
- **MSBR** , Th-U equilibrium cycle, **$T_{melt.}$=500°C**
  71.7%LiF - 16%BeF_2 - 12%ThF_4 - 0.3%UF_4
- **MSFR**, Th-U equilibrium cycle, **$T_{melt.}$=560°C**
  78%LiF - 17.6%ThF_4 - 4%UF_4 - 0.2%PuF_3
- **MSFR**, Pu started Th-U cycle, **$T_{melt.}$=625°C**
  78%LiF - 16%ThF_4 - 6%PuF_3
- **MCFR**, Pu started U-Pu cycle, **$T_{melt.}$=565°C**
  60%NaCl - 35%UCl_3 - 5%$PuCl_3$
- **MCFR**, Pu started Th-U cycle, **$T_{melt.}$=425°C**
  55%NaCl - 39%$ThCl_4$ - 6%$PuCl_3$
- Generally **solubility limits** (e.g. PuF_3) and **actinides density** compete with **melting temperature**.



LiF-ThF_4-PuF_3 ternary phase diagram w/ fixed 1% mol UF_4 concentration

*E. CAPELLI et al., “Thermodynamic Assessment of the LiF–ThF4–PuF3–UF4 System,” J. Nucl. Mater., **462**, 43 (2015).*

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.

Reprocessing strategies:

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

Fuel cycle assessment from neutronics perspective

For comparison with other reactors refer to:
Krepel, J., and Losa, E., Self-sustaining breeding in advanced reactors: Characterization of selected reactors, Encyclopedia of Nuclear Energy, (Greenspan, E., Ed.), Elsevier, 2021.

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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.

5 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 Fission Products (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.

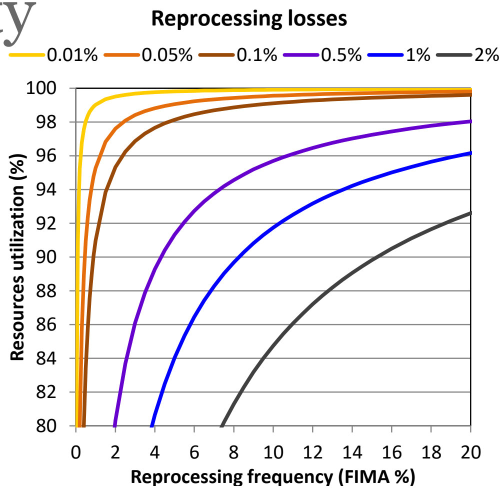
V. Transmutation capability
- “Neutron costs” and “speed” of synthetic actinides fission.
- Synthetic Ac. compatibility with the fuel and fabrication process.

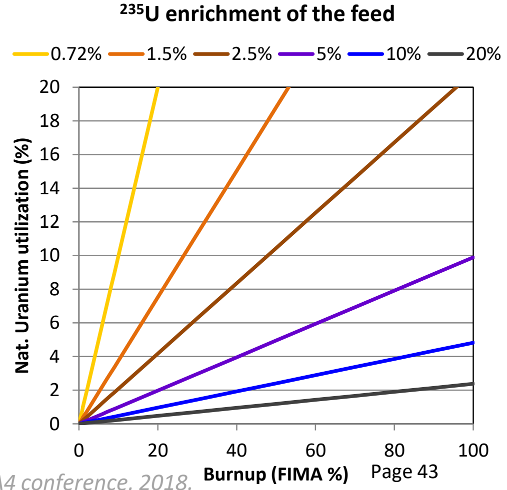
Krepel, J., Self-sustaining breeding in advanced reactors: Comparison of fuel cycle performance, Encyclopedia of Nuclear Energy, (Greenspan, E., Ed.), Elsevier, 2021.

Ac. recycling vs. breeding capability

- Fuel cycle can be closed for: **Burner, Convertor, and Breeder**.
- Recycling does **not make sense** for **B&B** reactor.
  However, the fuel can be recycled and used in other reactor.
- **Recycling** is the ultimate **waste reducing** option.
- **Recycling in a breeder** => **highest** resources **utilization.**
  It is limited by reprocessing losses.
- **Recycling in a convertor** => **medium** resources **utilization.**
  It is limited by enrichment process and reprocessing losses.
- **Recycling in a burner** => mainly **waste minimization.**
- **Fuel recycling / long irradiation** leads to:

equilibrium fuel composition.

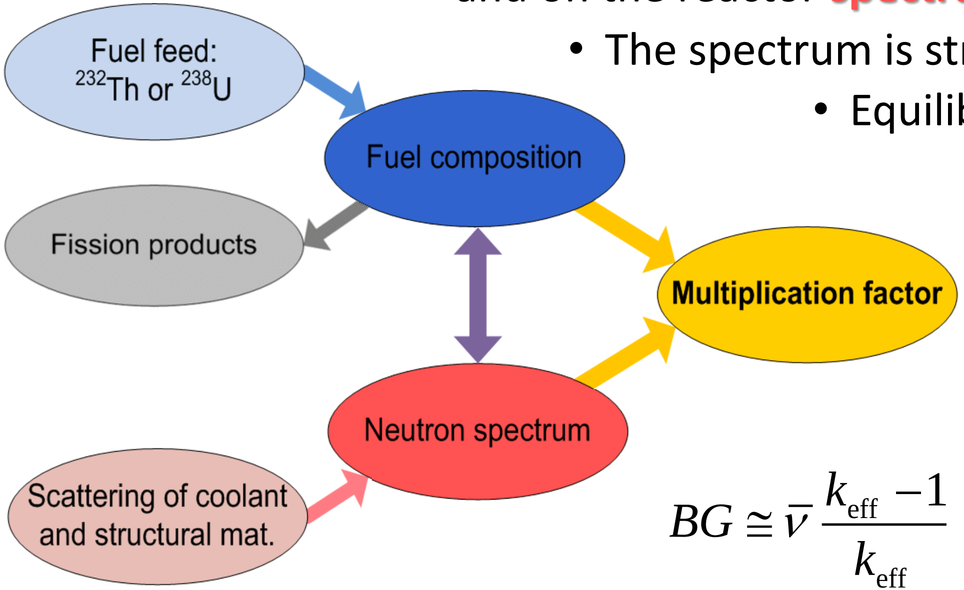




Krepel, J., et al., Fuel cycle sustainability of molten salt reactor concepts in comparison with other selected reactors. PHYTRA4 conference, 2018.

Equilibrium fuel composition

- When fuel cycle **parameters**: power, reprocessing scheme, feed composition, etc. are **fixed**, reactor will converge to equilibrium state / **fuel composition**.
  - The composition depends on **feed** type ^{238}U or ^{232}Th
    - and on the reactor **spectrum**.
      - The spectrum is strongly determined by **scattering materials**.
        - Equilibrium composition and spectrum determine the **multiplication factor**.



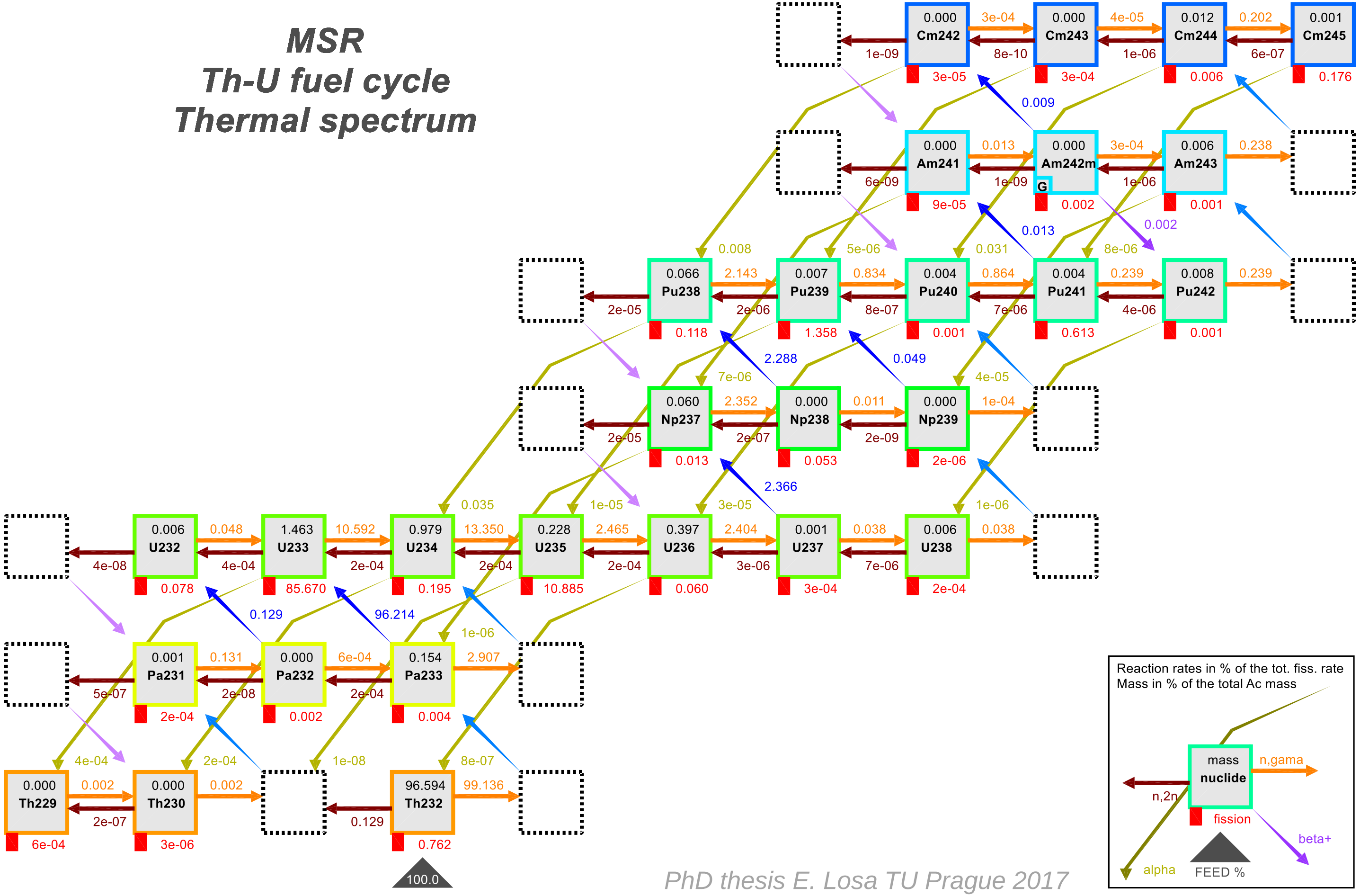
$$BG \simeq \bar{\nu}\,\frac{k_{\mathrm{eff}}-1}{k_{\mathrm{eff}}}$$

- Equilibrium is **inherent core state.** *(Bateman's matrix eigenstate: composition, spectrum, reactivity)*
- Equilibrium **reactivity** indicates neutron **efficiency** of the core and its **capability for breeding**.

Křepel, J., Losa, E., 2019. Closed U-Pu and Th-U cycle in sixteen selected reactors: evaluation of major equilibrium features. Ann. Nucl. Energy.

Equilibrium fuel composition (= irradiation chain)

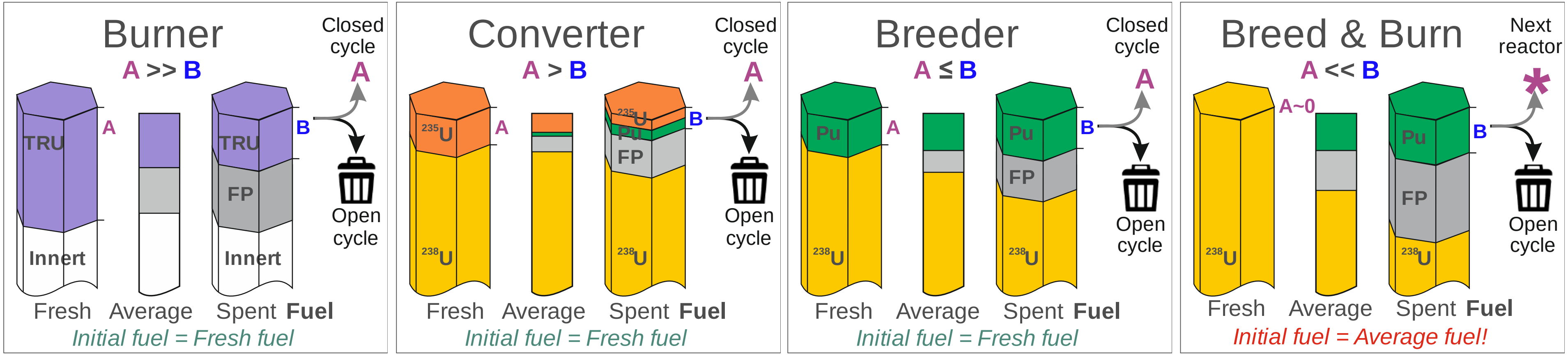
MSR
Th-U fuel cycle
Thermal spectrum



PhD thesis E. Losa TU Prague 2017

Reactor classification by breeding capability

Neutron economy



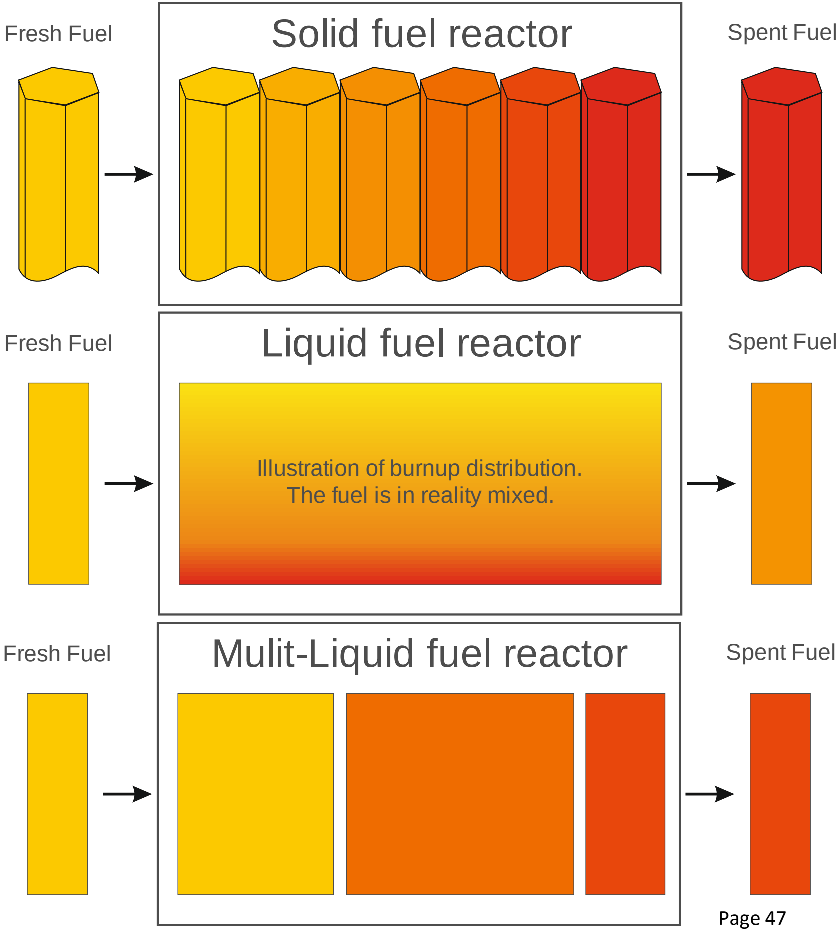
- Burner is typically used for synthetic Ac and excludes fertile isotopes as ^{238}U or ^{232}Th.
- Convertor, e. g. PWR or DMSR, is usually operates in open fuel cycle and burns ^{235}U.
- Breeder profit from neutronics advantages only in the closed cycle.
  For Iso-breeding (EU) or Break-even (US) reactor => A=B.
- Extreme breeder can be operated in Breed & Burn mode.
  It can have high fuel utilization even without reprocessing.

Breed & Burn (B&B): special open cycle mode

- Initially the fissile fuel will be **bred** and firstly later it will be **burned**.
- The B&B cycle in **liquid fuel** reactor substantially differs from **solid fuel.**

  Discharged fuel:
  Most burned in solid fuel case
  Average burned in liquid fuel case.

- To increase the burnup and reduce the core size (**single-fluid** layout can be bulky), **multi-fluid** layout can be used.



Hombourger, B., Křepel, J., Mikityuk, K., Pautz, A., 2017. On the Feasibility of Breed-and-Burn Fuel Cycles in Molten Salt Reactors, in: Proceedings of FR17. Yekaterinburg, Russian Federation.

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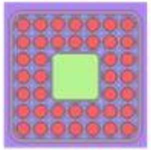
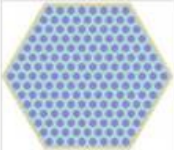
Equilibrium cycle comparison for 16 reactors

- **Assumptions:**
  1) infinite lattice.
  2) neglected fission products.
  3) design as is, no additional optimization.
  4) ENDF/B-VII.0
- 8 thermal and
- 8 fast reactors were compared in both U-Pu and Th-U cycles.

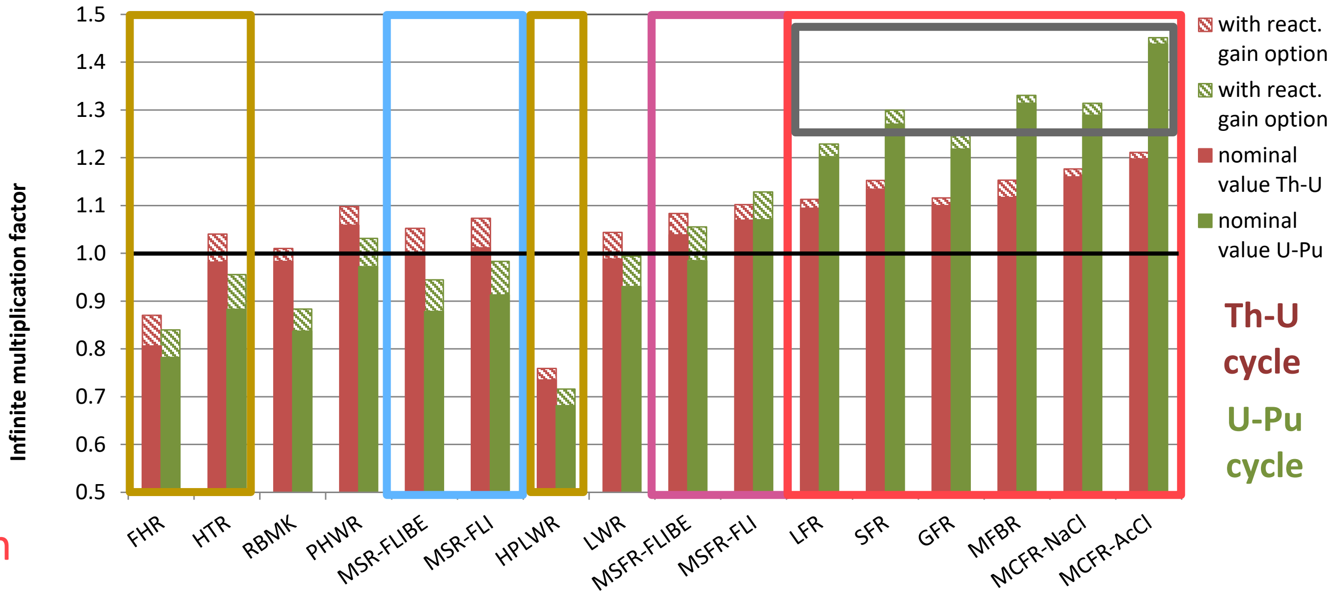
Solid fuel thermal reactors			
Reactor name (and label)	**Lattice geometry**	**Name and short name**	**Lattice geometry**
Fluoride high temperature reactor PB-AHTR (**FHR**)		Pressurized heavy water reactor (**PHWR**)	
High temperature reactor (**HTR**)		High performance light water reactor (**HPLWR**)	
Reaktor balshoi moshnosti kanalnyj (**RBMK**)		Light water reactor VVER-1000 (**LWR**)	
Liquid fuel thermal reactors			
Thermal MSR fueled by LiF-BeF$_2$-AcF$_4$ (**MSR-FLIBE**)		Thermal MSR fueled by LiF-AcF$_4$ (**MSR-FLI**)	

Solid fuel fast reactors			
Reactor name (and label)	**Lattice geometry**	**Name and short name**	**Lattice geometry**
European lead system (**LFR**)		Gas cooled fast reactor (**GFR**)	
European sodium fast reactor (**SFR**)		Metal fueled fast breeder reactor (**MFBR**)	
Liquid fuel fast reactors			
Molten salt fast reactor fueled by LiF-BeF$_2$-AcF$_4$ (**MSFR-FLIBE**)		Molten salt fast reactor fueled by NaCl-AcCl$_4$ (**MCFR-NaCl**)	
Molten salt fast reactor fueled by LiF-AcF$_4$ (**MSFR-FLI**)		Molten salt fast reactor fueled by AcCl$_4$ (**MCFR-AcCl**)	

Křepel, J., Losa, E., 2018. Closed U-Pu and Th-U cycle in sixteen selected reactors: evaluation of major equilibrium features. Submitted to Ann. Nucl. Energy.

Equilibrium multiplication factor (reactivity)

- Better performance: **Th-U in thermal** and **U-Pu in fast** spectra.
- Graphite moderated MSR only in Th-U.
- Fluorides fast MSFR possible in both cycles *(equal performance almost epithermal).*
- GenIV fast reactors can all breed in closed cycle.
- B&B possible for U-Pu.
- SCWR, FHR, HTR???



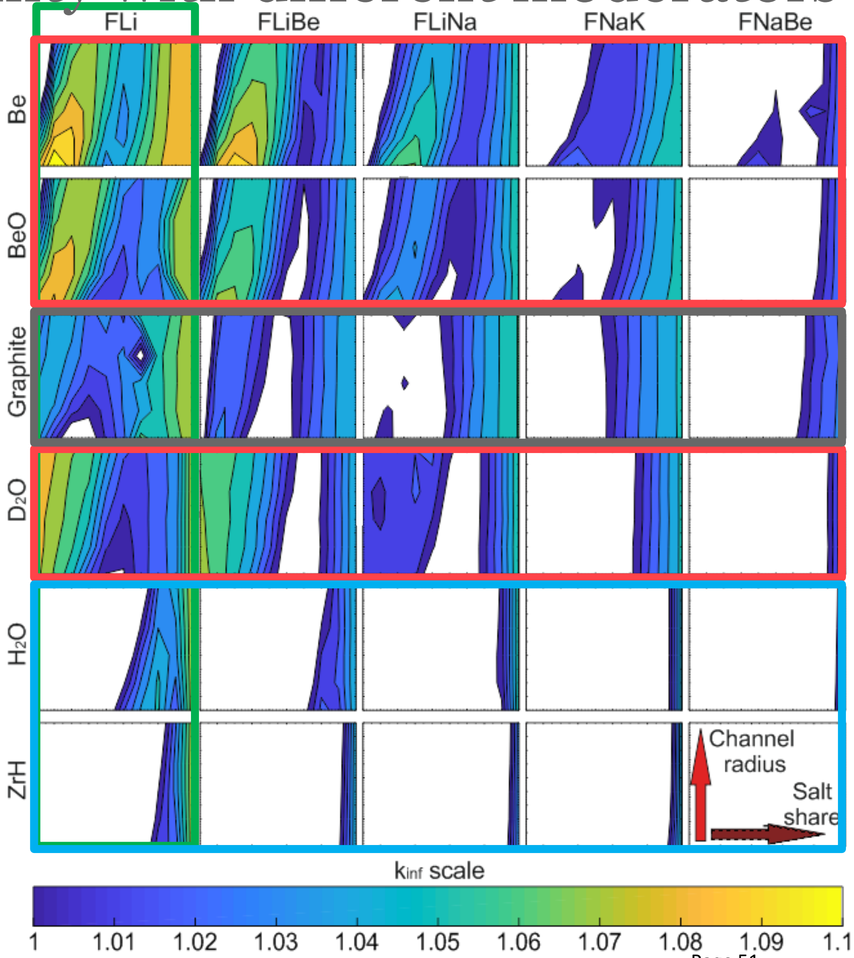
Breeding capability of moderated MSRs

For comparison with other reactors refer to:

Krepel, J., and Losa, E., Self-sustaining breeding in advanced reactors: Characterization of selected reactors, Encyclopedia of Nuclear Energy, (Greenspan, E., Ed.), Elsevier, 2021.

Th-U breeding capability with different moderators

- **5 fluoride salts** were analyzed with **6** selected **moderators**.
- Equilibrium $\boldsymbol{k_{inf}}$ is presented as a function of **salt share** and **channel radius**.
- **FLi** salt is neutronically the best.
- Good results for **Be, BeO, and D_2O**; however, they are not compatible with the salt without cladding (SiC..?).
- Hydrogen based moderators **ZrH and H_2O** not applicable for closed cycle.
- **Graphite** is not the best moderator, but the only one directly compatible with salt.



Hombourger, B.A., 2018. Ph.D. Thesis. EPFL Lausanne, Switzerland.

Neutronics impact of cladding (7LiF salt example)

- **LiF salt** combined with **Be** and **D_2O** moderators was selected to analyze the impact of cladding:

  Hastelloy,

  SS316,

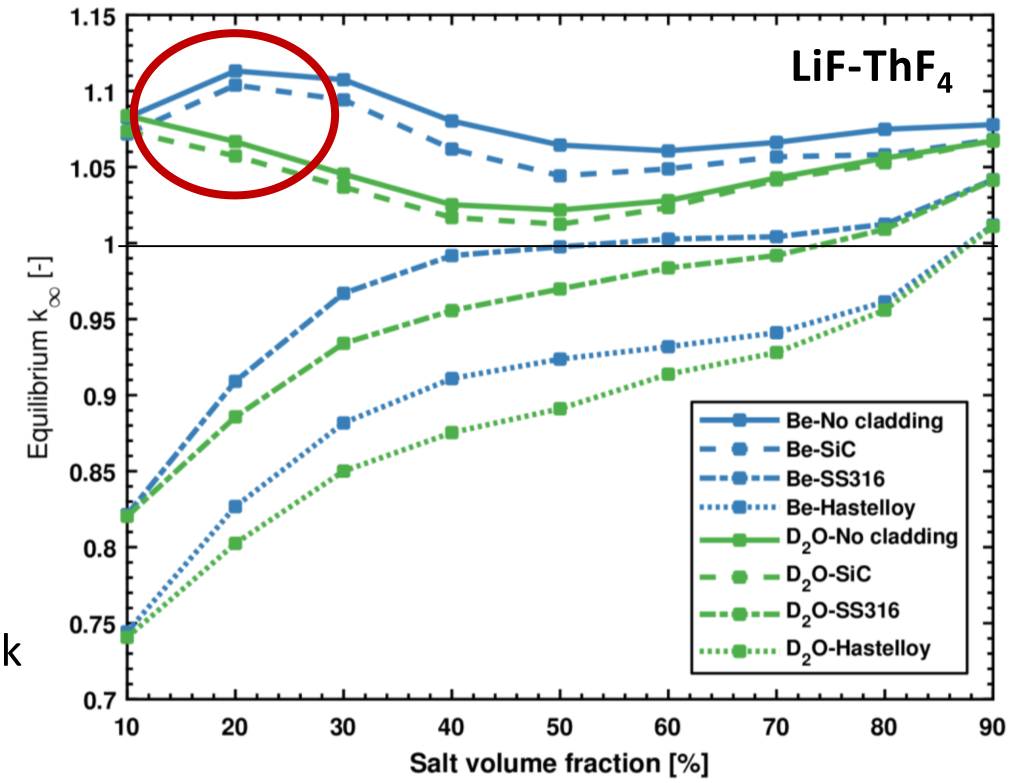
  and SiC.

- Only **SiC** seems to have acceptable low parasitic neutron capture.

- Purely from neutronics perspective **H**eavy **W**ater **B**oiling MSR would work

  HWB-MSR ☺

  Thesis topic for super hero…?



Hombourger, B.A., 2018. Ph.D. Thesis. EPFL Lausanne, Switzerland.

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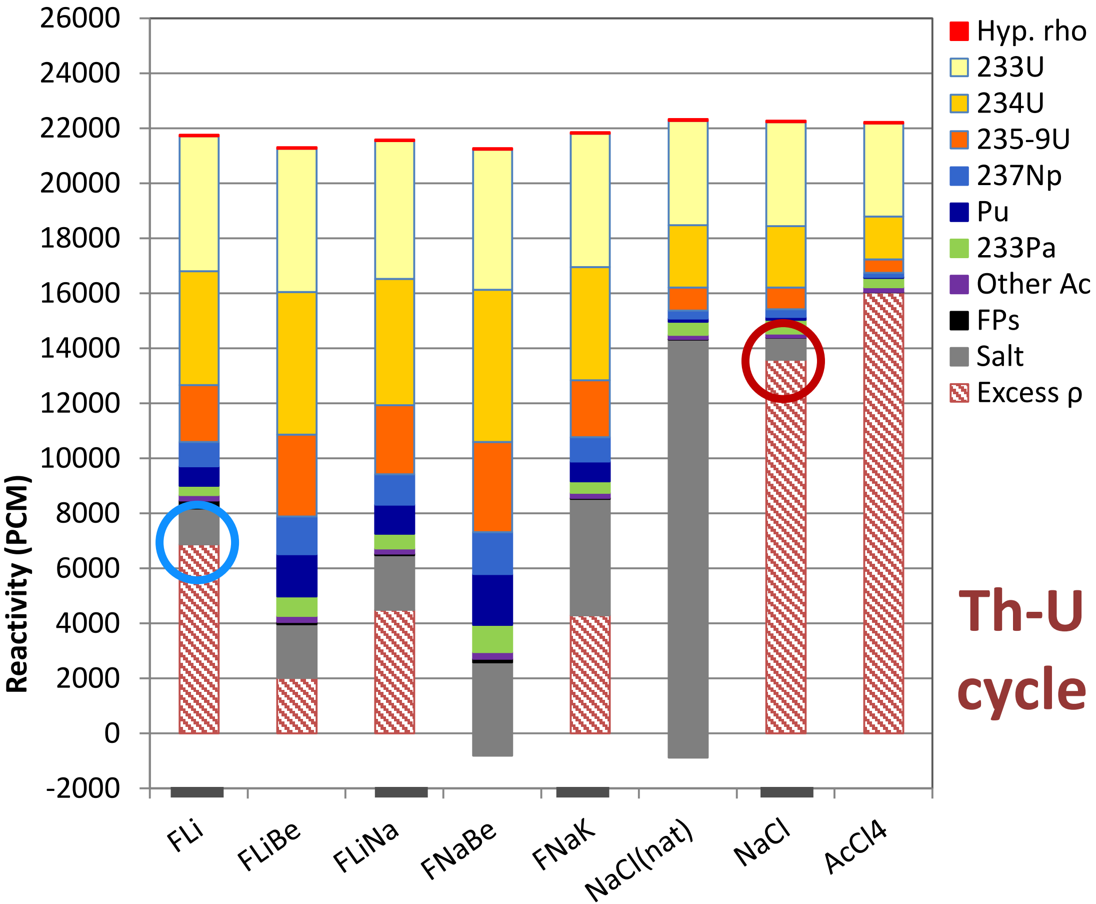
Breeding capability of homogeneous fast MSRs

Th-U cycle performance without moderator

- **8 salts** were evaluated:
  FLi, FLiBe, FLiNa, FNaBe, FNaK, NaCl (nat), $Na^{37}Cl$, $Ac^{37}Cl$.
  32% $AcCl_3$ 32% $AcCl_3$ 100% $AcCl_4$
- **4** options in **Th-U** (reasonable melting point and reactivity): **FLi, FLiNa, FNaK, $Na^{37}Cl$.**
- **$Na^{37}Cl$** provides the highest excess in Th-U of 13000 pcm.
- **FLi** is best fluoride salt with 6000 pcm.

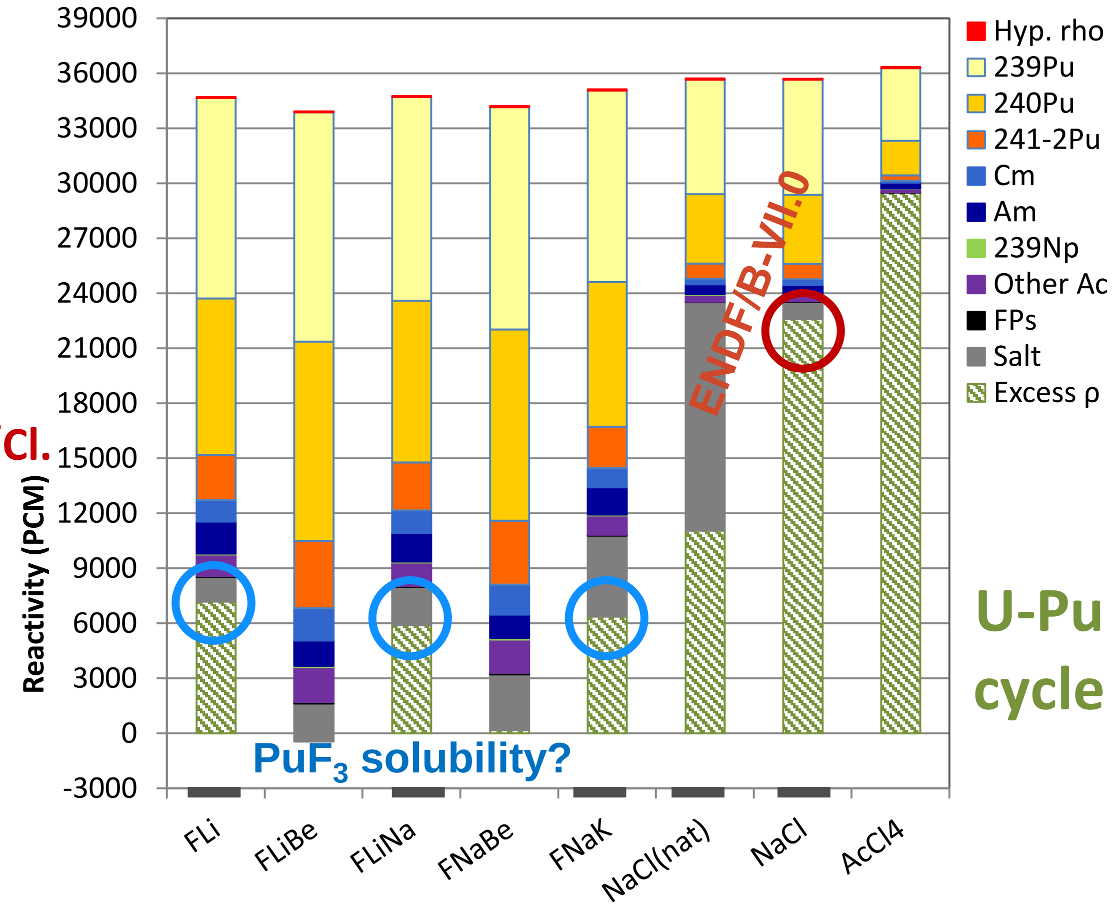


Th-U cycle

IAEA Technical Report Series, Status of Molten Salt Reactor Technology, document in preparation, International Atomic Energy Agency, 2021.

U-Pu cycle performance without moderator

- **8 salts** were evaluated:
  FLi, FLiBe, FLiNa, FNaBe, FNaK, NaCl (nat), $Na^{37}Cl$, $Ac^{37}Cl$.
  32% $AcCl_3$ 32% $AcCl_3$ 100% $AcCl_3$
- **5** options in **U-Pu** (reasonable melting point and reactivity):
  Fli, FLiNa, FNaK, NaCl (nat), $Na^{37}Cl$.
- **$Na^{37}Cl$** provides the highest overall excess of 22000 pcm.
- **FLi, FLiNa, FNaK** have similar performance of ~6000pcm, PuF_3 solubility is the major limiting issue.



U-Pu cycle

Self-sustaining breeder core size estimate

Core radius estimate in Th-U cycle

- Bare core criticality line.

$$k_{inf} = 1 + M^2B^2$$

Derived from Fermi theory of bare "thermal" reactor:

$$k_{eff} = k_{\inf} p_1 p_2 = k_{\inf} \frac{e^{-\tau B^2}}{1+L^2B^2} \cong k_{\inf} \frac{1}{1+\left(\tau + L^2\right)B^2} = k_{\inf} \frac{1}{1+M^2B^2}$$

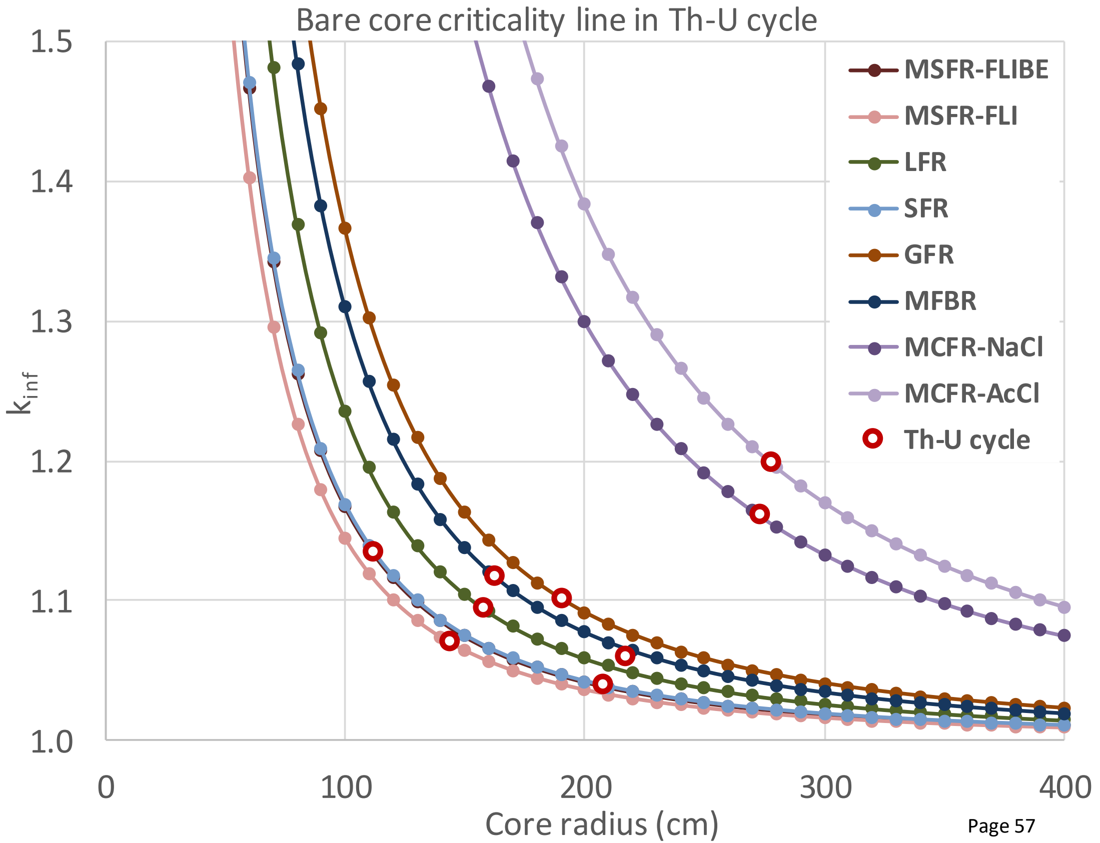
- Buckling for a cylinder:

$$B^2 = \left(\frac{\pi}{h}\right)^2 + \left(\frac{2.405}{r}\right)^2$$

- Minimal volume for given B^2:

$$\frac{h}{r} = \frac{\sqrt{2}\pi}{2.405} \cong 1.85$$

- **Core radius estimate in Th-U cycle =>**



Core radius estimate in U-Pu cycle

- Bare core criticality line.

$$k_{inf} = 1 + M^2B^2$$

Derived from Fermi theory of bare "thermal" reactor:

$$k_{eff} = k_{\text{inf}}\, p_1 p_2 = k_{\text{inf}} \frac{e^{-\tau B^2}}{1+L^2B^2} \cong k_{\text{inf}} \frac{1}{1+\left(\tau + L^2\right)B^2} = k_{\text{inf}} \frac{1}{1+M^2B^2}$$

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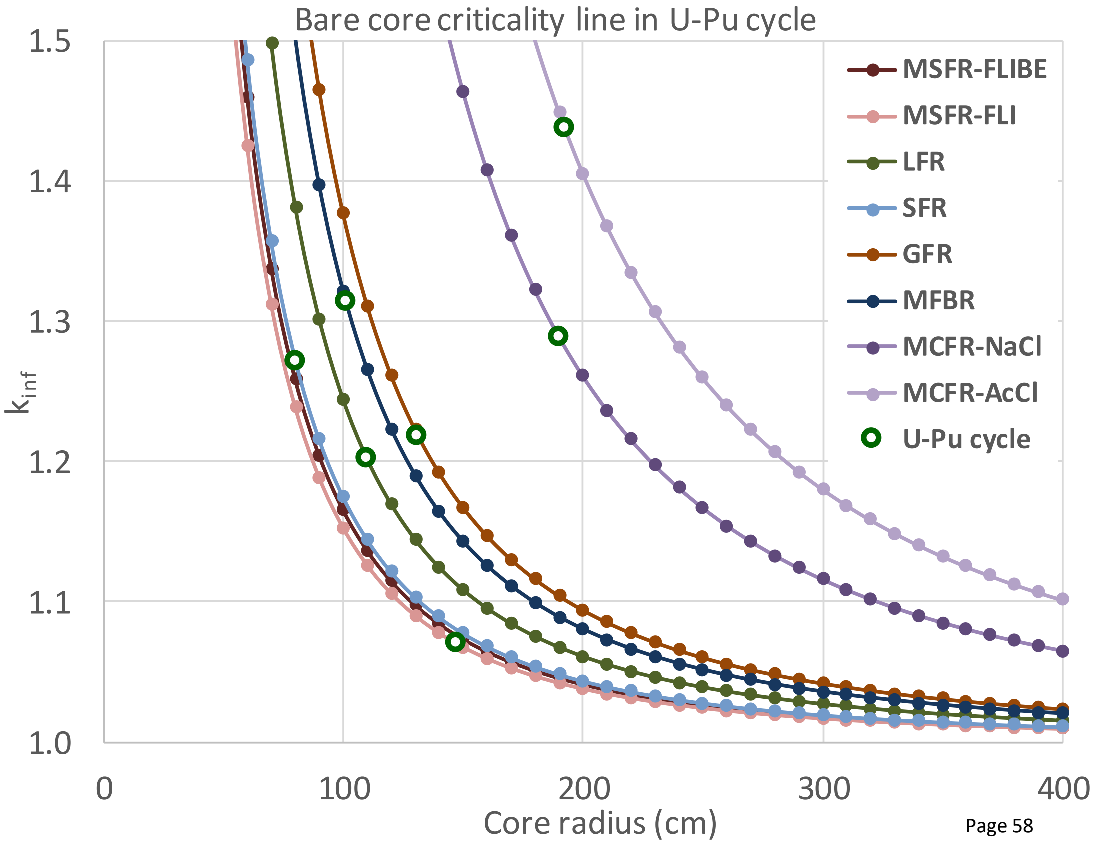
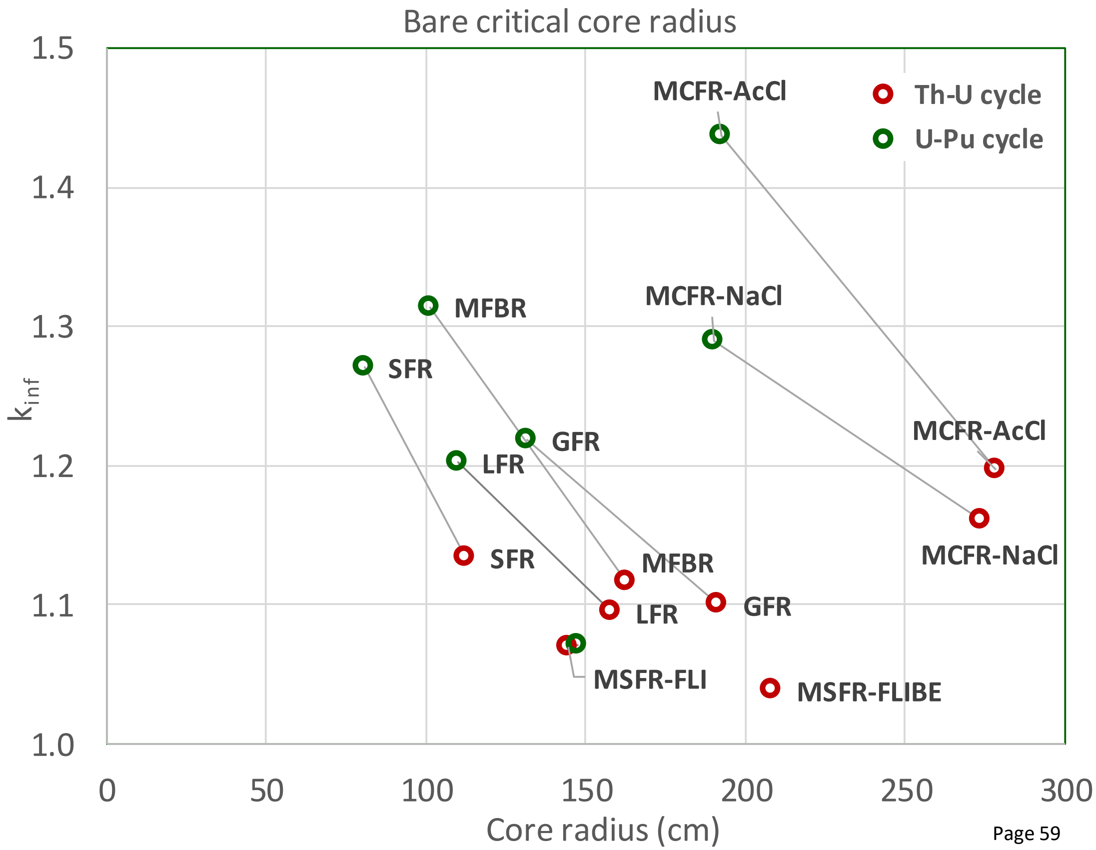
- **Core radius estimate in U-Pu cycle =>**



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

- 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-FLI** is the **smallest MSR** core and it has the same core size for both cycles. (**very soft fast spectrum**)



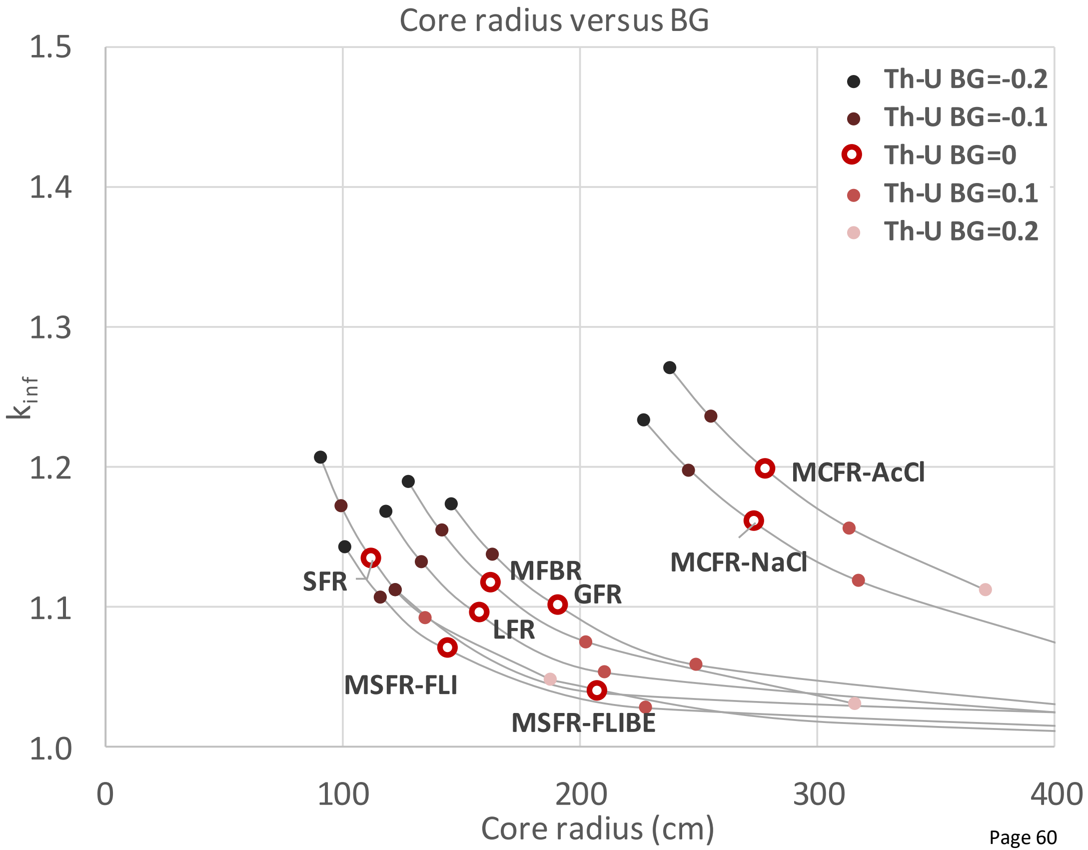
Core radius estimate in Th-U cycle

- Combining these two equations:

$$k_{eff} \cong k_{\inf} \frac{1}{1+M^2B^2}$$

$$BG_{per} \cong \bar{\nu} \frac{k_{\mathrm{eff}}-1}{k_{\mathrm{eff}}}$$

- Bare core size can be estimated for several BG values.
  Th-U cycle =>



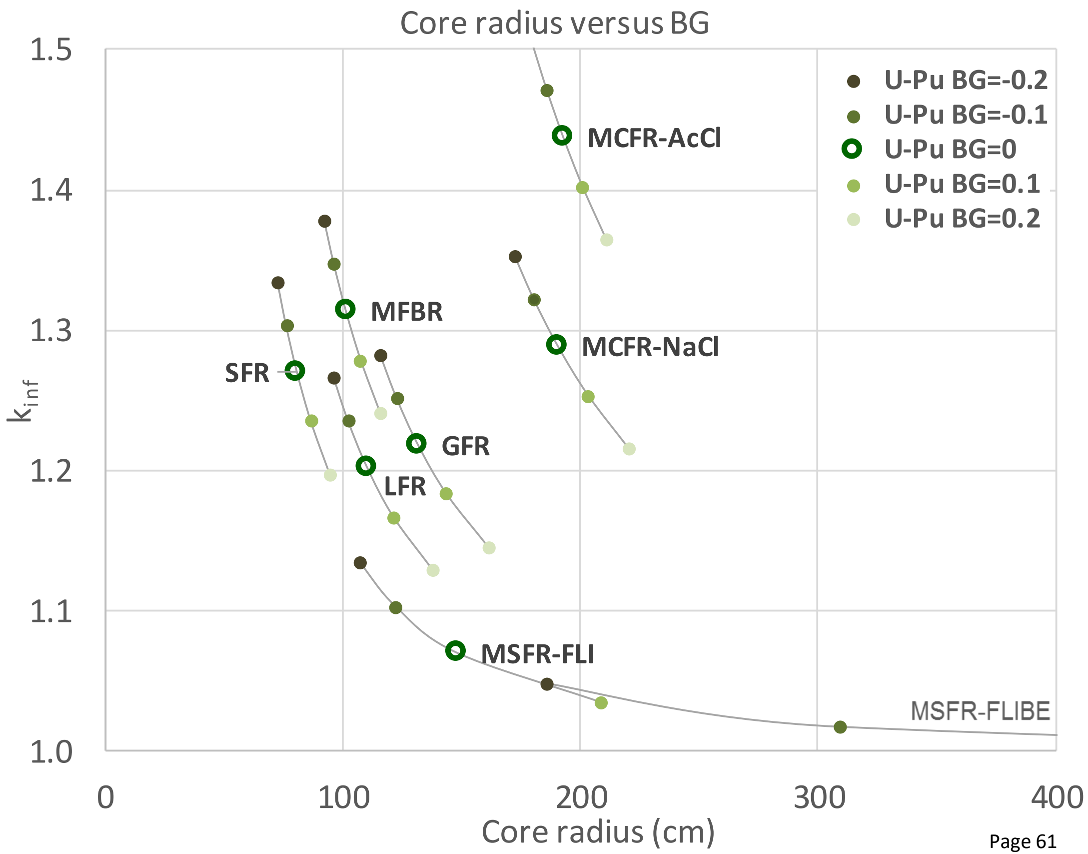
Core radius estimate in U-Pu cycle

- Combining these two equations:

$$k_{eff} \cong k_{\inf} \frac{1}{1+M^2B^2}$$

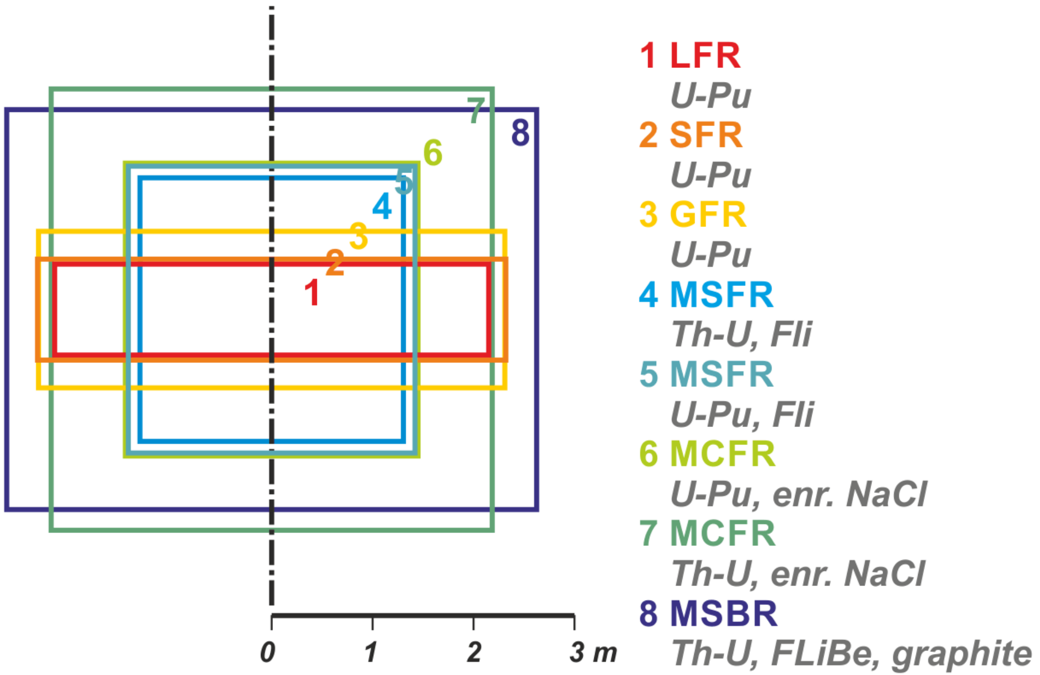
$$BG_{per} \cong \bar{\nu} \frac{k_{\mathrm{eff}}-1}{k_{\mathrm{eff}}}$$

- Bare core size can be estimated for several BG values.
  U-Pu cycle =>



Self-sustaining breeder in closed cycle

- Using 1m **hastelloy reflector** core size was estimated for **single-fluid** designs.
- It was compared with classical fast reactors.
- MSFR (**Fli**) in Th-U (4) is compact.
- MSFR (**Fli**) in U-Pu (5) is bigger.
- MCFR (**Na^{37}Cl**) in U-Pu (6) is comparable to MSFR in U-Pu (5).
- MCFR (**Na^{37}Cl**) in Th-U (7) is big.
- MSBR (ORNL design, 13% salt).



Hombourger, B.A., 2018. Ph.D. Thesis. EPFL Lausanne, Switzerland.

Transition to closed Th-U and U-Pu cycle

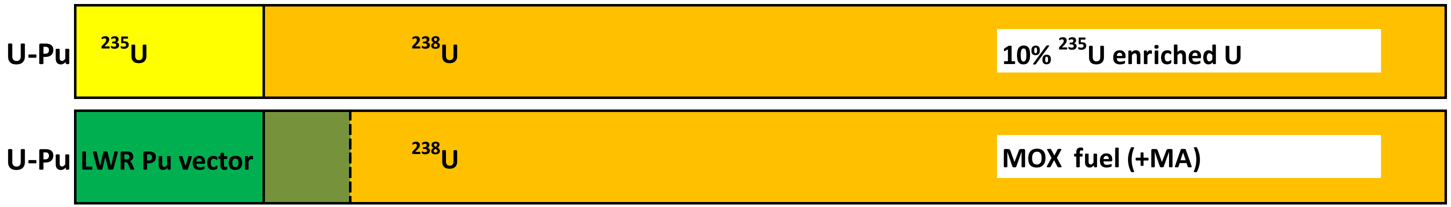
5 major fissile materials to start the Th-U cycle

Material	RG_Pu	LEU	~~HEU~~	U233	~~WG_Pu~~
Fissile isotope(s)	^{239}Pu, ^{241}Pu	^{235}U	^{235}U	^{233}U	^{239}Pu
Fissile isotope share	~60%	1-20%	21-95%	60-100%	>93%
“Availability”	medium	high	high	low	medium
Proliferation risk	medium	medium	high	high	high

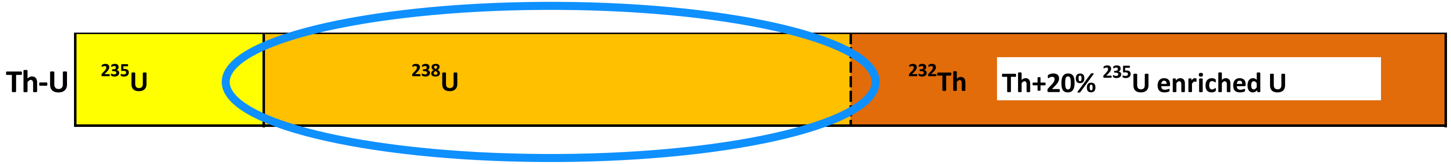
RG_Pu and LEU as initial fuel load

- Both **RG_Pu** and **LEU** are very natural option to start the **U-Pu** cycle.

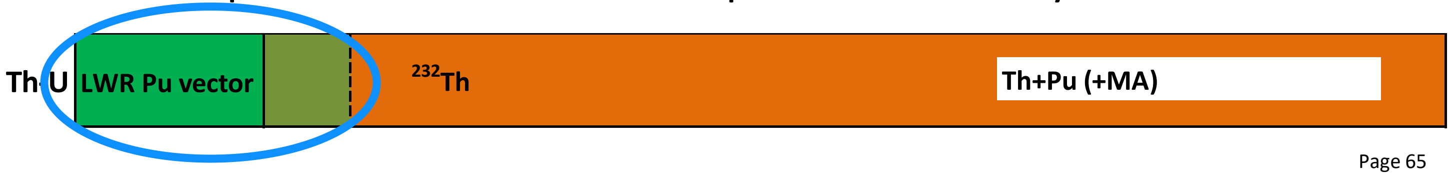
Fuel composition - initial cycles (10% ^{235}U equivalent)



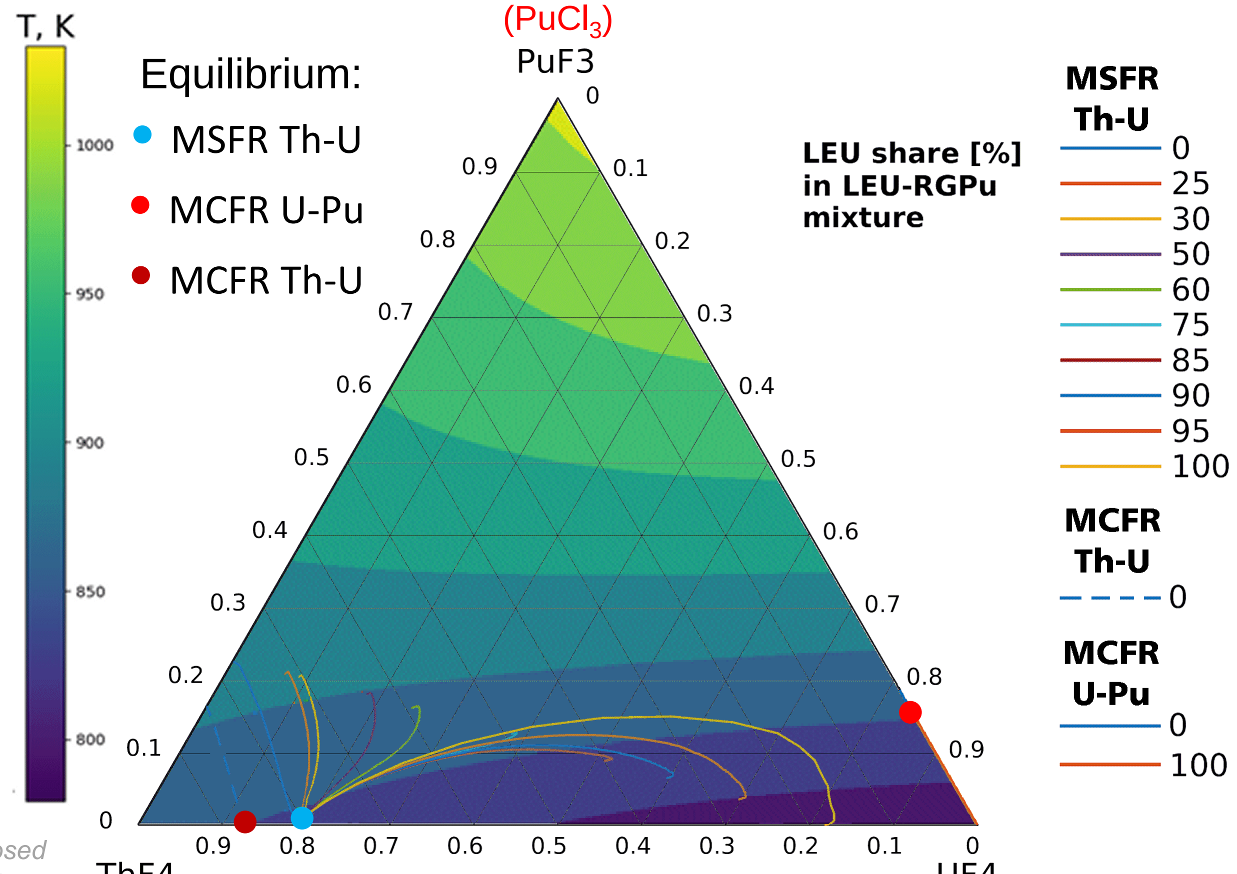
- Starting **Th-U** cycle with **LEU** induces **^{238}U presence** in the core.



- Starting **Th-U** cycle with **RG_Pu, LEU** or their mixture introduces strong **perturbation**.
- **Pu and $^{235 \& 238}U$** are not presented in the salt at equilibrium Th-U cycle.



Transition to Th-U cycle in MSFR (and U-Pu MCFR)



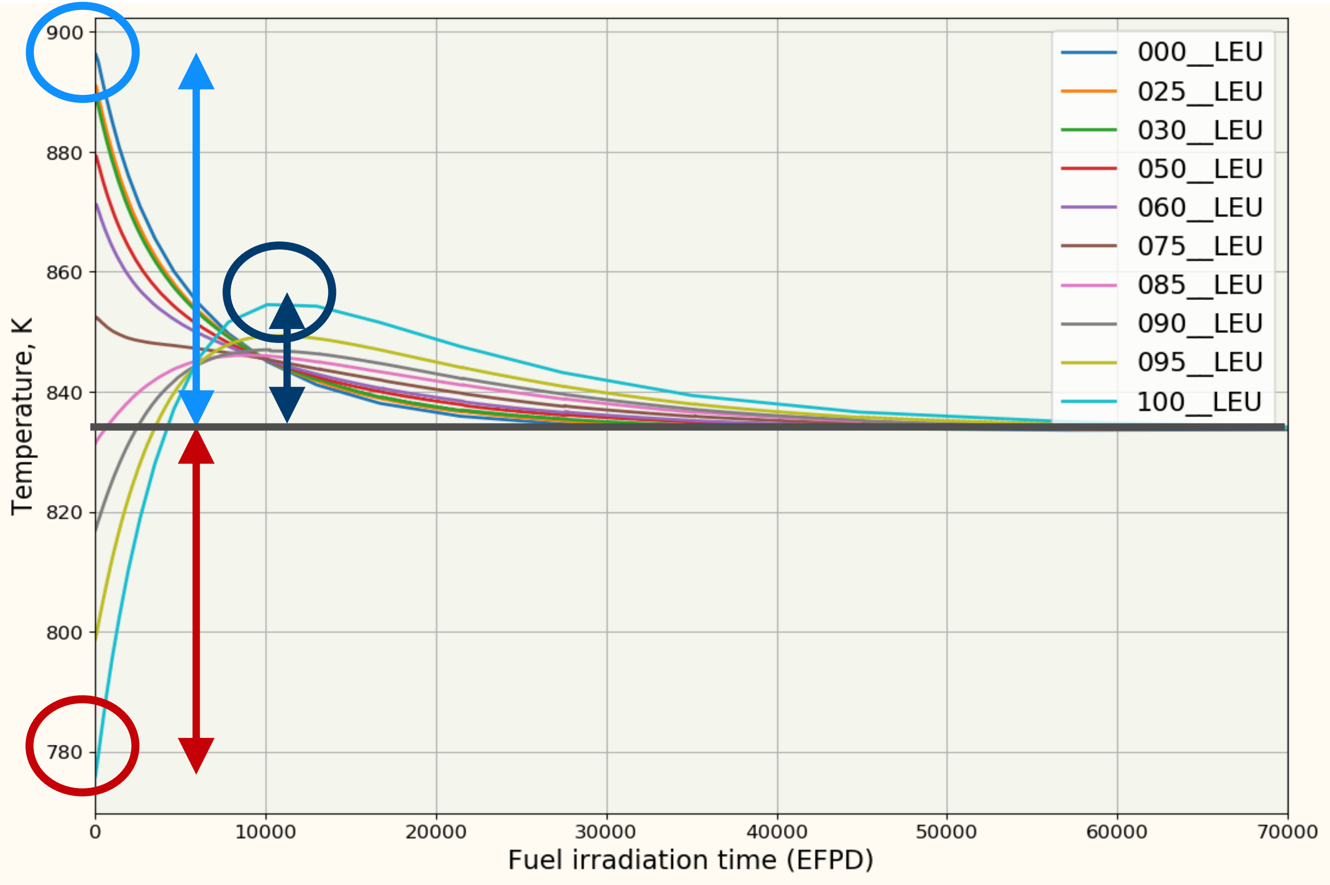
Liquidus temperature

Křepel, J. et al., Transition to closed Th-U fuel cycle in fluoride salts based fast MSR. ICAPP 2019.

Evolution of the liquidus temperature

- **PuF_3** presence increases the liquidus temperature.
- **UF_4** presence decreases the liquidus temperature.
- The temperature for equilibrium Th-U cycle is cca **835K.**
- The difference is:
  up to **-55K** and **+60K** or **+20K**.

LiF-ThF4-UF4-PuF3 (at **76.6%** LiF).



Křepel, J. et al., Transition to closed Th-U fuel cycle in fluoride salts based fast MSR. ICAPP 2019.

Self-sustaining breeding in open cycle (B&B)

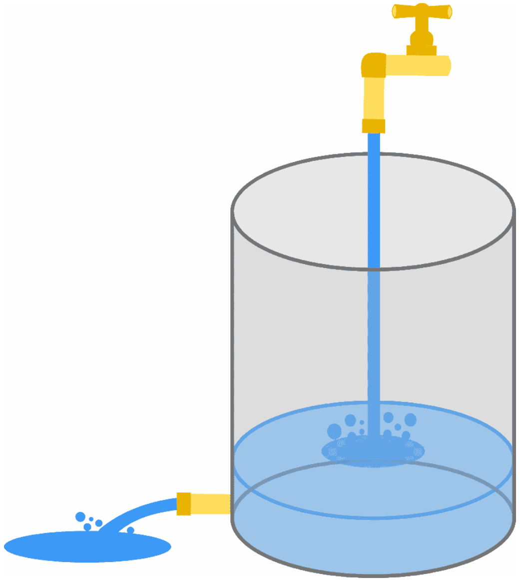


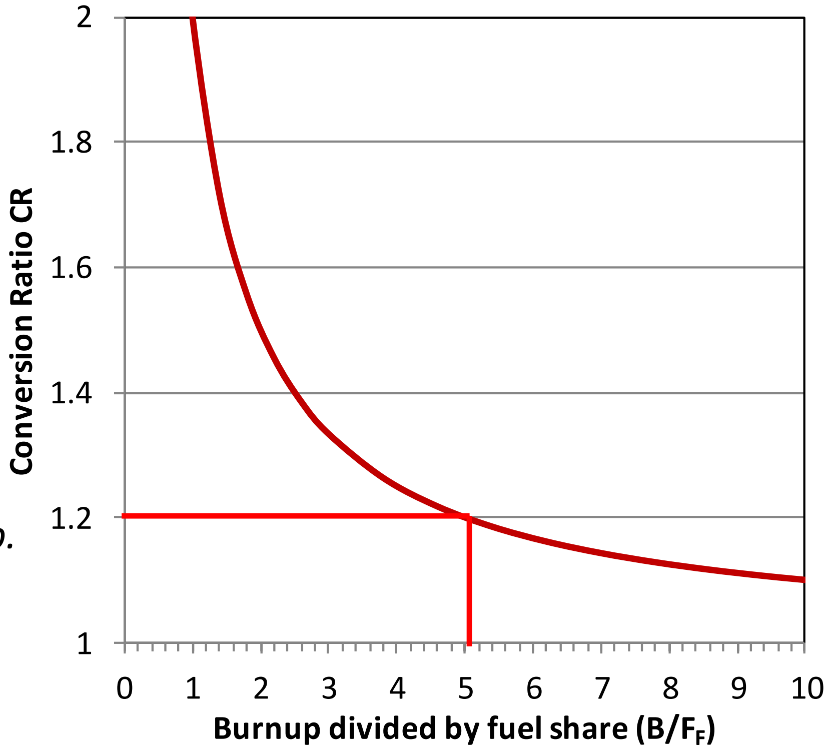
Illustration of tap-like reactor

https://www.subpng.com/

Trivial criteria for breed-and-burn cycle operation

- In B&B cycle conditions:
  1) fresh fuel is only fertile material
  2) spent fuel is not recycled.
- B&B trivial criterion (tautology): **I = II**
  I: Fissile Fuel $\mathbf{F_F}$ share in the discharged fuel.
  II: New fissile fuel bred in the discharged fuel.
- $$F_F = B(CR-1) \Rightarrow \frac{1}{CR-1} = \frac{B}{F_F}$$

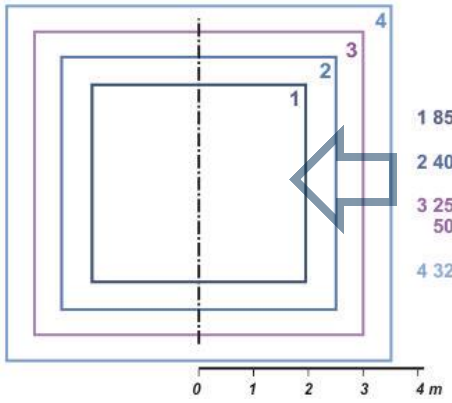
  where CR is conversion ratio and B is the fuel burnup.
- Reactor must be critical for CR, F_F, and B,
  e.g. for CR=1.2: F_F=10% ⇔ B=50% (1% ⇔ 5%)
- Fuel utilization in B&B cycle?
  It is equal to the burnup.

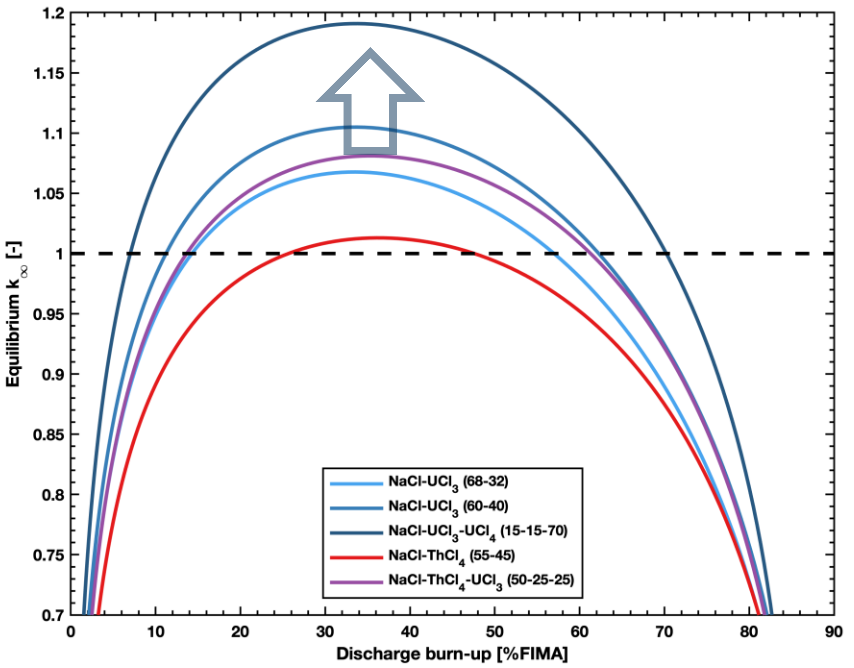


J. Krepel, B. Hombourger, E. Losa, Fuel cycle sustainability of Molten Salt Reactor concepts in comparison with other selected reactors, PHYTRA4, Marrakech, Morocco, September 17-19, 2018.

Self-sustaining breeder in open cycle (B&B)

- B&B is practically not possible in Th-U cycle.
- It is only possible in mixed U-Pu & Th-U cycle.
- B&B cores are bulky (chlorides = hard spectrum, but also high Migration area).
- The performance increases with growing actinides share in the core.

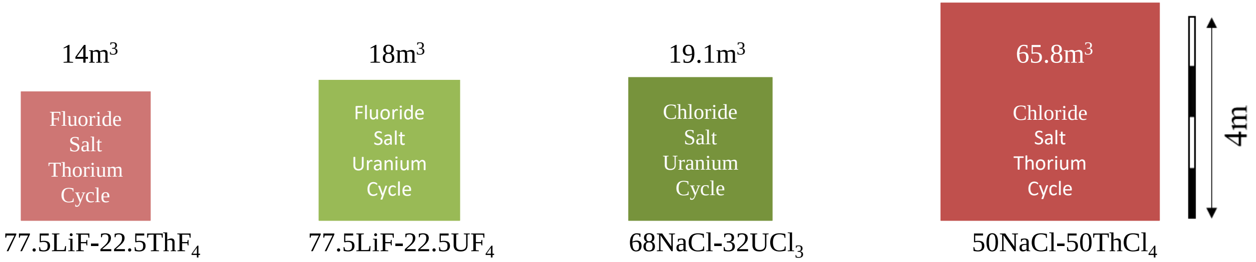




Hombourger, B. et al., 2019. Breed-and-Burn Fuel Cycle in Molten Salt Reactors. Submitted to special MSR edition of The European Physical Journal

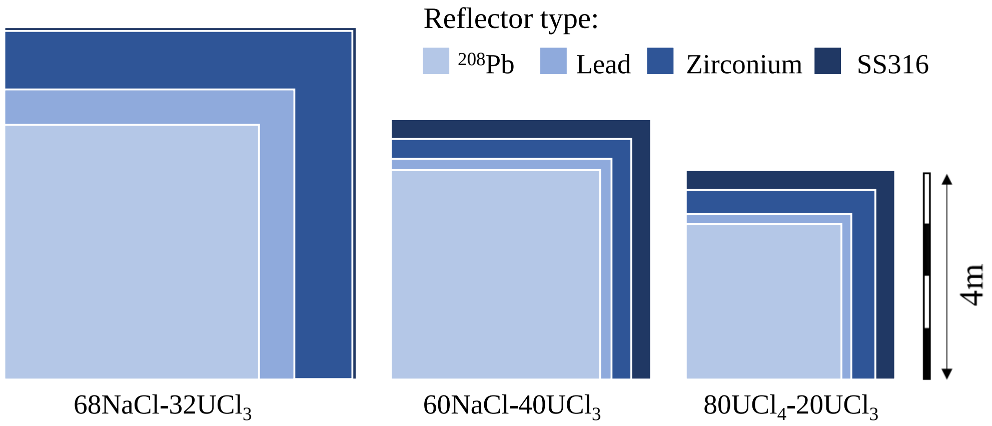
Core size comparison for closed and open B&B cycle

- Closed cycle



- Breed and burn open cycle

Critical core sizes



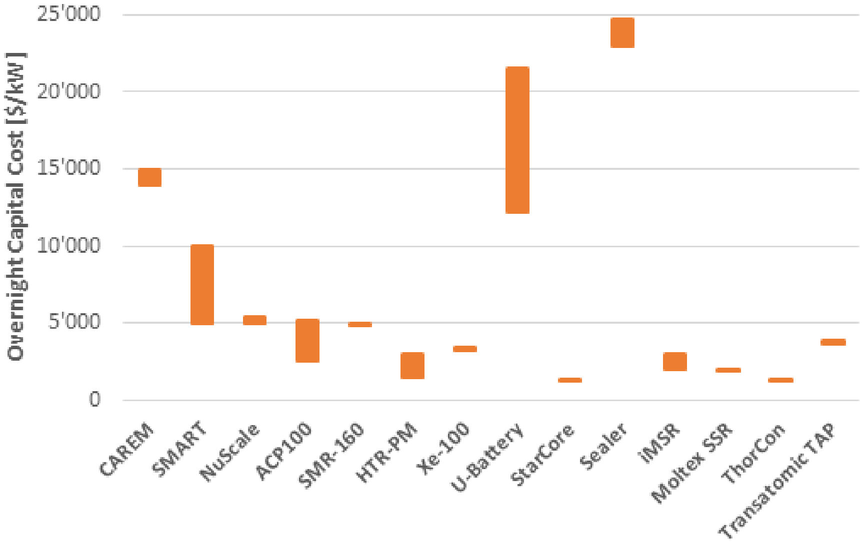
Hombourger, B.A., 2018. Ph.D. Thesis. EPFL Lausanne, Switzerland.

Summary – fuel cycle

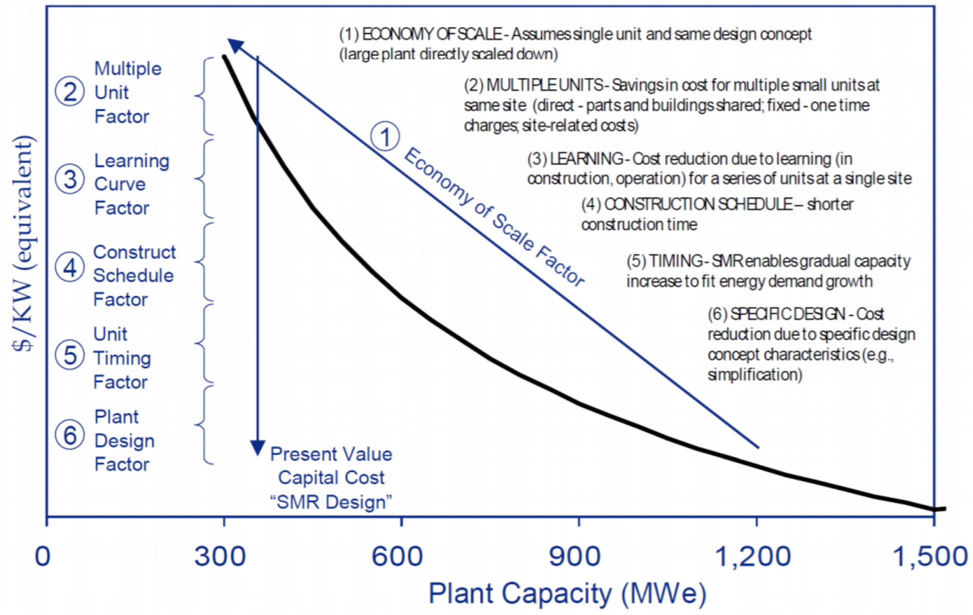
- MSR can be designed as very compact burner fueled by minor actinides or enriched uranium.
- MSR can be designed as a Th-U or U-Pu breeder.
- Fluoride salts are more convenient for Th-U cycle (thermal or fast).
- Chloride salts are more convenient for U-Th cycle (fast).
- Chloride salts are applicable for U-Pu breed-and-burn cycle (fast).
- The core size may be relatively large.
- The low-pressure vessel can be still manufactured in factory as a module.

Some general comments on size and modularity

Thesis of P. Dobrzynski, Estimating the Cost of Small Modular Reactors, 2017.

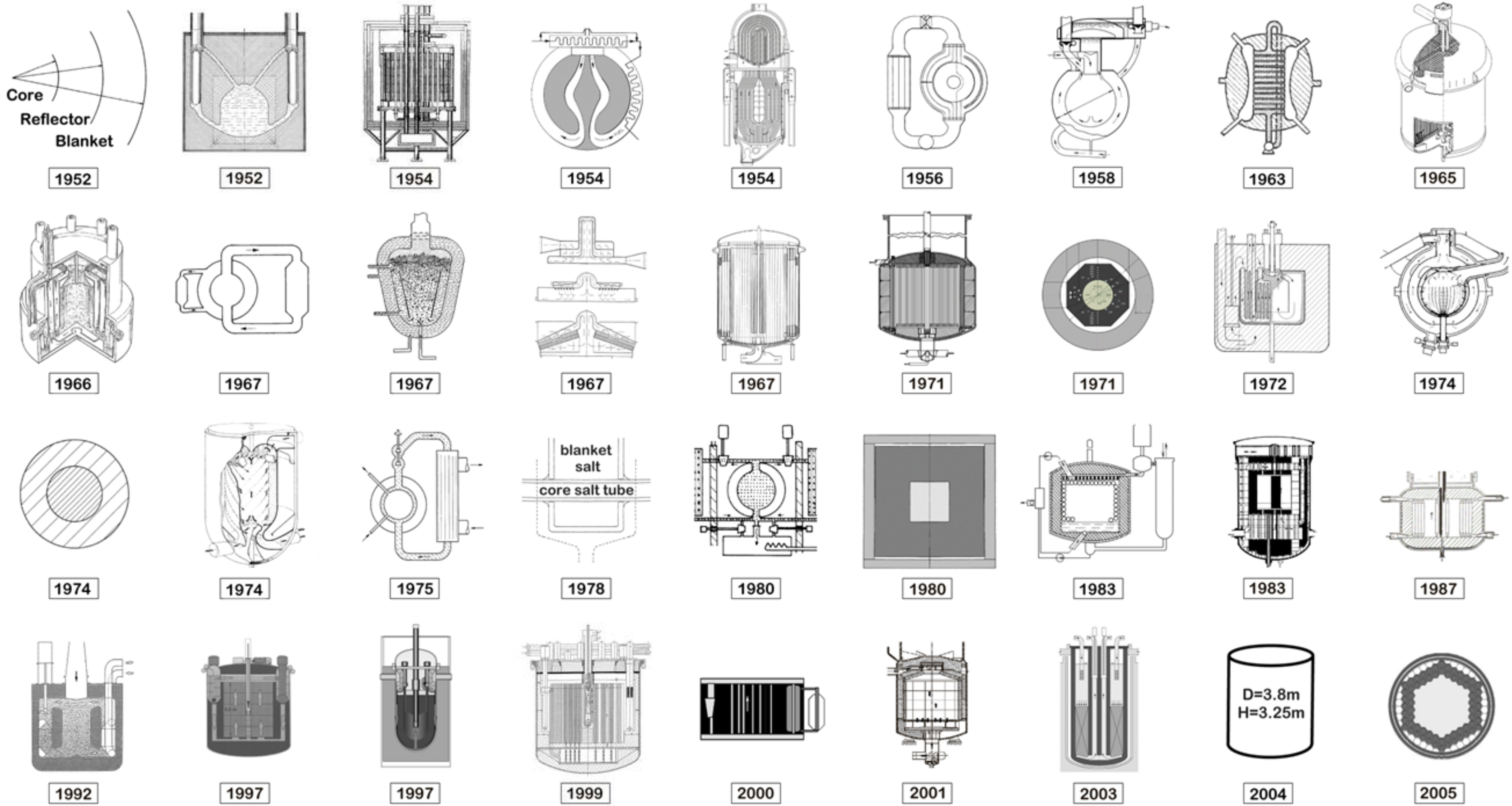


Available vendor estimates of the selected SMRs



Influence of SMR's features on the unit investment cost (Carelli, et al., 2008)

Recall of the variety



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General conclusion

- MSR is not a reactor, it is entire category of reactors.
- MSR has potential to be safe, sustainable and economic at once.
- MSR is not yet safe, sustainable and economic…
- We should firstly utilize the potential.
- Safety evaluation would be premature for many concepts.
- Economy was also not evaluated, but prototypes would be expensive.
- Sustainability, or actually breeding capability, was evaluated.
- MSR can breed in both open and closed cycle.
- MSR might be designed as modular reactor.
- Preferably Big Modular Reactor (BMR).

Wir schaffen Wissen – heute für morgen

Thank you.
Questions?

List of accomplished MSc and PhD theses

- Listed are also several theses accomplished in **cooperation** with other institutes or by visiting students from **POLIMI** and **CTU Prague**.

Name	Title	Thesis type / Status	Year
B. Hombourger	Parametric Lattice Study for Conception of a Molten Salt Reactor in Closed Thorium Fuel Cycle	Swiss nuclear MSc course / Defended	2013
C. Fiorina	The Molten Salt Fast Reactor as a Fast-Spectrum Candidate for Thorium Implementation	PhD thesis in cooperation with PSI / Defended at POLIMI	2013
V. Ariu	Heat exchanger analysis for innovative molten salt fast reactor	Swiss MSc course / Defended	2014
M. Aufiero	Development of advanced simulation tools for circulating-fuel nuclear reactors	PhD thesis in cooperation with PSI / Defended at POLIMI	2014
H. Kim	Static and transient analysis of Molten Salt Reactor Experiment using SERPENT-2 / TRACE / PARCS codes	Swiss nuclear MSc course / Defended	2015
J. Choe	Empirical Decay Heat Correlations and Fission Products Behavior in MSRs	Swiss nuclear MSc course / Defended	2015
J. Bao	Development of the model for the multiphysics analysis of Molten Salt Reactor Experiment using GeN-Foam code	Swiss nuclear MSc course / Defended	2016
D. Pyron	Safety Analysis for the Licensing of Molten Salt Reactors	Swiss nuclear MSc course / Defended	2016
E. Pettersen	Coupled multi-physics simulations of the MSFR using TRACE-PARCS	MSc thesis in cooperation with PSI / Defended at Univ. Paris-Saclay	2016
N. Vozarova	Behaviour of fission products in the molten salt reactor fuel	Swiss MSc course in cooperation with JRC Karlsruhe / Defended at ETHZ	2016
M. Zanneti	Development of new tools for the analysis and simulation of circulating-fuel reactor power plants	PhD thesis in cooperation with PSI / Defended at POLIMI	2016
E. Losa	U-Pu and Th-U Fuel Cycle Closure	PhD thesis in cooperation with PSI / Defended at TU Prague	2017
B. Hombourger	Small modular Molten Salt Fast Reactor design for closed fuel cycle	PhD thesis in frame of SNSF project at PSI Defended at EPFL	2018
M. Di Filippo	Development of dedicated MSR burn-up tool	Swiss nuclear MSc course / Defended	2018
V. Sisl	Thorium Utilization in the Fuel Cycle of Advanced Nuclear Reactors	MSc thesis in cooperation with PSI / Defended at TU Prague	2018
T. Koivisto	Assessment of waste burning in open cycle of two fluids chloride MSR	MSc thesis in cooperation with PSI / defended at Aalto University in Finland	2019
V. Raffuzzi	Modelling of batch-wise operation of European Sodium Fast Reactor and Breed&Burn Molten Salt Reactor	Swiss nuclear MSc course / Defended	2019
J. Dietz	Chemical-Thermodynamic Modelling of the MSR-Related Systems Under Normal and Accident Conditions	Swiss nuclear MSc course / Defended	2020
R. Gonzales	Improved methodology for analysis and design of Molten Salt Reactors	PhD thesis in frame of H2020 project SAMOFAR / Defended at EPFL	2021
J. Santora	Two-fluid Breed & Burn MCFR parametric study	Swiss nuclear MSc course / Defended	2022
F. Borys	Identification and evaluation of transmutation criteria for selected reactors	Swiss nuclear MSc course	2023
M. Krstovic	Extended Point Kinetics Solver For MSR	Swiss nuclear MSc course	2023

Burnup definition

Burnup definition

- 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

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

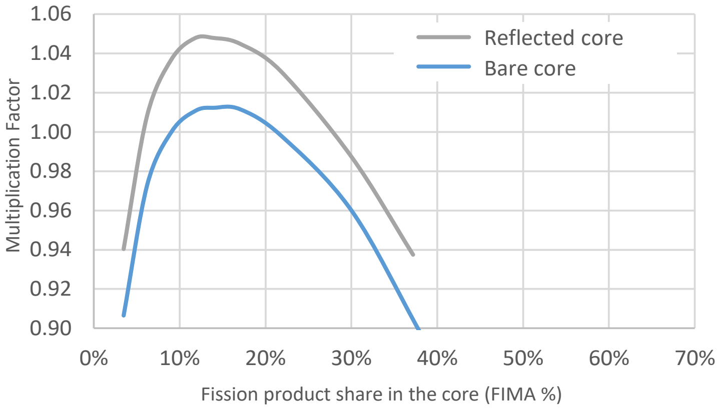
$$B_{FIMA\%}(t)=\frac{\int_0^t F(t)\,dt}{N_{Ac,core}(0)+\int_0^t \dot{N}_{Ac,in}(t)\,dt}=\frac{\int_0^t F(t)\,dt}{N_{Ac,core}(t)+\int_0^t \dot{N}_{Ac,out}(t)\,dt+\int_0^t F(t)\,dt}$$

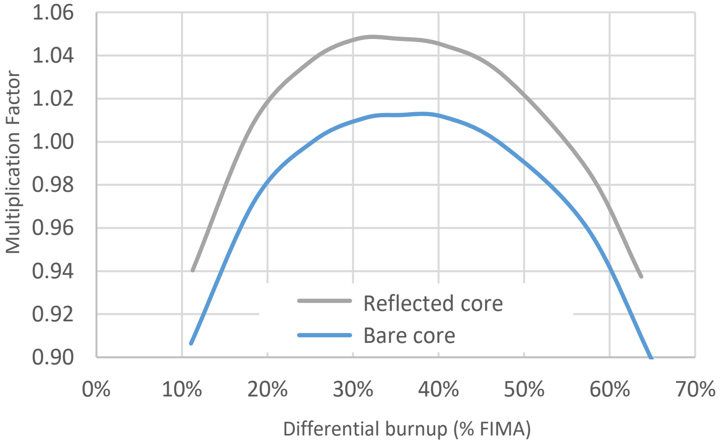
Krepel, J., Ragusa, J., Molten salt reactor physics: characterization, neutronic performance, multiphysics coupling, and reduced-order modeling, chapter 4 in a book: Molten Salt Reactors and Thorium Energy, 2nd Edition - June 1, 2023

Burnup definition

- Due to the continuous fission products removal by off-gas system, the burnup and the fission products share in the core differs.

$$FPs_{share,core} = \frac{M_{FPs,core}(t)}{M_{Ac,core}(t) + M_{FPs,core}(t)} \neq B_{FIMA\%}(t)$$





Krepel, J., Ragusa, J., Molten salt reactor physics: characterization, neutronic performance, multiphysics coupling, and reduced-order modeling, chapter 4 in a book: Molten Salt Reactors and Thorium Energy, 2nd Edition - June 1, 2023

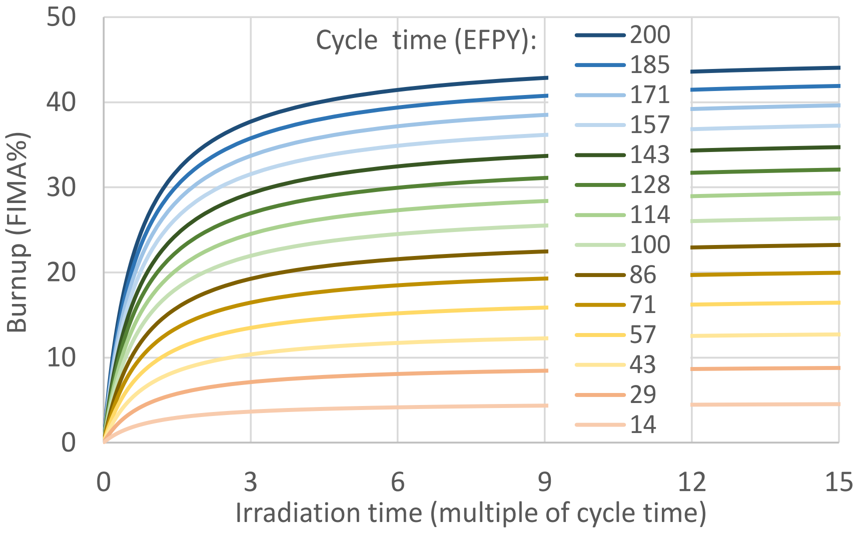
Burnup definition

- The differential and integral definition provide different values.
- The integral definition includes the initial core loading.
- For stabilized and long enough operation, they can be equal.

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

$$\Updownarrow$$

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



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