

Investigation of MSFR Burnup Characteristics using Monte-Carlo Neutronics Simulations

Marcin Sadowy (s223854), Ewout Voorrips (s243085)

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Project goal

The aim of this project is to assess the **burnup characteristics** of a cylindrical **Molten Salt Fast Reactor (MSFR)** using Monte-Carlo neutronics simulations, implemented in **Serpent2** v2.1.32 [2] with the **JEFF-3.1.1** Nuclear Data Library [3]. Several parameters, such as the effective multiplication factor, energy spectrum and radial and axial neutron flux are determined and compared with simulated burnup of up to **40 years**. Evolution of the inventory is simulated and a comparison is made between nuclear data for ^{149}Sm from JEFF-2.2 and JEFF-3.1.1. Finally, online fuel reprocessing is introduced, extracting insoluble noble metals and gaseous fission products through continuous helium bubbling. The results are ultimately compared to the **EVOL/MARS** neutronics benchmark [1].

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Context

MSFR is an advanced type of nuclear reactor originally developed by CNRS, France [1]. It uses a circulating **molten salt** as fuel and is based on the **fast neutron** spectrum. The salt is heated by **fission reactions** inside the salt and exchanges this heat in a heat exchanger, enabling power generation. Advantages of MSFR compared to more conventional solid-state fuel reactors include increased **safety**, higher heat conversion **efficiency** due to higher operating temperature and the possibility of using **thorium**, which is four times more abundant on earth than uranium, as a fertile material.

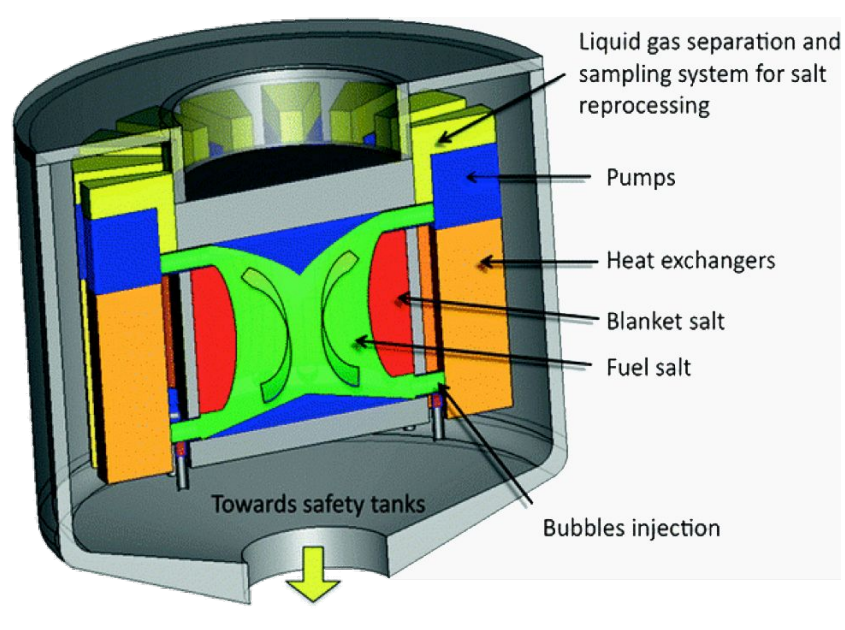


Fig. 1: Conceptual Design of an MSFR [1]

The benchmark to which our simulations are compared to uses the **thorium fuel cycle** ($^{232}\text{Th} \rightarrow ^{233}\text{Th} \rightarrow ^{233}\text{Pa} \rightarrow ^{233}\text{U}$). The reactor contains a fertile blanket, consisting of a (77.5 mol% LiF-22.5 mol% ThF_4) salt, which is used to breed more ^{233}U fuel. The fuel salt (initially 77.5 mol% LiF 22.5 mol% ThF_4 - ^{233}U) has an **inlet** temperature of **650°C** and an **outlet** temperature of **750°C**. In this project we consider a 3 GW_{th} reactor with a ^{233}U -started fuel.

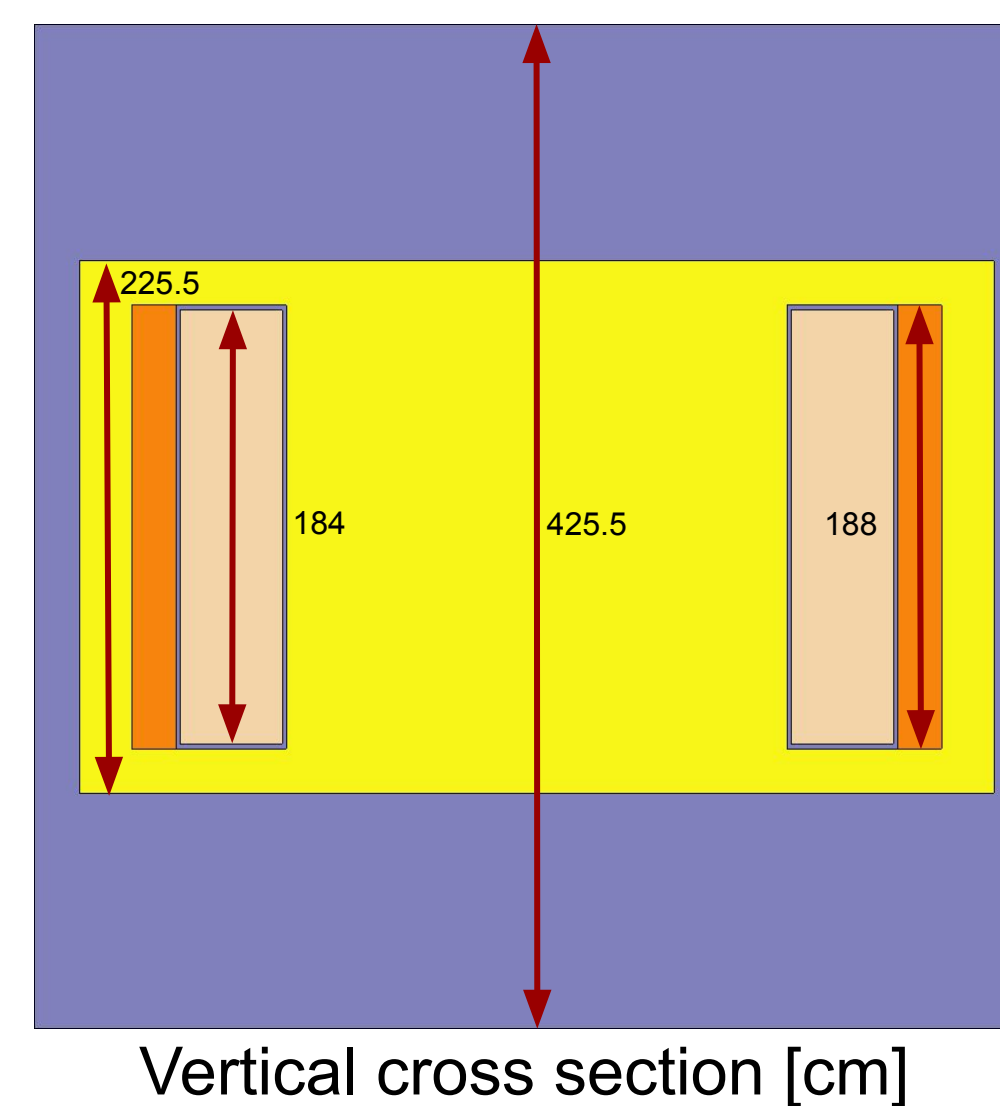
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Serpent2 modelling

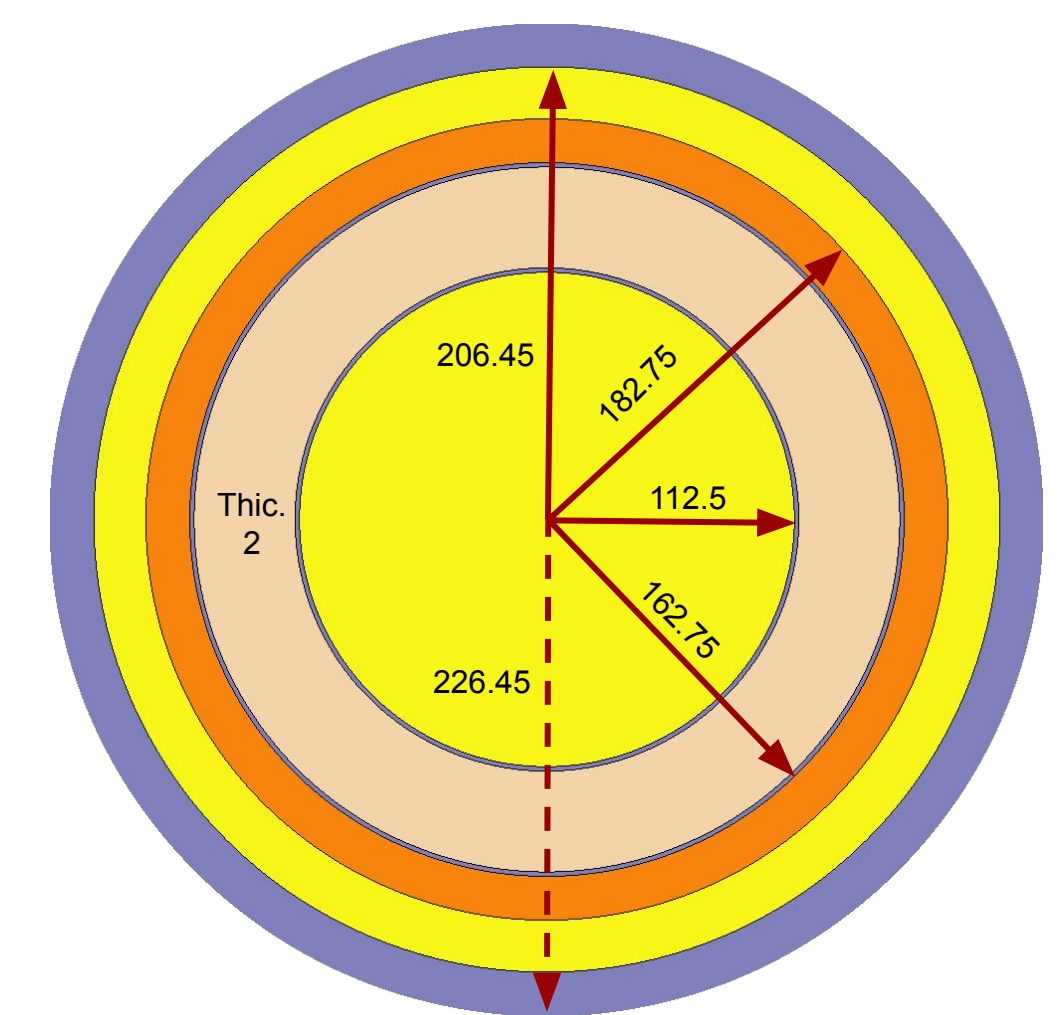
Initial HN concentrations

	^{232}Th [mol%]	^{233}U [mol%]
Steady-state	19.985	2.515
Depletion	19.945	2.555

- Neutron population of 100.000 per cycle
- 50 inactive cycles, 200 active cycles
- Black boundary conditions
- Unresolved resonance probability table on
- LiF- ThF_4 - ^{233}U fuel salt mixture (yellow)
- LiF- ThF_4 fertile blanket (beige)
- B_4C protective layer (orange)
- NiCrW Hastelloy reflector (purple)



Vertical cross section [cm]



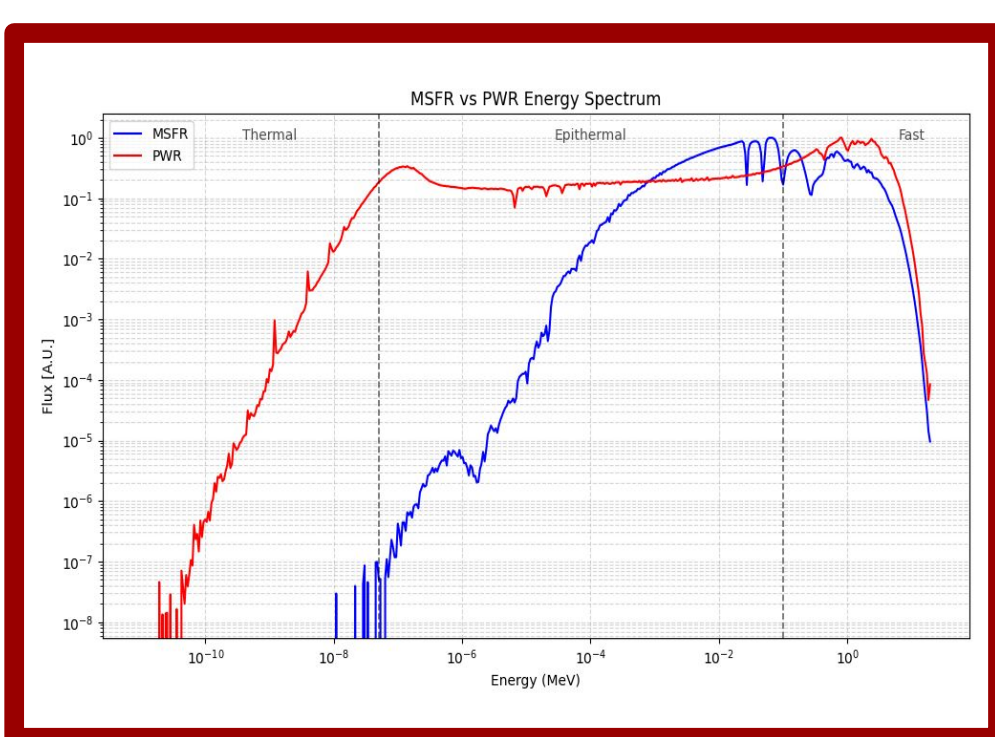
Horizontal cross section [cm]

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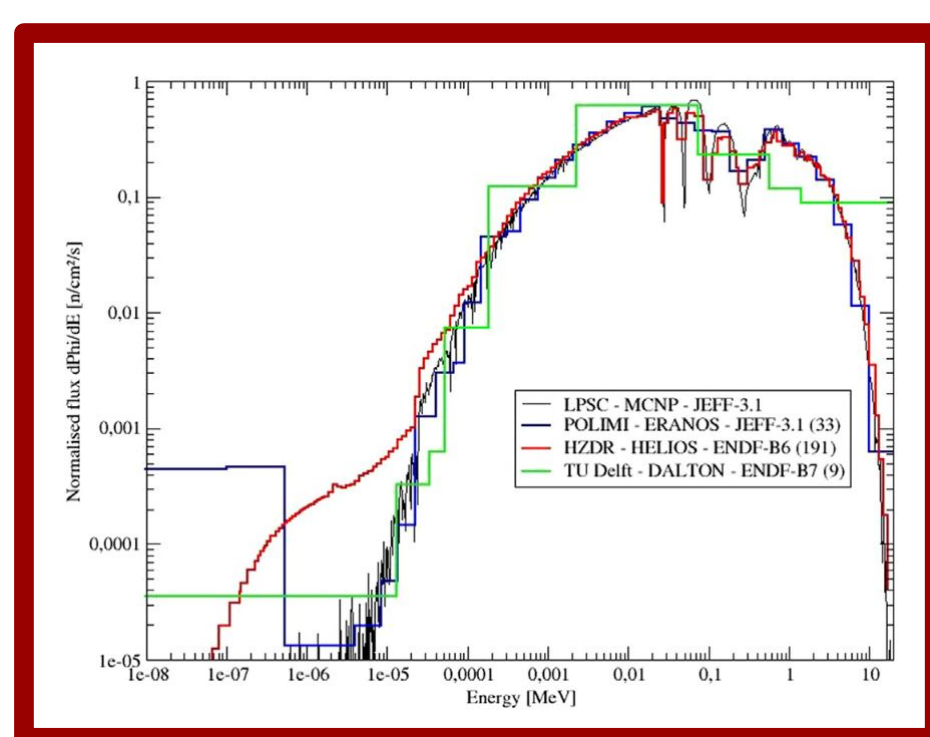
Results - Steady-state

Reaction rates

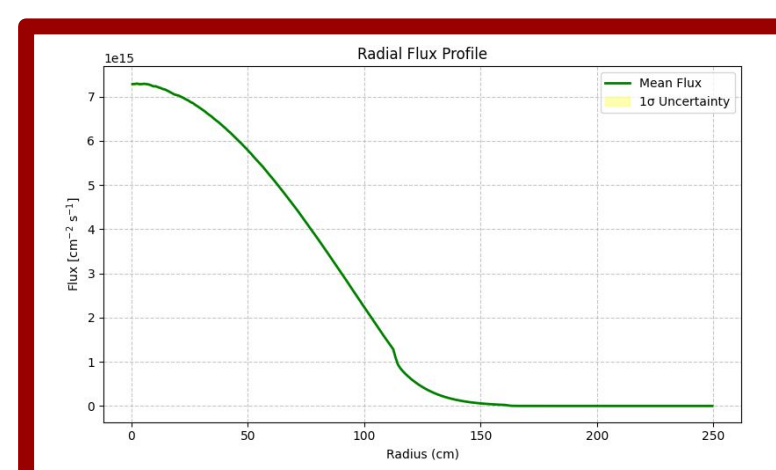
	Analog k_{eff} [-]	Mean Flux [$\text{cm}^{-2}\cdot\text{s}^{-1}$]	^{233}U f [mol/day]	^{233}U c [mol/day]	^{232}Th f [mol/day]	^{232}Th c [mol/day]
This work	0.99191 ± 26 pcm	3.56×10^{15}	13.2	1.38	0.24	15.8
POLIMI SERPENT JEFF-3.1	0.99406 ± 40 pcm	3.56×10^{15}	13.3	1.37	0.24	15.8



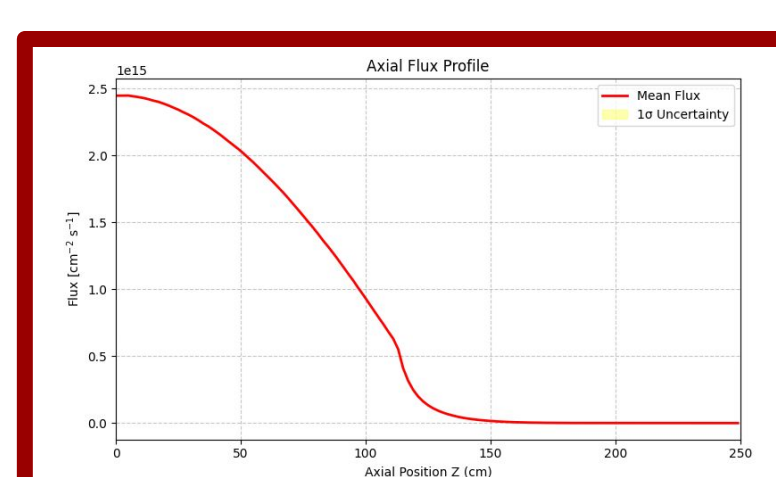
MSFR and PWR neutron energy spectra normalized to highest common value



Neutron energy spectra from EVOL/MARS benchmark [1]



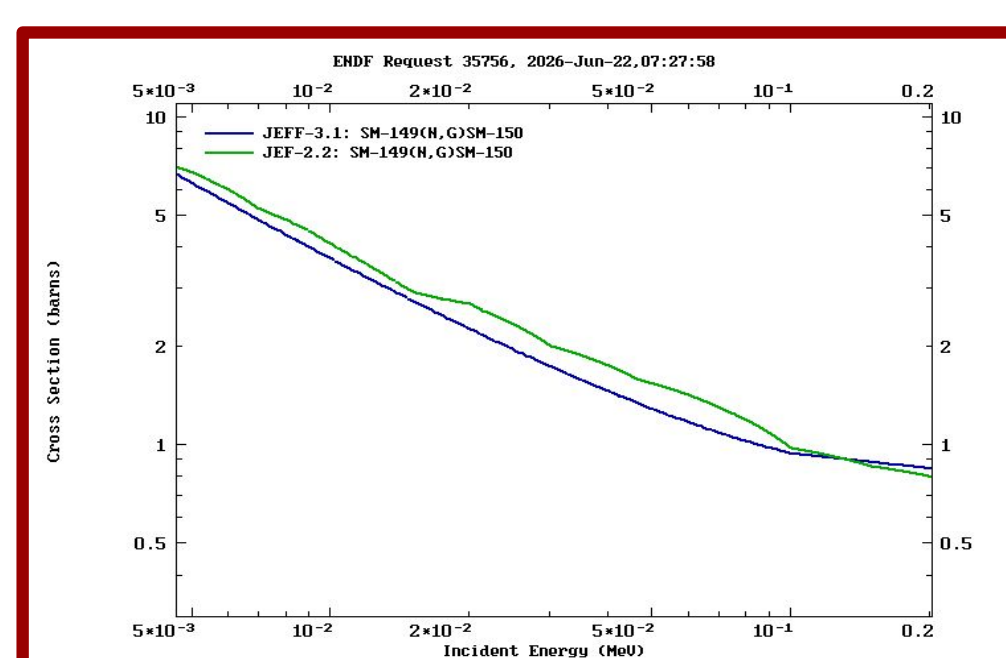
Radial flux distribution



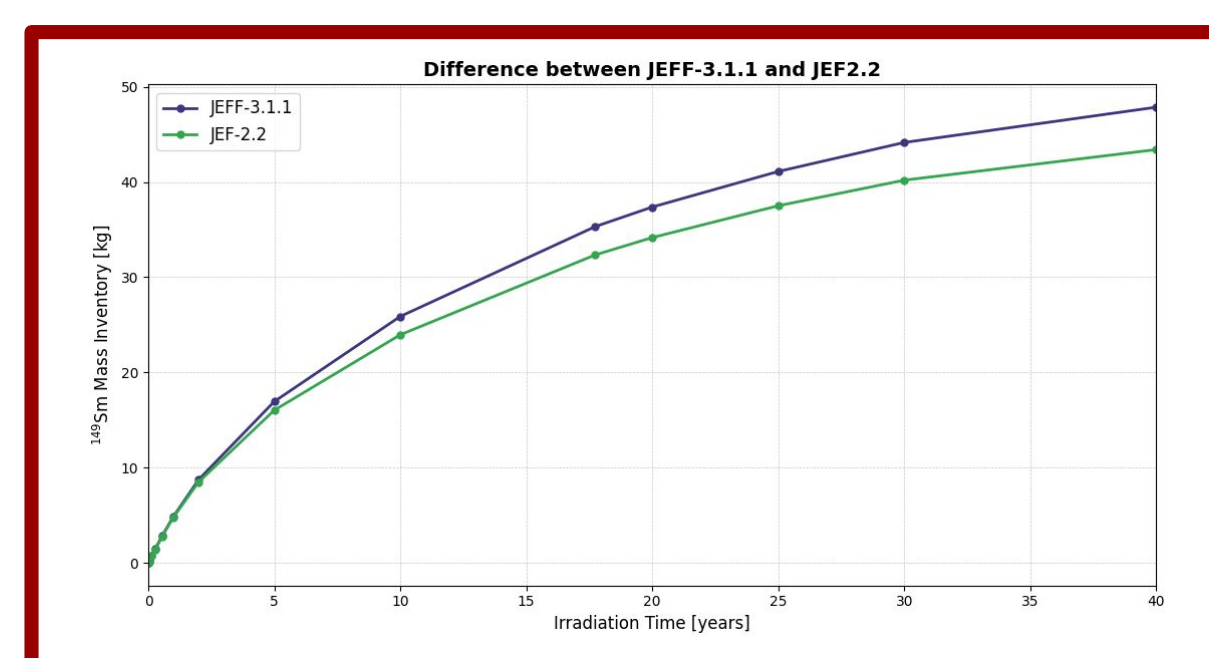
Axial flux distribution

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Impact of nuclear data library



^{149}Sm capture cross section data [4]



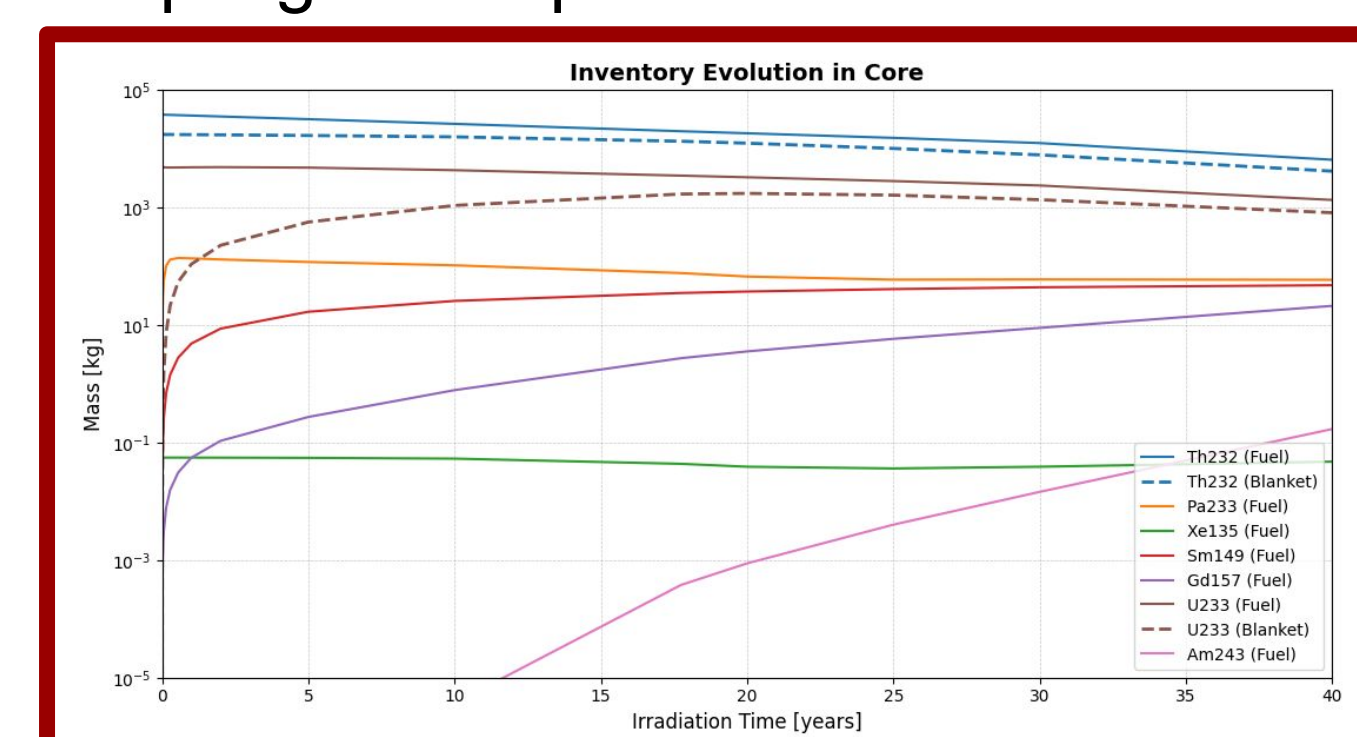
Inventory mass of ^{149}Sm

- Higher ^{149}Sm inventory with JEFF-3.1.1 explained by cross section data
- In relevant energy range the capture cross section is higher with JEF-2.2, meaning more ^{149}Sm is converted to ^{150}Sm .

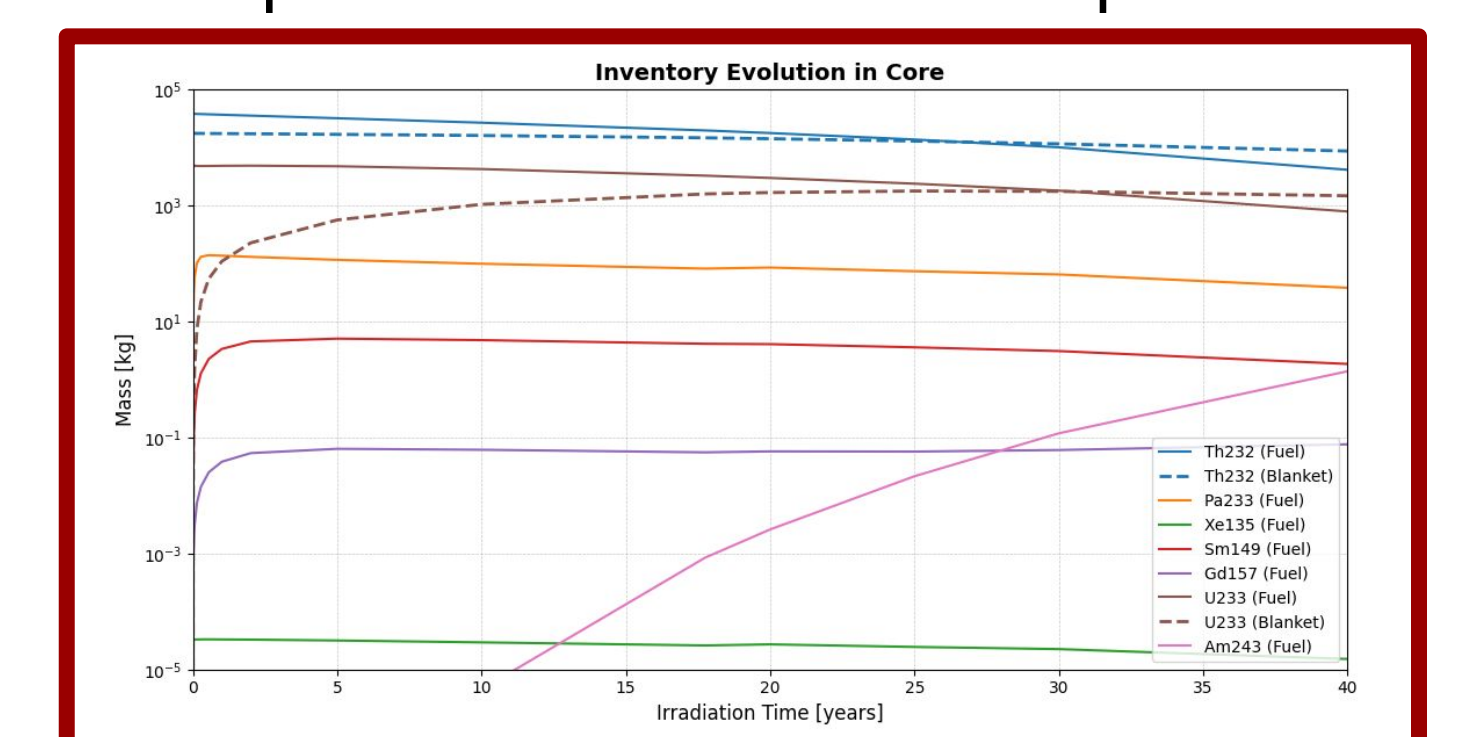
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Results - Depletion

Removal of elements $Z = 1, 2, 7, 8, 10, 18, 36, 41-47, 51, 52, 54$ and 86 through online helium bubbling extraction with $T_{1/2} = 30\text{s}$. Offline extraction removes $Z = 30-40, 48-50, 53, 55-70$ in a 450 day cycle. Bateman eqs. are solved using the CRAM solver method. No-reprocessing calculations used LELI predictor-corrector coupling and reprocessor calculations only used a predictor with linear extrapolation.



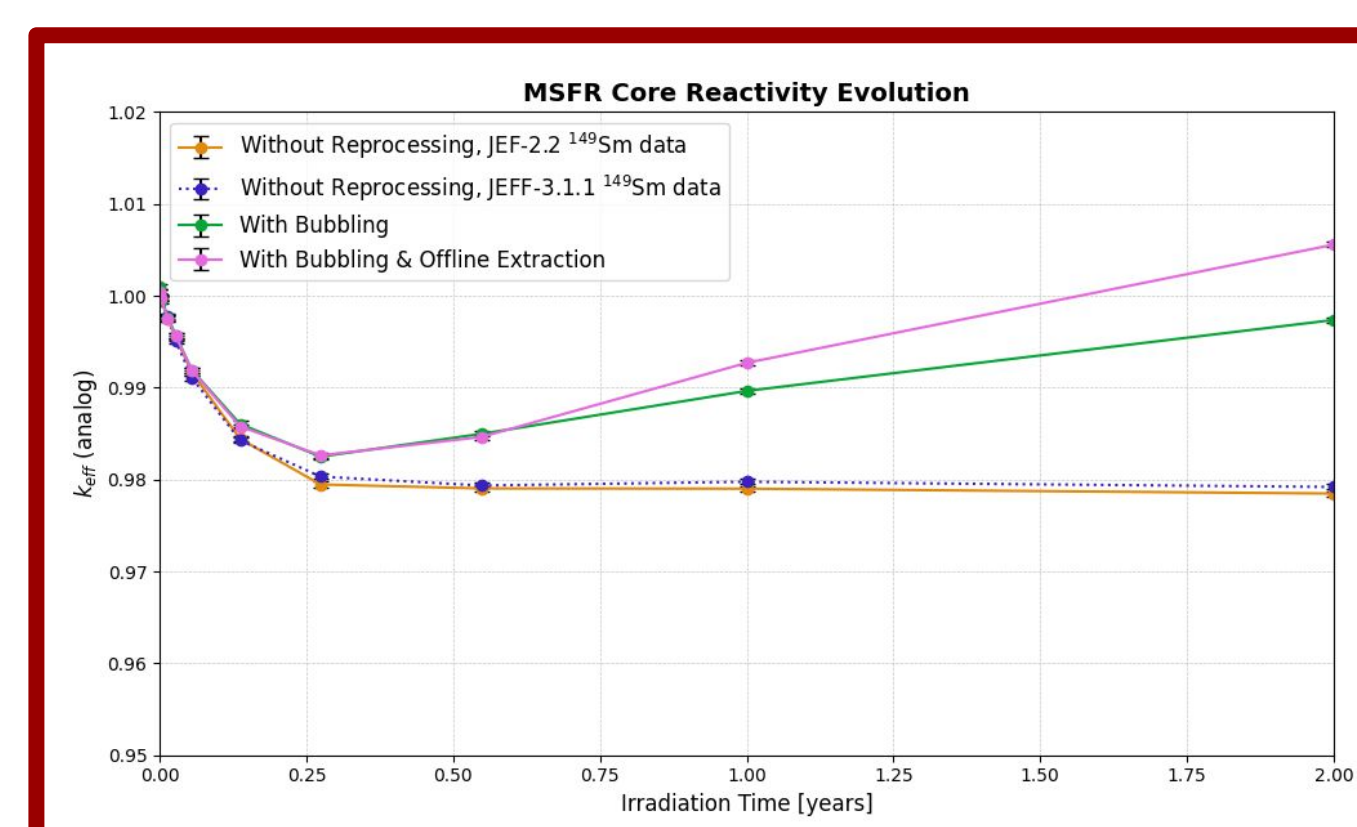
No reprocessing, JEFF3.1.1



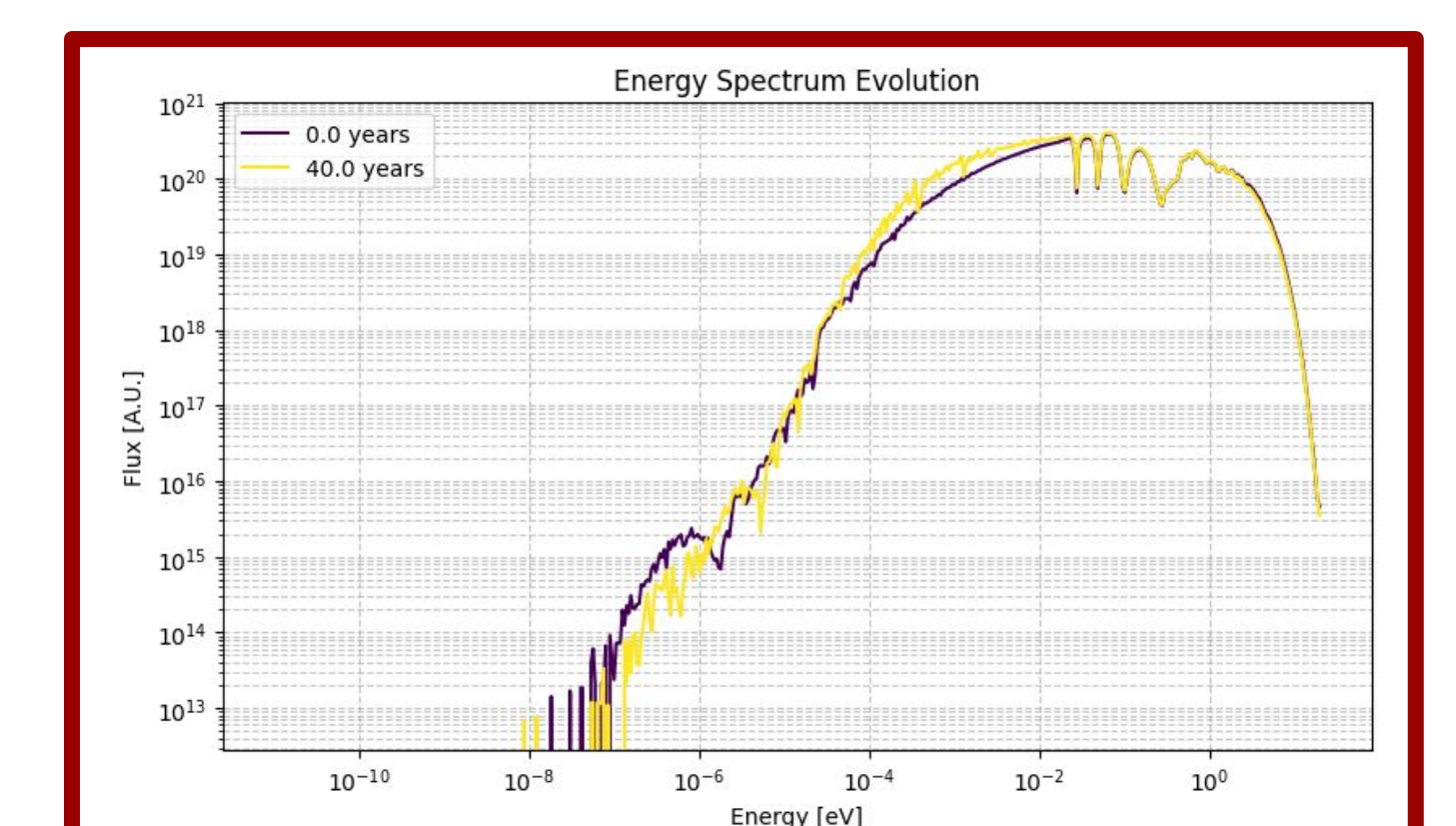
Online reprocessing

Lower ^{135}Xe concentration

k_{eff} increase from decreased poisoning



Evolution of k_{eff} as function of irradiation time



Start/end of cycle neutron energy spectra

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Conclusion

- Reactor core geometry and simulation parameters from EVOL/MARS benchmark
- k_{eff} mean flux and fission/capture reaction rates for ^{232}Th and ^{233}U in good agreement with the EVOL/MARS neutronics benchmark
- Steady depletion of fissile nuclides as expected without refueling
- Concentration of ^{233}U increased in fertile blanket
- Online extraction presented a large drop in ^{135}Xe concentration
- Part of initial k_{eff} drop mitigated, rising after initial settling period, settling at higher value long-term
- Neutron energy spectrum hardened over time, favoring epithermal region compared to beginning of cycle

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References

- Neutronic benchmark of the molten salt fast reactor in the frame of the EVOL and MARS collaborative projects. M. Brovchenko et al., EPJ Nuclear Sci. Technol. Volume 5, 2019
- Status of the Serpent Monte Carlo code in 2024, J. Leppänen et al., EPJ Nuclear Sci. Technol., 11:3, 2025
- Status of the JEFF Nuclear Data Library, A.J. Koning et al., JKPS:59:1057-1062, 2011
- Flex-DB-Ex - Flexible Database Explorer (Version 1.004) Zerkov, V., IAEA, 2008