OPENFOAM: MSFR REACTIVITY INSERTION & LOFA
OPENMC: NEUTRONIC BENCHMARK ON CEFR START-UP TESTS

JOINT ICTP-IAEA WORKSHOP ON OPEN-SOURCE NUCLEAR CODES FOR REACTOR ANALYSIS

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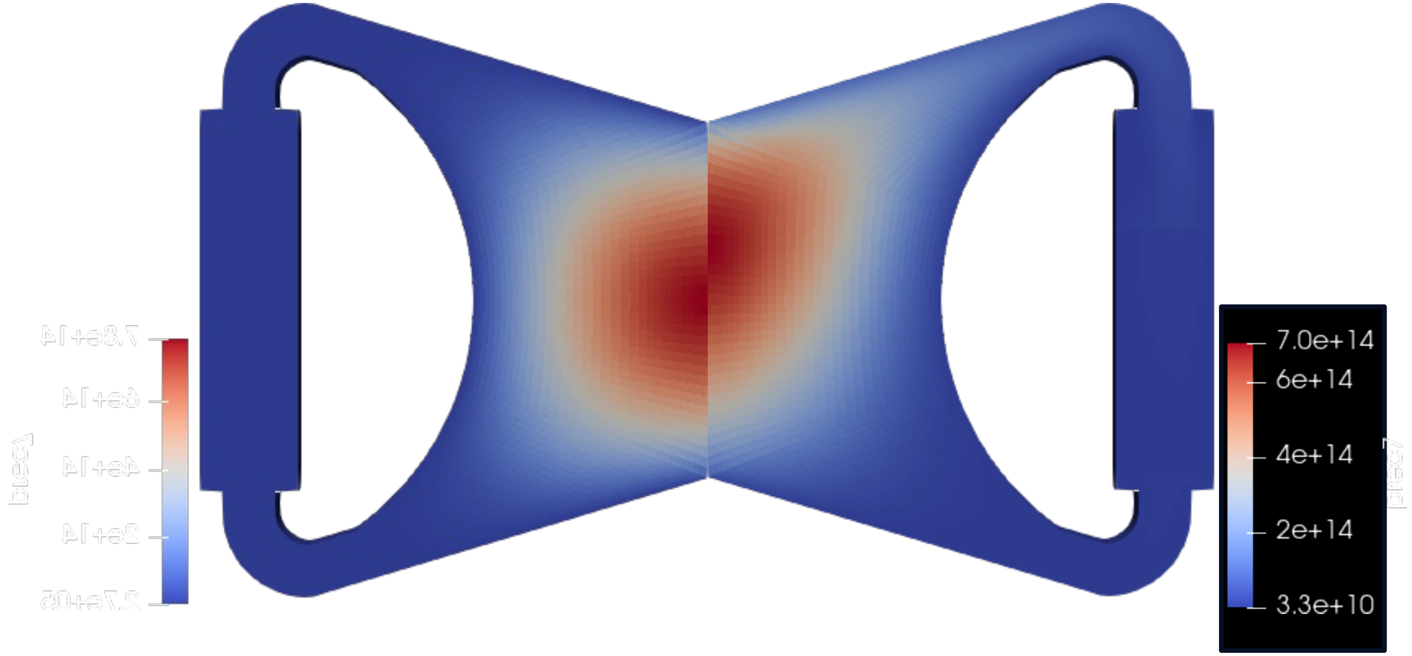
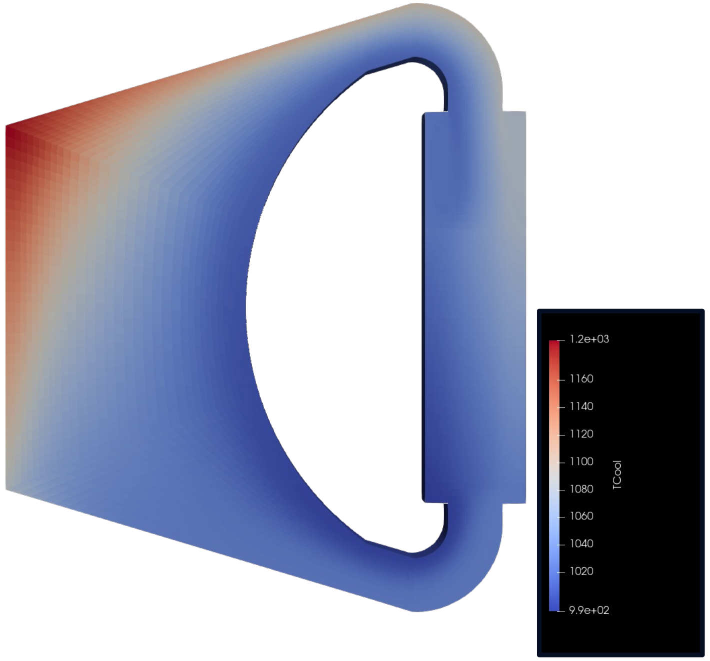
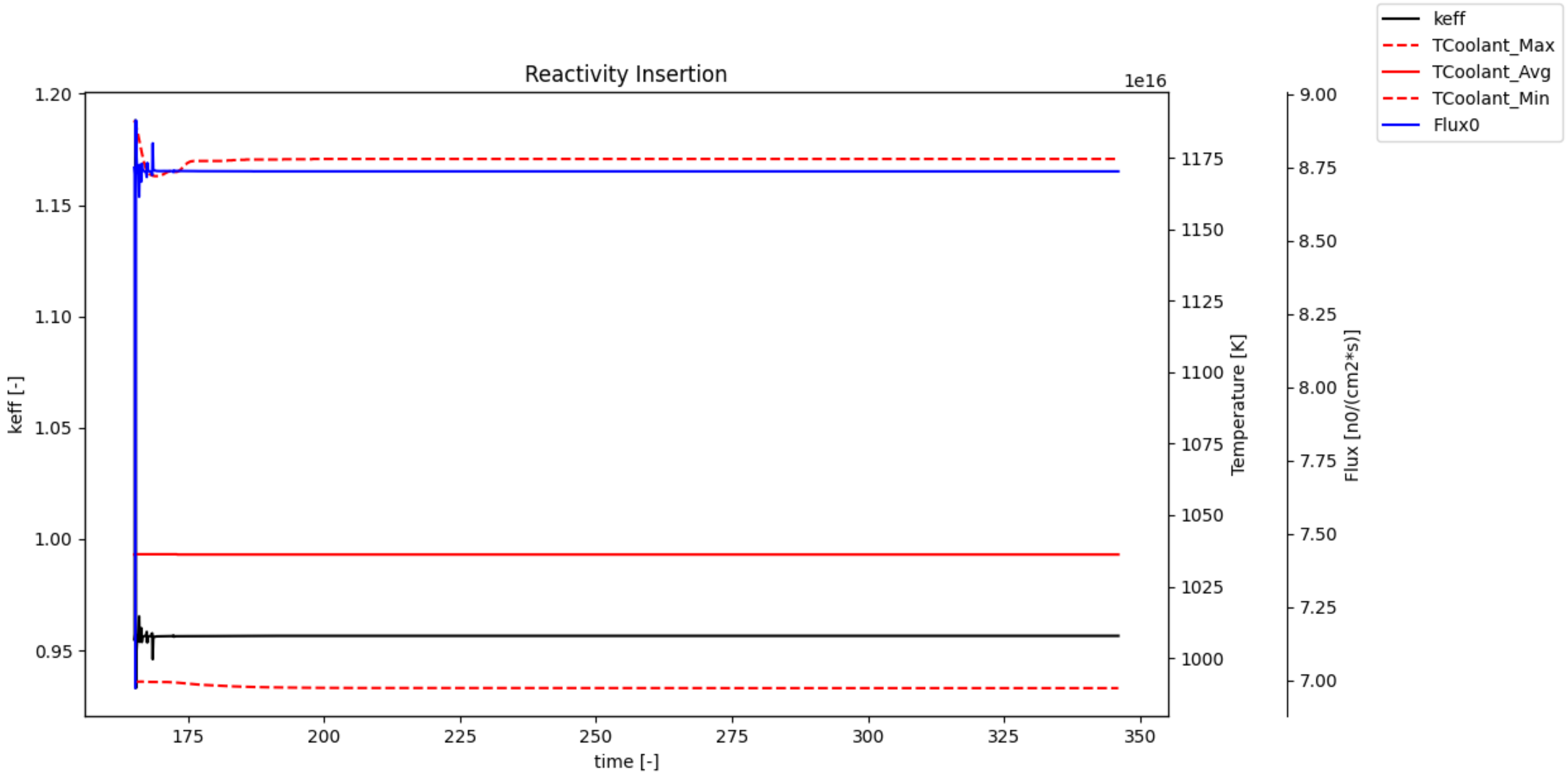
FUNCTIONAL TESTS

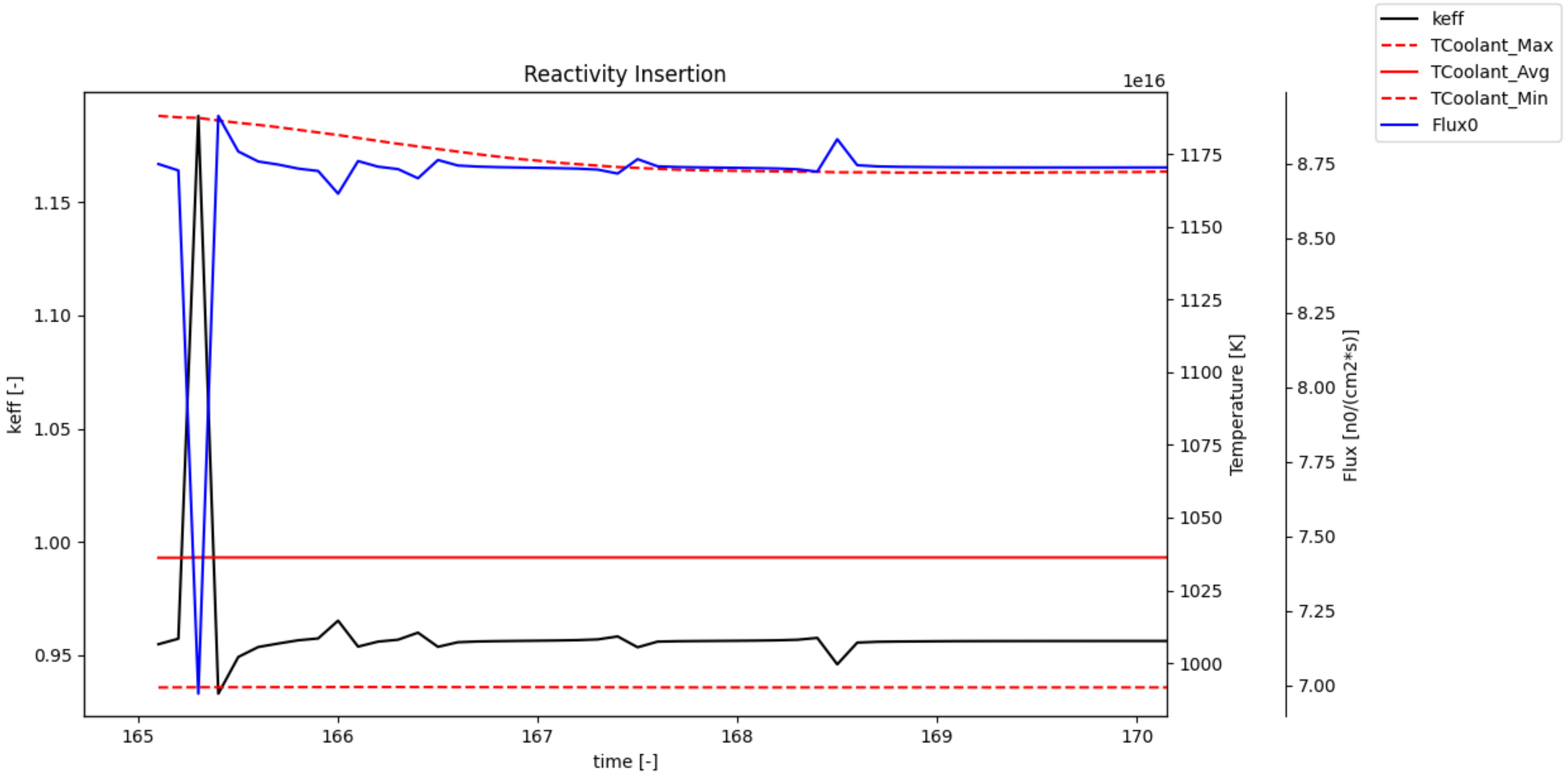




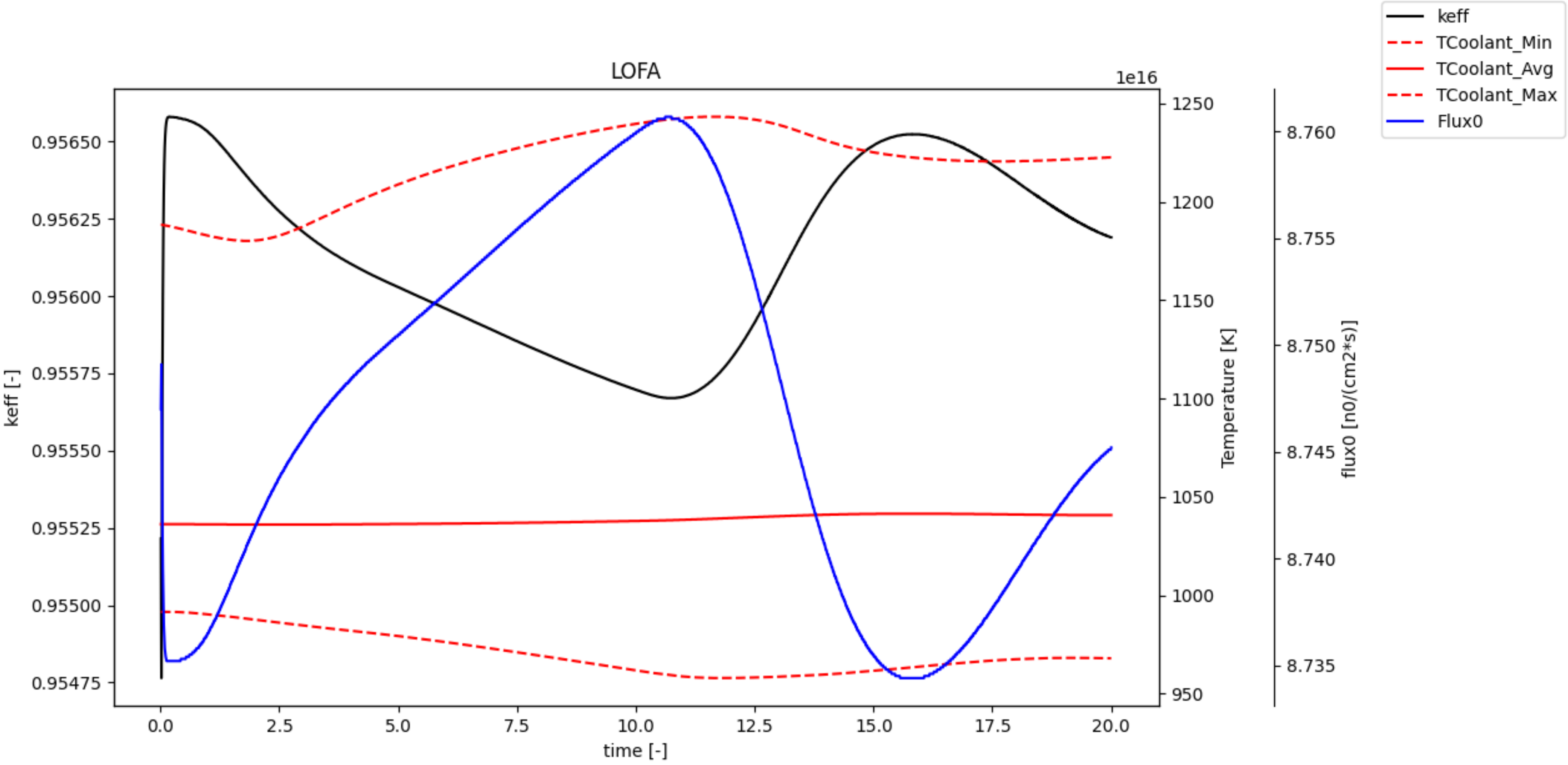
REACTIVITY INSERTION



REACTIVITY INSERTION



LOSS OF FLOW ACCIDENT

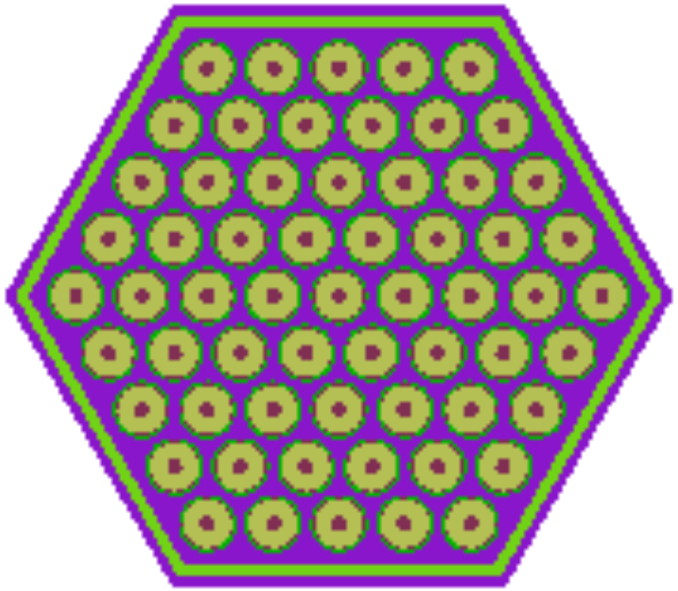


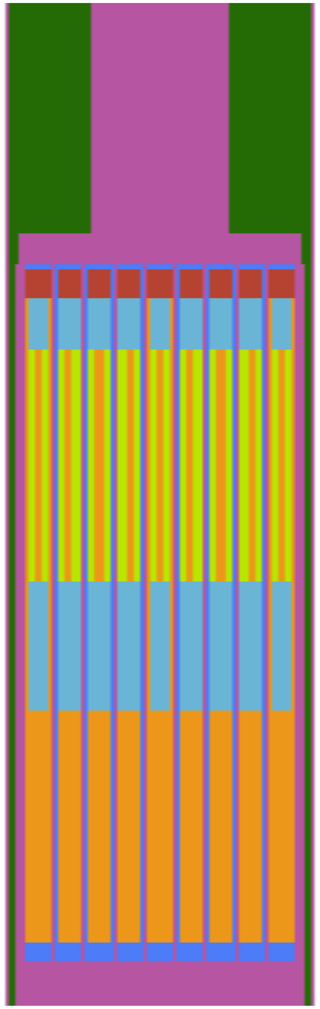
MSFR SIMULATION CONCLUSIONS

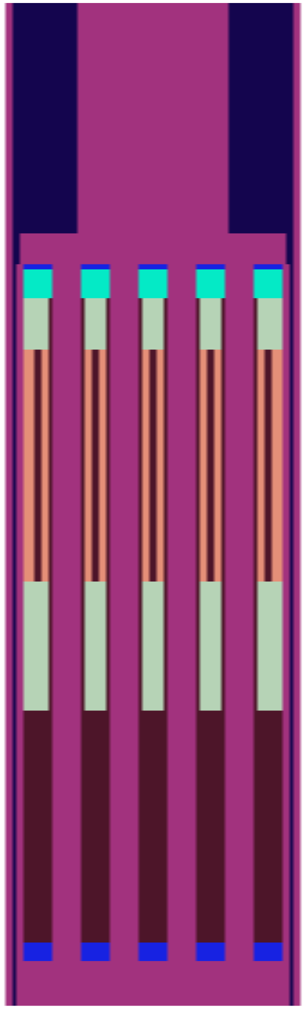
- The results are not to be trusted
- More sensitivity and convergence analysis needed

- OpenFoam's multi-region capability facilitates complex multi scale modelling and simulation
- GenFoam is a great tool for MSR reactors, since its multi-physics nature allows for simulation of thermal-hydraulics and neutronics calculations

OPENMC: GEOMETRY PLOTS







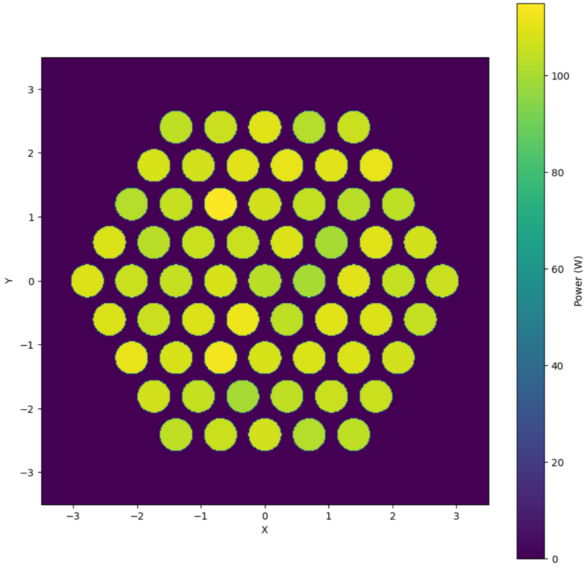
RESULTS

- k-effective (Collision) = 1.56156 +/- 0.00436
- k-effective (Track-length) = 1.55914 +/- 0.00466
- k-effective (Absorption) = 1.56008 +/- 0.00251
- Combined k-effective = 1.56021 +/- 0.00227
- Leakage Fraction = 0.02205 +/- 0.00045

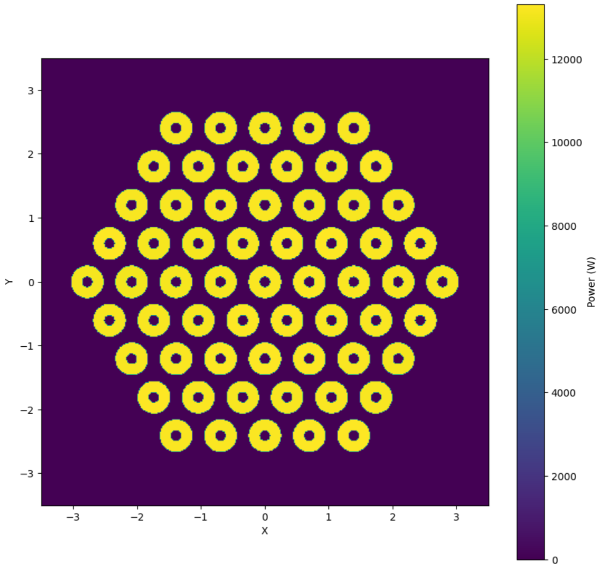
Large leakage; no time to further investigate

RESULTS: POWER DISTRIBUTION

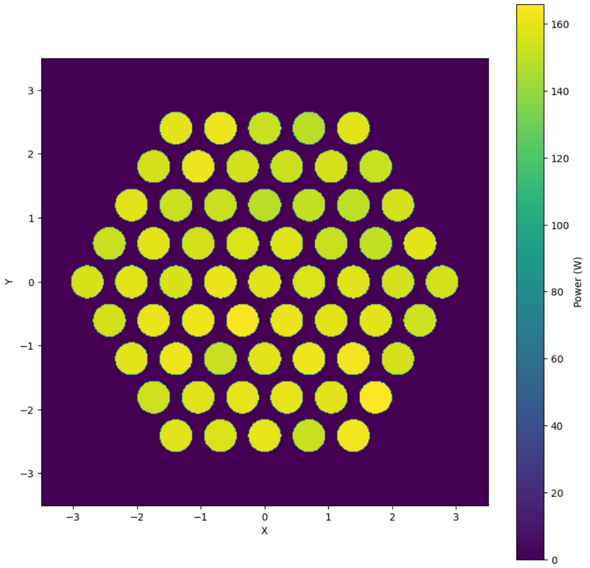
At BOC, the blanket (depleted uranium) is expected to have low fission compared to the fuel region, as we see in the plots. With operation time, the blanket will have more Pu-239 as a result from neutron absorption from U-238 and transmutation. The fission rate in that region may increase.



a) Lower blanket



b) Fuel



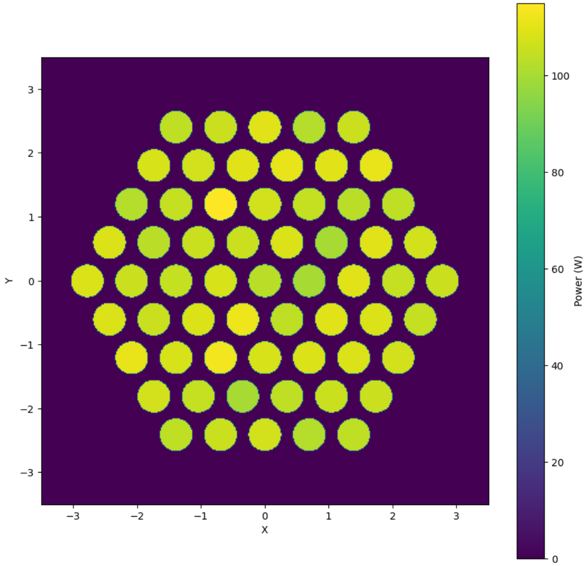
c) Upper blanket

Results are not normalized but the ratio is constant

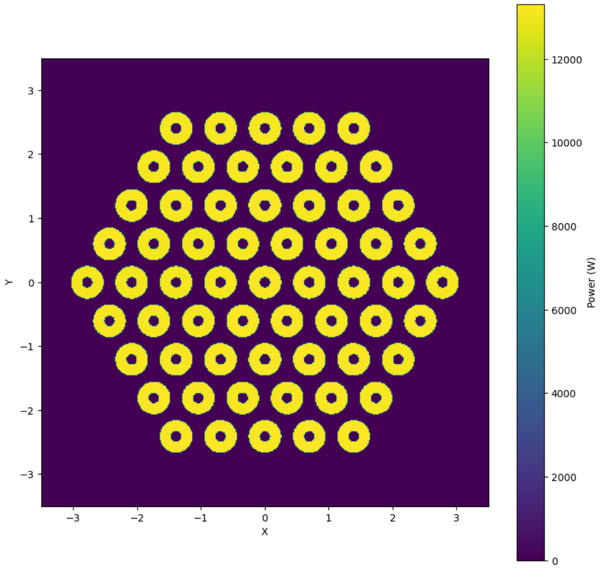
Material	Fission	Absorption	Scattering
Fuel	6.10e-01	7.81e-01	9.91e+00
Blanket	1.20e-02	1.11e-01	5.46e+00

RESULTS: POWER DISTRIBUTION

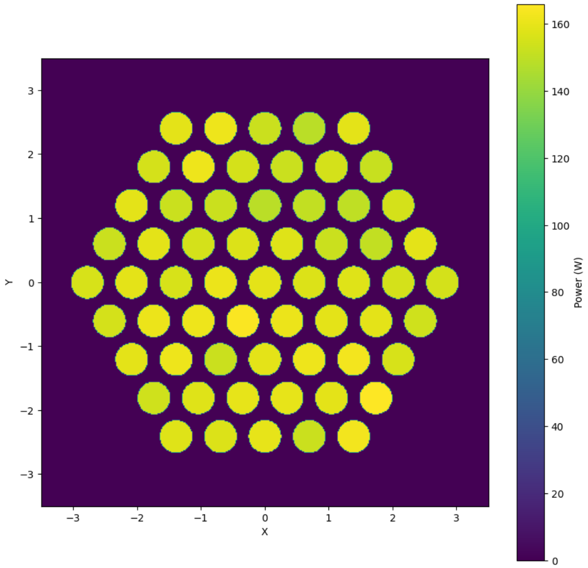
Reflective boundary conditions ensure that the difference between the power in the central pins and the periphery are not large.



a) Lower blanket



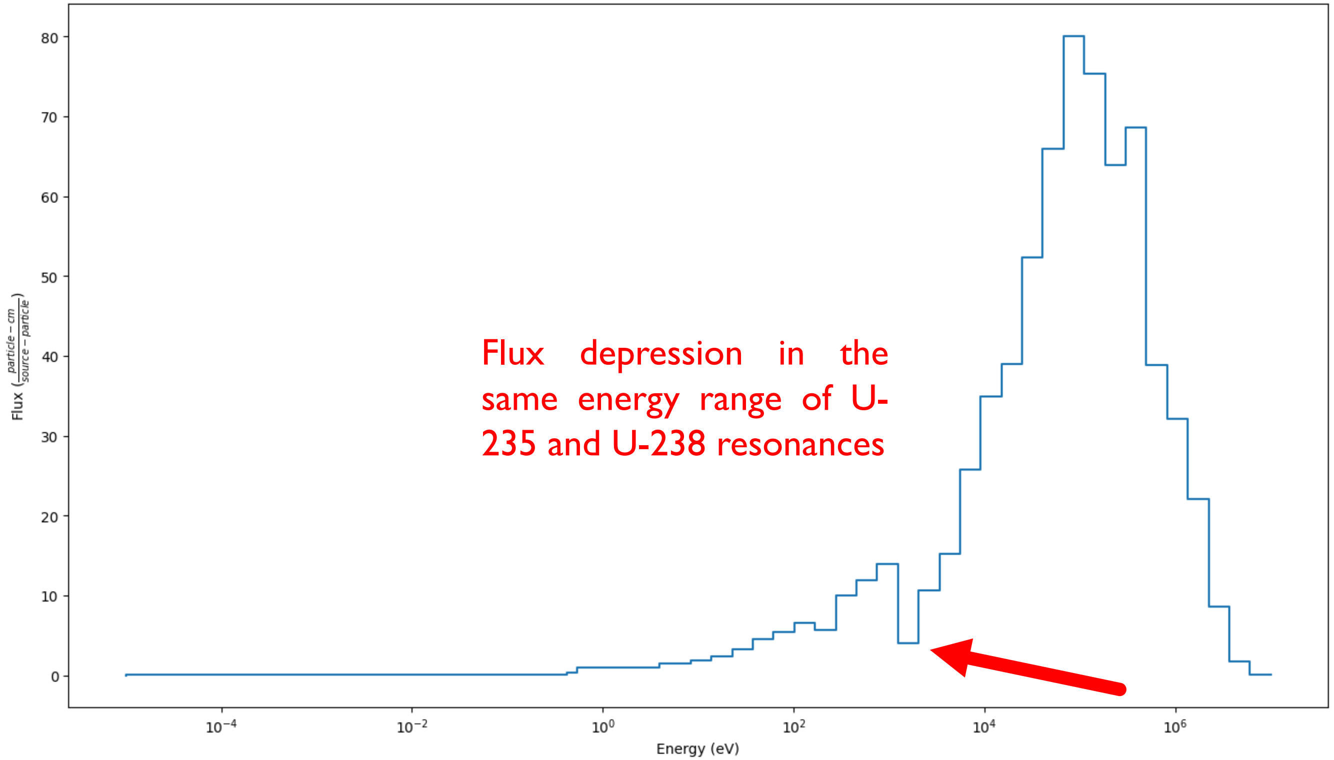
b) Fuel

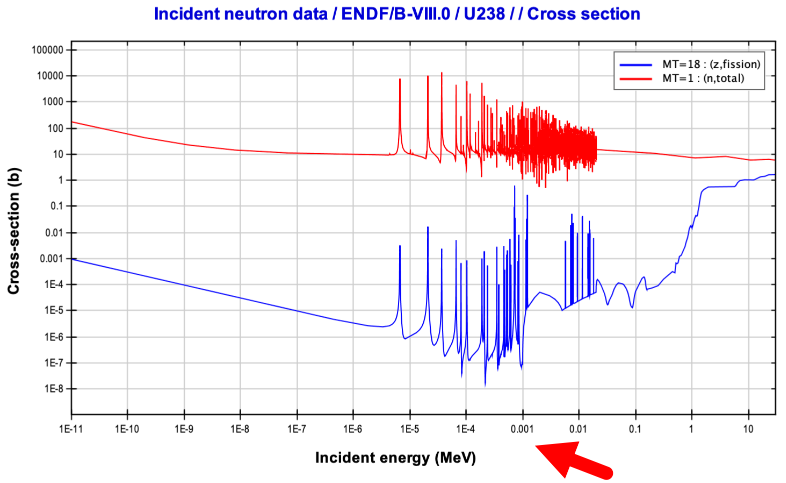


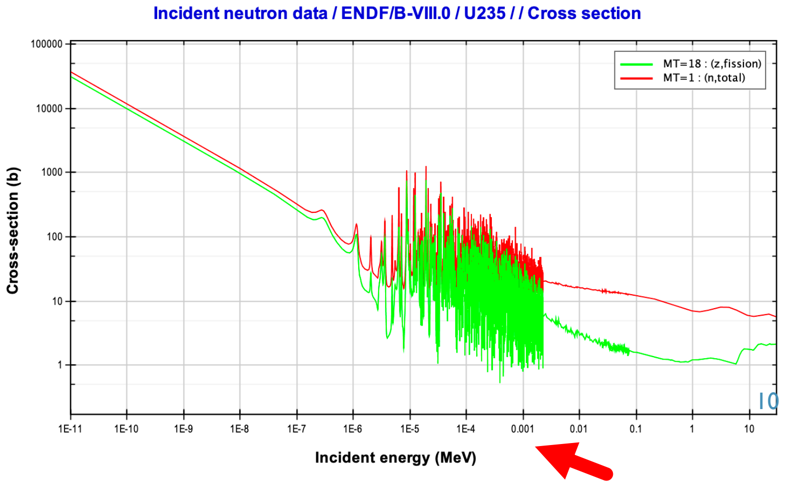
c) Upper blanket

The upper blanket length is 10 cm, while the lower blanket length is 25 cm; that means that the upper blanket has a smaller fuel contact to perimeter ratio, which may explain its higher power as compared to the lower blanket – since the plots show the mean value for the pins.

FLUX SPECTRUM

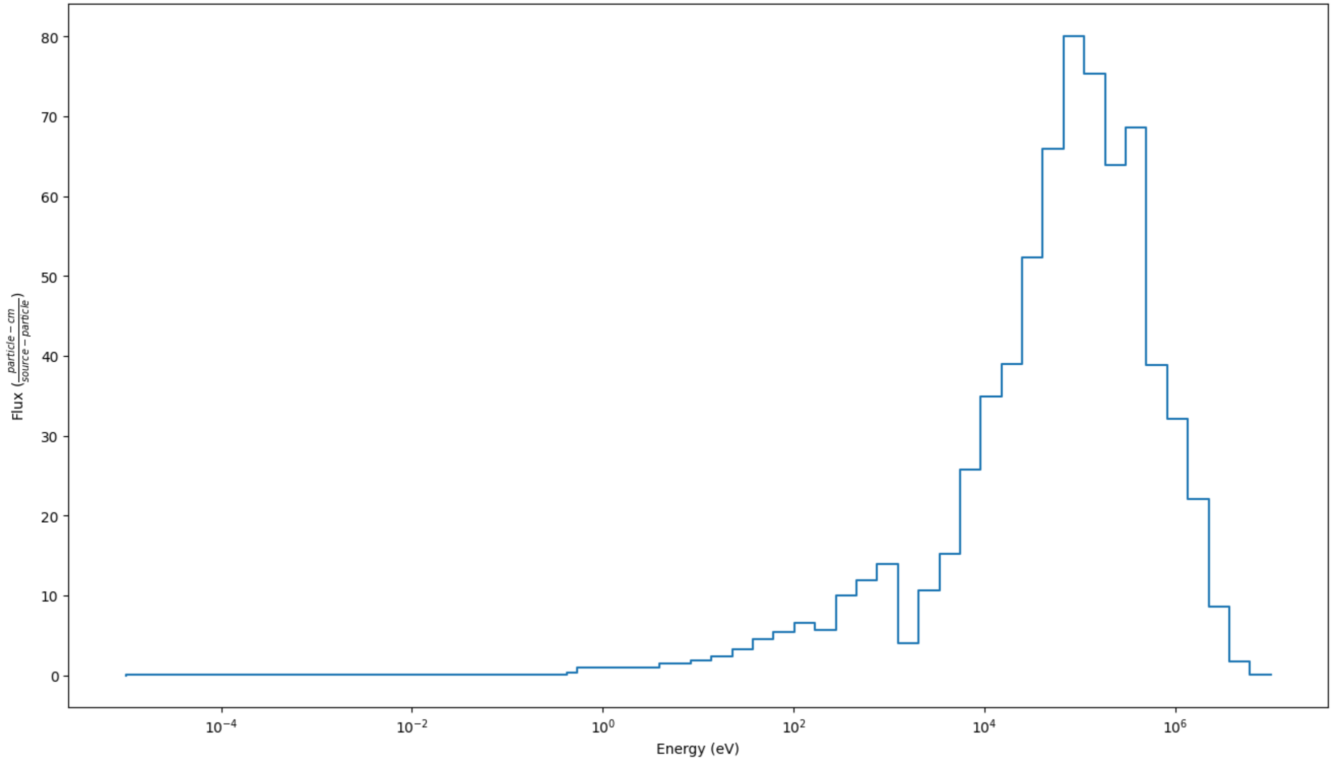


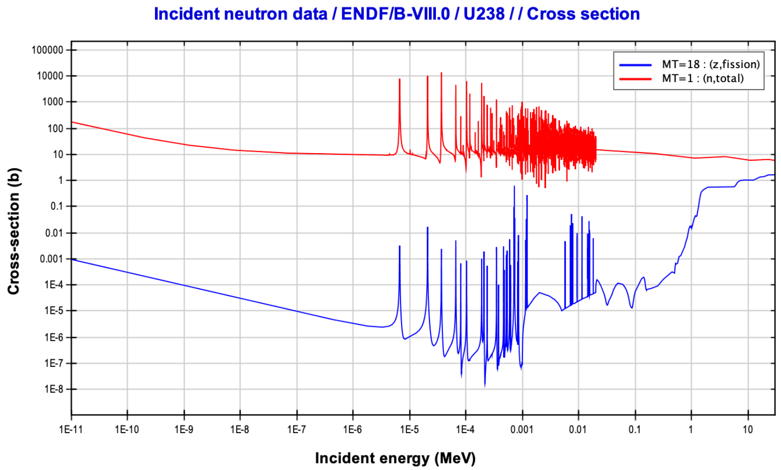


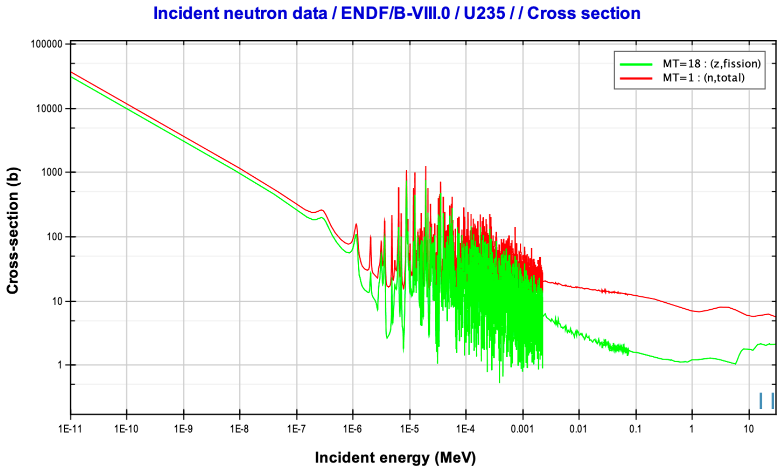


As expected, the majority of neutrons is in the fast spectrum (MeV), since they are born with this energy in average and in a fast reactor there is no moderation.

FLUX SPECTRUM







In the cross section plots generated by JANIS, we notice that while the fission cross section is almost coincident with the absorption cross section for U-235, for U-238 there is less fission; however, the absorption will result in transmutation of U-238 to Pu-239 – generating more fuel with operation time.

CONCLUSIONS

- The simulation showed good results in the range of the expected – we don't have the comparison data
- OpenMC is an incredible tool, especially considering:
  - The integration of many features that in other codes would have to be done separately, with different tools.
  - The high-level language is user-friendly, and the help tool is very convenient for users familiarized with MC neutronics codes – not having to go through a manual every time you need to check something saves a lot of time!
  - All the benefits of open source codes

ACKNOWLEDGEMENTS

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