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The Effect of Nuclear Data Discrepancies on Criticality Simulations of Molten Salt Fast Reactors

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Presentation Overview

- Background and Motivation
- Modeling Methodology
- Results and Discussion
- Conclusions and Future work

Background and Motivations

Terrapower's Molten Chloride Fast Reactor

- New MSFR concept using chloride salt
- Minimal Cl Nuclear Data and no updates between recent libraries.
- New Cl Data released in ENDF8.1

Research Objectives

- How does new nuclear data, and discrepancies between libraries, affect the criticality of neutronics modeling?

- What are the physical mechanisms — fission, capture, or leakage — driving those changes?

- Which isotopes and energy ranges are most responsible for the difference?

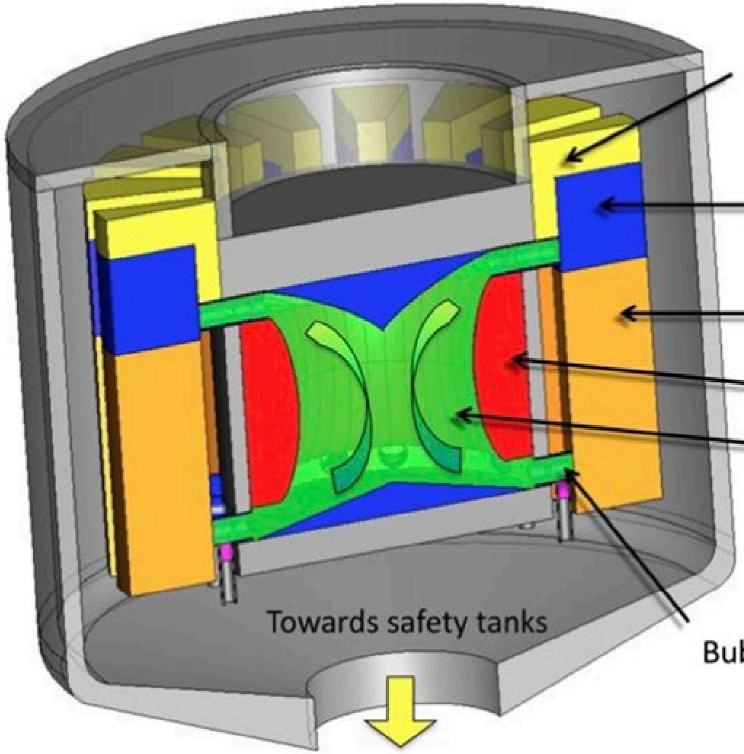
Modeling Methodology

Modeling Codes and Procedures

Nuclear Codes:

- NJOY2016
  - Difference in key MSFR isotopes
  - Groupr: 239 groupwise cross section
- Serpent
  - Utilized MSFR Benchmarks
  - Combined fuels from different studies
  - Criticality Simulation
  - In depth physics analysis
- OpenMC for K-eff Verification

MSFR Benchmark Concept



"Brovchenko et al. [3]

Parameter	Unit	Value
Total thermal power	MW_{th}	3000
Electric Power	MW_e	1500
Fuel salt volume	m^3	18 (9 in the core)
Blanket salt volume	m^3	7.3
Core dimensions	m	Radius: 1.1275 Height: 2.255
Blanket dimensions	m	Thickness: 0.5 Height: 1.88
Axial reflector thickness	m	1
Inlet/outlet fuel salt temperature	°C	650/750
Density	g/cm^3	4.1249
Melting Point	^oC	565

Criticality of 6 Fuels

Fuel	ENDF/B-VII.1	ENDF/B-VIII.0	Absolute Difference in[pcm]
^{233}U -Fluoride	0.927006 ±0.00004	0.94256 ±0.00004	-1555.4
TRU-Fluoride	0.994779 ±0.00004	0.993006 ±0.00004	177.3
TRU/ U^{enr} -Fluoride	1.04199 ±0.00004	1.0439 ±0.00004	-191.0
^{233}U -Chloride	0.976869 ±0.00008	0.981182 ±0.00008	-431.3
TRU-Chloride	1.08409 ±0.00007	1.08876 ±0.00008	-467.0
TRU/ U^{enr} -Chloride	0.974182 ±0.00007	1.01372 ±0.00007	-3953.8

What can we unveil about the physics contributing the absolute difference?

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Results and Discussion

Neutron Flux Distribution (1/2)

Spatial Flux Distribution

- **Tally setup:** 2D mesh flux tallies to map neutron flux across the radial (R–Z) plane and azimuthal angles at the core midplane (Z = 200.688 cm).
- **Postprocessing:** Flux normalized per source neutron to yield absolute flux distributions.
- Data interpolated and visualized via MATLAB contour mapping.

Energy dependent Flux

- **Tally setup:** Continuous-energy flux tallies across fuel, moderator, and reflector to capture spatial energy gradients.
- **Postprocessing:** Collapsed flux into logarithmic energy bins to generate the neutron spectrum.
- Compared spectral shapes between data libraries.

Flux Distribution (2/2)

Spatial Flux Distribution

Energy dependent Neutron Flux

TRU/U^{enr} -Chloride

Integral Reaction Rates(1/1)

- Integrate flux across microscopic cross sections across energy to obtain total reaction rates
- Looked at fission, capture, absorption, and leakage cross sections

Integral Reaction Rate	ENDF/B-VII.1	ENDF/B-VIII.0	Relative % difference
Fission	0.34468	0.358892	-4.12
Absorption	0.998896	0.998558	0.03
Capture	0.654216	0.639667	2.22
Leakage	0.001103	0.001440	-30.5

Sensitivity Analysis (1/2)

- **Perturbation:** Key reactions: Elastic scattering, fission, capture, and radiative capture

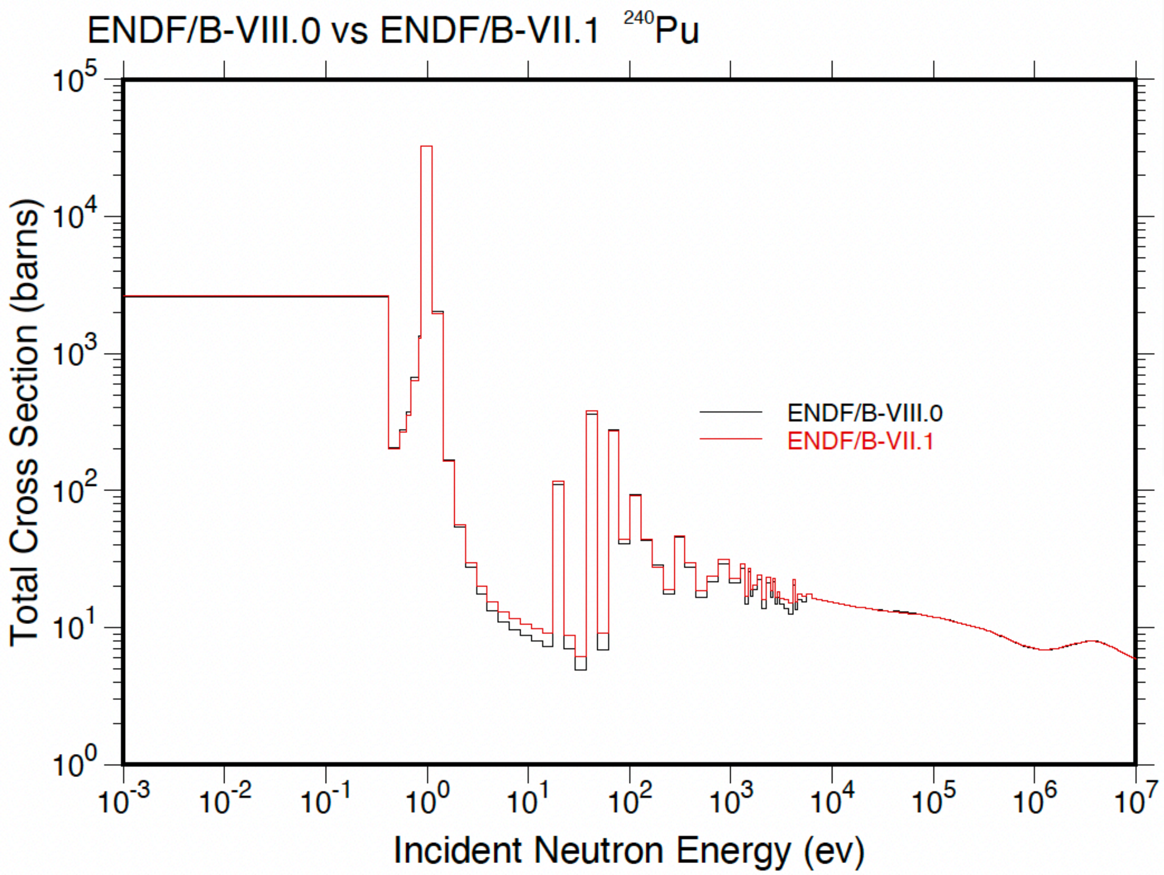
- Each index represents the change in k-eff per unit cross-section change, collapsed to 8-group energy bins.

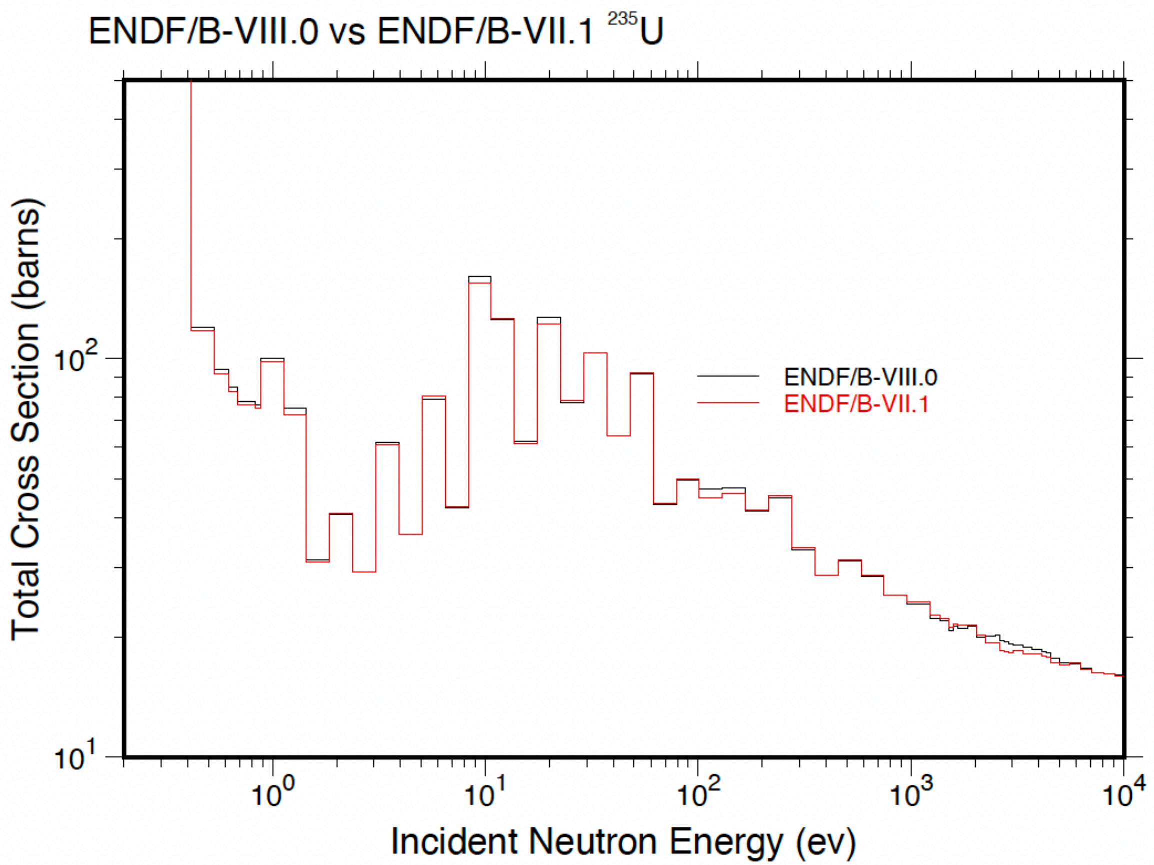
- Identifies which nuclear data by isotope and reaction type most influence reactivity

Sensitivity Analysis (2/2)

Sensitivity Index of CI and total materials

Sensitivity Analysis (2/2)



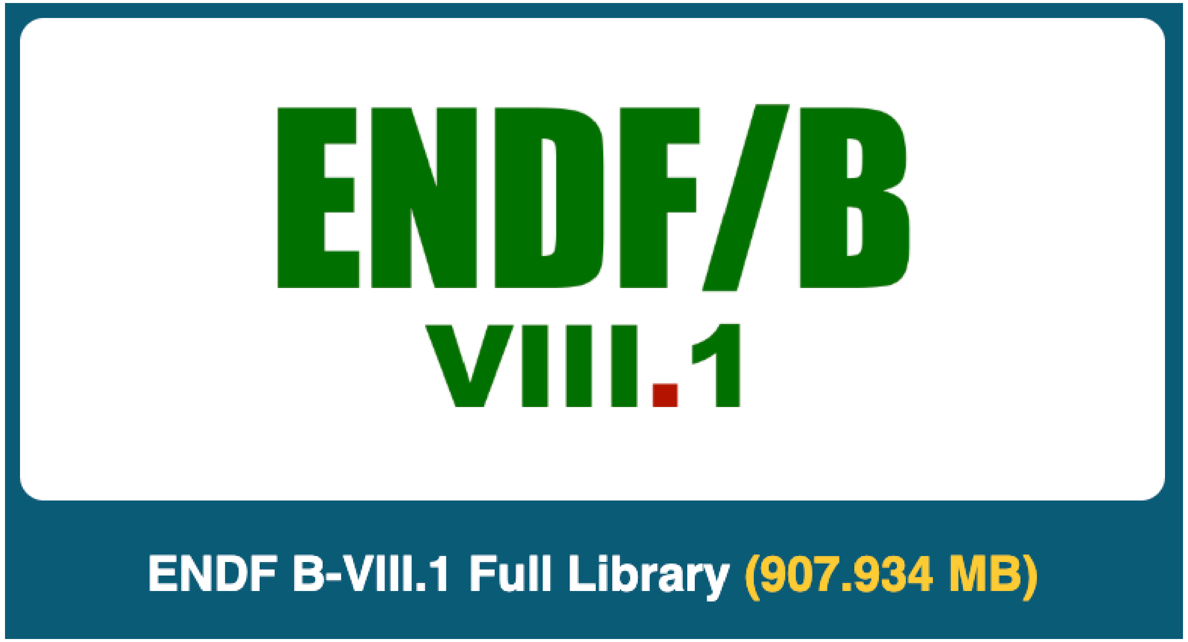


Conclusion and Future Work

Conclusions

- TRU/U^{enr} -Chloride fuel showed the largest k-eff deviation and most sensitivity to nuclear data variations.
- Data discrepancies between ENDF/B-VII.1 and VIII.0 altered fission and capture rates, driving k-eff differences
- Reaction rate differences were due to changes in fissionable materials
- System is sensitivity to fuel interactions with chloride base, when compared to fluoride salt counterpart

Future Work



- Investigate and refine nuclear data for key isotopes to reduce high-energy cross-section uncertainties.

- Expand this analysis to include new ENDF/B-VIII.1 data to compare the effects of new data on MCFR simulation accuracy and reliability.

- Develop improved evaluation methods to better capture energy-dependent effects impacting k-eff in molten-salt systems.

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- [1] National Nuclear Data Center. Evaluated Nuclear Data File (ENDF) nndc.bnl.gov.
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Thank you!
Any Questions?

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