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

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Introduction

Nuclear data libraries are the foundation for the neutron transport codes that simulate physics behavior of nuclear reactors. These libraries have different versions, released periodically with updated nuclear data. Upon investigation a noticeable difference between total cross sections (XS) of specific nuclides was observed when comparing different libraries of nuclear data. This study aims to investigate the potential effects of these differences on the effective multiplication factor (k_{eff}) in advanced reactor simulations. To achieve this objective, we used two different versions of XS libraries to model the same molten salt fast reactor (MSFR) with Serpent, a Monte Carlo neutron transport code. The MSFR design (see Fig. 1) from the EVOL project was selected as the reference model for the study of potential deviation in k_{eff} and other reactor physics parameters such as reaction rate and flux distribution. [1]

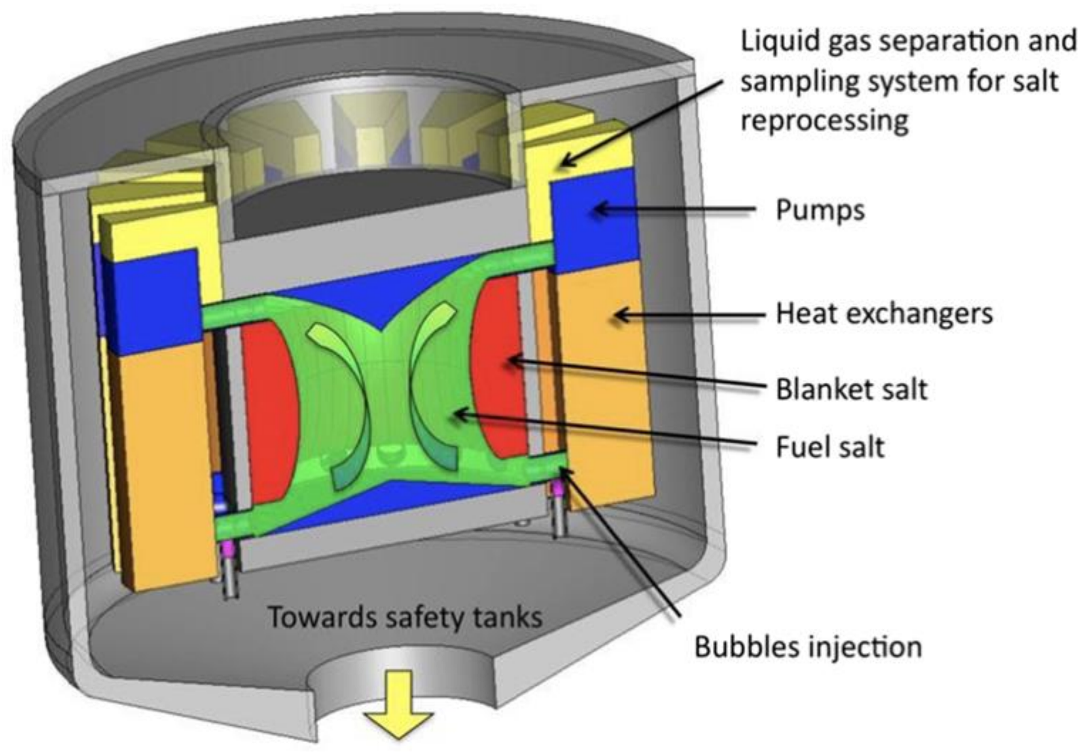

Fig. 1. Schematic MSFR design from the EVOL project [1].

Nuclear Data Discrepancies

Using Evaluated Nuclear Data File (ENDF) libraries ENDF/B-VII.1 and ENDF/B-VIII.0, the pointwise comparison of the total neutron XS showed some difference for uranium isotopes ^{238}U, ^{235}U, ^{233}U and chloride isotopes ^{35}Cl and ^{37}Cl. When collapsing the pointwise data into a groupwise format using NJOY2016, a nuclear data processing code [2], a more discrete difference can be seen in the resonance range as shown in Fig. 2.

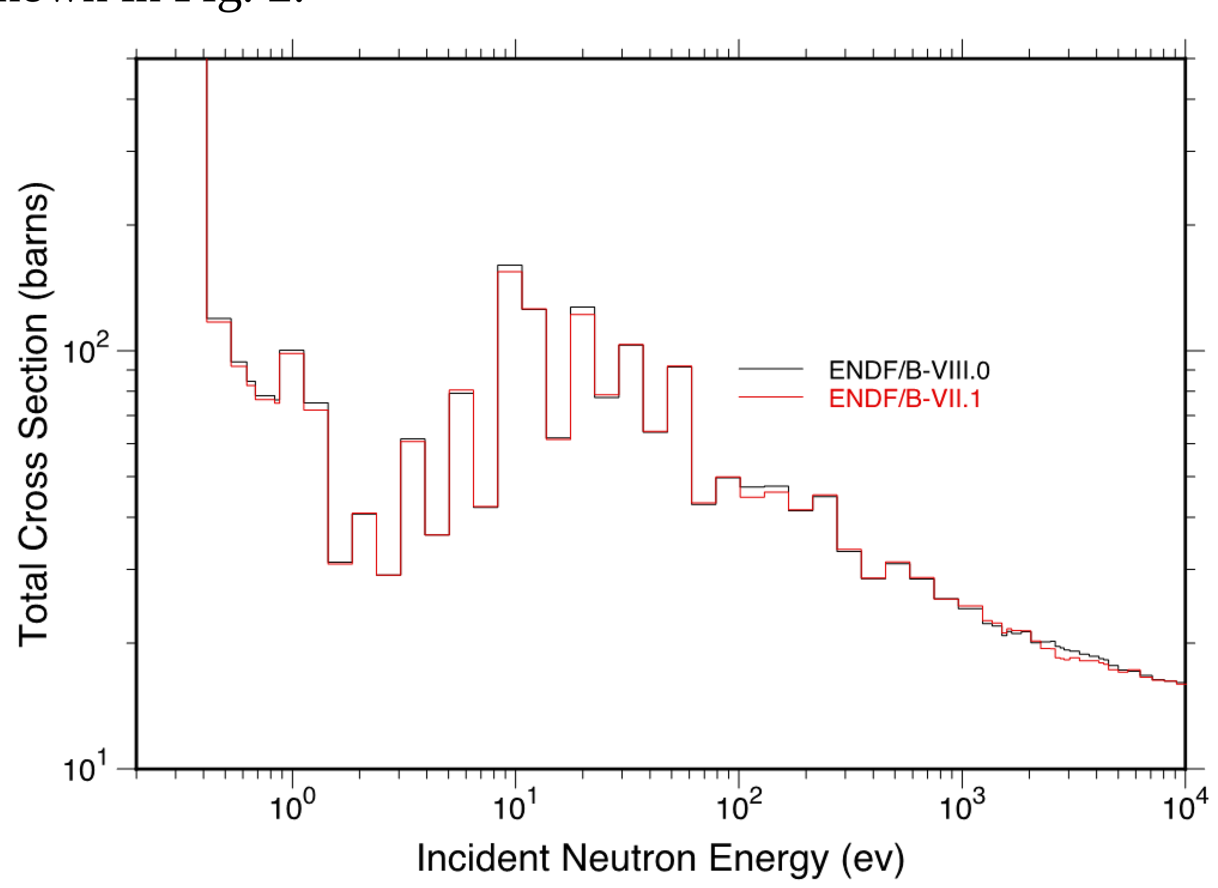

Fig. 2. Comparison of ENDF/B-VIII.0 and ENDF/B-VII.1 libraries for the 239 group total XS of ^{235}U zoomed on the resonance range using NJOY [2].

Reactor Model Specifications

This study looked at different compositions of both fluoride and chloride salt bases in combination with different fissile mixtures of uranium isotopes and transuranic nuclides (TRU). Three chloride based and three fluoride based fuels, totaling six fuel compositions, were considered in the study. Surrounding the fuel salt is the fissile blanket which is composed of a chloride thorium mixture. The geometry of the fuel, blanket, and other core components can be seen in Fig. 1 and Fig. 3.

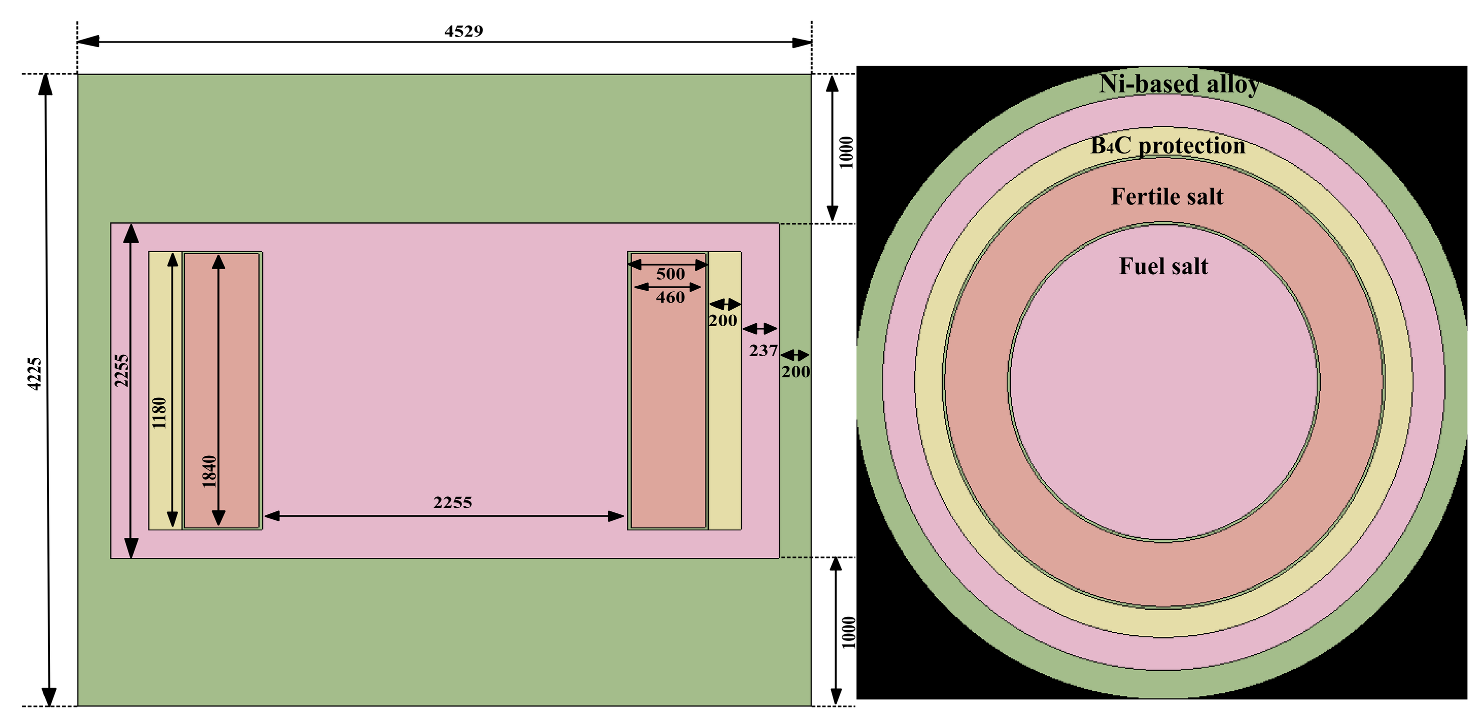

Fig. 3. Axial planar view (left) and radial planar view (right) of the MSFR core model (dimensions given in mm) [3].

Modeling Method and Procedure

Serpent [4] was used to model and simulate the reactor physics that is needed to determine the multiplication factors and other neutronics parameters. We first compared the k_{eff} of each of the six fuel compositions. The preliminary neutronics calculation results indicate the greatest difference in k_{eff} between the two libraries was existed in the TRU/U^{enr}-Chloride mixed fuel. To determine potential causes of this difference, we further investigated other physics measurements such as the flux mesh distribution and the total integral reaction rates.

Results

The largest difference in k_{eff} between ENDF/B-VII.1 and ENDF/B-VIII.0 data libraries was for TRU/U^{enr}-Chloride mixed fuel, and the results and absolute difference are shown in Tabel I.

Table I. The k_{eff} values of the TRU/U^{enr}-Chloride fuel.

Fuel	ENDF/B-VII.1	ENDF/B-VIII.0	Absolute Difference in [pcm]
TRU/ U^{enr} -Chloride	0.974182 ±0.00007	1.01372 ±0.00007	-3953.8

The first set of parameters that we investigated to explain the large difference in the TRU/U^{enr}-Chloride fuel was in the flux and associated energy spectrum in the reactor core. The spatial flux distribution over the entire core is visualized in Fig. 4. The comparison of the flux spectra over the core is shown in Fig. 5. The second set of parameters that was investigated was the various reaction rates in the core. The four reaction rates considered were the total reaction, fission, neutron absorption, and capture rate. The results and the relative difference between the libraries are summarized in Table II. All these results indicate appreciable differences due to the nuclear data discrepancies existed in the two versions of data libraries.

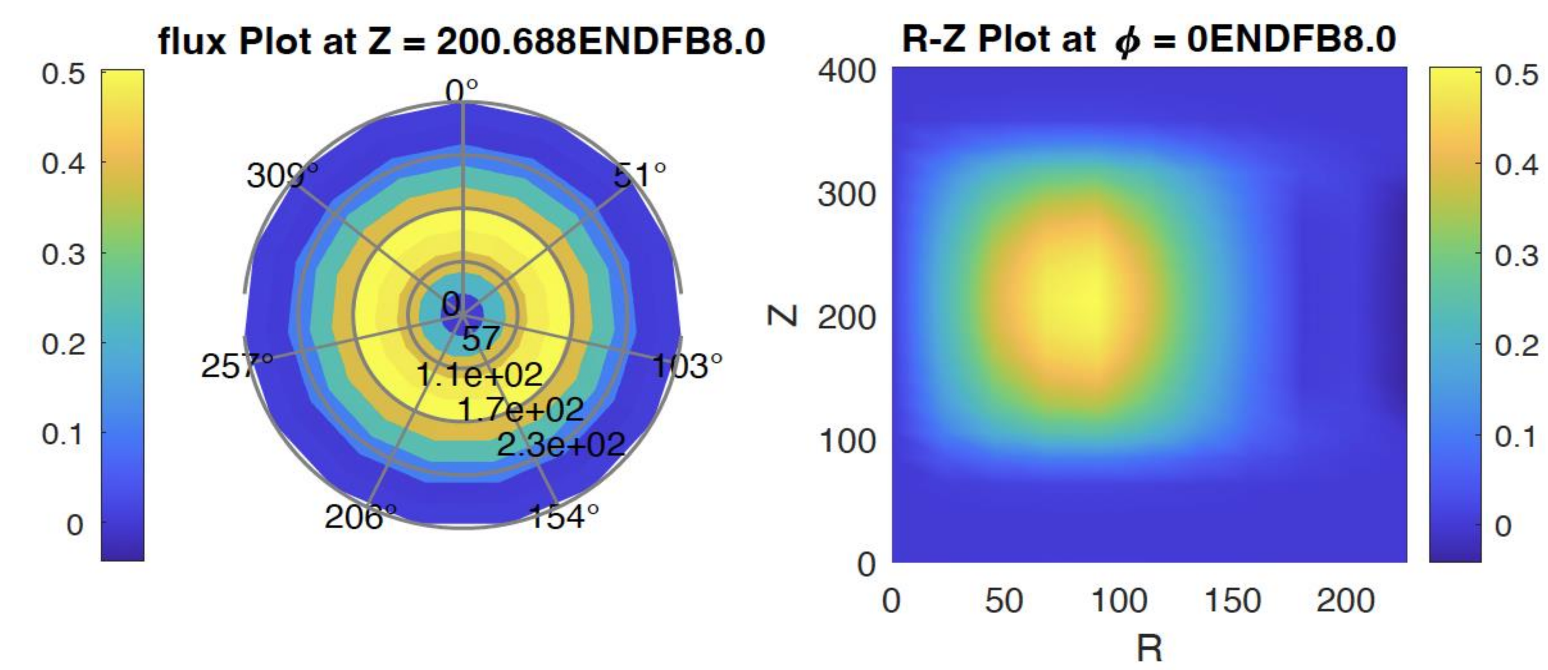

Fig. 4. Radial and axial planar view of the flux distribution modeled with ENDF/B-VIII.0.

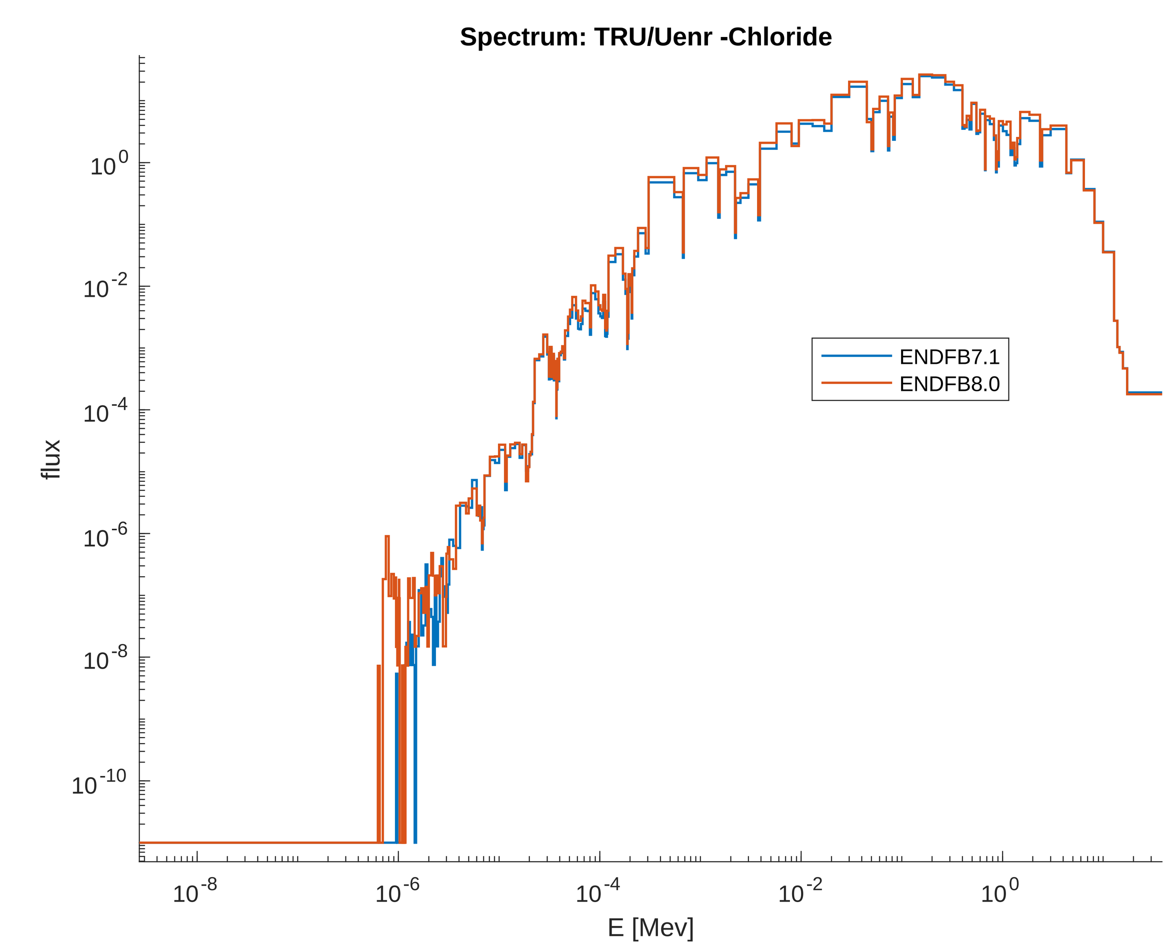

Fig. 5. Comparison of flux spectra with ENDF/B-VII.1 and ENDF/B-VIII.0 data libraries.

Table II. Comparison of neutron reaction rates with ENDF/B-VII.1 and ENDF/B-VIII.0 libraries .

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
Total	48.5788	54.8891	-12.99

Acknowledgments

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