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A Comparative Study on Sodium Void Reactivity Worth for ABTR-250 Core using Different Nuclear Cross Section Libraries

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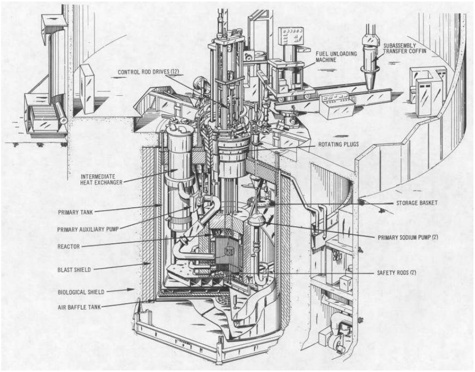
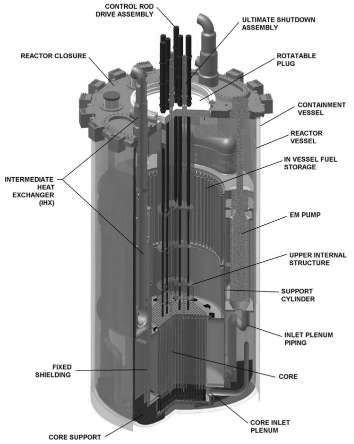
1. Introduction

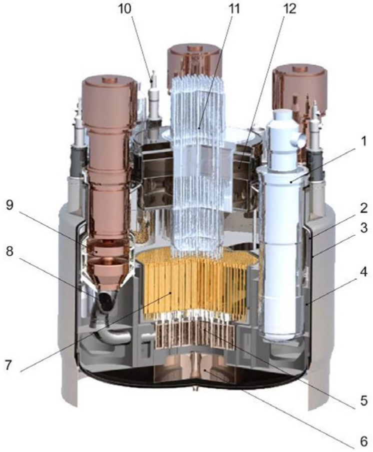
- **Sodium-cooled Fast Reactor (SFR)**
  - A SFR uses liquid sodium as the reactor coolant.
  - Sodium has a high atomic mass number and good neutron economy, which can harden the fast neutron spectrum.
  - It also features high thermal conductivity and boiling point (883°C).
  - SFR has been developed historically since the 1960s.
- **Sodium Void Reactivity Worth (SVR)**
  - The sodium coolant may boil or be lost.
  - This results in a **reactivity change (Δρ)** due to
    - Spectrum hardening: reduced capture reaction -> positive reactivity effect
    - Increased neutron leakage: reduced scattering -> negative reactivity effect
  - The SVR depends on core design, fuel composition, and the **accuracy of cross-sections** in the fast neutron energy range.



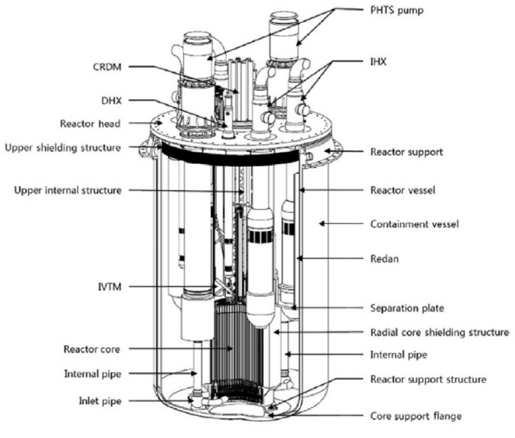
EBR-II
(U.S. ANL from 1964 to 1994)



PRISM
(U.S. BNL)



BN-1200
(Russia)



PGSFR
(Korea, KAERI)

Fig. World-wide SFRs.

1. Introduction

- **250 MWth Advanced Burner Test Reactor (ABTR-250)**
  - A representative SFR designed by Argonne National Laboratory (ANL)
  - ABTR was designed based on ANL's experience in designing, constructing, and operating SFRs.
- **Oak Ridge National Laboratory (ORNL) benchmark report**
  - As a U.S. NRC project, the SNL MELCOR team and the ORNL SCALE team are collaborating to assess the modeling and simulation capabilities for accident progression for non-LWR cases.
  - ORNL reported that the old ENDF/B-V library can lead to a positive sodium void worth; conversely, newer ENDF/B-VI and VII.0 libraries can result in a negative worth for the same configuration.
- **Objectives in this study**
  - We will **compare the void worths using the ENDF/B-V to -VIII.1** libraries for ABTR-250.
  - We also **validate against experimental data** (ZEBRA reactor).

2. Computational Method and Model

- **ENDF/B cross-section library**
  - ENDF: Evaluated Nuclear Data File
    - Originally, ENDF/A was for development and ENDF/B was the official release library since 1968.
  - The library contains evaluated **nuclear reaction data** primarily for incident neutrons, protons, and photons on isotopes from hydrogen ($_{1}H$) to fermium ($_{100}Fm$).
  - The data are processed into various formats. Notably, it is often converted into the **ACE (A Compact ENDF) format** for use in Monte-Carlo codes (MCNP, **Serpent**, OpenMC, McCARD, … etc.).

Table. History of ENDF/B versions.

Version	Release year	No. of Materials	Features
ENDF/B-I	1968	58	First version
ENDF/B-V	1979 (V.2 1985)	201	ANL used this with DIF3D/REBUS-3 code system.
ENDF/B-VI	1990 (VI.8 2001)	328	Energy release per fission included & Improvements in fast neutron XS
ENDF/B-VII.0	2006	393	Improvements XS and delayed neutron fraction in fast spectrum
ENDF/B-VII.1	2011	423	Improved actinide evaluations for isotopes of U, Np, Pu, and Am in fast reactors
ENDF/B-VIII.0	2018	557	New evaluations for key isotopes (^{1}H, ^{16}O, ^{56}Fe, ^{235}U, ^{238}U, and ^{239}Pu) & Major improvements in the fast neutron energy region
ENDF/B-VIII.1	2024	557	Newest release

2. Computational Method and Model

- **Serpent 2**
  - Continuous-energy Monte-Carlo neutron transport code, developed by VTT, Finland
  - Serpent uses **ACE format** cross-section libraries.
  - We use Serpent for all transport calculations in this study.
  - To achieve accurate Monte-Carlo uncertainty levels, we set
    - 100 inactive and 4,000 active cycles with 400,000 histories per cycle,
    - achieving a standard deviation in the k_{eff} of $\sim$ 2 pcm.

2. Computational Method and Model

- **ABTR-250**
  - 250 MWth **SFR** designed by ANL
  - It was designed to demonstrate transmutation of **transuranics (TRU)**.
  - ABTR core was designed to operate in **4-months cycles** using **U/TRU-Zr10% metallic fuel**, with a TRU content of ~20 wt%.

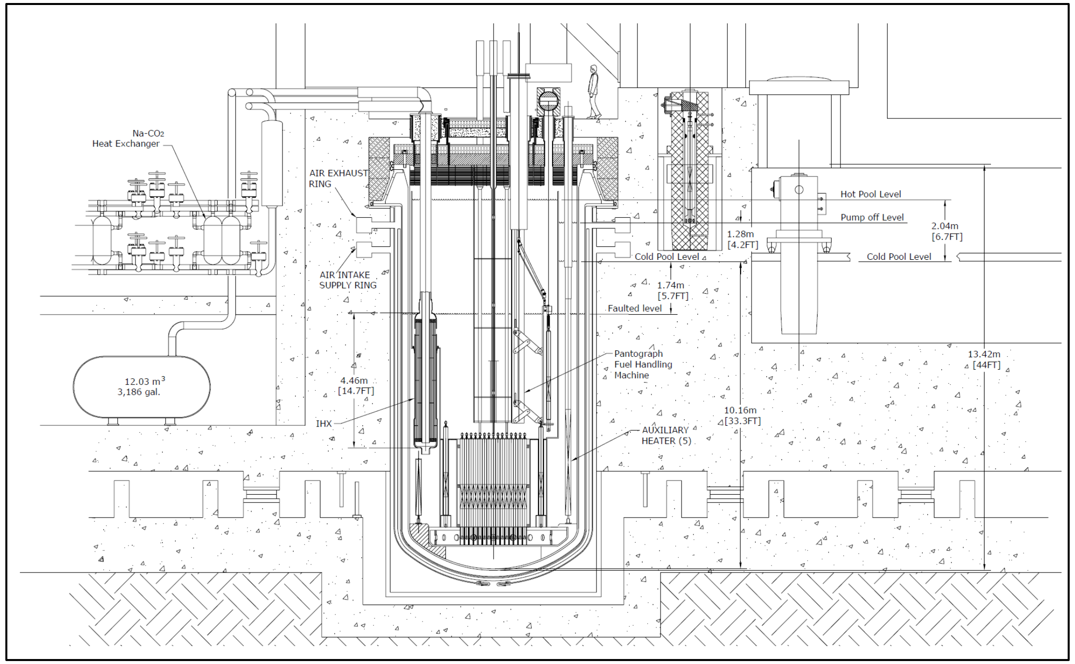


Fig. Elevation view of the primary system of ABTR-250.

Table. Key characteristics of ABTR-250.

Design parameter	Value
Reactor power (MWth)	250
Fuel material	U/TRU-Zr10%
TRU fraction	~ 20 wt%
Coolant	Na
Structural material	HT-9
Fuel slug diameter (cm)	0.7002
Fuel rod diameter (cm)	0.8114
Fuel pin pitch (cm)	0.9134
Active core height (cm)	84.4108
Number of pins per FA	217
Assembly pitch (cm)	14.685

2. Computational Method and Model

- **ABTR-250 core design**
  - ANL originally designed the ABTR.
    - Code: DIF3D (nodal)
    - XS Library: ENDF/B-V.2
    - Geometry: homogeneous model
- **Benchmark calculations at BOEC**
  - INL:
    - Code: Serpent 2 (Monte-Carlo)
    - XS Library: ENDF/B-VII.1
    - Geometry: heterogeneous model
  - ORNL:
    - Code: SCALE-Shift (Monte-Carlo)
    - XS Library: ENDF/B-VII.1
    - Geometry: heterogeneous model

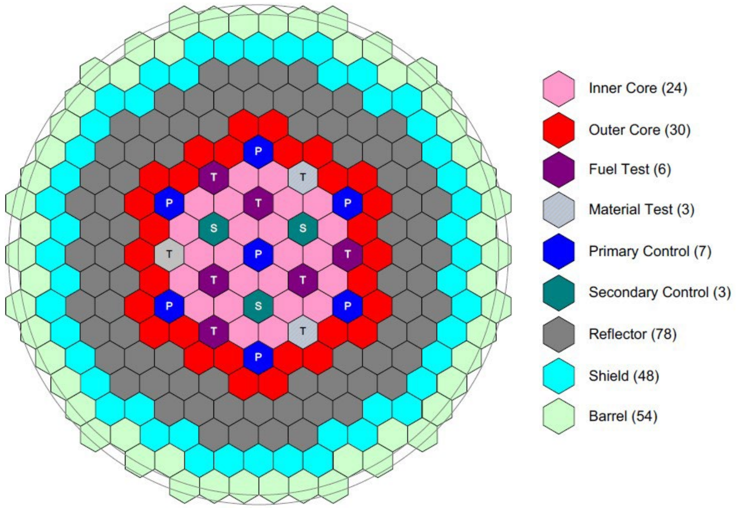


Fig. ANL homogeneous model. (for nodal code)

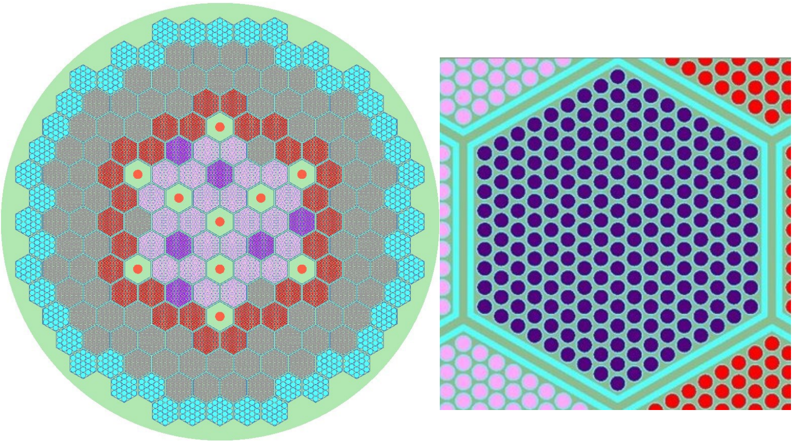


Fig. ORNL heterogeneous model. (for Monte-Carlo code)

Table. Eigenvalue comparison (ORNL report).

Code	**XS Library**	k_{eff}	**Difference (pcm)**
Serpent	ENDF/B-VII.1	1.03055 ± 0.00002	(ref)
SCALE	ENDF/B-VII.1	1.03019 ± 0.00004	-37 ± 4
SCALE	ENDF/B-VIII.0	1.03152 ± 0.00004	97 ± 4

Table. Void worth comparison (ORNL report).

Code	**XS Library**	**Void worth**	β_{eff} **(pcm)**
DIF3D	ENDF/B-V.2	1.75 $	330
SCALE	ENDF/B-VII.1	-0.451 $	331 ± 5

in ORNL report

3. Results (Model Verification)

- **For comparison with benchmark calculations, the eigenvalue were compared with ORNL and INL reports.**
  - We compared our calculations with ORNL and INL reports using the same XS library (ENDF/B-VII.1).
  - Our calculation differed by -14 pcm from ORNL and +22 pcm from INL, which can be considered sufficient agreement.

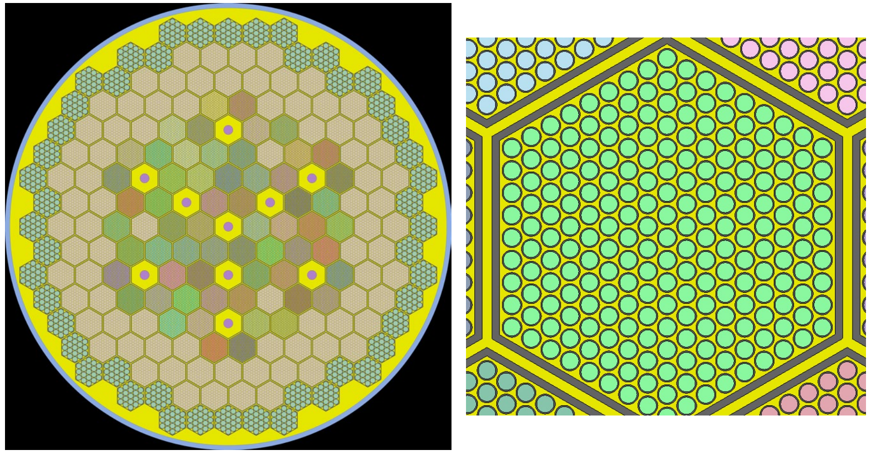


Fig. Our Serpent 2 modeling.

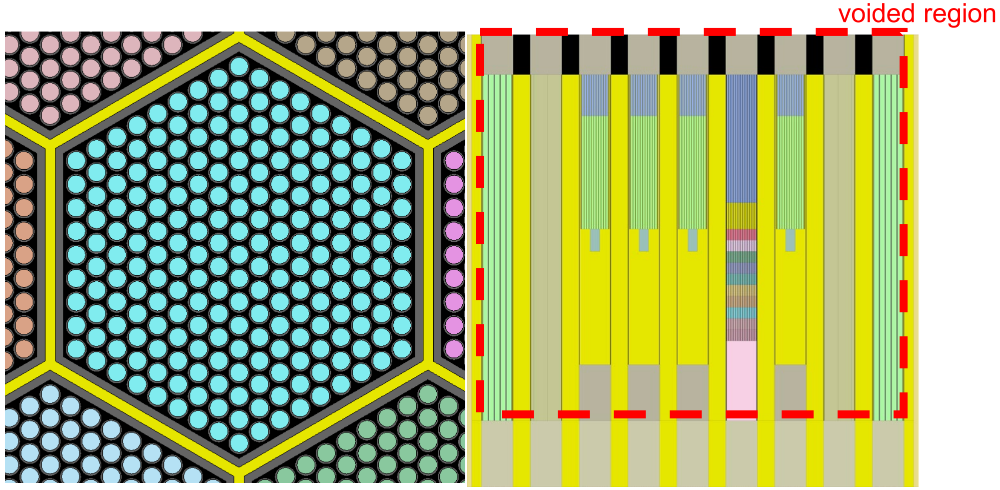


Fig. Sodium coolant voiding modeling.

Table. Eigenvalue comparison with previous reports.

ENDF/B-VII.1	ORNL	INL	This study
Computer code	Shift	Serpent 2	Serpent 2
k_{eff}	1.03019	1.03055	1.03033
Difference (pcm)	-14	22	(ref)
β_{eff}	331	330	336

3. Results (Model Verification)

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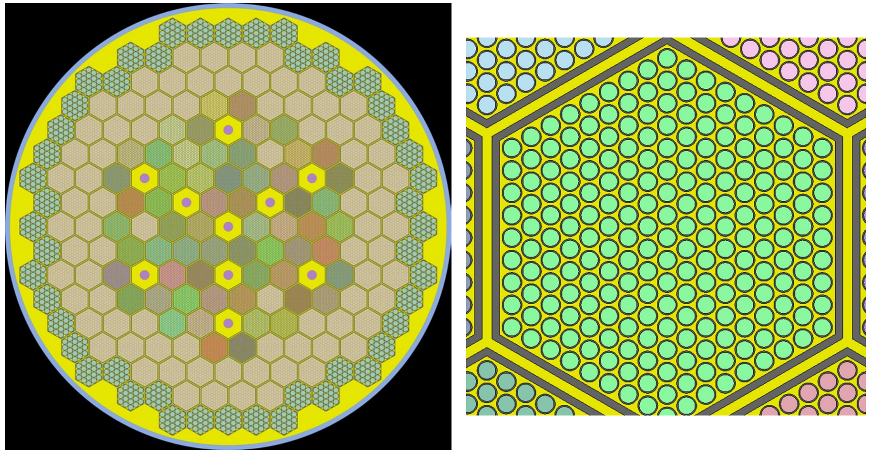
    - We compared our calculations with ORNL and INL reports using the same XS library (ENDF/B-VII.1).
    - Our calculation differed by -14 pcm from ORNL and +22 pcm from INL, which can be considered sufficient agreement.
    - Our SVR was almost the same value as ORNL's, which used the same ENDF/B-VII.1 library.

Table. SVR comparison with previous reports.

	ANL	ORNL	This study
Computer code	DIF3D	Shift	Serpent 2
ENDF/B version	V.2	VII.1	VII.1
SVR (pcm)	577	-149	-164



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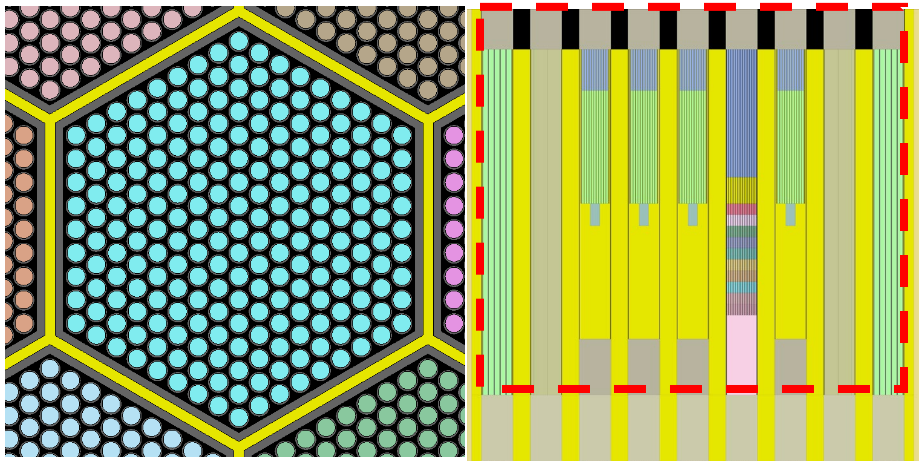

Fig. Sodium coolant voiding modeling.

3. Results (Eigenvalue and Void Worth)

- **We performed calculations using ENDF/B libraries from V to the latest VIII.1, under the same geometry and material conditions.**
- **Eigenvalues:**
  - Largest VI.8 / Smallest VII.1
- **Sodium void worths**
  - The ENDF/B-V was the only one to show a positive value, consistent with the ANL report.

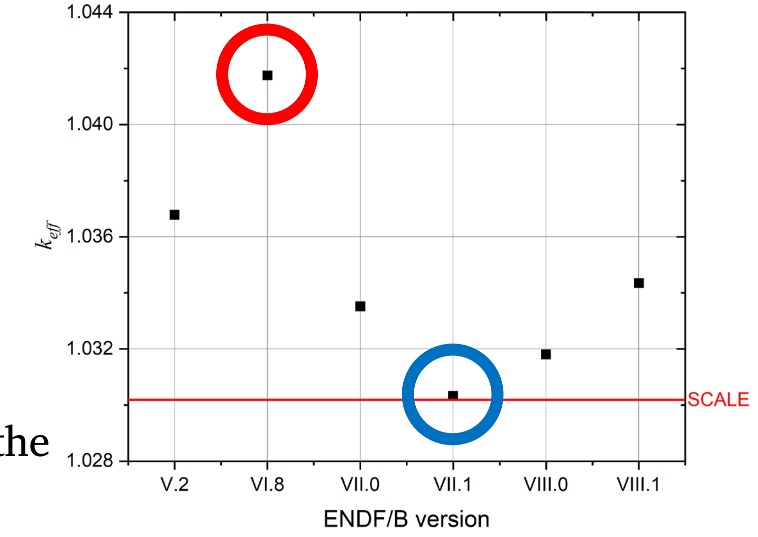

Fig. Serpent 2 k_{eff} results.

Table. Serpent 2 results and differences with SCALE-Shift for k_{eff}, β_{eff}, and void worths.

ENDF/B version	V	VI.8	VII.0	VII.1	VIII.0	VIII.1
k_{eff}	1.03678	1.04175	1.03352	1.03033	1.03181	1.03435
β_{eff} (pcm)	-	338	336	336	336	336
SVR (pcm)	182	-30	-103	-164	-107	-68
Differences with SCALE-Shift (VII.1)						
k_{eff} (pcm)	659	1156	333	14	162	416
β_{eff} (%)	-	2.22%	1.53%	1.57%	1.53%	1.42%
SVR (%)	-222%	-80%	-31%	10%	-28%	-54%

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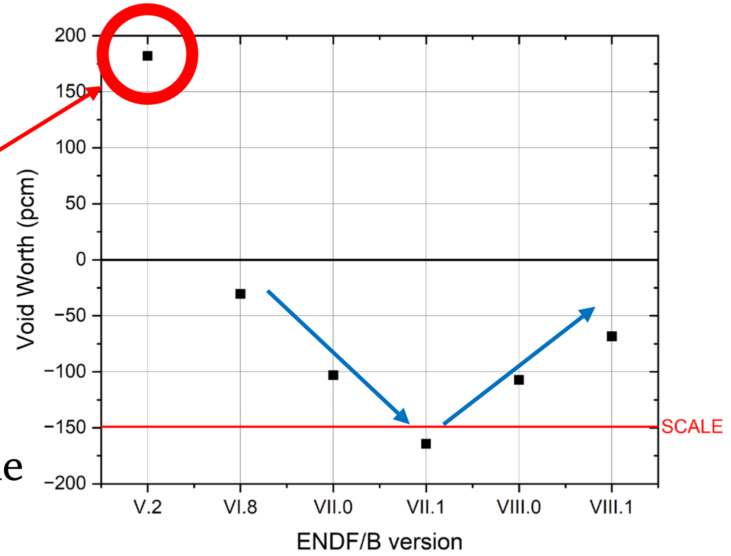

Fig. Serpent 2 SVR results.

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3. Results (Neutron Spectrum)

- The above eigenvalue and void worth results are caused by differences in reaction rates according to the neutron flux energy spectrum among the libraries.
- It can be seen that the ABTR-250 core has a spectrum in the epithermal-fast region.

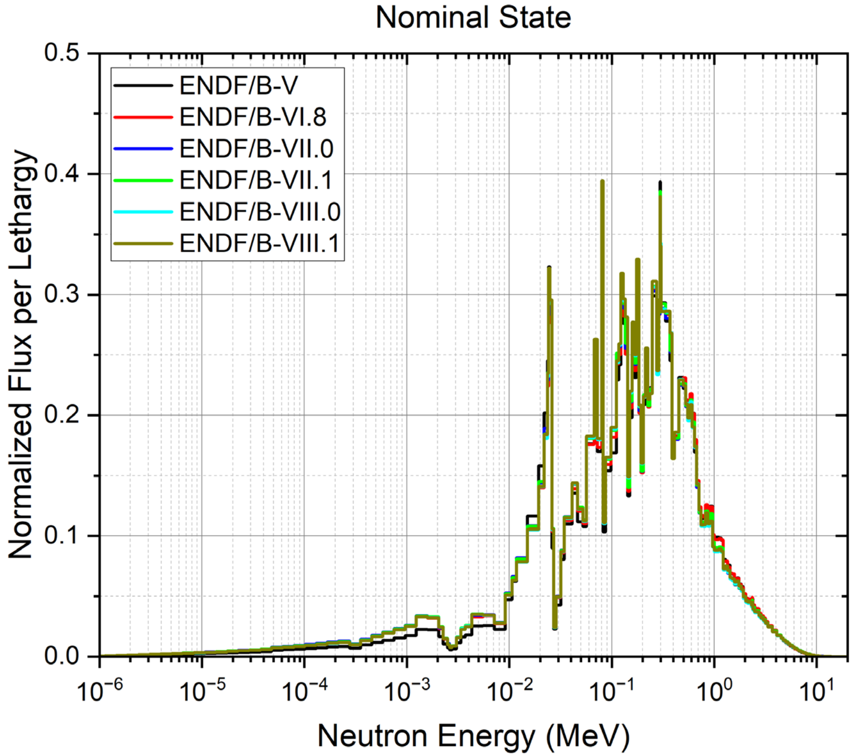

Fig. Comparison of neutron flux spectra for different ENDF/B libraries.

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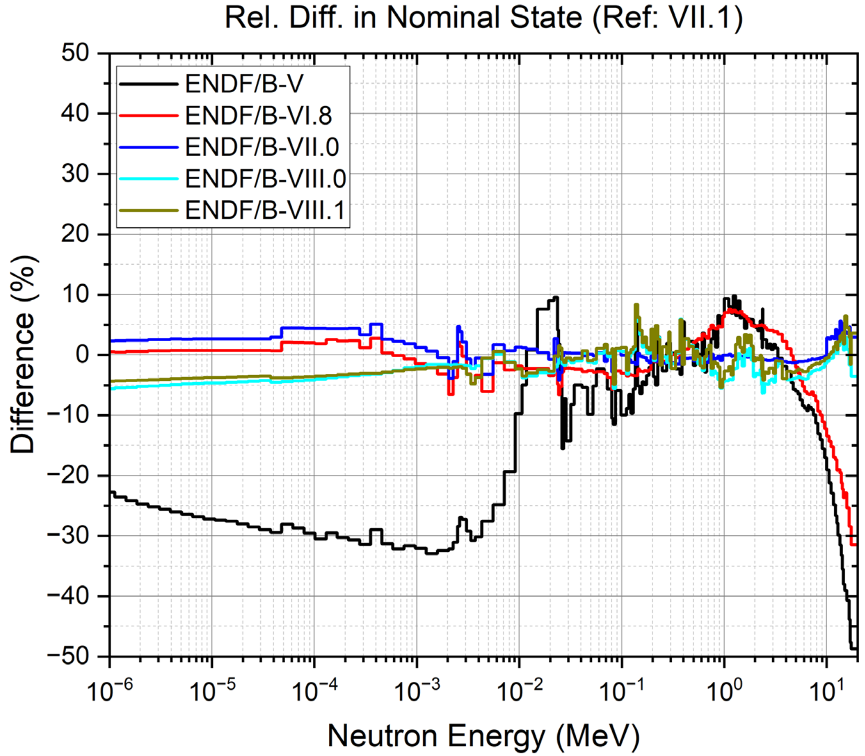
- **The right figure shows the relative differences from VII.1.**
  - The old V and VI.8 versions have large differences in fast region.
  - The VII.0 version shows a similar spectrum within ~1.5% difference.
  - The VIII.0 and VIII.1 versions show little differences in fast region, within ~3% on average.



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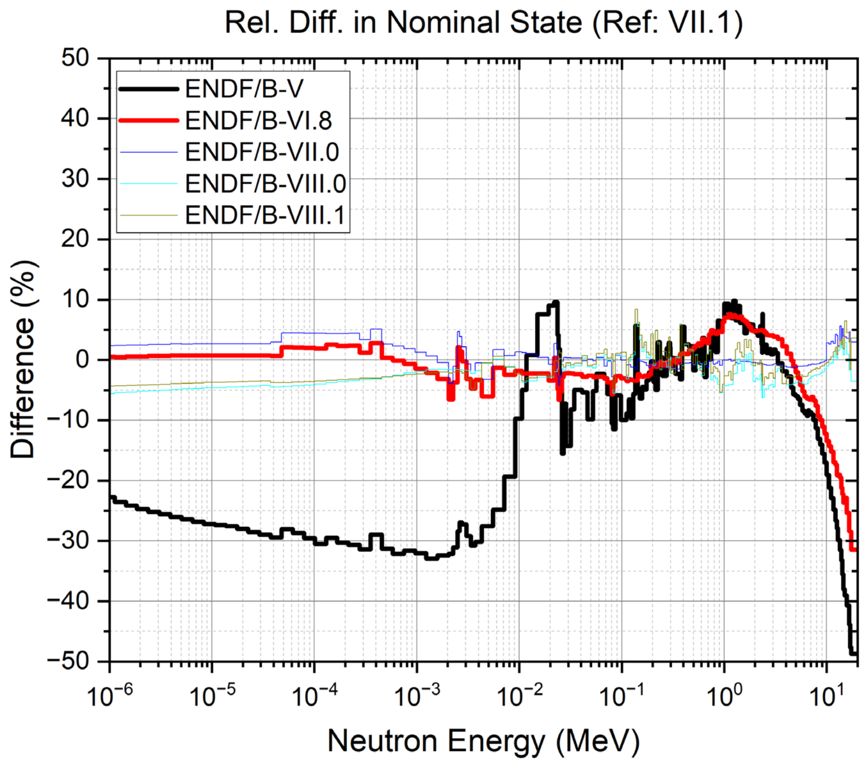

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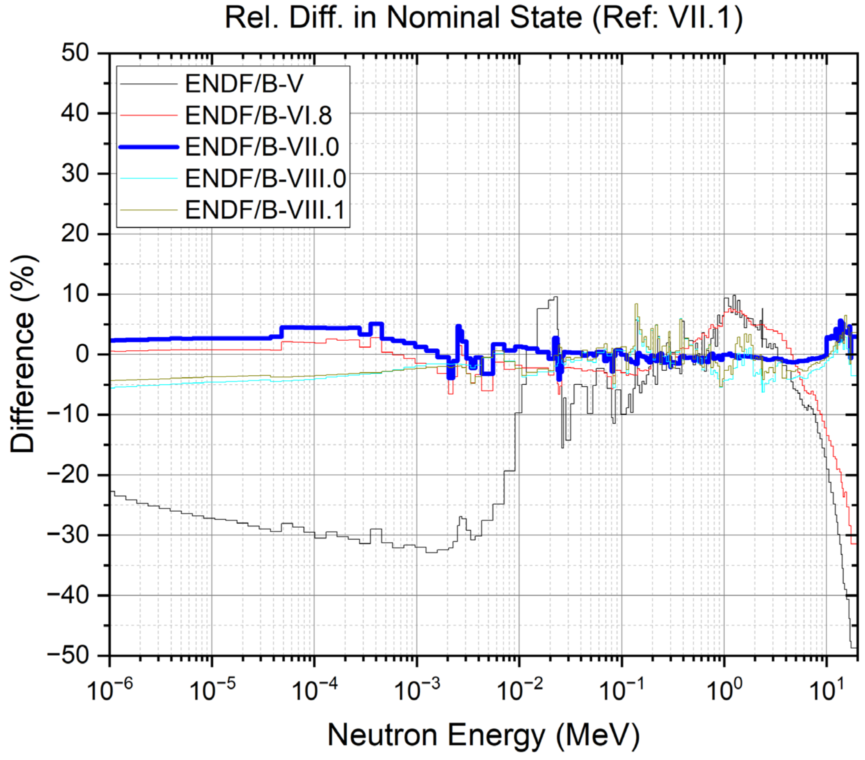

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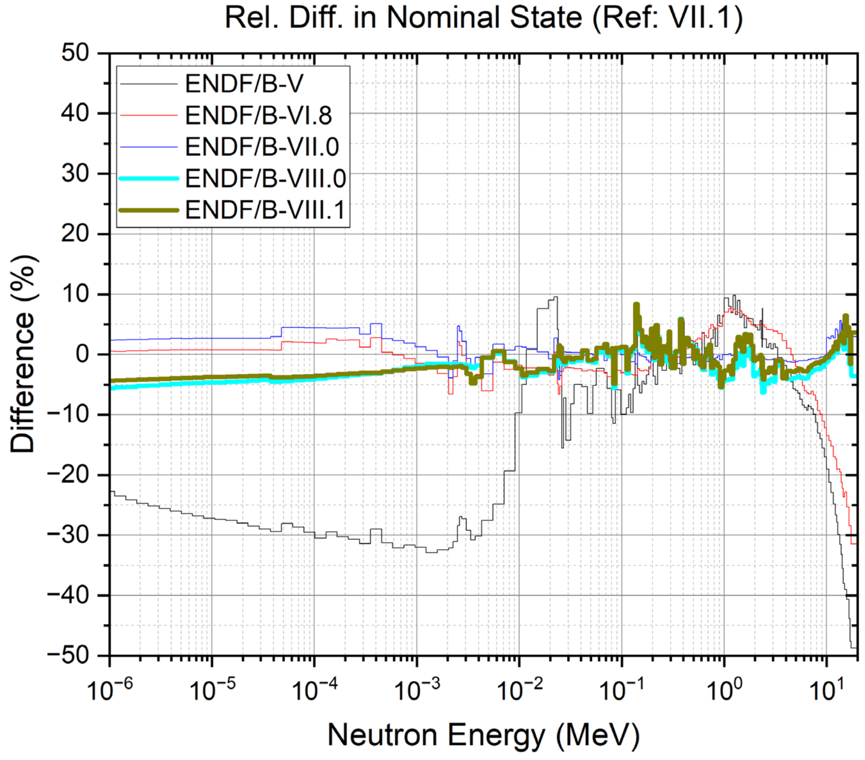


Fig. Comparison of neutron flux spectra for different ENDF/B libraries.

3. Results (Neutron Balance)

- **Neutron balance method for reactivity decomposition is based on the integral neutron reaction rates.**
  - Neutron balance equation:

$$-\int \nabla D \nabla \phi \, dV + \int \Sigma_c \phi \, dV + \int \Sigma_f \phi \, dV = \frac{1}{k_{eff}} \int \nu \Sigma_f \phi \, dV + \int \Sigma_{n,2n} \phi \, dV$$

  - Reaction rates: **Leakage** (L), **Capture** (C), **Fission** (F), **n,2n** (N), **Fission neutron production** (P)
  - Reactivity normalized by fission neutron production:

$$\rho(L, C, F, P) = 1 - \frac{L + C + F - N}{P} = 1 - (l + c + f - n)$$

Table. Comparison of changes in normalized reaction rates from nominal to voided states.

V - N (pcm)	**V**	**VI.8**	**VII.0**	**VII.1**	**VIII.0**	**VIII.1**
Leakage (l)	**2152**	2186	2214	**2315**	2291	2272
Capture (c)	**-2288**	-2115	-2057	**-2097**	-2139	-2148
Fission (f)	-35	-28	-40	-39	-32	-42
n,2n (n)	11	13	14	14	13	14
Reactivity change	-182	30	103	164	107	68

- **Zero Energy Breeder Reactor Assembly (ZEBRA)**
  - SFR operated from 1962 to 1982 by the United Kingdom Atomic Energy Authority (UKAEA) to perform critical experiments
  - ZEBRA-LMFR-EXP-001 experiment had the purpose to evaluate the nominal state and the sodium-voided state.

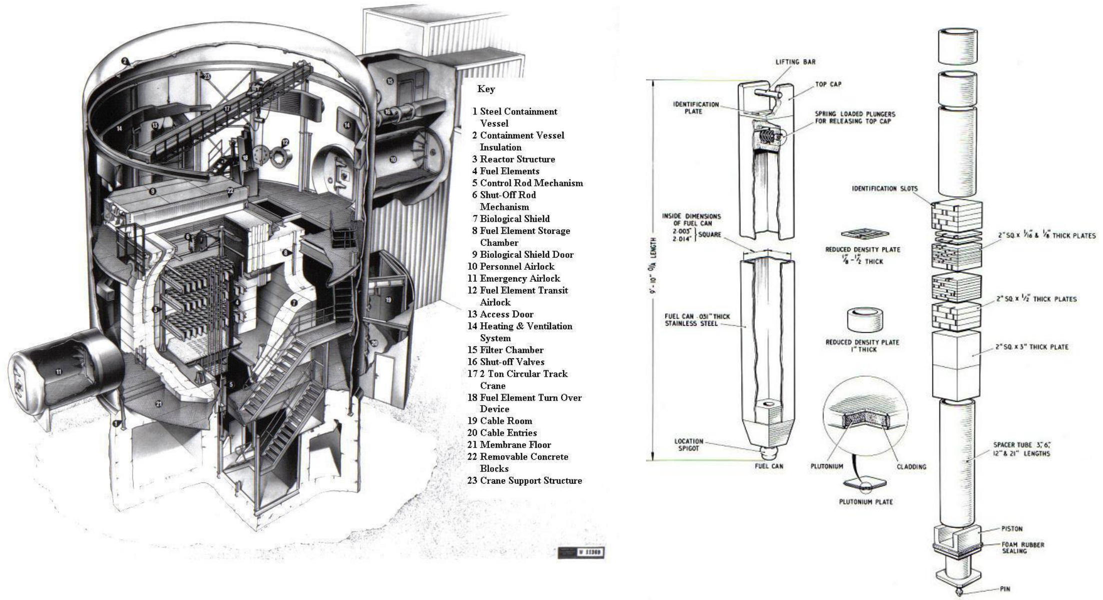


Fig. ZEBRA facility and diagram of a fuel element.

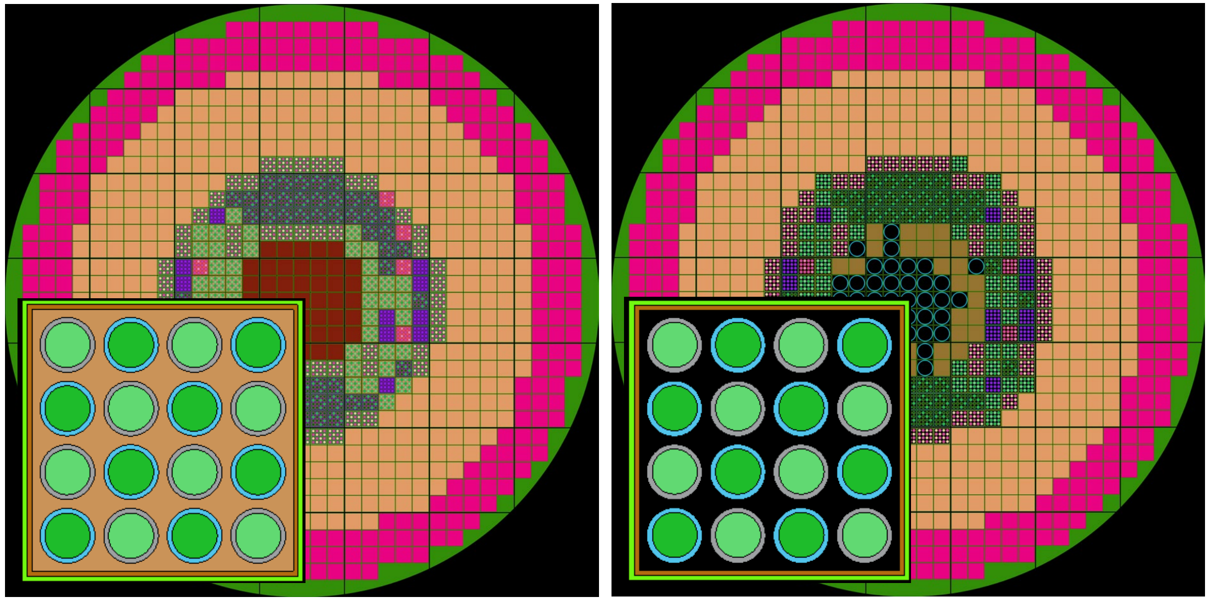
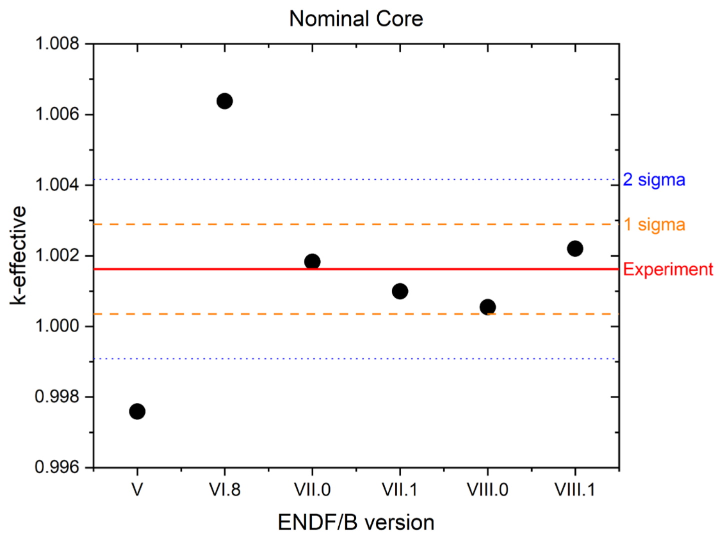


Fig. Serpent 2 geometry models of nominal and voided cores. (Black: no material)

- **Zero Energy Breeder Reactor Assembly (ZEBRA)**
  - The k_{eff} for the old **ENDF/B-V and VI.8** are outside 2 σ of the experimental data.
  - The k_{eff} **from VII.0 onward** are within 2 σ of the experimental data.
  - The k_{eff} for **newest VIII.1** is only within 1 σ of both experiment cases.

Table. Serpent 2 k_{eff} results in ZEBRA cases.

Nominal	**V**	**VI.8**	**VII.0**	**VII.1**	**VIII.0**	**VIII.1**
Experiment			1.00162 ± 127 pcm			
k_{eff}	0.997585	1.00638	1.00183	1.00099	1.00054	1.00220
Diff. (pcm)	-404	472	21	-63	-108	58
Voided						
Experiment			1.00132 ± 127 pcm			
k_{eff}	1.00039	1.00633	0.999884	0.998884	0.999506	1.00176
Diff. (pcm)	-93	497	-143	-244	-181	44



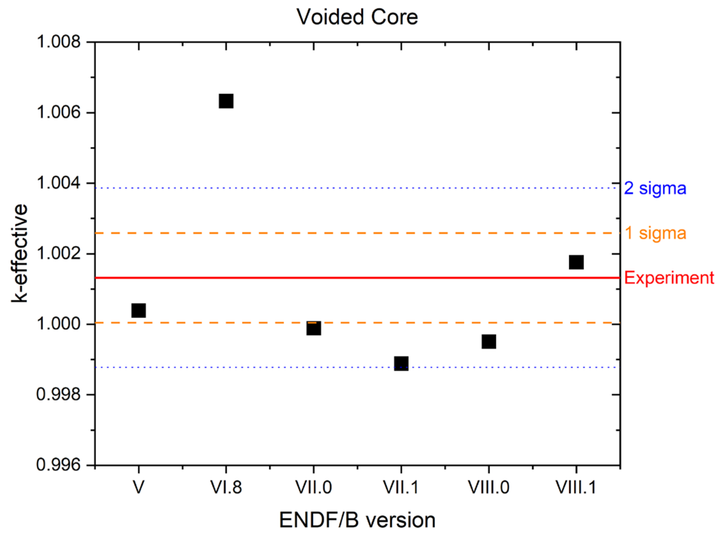
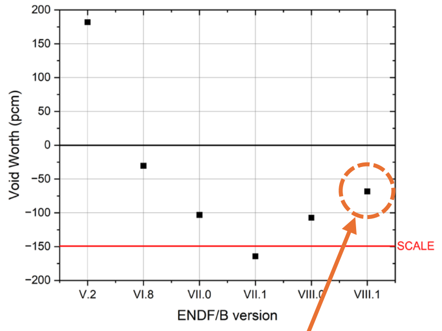


Fig. Comparison of k_{eff} for ENDF/B libraries in ZEBRA.

4. Conclusions

- **We compared the void worths using the ENDF/B cross-section libraries V - VIII.1 for ABTR-250.**
- **Only ENDF/B-V showed a positive void worth, and others showed negative values.**
  - ENDF/B-V: +182 pcm
  - ENDF/B-VII.1 showed the lowest value (-164 pcm), and the newest ENDF/B-VIII.1 showed a larger magnitude (-68 pcm).



- **The neutron spectrum and cross-section change across library generations.**
  - The old ENDF/B-V and VI.8 libraries showed very different spectra than newer ones.
  - The newest ENDF/B-VIII.0 and VIII.1 libraries expand the energy range.
- **The experimental reactor (ZEBRA) supports the reliability of libraries from VII.0 onward.**
  - Old V and VI.8 were not within 2 σ of the experimental data.
  - Newest VIII.1 was within 1 σ of the experimental data.

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Thanks for your listening!

Appendix (Cross-section)

- **We compared the major cross-sections of ENDF/B-V, VII.1, and VIII.1 for ABTR-250.**
- **U-235 (n,f) / Pu-239 (n,f) / Na-23 (n,tot)**

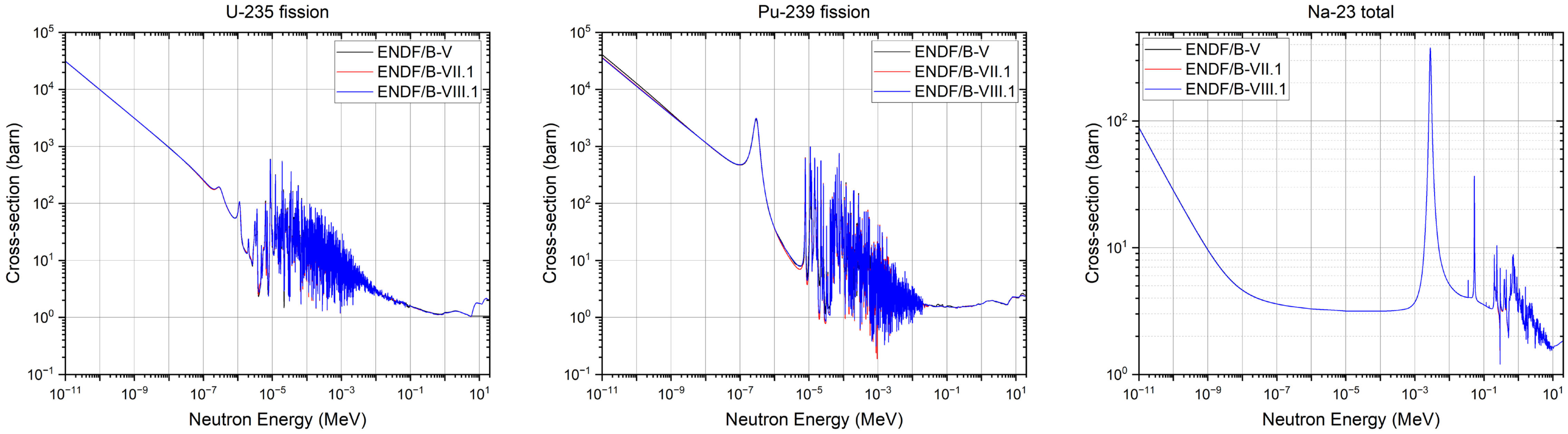
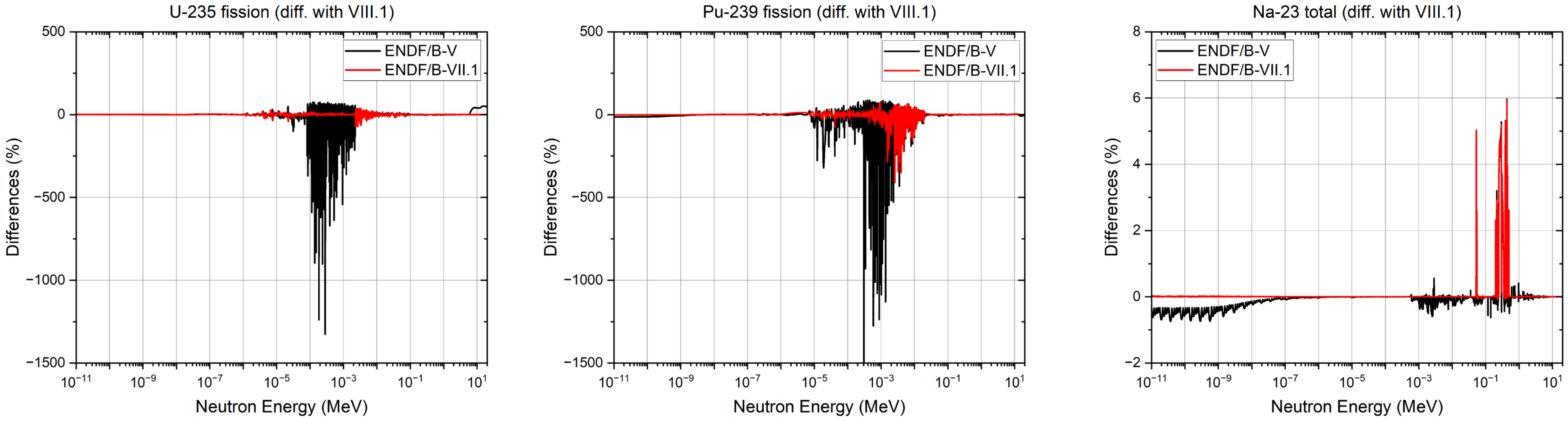


Fig. Comparison of major XSs for ENDF/B-V, VII.1, VIII.1.

Appendix (Cross-section)

- **We compared the major cross-sections of ENDF/B-V, VII.1, and VIII.1 for ABTR-250.**
  - Difference with VIII.1 $= \frac{\text{XS}-\text{VIII.1}}{\text{VIII.1}} \times 100\ (\%)$
  - For ^{235}U fission, ENDF/B-V has an average difference of ~44% in range from 10^{-5} to 10^{-2} MeV, and ENDF/B-VII.1 shows a similar but ~6% average difference from ~2 keV to 150 keV.
  - ^{239}Pu fission XSs show larger differences than ^{235}U.
  - ^{23}Na total XSs show small differences, with a maximum of ~6%.



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