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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 Reactors (SFRs) are a reliable candidate for a Gen-IV reactor type with a fast neutron spectrum [1]. A key safety parameter in SFRs is the sodium void worth, which describes the change in reactivity if the sodium coolant is lost. This phenomenon can occur from boiling due to the relatively low boiling point of Sodium [2]. This can lead to positive reactivity feedback, which poses a threat to reactor safety.

Argonne National Laboratory (ANL) reported the neutronic analyses of the 250 MWth Advanced Burner Test Reactor (ABTR-250), a representative SFR design [3]. A benchmark study by Oak Ridge National Laboratory (ORNL) concluded that the core has a negative sodium void worth [4]. However, the original design report from ANL calculated a positive value. The ORNL report suggested this discrepancy was because ANL used the older ENDF/B-V cross-section library.

This study will perform a detailed analysis of the sodium void worth for the ABTR-250 core by using various ENDF/B cross-section libraries. We will use the Monte-Carlo code Serpent 2 to compare the results obtained from these libraries, ranging from the older version V to the latest version VIII.1 [5–10].

2. Computational Methods and Model

All neutron transport calculations are performed with Serpent 2, a Monte Carlo code. We compare results while varying only the ENDF/B cross-section libraries.

2.1 Serpent 2

Serpent 2 is a continuous-energy Monte Carlo neutron transport code developed at the VTT Technical Research Centre of Finland [11]. Serpent 2 can tally neutron energy using detector definition (*det* card) and reaction rate (*set arr* card) to analyze neutron spectrum and balance in both nominal and voided coolant states.

To ensure accurate Monte Carlo transport calculation, we set 100 inactive cycles and 4,000 active cycles with 400,000 histories per cycle, achieving a standard deviation in the effective multiplication factor (k_{eff}) of about 2 pcm.

2.2 ENDF/B library

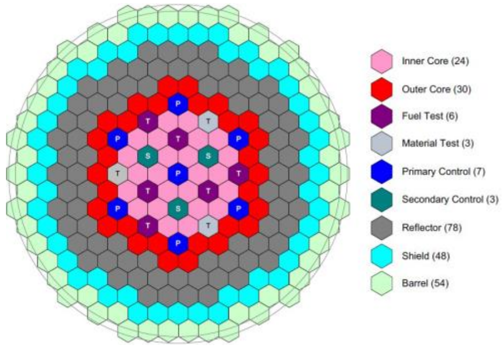
Serpent 2 uses ACE-format cross sections, a format also commonly used by Monte-Carlo codes, like MCNP and OpenMC. Our lab uses the NJOY code to produce ACE-format data from ENDF/B libraries [12].

We will analyze the void worth using the ENDF/B libraries ranging from version V to the latest version VIII.1, under the same geometry and material conditions.

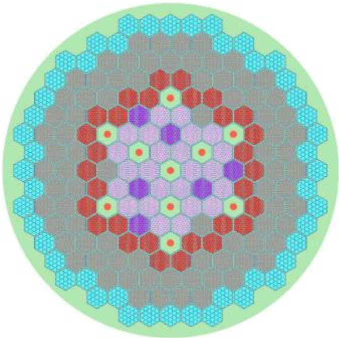
2.3 ABTR-250

To evaluate the sodium void worth of various ENDF/B libraries, the ABTR-250 was selected as the SFR core design. ABTR-250 was designed by ANL and is characterized by using transuraniscs (TRU) from spent LWR fuel as fuel. The fuel is U/TRU-Zr10% metallic fuel with a TRU content of approximately 20%, designed for a 4-month operating cycle.

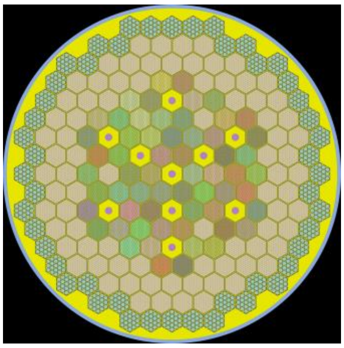
The original design report considered homogenized fuel assemblies [3], while Idaho National Laboratory (INL) recently performed an analysis using a heterogeneous design with heterogeneous assemblies and structurally homogenized plugs [13]. This design was based on the ABTR benchmark report, which provided detailed design parameters [14]. ORNL adopted the heterogeneous design based on the INL report [4]. Following these approaches of INL and ORNL, we also adopted the heterogeneous design according to the ABTR benchmark specification for our calculations. Figure 1 shows both the homogeneous and heterogeneous models of the ABTR core.



(a) Homogeneous model of ABTR-250 [3]



(b) Heterogeneous model of ABTR-250 [4]



(c) Our Serpent 2 modelling

Figure 1. ABTR core radial configurations

The neutronic analysis of ABTR-250 was reported by ANL using the diffusion code DIF3D with the ENDF/B-V library [3], by INL using the ENDF/B-VII.1 library and Serpent 2, and by ORNL using the ENDF/B-VII.1 continuous-energy library and Shift, a Monte-Carlo transport solver module in the SCALE code system [4]. ANL and ORNL performed analyses of the sodium void worth, and this study conducts a comparative analysis with these results, only at the beginning of equilibrium cycle (BOEC).

3. Results

3.1 Model Verification

For comparison with benchmark calculations, the eigenvalue results were compared with the ORNL and INL reports and are summarized in Table I. To verify the core design, all cases employed the ENDF/B-VII.1 cross-section library with Monte Carlo transport codes. Our calculation showed differences of -14 pcm from ORNL and 22 pcm from INL using same Serpent 2 code, which can be considered sufficient agreement.

Table I. Eigenvalue comparison with previous reports

ENDF/B-VII.1	ORNL[4]	INL[13]	This study
Computer code	Shift	Serpent	Serpent
k_{eff}	1.03019	1.03055	1.03033
Difference (pcm)	-14	22	(ref)
β_{eff} (pcm)	331	330	336

The methodology for calculating the sodium void worth was described the ABTR design report [3]. Our results were compared with those of previous works in Table II. The results showed almost the same worth as ORNL, which used the same ENDF/B-VII.1 library. However, ANL, which had used the older ENDF/B-V library, reported the opposite result with a positive worth. This dramatic result will be examined in the following section to determine whether it is due to the diffusion code or the changes in the cross-section library.

Table II. SVR comparison with previous reports

	ANL[3]	ORNL[4]	This study
Computer code	DIF3D	Shift	Serpent
ENDF/B version	V	VII.1	VII.1
Void worth (pcm)	577	-149	-164

3.2 Eigenvalues and Void Worths

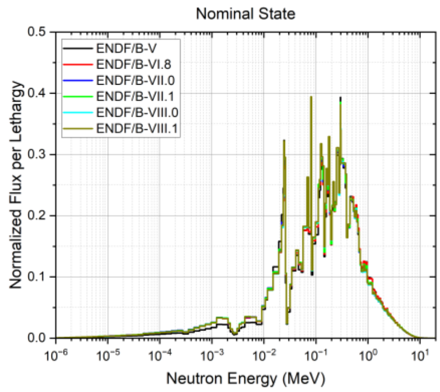
In this section, the eigenvalue, effective delayed neutron fraction (β_{eff}), and sodium void worths obtained with different ENDF/B libraries are compared. Table III summarizes the results calculated with Serpent 2 and the differences of these results compared with the SCALE/Shift calculations as a reference.

The ENDF/B-VI.8 library yields the largest k_{eff}, while the smallest k_{eff} is calculated with VII.1. The effective delayed neutron fraction shows nearly identical values across all libraries, except for the result for the older V version was not printed in the Serpent output. Most of the calculated sodium void worths were negative. Notably, the ENDF/B-V was the only one to show a positive value, which is consistent with the results from ANL report. The various libraries had different worths, and there was a difference of about 60% between VII.1 with the most negative value and the most recent VIII.1. Therefore, it is shown that the cross-section library must be selected carefully in the core design of SFRs.

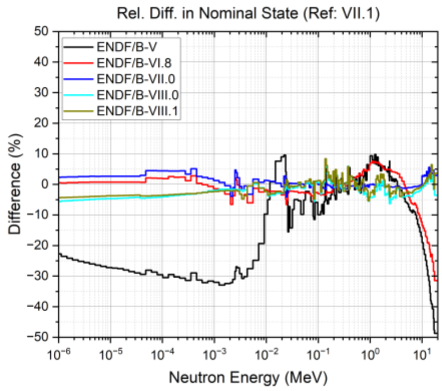
3.3 Neutron Spectrum and Balance Analysis

The above eigenvalue and void worth results are caused by the differences in reaction rates according to the neutron flux energy spectrum among the libraries.

Figure 2 shows the neutron flux spectrum in the core for the ENDF library cases. It can be seen that the ABTR-250 core has a spectrum in the epithermal-fast region. When the relative differences were analyzed with the ENDF/B-VII.1 library as a reference, ENDF/B-V shows large differences in all energy regions, and the libraries up to ENDF/B-VI.8 show large differences in the fast region. The libraries from ENDF/B-VII.0 onward show differences within 3% on average.



(a) Normalized neutron spectra per lethargy



(b) Relative differences with ENDF/B-VII.1

Figure 2. Comparison of neutron flux spectra for difference ENDF/B libraries

For the analysis of the void worth, we introduced the neutron balance method [15,16]. This method can analyze reactivity by decomposing the neutron reaction in the core into leakage (L), capture (C), fission (F), and (n,2n) (N) rates. The neutron balance equation is expressed each reaction rate normalized by the fission neutron production rate (F_p), as follows:

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

where l, c, f, and n are normalized leakage, capture, fission, and (n,2n) reaction rates, respectively. Table IV decomposes the reaction rates between the coolant nominal and the voided states for each library. In all cases, the effects of leakage and capture were dominant. For the ENDF/B-V case, which is the only one with a positive sodium void worth, it shows that the reactivity increase from the reduced capture has a larger contribution than the reactivity decrease from leakage.

4. Conclusions

This study compared the sodium void worth, an important parameter for SFRs, using various ENDF/B libraries. The ABTR-250 was adopted as the SFR core design. ANL reported a positive sodium void worth using the ENDF/B-V library in the original design report, whereas ORNL, which performed the benchmark calculation for ABTR, showed a negative value using the more recent ENDF/B-VII.1.

We performed a comparative analysis of the void worths from the old ENDF/B version V to the latest version VIII.1 using the Monte-Carlo code Serpent 2. Our results showed a positive value when using the ENDF/B-V library, consistent with the previous analysis by ANL, while the subsequent libraries resulted in negative values.

Analysis of the neutron flux energy spectrum inside the core revealed that the older ENDF/B versions V and VI.8 showed large differences in the fast region compared to more recent versions. The neutron balance analysis showed that for ENDF/B-V, the only case with a positive void worth, the reactivity increase from capture reaction rate change had a larger contribution than the reactivity decrease from leakage change.

A limitation of this study is that while the calculations were performed using the ABTR benchmark core design, this reactor has not been physically built. Therefore, it is unknown which calculated result is closest to reality. Our future work will aim to determine which ENDF/B library is the most accurate through comparison with validated experimental data.

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Table III. Serpent 2 results and differences from 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
Void worth (pcm)	182	-30	-103	-164	-107	-68
			Differences wih Shift (VII.1)			
k_{eff} (pcm)	659	1156	333	14	162	416
β_{eff} (%)	-	2.22%	1.53%	1.57%	1.53%	1.42%
Void worth (%)	-222%	-80%	-31%	10%	-28%	-54%

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

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