An Investigation of the Effect of ENDF/B-VIII.1 Cross Section Data on Criticality Estimation of i-SMR Core

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1. Introduction

In our country, the national project on i-SMR design and development has been launched and actively researched. The i-SMR core has very different design characteristics due to the use of boron-free operation and In-Vessel CRDM [1]. For example, much of the excess reactivity should be compensated by burnable absorber, which requires use of innovative burnable poison rods, and the insertion of control rods even during normal power operation can lead to significant distortion of axial power distribution. Also, it is important to secure sufficient reactivity margin from control rods because the use of the In-Vessel CRDM limits the number of control rods. Under these circumstances, the accurate estimation of the core reactivity is important to secure reactivity margins.

Recent releases of the ENDF/B-VIII.1 library have introduced neutron and thermal scattering cross section data tailored to various uranium-235 enrichments (5.0 , 10.0, 20.0 wt.%) [2]. The objective of this work is to evaluate the effect of these updated neutron and thermal scattering cross sections on the reactivity in 2D fuel assembly models and a full-core i-SMR model, in comparison to the ENDF/B-VII.1 library. To evaluate the differences arising from different versions of neutron cross sections, we performed criticality calculations. Furthermore, to measure the effect of the thermal scattering cross section, we conducted another criticality calculation by simply replacing the ENDF/B-VII.1 thermal scattering cross section with the ENDF/B-VIII.1 data for H_2O and UO_2. For both ENDF/B-VII.1 and ENDF/B-VIII.1, the ENDF data were processed with NJOY2016 (version 78) to generate pointwise continuous cross sections for Monte Carlo code.

2. Codes and Modeling

2.1 Cross section processing and Calculation Codes

ENDF/B-VII.1 and ENDF/B-VIII.1 nuclear data libraries were employed in this work. All the cross section data were reprocessed using NJOY2016 (version 78) to generate the ACE files required for Monte Carlo simulations [3]. The thermal scattering data used in the calculations, summarized in Table I, includes uranium in UO_2, oxygen in UO_2, and hydrogen in H_2O [4].

For the criticality calculations, the continuous-energy Monte Carlo code Serpent2 was used. Serpent2 is a three-dimensional continuous-energy neutron and photon transport code, which has been under development at VTT Technical Research Centre of Finland since 2004.

Table I. S(α,β) cross section of ENDF/B-VII.1 and ENDF/B-VIII.1

Library	Cross section
Uranium in UO_2 and Oxygen in UO_2	
ENDF/B-VII.1	Natural uranium
ENDF/B-VIII.1	Natural uranium
ENDF/B-VIII.1	Enriched uranium (5 wt.%)
ENDF/B-VIII.1	Enriched uranium (10 wt.%)
ENDF/B-VIII.1	Enriched uranium (20 wt.%)
Hydrogen in H_2O	
ENDF/B-VII.1	Protium (^1H)
ENDF/B-VIII.1	Protium (^1H)

2.2 Modeling of 2D Fuel Assembly and i-SMR core

To investigate the effects of ENDF/B-VIII.1 neutron cross sections and thermal scattering cross sections, a simple 2D Fuel Assembly (FA) was modeled. As shown in **Fig.1**, the FA follows the Westinghouse 17x17 type assembly design. The dimensional and material information for the FA can be found in **Table I**. Zircaloy-4 was used for both the cladding material and guide tubes, and the uranium enrichments range from 5 to 20 wt.%. For the purposes of modeling, spacer grids and dimples were ignored.

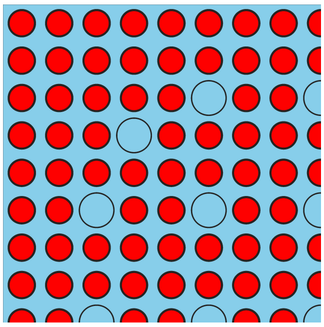

Fig. 1. Configuration of fuel assembly

Table II. Parameters of fuel assembly

Values	
Number of fuel rods	264
Number of thimbles	24
Number of Instrument thimbles	1
Guide tube material	Zircaloy-4
Instrument thimble material	Zircaloy-4
Lattice pitch (cm)	1.2600
Fuel Rod Diameter (cm)	0.9500
Pellet Diameter (cm)	0.8190
Diametral Gap (cm)	0.0166
Clad Thickness (cm)	0.0572
Clad material	Zircaloy-4
Density (g/cm3)	10.223
^{235}U enrichment (wt.%)	5 ~ 20

The i-SMR full core criticality calculation was performed to investigate the effect of ENDF/B-VIII.1 neutron cross sections and thermal scattering cross sections on the core reactivity under the practical situations. The core which has ~5 wt.% uranium enrichment fuels was modeled. **Fig. 2** shows the axial and radial configuration of i-SMR core.

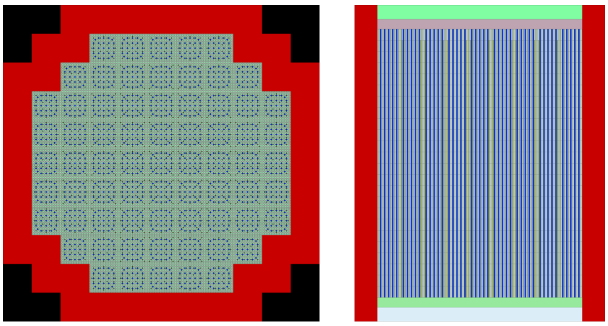

Fig. 2. Radial (left) and axial (right) configuration of i-SMR

And the design parameters of i-SMR can be found in **Table III** [5].

Table III. Design Parameters of i-SMR

Values	
Core	
Active core height	240 cm
Number of FAs	69
Fuel management scheme	2 batches
Radial reflector	Stainless steel 304
Core inlet temperature	295.5 °C
Fuel assembly and fuel rod	
Assembly array	17x17 (Westinghouse type)
Fuel rod pitch	1.26 cm
Number of fuel rod per FA	264
Number of guide tubes per FA	25
Fuel pellet density	10.220 g/cm^3
^{235}U enrichment	~ 5.0 wt.%

3. Results

3.1 Critical calculation of 2D Fuel Assembly

We performed 2D single FA criticality calculations with reflective boundary condition. The fuel enrichment ranged from 5.0 wt.% to 20.0 wt.%. ENDF/B-VII.1 was employed to compare the result of the critical calculations from ENDF/B-VIII.1. ENDF/B-VIII-1 provide various thermal scattering cross sections along with enrichment of uranium dioxide (5.0, 10.0, 20.0 wt.%). **Table IV** presents the results of 2D FA criticality calculations, varying the enrichment of each fuel assembly. For the regular cases with ENDF/B-VIII-1 neutron cross section, the thermal scattering cross section was applied according to the specific uranium enrichment for each case. The cases with ENDF/B-VII.1 use thermal scattering cross section for uranium-238. Additionally, we performed the calculations that combined neutron cross sections from ENDF/B VII.1 with thermal scattering cross sections from ENDF/B VIII.1. Furthermore, for the ENDF/B VIII.1 library, we carried out an additional calculation using its natural uranium thermal scattering cross section. For the regular cases, the results differ by approximately 150 pcm, reaching up to 215 pcm. Overall, the variations due to changes in the neutron cross sections version are more significant than those arising from changes in the thermal scattering cross sections. It is noted that a simple replacement of the thermal scattering cross section with ENDF/B-VIII.1 for the enriched UO_2 and natural UO2 thermal scattering ones in the ENDF/B-VII.1 calculations led to only a 2-15 pcm deviation from the original ENDF/B-VII.1 results and the replacement of the thermal scattering cross sections for enriched UO2 ones of ENDF/B-VIII.1 with the ones for natural UO2 ones of ENDF/B-VIII.1 in the ENDF/B-VIII.1 calculations also led to only a 3-5 pcm.

Table IV. Results of 2D PWR Fuel Assembly criticality calculations with ENDF/B-VIII.1 and ENDF/B-VII.1.

Enrichment	Infinite Multiplication Factor		Diff. (pcm)
	ENDF/B-VII.1	ENDF/B-VIII.1	
5 wt.%	1.43386 (±0.0006)	1.43067 (±0.00006)	155.5
	1.43418* (±0.0006)	-	-
	1.43409** (±0.0006)	1.43061** (±0.00006)	169
10 wt.%	1.51835 (±0.0006)	1.51384 (±0.0006)	196.2
	1.51865* (±0.0006)	-	-
	1.51847** (±0.0005)	1.51395** (±0.0006)	196.6
20 wt.%	1.57841 (±0.0005)	1.57329 (±0.0006)	206.2
	1.57844* (±0.0005)	-	-
	1.57853** (±0.0005)	1.57317** (±0.0006)	215.8

*: Using enriched UO_2 thermal scattering XS (ENDF/B-VIII.1)
**: Using natural UO_2 thermal scattering XS (ENDF/B-VIII.1)

3.2 Whole core criticality calculation of i-SMR

To investigate the effect of ENDF/B-VIII.1 neutron cross sections and thermal scattering cross sections under typical operating conditions, criticality calculations for the i-SMR core were performed. The i-SMR core had an average uranium enrichment of approximately 5 wt.%, and the corresponding thermal scattering cross sections for 5.0 wt.% of uranium dioxide were applied in the calculations. The Serpent2 Monte Carlo simulation used 400 million neutron histories. As shown in **Table V**, the differences between the calculation applying ENDF/B-VIII.1 neutron cross sections and thermal cross sections and the calculation using ENDF/B-VII.1 were up to 51 pcm. In the ENDF/B-VII.1 calculation, simple replacement of the thermal scattering cross section led to just a 3-5 pcm deviation from the original ENDF/B-VII.1 results. In case of ENDF/B-VIII.1, the replacement of the thermal scattering cross section data also led to just a 17 pcm deviation from the regular one.

Table V. Results of whole core criticality calculation with ENDF/B-VIII.1 and ENDF/B-VII.1

Effective Multiplication Factor		Diff. (pcm)
ENDF/B-VII.1	ENDF/B-VIII.1	
1.02989 (±0.00005)	1.03043 (±0.00005)	51
1.02992* (±0.00005)	-	-
1.02984** (±0.00006)	1.03024** (±0.00006)	37

*: Using enriched UO_2 thermal scattering XS (ENDF/B-VIII.1)
**: Using natural UO_2 thermal scattering XS (ENDF/B-VIII.1)

4. Conclusions

In this paper, we examined the effects of ENDF/B-VIII.1 neutron cross sections and thermal scattering cross sections on the reactivity of fuel assemblies and core of i-SMR. The results indicate that the differences in the neutron cross section versions had a much greater impact than differences in the thermal scattering cross section versions. The differences in the neutron cross section libraries led to differences of 155-215 pcm in the assembly calculations and about 51 pcm in the core calculations. In contrast, variations in the thermal scattering cross sections caused differences of only 3-17 pcm in both the assembly and core.

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REFERENCES

[1] H.O. Kang, B.J. Lee, S.G. Lim, Light water SMR development status in Korea, Nuclear Engineering and Design 419 (2024) 112966.
[2] N.A. Gibson, W. Haeck, N.A. Kleedtke, D. Neudecker, D.K. Parsons, T. Saller, P. Talou, Release of ENDF/B-VIII.1β1-Based ACE Data Files, Los Alamos National Laboratory (LANL), Los Alamos, NM (United States), 2023. https://doi.org/10.2172/1968203.
[3] D. Brown, ENDF-6 Formats Manual - Data Formats and Procedures for the Evaluated Nuclear Data Files ENDF/B-VI, ENDF/B-VII and ENDF/B-VIII, 2023. https://doi.org/10.2172/2007538.
[4] D. Parsons, C. Toccoli, Re-Release of the ENDF/B VIII.0 S(α,β) Data Processed by NJOY2016, 2020. https://doi.org/10.2172/1634930.
[5] W.J. LEE, S.G. Hong "i-SMR Core Design Option Study with Gadolinium Nitride Coating Burnable Absorber", Transactions of the Korean Nuclear Society Autumn Meeting Changwon, Korea, October 24-25, 2024