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Analysis of a Metallic Plutonium Fueled BFS-55-1 Critical Assembly with Different Evaluated Nuclear Data Files

Jaewoon Yoo*, Sang-Ji Kim, Yeong-Il Kim and Dohee Hahn
Korea Atomic Energy Research Institute, 1045 Daedeog-Daero, Yuseong-gu, 305-353 Daejeon, Korea
**Corresponding author: jwyoo@kaeri.re.kr*

1. Introduction

The uncertainty of basic nuclear data occupies a considerable portion in a fast reactor analysis compared with other uncertainties coming from the methodologies, geometrical approximation, and so on. For this reason, effort has been devoted to a cross section adjustment for improving the calculation accuracy.

The cross section adjustment should be based on many results from a integral measurement and the selection of a appropriate basic cross section should precede that work. In this context, two metallic uranium fueled critical assemblies were analyzed before[1].

The metallic plutonium fueled BFS-55-1 critical assembly[2] is analyzed with the different nuclear data files to compensate for the limited number of the critical assembly and to provide basic information for selecting the appropriate cross section.

2. Description of BFS-55-1 Critical Assembly

The BFS-55-1 critical assembly is a metallic plutonium fueled core with a single enrichment. The critical experiment was carried out in 1987 to investigate the characteristics of a metallic Pu-fueled breeder core.

The unit fuel cell consists of 2 highly enriched plutonium metal disks, 3 U-238 disks, 3 sodium pellets and 1 stainless steel disk. The volume fraction of sodium is about 0.3. The average plutonium enrichment in the fuel cell is about 10wt%.

The core is divided into two regions: core and blanket as shown in Figure 1. The core region is surrounded by two succeeding axial blankets and a radial UO_2 blanket.

3. Calculation method

3.1. Nuclear data libraries

Three up to date nuclear data files, ENDF/B-VII.0, JEFF-3.1 and JENDL-3.3 are used for the analysis in addition to the ENDF/B-VI.6 file which has been used for the analysis of the sodium cooled fast reactor at KAERI. All the evaluated nuclear data files are processed into a MATXS format by the NJOY code, in which a KALIMER-150 neutron spectrum is used as a weighting function in the GROUPR module to be consistent with older KAFAX-E66 library.

3.2. Effective cross section generation

TRANSX code[3] is used for an effective cross section generation. The unit fuel cell is described as either an homogeneous mixture or an axial plane as described in Figure 1. Resonance self shielded effective cross sections are processed for the corresponding model by a narrow resonance approximation.

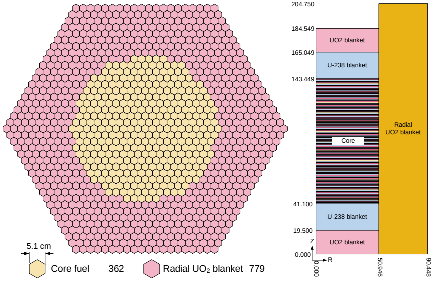

Figure 1 Radial and axial layout of BFS-55-1 critical assembly

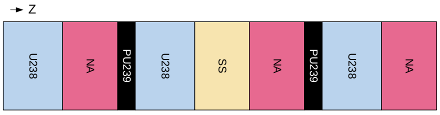

Figure 2 Axial configuration of the BFS-55-1 unit fuel cell

3.3. Core calculation

Criticality and other reactivity parameters are mainly calculated by either the TWODANT[4] S_N transport or DIF3D[5] finite difference methods with a geometry approximated into the R-Z geometry. All the calculations are carried out with 150 group constants without collapsing.

The k-effective calculated by the diffusion method is corrected to account for the neutron transport effect, which is carried out by comparing the results of the DIF3D calculation with those of the TWODANT in the same R-Z geometry.

4. Results and Discussion

4.1. k-effective

The k-effective was calculated with two core configurations; one is the axially heterogeneous configuration and the other is the homogeneous configuration. The heterogeneity effect is defined as the difference in the reactivities between two configurations.

As shown in Table I, the C/E of k-effective for the heterogeneous configuration is very poor but the accuracy is improved for the homogeneous one. Main reason for the discrepancies for the heterogeneous configuration is mainly from the underestimation of the heterogeneity effect, which had been evaluated as 1800 pcm by the IPPE side. It was decided that the pellet material properties should be re-evaluated after a discussion with IPPE scientist.

Table I C/E of k-effective for BFS-55-1

	ENDF/B-VII.0	JEFF-3.1	JENDL-3.3	ENDF/B-VI.6
Heterogeneous	0.98557	0.99057	0.98224	0.99543
Homogeneous	0.99503	0.99966	0.99201	1.00595
Hetero. effect [pcm]	878	917	846	772
Transport effect [pcm]	215	207	182	205

4.2. Spectral indices and reaction rate ratios

The spectral indices and reaction rate ratios were measured in the core center by using the segment chamber for a fission reaction and by a foil irradiation for the fertile capture reaction. Table II shows the C/E values of the spectral indices and the reaction rate ratios. The up-to-date libraries tend to underestimate the U-238 fission reaction but a improvement was shown for the fission reaction of Pu-240 and Pu-241.

Table II C/E of spectral indices and reaction rate ratios

	1σ [%]	ENDF/B-VII.0	JEFF-3.1	JENDL-3.3	ENDF/B-VI.6
F28/F25$^{1)}$	1.68	0.951	0.945	0.967	1.009
F49/F25	1.40	0.993	1.000	0.992	1.003
F40/F49	3.15	0.996	1.017	0.974	1.055
F41/F49	1.98	0.994	0.996	1.005	0.983
C28/F25	2.60	1.020	1.011	1.021	1.008
C28/F49	2.60	0.994	0.977	0.994	0.972

$^{1)}$ XYZ/XYZ, X=reaction type (F=fission, C=capture), Y=last digit of atomic number, Z=last digit of mass number

4.3. Sodium void reactivity

The sodium void reactivity was measured by substituting the sodium pellets located at the core radial center with vacant pellets. The measurement was carried out with three axial configurations. 3 central fuel cells and 4 fuel cells adjacent to the core center are replaced in Case1 and Case2, respectively. 4 fuel cells located at the bottom and top of the core are replaced in Case3. The actual sodium void reactivities for Case1 and Case2 are positive but for Case3 the actual value is close to zero because the neutron leakage term becomes significant at void.

All the results of the sodium void reactivity calculation are satisfactory for the axially heterogeneous configuration as shown in Table III. Among them, JEFF-3.1 shows the best performance in the sodium void reactivity evaluation owing to the adjustments of the inelastic and elastic cross sections of Na-23. The results for the homogeneous configuration is highly overestimated for the sodium void reactivity but all the results except Case3 are conservative from the viewpoint of a core design.

Table III C/E of sodium void reactivity

	Case1	Case2	Case3$^{1)}$
Uncertainty (1σ) [%]	7.1	7.7	0.5$^{2)}$
Heterogeneous configuration			
ENDF/B-VII.0	0.966	0.972	-1.738
JEFF-3.1	0.991	1.023	-1.692
JENDL-3.3	1.036	0.932	-6.781
ENDF/B-VI.6	1.012	1.032	-1.248
Homogeneous configuration			
ENDF/B-VII.0	1.271	1.329	0.003
JEFF-3.1	1.296	1.349	-0.494
JENDL-3.3	1.286	1.337	-0.511
ENDF/B-VI.6	1.300	1.369	0.393

$^{1)}$ C-E for Case3 in cent
$^{2)}$ The unit of uncertainty is in cent

5. Conclusions

The metallic plutonium fueled BFS critical assemblies (BFS-55-1) were analyzed with four different evaluate nuclear data libraries. The calculated k-effective shows relatively large discrepancies in the axially heterogeneous configuration. The material properties such as the impurity content in the pellet should be re-evaluated for a future analysis. Improvement in the fission cross section of Pu-240 and Pu-241 was found in the up-to-date nuclear data files and the accuracy of the sodium void reactivity was satisfactory.

Results of this study could be utilized for the selection of isotope-wise nuclear data that will be used for a Korean sodium cooled fast reactor design and provide fundamental data to investigate the characteristics of the most up-to-date nuclear data files.

Acknowledgement

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