Analysis of Metallic Uranium Fueled BFS Experiments using McCARD

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1. Introduction

The BFS-1 critical facility was built at Institute of Physics and Power Engineering (IPPE) of Russia for benchmarking sodium cooled fast reactor cores. As a part of the cooperation program between Korea Atomic Energy Research Institute (KAERI) and IPPE, the BFS-73-1[1] and BFS-75-1[2] critical assemblies were mounted in the BFS-1 critical facility to examine the KALIMER-150 fuels in 1997 and 1999. The average uranium enrichments of the two critical assemblies is 18.5 wt% that is similar to that of KALIMER-150. The analysis of the two core has been carried out by using the K-CORE and ECCO/ERANOS2.1 [3] code system. The objective of this study is to analyze the BFS-73-1 and BFS-75-1 experiments with the Monte Carlo code of Seoul National University, McCARD [4].

2. BFS 73-1 and BFS-75-1 Experiments

In the BFS-73-1 and BFS-75-1 experiments, the metallic uranium fuels are located at the center of the core enclosed by axial and radial blankets with depleted uranium. The fuel cell consists of enriched uranium and depleted uranium pellets with sodium pellets as the coolant. The BFS-73-1 consists of 425 fuel rods, 897 radial blanket rods and 14 control rods while the BFS-75-1contsists of 91 LEZ fuel rods(15.11wt%) in the inner core and 162 HEZ fuel rods(19.96wt%) in the outer core. In the core and reflector region, the steel sticks are placed around the fuel rods. These are all room temperature criticals.

3. Results and Discussion

All the McCARD Monte Carlo Calculations were performed employing 10,000 neutron histories per a cycle, 1000 active cycles, and 50 inactive cycles. For efficient calculations, the steel sticks and air regions outside fuel and reflector rods are homogenized into one cell. For both experiments, the calculations for the homogeneous and heterogeneous configurations were performed.

3.1 k-effective

Table I shows the C/E values for each evaluated nuclear data library used and each configuration. For heterogeneous configuration of BFS-73-1, the k-effective of the ENDF/B-VII case differs from the experimental value by 583 pcm while that of the ENDF/B-VI.8 case differ by 834 pcm. For the heterogeneous configuration of BFS-75-1, the k-effective of ENDF/B-VII differs from by 545 pcm while that of ENDF/B-VI.8 by 295 pcm. The difference between the experimental and calculated values comes from the homogenization effect for steel sticks and the uncertainty of nuclear data library.

Table I. C/E values of k-effective

Case	ENDF/B-VI.8	JENDL-3.3	ENDF/B-VII.0
BFS-73-1 Heterogeneous	0.99167 ±0.00017	0.98253 ±0.00017	0.99417 ±0.00017
BFS-73-1 Homogeneous	0.98804 ±0.00017	0.97830 ±0.00016	0.99128 ±0.00017
BFS-75-1 Heterogeneous	0.99705 ±0.00017	0.98724 ±0.00016	0.99455 ±0.00017
BFS-75-1 Homogeneous	0.99376 ±0.00017	0.98223 ±0.00016	0.99051 ±0.00016

To estimate the uncertainty in k-effective, a sensitivity and uncertainty(S/U) analysis was performed using the S/U analysis module [5] of McCARD and the covariance data based on JENDL-3.3[6]; U^{235}, U^{238}, Na^{23}, Cr^{52}, Fe^{56} and Mn^{55}. For BFS-73-1, the uncertainty of k-effective due to the nuclear data uncertainties is estimated to 884 pcm. Figure 1 shows the contribution of nuclear data uncertainties to $\sigma(k_{eff})$ by the cross section type and by nuclide for BFS-73-1. The uncertainties of the U^{238} inelastic scattering cross section and the U^{238} capture cross section turned out to contribute most to $\sigma(k_{eff})$.

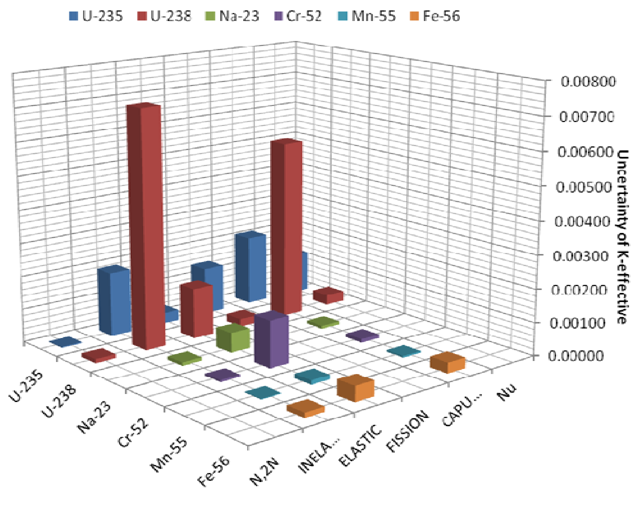

Fig. 1. Contribution of nuclear data uncertainties to $\sigma(k_{eff})$ by cross section type and nuclide for BFS-73-1

3.2 Axial Fission Rate Distribution

Figure 2 shows the axial U-235 fission rate distribution for the heterogeneous configuration of BFS-75-1 obtained using ENDF/B-VII. The fission rate distributions are normalized to the maximum values so that the maximum fission rate is unity. The axial U-235 fission rate distributions agree well within a single standard deviation the experimental value.

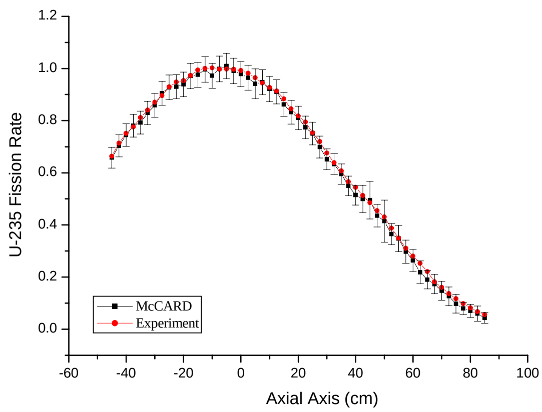

Fig. 2. Axial U-235 fission rate distribution for heterogeneous configuration of BFS-75-1 using ENDF/B-VII

3.3 Spectral Indices and Reaction Rate Ratios

Table II shows the C/E value of spectral Indices and reaction rate ratios for BFS-75-1. The results of ENDF/B-VII and JENDL-3.3 agree well within a single standard deviation while the results of ENDF/B-VI.8 agree within two standard deviations. Overall, the accuracy of spectral indices and reaction rate ratio is improved when the ENDF/B-VII nuclear data library is used.

Table II. C/E values of spectral indices for BFS-75-1

Spectral Indices	Uncertainty [%]	ENDF/B-VII	ENDF/B-VI.8	JENDL-3.3
F28/F25	1.563	1.032±0.191	1.105±0.195	1.092±0.197
F40/F49	2.271	1.075±0.110	1.160±0.116	1.080±0.110
F48/F49	1.275	1.061±0.081	1.094±0.084	1.050±0.081
F49/F25	1.170	0.993±0.069	1.009±0.070	0.993±0.070
F37/F49	1.987	1.026±0.111	1.081±0.114	1.042±0.112
F53/F49	4.294	0.994±0.135	1.182±0.151	1.019±0.137

*F28/F25 = $\sigma_f(U^{238})$ / $\sigma_f(U^{235})$, F40/F49 = $\sigma_f(Pu^{240})$ / $\sigma_f(Pu^{239})$,
F48/F49 = $\sigma_f(Pu^{238})$ / $\sigma_f(Pu^{239})$, F49/F25 = $\sigma_f(Pu^{239})$ / $\sigma_f(U^{235})$,
F37/F40 = $\sigma_f(Np^{237})$ / $\sigma_f(Pu^{240})$, F53/F49 = $\sigma_f(Am^{243})$/ $\sigma_f(Pu^{239})$

3.4 Delayed Neutron Fraction

Table III shows the delayed neutron fraction (β_{eff}) for BFS-73-1 heterogeneous configuration. The β_{eff} calculated by McCARD using ENDF/B-VI is within the error of experimental value.

Table III. Delayed Neutron Fraction for BFS-73-1

Case	β_{eff}
McCARD (ENDF/B-VI)	0.00715 ± 0.00003
Experiment (Cf-source)	0.00720 ± 0.00030
Experiment (Rossi-α)	0.00740 ± 0.00040

4. Conclusions

The Monte Carlo analysis of the two metallic uranium critical assemblies was performed using McCARD. The comparison between McCARD and experimental results for k-effective, axial fission rate distribution, spectral indices and delayed neutron fraction shows a good agreement. This study validates the application of McCARD to the analysis of metallic uranium fueled fast reactor cores.

Acknowledgement

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