3.8 Comparison of BFS-73-1 Benchmark Test using JENDL-3.2, JEF-2.2 and ENDF/B-VI.3

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A comparative analysis for BFS-73-1 critical assembly, which is a benchmark of a fast reactor core with metal uranium fuel of 18.5% enrichment, was carried out with TWODANT code and three current evaluated data libraries of JENDL-3.2, JEF-2.2 and ENDF/B-VI.3. The results for criticality, spectral indices and reaction rate distributions are intercompared along with experimental data.

1. Introduction

As a part of critical benchmarks for developing the KALIMER(Korea Advanced LIquid MEtal Reactor), experimental studies on metal fueled liquid metal reactor core characteristics are carried out under contract between Korea Atomic Energy Research Institute and Institute of Physics and Power Engineering in Russia. The BFS-73-1 critical assembly is a benchmark core with metal uranium fuel of 18.5% enrichment. K_{eff}, spectral indices, axial and radial fission rate distributions of U-235 and U-238, $\mathbf{b}_{eff}$, Doppler effects of U-238 and sample reactivity worth etc. were measured in this experiment. An analysis of the experiment[1] has been performed using DIF-3D code and JEF-2.2-based KAFAX-F22[2] data set. In order to survey the impact of nuclear data on the BFS-73-1 benchmark test, a comparative analysis using JENDL-3.2, JEF-2.2 and ENDF/B-VI.3 was carried out and the results are presented in this symposium.

2. Benchamrk core of BFS-73-1

The BFS-73-1 was constructed at BFS-1 facility in 1997. The core is a homogeneous system, without control rods in the core. Axial blankets of ~50 cm thickness are located above and below the core, which are consisted of depleted uranium dioxide pellets. A radial blanket of ~35 cm thickness, surrounding the core, also contains only depleted dioxide uranium pellets in steel tubes. The core is composed of fuel elements with round stainless steel sticks between tubes, and the radial blanket is assembled with tubes of UO_2 pellets and triangular stainless steel sticks between these tubes. Table 1 shows some parameters of BFS-73-1 critical core. And layout of the core, and the fuel and blanket rods is shown in Fig. 1.

Table 1. Parameters of BFS-73-1 critical core

No. of rods in core	• 379 with 36.41% uranium enriched • 46 with 36.45% uranium enriched • 760 round steel sticks
No. of rods in blanket	• 708 UO_2 • 1400 triangular steel sticks
Temperature(°C)	• +19 °C
Position of control rods	• All rods are in the upper position
K_{eff}	• 1.0008

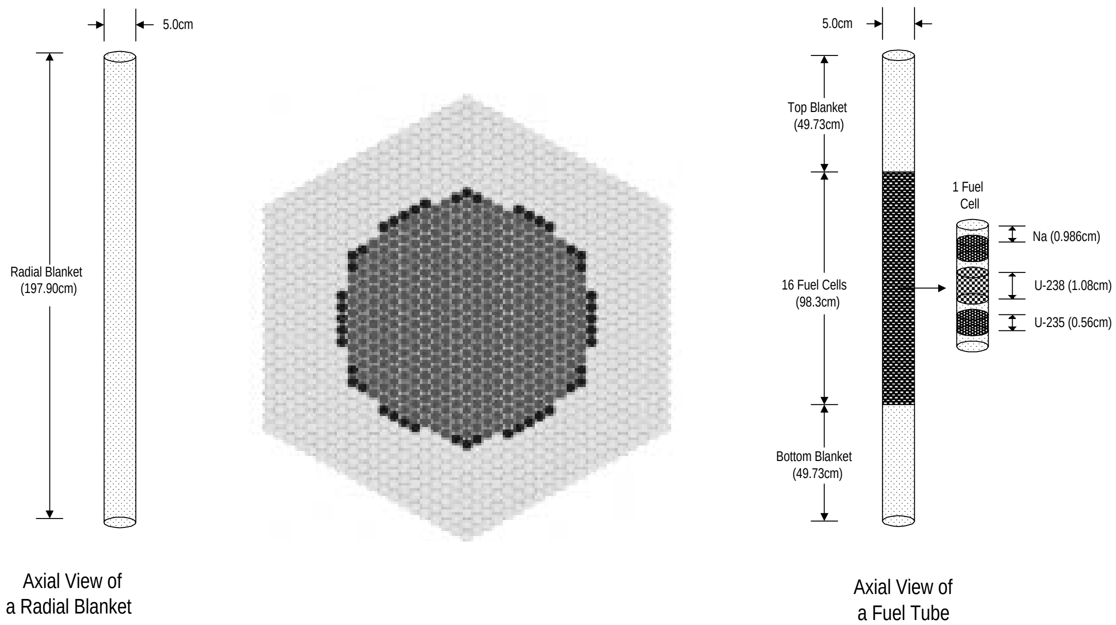

Figure 1. Layout of BFS-73-1

3. Data processing

Three multigroup cross section data sets were processed with the NJOY94.105[3] system from JENDL-3.2, JEF-2.2 and ENDF/B-VI.3, respectively. The data sets have MATXS-format and LANL 80-group structure . The MATXS-format data can be processed to ANISN-format data using TRANSX 2.15[4] pre-processing code for neutron transport calculations.

4. Calculation

Neutron transport calculations using each of these data sets were carried out with pre-processing code TRANSX 2.15 for MATXS-format data library and 2-dimensional discrete ordinates code TWODANT[5] using R-Z model of BFS-73-1 critical assembly. Figure 2 shows the R-Z model for the calculations. At first, the 80-group data were collapsed to 25-group with the flux calculated by R-Z model, coarse-mesh and P_3-S_8 approximations. And then, 2-dimensional 25-group fine-mesh(width of meshes: ~0.6cm for radial and ~0.5 cm for axial direction) calculations were performed with P_3-S_8 approximations. In this report, the results for criticality, spectral indices, and axial and radial reaction rate distributions of U-235 and U-238 are intercompared along with experimental data.

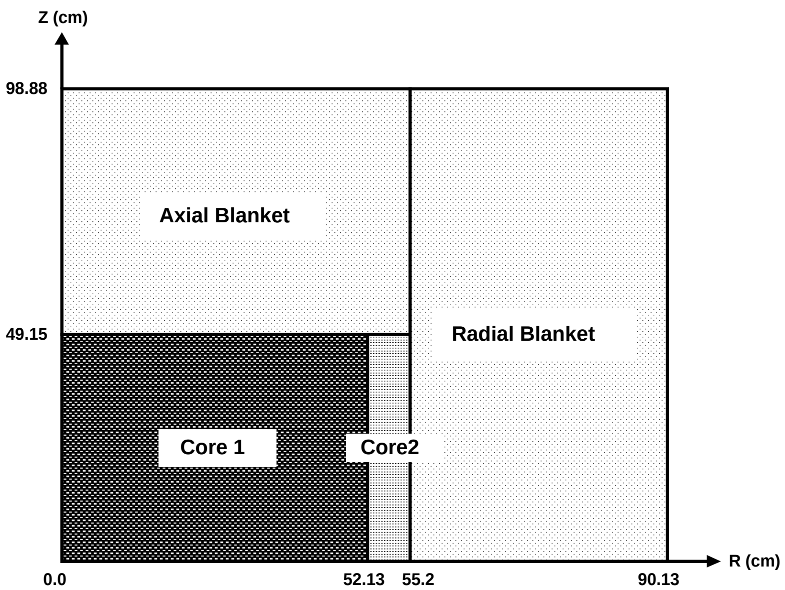

Figure 2. Calculational R-Z model of BFS-73-1

5. Results and Discussion

Calculated values of K_{eff} and central reaction rate ratios are compared with the measured data. The result is presented in Table 2, as the ratio of calculation to experiment(C/E). K_{eff} value using JEF-2.2 shows a little active result. Calculated central reaction rate ratios of U-238 capture to Pu-239 fission cross section using the three data sets underestimate the experimental values. And the value of s_f^{238}/s_f^{235} using ENDF/B-VI.3 overestimates the measured data. All of s_f^{239}/s_f^{235} are within measurement error range. C/E values of fission rate distribution of U-235 and U-238 in the core and blanket region are shown in Fig.3~6. Figure 3~4 and Figure 5~6 are for radial and axial distributions, respectively. Measurement uncertainties for U-235 fission rate distributions are ~2% in the core and they increase up to 4% in the blankets. For U-238, the uncertainties are ~3% in the core and ~7% in the blankets. For the most part, C/E-values of fission reaction rate distribution using the three data sets show reasonable tendency except U-235 fission rate in radial blanket, considering the measurement errors. However, the results of fission rate distributions using ENDF/B-VI.3 are slightly better than those of JENDL-3.2 and JEF-2.2.

Table 2. C/E-values of Keff and spectral indices

	Exp. Error(%	JENDL-3.	JEF-2.2	ENDF/B-VI.3
K_{eff}		0.9978	1.0064	1.0029
s_f^{238}/s_f^{235}	1.65	1.011	1.013	1.038
s_c^{238}/s_f^{239}	1.6	0.972	0.971	0.954
s_f^{239}/s_f^{235}	1.6	0.983	0.989	0.992

Acknowledgements

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References

[1] Hoon Song et al., “Analysis of BFS-73-1 Experiment,” KAERI/TR-1138/98 (1998).

[2] J.D. Kim and C.S. Gil, “Development and Benchmark of Multi-group Library for Fast Reactor using JEF-2.2,” KAERI/TR-842/97 (1997).

[3] R.E. MacFarlane and D.W. Muir, “The NJOY Nuclear Data Processing System, Version 91,” Los Alamos National Laboratory, LA-12740-M (Oct. 1994).

[4] R.E. MacFarlane, “TRANSX 2: A Code for Interfacing MATXS Cross-Section Libraries to Nuclear Transport Codes,” Los Alamos National Laboratory, LA-12312-MS (Dec. 1993).

[5] R.E. Alcouffe, et al, “User’s Guide for TWODANT: A Code Package for Two-Dimensional, Diffusion-Accelerated, Neutron Transport,” Los Alamos National Laboratory, LA-10049-M (Feb. 1990).

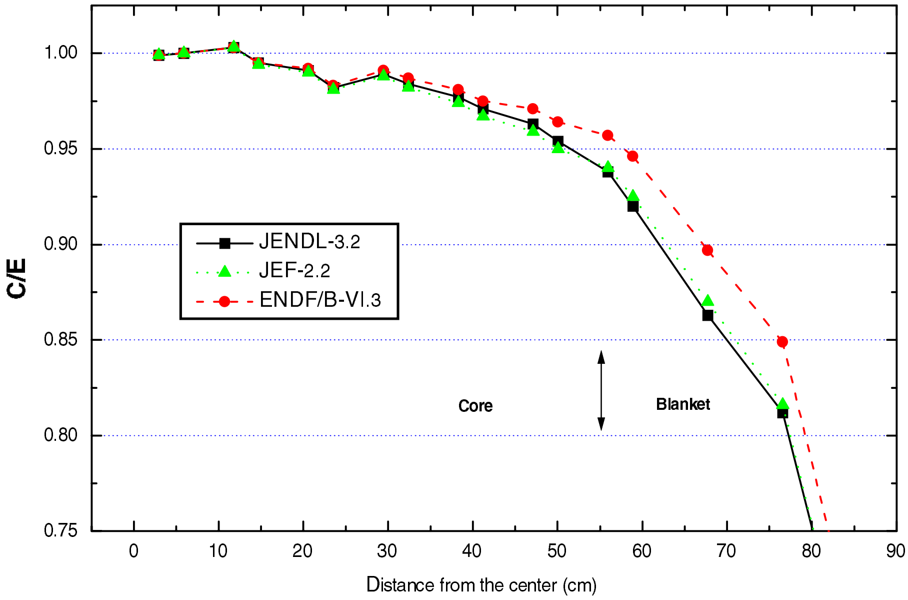

Fig. 3. Radial distribution of U-235 fission rate

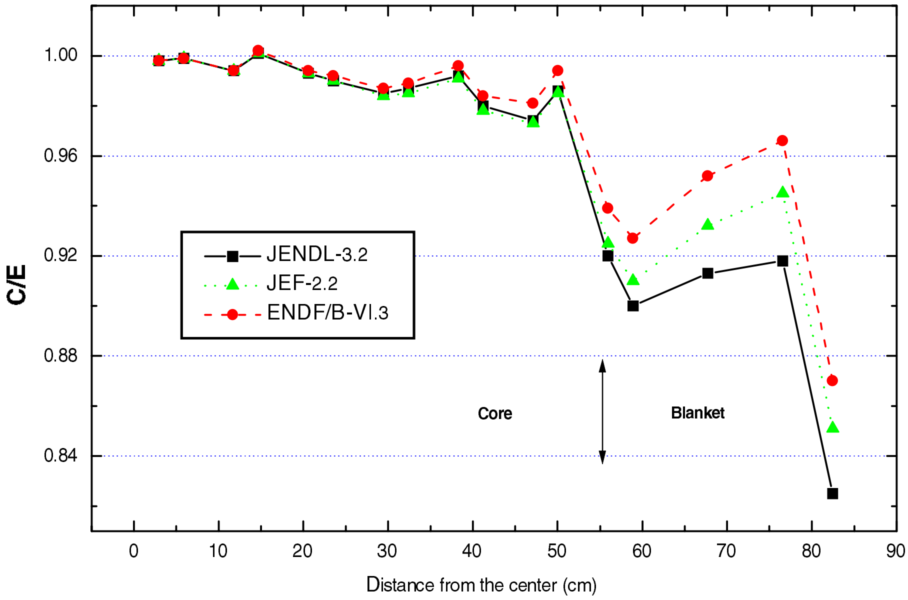

Fig. 4. Radial distribution of U-238 fission rate

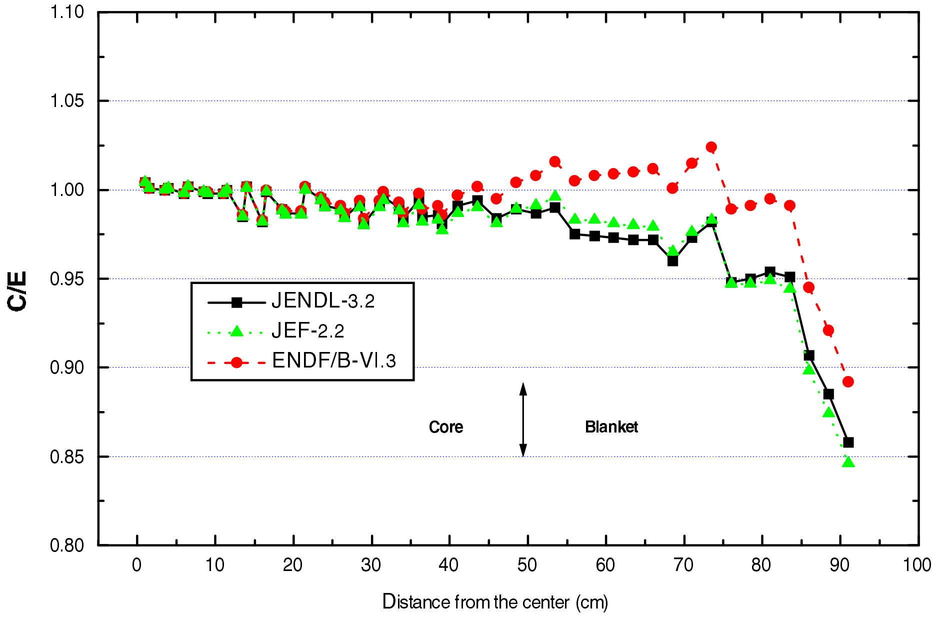

Fig. 5. Axial distribution of U-235 fission rate

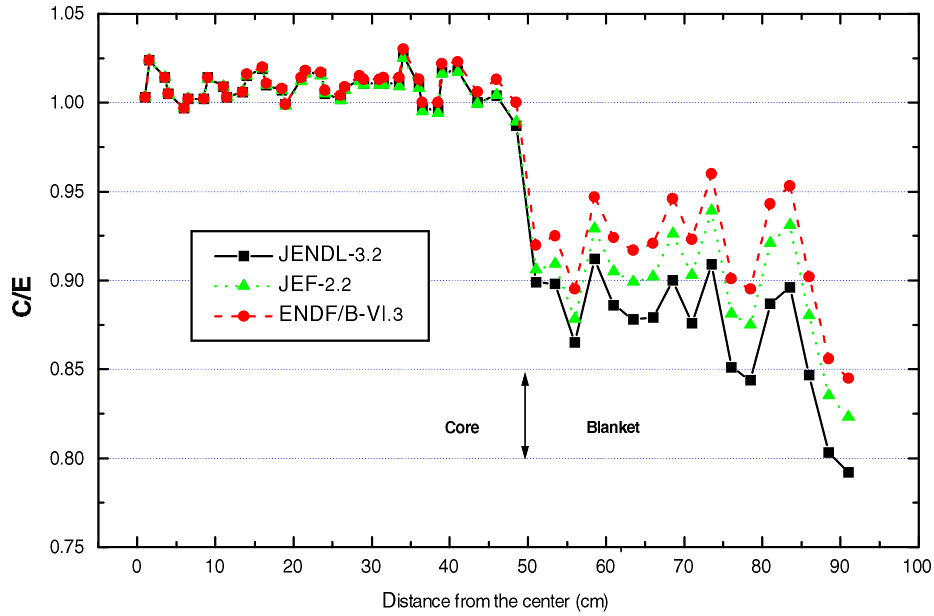

Fig. 6. Axial distribution of U-238 fission rate