Validation of JENDL-3.3 by Criticality Benchmark Testing

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In the thermal uranium core, the keff-values of STACY, TRACY and JRR-4 overestimated with JENDL-3.2 were improved significantly by decreasing of about 0.6 % with JENDL-3.3. This is due to modification of the fission spectrum and thermal fission cross section of ^{235}U from JENDL-3.2 to JENDL-3.3 data. For the uranium fast cores, the discrepancies of keff values between JENDL-3.2 and 3.3 were very small. In the thermal Pu cores of TCA, the keff-values calculated with JENDL-3.3 were in good agreement with the experimental values. For Pu fuel cores of ZPPR-9 and FCA-XVII, the keff values calculated with JENDL-3.3 became larger 0.2 % than those for JENDL-3.2. In small fast cores with U-233 fuel, the keff-values overestimated with JENDL-3.2 were improved considerably with JENDL-3.3, due to reevaluation of U-233 fission cross sections in the high energy region.

1. INTRODUCTION

In order to accurately assess neutronic behavior of various types of reactors, it is necessary to validate both calculation methods and nuclear data by analyzing integral experimental data. On the nuclear data development, the updated version of JENDL-3 nuclear data library, JENDL-3.3, has been tentatively released to test the validation in early 2000.

In fast reactor benchmark calculations, the selected cores are LMFBR mock-up cores ZPPR-9 and FCA-XVII-1, and small fast reactor cores such as JEZEBEL, GODIVA and FLATTOP with hard neutron spectrum. For thermal reactors, the JRR-4, TRX and TCA cores with water moderated lattice of U or Pu fuel, and furthermore the STACY and TRACY with uranyl nitrate solution were selected.

All the benchmark calculations were performed with the continuous energy Monte Carlo code MVP to reduce the uncertainties of core geometrical modeling and data processing for multigroup cross sections production.

The objective of this benchmark calculations is validation of JENDL-3.3 released tentatively to assess neutronic behavior of various types of reactors, to validate both calculation methods and nuclear data by analyzing integral experimental data. We confirmed the JENDL-3.3 shows the best results based on the comparative studies with the other nuclear data; JENDL-3.2, ENDF/B-VI and JEF-2.2.

2. BENCHMARK CORES AND ITS CHARACTERISTICS

Thermal neutron cores: For the thermal reactors, JRR-4, TCA and TRX as the water moderated lattice cores of U or Pu fuel, and STACY and TRACY as the critical cores with uranyl nitrate solution were selected. In the uranium fueled cores, TCA-150U, -183U, -248U, -300U are low-enriched UO_2 lattice, and TRX-1 and 2 are uranium-metal hexagonal lattice. And STACY, TRACY and JRR-4 are high-enriched research reactors with 10 and 20%EU. For these high-enriched cores, JENDL-3.2 overpredicted significantly the keff-values. As the plutonium fueled cores, TCA-242Pu, -298Pu, -424Pu and -555Pu are PuO_2 lattice with low Pu contents. U enrichment and Pu content in the thermal cores are as follows: TCA-UO2 is 2.6%EU, TRX-U 1.3%EU, JRR-4 20%EU, STACY and TRACY the uranyl solution of 10%EU, and TCA-MOX 3%Pu content.

Fast neutron cores: For the fast reactors, we selected JEZEBEL, GODIVA, FLATTOP and BIGTEN of small and very hard neutron spectrum cores with U-235, Pu and U-233, and ZPPR-9, ZPPR-13A and FCA-XVII-1 of large FBR mock-up cores. Table 1 shows the geometrical sizes for the small cores. In the U-235 cores, GODIVA is the bare sphere of highly enriched U, FLATTOP-25 the bare sphere of highly enriched U with U-reflector and BIGTEN the cylinder of 10%EU with U-reflector. In the U-233 cores, JEZEBEL-23 is the bare sphere of 98% enriched U-233 fuel and FLATTOP-23 the bare sphere of U-233 fuel with U-reflector. In the Pu cores, JEZEBEL is the bare sphere of Pu fuel, JEZEBEL-Pu the bare sphere of Pu fuel with 20%Pu-240, FLATTOP-Pu the bare sphere of Pu fuel with U-reflector and THOR the bare sphere of Pu fuel with Th-reflector. As large size LMFBR cores, ZPPR-9 is the reference core of MOX-FBR in JUPITER program, ZPPR-13A the radial heterogeneous core and FCA-XVII-1 the MOX-FBR mockup core with a highly-enriched uranium driver region.

Table1 Geometrical sizes of small fast reactors

Core	sphere radius(cm)	reflector thicness(cm)
GODIVA	8.741	
FLATTOP-25	6.116	24.13
JEZEBEL	6.385	
JEZEBEL-Pu	6.660	
FLATTOP-Pu	4.533	24.13
JEZEBEL-23	5.983	
FLATTOP-23	4.317	24.13
THOR	sphere:r=5.310, cylinder-radius:26.65, Height:53.30	
BIG10	cylinder radius: 41.91, Height: 96.428	

3. BENCHMARK CALCULATION

In this calculation, the continuous energy Monte Carlo code, MVP[1] is used from the point of view to reduce the uncertainties of core geometrical modeling and data processing for multigroup cross sections production, and as the calculation time is the shortest. All the benchmark cores are calculated with MVP library generated based on JENDL-3.3.

The calculation conditions are as follows: Statistical error of keff is 0.02%. The calculation CPU time is 10 h for TCA, 1-6h for small fast cores and 18h for large FBRs, respectively. History No. per one batch is 20,000. Initial batch No. is 400,000. The highest energy is 20 MeV. The lowest energy is 10^{-5} eV. Thermal cut energy is below 4.5 eV, and $S(\alpha,\beta)$ of ENDF/B-III data is used. Furthermore, the unresolved resonance region is treated by the probability table method.

We studied that the difference between keff-values calculated with MVP and MCNP4B using JENDL-3.2 data. The preliminary results of the ratios of MVP to MCNP4B for the keff-values are as follows: 1.0013 for the cell calculation of UO_2 pin, 0.99998 for STACY, 1.0008 – 1.0021 for TCA-UO2, 1.0003 – 1.00224 for TCA-PuO_2, 1.002 for GODIVA and 0.99936 for JEZEBEL. That is, the results of MVP become larger 0.1 – 0.2 % for U-cores and smaller 0.1 % for small fast neutron Pu core. As this causes, we can consider the differences between both codes for unresolved resonance PT method, $S(\alpha,\beta)$, pointwise cross section production error, fission spectrum and inelastic scattering cross sections.

4. RESULTS AND DISCUSSIONS

The calculated results are shown in Tables 2, 3, 4 and 5. In these tables, the results calculated with JENDL-3.3 are compared with those calculated with JENDL-3.2, ENDF/B-VI.5 and JEF-2.2. We

Table 2 Comparison of the C/E(keff)-values for U-fuel thermal cores

Core	JENDL-3.3	JENDL-3.2	ENDF/B-VI.5	JEF-2.2
TCA150U	0.9960	1.0005	0.9921	1.0005*
TCA183U	0.9960	1.0011	0.9925	1.0014*
TCA248U	0.9967	1.0017	0.9935	1.0004*
TCA300U	0.9966	1.0011	0.9940	0.9983*
TRX-1	0.9920	0.9951	0.9902	0.9958*
TRX-2	0.9922	0.9954	0.9905	0.9937*
STACY	1.0036	1.0079	1.0002	*without U234: 0.15%
TRACY	1.0034	1.0082		

can see that the keff-values of STACY, TRACY[2] and JRR-4 in the highly enriched U fuel cores overestimated with JENDL-3.2 are improved significantly by decreasing of about 0.6 % with

JENDL-3.3. This is due to modification of the fission spectrum and thermal fission cross sections from JENDL-3.2 to JENDL-3.3 data. The keff-values of TRX cores are underestimated by all the nuclear data libraries, especially ENDF/B-VI.5. The keff-values for PuO_2 fuel cores of TCA calculated with all the nuclear data libraries are in very good agreement with the experiments.

Table 3 Comparison of the C/E-values (keff) for Pu-fuel thermal cores

Core	JENDL-3.3	JENDL-3.2	ENDF/B-VI.5	JEF-2.2
TCA242Pu	0.9966	0.9959	0.9946	0.9952
TCA298Pu	0.9975	0.9968	0.9948	0.9977
TCA424Pu	0.9985	0.9978	0.9954	
TCA555Pu	0.9988	0.9987	0.9967	

Table 4 Comparison of the C/E(keff)-values calculated for small cores

Cores	JENDL-3.3	JENDL-3.2	ENDF/B-VI.5	JEF-2.2
U-235 fuel				
GODIVA	1.0032	1.0030	0.9965	0.9953
FLATTOP-25	0.9983	0.9986	1.0037	0.9917
BIGTEN	0.9976	0.9984	1.0149	1.0044
Pu fuel				
JEZEBEL	0.9972	0.9972	0.9972	0.9970
JEZEBEL-Pu	1.0020	1.0015	0.9987	0.9990
FLATTOP-Pu	0.9923	0.9928	1.0041	0.9889
THOR	1.0068	1.0061	1.0059	0.9800
U-233 fuel				
JEZEBEL-23	1.0039	1.0129	0.9933	0.9641
FLATTOP-23	1.0003	1.0069	1.0028	0.9710

Table 5 Comparison of the keff and C/E-values for the ZPPR-9 and FCA-XVII-1 cores. The experimental keff-values are 1.00106 for ZPPR-9 and 0.9992 for FCA-XVII-1, respectively.

Nuclear data	Keff		C/E-value	
	ZPPR-9	FCA-XVII-1	ZPPR-9	FCA-XVII-1
JENDL-3.3	0.9976	1.0029	0.9965	1.0037
JENDL-3.2	0.9953	1.0011	0.9942	1.0019
ENDF/B-VI.5	1.0040	1.0104	1.0029	1.0112
JEF-2.2	0.9964	1.0073	0.9953	1.0081

The keff-values calculated for the fast neutron cores are shown in Tables 4 and 5. In the U-235 fuel cores, the keff-values for JENDL-3.3 and 3.2 show very good agreement with the experiments. For ZPPR-9 of large Pu FBR mock up core, JENDL-3.3 improves by 0.2 % the keff-value underestimated with JENDL-3.2. The causes are investigated by a sensitivity analysis with changing the nuclides from JENDL-3.2 to JENDL-3.3 as shown in Fig. 1. From this figure, the most important effect on keff is due to Iron of which total cross sections become large at the MeV to 100 keV region in JENDL-3.3, comparing with JENDL-3.2 data described in the previous paper by Nakagawa and Shibata. For the U-233 core, JENDL-3.3 shows the best estimation for keff-values in all the libraries.

5. CONCLUDING REMARKS

Thermal neutron spectrum core

U235-fuel: the keff-values of STACY, TRACY and JRR-4 overestimated with JENDL-3.2 were improved significantly by decreasing of about 0.6 % with JENDL-3.3. This is due to modification of the fission spectrum and thermal fission cross sections from JENDL-3.2 to JENDL-3.3 data. In TCA, the keff-values calculated from JENDL-3.3 were in good agreement with the experiments.

Pu-Fuel: the keff-values of TCA calculated from JENDL-3.3 were in good agreement with the experimental values.

Fast neutron spectrum cores

U235-fuel: the keff-values for JENDL-3.3 were in good agreement with the experimental values and the discrepancies between JENDL-3.2 and 3.3 were very small.

Pu-fuel: in ZPPR-9 and FCA-XVII-1, the keff values calculated with JENDL-3.3 became larger 0.2 % than those for JENDL-3.2.

U233-fuel: the keff-values overestimated with JENDL-3.2 were improved considerably with JENDL-3.3, due to reevaluation of U-233 fission cross sections in the high energy region.

Comparison of the results for JENDL-3.3 with those for ENDF/B-VI.5

For thermal and fast cores, and U-235, Pu and U-233 fuel cores, the keff-values for JENDL-3.3 showed better agreement with the experiments than those for ENDF/B-VI.5 as shown in Fig. 2.

Future works: Remained Benchmark calculations are as follows:

Reaction rate ratios and distributions. Reactivities for void, control rod and Doppler for FBRs.

Estimation of buildup nuclides by burnup calculation.

Comparative study between the discrepancies of keff-values calculated with MVP and MCNP.

Production of the multigroup cross section libraries for SRAC-J3.3 and JFS3-J3.3

REFERENCES

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(2) Miyoshi Y., Kaneko K., Nakamura et al.: private communication.

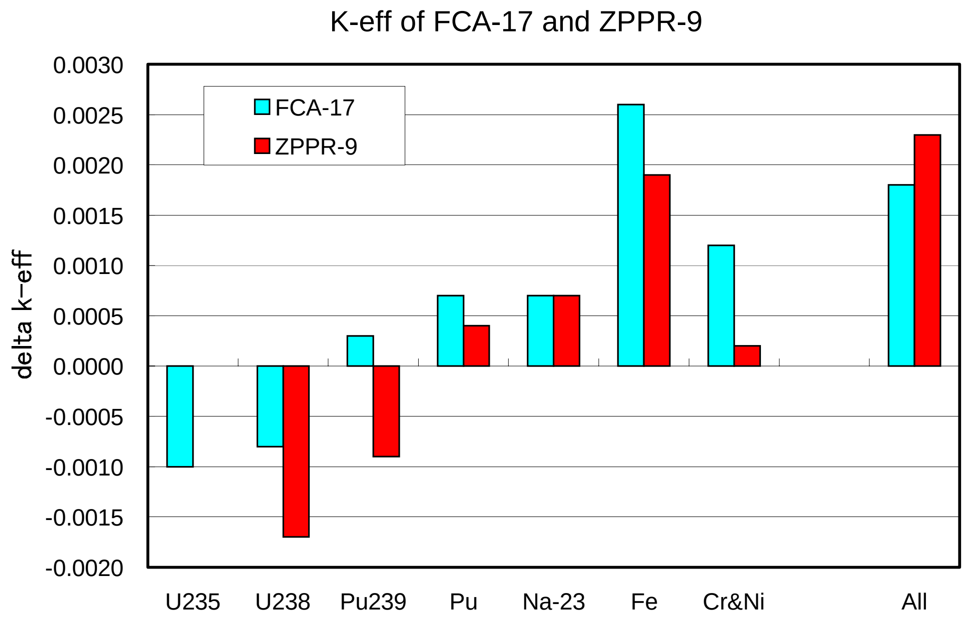

Fig. 1 Sensitivity analysis of nuclear data of nuclides changed from JENDL-3.2 to 3.3 in large FBR mockup cores of ZPPR-9 and FCA-XVII-1

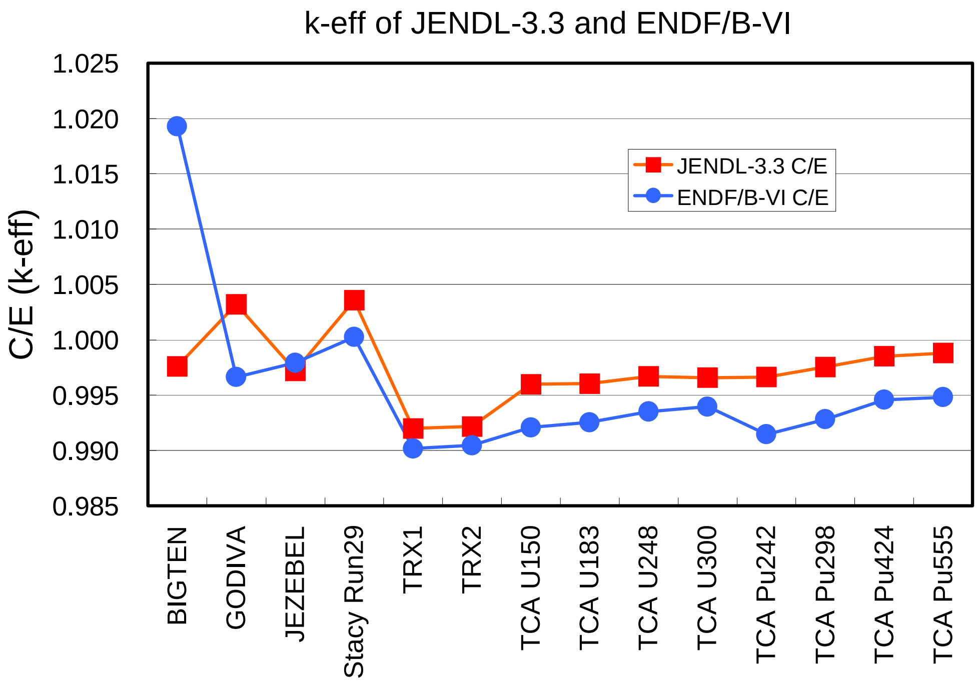

Fig. 2 Comparison of the (C/E-1) values for thermal cores calculated with JENDL-3.3 and ENDF/B-VI.5