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An Analysis of Fast Critical Experiments Using JEF-1-Based 50-Group Constant Set

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JEF-1의 50군 단면적에 의한 고속 임계실험 해석

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Abstract

JEF-1-based 50-group cross section set for fast reactor calculations was generated using NJOY system. The set was then examined by analyzing measured integral quantities such as criticality and central reaction rate ratios for 27 fast critical assemblies. The calculated results using the new set were also compared with those of ENDF/B-IV or-V-based fast set.

In general, the JEF-1-based set shows an improvement in predicting measured integral quantities in comparison with the previous set. With a few exceptions, JEF-1 results are comparable to those of ENDF/B-V.

요 약

NJOY 코드 씨스템으로 JEF-1평가 핵자료를 처리하여 고속로용 50군 군정수 SET을 생산하였다. 이를 이용하여 스물일곱가지 고속 임계로심 실험에서 얻어진 임계도 및 노심 중앙에서의 반응율비를 계산하고 측정치와 비교 · 분석하였다. 아울러 ENDF/B-IV와-V 자료로 해석한 결과와도 비교 · 검토하였다. 일반적으로, 임계실험의 적분량 추정에서 JEF-1의 결과는 지금까지 사용해온 ENDF/B-IV의 결과보다 개선되었고, ENDF/B-V의 결과에 근접하고 있다.

1. Introduction

In the 1980s, ENDF/B-IV[1] data were a main source for performing thermal/fast reactor analysis. In fact, both ENDF/B-IV-based KAERI-26G[2] and LIB-IV[3] fast sets have been used to perform fast reactor physics calculations in KAERI. At the beginning of 1990s, new versions of evaluated nuclear data libraries, such as ENDF/B-VI[4], JENDL-3[5], JEF-1[6] or BROND-2[7], have been released to all users in the world without restrictions.

JEF-1, the first version of the Joint Evaluated File, had been developed at the OECD/NEA Data Bank in co-operation with several laboratories in the member countries. It was finalized in 1986 and released in 1990 to the IAEA Nuclear Data Section for distribution to scientists in IAEA member states. For an application purpose to fast reactor physics calculations, a new JEF -1–based multigroup cross section set was processed by using NJOY code system[8]. The quality of data sets can be determined by comparing their performance in predicting measured integral quantities. It is also very desirable to compare the calculated results using the newly generated data set with the corresponding ones using the earlier data sets for several of fast reactor benchmarks recommended. In order to predict the performance of the JEF -1–based multigroup cross section set relative to the earlier ENDF/B–IV–based set, both the fast data sets were examined by analyzing a wide range of fast critical experments. The calculated–to–experimental ratios using the JEF–1–based set were also compared with those of ENDF/B–IV based LIB–IV set, and of ENDF/B–V–based LIB–V[9] set obtained from literature.

In addition, some JEF–1 results were addressed by comparing with ENDF/B–VI benchmark results from a recent report.[10]

2. Data Processing Procedure

NJOY is well known as the most general purpose and versatile nuclear data processing code system, and the NJOY91 is the latest version released.

A new 50 neutron group cross section set for fast reactor analysis has been generated. The set was processed by using NJOY91.38 starting from JEF–1 nuclear data. Figure 1 shows the processing scheme in detail.

The MODER module is used to convert ENDF–formatted data from the NJOY blocked–binary mode to formatted mode and vice versa. The RECONR module reconstructs ENDF/B resonance representations and interpolation laws so as to obtain a pointwise ENDF–formatted tape where all cross sections are represented within specified tolerances by tables with linear interpolations. In BROADR module, the cross sections are Doppler broadened and thinned. A tolerance of 0.1 percent for the above two modules run was used for the reconstructions, linearizations and thinnings. The UNRESR module is used when self–shielded average cross sections for the unresolved resonance energy region are required. The GROUPR module is used to obtain self–shielded multigroup cross sections and group–to–group scattering matrices. The energy bounds for the 50 groups are identical to those of the LIB–IV groups. The structure used quarter lethargy spacing in the middle energy range, with one–half lethargy groups above 500 keV and below 275 eV. This proces-

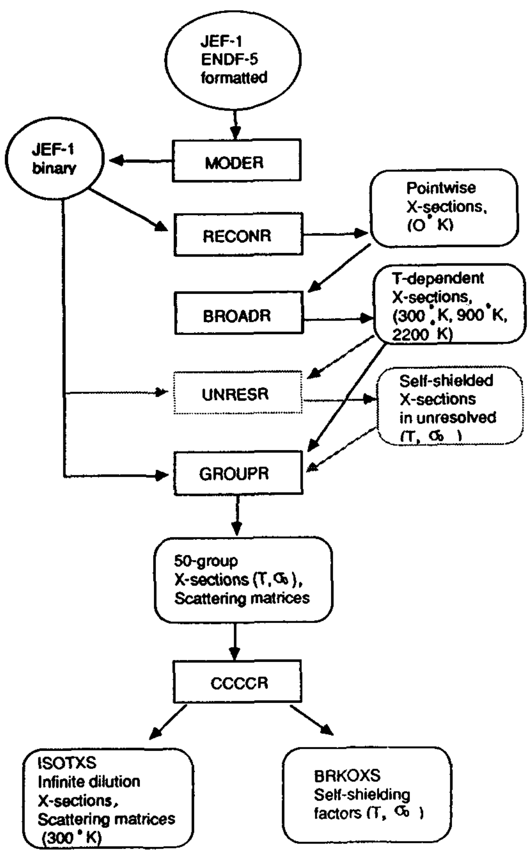

Fig. 1. Flow Diagram for Generating CCC–Format SPHINX Library.

sing methodology is also similar to LIB–IV except the weighting spectrum. An adequate weighting spectrum with respect to a broad class of applications was used to collapse the point cross sections to 50–group data. It has a fusion peak at high energies, followed by a fission spectrum, a slowing–down spectrum typical to a fast reactor, and a thermal tail. Figure 2 shows the weighting spectrum. The final step is to use the CCCCR module to format the multigroup data for CCCC standard interface files, ISOTXS and BRKOXS. The ISOTXS consists of infinite–dilution cross sections and scattering matrices, and the BRKOXS contains Bondarenko–type self–shielding factors.

All basic nuclear data used in this work are based on the JEF–1 file with exception of Nb–93(ENDF/B–IV) and Sn(ENDL–84).

In the fast reactor calculations for uranium– or plutonium–fueled core, the fission source spectrum for a main fissile isotope has been used. Three kinds of fission source spectra processed are shown in Table 1 with the 50 neutron energy group boundaries.

3. Characteristics of Critical Assemblies

Twenty–seven fast critical assemblies were analyzed in this study. 22 assemblies are those

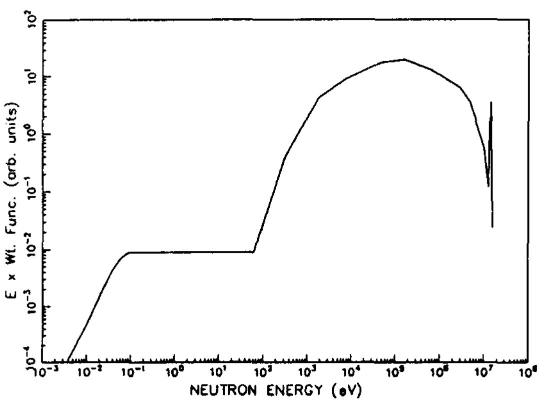

Fig. 2. Neutron Weight Function.

recommended by the Cross Section Evaluation Working Group(CSEWG)[11] as benchmark problems for fast reactors. And 5 non–CSEWG assemblies, which are often used for the fast reactor benchmark, were also added : a mock–up experiment of JOYO(FCA–V–2)[12] and four ZPR–3 assemblies 49, 50, 53 and 54[13]. These assemblies have a variety of the core composition, size and spectra. The selected critical assemblies are listed in Table 2, together with their key characteristics for the one–dimensional spherical models. The experiments include 16 plutonium–fueled and 11 uranium–fueled assemblies, including two ^{233}U–fueled assemblies. The assemblies are ordered in Table 2 by increasing core volumes. Table 2 also gives the principal contents of the fuel and reflector as well as the core radii and reflector thicknesses. The fuel fertile–to–fissile concentration ratios vary from 0.02 for ^{233}U cores to 8.8 for BIG TEN. Note that the experiments vary from a small bare sphere of 0.34 liter to 4000 liter core similar to that of a largy liquid metal fast breeder reactor.

4. Analysis

Measured integral quantities for selected 27 assemblies were analyzed by using SPHINX code system[14]. The SPHINX incorporates both one-dimensional diffusion and transport theory. The transport theory space calculation is that of the ANISN[15] code. In this analysis all calculations performed with spherical geometry model were based on the transport theory. All transport calculations were performed with P_3S_{16} approximation, using each of the 50–group JEF-1– and ENDF/B–IV–based cross section sets. Data analyzed in this study are effective multiplication factors(k_{eff}) and spectral indices at core center.

Table 1. 50–Group Boundaries and Fission Spectra

Group	Energy Boundary (Upper Limit)	U–235	U-233	Pu–239
1	1.9971E+07	1.3016E−03	1.5708E−03	2.5148E−03
2	1.0000E+07	2.4766E−02	2.5070E−02	3.1336E−02
3	6.0653E+06	1.1400E−01	1.1242E−01	1.1947E−01
4	3.6788E+06	2.1362E−01	2.0755E−01	2.0930E−01
5	2.2313E+06	2.2971E−01	2.2818E−01	2.2284E−01
6	1.3534E+06	1.7691E−01	1.7699E−01	1.7348E−01
7	8.2085E+05	1.1178E−01	1.1185E−01	1.1185E−01
8	4.9787E+05	3.6246E−02	3.6839E−02	3.6839E−02
9	3.8774E+05	2.6579E−02	2.7602E−02	2.6863E−02
10	3.0197E+05	1.9206E−02	2.0425E−02	1.9500E−02
11	2.3518E+05	1.3721E−02	1.4922E−02	1.3986E−02
12	1.8316E+05	9.7169E−03	1.0775E−02	9.9367E−03
13	1.4264E+05	6.8345E−03	7.7031E−03	7.0083E−03
14	1.1109E+05	4.7820E−03	5.4614E−03	4.9147E−03
15	8.6517E+04	3.3323E−03	3.8462E−03	3.4312E−03
16	6.7380E+04	2.3148E−03	2.6941E−03	2.3872E−03
17	5.2475E+04	1.6041E−03	1.8792E−03	1.6564E−03
18	4.0868E+04	1.1095E−03	1.3063E−03	1.1469E−03
19	3.1828E+04	7.6626E−04	9.0573E−04	7.9276E−04
20	2.4788E+04	5.2863E−04	6.2671E−04	5.4730E−04
21	1.9305E+04	3.6437E−04	4.3295E−04	3.7747E−04
22	1.5034E+04	2.5098E−04	2.9874E−04	2.6013E−04
23	1.1709E+04	1.7279E−04	2.0593E−04	1.7916E−04
24	9.1188E+03	1.1891E−04	1.4185E−04	1.2334E−04
25	7.1018E+03	8.1805E−05	9.7657E−05	8.4877E−05
26	5.5308E+03	5.6264E−05	6.7202E−05	5.8395E−05
27	4.3074E+03	3.8690E−05	4.6230E−05	4.0167E−05
28	3.3546E+03	2.6602E−05	3.1795E−05	2.7625E−05
29	2.6126E+03	1.8288E−05	2.1863E−05	1.8998E−05
30	2.0347E+03	1.2571E−05	1.5031E−05	1.3065E−05
31	1.5846E+03	8.6404E−06	1.0333E−05	8.9849E−06
32	1.2341E+03	5.9384E−06	7.1023E−06	6.1795E−06
33	9.6112E+02	4.0810E−06	4.8804E−06	4.2507E−06
34	7.4852E+02	2.8043E−06	2.9701E−06	2.9246E−06
35	5.8295E+02	1.9269E−06	1.5747E−06	2.0128E−06
36	4.5400E+02	1.3238E−06	8.3679E−07	1.3859E−06
37	3.5358E+02	9.0939E−07	4.4602E−07	9.5475E−07
38	2.7537E+02	1.0535E−06	3.6695E−07	1.1125E−06
39	1.6702E+02	4.9655E−07	1.0723E−07	5.3117E−07
40	1.0130E+02	2.3373E−07	3.2373E−08	2.5551E−07
41	6.1442E+01	1.1142E−07	1.0245E−08	1.2440E−07
42	3.7267E+01	5.2629E−08	0.0000E+00	6.1723E−08
43	2.2603E+01	2.4860E−08	0.0000E+00	3.1509E−08
44	1.3710E+01	1.1743E−08	0.0000E+00	1.6745E−08
45	8.3153E+00	3.4283E−10	0.0000E+00	9.3768E−09
46	5.0435E+00	1.6192E−10	0.0000E+00	5.5823E−09
47	3.0590E+00	7.6150E−11	0.0000E+00	3.5413E−09
48	1.8554E+00	3.5223E−11	0.0000E+00	2.3810E−09
49	1.1254E+00	6.3729E−13	0.0000E+00	1.6786E−09
50	6.8256E−01	0.0000E+00	0.0000E+00	5.2107E−09
	1.0000E−05			

Table 2. Fast Critical Assembly Characteristics on Spherical Model

Assembly	Main Fissile Fuel	Fertile -to- Fissile Ratio	Approx. Core Volume (liter)	Core Radius (cm)	Reflector Thickness (cm)	Comments
FLATTOP-23	U*	0.019	0.34	4.317	19.52	98.13 % U-233 metal, natural U reflector
FLATTOP-PU	Pu	0.050	0.39	4.533	19.597	Pu metal with 4.5 % Pu-240, natural Ureflector
THOR	Pu	0.054	0.63	5.310	23.57	Pu metal with 5.1 % Pu-240, Th-232 reflector
JEZEBEL-23	U*	0.019	0.90	5.983	-	Bare sphere with 98.13 % U-233 metal
FLATTOP-25	U	0.072	0.96	6.116	18.014	U metal, natural U reflector
JEZEBEL	Pu	0.047	1.09	6.385	-	Bare sphere with 4.5 % Pu-240 metal
JEZEBEL-PU**	Pu	0.258	1.20	6.599	-	Bare sphere with 20.1 % Pu-240 metal
GODIVA	U	0.066	2.80	8.741	-	Bare sphere with 93.77 % Pu-235 metal
VERA-11A	Pu	0.05	12	13.99	43.0	No U in core, diluted with graphite
VERA-1B	U	0.08	30	19.138	39.452	Enriched U, diluted with graphite
ZPR-3-6F	U	1.1	50	22.995	30.5	About 1 : 1 fertile to fissile
ZEBRA-3	Pu	8.6	60	23.68	30.5	Hard spectrum(80 % above 100 keV)
ZPR-3-12	U	3.8	100	28.76	30.5	4 : 1 U-C system, soft spectrum
SNEAK-7A	Pu	3.0	110	20.50	30.0	
BIG TEN	U	8.8	120	30.48	15.24	10 % U-235 metal, depleted U reflector
ZPR-3-11	U	7.5	140	31.61	30.0	U metal
ZPR-3-54	Pu	1.6	190	35.839	37.343	Similar to ZPR-3-35 except Fe reflector
FCA-V-2	Pu	2.3	200	36.232	30.0	JOYO Mock-up
ZPR-3-53	Pu	1.6	220	37.546	37.33	Similar to ZPR-3-54, except U reflector
SNEAK-7B	Pu	7.0	310	40.64	30.0	
ZPR-3-50	Pu	4.5	340	43.43	40.34	Similar to ZPR-3-49, except additional C added
ZPR-3-48	Pu	4.5	410	45.245	30.0	Graphite added to soften spectrum
ZEBRA-2	U	6.2	430	45.45	31.7	6 : 1 U-C system
ZPR-3-49	Pu	4.5	450	47.53	36.43	Similar to ZPR-3-48, except sodium removed
ZPR-3-56B	Pu	4.6	610	52.72	34.34	PuO_2-UO_2, predominantly Ni reflector
ZPR-6-7	Pu	6.5	3100	88.16	33.81	PuO_2, LMFBR design, L/D=0.9
ZPR-6-6A	U	5.0	4000	95.67	33.81	UO_2, LMFBR design, L/D=0.8

* U-233

** JEZEBEL-PU(20.1)

5. Results and Discussion

Calculated values were compared with those of experiments, and some statistical quantities were used to extract the characteristics between the new fast set and the earlier ones. The quantities are statistical average and average of absolute difference from unity for the ratio of calculated to experimental(C/E) values. In addition, it is very valuable to compare the JEF-1 results with the corresponding ENDF/B-V results for several of the CSEWG benchmark assemblies to predict the performance of JEF-1-based cross section set relative to ENDF/B-V-based one. For this, ENDF/B-V results obtained from Ref. 9 were also intercompared with the results obtained from JEF-1 and ENDF/B-IV.

5.1. Effective Multiplication Factors

The results of k_{eff} calculations are presented in Table 3 as the value of C/E, together with those

Table 3. Ratios of Calculated-to-Experimental Value of K_{eff}

Assembly	ENDF/B-IV	JEF-1	ENDF/B-V*
FLATTOP-23	.9824	.9780	-
FLATTOP-PU	1.0055	1.0030	-
THOR	.9829	.9934	-
JEZEBEL-23	.9650	.9687	-
FLATTOP-25	1.0142	.9982	-
JEZEBEL	1.0001	1.0105	1.0111
JEZEBEL-PU	.9944	1.0028	-
GODIVA	1.0082	.9987	1.0028
VERA-11A	.9890	.9862	.9884
VERA-1B	.9958	.9830	.9923
ZPR-3-6F	1.0154	1.0056	1.0101
ZEBRA-3	1.0060	1.0044	1.0090
ZPR-3-12	1.0107	1.0054	1.0061
SNEAK-7A	1.0054	1.0073	1.0056
BIG TEN	1.0167	1.0093	-
ZPR-3-11	1.0155	1.0084	1.0121
ZPR-3-54	.9880	1.0067	-
FCA-V-2	.9991	.9984	-
ZPR-3-53	9873	.9920	-
SNEAK-7B	1.0065	1.0072	1.0048
ZPR-3-50	.9895	.9941	-
ZPR-3-48	.9885	.9922	.9883
ZEBRA-2	.9931	.9912	.9988
ZPR-3-49	1.0057	1.0102	-
ZPR-3-56B	1.0085	1.0185	1.0106
ZPR-6-7	.9856	.9889	.9841
ZPR-6-6A	.9950	.9972	.9854
Average over all assemblies	.9983	.9985	
Average |C/E-1.0|	.0105	.0089	
Average ove 15 assemblies	1.0016	1.003	1.0006
Average |C/E-1.0|	.0086	.0087	.0090

* Values are from Ref. 9.

of ENDF/B-V. The ^{233}U-fueled cores, FLATTOP-23 and JEZEBEL-23, give calculated k_{eff}'s that are much lower than the measured value of unity. JEZEBEL-23 assembly was calculated to be more than 3% subcritical. The same tendency of the JEF-1 and ENDF/B-IV may be attributed to the basic data source of ^{233}U. ^{233}U data of JEF-1 are originated from ENDF/B-IV data. Disregarding ^{233}U-fueled assemblies gives an average C/E value of k_{eff} of 1.0005 with JEF-1 and an average deviation from unity of 0.0074.

As for two assemblies having an environment similar to that of a large liquid metal fast breeder reactor such as ZPR-6-7 and ZPR-6-6A, is clear that the JEF-1 set was an improvement in prediction of criticality comparing with the previous sets. With the exception of ^{233}U-fueled assemblies, JEF-1-based fast set gives a reasonable result comparable to the ENDF/B-V's in evaluation of eigenvalues.

5.2. Central Reaction Rate Ratios

For all assemblies, the central fission reaction rates in ^{238}U, ^{239}Pu, ^{240}Pu, ^{233}U, ^{234}U, ^{236}U and ^{237}Np, and the central neutron capture reaction rate in ^{238}U, *relative to the central fission reaction* rate in ^{235}U(F28/F25, F49/F25, F40/F25, F23/F25, F24/F25, F26/F25, F37/F25 and C28/F25), were calculated with both JEF-1-based set and ENDF/B-IV-based set. The respective C/E values for various reactions are presented in Table 4 through Table 11.

● F28/25

In non-CSEWG assemblies, calculated values for $^{238}U/^{235}U$ fission ratio are larger by more than 20% comparing with the experimental values. The results for ^{233}U-fueled assemblies are obtained to be ~10% lower than the measured values. Except the above assemblies, it is clear that JEF-1 is an improvement over ENDF/B-IV and-V.

● F49/F25

JEF-1 results are generally appeared to be better than those of ENDF/B-IV. For the CSEWG assemblies, the average of C/E values is unity for JEF-1 and the average error is smaller than 1.6%,

Table 4. Calculated–to–Experimental Values of Central Fission Rate in ^{238}U Normalized to ^{235}U Fission Rate

Assembly	Experiment	ENDF/B–IV	JEF–1	ENDF/B–V*
FLATTOP–23	.191	.8917	.9076	–
FLATTOP–PU	.18	.9186	.9714	–
THOR	.195	.9282	.9870	–
JEZEBEL-23	.2131	.8972	.9133	–
FLATTOP–25	.149	1.0273	1.0573	–
JEZEBEL	.2137	.9067	.9682	.9167
JEZEBEL–PU	.206	.9194	.9762	–
GODIVA	.1647	1.0297	1.0596	1.0364
VERA–11A	.102	.8763	.9173	.8996
VERA–1B	.086	.9670	.9927	.9535
ZPR–3–6F	.078	1.0324	1.0415	1.0342
ZEBRA-3	.0461	1.0158	.9911	1.0293
ZPR–3–12	.047	1.0962	1.0847	1.0917
SNEAK-7A	.0448	.9953	.9795	.9766
BIG TEN	.0373	1.0576	1.0391	–
ZPR-3-11	.038	1.0795	1.0645	1.0745
ZPR-3-54	.0254	1.2185	1.1949	–
FCA–V-2	.0396	1.1634	1.1270	–
ZPR–3–53	.0254	1.2185	1.2067	–
SNEAK–7B	.033	1.0558	1.0203	1.0340
ZPR-3-50	.0251	1.2108	1.1865	–
ZPR–3–48	.0326	1.0758	1.0497	1.0730
ZEBRA–2	.032	1.1078	1.0847	1.0603
ZPR–3–49	.0345	1.1046	1.0904	–
ZPR-3-56B	.0308	1.0042	.9636	.9971
ZPR–6–7	.02202	1.0345	.9918	1.0377
ZPR–6-6A	.02411	1.0212	.9946	.9996
Average over all assemblies		1.0316	1.0315	
Average |C/E–1.0|		.0835	.0653	
Average over 15 assemblies		1.0199	1.0129	1.0143
Average |C/E–1.0|		.0538	.0411	.0485

* Values are from Ref. 9.

which gives accurate results comparable to the ENDF/B–V results. However, the results for both ZPR–3–53 and 54, non–CSEWG assemblies, are ~4% lower than the measured values.

● F49/f25

JEF–1 results are generally appeared to be better than those of ENDF/B–IV. For the CSEWG assemblies, the average of C/E values is unity of JEF–1 and the average error is smaller than 1.6%, which gives accurate results comparable to the ENDF/B–V results. However, the results for both ZPR–3–53 and –54, non–CSEWG assemblies, are ~4% lower than the measured values.

● F40/F25 and F24/F25

Calculated results for both indices are generally overestimated for most of the assemblies. The overestimations of the ENDF/B–V's are higher than those of the JEF–1's. Remarkable overpre-

Table 5. Calculated–to–Experimental Values of Central Fission Rate in ^{239}Pu Normalized to ^{235}U Fission Rate

Assembly	Experiment	ENDF/B–IV	JEF–1	ENDF/B–V*
FLATTOP–25	1.37	.9901	1.0117	–
JEZEBEL	1.448	.9622	.9863	.9724
GODIVA	1.402	.9847	1.0074	.9943
VERA–11A	1.18	.9803	1.0116	1.0006
VERA–1B	1.2	9521	.9773	.9642
ZPR–3–6F	1.22	1.0224	1.0400	1.0384
ZEBRA-3	1.19	.9903	1.0034	1.0092
ZPR–3–12	1.12	.9976	1.0119	1.0129
SNEAK-7A	1.016	.9666	.9837	.9835
BIG TEN	1.185	.9891	1.0018	–
ZPR-3-11	1.19	.9875	.9998	1.0008
ZPR-3-54	.928	.9302	.9547	–
FCA–V-2	1.104	.9704	.9816	–
ZPR–3–53	.928	.9297	.9557	–
SNEAK–7B	1.012	.9931	1.0025	1.0099
ZPR-3-50	.903	.9878	1.0090	–
ZPR–3–48	.976	.9945	1.0109	1.0137
ZEBRA–2	.987	1.0091	1.0230	1.0157
ZPR–3–49	.986	1.0105	1.0286	–
ZPR-3-56B	1.028	.466	.9554	.9657
ZPR–6–7	.9425	.9763	.9864	.9955
Average over all assemblies		.9796	.9973	
Average |C/E–1.0|		.0244	.0181	
Average over 14 assemblies		.9831	1.0000	.9983
Average |C/E–1.0|		.0214	.0158	.0161

* Values are from Ref. 9.

Table 6. Calculated–to-Experimental Values of Central Fission Rate in ^{240}Pu Normalized to ^{235}U Fission Rate

Assembly	Experiment	ENDF/B–IV	JEF–1	ENDF/B–V*
VERA–11A	.475	1.0899	1.1186	1.1373
VERA–1B	.399	1.2419	1.2438	1.2571
ZPR–3–6F	.53	1.0352	1.0168	1.0640
ZEBRA-3	.373	1.0374	.9908	1.0885
ZPR–3–12	.34	1.0801	1.0267	1.1074
SNEAK-7A	.174	1.2152	1.1864	–
BIG TEN	.174	1.2137	1.1942	–
ZPR-3-11	.159	1.3821	1.3322	–
ZPR-3-54	.243	1.0886	1.0493	1.1078
ZEBRA-2	.237	1.1429	1.1429	1.1068
ZPR-3-56B	.282	.8671	.8671	.8812
Average over all assemblies		1.1267	1.0967	
Average |C/E–1.0|		.1509	.1297	
Average over 8 assemblies		1.0729	1.0438	1.0938
Average |C/E–1.0|		.1061	.0892	.1234

* Values are from Ref. 9.

Table 7. Calculated-to-Experimental Values of Central Fission Rate in ^{233}U Normalized to ^{235}U Fission Rate

Assembly	Experiment	ENDF/B-IV	JEF-1	ENDF/B-V*
FLATTOR-25	1.6	.9432	.9471	–
JEZEBEL	1.578	.9476	.9502	.9867
GODIVA	1.59	.9472	.9513	.9855
VERA-11A	1.49	.9981	1.0059	1.0195
VERA-1B	1.433	1.0359	1.0433	1.0537
ZPR-3-6F	1.53	.9909	.9954	1.0229
ZEBRA-3	1.542	.9843	.9905	1.0104
ZPR-3-12	1.48	1.0118	1.0175	1.0277
BIG TEN	1.58	.9616	.9676	–
ZPR-3-11	1.52	.9998	1.0054	1.0250
ZEBRA-2	1.453	1.0077	1.0149	1.0062
ZPR-3-56B	1.478	.9830	.9890	.9899
Average over all assemblies		.9842	.9899	
Average |C/E-1.0|		.0250	.0247	
Average over 10 assemblies		.9906	.9964	1.0127
Average |C/E-1.0|		.0205	.0211	.0203

* Values are from Ref. 9.

Table 8. Calculated-to-Experimental Values of Central Fission Rate in ^{234}U Normalized to ^{235}U Fission Rate

Assembly	Experiment	ENDF/B-IV	JEF-1	ENDF/B-V*
ZPR-3-6F	.451	1.0746	1.1101	1.1723
ZEBRA-3	.346	1.0018	1.0095	1.1220
ZPR-3-12	.305	1.0759	1.0773	1.1675
ZPR-3-11	.31	1.0635	1.0680	1.1661
ZEBRA-2	.153	1.4886	1.4703	1.5471
ZPR-3-56B	.195	1.0530	1.0503	1.1554
Average over all assemblies		1.1262	1.1309	1.2217
Average |C/E-1.0|		.1262	.1309	.2217

* Values are from Ref. 9.

Table 9. Calculated-to-Experimental Values of Central Fission Rate in ^{236}U Normalized to ^{235}U Fission Rate

Assembly	Experiment	ENDF/B-IV	JEF-1	ENDF/B-V*
VERA-1B	.134	1.3073	1.3151	1.3396
ZEBRA-3	.099	1.0858	1.0435	1.1475
ZPR-3-11	.12	.8043	.7780	.8358
ZEBRA-2	.093	.8527	.8312	.8624
ZPR-3-56B	.0639	1.0737	1.0504	1.1374
Average over all assemblies		1.0247	1.0036	1.0645
Average |C/E-1.0|		.1620	.1600	.1853

* Values are from Ref. 9.

Table 10. Calculated-to-Experimental Values of Central Fission Rate in ^{237}Np Normalized to ^{235}U Fission Rate

Assembly	Experiment	ENDF/B-IV	JEF-1	ENDF/B-V*
FLATTOP-23	.89	.9581	.9162	–
FLATTOP-PU	.84	.9823	.9600	–
THOR	.92	.9683	.9547	–
JEZEBEL-23	.977	.9503	.9538	–
FLATTOP-25	,76	1.0651	1.0281	–
JEZEBEL	.962	.9700	.9578	.9891
JEZEBEL-PU	.92	1.0005	.9815	–
GODIVA	.837	1.0476	1.0166	1.0640
VERA-11A	.43	1.1818	1.1498	1.2281
VERA-1B	.38	1.2744	1.2094	1.2850
ZEBRA-3	.353	1.0659	.9527	1.1176
BIG TEN	.316	1.0884	.9668	–
ZPR-3-11	.33	1.0755	.9574	1.1036
ZEBRA-2	.214	1.1785	1.0359	1.1439
Average over all assemblies		1.0576	1.0000	
Average |C/E-1.0|		.0820	.0628	
Average over 7 assemblies		1.1134	1.0399	1.1331
Average |C/E-1.0|		.1219	.0777	.1362

* Values are from Ref. 9.

Table 11. Calculated-to-Experimental Values of Central Fission Rate in ^{238}U Normalized to ^{235}U Fission Rate

Assembly	Experiment	ENDF/B-IV	JEF-1	ENDF/B-V*
THOR	.083	.8698	.8124	–
VERA-11A	.158	.7617	.7303	.7203
VERA-1B	.135	.9139	.8864	.9156
ZPR-3-6F	.104	.9389	.9387	.9260
ZPR-3-12	.123	.9676	9721	.9659
SNEAK-7A	.1376	.9868	.9797	.9913
BIG TEN	.11	.9807	.9961	–
ZPR-3-11	.112	.9634	.9795	.9696
FCA-V-2	.14	.9177	.9249	–
SNEAK-7B	.131	1.0287	1.0294	1.0298
ZPR-3-48	.138	.9736	.9720	.9768
ZEBRA-2	.136	.9777	.9768	.9544
ZPR-6-7	.132	1.0784	1.0785	1.0886
ZPR-6-6A	.1378	1.0322	1.0342	1.0479
Average over all assemblies		.9565	.9508	
Average |C/E-1.0|		.0634	.0695	
Average over 11 assemblies		.9657	.9616	.9624
Average |C/E-1.0|		.0596	.0642	.0679

* Values are from Ref. 9.

dictions for F40/F25 are observed especially for non-CSEWG assemblies.

● F23/F25

Since the source cross section data of ^{233}U for JEF-1 are from ENDF/B-IV as previously mentioned, there are no differences between the results of JEF-1 and ENDF/B-IV. Underpredictions for metal-fueled hard cores are improved by using ENDF/B-V.

● F26/F25 and F37/F25

There is still a large dispersion in C/E values for $^{236}U/^{235}U$ fission ratios. The result for $^{237}Np/^{235}U$ fission ratios is not improved with the use of ENDF/B-V, which remains unsolved.

● C28/F25

C/E values for the major part of calculated assemblies are lower than unity. Remarkable underpredictions are observed especially in the small size cores. However, all the values for ZPR-6-7 using the three sets are overestimated by about 10 percent.

● C28/F49

The ratio of the ^{238}U capture to ^{239}Pu fission rate is an index calculated and experimental values of the central capture in ^{238}U normalized to the ^{239}Pu fission rate(C28/F49) is given in Table 12. The calculated results underestimate the experimental values for most of the assemblies, and the three sets show a similar tendency for all assemblies. Average C/E value over all assemblies using JEF-1 is ~2% lower than that of ENDF/B-IV, and the same tendency is observed also for the ENDF/B-V case. It is noted that the average C/E value for U-fueled assemblies is ~10% lower than that for Pu-fueled assemblies, and especially ~10% overprediction for ZPR-6-7 still remains in the results obtained using the three data sets.

In evaluating the spectral indices, we can also find out the same results of over- or under-predictions for the non-CSEWG assemblies from both Hardie's study[13] using ENDF/B-IV data set and Takano's study[16] using JFS-3-J2 fast set. The experimental reaction rate data of non-CSEWG assemblies are not likely to be adequate for fast reactor benchmark.

5.3. Comparison with Some ENDF/B-VI Results

R.Q. Wright et al.[10] have recently reported a result of ENDF/B-VI fast reactor data testing at the Oak Ridge National Laboratory(ORNL). The ten benchmarks include three plutonium-fueled and seven uranium-fueled assemblies, including one ^{233}U-fueled assembly.

The ORNL eigenvalue of 0.9935 for the ^{233}U-fueled JEZEBEL-23 represents an improvement compared with the k_{eff} of 0.9687 using JEF-1.

The ORNL calculated-to-experimental values for central reaction rate ratios from the ENDF/B-VI calculations are given in Table 13, together with the JEF-1 results. The JEF-1 underpredictions of F28/F25 and F37/F25 for both JEZEBEL and JEZEBEL-23 assemblies, and F23/F25 for metal-fueled core selected by the ORNL benchmark show improvement with the use of the ENDF/B-VI data. The large differences in k_{eff} values from unity using JEF-1 for hard cores are improved by using ENDF/B-VI. Considering the central reaction rate ratios for the soft cores in Table 13, JEF-1 results are accurate enough to be comparable with the result of ENDF/B-VI.

6. Conclusions

JEF-1 file was processed with NJOY91.38 code system and a 50-group cross section set was generated for fast reactor calculations. Measured integral quantities such as k_{eff} and reaction rate

Table 12. Calculated–to–Experimental Values of Central Fission Rate in ^{238}U Normalized to ^{239}U Fission Rate

Assembly	Experiment	ENDF/B–IV	JEF–1	ENDF/B–V*
VERA-1B	.122	.8850	.8364	.8762
ZPR–3–6F	.085	.9209	.9052	.8941
ZPR–3–12	.11	.9685	.9592	.9518
SNEAK-7A	.135	1.0241	.9990	1.0111
ZPR–3–11	.094	.9768	.9810	.9702
FCA–V–2	.1268	.9457	.9423	–
SNEAK-7B	.129	1.0395	1.0303	1.0233
ZPR-3-48	.141	.9818	.9643	.9667
ZEBRA–2	.138	.9675	.9533	.9384
ZPR-6-7	.14	1.1050	1.0937	1.0979
Average over all assemblies		.9815	.9665	
Average $\lvert C/E-1.0 \rvert$		.0522	.0583	
Average over 9 assemblies		.9855	.9692	.9700
Average $\lvert C/E-1.0 \rvert$		.0520	.0584	.0594

* Values are from Ref. 9.

Table 13. Comparison of Calculated–to–Experimental Values of Central Reaction Rate Ratios from JEF–1 and ENDF/B–VI

Assembly	F28/F25	F37/F25	F23/F25	F49/F25	C28/F25
JEZEBEL	0.968	0.958	0.950	0.986	
	(0.983)*	(0.988)	(0.999)	(0.982)	
JEZEBEL–23	0.913	0.914			
	(1.011)	(1.010)			
JEZEBEL–Pu	0.976	0.982			
	(0.987)	(1.019)			
GODIVA	1.060	1.017	0.951	1.007	
	(0.981)	(0.987)	(1.001)	(0.988)	
FLATTOP-25	1.057	1.028	0.947	1.012	
	(0.992)	(1.006)	(0.997)	(0.994)	
ZPR–3–12	1.085		1.018	1.012	0.972
	(1.085)		(1.031)	(0.996)	(0.949)
ZPR–3–11	1.064		1.005	1.000	
	(1.093)		(1.036)	(0.995)	
BIG TEN	1.073	0.967	0.968	1.002	0.996
	(1.073)	(1.074)	(0.997)	(0.997)	(0.950)
ZPR–6-6A	0.995				1.034
	(0.876)				(0.991)
ZPR-6-7	0.992			0.986	1.078
	(1.001)			(0.958)	(1.004)

* Values in parentheses are results using ENDF/B–VI from Ref. 10.

raios at core center for most of the CSEWG fast reactor benchmarks and some non-CSEWG fast critical assemblies were calculated using the newly generated JEF-1-based set and the previous ENDF/B-IV-based set. The correlations between calculated and experiment using the new set were then compared with those of thé previous sets, from which the following conclusions are withdrawn :

- The JEF-1-based set is an improvement in prediction of criticality in comparison with the previous set. With a few exceptions, JEF-1 results are comparable with those of ENDF/B-V.
- Significant discrepancies between calculated and experimental reaction rate values are observed in the results for the non-CSEWG assemblies.
- Calculated values of ^{239}Pu to ^{235}U fission central reaction rate ratio for most of the CSEWG assemblies agree well with the experimental values within average error of ~1.6%.
- In the evaluation of ^{238}U caputure to ^{239}Pu fission reaction rate ratio, ~10% overprediction for ZPR-6-7 assemby still remains in the results obtained using the three data sets.

In summary, it is clear that JEF-1-based set shows an improvement over ENDF/B-IV-based set and gives reasonable results comparable to ENDF/B-V in the fast reactor physics calculations.

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