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V BENCHMARK TESTING

Thermal and Fast Reactor Benchmark Testing of ENDF/B-6.4

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Introduction

In 1995 and 1996, CNDC made homogeneous fast and thermal reactor benchmark testing of CENDL-2 and B-6.2(ENDF/B-6 version 2), respectively[1, 2]. It proved that ^{238}U data of CENDL-2 are better than those of B-6.2.

For fast reactor benchmarks, B-6.2 shows 1~2 % larger k_{eff} than CENDL-2 for the cores with ^{238}U fuel and reflector. The difference is mainly caused by ^{238}U data, especially its inelastic data. The inelastic scattering cross sections of ^{238}U from B-6.2 makes the fast neutron spectrum hardened and increases of neutron production rate. In thermal reactor benchmark testing, the k_{eff} values calculated using B-6.2 for both lattice assembly TRX-1 and TRX-2 with metal uranium fuel rod are underestimated from 0.6 % to 1 %. And for BAPL-UO_2-1, -2 and -3 with uranium oxide fuel rod lattice, the k_{eff} values calculated using B-6.2 are underestimated by about 0.2 % to 0.5 %.

Last year, CNDC received new version of B-6.4 (ENDF/B version 4). In order to understand the state of development of ENDF/B-6, it is necessary to carry out fast and thermal reactor benchmark testing of B-6.4 again.

The benchmark testing for B-6.4 was done with the same benchmark experiments and calculating method as for B-6.2[1, 2]. The effective multiplication factors k_{eff}, central reaction rate ratios of fast assemblies and lattice cell reaction rate ratios of thermal lattice cell assemblies were calculated and compared with testing results of B-6.2 and CENDL-2.

It is obvious that ^{238}U data files are most important for the calculations of large fast reactors and lattice thermal reactors. However, ^{238}U data in the new version of

ENDF/B-6 have not been renewed. Only data of ^{235}U, ^{27}Al, ^{14}N and ^{2}D have been renewed in ENDF/B-6.4. Therefor, it will be shown that the thermal reactor benchmark testing results are remarkably improved and the fast reactor benchmark testing results are not been improved.

1 Thermal Reactor Benchmark Testing

1.1 Multigroup Constant Generations and Benchmark Calculations

NSLINK code system[3] were used to process B-6.4 and generating 123 group cross sections in AMPX master library format. A modified code system[4] PASC-1 was used in the calculations. The first step, it calculates a lattice-cell spectrum of 123 groups in P_3 S_8 for heterogeneous assembly or infinite medium spectrum for homogeneous assembly and produces spectrum averaging cross section set of 48 groups. The second step, it is used for critical calculations. For heterogeneous assembly, the experimental total buckling is used to account for leakage correction. The calculated integral parameters include k_{eff} and lattice-cell reaction rate ratios ρ^{28}, δ^{25}, δ^{28} and C^{*}.

1.2 Effective Multiplication Factors

Table 1 presents the calculated k_{eff} values of 13 thermal reactor benchmark assemblies for B-6.4 obtained by CNDC along with the values of k_{eff} published for benchmark testing of B-6.2 and CENDL-2. In fact, new version data of ^{27}Al, ^{14}N and ^{2}D were changed only in the high energy region, and the variances between old and new versions do not affect the calculated results for thermal benchmarks testing.

In Table 1 all of the testing results for B-6.2 indicate that the calculated values of k_{eff} are underestimated notably. Naturally, the data of some important nuclides, for example, ^{238}U and ^{235}U in ENDF/B-6 library, should be reevaluated. In fact, ^{238}U data are most important but it has never been changed from B-6.2 to B-6.4. For ^{235}U in the B-6.4 the data files of σ_t, σ_f, σ_c and σ_e below 1 keV were changed and ν_f values below 10 keV were increased by 0.082 %, as compared with that of B-6.2. The variance between two versions results in increasing the calculated k_{eff} values.

For different assemblies the variances of the calculated values of k_{eff} are different. Let Δk_{eff} express the calculated value of k_{eff} using B-6.4 decreased by using B-6.2. For the light water moderated assemblies the Δk_{eff} values of TRX-1 and TRX-2 are less than these of BAPL-UO_2-1, -2 and -3. And the Δk_{eff} values of ZEEP-1, -2 and -3 with heavy water moderator are the biggest, compared with these assemblies with light water moderator. That is to say, k_{eff} is gradually increased with that spectrum of assembly softens. It is the reason that the more neutrons are absorbed in the light-water moderated system. Therefor, the increments of Δk_{eff} for assemblies ZEEP-1,-2 and -3 with heavy-water moderator are from 0.31% to 0.25%, respectively, and they are about two times larger than that for the others with light-water moderator.

Table 1 Results of k_{eff} Calculations

Assembly	CENDL-2	ENDF/B6.4	ENDF / B - 6 . 2					Δk_{eff}
	CNDC Sn, 123 g Ref. 2	CNDC Sn, 123 g this work	CNDC Sn, 123 g Ref. 2	ORNL Sn,123g Ref. 5	LANL Sn, 69g Ref. 6	JAERI MC Ref. 7	KAERI WIMS Ref. 8	
ORNL-1	0.9995	0.9978	0.9971	0.9965	0.9969	—	—	0.0007
ORNL-2	0.9990	0.9973	0.9968	0.9964	0.9967	—	—	0.0005
ORNL-3	0.9959	0.9943	0.9938	0.9935	—	—	—	0.0005
ORNL-4	0.9973	0.9957	0.9952	0.9950	—	—	—	0.0005
ORNL-10	0.9944	0.9932	0.9928	0.9961	0.9972	—	—	0.0004
TRX-1	0.9968	0.9923	0.9909	0.9894	0.9869	0.9919	0.9908	0.0014
TRX-2	0.9993	0.9947	0.9939	0.9915	0.9891	0.9925	0.9924	0.0008
BAPL-UO_2-1	0.9998	0.9966	0.9949	0.9975	0.9949	—	0.9952	0.0017
BAPL-UO_2-2	1.0007	0.9975	0.9957	0.9971	0.9959	—	0.9951	0.0018
BAPL-UO_2-3	1.0027	0.9992	0.9979	0.9972	0.9974	—	0.9960	0.0013
ZEEP-1	1.0019	1.0029	0.9998	—	—	—	—	0.0031
ZEEP-2	1.0001	1.0008	0.9981	—	—	—	—	0.0027
ZEEP-3	0.9987	0.9994	0.9969	—	—	—	—	0.0025

Note: $\Delta k_{eff} = k_{eff}(B6.4) - k_{eff}(B6.2)$

^{238}U data play an important role in thermal reactor calculations because its contents in the fuel rod of lattice-cell assembly are more than 96 % for light-water moderator system and 99 % for heavy-water moderator system, respectively. It is obvious that improving calculation results are essentially impossible when data of ^{238}U are neverre-evaluated .

1.3 Lattice Cell Reaction Rate Ratios

The lattice cell reaction rate ratios ρ^{28}, δ^{25}, δ^{28} and C^* of five lattice assemblies with light water moderator were calculated using B-6.4. The calculated results together with B-6.2 from CNDC[2] and KAERI[8] are given in Table 2.

Table 2 Calculation results of lattice-cell reaction rate ratios (C / E)

Assembly	ρ^{28} epithermal / thermal ^{238}U capture				δ^{25} epithermal / thermal ^{235}U fission			
	CENDL-2	ENDF / B-6.2		ENDF/B-6.4	CENDL-2	ENDF / B-6.2		ENDF/B-6.4
	CNDC	CNDC	KAERI	CNDC	CNDC	CNDC	KAERI	CNDC
TRX-1	1.0533	1.0448	1.0490	1.0343	1.0049	1.0102	0.9960	1.0034
TRX-2	1.0342	1.0248	1.0390	1.0177	0.9899	0.9945	0.9530	0.9881
BAPL-UO_2-1	1.0475	1.0373	1.0360	1.0257	0.9957	1.0000	0.9640	0.9926
BAPL-UO_2-2	1.0813	1.0704	1.0710	1.0571	1.0021	1.0065	0.9710	0.9990
BAPL-UO_2-3	1.0451	1.0374	1.0350	1.0236	1.0054	1.0094	0.9810	1.0023
	δ^{28} ^{238}U fission / ^{235}U fission				C^{*} ^{238}U capture / ^{235}U fission			
TRX-1	1.0148	1.0433	1.0610	1.0389	1.0039	1.0113	1.0150	1.0042
TRX-2	0.9812	1.0025	1.0026	0.9999	0.9919	0.9998	1.0050	0.9955
BAPL-UO_2-1	0.9977	0.9782	1.0000	0.9721	—	—	—	—
BAPL-UO_2-2	0.9174	0.9353	0.9570	0.9300	—	—	—	—
BAPL-UO_2-3	0.9223	0.9381	0.9650	0.9340	—	—	—	—

Table 2 shows that the lattice cell reaction rate ratios of calculation value to experimental value, that is C / E, for TRX-1 and -2 using B-6.4 are obviously improved, compared with using B-6.2. For example, the values of ρ^{28} for TRX-1 and -2 are decreased by 1% and 0.7%, respectively. Both of δ^{25} for two assemblies are decreased by 0.7%. δ^{28} and C^{*} are also improved. For assemblies with UO_2 fuel rods, however, improvements of values of δ^{25} are not obvious, except ρ^{28}. It seems that values of δ^{28} become a little bad.

In order to analyze these results conveniently lattice cell spectrum average cross sections and fluxes of six groups are calculated. The sixth group among them is the thermal group and the cut-off of thermal energy is 0.625 eV. The data of ^{235}U and ^{238}U cross sections and fluxes of fifth and sixth group from B-6.4 and B-6.2 are listed in Table 3, respectively.

First of all, it is found from Table 3 that both of the calculated thermal or epithermal average flux for same assembly using B-6.4 are decreased, as compared with using B-6.2. Besides, there are large differences in the calculated spectra for different assemblies. It can be seen that the changes of lattice cell parameters come from the changes of ^{235}U cross sections and flux spectrum from B-6.2 to B-6.4. And the changes are different for different assemblies.

^{235}U fission cross sections in the thermal energy region are increased 0.41 barn for TRX-2 and 0.528 barn for BAPL-UO_2-1. And its epithermal fission cross sections are decreased about 0.2 barn for all four assemblies. The differences of thermal fluxes for assemblies between B-6.4 and B-6.2, however, are small, therefore δ^{25} (epithermal / thermal ^{235}U fission) using B-6.4 is decreased 0.7% and is more close to experimental value.

Table 3 Comparison of lattice cell average fluxes and thermal and epithermal cross sections between B-6.2 andB-6.4

Assembly	TRX-1		TRX-2		BAPL-UO_2-1		BAPL-UO_2-3	
Lattice cell pitch	1.806/cm		2.174/cm		1.5578/cm		1.8057/cm	
E Upper /eV	1.324E3	0.625	1.324E3	0.625	1.324E3	0.625	1.324E3	0.625
E Lower /eV	0.625	1E-5	0.625	1E-5	0.625	1E-5	0.625	1E-5
B-6.4 average flux	5.7904	6.0753	5.4859	9.2544	7.5588	8.5173	6.6825	11.5766
B-6.2 average flux	5.7935	6.0852	5.4879	9.2640	7.5610	8.5284	6.6837	11.5864
$^{B-6.4}$flux – $^{B-6.2}$flux	–0.0031	–0.0099	–0.0020	–0.0096	–0.0022	–0.0111	–0.0012	–0.0098
^{235}U B-6.4 σ_c/b	15.196	56.075	15.300	56.913	16.007	63.690	16.172	66.438
^{235}U B-6.2 σ_c/b	14.048	56.101	14.157	56.973	14.796	63.732	14.962	66.514
$\Delta_c = {}^{B-6.4}\sigma_c - {}^{B-6.2}\sigma_c$	+1.148	–0.026	+1.143	–0.060	+1.211	–0.042	+1.210	–0.076
^{235}U B-6.4 σ_f/b	28.519	329.962	28.693	335.485	29.618	374.053	29.874	390.583
^{235}U B-6.2 σ_f/b	28.715	329.469	28.890	335.075	29.815	373.525	30.074	390.121
$\Delta_f = {}^{B-6.4}\sigma_f - {}^{B-6.2}\sigma_f$	–0.196	+0.493	–0.197	+0.410	–0.197	+0.528	–0.200	+0.462
^{238}U B-6.4 σ_c/b	1.0773	1.6252	1.7533	1.6394	2.2915	1.8313	2.4233	1.8983
^{238}U B-6.2 σ_c/b	1.0778	1.6253	1.7539	1.6395	2.2920	1.8314	2.4237	1.8984
$\Delta_c = {}^{B-6.4}\sigma_c - {}^{B-6.2}\sigma_c$	–0.0005	–0.0001	–0.0006	–0.0001	–0.0005	–0.0001	–0.0004	–0.0001

^{235}U thermal capture cross sections are decreased 0.026 barns for TRX-1 and 0.076 barns for BAPL-UO_2-3, and its epithermal capture cross sections are increased about 1.148 and 1.121 barns. It results in a decrement of epithermal fluxes. Owing to the difference of spectrum, for the same assembly the average ^{238}U capture cross sections using B-6.4 have decrement, though the data of ENDF/B-6 have not been changed. Consequently, ρ^{28} (epithermal / thermal ^{238}U capture) obtained using B-6.4 is decreased about 1% and is more close to experimental value. The most obvious improvement is ρ^{28}. It is natural that C^* is improved.

Furthermore, all σ_e values of ^{235}U in the thermal and epithermal regions are decreased by tens mb. All the σ_t values of ^{235}U in the thermal and epithermal regions are also increment because of the increment of thermal σ_f and epithermal σ_c.

2. Fast Reactor Benchmark Testing

2.1 Multigroup Constant Generations and Benchmark Calculations

NSLINK code system was applied to processing B-6.4 and generating 175 group library with VITAMIN-J energy group structure in AMPX master library format.

The PASC-1 code system was used in the calculations. Firstly, it performs a resonance self-shielding calculation based on the Bondarenko method and generates problem-dependent master data set. Then, it calculates k_{eff} and central reaction rate ratios with 175 groups in P_3 S_{32}.

2.2 Effective Multiplication Factors

In the fast reactor benchmarks testing, only the data files of ^{235}U and ^{241}Pu have been changed from B-6.2 to B-6.4. The variances of ^{241}Pu data matters little to testing results, because its content is too small in fast reactor concerned. The changes of cross section data of ^{235}U are merely arisen below 2 keV. Considering fast reactors the flux above this energy is much higher than that below this energy, so that there are no obvious changes for the recalculated reactor parameters.

Table 4 presents the calculated k_{eff} values of nine homogeneous assemblies for B-6.4 obtained by CNDC along with the values of k_{eff} published for benchmark testing of B-6.2[1],[5],[6],[7],[9]. As expected, the calculated k_{eff} values are almost the same with the two libraries of different versions. That is to say, there is no any improvement for the fast reactor testing results from B-6.2 to B-6.4.

It is well known that the results for uranium fuel system from CENDL-2 are better than those from B-6.4. The data of ^{238}U from CENDL-2 used in calculations gives good results for all of uranium fuel assemblies with hard and soft spectra. The k_{eff} value of BIG-10 for B-6.4 or B-6.2 was overestimated by 2%, because the calculated spectrum is too hard.

In fast reactor benchmark testing, both B-6.4 and B-6.2 show 1~2% larger k_{eff} than CENDL-2 for the cores with ^{238}U fuel and reflector. The difference is mainly caused from ^{238}U data, especially its inelastic data. Inelastic scattering cross sections of ^{238}U from ENDF/B-6 make harder fast neutron spectrum and increases neutron production rate. Consequently, it leads the k_{eff} value to a increment. Dr. Takano gave the same conclusion as our results for B-6.2[7]. Therefor, It is to say the data of ^{238}U of CENDL-2 is much better than that of ENDF/B-6.

Table 4 Results of k_{eff} calculations for fast reactor benchmark testing

ASSEMBLY	CENDL-2	B-6.4	ENDF / B-6.2				
	CNDC	CNDC	CNDC	ORNL	LANL	M. Caro	JAERI
	Ref. 1	This work	Ref. 1	Ref. 5	Ref. 6	Ref. 9	Ref. 7
GODIVA	1.00003	0.99946	0.99946	0.9960	0.9983	0.9954	0.9965
FLATTOP-25	1.00142	1.00782	1.00785	1.0018	1.0030	1.0007	1.0073
BIG-10 C	0.99541	1.01576	1.01576	1.0171	1.0105	1.0063	1.0149
C / E	0.99940	1.01984	1.01984	—	—	—	—
JEZEBEL	1.00430	1.00053	1.00056	0.9970	0.9989	0.9960	0.9972
JEZEBEL-Pu	1.00391	1.00181	1.00261	0.9980	0.9981	0.9893	0.9987
FLATTOP-Pu	1.00066	1.00883	1.00886	1.0029	1.0055	1.0025	1.0041
JEZEBEL-23	0.99463	0.99458	0.99458	0.9934	0.9940	0.9929	0.9933
FLATTOP-23	1.00187	1.00645	1.00645	1.0032	1.0041	1.0026	1.0028
THOR	1.00925	1.00721	1.00721	—	—	1.0056	1.0059

2.3 Central Reaction Rate Ratios

Table 5 represents the results of central reaction rate ratios of nine assemblies which were calculated by CNDC for B-6.2, B-6.4 and CENDL-2. The reaction rates are all relative to that of fission of ^{235}U. The C/E represents a ratio of calculation value to experimental value.

It is clear that there are no obvious variations in the calculated results between B-6.2 and B-6.4, except F28. For assembly JEZEBEL-Pu F28 is underestimated by 0.24% for B-6.4, compared with B-6.2. It arises from the variations of ^{241}Pu data in both of ENDF/B-6 libraries. The variations from B-6.2 to B-6.4 do not result in a obvious difference of two calculation values of F28 for JEZEBEL, because the nuclear density of ^{241}Pu is only 0.000117(nuclei/b-cm), about twelve times less than JEZEBEL-Pu.

Considering calculation results for CENDL-2, very satisfactory results were obtained for three uranium fuel assemblies. Especially, F28 and C28 for BIG-10 are much better than that from both of ENDF/B-6 libraries. F28 of BIG-10 using B-6.4 or B-6.2 is about 5 percent larger than experimental value but C28 is about 5 percent less than it, but the results for other assemblies with harder spectra are satisfactory.

Table 5 Central Reaction Rate Ratios (*C/E*)

Assembly	Exp.	CENDL-2	B-6.2	B-6.4	Assembly	Exp.	CENDL-2	B-6.2	B-6.4
GODIVA	F28 .1647	0.9866	0.9879	0.9879	JEZEBEL	F28 .2137	0.9708	0.9839	0.9836
	F49 1.402	0.9971	0.9883	0.9883		F49 1.448	0.9941	0.9818	0.9812
	F 37 0.837	0.9719	0.9883	0.9883		F37 0.962	0.9828	0.9874	0.9873
	F 23 1.590	0.9999	1.0002	1.0002		F23 1.578	1.0016	0.9987	0.9987
FLATTOP -25	F 28 0.149	0.9993	0.9968	0.9966	JEZEBEL -Pu	F28 0.206	0.9861	0.9941	0.9917
	F 49 1.370	1.0020	0.9953	0.9953		F37 0.920	1.0116	1.0164	1.0157
	F 37 0.760	0.9937	1.0141	1.0141	FLATTOP -Pu	F28 0.180	0.9733	0.9909	0.9908
	F 23 1.600	0.9920	0.9936	0.9936		F 37 0.840	0.9821	0.9987	0.9986
BIG-10	F 28 .0373	1.0046	1.0512	1.0512	JEZEBEL -23	F 28 .2131	1.0588	1.0560	1.0558
	C28 0.110	1.0032	0.9475	0.9473		F 37 0.977	0.9821	1.0256	1.0257
	F 49 1.185	0.9704	0.9948	0.9948	FLATTOP -23	F 28 0.191	1.0473	1.0453	1.0455
	F 37 0.316	0.9410	1.0639	1.0639		F 37 0.890	1.0111	1.0331	1.0330
	F 23 1.580	0.9850	0.9954	0.9954	THOR	F 28 0.195	0.9620	0.9760	0.9759
						C28 0.083	0.8471	0.8413	0.8413
						F 37 0.920	0.9512	0.9548	0.9548

Note : F28, ratio of ^{238}U fission rate to ^{235}U fission rate,
C28, ratio of ^{238}U capture rate to ^{235}U fission rate,
F49, ratio of ^{239}Pu fission rate to ^{235}U fission rate,
F37, ratio of ^{237}Np fission rate to ^{235}U fission rate,
F23, ratio of ^{233}U fission rate to ^{235}U fission rate.

3. Conclusions

To sum up, some conclusions can be obtained :

(1) Obvious improvement has been made in the thermal reactor benchmark testing of B-6.4, but the calculated values of k_{eff} are still 0.0017 lower for homogeneous assemblies, 0.0045 for TRX-1 and -2 and 0.0032 for BAPL-UO_2-1, -2 and -3, respectively, less than that of CENDL-2.

(2) For fast reactor benchmarks testing of B-6.4, there is no any improvement, as compared with B-6.2, both of ENDF/B versions show 1~2% larger k_{eff} than that of CENDL-2 for the cores or reflectors with ^{238}U fuel. The results from CENDL-2 are much better than those from Both ENDF/B-6.

(3) For the ^{235}U data of B-6.4, the values of σ_f in the thermal energy region and ν_f values below 10 keV are increased, compared with that of B-6.2. The variances between two versions result in the increase calculated k_{eff} values for thermal reactor benchmark testing but yet it does not avail for fast reactor benchmark testing of B-6.4.