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HOMOGENEOUS FAST REACTOR BENCHMARK TESTING OF CENDL-2 AND ENDF/B-6

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1995

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Abstract

How to choose correct weighting spectrum has been studied to produce multigroup constants for fast reactor benchmark calculations. A correct weighting option makes us obtain satisfying results of K_{eff} and central reaction rate ratios for nine fast reacto:. benchmark testing of CENDL–2 and ENDF / B–6.

Introduction

Recently, the revised nuclear data file[1] of ^{238}U was produced for CENDL–2.1. In order to do the validation for CENDL–2, especially for ^{238}U, it should be necessary to choose several sets of benchmark experiments which includes homogeneous and heterogeneous fast reactors, thermal reactors, fusion reactors and others. First of all, homogeneous fast reactor benchmark testing of CENDL–2 and ENDF / B–6 are given in this paper. The remainder of data testing will be released in the Communication of Nuclear Data Progress one after another.

Nine homogeneous fast assemblies with simple compositions and geometries are used in this data testing. They are recommended by CSEWG in the United States[2]. The effective multiplication factors and central reaction rate ratios of these assemblies were calculated and compared with others. It is worth notice that correct option of weighting spectrum used in generating multigroup constants is very important. The concerned calculational results will be discuss.

1 Description of Benchmark Assemblies

Nine fast critical reactors were used in this study. Their main characteristics are given in Table 1. All of these assemblies have simple geometry and uniform compositions, which facilitate calculational testing, especially for the uranium and plutonium isotope cross sections in the fission source range. Besides, BIG–10 with larger core volume and softer core spectrum is best suited to test ^{238}U cross sections of resonance region and above fission threshold.

Table 1 Critical assembly characteristics

	CORE		REFLECTOR	
ASSEMBLY	FUEL	RADIUS (cm)	MATERIAL	THICKNESS (cm)
GODIVA	Enriched U 92% ^{235}U	8.741	No	0.0
FLATTOP–25	Enriched U 91% ^{235}U	6.116	Natural U	18.041
BIG–10	Enriched U 10% ^{235}U	30.480	Depleted U	15.240
JEZEBEL	Pu	6.385	No	0.0
JEZEBEL–Pu	Pu, 20% ^{240}Pu	6.65985	No	0.0
FLATTOP–Pu	Pu	4.533	Natural U	19.597
JEZEBEL–23	^{233}U	5.983	No	0.0
FLATTOP–23	^{233}U	4.371	Natural U	19.520
THOR	Pu	5.310	^{233}Th	24.570

2 Theoretical Method

2.1 Generations of Multigroup Constants

NJOY–91.91[3] and MILER[4] code system were applied to processing evaluated nuclear data and generating 175 group cross sections with VITAMIN–J energy structure in the AMPX master library format from CENDL–2 and ENDF / B–6. NJOY–91.91 can produce infinitely multigroup averaged cross sections, transfer matrices and self–shielding factors dependent on reactions, temperature and σ_0. The output data file of multigroup cross sections from module GROUPR of NJOY is called the GENDF in ENDF / B format. The MILER read two GENDF data files independent and dependent on temperature, respectively. And then the two files are converted into a multigroup cross section data file with Bondarenko self–shielding factors in the AMPX master library format.

In order to test weighting spectrum effect on generating averaged cross–sections, three weighting functions, i. e. W–A (thermal maxwelliam + $1/E$ + fission spectrum), W–B (thermal + $1/E$ + fast reactor + fission + fusion) and W–C (VITAMING–Eweighting function, described in the option 11 of module GROUPR in the NJOY–91.91), were used in running code NJOY, respectively. From our calculational results it has been pointed out that the more close to the calculated reactor core spectrum the weighting function is, the more accurate values calculated integral parameters of the reactor become.

2.2 Benchmark Calculations

First of all, a problem–dependent AMPX working library is produced from the AMPX master library by such modules as AJAX–C, BONAMI–C, and NITWAL–S in the modified code system PASC–1[5].

The module AJAX–C can select the concerned multigroup data from AMPX master library to produce a new master library. The BONAMI–C performs a resonance self–shielding calculation based on the Bondarenko method and generates problem–dependent master data set. The NITAWL–S converts the AMPX master library into a AMPX working library. The XSDRNPM–C is a modified version of one–dimensional transport code XSDRNPM–S in the PASC–1 code system[6]. The modified XSDRNPM–C can calculate central reaction rate ratios of fast critical reactors.

Finally, the XSDRNPM–C was used in calculating K_{eff} and central reaction rate ratios with 175 groups in P_3 S_{32}.

3 Weighting Spectrum Effect

As mentioned above three weighting spectra have already specified to generate three sets of 175 group cross sections in the VITAMIN–J energy structure from CENDL–2. Three weighting functions, which are called weighting *A*, *B*, and *C*, respectively, are shown in the Fig. 1. Three uranium fuel assemblies were used in this study. The calculated results are listed in Table 2.

Table 2 Effects of weighting spectra on integral parameters

ASSEMBLY	GODIVA		FLATTOP–25		BIG–10		
	K_{eff}	F28	K_{eff}	F28	K_{eff}	F28	C28
EXP.	1.00000 ± 0.1%	0.1647 ± 1.1%	1.00000 ± 0.1%	0.149 ± 1.34%	0.996 ± 0.2%	0.0373 ± 1.07%	0.1100 ± 2.73%
W–A C	0.99681	0.1594	0.99737	0.1462	0.99415	0.03726	0.1103
C/E	0.99681	0.9678	0.99737	0.9812	0.99814	0.9989	1.0027
W–B C	0.99656	0.1588	0.99753	0.1457	0.99541	0.03747	0.1104
C/E	0.99656	0.9642	0.99753	0.9779	0.99940	1.0046	1.0032
W–C C	1.00003	0.1625	1.00142	0.1489	1.00211	0.03799	0.1100
C/E	1.00003	0.9866	1.00142	0.9991	1.00800	1.0161	1.0000

We also draw a picture with three reactor core spectra shown as Fig. 2, so as to further clarify the effects of different weighting function on integral parameters and to better understand the relationship between weighting and reactor spectrum. For convenience, all of spectra of weighting and reactor cores were normalized to the flux of the fission threshold energy group of ^{238}U.

As GODIVA is a very small bare metal sphere assembly of high enriched uranium, its spectrum is very hard and very close approximation to weighting spectrum C. The volume of core of FLATTOP–25 is only 0.96 liters. Therefore the core spectrum of FLATTOP–25 is also hard and the same spectrum as GODIVA has. Consequently, the calculated results for the harder weighting C are reasonable. Fortunately, they have been also better than that using weighting A and B. Owing to the fact that the weighting B is softer, the fission contribution of ^{238}U in the high energy range has been underestimated. It is the reason why K_{eff} and F28 for the weighting B have been decreased by about 0.4% and 2%, respectively, as compared with that for weighting C. At the same time, weighting A is hard, too. The excessive hard spectrum results in that fission contributions of ^{235}U have been underestimated and secondary fission spectrum neutrons have been decreased so as to decrease fission rate of ^{238}U. And the value of K_{eff} for system has been underestimated, too.

Because the BIG–10 has the larger core volume of 119 liters, its core spectrum is softened. It is very famous intermediate energy standard neutron field, yet. It is necessary that we should make use of the weighting B with softer fast reactor spectrum to generate multigroup cross sections. Obviously, the calculated results for the weighting B are reliable. Using the harder weighting C, the K_{eff} and F28 were overestimated by 0.8% and 1.2%, respectively. It was unexpected that using the hardest weighting A we obtained the lowest value of K_{eff}. In fact, the hard core spectrum results in increasing leakage neutrons from core and decreasing fission contribution of ^{235}U.

It will be seen from these results that a good selection of weighting function should be suitable to the calculated reactor spectrum. That is to say, the weighting function used in generating multigroup cross sections must approximate to the spectrum of the assembly as far as possible, especially for benchmark testing of nuclear data. It is the correct weighting option that makes us obtain satisfying results about the benchmark testing of CENDL−2 for three homogeneous uranium fuel assemblies.

4 Calculational Results of Integral Parameters

According to analyses in the preceding paragraph, three weighting functions were used for generating 175 group cross sections from CENDL−2 and ENDF / B−6. Transport calculations of 175 groups in P_3 S_{32} for nine fast critical assemblies listed in the Table 1 have been carried out using the benchmark calculational method described in the paragraph 2.2. The values of K_{eff} and central reaction rate ratios for these assemblies have been obtained.

4.1 Effective Multiplication Factors

Table 3 presents the calculated values of K_{eff} of nine homogeneous assemblies for CENDL−2 and ENDF / B−6 obtained by CNDC along with the values of K_{eff} published for benchmark testing of ENDF / B−6, JEF−2 and JENDL−3[7, 8].

The results of first two lines are right, because the correct weighting options were used and the transport calculations with resonance self−shielding processing are rigorous, too. Naturally, it is that are results of homogeneous fast reactor benchmark testing of CENDL−2 and ENDF / B−6. It may be true that the results from CENDL−2 are better than others. The data of the new evaluated ^{238}U of CENDL−2 used calculations lead to good results for all of uranium fuel assemblies with hard and soft spectra. The K_{eff} value of BIG−10 for ENDF / B−6 was overestimated by 2 %, because the calculated spectrum is too hard.

The calculated K_{eff} values of two plutonium metal bare sphere assemblies for CENDL−2 were overestimated by about 0.4 percent. However, the good results of that for ENDF / B−6 were obtained. Itis interesting that the calculated value of K_{eff} of FLATTOP−Pu with natural uranium reflector for CENDL−2 is much better than that for all of other evaluated libraries.

Table 3 Results of K_{eff} calculations

	CNDC				Ref. 7		Ref. 8
Assembly	CENDL−2*	ENDF / B−6*	ENDF / B−6△	ENDF / B−6□	ENDF / B−6	JEF−2	JENDL−3
GODIVA	1.00003	0.99946	0.99626	0.99626	0.9954	0.9934	1.0066
FLATTOP−25	1.00142	1.00785	1.00356	1.00101	1.0007	0.9898	1.0033
BIG−10 C	0.99541	1.01576	1.01693	1.00555	1.0063	0.9928	1.0038
C / E	0.99940	1.01984	1.02101	1.00959			
JEZEBEL	1.00430	1.00056	0.99753	0.99753	0.9960	0.9952	1.0001
JEZEBEL−Pu	1.00391	1.00261	1.00040	1.00040	0.9893	0.9898	0.9963
FLATTOP−Pu	1.00066	1.00886	1.00424	1.00742	1.0025	0.9887	0.9974
JEZEBEL−23	0.99463	0.99458	0.99301	0.99301	0.9929	0.9756	1.0206
FLATTOP−23	1.00187	1.00645	1.00341	1.00470	1.0026	0.9836	1.0175
THOR	1.00925	1.00721	1.00389	1.00719	1.0056	0.9797	0.9985

Note :

* W−C was used in generating multigroup constants for assemblies, except W−B for BIG−10. Transport calculations with resonance self−shielding.

△ W−A was used in generating multigroup constants with resonance self−shielding processing.

□ W−A was used in generating multigroup constants without resonance self−shielding processing.

4.2 Central Reaction Rate Ratios

The Table 4 presents the calculated results of central reaction rate ratios for nine assemblies. The reaction rates are all relative to that of fission of ^{235}U.

Table 4 Central reaction rate ratios (C / E)

			CNDC			Ref. 7		Ref. 8
ASSEMBLY		EXP.	CENDL–2*	ENDF / B–6*	ENDF / B–6$^{\triangle}$	ENDF / B–6	JEF–2	JENDL3
GODIVA	F28	0.1647	0.9866	0.9879	0.9686	0.9541	0.9535	1.0006
	F49	1.402	0.9971	0.9883	0.9871	0.9860	0.9922	
	F37	0.837	0.9719	0.9883	0.9805	0.9742	0.9609	
	F23	1.590	0.9999	1.0002	1.0010	1.0016	0.9676	
FLATTOP–25	F28	0.149	0.9993	0.9968	0.9759	0.9655	0.9708	1.0697
	F49	1.370	1.0020	0.9953	0.9945	0.9936	0.9983	
	F37	0.760	0.9937	1.0141	1.0069	1.0016	0.9868	
	F23	1.600	0.9920	0.9936	0.9947	0.9949	0.9621	
BIG–10	F28	0.0373	1.0046	1.0512	1.0657	1.0519	1.0142	1.0195
	C28	0.110	1.0032	0.9475	0.9818	0.9836	0.9998	
	F49	1.185	0.9704	0.9948	0.9992	0.9985	0.9872	
	F37	0.316	0.9410	1.0639	1.0780	1.0724	0.9720	
	F23	1.580	0.9850	0.9954	0.9973	0.9972	0.9773	
JEZEBEL	F28	0.2137	0.9708	0.9839	0.9736	0.9600	0.9528	0.9944
	F49	1.448	0.9941	0.9818	0.9838	0.9836	0.9893	
	F37	0.962	0.9828	0.9874	0.9932	0.9889	0.9624	
	F23	1.578	1.0016	0.9987	0.9998	1.0005	0.9659	
JEZEBEL–Pu	F28	0.206	0.9861	0.9941	0.9888	0.9675	0.9651	1.0063
	F37	0.920	1.0116	1.0164	1.0226	1.0169	0.9903	
FLATTOP–Pu	F28	0.180	0.9733	0.9909	0.9817	0.9730	0.9734	1.0117
	F37	0.840	0.9821	0.9987	1.0050	1.0042	0.9766	
JEZEBEL–23	F28	0.2131	1.0588	1.0560	1.0192	1.0081	0.9348	1.0619
	F37	0.977	0.9821	1.0256	1.0128	1.0079	0.9393	1.0192
FLATTOP–23	F28	0.191	1.0473	1.0453	1.0099	1.0030	0.9384	1.0696
	F37	0.890	1.0111	1.0331	1.0209	1.0184	0.9483	1.0300
THOR	F28	0.195	0.9620	0.9760	0.9657	0.9559	0.9781	1.0034
	C28	0.083	0.8471	0.8413	0.8471	0.8500	0.8301	
	F37	0.920	0.9512	0.9548	0.9605	0.9580	0.9633	

Note : ' * ' and '△' represent the same meaning as that in the Table 3.

Considering calculational results for CENDL–2, very satisfying results were obtained for three uranium fuel assemblies. Especially, F28 and C28 for BIG–10 are much better than that from other evaluated libraries. F49 for BIG–10 has been about 3 percent less than experimental value, although that for other assemblies with harder spectra are satisfying. The calculated values of F37 for CENDL–2 are generally underestimated, as compared with that for ENDF / B–6.

The calculated central reaction rate ratios for all the enriched uranium and plutonium fuel assemblies for ENDF / B–6 have been good, except that for BIG–10. Ours calculated values of F28 and C28 for BIG– 10 are 5.1% higher and 5.2% lower than experimental values, respectively. It may result from that slowing–down power of ^{238}U in high energy region is too weak. The calculated reaction rate ratios for assembly THOR are underestimated, especially, the calculated C28 is about 15 percent lower than the experimental value.

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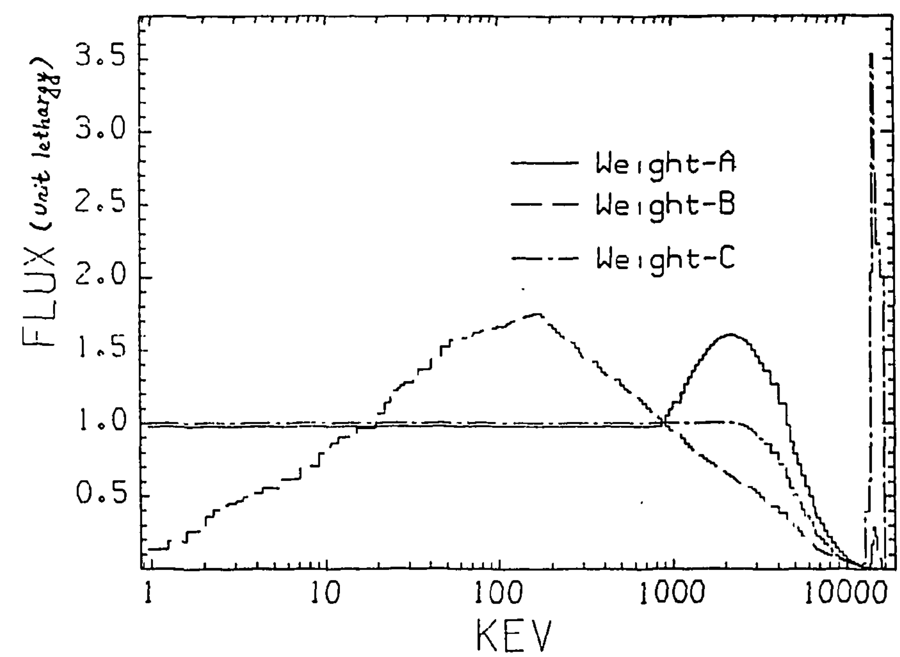

Fig. 1 Comparison of neutron weight spectrum

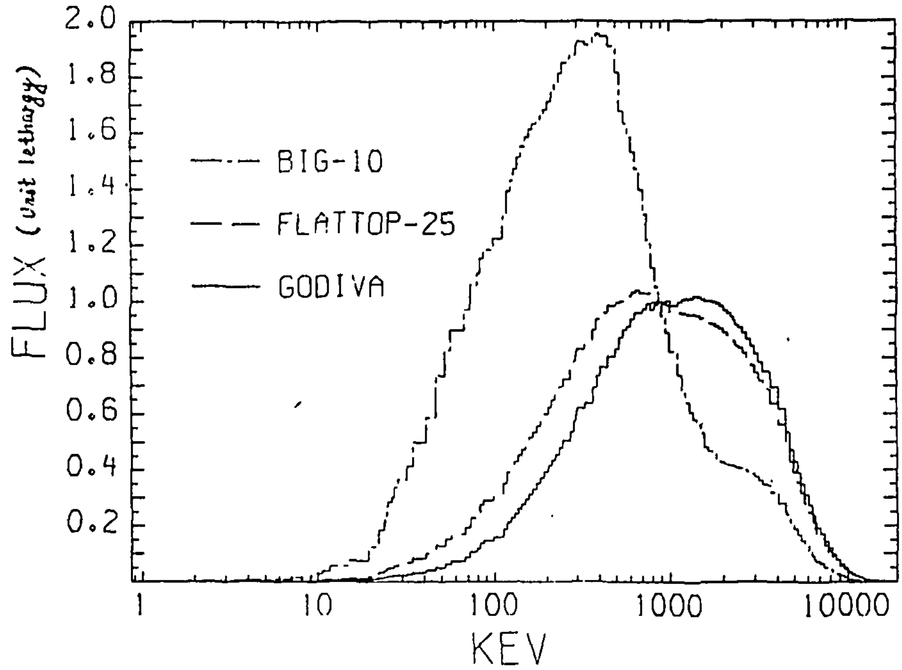

Fig. 2 Comparison of neutron flux spectrum

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