Journal of the Korean Nuclear Society
Volume 14, Number 4, December, 1982

《**Original**》

Generation and Benchmark Test of 26-group Constant Set for Fast Reactor Calculations

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(Received August 27, 1982)

고속로용 26군 군정수라이브러리 생산 및 벤치마크 계산

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(1982. 8. 27 접수)

Abstract

An ABBN-type 26-group constant set, KAERI-26G, which can be reliably applicable to fast reactor calculations has been generated using the nuclear data of ENDF/B-IV or ENDL-78 and a processing code ETOX-K4. The KAERI-26G set was evaluated by analysing measured integral quantities such as effective multiplication factor, central reaction-rate ratio, and central reactivity coefficient for a variety of critical assemblies. And these calculated quantities were compared with results from other workers using similar-type sets.

요 약

평가핵자료파일 ENDF/B-IV 및 ENDL-78과 군정수 계산용 전산코드 ETOX-K4를 이용하여 고속로 핵계산을 위한 ABBN형 26군 군정수세트 KAERI-26G를 생산하였다. 이 세트를 이용하여 여러형태의 고속임계로심들에 대한 유효증배계수, 반응율비, 반응도계수 등의 적분실험치를 해석하고 기 연구발표된 같은 형의 군정수세트에 대한 연구결과와도 비교하여 생산된 군정수세트의 신뢰도를 평가하였다.

1. Introduction

The methods of generating multigroup constant sets and the multigroup approximation methods of performing reactor calculations have undergone extensive development. Multigroup constants of the materials in a fast reactor depend on the particular compositions of each region in the reactor and hence, composition-independent group constant set does not suffice for all designs.

For detailed design analysis of a fast reactor, the multigroup constant set of effective cross sections should be developed for the particular reactor composition and design being analysed. Fast reactor design analyses have been successfully performed using the self-shielded multigroup constant set of Bondarenko et al. (ABBN set),[1] in reactor developing countries.

Briefly, the principal advantage of ABBN set (also called "Bondarenko set" or "Russian set") is that calculations can be made for reactors of various compositions using the same set of multigroup constants. The main approach used in the developing countries of a fast reactor to obtain multigroup constants for design purpose was ABBN-type.[2~5]

ABBN-type group constants differ from the usual multigroup cross sections by the inclusion of table of temperature and composition dependent correction factor (also called "f-factor" or "self-shielding factor"). Effective group cross sections are then obtained by multiplying the regular infinitely dilute (i.e. pseudo-composition-independent) cross sections by self-shielding factors that were determined by the medium's temperature and composition.

An ABBN-type 26-group constant set, KAERI-26G, was generated. The set was evaluated by analysing measured integral quantities for 17 fast critical assemblies. The integral quantities investigated in this study were effective multiplication factor, central reaction-rate ratio, and central reactivity coefficient. And these calculated quantities were compared with results obtained recently by other workers using their sets.

The detailed specifications for the multigroup constant set, the processing method, and the analysing performed will be discussed in the following sections.

2. Calculational Procedure

2.1. Generation of Group Constants

Detailed nuclear calculations of a fast reactor have been recently performed using the multigroup constant set of about 40- to 70-group structure.[6~9] However, because of computer time consuming of these, the 26-group structure of ABBN set has been commonly used to calculations of fast reactor design base. Therefore, the same group structure of ABBN set was selected for KAERI-26G set. The structure is provided for the energy range from thermal to 10.5 MeV.

In the average procedure, a combined 1/E and fission neutron spectrum was used for weighting. The fission spectrum joins 1/E at 0.8 MeV, which is lower boundary of 5th group, and last 26th group data are 2,200 m/s cross sections.

The three temperature values used by original ABBN set, such as 300, 900 and 2,100°K, adequately cover most reactor temperature and can reflect self-shielding factor behavior sufficiently. Therefore, the same values were selected here.

And the selection of σ_0 values, which are the total cross sections per atom of all other isotopes, is more difficult and requires lots of experience. Self-shielding factors are needed for a wide range of σ_0 values. But general multigroup codes allow limited number of σ_0 values for each reaction type, such as 1DX code[10] allows only six σ_0 values. In KAERI-26G process, six σ_0 values for each nuclide, were taken from ref. 8.

In order to generate group constants, ENDF/B-IV[11] and ENDL-78[12] were used for basic nuclear data and ETOX-K4 code[13] which is an extended version of ETOX-3 code,[14] for processing.

In the fission source calculations, the nuclear temperature of 1.35 MeV was used for uranium-235 and 1.4 MeV for plutonium-239.

2.2. Calculation of Integral Quantities of Fast Critical Assemblies.

In order to assess the reliability of the generated KAERI-26G set to the design calculation of a fast reactor, benchmark test calculations of fast critical experiments were performed for 14

Table 1. Characteristics of Selected Fast Critical Assemblies.

Assembly	Fissile Fuel	Fertile -to-Fiss. Ratio	Approx. Core Vol. (liter)	Laboratory
VERA-1B	U	0.081	30	AWRE(UK)
ZPR-3-6F	U	1.1	50	ANL(USA)
ZPR-3-12	U	3.8	100	ANL
ZPR-3-11	U	7.5	140	ANL
ZEBRA-2	U	6.2	430	AWRE
ZPR-6-6A	U	5.0	4000	ANL
VERA-11A	Pu	0.05	12	AWRE
ZEBRA-3	Pu	8.6	60	AWRE
SNEAK-7A	Pu	3.0	110	KFK(WG)
ZPR-3-53	Pu	1.6	220	ANL
SNEAK-7B	Pu	7.0	310	KFK
ZPR-3-50	Pu	4.5	340	ANL
ZPR-3-48	Pu	4.5	410	ANL
ZPR-3-49	Pu	4.5	450	ANL
ZPR-3-56B	Pu	4.6	610	ANL
ZPPR-2	Pu	6.5	2400	ANL
ZPR-6-7	Pu	6.5	3100	ANL

critical assemblies[15] designated by the Cross Section Evaluation Working Group (CSEWG) as "Phase II fast reactor data testing criticals" and three ZPR-3 assemblies,[16] 49, 50 and 53 which were experimental cores of ANL for cross section check. Only the principal characteristics of the assemblies used in this study are included in Table 1.

The fast critical assemblies have a variety of characteristics, from 12 to 4,000 liters in core size, from almost no fertile material to a fertile-to-fissile ratio of >8, and uranium- and plutonium-fueled cores.

The calculations for all the selected assemblies except ZPPR-2 were performed by one-dimensional diffusion-theory in spherical geometry. The model for assembly ZPPR-2 was one dimensional cylindrical geometry. Therefore, several factors such as one-to-two dimensional, diffusion-to-transport theory, and heterogeneity correction factors, were applied to calculated effective multiplication factors (K_{eff}).

A one-dimensional diffusion code, 1DX, was used to calculate K_{eff}'s, real and adjoint fluxes for perturbation, and central reaction-rate ratios. And then, a perturbation code, PERT-V[17] was used to calculate material reactivity worths at the core center.

A total of 80 mesh intervals were used in the one-dimensional diffusion calculations. The single exception was assembly ZPPR-2 in which case they were 90 mesh intervals.

3. Result and Discussion

3.1. Group Constant Set

For fast reactor calculations, an ABBN-type 26-group constant set, KAERI-26G, consisting of a total of 34 nuclides was generated. Almost all the basic nuclear data adopted for the generation of the set are ENDF/B-IV, and some of them (i.e. Ti, Ga and Sn) are ENDL-78.

The set contains the following nuclides;

H-1, Be-9, B-10, B-11, C-12, O-16, Na-23, Mg, Al-27, Ti, Si, V, Cr, Mn-55, Fe, Ni, Cu, Ga, Nb-93, Mo, Sn, Pb, Th-232, U-233, U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241.

The included data are infinitely dilute cross sections, inelastic transfer matrices, self-shielding factors of total, elastic, capture and fission cross sections.

And the set contains two fission spectra, one for U-235 and the other for Pu-239.

3.2. Effective Multiplication Factor

The results of calculated K_{eff}'s for experimentally critical systems using 1DX and KAERI-26G set are given in Table 2 in the form of C/E values (the ratio of calculated to experimental value), together with those obtained from original ABBN,[16] HEDL(ENDF/B-III[16]), LIB-V(ENDF/B-V[9]) and JFS-V-II set[7] by studies of Hardie et al., Kidman and Takano et al. The calculated values are corrected using the factors given in ref. 16, because the correction

Table 2. Comparison of Calculated Keff Values of Fast Critical Assemblies obtained by Different Sets.

Assembly	Corrected K_{eff}				
	KAERI-26G	ABBN	HEDL	JFS-V-II	LIB-V
VERA-1B	1.00129	1.0242	1.0026	1.00360	0.9980
ZPR-3-6F	1.01189	1.0274	1.0087	1.01661	1.0079
ZPR-3-12	1.00766	1.0222	1.0017	1.00697	1.0053
ZPR-3-11	1.01315	1.0072	0.9924	1.00796	1.0121
ZEBRA-2	0.99998	1.0121	0.9902	0.98523	0.9982
ZPR-6-6A	1.00012	1.0250	0.9988	1.00191	0.9916
VERA-11A	0.99259	1.0186	0.9935	0.99240	0.9948
ZEBRA-3	1.00205	1.0051	0.9816	0.99797	1.0073
SNEAK-7A	1.00397	1.0376	1.0006	1.00508	1.0080
ZPR-3-53	0.99682	1.0387	1.0008	0.99650	
SNEAK-7B	1.00166	1.0287	0.9893	1.00436	1.0068
ZPR-3-50	0.99689	1.0386	0.9940	0.99848	
ZPR-3-48	1.00360	1.0418	0.9997	1.00306	1.0067
ZPR-3-49	1.00427	1.0401	0.9985	1.00416	
ZPR-3-56B	0.98975	1.0271	0.9948	0.99668	1.0041
ZPPR-2	1.00067	1.0452	0.9994	1.00874	1.0019
ZPR-6-7	0.99679	1.0420	0.9926	1.00332	0.9986
Statistical Values					
U-fueled Cores					
A.V.	1.00568	1.0197	0.9991	1.00371	1.0022
S.D.	0.00549	0.0074	0.0063	0.00949	0.0069
Pu-fueled Cores					
A.V.	0.99901	1.0330	0.9950	1.00098	1.0035
S.D.	0.00461	0.0116	0.0056	0.00463	0.0045
All Cores					
A.V.	1.00136	1.0283	0.9964	1.00194	1.0030
S.D.	0.00588	0.0121	0.0061	0.00688	0.0057

factors are independent of group constant sets.

From Table 2, the statistical average value (A.V.) and the standard deviation(S.D.) of KAERI-26G set for all the assemblies are 1.0014 and 0.5%, respectively. The KAERI-26G set gives calculated K_{eff} that is a little higher than measured value by ~0.14% on the average. However, as far as plutonium-fueled assemblies are concerned, the average value of K_{eff}'s is 0.9990 and the standard deviation 0.46%.

If we could take the target accuracy of 0.5%[18] requested from nuclear reactor designer, the above result of plutonium-fueled cores seems to be excellent. Considering the experimental errors of ~0.3%, the average value and the standard deviation obtained from all the assemblies are also remarkable values of predicting criticality.

ABBN set gives calculated K_{eff}'s that are much higher than the measured values by ~2.8% on the average. The over-reactive character of original ABBN set was reevaluated by Orlov et al.[19] in 1970. In the processing, the infinitely dilute cross sections were reevaluated only for U-235 and Pu-239. Cuculeanu et al.[20] show that the revised set obtains the results of 3.3% underprediction from original ABBN set.

The results of other three sets also show a good one of calculating K_{eff}'s for fast critical assemblies. The ENDF/B-V is a latest version of ENDF/B and not released until now. However, there are no major changes in the value of K_{eff} obtained using LIB-V set.

3.3. Central Reaction-Rate Ratio

The calculated central reaction-rate ratios in the form of C/E value for the most part of critical assemblies are presented in Table 3, which includes the fission rates in U-238, Pu-239 and Pu-240, and the capture rate in U-238, relative to the fission rate in U-235. And the ratio of the capture rate in U-238 to the fission rate in Pu-239 is also included in the Table. For comparative study, the results from HEDL, LIB-V, and JFS-V-II set are summarized in Table 4.

The average C/E value of $\langle\sigma_f^{238}/\sigma_f^{235}\rangle$ is overestimated by $\sim$5% and the standard deviation is $\sim$8%. We can also detect much overestimated trend at some assemblies, such as VERA-1B, ZPR-3-53 and ZPR-3-50, and the results using other sets show the same trend. Therefore, the measured data should be checked more carefully for such cases. Disregarding such extreme cases gives only $\sim$2% overestimation on the average. Since the fission cross section of U-238 is a threshold reaction, it is sensitive to high energy spectrum. The fact that C/E values of $\langle\sigma_f^{238}/\sigma_f^{235}\rangle$ are high suggests that a part

Table 3. Comparison of Calculated and Experimental Values for Central Reaction-Rate Ratios(C/E).

Assembly	$\langle\sigma_f^{238}/\sigma_f^{235}\rangle$	$\langle\sigma_f^{239}/\sigma_f^{235}\rangle$	$\langle\sigma_f^{240}/\sigma_f^{235}\rangle$	$\langle\sigma_c^{238}/\sigma_f^{235}\rangle$	$\langle\sigma_c^{238}/\sigma_f^{239}\rangle$
VERA-1B	1.2181	1.0668	1.2682	0.9532	0.8966
ZPR-3-6F	1.0217	1.0243	0.9710	0.9472	0.9276
ZPR-3-12	1.0804	0.9999		0.9656	0.9641
ZPR-3-11	1.0856	0.9886	1.0387	0.9690	0.9814
ZEBRA-2	1.0463	0.9994	1.0389	0.9562	0.9553
ZPR-6-6A	0.9443			1.0088	
VERA-11A	1.1006	1.0649	0.9801		
ZEBRA-3	1.0198	0.9866	0.9865		
SNEAK-7A	0.9292	0.9584		0.9734	1.0189
ZPR-3-53	1.1350	0.9141	1.0822		
SNEAK-7B	0.9862	0.9874			
ZPR-3-50	1.1462	0.9775	1.2509	1.0208	1.0374
ZPR-3-48	1.0427	0.9899	1.0030		
ZPR-3-49	1.0747	1.0061		0.9593	0.9389
ZPR-3-56B	0.9602	0.9442	0.7960		
ZPPR-2	1.0708	0.9791	1.0387		
ZPR-6-7	0.9284	0.9596		1.0271	1.0482
Statistical Values					
U-fueled Cores					
A.V.	1.0661	1.0158	1.0796	0.9667	0.9450
S.D.	0.0826	0.0281	0.1126	0.0202	0.0298
Pu-fueled Cores					
A.V.	1.0358	0.9789	1.0196	0.9952	1.0109
S.D.	0.0741	0.0364	0.1258	0.0293	0.0428
All Cores					
A.V.	1.0465	0.9904	1.0413	0.9781	0.9743
S.D.	0.0785	0.0381	0.1245	0.0280	0.0488

Table 4. Comparison of Statistical C/E Values for Central Reaction-Rate Ratios obtained by Different Sets.

Reaction-Rate Ratio	Stat. Value	Set			
		KAERI-26G	HEDL	LIB-V	JFS-V-II
$<\sigma_f^{238}/\sigma_f^{235}>$	A.V.	1.0465	0.983	1.0644	1.0222
	S.D.	0.0785	0.090	0.0638	0.0755
		(17)*	(17)	(13)	(17)
$<\sigma_f^{239}/\sigma_f^{235}>$	A.V.	0.9904	0.986	1.0157	0.9859
	S.D.	0.0381	0.037	0.0352	0.0365
		(16)	(16)	(12)	(16)
$<\sigma_f^{240}/\sigma_f^{235}>$	A.V.	1.0413	0.980	1.0912	1.0679
	S.D.	0.1245	0.110	0.0934	0.1114
		(11)	(11)	(8)	(11)
$<\sigma_c^{238}/\sigma_f^{235}>$	A.V.	0.9781	0.988	0.9859	0.9821
	S.D.	0.0280	0.034	0.0392	0.0204
		(10)	(10)	(10)	(10)
$<\sigma_c^{238}/\sigma_f^{239}>$	A.V.	0.9743	0.999	0.9682	0.9938
	S.D.	0.0488	0.063	0.0592	0.0422
		(9)	(9)	(8)	(9)

* The value in parenthesis is the number of cases calculated.

of the discrepancy may be caused by incorrectly calculated spectra.

The C/E value of $<\sigma_f^{239}/\sigma_f^{235}>$ is underestimated by ~1% on the average and the standard deviation is ~3.8% These results are better than those from other sets as shown in Table 4.

There is a large scatter in the C/E values of $<\sigma_f^{240}/\sigma_f^{235}>$ from 0.796 for ZPR-3-56B to 1.268 for VERA-1B, and standard deviation is over 12%. The average value of $<\sigma_f^{240}/\sigma_f^{235}>$ for uranium-fueled cores gives higher than that for plutonium-fueled cores by ~6%. And the results from LIB-V set are poorer than those from KAERI-26G set. From these, a reevalua-

Table 5. Delayed Neutron Fission Spectra in the 26-Group.

Group	Spectrum*	
	U-235	Pu-239
5	0.1205	0.1173
6	0.3475	0.3402
7	0.2780	0.2954
8	0.1631	0.1516
9	0.0705	0.0736
10	0.0162	0.0173
11	0.0030	0.0036
12	0.0012	0.0010

*All other group data are 0.0.

Table 6. Calculated Values of the Effective Delayed Neutron Fraction, the Neutron Generation Time and the Inhour to ΔK/K Conversion Factor.

Assembly	β_{eff}	Generation Time(sec)	No. of Inhours per %ΔK/K
VERA-1B	0.007714	0.10856	398.88
ZPR-3-6F	0.007502	0.07571	431.13
ZPR-3-12	0.007587	0.10183	441.58
ZPR-3-11	0.007443	0.06748	470.95
ZEBRA-2	0.007520	0.21223	446.86
ZPR-6-6A	0.007306	0.51647	434.90
ZEBRA-3	0.004702	0.06098	799.10
SNEAK-7A	0.003770	0.17708	888.20
ZPR-3-53	0.003282	0.48876	937.37
SNEAK-7B	0.004322	0.16458	813.08
ZPR-3-50	0.003704	0.35694	905.05
ZPR-3-48	0.003776	0.26766	900.97
ZPR-3-49	0.003823	0.24015	898.36
ZPR-3-56B	0.003351	0.50015	954.94
ZPPR-2	0.003493	0.46924	936.64
ZPR-6-7	0.003540	0.50790	935.65

tion of Pu-240 fission cross section data or a careful checking of measured data would be very desirable.

$<\sigma_c^{238}/\sigma_f^{239}>$ value is an important component of the breeding ratio for fast breeder reactors. The differences in the results of $<\sigma_c^{238}/\sigma_f^{239}>$ value for different fueled cores are still detected, and these are also similar to the result of $<\sigma_c^{238}/\sigma_f^{235}>$ value. The average C/E value of $<\sigma_c^{238}/\sigma_f^{235}>$ and $<\sigma_c^{238}/\sigma_f^{239}>$ for uranium-fueled cores is ~3 and ~7% less than for plutonium-fueled cores, respectively. About plutonium-fueled cores considered the average value of $<\sigma_c^{238}/\sigma_f^{239}>$ is overestimated by ~1% and the standard deviation is ~4%. Therefore, the breeding ratio of a plutonium-fueled core will be predicted within an error of ~5%. The underestimated trend of $<\sigma_c^{238}/\sigma_f^{235}>$ or $<\sigma_c^{238}/\sigma_f^{239}>$ value is thought to be due mostly to the capture cross section data of U-238.

3.4. Central Reactivity Coefficient

Small sample central reactivity worths for a number of materials in critical assemblies were calculated.

To allow a more consistent comparison between measurement and calculation, the worths were converted back from $\Delta K/K$ values to inhours. The inhours to $\Delta K/K$ conversion factors were calculated in this study. The all delayed-neutron data, except delayed-neutron fission spectrum which was collapsed from Kidman's study,[9] required for this factor were obtained

Table 7. Comparison of Calculated and Experimental Values for Central Reactivity Coefficients(C/E).

Assembly	U-235	U-238	Pu-239	Cr	Fe	Ni	Al	B-10	Na	C	Mn	Mo
VERA-1B	0.9422	1.5439	0.9640				1.1814	0.9974	0.4660	1.0829		
ZPR-3-6F	0.8370	2.9993	1.0459	1.8733	1.1102	1.8112	0.9634	0.9602	0.3440	0.7642	0.4666	−1.2534
ZPR-3-12	0.9565	0.8744	0.9899		1.1491	1.2067	1.4987			0.3420		1.1165
ZPR-3-11	1.0671	0.9586	1.0170	1.3270	1.3124	1.3930	1.5591	0.9251	2.0210	1.6251	1.4013	1.2110
ZEBRA-2	1.0737	1.0196	1.0491	1.4903	1.3541	1.0519	1.2963	0.8411	2.0161	0.5838	2.1799	
ZPR-6-6A	1.0970	1.1502	1.0478				0.9360	2.3933	0.4398			
ZEBRA-3	1.1342	0.9758	1.1180				1.0074	0.9672	1.2970	1.0257		
SNEAK-7A	1.0902	1.1905	1.0929		0.9398			1.0140				
ZPR-3-53	1.2102	0.9205	1.1278	1.7327	2.1780	1.3190		0.7348	1.0544			
SNEAK-7B	1.0421	1.0901	1.0310		1.0726			0.7814				
ZPR-3-50	1.0670	0.8770	1.0514	1.6687	1.3167	1.3005		0.8164	−1.6115			
ZPR-3-48	1.1293	1.0916	1.1135	1.5122	1.2276	1.3756	1.2056	1.0266	1.6693	4.4862	1.8626	1.3414
ZPR-3-49	1.0862	0.9584	0.9966	1.3898	0.9610	1.1101		0.8777	0.4051			
ZPR-3-56B	1.1252	1.1424	1.1739	1.2227	0.9984	1.2433		0.9470	1.8779	1.5158		
ZPR-6-7	1.1329	1.0205	1.1469	1.4361	1.1628	1.2242	1.3267	1.1287	1.0884	1.6074	1.3847	
Statistical Values												
U-fueled Cores												
A.V.	0.9956	1.1093*	1.0184					0.9320				
S.D.	0.0922	0.2352	0.0335					0.0518				
Pu-fueled Cores												
A.V.	1.1130	1.0296	1.0947					0.9438				
S.D.	0.0461	0.0996	0.0543					0.1117				
All Cores												
A.V.	1.0661	1.0581*	1.0641	1.5170	1.2319	1.3036	1.2548	0.9395				
S.D.	0.0894	0.1661	0.0601	0.1951	0.3147	0.1978	0.1978	0.0949				

* neglecting ZPR-3-6F

Table 8. Comparison of Statistical C/E Values for Central Reactivity Coefficients obtained by Different Sets.

Material	Considered Core	Stat. Value	Set		
			KAERI -26G	HEDL	LIB-V
U-235	U-fueled	A. V.	0.9956	1.014	1.0106
		S. D.	0.0922	0.121	0.0994
	Pu-fueled	A. V.	1.1130	1.194	1.0686
		S. D.	0.0461	0.043	0.0574
	All	A. V.	1.0661	1.122	1.0396
		S. D.	0.0894	0.121	0.0862
			(15)*	(15)	(12)
U-238	U-fueled	A. V.	1.1093	1.205	1.1217
		S. D.	0.2352	0.208	0.2549
	Pu-fueled	A. V.	1.0296	1.205	1.0485
		S. D.	0.0996	0.115	0.0546
	All	A. V.	1.0581	1.205	1.0817
		S. D.	0.1661	0.154	0.1803
			(14)	(14)	(11)
Pu-239	U-fueled	A. V.	1.0184	1.018	1.0463
		S. D.	0.0335	0.062	0.0466
	Pu-fueled	A. V.	1.0947	1.172	1.1186
		S. D.	0.0543	0.051	0.0561
	All	A. V.	1.0641	1.111	1.0802
		S. D.	0.0601	0.094	0.0648
			(15)	(15)	(12)
Cr	All	A. V.	1.5170	1.467	1.6992
		S. D.	0.1951	0.211	0.3755
Fe	All	A. V.	1.2319	1.047	1.2151
		S. D.	0.3147	0.325	0.1703
Ni	All	A. V.	1.3036	1.237	1.2960
		S. D.	0.1978	0.168	0.2085
Al	All	A. V.	1.2548		1.2477
		S. D.	0.1978		0.2956
B-10	All	A. V.	0.9395	0.975	0.9423
		S. D.	0.0949	0.119	0.0712

* The value in parenthesis is the number of cases calculated.

from ENDF/B-IV. Table 5 gives the 26-group delayed-neutron fission spectrum for U-235 and Pu-239 fission.

A one-dimensional PERT-V calculation for each assembly was made to obtain the value of effective delayed neutron fraction (β_{eff}), the conversion factor from inhours to $\Delta K/K$ and the neutron generation time. Table 6 gives the results of these calculations.

Calculated central reactivity coefficients are compared with the experimental values in Table 7 and the summarized results obtained from other two sets, HEDL and LIB-V, are also given in Table 8, respectively.

Although the average C/E value of central reactivity coefficient of U-235 for all uranium-fueled assemblies is close to unity discrepancy of U-235 reactivity worth between the calculated and the experiment for each plutonium-fueled assembly is still apparent. And there is also a similar trend in the result of Pu-239.

The standard deviation of U-238 is$\sim$17%, and it is found that C/E values of U-238 differ from unity by more than 10% in about half of considered assemblies. But the reactivity coefficients for U-238 using LIB-V set are not also revised at all. As mentioned in the reaction-rate ratio, an adjusting of U-238 data may be suggested.

Average C/E value of reactivity coefficients of Cr are very high compared to Fe, Ni and Al values. Such differences are also detected in the results from HEDL and LIB-V set. Calculated reactivity coefficients of B-10 are low compared to experimental values. The underprediction of $\sim$6% and the standard deviation of $\sim$10% of B-10 are not removed in the calculations using LIB-V set.

Finally, reactivity coefficient values of the remaining four materials presented in the Table 7 are still hard to analyse.

4. Conclusion

An ABBN-type 26-group constant set, KAERI-26G, consisting of a total of 34 nuclides, which can be reliably applicable to the calculation of fast critical assemblies and of large fast breeder power reactors, has been generated using the nuclear data of ENDF/B-IV or ENDL-78 and a processing code, ETOX-K4.

Based on the analyses of many fast critical assemblies using KAERI-26G set, and the comparative trend evaluations of those using similar type sets, the following conclusions can be made.

1) The average K_{eff} values of 11 plutonium-fueled fast critical assemblies is 0.9990 with a standard deviation of 0.46%, and the value of all 17 assemblies considered is 1.0014 with a standard deviation of 0.59%.

2) As a result of comparative analyses for central fission rate in Pu-240 normalized to the U-235 fission rate, a reevaluation of Pu-240 fission data or a careful checking of measured data would be very desirable.

3) The underestimated trend of the central capture rate in U-238 normalized to the U-235 or Pu-239 fission rate is thought to be due mostly to the capture data of U-238. However, the breeding ratio of plutonium-fueled fast breeder reactor will be predicted within an error of about 5%.

4) Calculated central reactivity worths of U-235 and Pu-239 for plutonium-fueled critical assemblies gives higher than those for uranium-fueled assemblies by about 10% on the average.

In summary, it was verified that the KAERI-26G set was a powerful tool in predicting fast reactor characteristics.

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