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Substantiation of Data Files of JEFF-3.1.2 for Safety Analysis of TRIGA Mark-II Reactor through the Scrutiny of Integral Parameter of Benchmark Lattices TRX and BAPL

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Abstract: The aim of this analysis is to bear out the nuclear data files of JEFF-3.1.2 for theoretical safety analysis of a 3 MW TRIGA MARK-II research reactor is custom-made at AERE, Dhaka, Bangladesh through the study of integral parameters of benchmark lattice TRX and BAPL of thermal reactor. The basic evaluated nuclear data files of JEFF-3.1.2 are selected for TRIGA reactor and processed by using nuclear data processing code NJOY99.0. Different cross-sections of U-235 and U-238 are computed from the NJOY output of the evaluated nuclear data library. The 69 group cross-section library is engendered from the processed file for reactor code WIMSD-5B. From the generated 69 group cross-section library, the integral parameters of yardstick lattices TRX and BAPL are premeditated by using cell code WIMSD-5B. The calculated integral parameters are compared to the deliberated values as well as the consequences of Monte Carlo Code MCNP. From the assessment it is found that all the integral parameters are in good concurrence with some suspicions. Through benchmarking the integral parameters of TRX and BAPL lattices this analysis reflects the support to the evaluated nuclear data files of JEFF-3.1.2 for safety analysis of TRIGA Mark-II research reactor at AERE, Dhaka, Bangladesh.

Keywords: BAPL, JEFF-3.1.2, NJOY99.0, TRIGA MARK-II, TRX and WIMSD-5B

1. Introduction

The Evaluated Nuclear Data Files (ENDF) system was built up for the storage and repossession of evaluated nuclear data to be used for applications of nuclear technology [1]. A vital corollary of each appraisal must be completed for its intended purpose. These applications run many features of the system including the selection of materials to be comprised, the information used, the formats used and the testing is required before a library is released. If the entailed data are not obtainable for various finicky reactions, the evaluator should provide them by using nuclear models [2]. The evaluated data sets are prepared in ENDF format and converted into forms appropriate for testing and actual applications using processing codes. Processing codes that generate point-wise and group averaged cross sections for use in neutronics calculations from an ENDF library are available. The basic data formats for an ENDF library are developed in such a manner that few constraints are placed on using the data as input to the codes that generate any of the secondary libraries [3]. The computer code NJOY99.0 [4] is used for converting nuclear data in ENDF-6 format into libraries as Joint Evaluated Fission and Fusion Data Library (JEFF) in Europe [5], Japanese evaluated nuclear data library (JENDL) [6], Chinese evaluated nuclear data library (CENDL) [7], ENDF/B-VI in USA [8] and BROND in Russia [9]. Joint Evaluated Fission and Fusion Data Library is high quality nuclear data libraries for accessible and prospective nuclear energy systems and this library involves evaluation efforts that cover the main nuclear data needs in the fields of fission and fusion applications [10]. The updated version JEFF-3.1.2 which is modified edition of JEFF-3.1.1 [11] is our interest in this present research. To study the safety analysis of Triga Mark –II research reactor by JEFF-3.1.2 data library, a careful verification is required for the usable data files of that library. The WIMS code is a freely accessible thermal reactor physics lattice-cell code used widely especially by scientists for thermal research and power reactor applications. In the present work the 69 group cross-section library is generated by using computer program NJOY99.0 and WIMSD-5B [12, 13]. The bench mark lattice TRX and BAPL is used for benchmarking the integral parameter for the generated library. The calculated integral parameters are compared to the standards values.

2. Methods

The tools of this study are computer program: NJOY99.0, WIMSD-5B; evaluated nuclear data library: JEFF-3.1.2; benchmark lattices: Thermal Reactor-one region lattice (TRX) and Bettis Atomic Power Laboratory-one region lattice (BAPL).

2.1. Computer Code NJOY99.0

The nuclear data processing system NJOY having new version NJOY99.0 is a modular computer code used for data processing in ENDF-6 format. One of the common applications of NJOY99.0 is to generate 69 group cross-section library from basic nuclear data library. The 69 group cross-section library for WIMSD-5B code from basic data files of JEFF-3.1.2 is created by NJOY99.0.

Table 1. *Properties of TRX benchmark lattice.*

Segment	External radius in cm	Nuclei	Concentration (E 24 atoms/cm^3)
Fuel	0.4915	^{235}U	6.2530E-04
		^{238}U	4.7205E-02
Void	0.5042	---------	------
Clad	0.5753	Al	6.025E-02
Moderator	*	^{1}H	6.676E-02
		^{16}O	3.338E-02

*Lattices spacing are 1.8060 cm& 2.1740 cm in triangular arrays

Table 2. *Properties of BAPL benchmark lattice.*

Segment	External radius in cm	Nuclei	Concentration (E 24 atoms/cm^3)
Fuel	0.4864	^{235}U	3.1120E-04
		^{238}U	2.3127E-02
Void	0.5042	---------	------
Clad	0.5753	Al	6.025E-02
Moderator	**	^{1}H	6.676E-02
		^{16}O	3.338E-02

**Lattices spacing are 1.5578cm, 1.6523cm and 1.8057 cm

Table 3. *Concern isotope of TRIGA with the respective material ID.*

SL. NO.	Isotope	Material ID.
01	1-H-1	125
02	5-B-10	525
03	6-C-12	625
04	7-N-14	725
05	8-O-16	825
06	13-Al-27	1325
07	14-Si-28	1425
08	24-Cr-52	2431
09	25-Mn-55	2525
10	26-Fe-56	2631
11	28-Ni-58	2825
12	40-Zr-91	4028
13	68-Er-166	6837
14	68-Er-167	6840
15	82-Pb-207	8234
16	92-U-235	9228
17	92-U-238	9237

2.2. Reactor Code WIMSD-5B

WIMS consisting of a lattice transport code and the associated library WILLIE is used to unravel various thermal reactor problems. The unique WIMSD structure is used with 14 fast group between 10 MeV and 9.11 keV; 13 resonance group between 9.118 keV and 4eV; and 42 thermal groups from 4 eV and 0 eV [14]. Rejoinder of U-235 and U-238 is taken to compute the integral parameters of benchmark lattices by using WIMSD-5B code.

2.3. Benchmark Lattices

The H_2O- moderated uranium lattices TRX-1 and TRX-2 [15] and H_2O-moderated uranium oxide critical lattices BAPL-UO_2-1, BAPL-UO_2-2 and BAPL-UO_2-3 is used for benchmarking of several integral parameter. BAPL-1, BAPL-2 and BAPL-3 used uranium oxide fuel enriched 1.311wt%; TRX-1, TRX-2 used uranium metal fuel in U-235 enriched to 1.305wt%. These five lattices are called benchmark lattice. The material and dimensional properties of TRX benchmark lattices are listed in Table-1 [16] and properties of BAPL lattices listed in Table-2 [17]. The interaction of U-235 and U-238 nuclei at 300K is used to compute the integral parameter of the benchmark lattices using the reactor code WIMSD-5B. The WIMSD-5B is also used to determine neutron cross-section in thermal as well as epithermal range of U-235 and U-238 isotopes for each benchmark lattices.

2.4. Calculation Techniques

The exactness of the processed Group-wise Evaluated Nuclear Data File (GENDF) is analyzed to demonstrate the worth of the previously evaluated data. The chain of NJOY99.0 modules [18], which have been used to generate the 69-group cross section library, is represented by flow chart in Fig.-1. The data strips are routed using NJOY99.0, which can rich the new quality feature of the database. The isotopes listed in Table-2 are concern to the TRIGA Mark-II at AERE, Dhaka, Bangladesh. These elements have been processed in RECONR- BROADR- UNRESR- THERMR-GROUPR- WIMSR cycle by Pentium-IV PC in DOS command mode [19]. Using the WILLIE and WIMSD-5B code 69-group cross-section library is generated from the processed isotope of JEFF-3.1.2. Fission cross-section, absorption cross-section, captured cross-section of U-235 and U-238 are computed for TRX-1, TRX-2, BAPL-UO_2-1, BAPL-UO_2-2 and BAPL-UO_2-3 lattices through the generated 69-group cross-section libraries of JEFF-3.1.2 by using WIMSD-5B. The integral parameters $\rho^{28}, \delta^{25}, \delta^{28}$ and C^* of TRX and BAPL lattices are represented in equations 2 to 5 [20]. The effective multiplication factor is noted by equation 1. The integral parameter of TRX and BAPL lattices of thermal reactor are calculated using this equation. The evaluated values of the integral parameters have been compared with the experimental values by cross-section evaluated working group (CSEWG) [21]. The overall analysis is performed at the department of Physics, Jahangirnagar University, Bangladesh.

k_{eff}= (neutron production from fission in one generation) / (neutron absorption in the preceding generation + neutron leakage in the preceding generation) (1)

ρ^{28} = Ratio of epithermal to thermal neutron captures cross-section of ^{238}U

$$= (\Sigma_c)^{38}_{epth}/(\Sigma_c)^{38}_{th} = (\Sigma_a - \Sigma_f)^{38}_{epth}/(\Sigma_a - \Sigma_f)^{38}_{th} \quad (2)$$

δ^{25} = Ratio of epithermal to thermal neutron fission cross section of ^{235}U

$$= (\Sigma_f)^{35}_{epth}/(\Sigma_f)^{35}_{th} \quad (3)$$

δ^{28} = Ratio of ^{238}U fission to ^{235}U fission

$$= (\Sigma_f^t)^{38} \ / \ (\Sigma_f^t)^{35} \quad (4)$$

C^*= Ratio of ^{238}U captures to ^{235}U fissions

$$= (\Sigma_c^t)^{38}/(\Sigma_f^t)^{35} = (\Sigma_a^t - \Sigma_f^t)^{38}/(\Sigma_f^t)^{35} \quad (5)$$

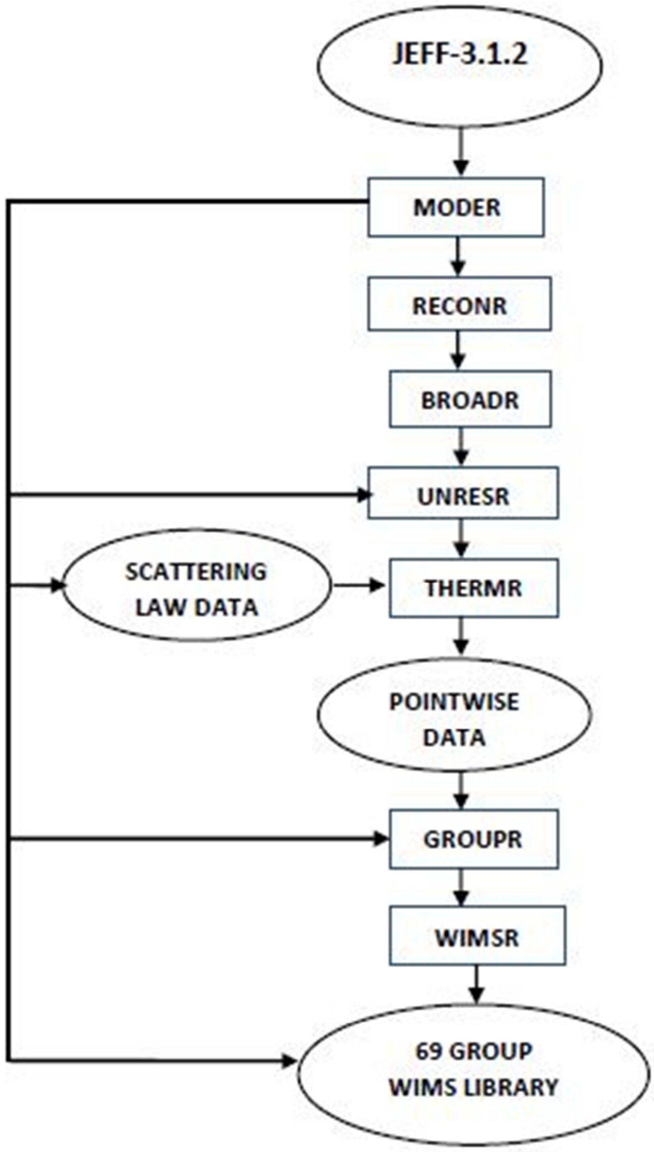

Figure 1. *Flow chart of nuclear data processing code NJOY99.0.*

3. Results

The NJOY output of the two isotopes U-235 and U-238 are compared in Tables 4 to 5 within the thermal range. The calculated epithermal absorption cross section, thermal absorption cross section, total absorption cross section, epithermal fission cross section, thermal fission cross section, epithermal capture cross section and total fission cross section for neutrons on U-235 and U-238 of benchmark lattices TRX and BAPL are plotted in Figs. 2 to 6. The values of effective multiplication factor k_{eff} for TRX and BAPL lattices are calculated and listed in Tables 6 & 7. The calculated values of other integral parameters ρ^{28}, δ^{25}, δ^{28} and C^* for TRX and BAPL lattices are summarized in Tables 8 & 9 and compared with experimental values by CSEWG.

In the horizontal axes of graphs from Figs. 2 to Fig. 6, the symbols a, b, c, d, e, f and g represent epithermal absorption cross-section, thermal absorption cross-section, total absorption cross section, epithermal fission cross-section, thermal fission cross-section, epithermal capture cross-section and total fission cross-section, respectively.

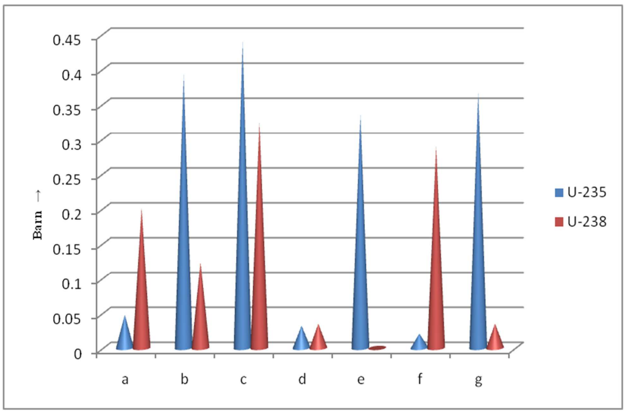

Figure 2. *Cross-section of U-235 & U-238 for TRX-1 lattice.*

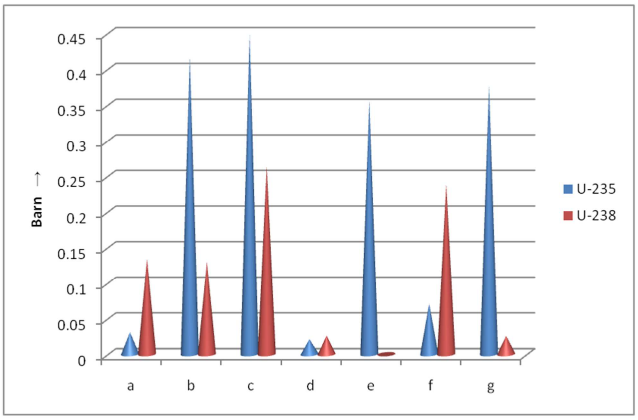

Figure 3. *Cross-section of U-235 & U-238 for TRX-2 lattice.*

Table 4. *Absorption & Fission cross-section from NJOY out of U-235 & U-238 in JEFF-3.1.2.*

Energy group no.	Thermal energy (eV)	Absorption cross-section in barn		Fission cross-section in barn	
		U-235	U-238	U-235	U-238
28	4.000	43.533	0.64026	26.933	3.1331E-06
29	3.300	30.120	0.51490	23.401	3.1620E-06
30	2.600	14.579	0.46582	11.295	3.1373 E-06
31	2.100	26.410	0.45476	15.919	3.6127E-06
32	1.500	24.804	0.46632	19.098	3.9454E-06
33	1.300	73.260	0.47913	53.315	4.1586E-06
34	1.150	136.64	0.48902	106.58	4.2940E-06
35	1.123	113.36	0.49210	105.38	4.3363E-06
36	1.097	116.77	0.49512	96.053	4.3779E-06
37	1.071	101.49	0.49814	85.271	4.4194E-06
38	1.045	89.239	0.50110	76.214	4.4601E-06
39	1.020	80.565	0.50397	69.443	4.4995E-06
40	0.996	74.315	0.50766	64.841	4.5463E-06
41	0.972	69.948	0.51165	61.456	4.5954E-06
42	0.950	66.089	0.51704	58.452	4.6618E-06
43	0.910	62.608	0.52605	55.765	4.7719E-06
44	0.850	61.428	0.54052	54.969	4.9416E-06
45	0.780	65.862	0.57150	58.980	5.2942E-06
46	0.625	81.247	0.62384	72.252	5.8672E-06
47	0.500	111.78	0.68368	97.674	6.5038E-06
48	0.400	156.80	0.73736	133.49	7.0642E-06
49	0.350	198.13	0.77403	164.84	7.4432E-06
50	0.320	225.07	0.80114	184.35	7.7218E-06
51	0.300	238.33	0.82589	193.20	7.7752E-06
52	0.280	236.98	0.68145	191.04	8.3380E-06
53	0.250	222.38	0.91109	180.28	8.8425-06
54	0.220	212.86	0.98454	175.58	9.8585E-06
55	0.180	224.93	1.0985	189.27	1.0707E-05
56	0.140	264.47	1.2605	225.68	1.2358E-05
57	0.100	315.69	1.4407	271.11	1.4159E-05
58	0.080	359.60	1.5890	309.48	1.5637E-05
59	0.067	398.60	1.7189	343.13	1.6930E-05
60	0.058	437.03	1.8462	376.44	1.8196E-05
61	0.050	482.50	1.9984	415.38	1.9709E-05
62	0.042	536.91	2.1822	461.67	2.1536E-05
63	0.035	592.86	2.3730	508.92	2.3430E-05
64	0.030	652.42	2.5769	558.88	2.5453E-05
65	0.025	730.59	2.8459	624.01	2.8121E-05
66	0.020	839.86	3.2255	714.61	3.1885E-05
67	0.015	1007.7	3.8121	853.53	3.7699E-05
68	0.010	1318.1	4.9044	111.15	4.8481E-05
69	0.005	221.59	8.0839	1863.2	8.0005E-05

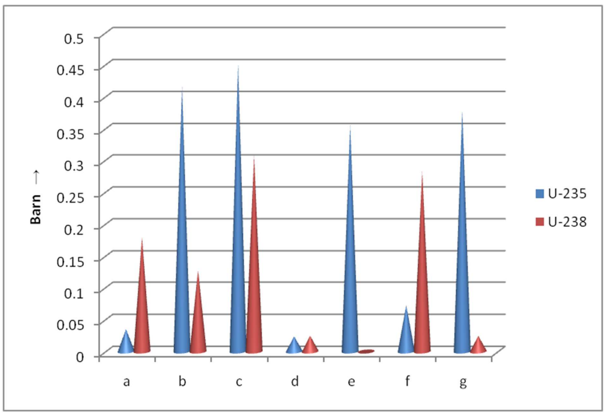

Figure 4. *Cross-section of U-235 & U-238 for BAPL-.1.*

Table 5. *Transport & total scattering cross-section from NJOY out of U-235 & U-238 in JEFF-3.1.2.*

Energy group no.	Thermal energy (eV)	Transport cross-section in barns		Total scattering cross-section barn	
		U-235	U-238	U-235	U-238
28	4.000	55.107	9.1126	10.970	8.0167
29	3.300	41.786	9.2503	11.066	8.2884
30	2.600	26.642	9.3123	11.319	8.3011
31	2.100	38.782	9.4295	11.839	8.5922
32	1.500	37.621	9.5217	11.405	8.0688
33	1.300	86.391	9.5762	11.381	7.8842
34	1.150	149.68	9.6143	5.6979	3.9807
35	1.123	144.30	9.6237	5.5339	3.9413
36	1.097	129.65	9.6329	5.5622	3.9685
37	1.071	114.36	9.6422	5.6450	4.0163
38	1.045	102.13	9.6512	5.5540	3.9455
39	1.020	93.480	9.6541	5.4849	3.8996
40	0.996	87.236	9.6428	5.5140	3.8899
41	0.972	82.877	9.6317	5.2317	3.6973
42	0.950	79.058	9.6361	7.7636	5.4683
43	0.910	75.664	9.6589	9.4923	6.6504
44	0.850	74.599	9.6911	10.201	7.1011
45	0.780	79.212	9.7591	12.097	8.3351
46	0.625	94.853	9.9317	12.212	8.2745
47	0.500	125.65	9.8992	12.283	8.1709
48	0.400	170.83	9.9640	11.161	7.3494
49	0.350	212.21	10.006	9.6537	6.3468
50	0.320	239.14	10.037	8.0837	5.3218
51	0.300	25.238	10.0065	8.2297	5.4283
52	0.280	25.102	10.103	10.095	6.6648
53	0.250	236.48	10.179	10.371	6.8318
54	0.220	227.06	10.276	11.596	7.6010
55	0.180	239.27	10.381	12.016	7.7886
56	0.140	278.95	10.538	12.487	8.0119
57	0.100	330.28	10.710	11.110	7.0761
58	0.080	374.23	10.843	9.8740	6.2623
59	0.067	413.25	10.958	8.5780	5.4284
60	0.058	451.67	11.056	8.3389	5.2650
61	0.050	497.09	11.152	8.6543	4.4503
62	0.042	551.57	11.370	8.5157	5.3526
63	0.035	607.68	11.643	7.4212	4.6604
64	0.030	667.32	11.886	7.8852	4.9441
65	0.025	745.52	12.159	8.3875	5.2502
66	0.020	854.78	12.521	8.9734	5.6058
67	0.015	1022.6	13.056	9.6824	6.0312
68	0.010	1333.1	14.183	10.749	6.6879
69	0.005	2231.1	17.547	12.002	7.4511

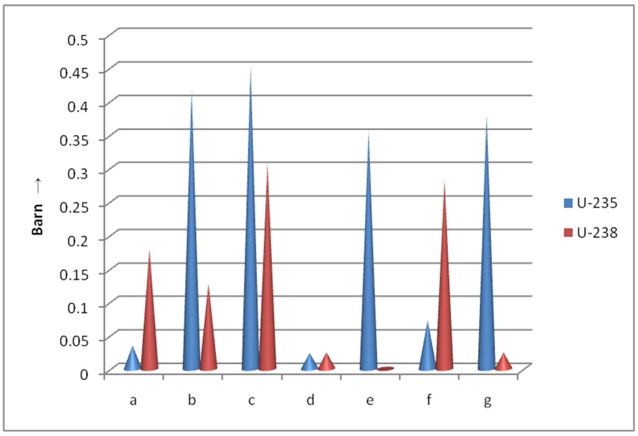

Figure 5. *Cross-section of U-235 & U-238 for BAPL-2 lattice.*

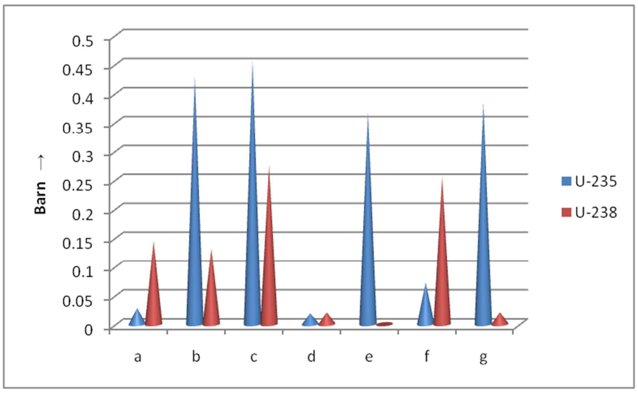

Figure 6. *Cross-section of U-235 & U-238 for BAPL-3 lattice.*

Table 6. *k_{eff} comparison of TRX benchmark lattices.*

Lattices	JEFF-3.1.2	Experiment (CSEWG, 1986)	Percentage of error
TRX-1	0.9853975	1.0000	1.46
TRX-2	0.9826511	1.0000	1.7

Table 7. *k_{eff} comparison of BAPL benchmark lattices.*

Lattices	JEFF-3.1.2	Experiment (CSEWG, 1986)	Percentage of error
BAPL-1	0.9828444	1.0000	1.7
BAPL-2	0.9849318	1.0000	1.5
BAPL-3	0.987897	1.0000	1.21

Table 8. *Integral parameter comparison of TRX benchmark lattices.*

Lattices	Integral Parameter	JEFF-3.1.2	Experiment (CSEWG, 1986)	Percentage of error
TRX-1	ρ^{28}	1.3466	1.3200	2
	δ^{25}	0.0958	0.0987	2.9
	δ^{28}	0.0949	0.0946	0.31
	C*	0.78848	0.7970	1.06
TRX-2	ρ^{28}	0.832	0.8370	0.59
	δ^{25}	0.05868	0.0614	4.4
	δ^{28}	0.0685	0.0693	1.1
	C*	0.6322	0.6470	2.2

Table 9. *Integral parameter comparison of BAPL benchmark lattices.*

Lattices	Integral parameters	JEFF-3.1.2	Experiment (CSEWG, 1986)	Percentage of error
BAPL-1	ρ^{28}	1.4767	1.3900	6
	δ^{25}	0.0811	0.08400	3.4
	δ^{28}	0.7540	0.0780	3.3
	C*	0.8250	……	….
BAPL-2	ρ^{28}	1.2101	1.1200	8.5
	δ^{25}	0.0661	0.0680	2.7
	δ^{28}	0.0649	0.0700	7.2
	C*	0.7460	….	….
BAPL-3	ρ^{28}	0.9440	0.9606	1.7
	δ^{25}	0.0507	0.0520	2.3
	δ^{28}	0.0533	0.0570	6.4
	C*	0.6616	…	….

4. Discussion

Very recent, only the integral parameters of TRX and BAPL lattices of JEFF-3.1.1 have been compared and the values of k_{eff} are very close to the experiment [22]. In the present work NJOY output for the two isotopes U-235 and

U-238 the group constants are consistent with each other. For each TRX and BAPL lattices the total absorption cross-section of U-235 is larger than that of U-238, epi-thermal fission cross-section of U-238 is marginal, but the thermal fission is completely absent. The captured cross-sections of U-235 are very low but thermal fission cross-section of U-235 is remarkably high. Captured cross-section of U-235 is very much lower than that of U-238. Moreover, the character cross-sections for each lattice are identical. From Tables 6 & Table 7, the calculated values of effective multiplication factor k_{eff} are very close to the experimental values, the maximum deviation is 1.7% for TRX-2 and BAPL-1 lattices. From Tables 8 & Table 9 it can be seen that the uncertainties of calculated values of the integral parameters of ρ^{28}, δ^{25}, δ^{28} and C^* for TRX-1 & TRX-2 lattices do not deviate by more than 5% from the experimental values. Only the values of ρ^{28} in BAPL-1 and BAPL-2 lattices, values of δ^{28} in BAPL-2 and BAPL-3 show more than 5% inaccuracy but rest of the values of the integral parameters are very close to the benchmark values.

5. Conclusion

This theoretical study compares with the benchmark integral parameters of thermal reactor metallic uranium (TRX) and uranium oxide (BAPL) lattices with the nuclear data library JEFF-3.1.2 by using NJOY'99.0 and WIMSD-5B codes. The results from these computations are compared with experimental values by CSEWG, and it is found that there are no significant differences between calculated and experimental values. The cross-sections for epithermal and thermal neutrons at each lattice are practically identical. The integral parameters are almost equal to the experimental values, except for few values.

In the experimental result by CSEWG, the values of C* are present in TRX lattice but absent for BAPL lattice, therefore judgment for value of C* for BAPL benchmark lattice are not achievable. To conclude, this analysis provides a clear confirmation of the nuclear data library JEFF-3.1.2 for the neutronic calculation of TRIGA mark-II research reactor. Therefore JEFF-3.1.2 is completely reliable for safety analysis of the TRIGA reactor at AERE, Dhaka, Bangladesh.

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