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Validation of Data Files of JENDL-4.0u for Neutronic Calculation of TRIGA Mark-II Reactor through the Investigation of Integral Parameter of Benchmark Lattices TRX and BAPL

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Abstract: *Evaluated Nuclear Data Files* **(***ENDF) can be used for neutronic analysis of thermal reactor such as research reactor. JENDL-4.0u (Japanese Evaluated Nuclear Data Library) is released by JAEA to read in ENDF-6 format. The aim of this analysis is to validate the nuclear data files of JENDL-4.0u for neutronics calculation of 3-MW TRIGA MARK-II research reactor through the analysis of integral parameters of TRX (Thermal Reactor-one region lattice) and BAPL (Bettis Atomic Power Laboratory-one region lattice) benchmark lattices of thermal reactor. In this paper the basic evaluated nuclear data files of JENDL-4.0u are used to generate 69-group cross section library for lattice transport code WIMSD-5B by nuclear data processing code NJOY99.0. From the generated 69 group cross-section library, the integral parameters (k_{eff}, $\rho^{28}, \delta^{25}, \delta^{28}$ and C*) of benchmark lattices TRX-1, TRX-2, BAPL-UO_2-1, BAPL-UO_2-2 and BAPL-UO_2-3 were calculated by WIMSD-5B code. The epithermal and thermal cross-section of U-235 and U-238 are calculated for each lattice. The calculated integral parameters are compared to the experimental values by Cross-Section Evaluated Working Group (CSEWG), USA. From the comparison it is found that the calculated values are very close to the experimental result obtained by CSEWG. This theoretical analysis provides the corroboration of evaluated data files of JENDL-4.0u for neutronics analysis of TRIGA MARK-II research reactor.*

Keywords: *BAPL, JENDL-4.0u, NJOY99.0, TRIGA MARK-II, TRX and WIMSD-5B*

I. Introduction

TRIGA is a high class successful reactor supplied by General Atomics, USA [1]. A 3 MW TRIGA MARK-II research reactor is commissioned at AERE, Dhaka, Bangladesh and designed for training, research and isotope production [2]. TRIGA MARK-II research reactors are light water reactors (L.W.R). ENDF are used for neutronic calculation of LWR configuration. Therefore, a continuous and careful verification is required for the usable data files of evaluated nuclear data library. The Japanese Evaluated Nuclear Data Library (JENDL) is built by the Nuclear Data Center of Japan Atomic Energy Agency (JAEA) with the support of Japanese Nuclear Data Committee. The first version of JENDL was released in 1977 as JENDL-1[3]. JENDL-4.0 was released in 2010 which contains neutron-induced reaction data for 406 nuclides [4], in the neutron energy range from 10-5 eV to 20 MeV. JENDL-4.0 is last updated at January, 2016 named JENDL-4.0u. JENDL-4.0u is the field of our present research. The WIMS codes [5] solve a wide variety of thermal reactor problem. This WIMS program consists of a lattice transport code and the associated library. The important use of this code is to generate 69 group cross-section libraries which provide the problem dependent solution to neutronic parameter of thermal reactors. The absorption cross-section, fission cross-section and total scattering cross-section in the thermal energy range for U-235 and U-238 isotopes are evaluated. In this study the 69 group cross-section library for the reactor code WIMSD-5B [6] is to be generated by the nuclear data processing code NJOY99.0 [7] using the data files of JENDL-4.0u. The calculated integral parameters were compared with the experimental values by cross-section evaluated working group (CSEWG) [8].

II. Tool And Technology

Four identified task associated with the nuclear data are; nuclear data production, evaluation, processing and its application. The First two are done by famous nuclear library of the world and last two are done by us by using computer code: NJOY99.0, WIMSD-5B; evaluated nuclear data library JENDL-4.0u; benchmark lattices TRX and BAPL.

2.1 NJOY99.0 Code System

The nuclear data processing system NJOY is a modular computer code used for converting nuclear data in ENDF-6 format into libraries as ENDF/B-VI in USA [9], JEFF in Europe [10], JENDL in Japan, CENDL in

China [11], BROND in Russia[12]. One of the common applications of NJOY99.0 is to generate 69 group cross-section libraries from basic nuclear data library. The 69 group cross-section library for WIMSD-5B code from basic data files of JENDL-4.0u is generated by NJOY99.0.

2.2 WIMSD-5B Code System

The Winfrith Improved Multigroup Scheme WIMS consisting of a lattice transport code and the associated library was widely used for solve reactor problem of various thermal reactors. The original WIMSD structure is used with 14 fast group between 10MeV and 9.11KeV; 13 resonance group between 9.118KeV and 4eV; and 42 thermal groups from 4eV and 0eV [13]. Reaction of U-235 and U-238 has been taken to calculate the integral parameters of TRX and BAPL lattices by using WIMSD-5B code. The cross-section data sets in thermal region was processed using WIMS library utility code WILLIE for U-235 and U-238 isotopes and was compared as well.

2.3 Benchmark Lattices

Two types of benchmark lattices H_2O- moderated uranium lattices TRX-1 & TRX-2 [14] as well as H_2O-moderated uranium oxide critical lattices BAPL-UO_2-1, BAPL-UO_2-2 and BAPL-UO_2-3 were used. The material and dimensional properties of benchmark lattices are listed in Table-1[15] and Table-2[16] respectively. In this study the isotopes in the data file JENDL-4.0u are processed to analyze the critical benchmark lattices TRX and BAPL. The reactions of U-235 and U-238 isotopes at 300K are used to calculate the integral parameter of TRX and BAPL lattices using the lattice code WIMSD-5B. The absorption cross-section, fission cross-section, transport cross-section and total scattering cross-section of U-235 and U-238 isotopes in the thermal and epithermal range for each TRX and BAPL lattices are determined using WIMSD-5B.

Table 1: Material and dimensional properties of TRX

Region	Outer radius in cm	Isotope	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 of 1.8060cm & 2.1740 cm respectively in triangular arrays

Table -2: Material and dimensional properties of BAPL

Region	Outer radius in cm	Isotope	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 of 1.5578, 1.6523 and 1.8057 cm respectively

III. Calculation Technique

Overall calculations of this study are performed through the generation of GENDF data and Benchmark calculation.

3.1 Generation of GENDF Data

The chain of NJOY99.0 modules [17] used to generate the 69-group cross section library is shown schematically in Fig-1. The accuracy of the processed Group-wise Evaluated Nuclear Data File (GENDF) was tested to verify the quality of the basic evaluated data. The data tapes were processed using NJOY99.0, which can handle the new feature of the database. The isotopes presented in the Table-3 are associated with the TRIGA Mark-II have been processed in the Pentium-IV PC at the department of physics of Jahangirnagar University, Bangladesh, in RECONR-BROADR-UNRESR-THERMR-GROUPR-WIMSR sequence [18].

3.2 Benchmark Calculation

The integral parameters $k_{eff,}$ $\rho^{28}, \delta^{25}, \delta^{28}$ and C* of TRX and BAPL lattices are calculated from generated 69-group cross-section libraries of JENDL-4.0u by using WIMSD-5B. The uranium oxide fuel enrichment in BAPL-1, BAPL-2, BAPL-3 are 1.311wt%; uranium metal fuel enrichment in TRX-1, TRX-2 are1.305wt%. The integral parameters were defined [19] below.

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

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

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

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

δ^{28} = Ratio of ^{238}U fission to ^{235}U fission = $(\Sigma_f^t)^{38} / (\Sigma^t_f)^{35}$
C^* = Ratio of ^{238}U captures to ^{235}U fissions = $(\Sigma_c^t)^{38} / (\Sigma^t_f)^{35}$ = $(\Sigma^t_a - \Sigma^t_f)^{38} / (\Sigma^t_f)^{35}$

The percentages of error are calculated by the difference between calculated value and experimental value divided by the experimental value times 100.

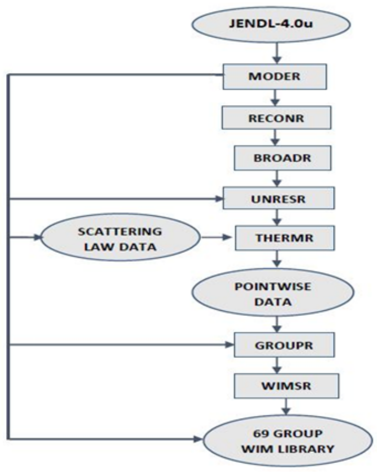

Fig.1. Flow chart of nuclear data processing code NJOY99.0

Table 3: TRIGA Mark-II associated isotope with the respective material ID

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

IV. Results and Discussions

The WIMS output of U-235 and U-238 isotopes in the thermal range are compared in Table 4 & 6. It can be observed that group constants are well agreed with each other. The calculated neutron cross-sections of U-235 and U-238 for each benchmark lattices TRX and BAPL are plotted in Fig. 2 to Fig.6 (Ep.A= epithermal absorption, Th.A= thermal absorption, T. A= total absorption cross section, Ep.F= epithermal fission, Th.F= thermal fission, T.Cp= epithermal capture and T.F=total fission). The thermal fission cross-section of U-238 for each lattices are almost absent. The captured cross-sections of U-235 are very low but fission cross-sections are significantly high. Fission and captured cross-section for epithermal and thermal neutrons are identical for each benchmark lattices. The calculated values of effective multiplication factor k_{eff} and other integral parameters for TRX and BAPL lattices are summarized and compared with experimental values by CSEWG are listed in Table 5 and Table 7 respectively.

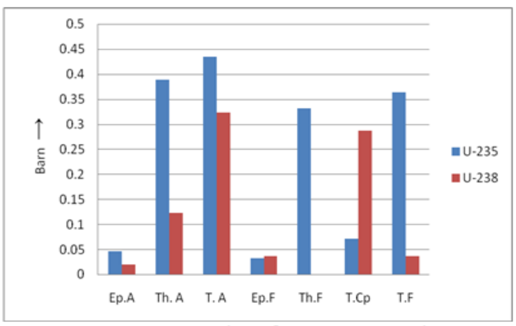

Fig 2: Neutron cross-section of U-235 & U-238 in TRX-1

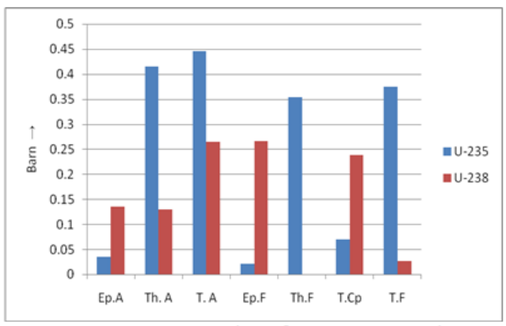

Fig 3: Neutron cross-section of U-235 & U-238 in TRX-2

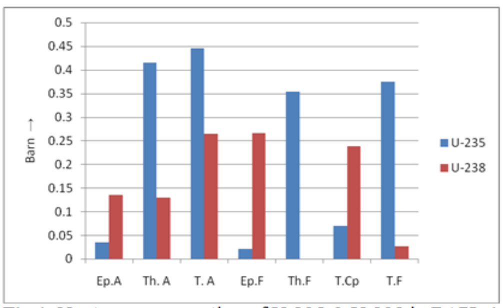

Fig 4: Neutron cross-section of U-235 & U-238 in BAPL-1

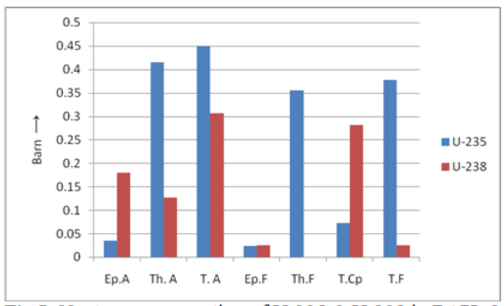

Fig 5: Neutron cross-section of U-235 & U-238 in BAPL-2

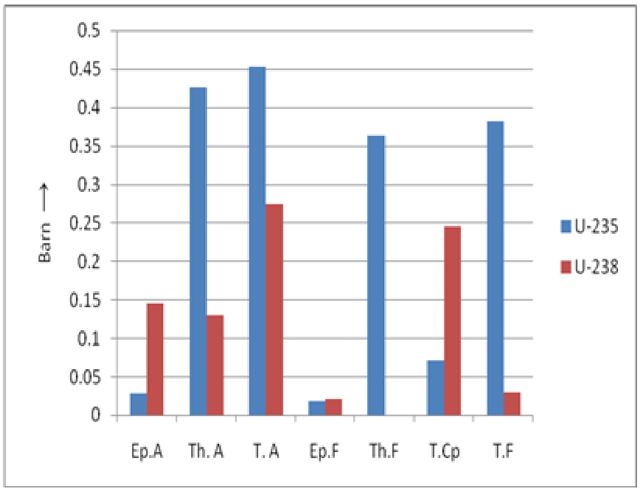

Fig 6: Neutron cross-section of U-235 & U-238 in BAPL-3

Table-4: Neutron cross-section for U-235 & U-238 isotope in JENDL-4.0u

Thermal energy (ev)	Absorption cross- section in barns		Fission cross- section in barns	
	U-235	U-238	U-235	U-238
4.000	4.3533E+01	6.4029E-01	2.6933E+01	2.2622E-06
3.300	3.0120E+01	5.1563E-01	2.3401E+01	2.2035E-06
2.600	1.4579E+01	4.6637E-01	1.1295E+01	2.2539E-06
2.100	2.6410E+01	4.5543E-01	1.5919E+01	2.4060E-06
1.500	2.4804E+01	4.6703E-01	1.9098E+01	2.5951E-06
1.300	7.3260E+01	4.8000E-01	5.3315E+01	2.7209E-06
1.150	1.3664E+02	4.8970E-01	1.0658E+02	2.8021E-06
1.123	1.3136E+02	4.9278E-01	1.0538E+02	2.8275E-06
1.097	1.1677E+02	4.9579E-01	9.6053E+01	2.8324E-06
1.071	1.0149E+02	4.9881E-01	8.5271E+01	2.8774E-06
1.045	8.9239E+01	5.0177E-01	7.6214E+01	2.9019E-06
1.020	8.0565E+01	5.0464E-01	6.9643E+01	2.9255E-06
0.996	7.4315E+01	5.0832E-01	6.4841E+01	2.9539E-06
0.972	6.9948E+01	5.1231E-01	6.1456E+01	2.9838E-06
0.950	6.6089E+01	5.1769E-01	5.8452E+01	3.0242E-06
0.910	6.2008E+01	5.2670E-01	5.5765E+01	3.0913E-06
0.850	6.1428E+01	5.4115E-01	5.4969E+01	3.1954E-06
0.780	6.5862E+01	5.7209E-01	5.8980E+01	3.4126E-06
0.625	8.1247E+01	6.2438E-01	7.2252E+01	3.7678E-06
0.500	1.1178E+02	6.8417E-01	9.7674E+01	4.1644E-06
0.400	1.5680E+02	7.3781E-01	1.3349E+02	4.5149E-06
0.350	1.9813E+02	7.4440E-01	1.6484E+02	4.7524E-06
0.320	2.2507E+02	8.0154E-01	1.8435E+02	4.9272E-06
0.300	2.3833E+02	8.2727E-01	1.9320E+02	5.0863E-06
0.280	2.3698E+02	8.6180E-01	1.9104E+02	5.3143E-06
0.250	2.2238E+02	9.1141E-01	1.8028E+02	5.6316E-06
0.220	2.1286E+02	9.8482E-01	1.7558E+02	6.0995E-06
0.180	2.2493E+02	1.0691E+00	1.8927E+02	6.8061E-06
0.140	2.6447E+02	1.2607E+00	2.2568E+02	7.8481E-06
0.100	3.1569E+02	1.4408E+00	2.7111E+02	8.9854E-06
0.080	3.5960E+02	1.5890E+00	3.0948E+02	9.9194E-06
0.067	3.9860E+02	1.7189E+00	3.4331E+02	1.0737E-05
0.058	4.3703E+02	1.8461E+00	3.7644E+02	1.1538E-05
0.050	4.8250E+02	1.9983E+00	4.1538E+02	1.2495E-05
0.042	5.3691E+02	2.1821E+00	4.6167E+02	1.3650E-05
0.035	5.9286E+02	2.3728E+00	5.0892E+02	1.4849E-05
0.030	6.5242E+02	2.5767E+00	5.5888E+02	1.6129E-05
0.025	7.3059E+02	2.8456E+00	6.2401E+02	1.7818E-05
0.020	8.3986E+02	3.2252E+00	7.1461E+02	2.0200E-05
0.015	1.0077E+03	3.8117E+00	8.5353E+02	2.3881E-05
0.010	1.3181E+03	4.8998E+00	1.1115E+03	3.0707E-05
0.005	2.2159E+03	8.0828E+00	1.8632E+03	5.0668E-05

Table-5: Integral parameter comparison for TRX benchmark lattices calculated by WIMSD-5B

Lattices	Integral parameter	JENDL-4.0u	Experiment (CSEWG,1986)	Percentage of error
TRX1	k_{eff}	0.97830	1.0000	2.1
	ρ^{28}	1.3517	1.3200	2.4
	δ^{25}	0.09585	0.0987	2.8
	δ^{28}	0.09958	0.0946	5.2
	C*	0.79023	0.7970	0.8
TRX2	k_{eff}	0.98149	1.0000	1.85
	ρ^{28}	0.8353	0.8370	1.32
	δ^{25}	0.05867	0.0614	4.4
	δ^{28}	0.0708	0.0693	2.16
	C*	0.63327	0.6470	2.1

Table-6: WIMS output for U-235 & U-238 isotope in JENDL-4.0u

Thermal energy (ev)	Transport cross-section in barns		Total scattering cross-section in barns	
	U-235	U-238	U-235	U-238
4.000	5.5107E+01	8.9832E+00	1.0970E+01	7.8943E+00
3.300	4.1788E+01	9.1191E+00	1.1066E+01	8.1583E+00
2.600	2.6642E+01	9.1806E+00	1.1319E+01	8.1738E+00
2.100	3.8782E+01	9.2967E+00	1.1839E+01	8.4646E+00
1.500	3.7621E+00	9.3885E+00	1.1405E+01	7.9495E+00
1.300	8.6391E+01	9.4427E+00	1.1381E+01	7.7679E+00
1.150	1.4968E+02	9.4804E+00	5.6997E+00	3.9220E+00
1.123	1.4430E+02	9.4898E+00	3.8436E+00	3.8832E+00
1.097	1.2965E+02	9.4990E+00	5.5622E+00	3.9101E+00
1.071	1.1436E+02	9.5080E+00	5.6450E+00	3.9571E+00
1.045	1.0213E+02	9.5172E+00	5.5540E+00	3.8874E+00
1.020	9.3480E+01	9.5200E+00	5.4849E+00	3.8422E+00
0.996	8.7236E+01	9.5090E+00	5.5140E+00	3.8326E+00
0.972	8.2877E+01	9.4981E+00	5.2317E+00	3.6429E+00
0.950	7.9058E+01	9.5026E+00	7.7637E+00	5.3879E+00
0.910	7.5664E+01	9.5252E+00	9.4929E+00	6.5526E+00
0.850	7.4599E+01	9.5572E+00	1.0201E+01	6.9967E+00
0.780	7.9212E+01	9.6248E+00	1.2097E+01	8.2127E+00
0.625	9.4853E+01	9.6972E+00	1.2212E+01	8.1531E+00
0.500	1.2565E+02	9.7646E+00	1.2283E+01	8.0512E+00
0.400	1.7083E+02	9.8294E+00	1.1161E+01	7.2417E+00
0.350	2.1221E+02	9.8713E+00	9.6537E+00	6.2539E+00
0.320	2.3914E+02	9.9026E+00	8.0837E+00	5.2439E+00
0.300	2.5238E+02	9.9301E+00	8.2297E+00	5.3489E+00
0.280	2.5002E+02	9.9686E+00	1.0095E+01	6.5672E+00
0.250	2.3648E+02	1.0043E+01	1.0371E+01	6.7312E+00
0.220	2.2706E+02	1.0140E+01	1.1596E+01	7.4898E+00
0.180	2.3927E+02	1.0246E+01	1.2016E+01	7.6747E+00
0.140	2.7895E+02	1.0402E+01	1.2487E+01	7.8948E+00
0.100	3.3028E+02	1.0575E+01	1.1110E+01	6.9727E+00
0.080	3.7423E+02	1.0708E+01	9.8740E+00	6.1708E+00
0.067	4.1325E+02	1.0823E+01	8.5789E+00	5.3491E+00
0.058	4.5167E+02	1.0921E+01	8.3393E+00	5.1881E+00
0.050	4.9709E+02	1.1018E+01	8.6543E+00	5.3707E+00
0.042	5.5157E+02	1.1235E+01	8.5157E+00	5.2747E+00
0.035	6.0768E+02	1.1507E+01	7.4217E+00	4.5923E+00
0.030	6.6732E+02	1.1750E+01	7.8852E+00	4.8719E+00
0.025	7.4552E+02	1.2022E+01	8.3875E+00	5.1736E+00
0.020	8.5478E+02	1.2384E+01	8.9734E+00	5.5240E+00
0.015	1.0226E+03	1.2920E+01	9.6824E+00	5.9431E+00
0.010	1.3331E+03	1.4047E+01	1.0749E+01	6.5902E+00
0.005	2.2311E+03	1.7408E+01	1.2002E+01	7.3423E+00

Table-7: Integral parameter comparison for BAPL benchmark lattices calculated by WIMSD-5B

Lattices	Integral parameters	JENDL-4.0u	Experiment (CSEWG,1986)	Percentage of error
BAPL 1	k_{eff}	0.97743	1.00000	2.2
	ρ^{28}	1.48610	1.3900	6.4
	δ^{25}	0.08143	0.084	3
	δ^{28}	0.07848	0.0780	0.6
	C*	0.82811	……	…..
BAPL 2	k_{eff}	0.98481	1.0000	1.5
	ρ^{28}	1.22360	1.1200	9
	δ^{25}	0.06634	0.0680	2.4
	δ^{28}	0.06721	0.0700	3.9
	C*	0.74501	….	….
BAPL 3	k_{eff}	0.98277	1.0000	1.72
	ρ^{28}	0.95002	0.96060	1.1
	δ^{25}	0.05090	0.0520	2.1
	δ^{28}	0.05487	0.0570	3.7
	C*	0.66349	…	…..

From the comparisons it is found that the values of integral parameters of TRX and BAPL benchmark lattices for JENDL-4.0u are generally well in agreement with the experimental values. The maximum uncertainty are found in k_{eff} is 2.2% for BAPL-1 lattices. Only the values (red marked in Table- 5 & Table-7) of δ^{28} in TRX-1 and ρ^{28} in BAPL-1 and BAPL-2 lattices provide more than 5% error. Moreover, all other integral parameters' values are very close to the experimental results.

V. Conclusion

This analysis deals with neutron cross-section for thermal neutron for U-238 & U-235 as well as the values of integral parameters of TRX & BAPL benchmark lattices for the last updated data files of JENDL-4.0u by using NJOY99.0 & WIMSD-5B code. From the comparison of the calculated result with the experimental result by CSEWG it is found that some values of the integral parameters are not close to the experimental results but there are no significant deviations for the other values of the calculated result. The values of C* are absent for BAPL lattice in the experimental result by CSEWG, therefore comparison for value of C* for BAPL benchmark lattice are not possible. Moreover, almost all other values of calculated results almost same to the experimental values. Considering overall observations it could be concluded that the updated library JENDL-4.0u is completely reliable for the neutronic calculation of the TRIGA reactor at AERE, Dhaka, Bangladesh. Therefore this present study gives the validation of the JENDL-4.0u data library for the study of TRIGA Mark -II research reactor.

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