A STUDY ON JUSTIFICATION OF SRAC 2006 CODE SYSTEM BY BENCHMARKING INTEGRAL PARAMETERS OF TRX AND BAPL CRITICAL LATTICES OF THERMAL REACTORS FOR TRIGA NEUTRONICS CALCULATION

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ABSTRACT

The aim of this study is to justify Standard Reactor Analysis Code System (SRAC2006) based on the evaluated nuclear data libraries such as Evaluated Nuclear Data File (ENDF/B-VI.8) and Joint European Fission-Fusion (JEFF-3.1) for core calculations of 3 MW TRIGA Mark-II Research Reactor at Atomic Energy Research Establishment (AERE), Bangladesh. The study was performed through the analysis of integral parameters of Thermal Reactor Cross section (TRX) and Bettis Atomic Power Laboratory (BAPL) benchmark lattices of thermal reactors. In integral measurements, the thermal reactor lattices TRX-1, TRX-2, BAPL-UO2-1, BAPL-UO2-2 and BAPL-UO2-3 were treated as experimental benchmarks for justifying the SRAC2006 code system. The integral parameters of the said lattices are calculated using the collision probability transport code SRAC-PIJ of the SRAC2006 code system at room temperature 20 0C based on the above libraries. The calculated integral parameters are compared to the measured values as well as the Monte Carlo Nuclear Particle (MCNP) values based on the Chinese Evaluated Nuclear Data Library (CENDL-3.0). It was found that the calculated results show a good agreement among the said data files as well as the experimental and the MCNP results. To enrich this justification study the group constants (in SRAC format) such as activation cross-section, fission cross-section, nu-fission cross-section, total cross-section, diffusion coefficients, absorption cross-section and scattering cross-section in fast and thermal energy range between the above data files for TRX and BAPL lattices were calculated using the SRAC-PIJ code in fixed source mode. By comparing the group constants it was found that the group constants were well consistent with each other. Therefore, this paper reflects the justification study of the SRAC2006 code system and it will be useful to meet up nuclear data for further core safety calculations of 3 MW TRIGA Mark-II research reactors at AERE, Savar, Dhaka, Bangladesh.

Key Words: SRAC2006, ENDF/B-VI.8, JEFF-3.1, TRX-1, TRX-2, BAPL-UO2-1, BAPL-UO2-2, BAPL-UO2-3, MCNP, SRAC-PIJ, Benchmark and validation.

1.0 INTRODUCTION

The 3 MW TRIGA Mark-II research reactor is most widely used for generation of neutrons due to its safety features, operational flexibility, wide capabilities and ability to utilize Low Enriched Uranium (LEU) fuels efficiently. This research reactor plays an important role in the development of nuclear science and technology. They are used to produce medical and industrial isotopes, for research in physics, biology and materials science, and for scientific education and training. They also occupy an indispensible place in nuclear power program.

For nuclear research and technology development to continue to prosper, research reactors must be safely and reliably operated, efficiently utilized, refurbished when necessary, and provided with adequate non-proliferating fuel cycle services. Number of endeavor had been taken in the past to the development and to the validation of adequate spectrum codes for safety and design calculation of light water reactors. Various accurate neutronics calculations have been performed using specific methods and nuclear data libraries. However, the repeated utilization of the existing code systems for calculations of Light Water Reactor (LWR) configurations require a continuous and careful verification and legalization of the quality of the results, especially when new evaluated nuclear data files are used.

This study deals with the validation study of the SRAC2006 code system based on evaluated nuclear data libraries ENDF/B-VI.8 and JEFF-3.1 for core safety analysis of 3 MW TRIGA Mark-II Research Reactor at AERE, Saver, Bangladesh. The study was performed through the analysis of integral parameters of TRX and BAPL benchmark lattices of thermal reactors. In integral measurements [1], the thermal reactor lattices TRX-1, TRX-2, BAPL-UO2-1, BAPL-UO2-2 and BAPL-UO2-3 were treated as experimental benchmarks for validating the SRAC2006 code system as well as evaluated nuclear data libraries ENDF/B-VI.8 and JEFF-3.1.Validation is an essential part of developing for accurate nuclear reactor physics calculations. The validation will be achieved through the analysis of the integral parameters of TRX and BAPL Benchmark Critical Experiment Lattices [1] of Thermal Reactors. The SRAC-PIJ [2] is a lattice transport code of the SRAC2006 code system [3] based on collision probability method and it is applicable to 16 different lattice modules. The integral parameters of TRX and BAPL Benchmark Lattices were calculated using the elementary code PIJ systematically based on evaluated nuclear data libraries JEFF-3.1 [4] and ENDF/B-VI.8 [5]. The calculated integral parameters were compared with the experimental values [1] and also MCNP results [6] based on CENDL-3.0 library respectively. It was found that in most cases, the calculated results show a good agreement among the said data files as well as the experimental [1] and the MCNP results [6]. To enrich this validation study the group constants (in SRAC format) such as activation cross-section, fission cross-section, nu-fission cross-section, total cross-section, diffusion coefficients, absorption cross-section and scattering cross-section in fast and thermal energy range between the above data files for TRX and BAPL lattices were calculated using the SRAC-PIJ code in fixed source mode.

2.0 THE PIJ CODE OF THE SRAC2006 CODE SYSTEM

The SRAC system [3] is designed and developed at Japan Atomic Energy Research Institute (JAERI) to permit overall neutronics calculations for various types of thermal reactors. The system covers production of effective microscopic and macroscopic group cross-sections, cell and core calculations including burn-up analyses. The collision probability method lattice transport code PIJ [2] is used to analyze this validation study. This code is applicable to 16 different lattice geometries and it does cell homogenization to obtain group constants needed for diffusion calculation of the whole TRIGA core in next. The input files of TRX and BAPL benchmark lattices of thermal reactors are modeled with SRAC-PIJ code, which support the optimized inputs suggested in the final report of the Library Update Project Stage-1 (WIMS). The integral parameters of TRX and BAPL benchmark lattices were calculated using the PIJ code based on 107 energy group (71 fast and 36 thermal) for both nuclear data files JEFF-3.1 and ENDF/B-VI.8 at room temperature 20 0C. The fast 71 group was divided into 4 energy groups and the thermal 36 group was also divided into three energy groups. The total 107 energy group was condensed into 7 energy groups. All calculations were performed in seven energy groups as shown in Table 1. Special consideration was made in development of energy group structure. In this case, the thermal cut-off energy was 0.60236 eV.

Table 1: Seven Energy Group Structure for Generation of Cross-sections Data Sets in SRAC-PIJ Code

Group No.	Energy (eV)		Flux Type
	Upper	Lower	
1	10.000E+06	8.2085E+05	Fast
2	8.2085E+05	5.5310E+03	
3	5.5310E+03	3.9279E+00	Epithermal
4	3.9279E+00	6.0236E-01	
5	6.0236E-01	2.5683E-01	Thermal
6	2.5683E-01	5.4520E-02	
7	5.4520E-02	1.0000E-05	

In addition, the group constants in fast and thermal energy range for TRX and BAPL lattices between ENDF/B-VI.8 and JEFF-3.1 libraries were calculated using the SRAC-PIJ code in fixed source mode where a flux calculation was carried out by separating fast groups 71 and thermal groups (36) in both cases. A fixed source problem by one-point fine-group equations with P_1 or B_1 approximation selected was solved for an imaginary media made by the homogenized cross-sections averaged in the whole system where a flat flux approximation and the fission spectrum of U-235 are assumed. The few-group diffusion coefficients were made from the inverse of the fine group transport cross-sections. After obtaining the few-group total cross-sections, the self-scattering cross-sections were adjusted to keep the neutron balance. The group constants as shown in Tables 5-8 and they were obtained individually in fast and thermal energy range for TRX and BAPL lattices of Thermal Reactors from cell homogenization in fixed source mode. The calculation scheme of SRAC2006 code system for this paper is shown in Figure 1.

3.0 EXPERIMENTAL FACILITY FOR TRX AND BAPL BENCHMARK LATTICES OF THERMAL REACTORS

For this analysis, two types of benchmark lattices were used: (i) H_2O-moderated uranium metal lattices TRX-1 and TRX-2 [7, 8] and (ii) H_2O moderated uranium oxide critical lattices BAPL-UO_2-1, BAPL-UO_2-2 and BAPL-UO_2-3 [9]. There are two types of systems such as TRX and BAPL systems.

4.0 SYSTEM DESCRIPTION OF TRX

These Cross Section Evaluation Working Group (CSEWG) experiments are moderated lattices of slightly enriched (1.3 wt. %) uranium rods with diameters of 0.98297 cm in a triangular pattern. Measured lattice parameters include ρ^{28}, δ^{25}, δ^{28} and C*. These lattices directly test the U-235 resonance fission integral and thermal fission cross section. They also test U-238 shielded resonance capture and the thermal capture cross section. They are sensitive to the U-238 fast fission cross-section, U-238 inelastic scattering and the U-235 fission spectrum. The scattering and thermal absorption cross sections of H_2O are also very important.

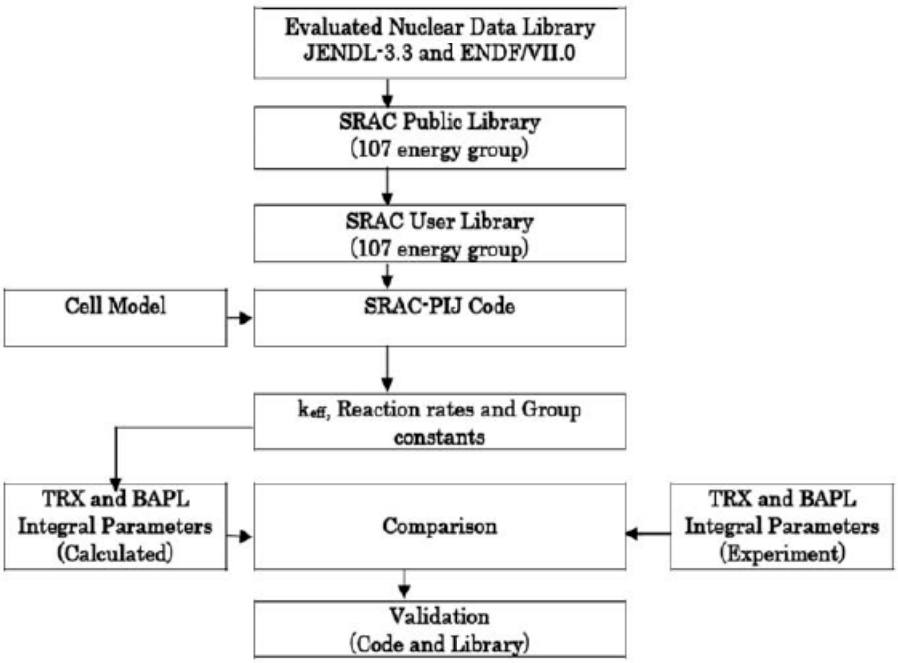

Fig. 1: The Calculation Scheme of the SRAC2006 Code System

4.1 System Description of BAPL

These CSEWG experiments consist of H_2O moderated critical lattices of 1.311 wt. % enriched uranium oxide rods with diameters of 0.9728 cm in a triangular pattern. The measured parameters include ρ^{28}, δ^{25}, δ^{28} and B2. Three lattices with moderator to fuel volume ratios of 1.43, 1.78, and 2.40 are specified.

4.2 Material and Dimensional Properties of TRX and BAPL

The material and dimensional properties of TRX [10] and BAPL [9] are given in the Table 2 and 3.

Table 2: Physical Properties of TRX [10]

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	6.0250E-02
Moderator	*	^{1}H	6.6760E-02
		^{16}O	3.3380E-02

***Lattices spacing of 1.8060, and 2.1740 cm in triangular arrays.**

Table 3: Physical Properties of BAPL [9]

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
		^{16}O	4.6946E-02
Void	0.5042		
Clad	0.5753	Al	6.0250E-02
Moderator	*	^{1}H	6.6760E-02
		^{16}O	3.3380E-02

***Lattices spacing of 1.5578, 1.6523 and 1.8057 cm in triangular arrays.**

1.0 BENCHMARK CALCULATIONS

The integral parameters of TRX and BAPL benchmark lattices of Thermal reactors were calculated using the lattice transport code SRAC-PIJ of the SRAC2006 code system based on JEFF-3.1 and ENDF/B-VI.8 evaluated data files. TRX-1 and TRX-2 used uranium metal fuel in ^{235}U enriched to 1.305 wt. % and BAPL-1, BAPL-2, and BAPL-3 used uranium oxide fuel enriched 1.311 wt. %. TRX and BAPL were water (H_2O) moderated. In integral measurements [46], the thermal reactor lattices TRX and BAPL were treated as experimental/standard benchmarks to validate the physics models used in computer codes (like the SRAC2006 code system) in conjunction with the associated nuclear data libraries JEFF-3.1 and ENDF/B-VI.8 used to describe the microscopic phenomena underlying the macroscopic behavior. The integral parameters analyzed based on reaction rate ratios were defined as below [11] and the measured thermal cut-off energy in this definition was 0.6250 eV:

k_{eff} = Effective neutron multiplication factor (finite medium).

ρ^{28} = Ratio of epithermal to thermal ^{238}U captures.
$= (\Sigma_c)^{38}_{epth} / (\Sigma_c)^{38}_{th}$

δ^{25} = Ratio of epithermal to thermal ^{235}U fission.
$= (\Sigma_f)^{35}_{epth} / (\Sigma_f)^{35}_{th}$

δ^{28} = Ratio of ^{238}U fission to ^{235}U fission.
$= (\Sigma_f^t)^{38} / (\Sigma_f^t)^{35}$

C* = Ratio of ^{238}U captures to ^{235}U fissions.
$= (\Sigma_c^t)^{38} / (\Sigma_f^t)^{35}$

To enrich this validation study the group constants or cross-sections data sets such as activation cross-section, fission cross-section, nu-fission cross-section, total cross-section, diffusion coefficients, absorption cross-section and scattering cross-section in SRAC format for TRX and BAPL Benchmark lattices of thermal reactors were calculated using the SRAC-PIJ code in fixed source mode in fast and thermal energy range based on the evaluated nuclear data libraries JEFF-3.1 and ENDF/B-VI.8.

2.0 RESULTS AND DISCUSSION

The calculated results of integral parameters for TRX and BAPL benchmark lattices of thermal reactors with experiment and MCNP results were summarized in Tables 3 and 4 respectively. For each of the benchmark lattices the measured as well as the calculated values of integral parameters and the associated % uncertainty were given. The effective multiplication factor k_{eff} is the most important benchmark integral parameter. It comprises all reactor physics parameters of the problem: geometry, isotopic composition, cross sections of all isotopes, spectrum etc. For this reason it is very sensitive to the SRAC-PIJ input modeling. In comparison to the experimental results it was found

that the calculated values of k_{eff} for TRX and BAPL lattices using the said data libraries were generally well agreed with each other. But the maximum uncertainty in k_{eff} was 0.393 % for TRX-2 lattice for JEFF 3-1 library. In most TRX cases, the calculated values of other integral parameters were fewer errors than the measured values.

In case of BAPL-2 and BAPL-3 the values of δ^{25} and δ^{28} were somewhat slightly over predicted in comparison to the experiment but show a similar trend with the MCNP results. Since the measurements of parameter C* were not available for BAPL lattices, the comparison was limited to TRX lattices. The differences in the results based on JEFF-3.1 and ENDF/B-VI.8 evaluated data libraries were relatively small. Most of the calculated parameters lie within the % uncertainty interval of the MCNP results as well as experimental values, which indicates that the calculation technique the SRAC-PIJ code of SRAC2006 code system based on the said data files is reliable for the neutronics calculation of Thermal reactors. In addition, the results of the cross section data sets (group constants in SRAC format) in fast and thermal energy range between the said data files were summarized in Tables 5– 8, respectively. From the analysis of the cross section data sets or group constants it was found that the group constants show a good agreement with each other.

Table 3: Comparison of calculated integral parameters for TRX lattices with the values of experiment and reference MCNP.

Lattices	Integral Parameters	Experiment (CSEWG,1986)	Reference MCNP	JEFF-3.1 SRAC-PIJ	ENDF/B-VI.8 SRAC-PIJ
TRX-1	k_{eff}	1.0000(0.30)	0.9975(-0.25)[a]	1.00237(0.237)[a]	0.99536(-0.464)[a]
	ρ^{28}	1.3200(1.60)	1.3608(3.09)	1.3426 (1.71)	1.3404 (1.547)
	δ^{25}	0.0987(1.00)	0.0980(-0.71)	0.0951 (-3.78)	0.0956 (-3.140)
	δ^{28}	0.0946(4.30)	0.0962(1.69)	0.0944 (-0.150)	0.0991 (4.756)
	C^*	0.7970(1.00)	0.7922(-0.60)	0.787 (-1.254)	0.797 (0.114)
TRX-2	k_{eff}	1.0000(0.10)	0.9982(-0.18)	1.00393 (0.393)	0.99828 (-0.172)
	ρ^{28}	0.8370(1.90)	0.8530(1.91)	0.833 (-0.405)	0.831 (-0.705)
	δ^{25}	0.0614(1.30)	0.0620(0.98)	0.0582 (-5.211)	0.0584 (-4.78)
	δ^{28}	0.0693(5.10)	0.0681(-1.73)	0.0697 (-1.87)	0.0763 (1.443)
	C^*	0.6470(0.93)	0.6387(-1.28)	0.633 (-2.163)	0.641 (-0.898)

[a](Error in %)=[(Calculated value- experimental value)/experimental value]x100

Table 4: Comparison of calculated integral parameters for BAPL lattices with the values of experiment and reference MCNP.

Lattices	Integral Parameters	Experiment (CSEWG, 1986)	CENDL-3.0 MCNP	JEFF-3.1 SRAC-PIJ	ENDF/B-VI.8 SRAC-PIJ
BAPL-1	k_{eff}	1.0000(0.10)	1.0023(0.23)[a]	1.00184(0.184)[a]	0.99558(-0.442)[a]
	ρ^{28}	1.3900(0.72)	1.3923(0.16)	1.417 (1.942)	1.414 (1.43)
	δ^{25}	0.0840(2.40)	0.0820(-2.39)	0.080 (-4.76)	0.0812 (-3.33)
	δ^{28}	0.0780(5.10)	0.0736(-5.61)	0.073 (-6.41)	0.0765(-1.923)
	C^*	…..	0.7972	0.8061	0.816
BAPL-2	k_{eff}	1.0000(0.10)	1.0021(0.21)	1.00255(0.255)[a]	0.99055(-0.145)[a]
	ρ^{28}	1.1200(0.89)	1.1602(3.59)	1.17 (4.46)	1.171 (4.55)
	δ^{25}	0.0680(1.50)	0.0669(-1.61)	0.0658 (-4.41)	0.066 (-2.94)
	δ^{28}	0.0700(5.70)	0.0633(-9.57)	0.0679 (-3.00)	0.0658 (-6.0)
	C^*	….	0.7274	0.732	0.741

BAPL-3	k_{eff}	1.0000(0.10)	1.0021(0.21)	1.00329(0.329)[a]	0.99810(-.192)[a]
	ρ^{28}	0.9060(1.10)	0.9130(0.77)	0.917 (1.21)	0.9144(1.38)
	δ^{25}	0.0520(1.90)	0.0515(-0.96)	0.050 (-3.846)	0.050 (-3.84)
	δ^{28}	0.0570(5.30)	0.0518(-9.12)	0.0526 (-7.71)	0.0539 (3.65)
	C^*	….	0.6511	0.653	0.661

[a](Error in %)=[(Calculated value- experimental value)/experimental value]x100

Table 5: Comparison of calculated group constants for TRX lattices in Fast energy range between evaluated data files using SRAC-PIJ code

Lattices	Group Constants in Fast Energy Range	JEFF-3.1	ENDF/B-VI.8
TRX-1	Activation cross-section	0.80390E-01	0.79012E-01
	Fission cross-section	0.29434E-02	0.29654E-02
	Nu-fission cross-section	0.77659E-02	0.77945E-02
	Total cross-section	0.29850E+00	0.29284E+0
	Diffusion coefficient 1	0.11167E+01	0.11383E+01
	Diffusion coefficient 2	0.11167E+01	0.11383E+01
	Absorption cross-section	0.11344E-01	0.11293E-01
	Scattering out cross-section	0.26920E-01	0.26451E-01
TRX-2	Activation cross-section	0.84166E-01	0.83084E-01
	Fission cross-section	0.22259E-02	0.22375E-02
	Nu-fission cross-section	0.59060E-02	0.59109E-02
	Total cross-section	0.28814E+00	0.28397E+00
	Diffusion coefficient 1	0.11569E+01	0.11738E+01
	Diffusion coefficient 2	0.11569E+01	0.11738E+01
	Absorption cross-section	0.83161E-02	0.82969E-02
	Scattering out cross-section	0.33572E-01	0.33135E-01

Table 6: Comparison of calculated group constants for TRX lattices in Thermal energy range between evaluated data files

Lattices	Group Constants in Thermal energy range	JEFF-3.1	ENDF/B-VI.8
TRX-1	Activation cross-section	0.44883E+01	0.44646E+01
	Fission cross-section	0.55460E-01	0.55198E-01
	Nu-fission cross-section	0.13511E+00	0.13450E+00
	Total cross-section	0.14370E+01	0.14534E+01
	Diffusion coefficient 1	0.23196E+00	0.22935E+00
	Diffusion coefficient 2	0.23196E+00	0.22935E+00
	Absorption cross-section	0.97147E-01	0.96962E-01
	Scattering out cross-section	0.42900E-03	0.45613E-03
TRX-2	Activation cross-section	0.48082E+01	0.47755E+01
	Fission cross-section	0.39048E-01	0.38835E-01
	Nu-fission cross-section	0.95128E-01	0.94630E-01
	Total cross-section	0.16902E+01	0.17069E+01
	Diffusion coefficient 1	0.19721E+00	0.19528E+00
	Diffusion coefficient 2	0.19721E+00	0.19528E+00
	Absorption cross-section	0.74099E-01	0.73871E-01
	Scattering out cross-section	0.29990E-03	0.32111E-03

Table 7: Comparison of calculated group constants for BAPL lattices in Fast energy range between evaluated data files

Lattices	Group Constants in Fast Energy range	JEFF-3.1	ENDF/B-VI.8
BAPL-1	Activation cross-section	0.81024E-01	0.79858E-01
	Fission cross-section	0.19270E-02	0.19343E-02
	Nu-fission cross-section	0.50605E-02	0.50603E-02
	Total cross-section	0.27940E+00	0.27481E+00
	Diffusion coefficient 1	0.11930E+01	0.12130E+01
	Diffusion coefficient 2	0.11930E+01	0.12130E+01
	Absorption cross-section	0.88987E-02	0.88613E-02
	Scattering out cross-section	0.21837E-01	0.21515E-01
BAPL-2	Activation cross-section	0.82534E-01	0.81482E-01
	Fission cross-section	0.17652E-02	0.17707E-02
	Nu-fission cross-section	0.46455E-02	0.46411E-02
	Total cross-section	0.27713E+00	0.27302E+00
	Diffusion coefficient 1	0.12028E+01	0.12209E+01
	Diffusion coefficient 2	0.12028E+01	0.12209E+01
	Absorption cross-section	0.81400E-02	0.81135E-02
	Scattering out cross-section	0 .24655E-01	0.24409E-01
BAPL-3	Activation cross-section	0.84132E-01	0.83233E-01
	Fission cross-section	0.15369E-02	0.15402E-02
	Nu-fission cross-section	0.40560E-02	0.40476E-02
	Total cross-section	0.27423E+00	0.27074E+00
	Diffusion coefficient 1	0.12155E+01	0.12312E+01
	Diffusion coefficient 2	0.12155E+01	0.12312E+01
	Absorption cross-section	0.70622E-02	0.70477E-02
	Scattering out cross-section	0.28506E-01	0.28195E-01

Table 8: Comparison of calculated group constants for BAPL lattices in Thermal energy range between evaluated data files

Lattices	Group Constants in Thermal energy range	JEFF-3.1	ENDF/B-VI.8
BAPL-1	Activation cross-section	0.45288E+01	0.45069E+01
	Fission cross-section	0.41025E-01	0.40811E-01
	Nu-fission cross-section	0.99946E-01	0.99444E-01
	Total cross-section	0.11978E+01	0.12100E+01
	Diffusion coefficient 1	0.27828E+00	0.27548E+00
	Diffusion coefficient 2	0.27828E+00	0.27548E+00
	Absorption cross-section	0.72782E-01	0.72605E-01
	Scattering out cross-section	0.36159E-03	0.38141E-03
BAPL-2	Activation cross-section	0.46594E+01	0.46339E+01
	Fission cross-section	0.37280E-01	0.37064E-01
	Nu-fission cross-section	0.90820E-01	0.90315E-01
	Total cross-section	0.13110E+01	0.13240E+01
	Diffusion coefficient 1	0.25425E+00	0.25177E+00
	Diffusion coefficient 2	0.25425E+00	0.25177E+00
	Absorption cross-section	0.68018E-01	0.67809E-01
	Scattering out cross-section	0.31879E-03	0.33755E-03

BAPL-3	Activation cross-section	0.48131E+01	0.47829E+01
	Fission cross-section	0.31870E-01	0.31668E-01
	Nu-fission cross-section	0.77642E-01	0.77165E-01
	Total cross-section	0.14636E+01	0.14772E+01
	Diffusion coefficient 1	0.22775E+00	0.22565E+00
	Diffusion coefficient 2	0.22775E+00	0.22565E+00
	Absorption cross-section	0.61042E-01	0.60808E-01
	Scattering out cross-section	0.26799E-03	0.28505E-03

3.0 CONCLUSIONS

The present study deals with not only the integral parameters but also the group constants of TRX and BAPL benchmark lattices. By comparing the calculated results with experiment as well as earlier published MCNP values (numerical benchmarks) and it was found that the calculated results of integral parameters show no significant differences among JEFF-3.1 and ENDF/B-VI.8 libraries as well as experiment and also earlier published MCNP results (numerically benchmarked) based on CENDL-3.0, which reflect that the numerical benchmark model developed by the lattice transport code SRAC-PIJ was a good confidence level in forecasting integral parameters of TRX and BAPL benchmark lattices of thermal reactors. But it was obvious that different evaluated nuclear data library was the cause of the slight difference between the calculated results.

In addition, the analysis of the group constants or cross-sections data sets in SRAC format for TRX and BAPL benchmark lattices of thermal reactors in fast and thermal energy range based on the said data files show identical with very insignificant differences with each other and this analysis was used to enrich this justification study. Therefore, this study reflects the justification study of the SRAC2006 code system and it will meet up nuclear data for further calculations of TRIGA Mark –II research reactor.

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