Comparison of JEFF-3.1.2 and JENDL-4.0u for TRIGA MARK-II Calculation Through the Benchmarking of Integral Parameter of TRX and BAPL Lattices of Thermal Reactor

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Abstract: The objective of this paper is to select an inimitable nuclear data library from JEFF-3.1.2 & JENDL-4.0u which are valid for theoretical safety analysis of TRIGA MARK-II research reactor. In this work the integral parameter (such as $k_{\text{eff}}$, $\rho_{28}$, $\delta_{25}$, $\delta_{28}$, $C^*$) of TRX and BAPL benchmark lattices of thermal reactors are compared with the experimental result by Cross Section Evaluation Working Group (CSEWG), USA. The nuclear data processing code NJOY99.0 is used to generate 69-group cross-section library from the basic evaluated nuclear data files JEFF-3.1.2 & JENDL-4.0u. TRX and BAPL benchmark lattices are modeled with optimized inputs, which are suggested in the final report of the WIMS Library Update Project Stage-I (WLUP). The integral parameters of five uranium-fuel thermal assemblies: TRX-1 and TRX-2 and BAPL-UO$_2$-1, BAPL-UO$_2$-2 & BAPL-UO$_2$-3 are calculated with the help of reactor lattice code WIMSD-5B based on the generated 69-group cross-section library. Form the comparison of the integral parameters with the experimental values it is found that the obtained results between the two libraries is nearly alike with some uncertainties. But the degrees of uncertainties for the values of integral parameters of JEFF-3.1.2 library are comparatively less. JEFF-3.1.2 is the better library and could be selected for the neutronic calculation of TRIGA Mark-II research reactor at AERE, Savar, Dhaka, Bangladesh.

Keywords: BAPL, JEFF-3.1.2, JENDL-4.0u, NJOY99.0, TRIGA MARK-II, TRX and WIMSD-5B

1. Introduction

A 3 MW TRIGA Mark-II research reactor has been operating in Bangladesh since 1986 at the Atomic Energy Research Establishment (AERE), Savar, Dhaka. The TRIGA Mark-II research reactor of Bangladesh Atomic Energy Commission (BAEC) is the first nuclear reactor in this country, which has been designed and constructed by the General Atomic of USA [1]. The installation of the reactor was started at the end of 1980 under a non-turnkey project, where local participation was about 50%. The reactor achieved its first criticality in the morning of 14 September 1986. This may be identified as a major event in the scientific annals of the country. The reactor was tested and commissioned fully at the end of the October 1986. Since its commissioning, the reactor has been used in various fields of research and utilization; such as, neutron activation analysis (NAA), neutron radiography, neutron scattering, production of radioisotopes, training of manpower, academic research etc [2]. Through these activities the reactor meets the need of the people of Bangladesh to some extent. To determine the neutron flux and the value of the effective multiplication factor ($k_{\text{eff}}$) for a reactor, one has to solve the neutron transport equation. Considering the rates at which neutrons of different energies moving in different directions enter and leave a small volume element derives this equation. The ENDF-6 library is the latest recommended evaluated nuclear
data file for use in nuclear science and technology applications. These advances focus on neutron cross sections, fission product yields, decay data and represent work by the USA Cross Section Evaluation Working Group in nuclear data evaluation that utilizes developments in nuclear theory, modeling, simulation and experiment [3, 4]. To study the neutron parameters including effective multiplication factor, neutron fluxes, power distributions, power peaking factors, excess reactivity and control rod worth calculation by using evaluated nuclear data library, it is very important to select the appropriate data library [5]. The available data libraries are Japanese evaluated nuclear data library (JENDL) [6], Joint Evaluated Fission and Fusion Data Library (JEFF) in Europe [7], ENDF/B-VI in USA [8], China evaluated nuclear data library (CENDL) [9] and BROND in Russia [10]. JEFF is high quality evaluated nuclear data library 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 [11]. Now JEFF-3.1.2 is available which is modified edition of JEFF-3.1.1 [12]. JENDL-4.0 was released in 2010 which contains neutron-induced reaction data for 406 nuclides. JENDL-4.0 is last updated at January, 2016 named JENDL-4.0u. A recent study provide the validation of the data files of JEFF-3.1.2 & JENDL-4.0u for the theoretical study of TRIGA Mark-II research reactor [13, 14]. The aim of this study is to choice a more efficient library for TRIGA calculation among the two valid libraries through the comparison of integral parameter of benchmark lattices TRX and BAPL by using two computer programs NJOY99.0 [15] and WIMSD-5B [16, 17].

2. Methods

The two computer program; NJOY99.0 and WIMSD-5B are used to compare the evaluated data files JEFF-3.1.2 and JENDL-4.0u through benchmarking TRX and BAPL benchmark lattices.

2.1. Computer Code NJOY99.0

The updated version NJOY99.0 of NJOY has the capability to process data in ENDF-6 format [18], which is used in JEFF-3.1.2 and JENDL-4.0u. The chain of NJOY99.0 modules [19] used to generate the 69-group cross section library from the basic data files of JEFF-3.1.2 & JENDL-4.0u is shown schematically in figure 1. NJOY directs the flow of data through the other modules and contains a library of common functions and subroutines used by the other modules. RECONR reconstructs point-wise (energy-dependent) cross sections from ENDF resonance parameters and interpolation schemes. BROADR adds temperature dependence to the point-wise cross sections generated by the RECONR module. UNRESR computes effective self-shielded point-wise cross sections in the unresolved energy range. THERMR computes cross sections and energy-to-energy matrices for free or bound scatterers in the thermal energy range. GROUPR generates self-shielded multi-group cross sections, group-to-group scattering matrices, photon-production matrices and charged-particle cross sections from point-wise input. ERRORR computes multi-group covariance matrices from ENDF uncertainties. MODER converts ENDF "tapes" back and forth between ASCII format and special NJOY blocked-binary format. WIMSR prepares libraries for the thermal reactor assembly codes WIMS-D.

### Table 1. Corresponding energy of 69 groups

| Groups | E_{max} (eV) |
|---|---|
| Fast Groups | 1.00000E+07 |
| Thermal Groups | 1.00000E+07 |
| 1 | 1.00000E+07 |
| 2 | 6.06550E+06 |
| 3 | 3.67900E+06 |
| 4 | 2.23100E+06 |
| 5 | 1.35300E+06 |
| 6 | 8.21000E+05 |
| 7 | 5.00000E+05 |
| 8 | 3.02500E+05 |
| 9 | 1.83000E+05 |
| 10 | 1.11000E+05 |
| 11 | 6.73400E+04 |
| 12 | 4.08500E+04 |
| 13 | 2.47800E+04 |
| 14 | 1.50300E+04 |
| Resonant Groups | 1.00000E+07 |
| 15 | 9.11800E+03 |
| 16 | 5.53000E+03 |
| 17 | 3.51910E+03 |
| 18 | 2.23945E+03 |
| 19 | 1.42510E+03 |
| 20 | 9.06899E+02 |
| 21 | 3.67263E+02 |
| 22 | 1.48729E+02 |
| 23 | 7.55014E+01 |
| 24 | 4.80520E+01 |
| 25 | 2.77000E+01 |
| 26 | 1.59680E+01 |
| 27 | 9.87700E+00 |
| | 1.00000E+00 |
| | 9.96000E-01 |
| | 9.10000E-01 |
| | 8.50000E-01 |
| | 7.80000E-01 |
| | 6.25000E-01 |
| | 5.00000E-01 |
| | 4.00000E-01 |
| | 3.50000E-01 |
| | 3.00000E-01 |
| | 2.80000E-01 |
| | 2.50000E-01 |
| | 2.00000E-01 |
| | 1.80000E-01 |
| | 1.40000E-01 |
| | 1.00000E-01 |
| | 8.00000E-02 |
| | 6.70000E-02 |
| | 5.80000E-02 |
| | 5.00000E-02 |
| | 4.20000E-02 |
| | 3.50000E-02 |
| | 3.00000E-02 |
| | 2.50000E-02 |
| | 2.00000E-02 |
| | 1.50000E-02 |

### Table 2. k_{eff} assessment of TRX benchmark lattices

| Lattices | JEFF-3.1.2 | JENDL-4.0u | Experiment (CSFSG, 1986) |
|---|---|---|---|
| TRX-1 | 0.9853975 (1.46%) | 0.978300 (2.17%) | 1.0000 |
| TRX-2 | 0.9826511 (1.73%) | 0.981949 (1.85%) | 1.0000 |
2.2. Reactor Code WIMSD-5B

The new version code WIMS-D, formally to be identified as WIMSD-5B, developed on the basis of the old WIMSD version of Atomic Energy Authority (AEA) Technology; was implemented on operating system with Lahaey F7713 FORTRAN compiler. In this version, additional possibilities proposed by the code users have been included. The unique WIMSD structure is used with 69 energy group; i.e. 14 fast group, 13 resonance group and 42 thermal groups [20]. The associated energy of 69 groups is presented in Table 1. Reaction of U-235 and U-238 has been taken to calculate the integral parameters of TRX and BAPL lattices using WIMSD-5B code. The cross-section data sets in thermal region have been processed using WIMS library utility code WILLIE for U-235 and U-238 isotopes and compared as well.

2.3. Benchmark Lattices

Two types of benchmark lattices H2O- moderated uranium lattices and H2O-moderated uranium oxide critical lattices are used to benchmarking the evaluated nuclear data files. TRX-1 & TRX-2 [21] are the H2O- moderated uranium lattices as well as BAPL-UO2-1, BAPL-UO2-2 and BAPL-UO2-3 are H2O-moderated uranium oxide critical lattices. The material and dimensional properties of benchmark lattices are listed in Table 3 [22] and Table 4 [23] respectively. The interaction 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 for the two nuclear data libraries. The absorption cross-section, fission cross-section and capture cross-section of U-235 and U-238 in the thermal and epithermal range for each TRX and BAPL lattices have been determined using WIMSD-5B.

2.4. Calculation Techniques

The isotopes listed in the Table 5 are linked with the TRIGA Mark-II at AERE, Dhaka, Bangladesh [24]. These isotopes are processed using NJOY99.0, which can touch the new attribute of the database, in RECONR- BROADR- THERMR- GROUPR- WIMSR cycle by Pentium-IV PC in DOS commend mode at the department of Physics, Jahangimargar University, Bangladesh [25]. Using the WILLIE and WIMSD-5B code two 69-group cross-section libraries are generated from the processed isotope of JEFF-3.1.2 & JENDL-4.0u. For TRX-1, TRX-2, BAPL-UO2-1, BAPL-UO2-2 and BAPL-UO2-3 lattices; the fission cross-section, absorption cross-section, captured cross-section of U-235 and U-238 are computed through the two generated 69-group cross-section libraries by using WIMSD-5B. The effective multiplication factor $k_{eff}$ defined by equation 1 [26] and the other integral parameters ($k_{eff}$, $\bar{\rho}_{th}^{\infty}$, $\delta_{th}^{38}$, $\delta_{th}^{28}$, $C$) of TRX and BAPL lattices listed [27] in Table 2 are calculated and compared with the experimental values by CSEWG [28]. The library provides integral value of TRX and BAPL lattices in minimum deviation from the standard values will be the expected library.

\[ k_{eff} = \frac{\text{neutrons production from fission in one generation}}{\text{neutron absorption in the preceding generation} + \text{neutron leakage in the preceding generation}} \] (1)

\[ \bar{\rho}_{th}^{\infty} = \text{Ratio of epithermal to thermal neutron captures cross-section of U} \] (2)

\[ \delta_{th}^{38} = \frac{\Sigma_{th}^{38}}{\Sigma_{th}^{38}} \text{th} = \frac{(\Sigma_{th} - \Sigma_{th})^{38}}{(\Sigma_{th} - \Sigma_{th})^{38}} \text{th} \] (2)

\[ \delta_{th}^{28} = \text{Ratio of epithermal to thermal neutron fission cross} \] (2)
section of $^{235}\text{U}$

$$= \left(\Sigma_f\right)_{\text{ep}}^{35}/(\Sigma_f^{35})_{\text{th}}$$

$$\delta^{28} = \text{Ratio of } ^{238}\text{U fission to } ^{235}\text{U fission}$$

$$= \left(\Sigma_f^{38}\right)/\left(\Sigma_f^{35}\right)$$

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

$$= \left(\Sigma_c^{38}\right)/\left(\Sigma_f^{35}\right) = \left(\Sigma_a^{38}\right)/(\Sigma_f^{35})$$

3. Results & Discussions

3.1. Cross-Section Comparison

The calculated neutron cross-section of 92-$^\text{U}$-$^\text{235}$ of TRX-1, TRX-2, BAPL-UO$_2$-1, BAPL-UO$_2$-2 and BAPL-UO$_2$-3 benchmark lattices for JENDL-4.0u and JEFF-3.1.2 are plotted in figure 2 to figure 6 (Ep.A= epithermal absorption cross section, Th.A= thermal absorption cross section, T. A= total absorption cross section, Ep.F= epithermal fission cross section, Th.F= thermal fission cross section, T.Cp= total capture cross section and T.F= total fission cross section). The neutron cross-section of 92-$^\text{U}$-$^\text{238}$ for JENDL-4.0u and JEFF-3.1.2 for each benchmark lattice are plotted in figure 7 to figure 11. It is observed that the cross-sections of U-235 are identical for both JENDL-4.0u and JEFF-3.1.2 in each benchmark lattice. For U-238 the cross-sections are almost similar for each lattice of both library but total captured cross-section in BAPL lattice is slightly larger and total fission cross-section is smaller in JEFF-3.1.2 data library.

3.2. $K_{\text{eff}}$ Comparison

The evaluated value the effective multiplication factor for JENDL-4.0u and JEFF-3.1.2 library for TRX benchmark lattice are listed in Table 2. Table 6 represents the evaluated...
The value of $k_{\text{eff}}$ for BAPL benchmark lattice of both nuclear data library. The value enclosed in first parenthesis indicates the percentage of variation of the evaluated value from the corresponding standard value. It is observed that the values of $k_{\text{eff}}$ in TRX-1 & TRX-2 lattices are closer to the standard values for JEFF-3.1.2 than the JENDL-4.0u library. The value of $k_{\text{eff}}$ is equal in BAPL-2 lattice of both library but deviation of $k_{\text{eff}}$ is larger in BAPL-1 & BAPL3 lattice for JENDL-4.0u.

Figure 5. Cross-section comparison of U-235 for BAPL-2 lattice.

Figure 6. Cross-section comparison of U-235 for BAPL-3 lattice.

Figure 7. Cross-section comparison of U-238 for TRX-1 lattice.

Figure 8. Cross-section comparison of U-238 for TRX-2 lattice.

Figure 9. Cross-section comparison of U-238 for BAPL-1 lattice.

Figure 10. Cross-section comparison of U-238 for BAPL-2 lattice.

Figure 11. Cross-section comparison of U-238 for BAPL-3 lattice.
3.3. Comparison of $\rho_{28}$, $\delta_{25}$, $\delta_{28}$, C*

$\rho_{28}$, $\delta_{25}$, $\delta_{28}$ and C* are the integral parameter of benchmark lattice. The values of the integral parameter of TRX & BAPL benchmark lattices for two libraries are listed in table 7 & table 8 respectively. The values of $\delta_{25}$ are equal in TRX-2 lattice. The values of C* are slightly closer to the experimental values by CSEWG in TRX-1, TRX-2 lattices for JENDL-4.0u than JEFF-3.1.2 data library. The values of $\rho_{28}$, $\delta_{25}$, $\delta_{28}$ are found closer to the experimental values by CSEWG for TRX-1 and TRX-2 lattices. The percent of error are more for the values of $\rho_{28}$, $\delta_{25}$ in BAPL-3 lattice but the other values of BAPL-1, BAPL-2 and BAPL-3 more closely for JEFF-3.1.2 than JENDL-4.0u. The maximum deviation of the values of all integral parameters from the experimental values is found for JENDL-4.0u is 9% for $\rho_{28}$ in BAPL-2 lattice. Therefore, almost the integral parameters are closer to the standard values for JEFF-3.1.2 than JENDL-4.0u.

4. Conclusion

This analysis deals with comparison of neutron cross-section of U-235& U-238 and integral parameters of TRX & BAPL benchmark lattices with the experimental values by CSEWG for evaluated nuclear data libraries JENDL-4.0u & JEFF-3.1.2 by using nuclear data processing code NJOY99.0 as well as reactor lattice code WIMSD-5B. From the comparison of cross-section of U-235& U-238 for five benchmark lattices it is found that there are no remarkable variations observed in the two libraries JENDL-4.0u & JEFF-3.1.2. The value of the effective multiplication factor ($k_{eff}$) plays an important role to control the reactor directly and $k_{eff}=1$ provides the reactor is in critical condition. For both libraries the $k_{eff}$ is less than 1 indicates reactor is in subcritical stage. The deviation of value of $k_{eff}$ form critical condition is same only for one lattice for both library and deviation is minor for four benchmark lattices for JEFF-3.1.2 than JENDL-4.0u. This implies the values of $k_{eff}$ for JEFF-3.1.2 are very closer to the critical condition i.e. closer to the experimental values by CSEWG. The Comparison of the values of $\rho_{28}$, $\delta_{25}$, $\delta_{28}$ and C* provide that the deviation of that calculated value for TRX & BAPL lattices are smaller in JEFF-3.1.2 than JENDL-4.0u library. The overall observations reflect that almost values of integral parameters are comparatively closer to the standard values for the evaluated nuclear data library- JEFF-3.1.2. Therefore it could be concluded that JEFF-3.1.2 is to be chosen for theoretical safety analysis of TRIGA reactor at AERE, Dhaka, Bangladesh for better performance.

| Lattices | JEFF-3.1.2 | JENDL-4.0u | Experiment (CSEWG, 1986) |
|----------|------------|------------|--------------------------|
| TRX-1    | $\rho_{28}$| 1.3466 (2.0%) | 1.3517 (2.4%) | 1.3200 |
|          | $\delta_{25}$ | 0.09578 (2.9%) | 0.09573 (3.5%) | 0.0987 |
|          | $\delta_{28}$ | 0.09958 (3.2%) | 0.09958 (2.9%) | 0.0996 |
|          | C*         | 0.78948 (1.0%) | 0.79023 (0.8%) | 0.7970 |
| TRX-2    | $\rho_{28}$ | 0.832 (0.59%) | 0.835 (3.2%) | 0.8370 |
|          | $\delta_{25}$ | 0.0586 (4.4%) | 0.0586 (4.4%) | 0.0614 |
|          | $\delta_{28}$ | 0.0685 (1.1%) | 0.0708 (2.16%) | 0.0693 |

| Lattices | JEFF-3.1.2 | JENDL-4.0u | Experiment (CSEWG, 1986) |
|----------|------------|------------|--------------------------|
| BAPL-1   | $\rho_{28}$ | 1.4767 (6.0%) | 1.48610 (6.4%) | 1.3900 |
|          | $\delta_{25}$ | 0.0821 (2.2%) | 0.08143 (3.0%) | 0.08400 |
|          | $\delta_{28}$ | 0.0774 (8.8%) | 0.0772 (1.0%) | 0.0780 |
|          | C*         | 0.8250 | 0.82811 | ...... |
| BAPL-2   | $\rho_{28}$ | 1.2101 (8.5%) | 1.2230 (9.0%) | 1.1200 |
|          | $\delta_{25}$ | 0.0661 (2.7%) | 0.06601 (3.0%) | 0.0680 |
|          | $\delta_{28}$ | 0.0679 (3.0%) | 0.06721 (3.9%) | 0.0700 |
|          | C*         | 0.7460 | 0.74501 | ...... |
| BAPL-3   | $\rho_{28}$ | 0.9440 (1.7%) | 0.95002 (1.1%) | 0.9606 |
|          | $\delta_{25}$ | 0.0507 (2.3%) | 0.05090 (2.1%) | 0.0520 |
|          | $\delta_{28}$ | 0.0553 (2.9%) | 0.05487 (3.7%) | 0.0570 |
|          | C*         | 0.6616 | 0.66349 | ...... |

References

[1] S. I. Bhuiyan, et al., “Generation of a library for reactor calculations and some applications in core and safety parameter studies of the 3-MW TRIGA MARK-II research reactor”, Nuclear Technology, Vol. 97, pp. 253, March 1992.

[2] M. K. Alam, M. N. Islam and M. A. Zaman, “Study of internal defects and water absorption behavior of single layer Italian tiles using neutron radiography facility of 3 MW TRIGA MK-II research reactor”, Journal of Bangladesh Academy of Sciences, Vol. 31, 2007.

[3] O. Allaoui, et all., “Validation of ENDF/B-VII.0 nuclear data library for shielding calculations using the Monte Carlo method”, International Journal of Advanced Research, Vol. 2, pp. 55-62, 2014.

[4] ENDF-6 Formats Manual, BNL-90365-2009, Brookhaven National Laboratory, pp.3-23, 2009.

[5] M. N. Uddin, M. M. Sarker, M. J. H. Khan, S. M. A. Islam, “Computational analysis of neutronic parameters for TRIGA Mark-II research reactor using evaluated nuclear data libraries” Annals of Nuclear Energy, Vol. 37, pp.302-309, 2010.

[6] K. Shibata, et al., “JENDL-4.0: A new library for nuclear science and engineering”, Journal of Nuclear Science and Technology, Vol. 48, pp. 1-30, 2011.

[7] A. J. Koning, et all., “Status of the JEFF nuclear data library”, Journal of the Korean Physical Society, Vol. 59, pp. 1057-1062, 2011.

[8] ENDF-B/V-VI: The US Evaluated Nuclear Data Library, BNL-NCS-60496, Brookhaven National Laboratory. 1993.
[9] Z. Youxiang, L. Tingjin, Z. Jingshang and L. Ping, “CENDL-3: Chinese evaluated nuclear data library, version 3”, Journal of Nuclear Science and Technology, Vol. 39, pp. 37-39, 2002.

[10] BROND-2. Library of recommended evaluated neutron data, VANT, Ser. Nucl. Const., N 2-3, 13, 1991.

[11] A. J. Koning et al., “The JEFF evaluated nuclear data project”, Proceedings of the International Conference on Nuclear Data for Science and Technology, ND2007, France, 22-27 April 2007.

[12] A. Santamarina, D. Bernard, Y. Rugama, “Validation Results from JEF-2.2 to JEFF-3.1.1”, JEFF Report 22, 2009.

[13] M. M. Islam, M. M. Haque and S. M. A. Islam, “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”, American Journal of Modern Physics, Vol. 5, pp. 35-41, 2016.

[14] M. M. Islam, M. M. Haque and S. M. A. Islam “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”, IOSR Journal of Applied Physics, Vol. 8, pp. 18-24, 2016.

[15] R. E. MacFarlane and D. W. Muir, “NJOY99.0: Code System for Producing Pointwise and Multigroup Neutron and Photon Cross sections from ENDF/B”, RSICC Code Package PSR-480. Los Alamos National Laboratory, Los Alamos, New Mexico, USA, 1999.

[16] J. R. Askew, F. J. Fayers, P. B. Kemshell, “A general description of the lattice code WIMS”, Journal of the British Nuclear Energy Society, Vol. 5, pp. 564, 1966.

[17] T. Kulikowska, “WIMSD-5B: A neutronic code for standard lattice physics analysis”, Distributed by NEA Data Bank. Saclay, France, 1996.

[18] P. F. Rose and C. L. Dunford, ENDF/B-122 Data Formats and Procedures for the Evaluated Nuclear Data File ENDF-6. BNL-44945, Brookhaven National Laboratory, USA. 1990.

[19] M. Halder and S. M. T. Islam, “Comparative study of generated wimsd-5b multigroup constants library based on JENDL-3.2 with JEFF-3.1.1, CENDL-3.0 and original WIMS and validation of generated library through some benchmark experiments analysis”, IOSR Journal of Applied Physics, Vol. 8, pp. 39-43, 2016.

[20] F. Leszczynski, “Description of Wims Library Update Project (WLUP)”, 2002 International Meeting on Reduced Enrichment for Research and Test Reactors, Bariloche, Argentina, November 3-8, 2002.

[21] J. Hardy, Jr. D. Klein, J. J. Volpe, “A study of physics parameters in several water-moderated lattices of slightly enriched and natural uranium”, Nuclear Science and Engineering, vol. 40, pp. 101-115, 1970.

[22] R. L. Hellens and G. A. Price, “Reactor physics data for water-moderated lattices of slightly enriched uranium”, Reactor Technology Selected Reviews-1964, 529.

[23] J. R Brown, D. R. Harris, F. S. Frantz, J. J Volpe, J. C. Andrews and B. H. Noordhoff, Kinetics and buckling measurements in lattices of slightly enriched U or UO2 rods in H2O. WAPD-176, January, 1958.

[24] S. Rana et. al., “Analysis of structural integrity on the renovated primary cooling system of the 3MW TRIGA Mark-II research reactor”, Proceeding of the International Conference on Mechanical Engineering & 14th Annual Paper Meet (6IMEC&14APM), Dhaka, Bangladesh, 28-29 September, 2012.

[25] M. N. Uddin, M. M. Sarker, M. J. H. Khan, S. M. A. Islam, “Validation of CENDL and JEFF evaluated nuclear data files for TRIGA calculations through the analysis of integral parameters of TRX and BAPL benchmark lattices of thermal reactors”, Annals of Nuclear Energy, Vol. 36, pp. 1521-1526, 2009.

[26] J. T. Lee, J, Kim, M. Cho, “Multigroup calculations for TRIGA-type reactor analysis”, Journal of the Korean Nuclear Society, Vol. 10, pp.87-92, 1978.

[27] R. Sher, S. Fiarman, “Studies of Thermal Reactor Benchmark Data Interpretation: Experimental Corrections”, EPRI NP-209, US, October 1976.

[28] Cross Section Evaluation Working Group (CSEWG), “Benchmark specifications with supplements”, Brookhaven national laboratory, National Nuclear Data Center, Upton, New York 11973, BNL-19302, II. ENDF-202, USA, November. 1, 1986.