Orinigal Article

Criticality and Safety Parameter Studies of a 3 MW TRIGA Mark-II Research Reactor and Validation of the Cross Section Library Generated From JENDL and JEF

Md. Manirul Islam [1] *, S.M.T. Islam [2], M.M. Islam [3], Md. Tarek Hossain [4]

1. Govt. B M College of National University, Bangladesh.
2. Govt. B M College of National University, Bangladesh.
3. Govt. B M College of National University, Bangladesh.
4. Barishal Engineering Collage of University of Dhaka, Bangladesh.

* Correspondence: manir.074@gmail.com

Abstract: Criticality and safety parameter play an important role in operation of a nuclear reactor. The criticality and safety parameter studies of TRIGA Research Reactor are very essential for the reactor users. In order to compare the effects of neutron data libraries and reactor calculation codes, various reactor parameters of light water reactor were analyzed. Cell and global calculation codes like WIMSD-5B and CITATION were used in the present study. All the calculations were carried out systematically with the cross section library generated from basic evaluated nuclear data files such as JENDL-3.3 and JEF-2.2. This study determined WIMS package uses to generate few group neutron macroscopic cross section (cell constants) for all of the materials in the core and its immediate neighbourhood and 3-D diffusion code CITATION uses to perform the global analysis of the core to study multiplication factor, neutron flux and power distribution, power peaking factors etc and to recommend that JENDL-3.3 and JEF-2.2 libraries are reliable for thermal reactor calculations. The computational methods, tools and techniques, customization of cross section libraries, various models for cells and super cells and a lot of associated utilities have been standardized and established or validated for the overall core analysis and coupled thermal hydraulic studies. The agreement of the thermal and epi-thermal fluxes with the experiment is reasonably good except at a few points. The calculated total peaking factor shows fairly good agreement with SAR and is more conservative than that in the SAR. The calculated results show no significant differences between JENDL-3.3 and JEF-2.2 libraries. Hence, both the libraries JENDL-3.3 and JEF-2.2 are sufficiently reliable for the criticality and Safety parameters studies of thermal reactor calculations. The methodologies and the strategy established in this study are being used for thermal hydraulic calculations of the 3 MW TRIGA Mark-II research reactors.

Keywords: safety parameter, 3 MW TRIGA, JENDL, JEF

1. INTRODUCTION

Neutron data libraries and reactor calculation codes play an important role in the prediction of criticality and safety parameters of nuclear multiplying systems. In order to compare the effects of neutron data libraries and reactor calculation codes, various reactor parameters like neutron flux, power distribution, power peaking factors and determination of hot spot of light water reactor were analyzed. Cell and global calculation codes like WIMSD-5B and CITATION were used in the present study for the analysis of 3 MW TRIGA Mark-II research reactor. The overall strategy of this study is: (i) Generation of problem dependent cross section library from basic Evaluated Nuclear Data Files such as JEF-2.2 [1] and JENDL-

3.3 with NJOY99.0, (ii) To use WIMS package [2] to generate few group neutron macroscopic cross section (cell constants) for all of the materials in the core and its immediate neighbourhood, (iii) To use 3-D diffusion code CITATION [3] to perform the global analysis of the core to study the criticality and safety parameter of TRIGA Mark-II research reactor [3], and (iv) To compare criticality and safety parameter values obtained by using basic evaluated nuclear data files JENDL-3.3[4] and JEF-2.2 with the experimental values. The hot spot is found physically at the fuel position C4. It is observed that the power peaking factors strongly depend on core configuration and must be calculated from case to case. It is also observed that the keff value obtained by using the library JEF-2.2 yield comparatively better agreement with the experimental result. The agreement between the thermal and epithermal flux distributions as obtained from the measurements and calculations are quite satisfactory, which reflect the validation of the computer codes and associated data libraries JENDL-3.3 and JEF-2.2 for criticality and safety parameter studies of TRIGA Mark-II research reactor at AERE, Savar, Dhaka, Bangladesh.

2. MATERIALS AND METHODS

The famous laboratories of the world have performed the first two steps. Other two were performed in this study. Calculations were carried out by using the following computer codes and along with the associated data libraries, i.e. the reactor lattice code WIMSD-5B & the 3-D diffusion code CITATION. In early sixties a set of different computer programs for lattice cell calculations has been developed at Atomic Energy Authority (AEA) Technology, Winfrith, United Kingdom by different authors [6-15]. The first version of the code available free of charge through the NEA (Nuclear Energy Agency) Data Bank was WIMSD. The latest version, known as WIMSD-5B, was released from Winfrith in 1998 [1,2] for distribution by the OECD/NEA Data Bank [16-20]. A new version of the code WIMS has been developed on the basis of the old WIMS version of AEA Technology Winfrith formally to be identified as WIMSD-5B9. In this version additional possibilities proposed by the code users have been included. The new version of WIMS code under DOS with Lahey FORTRAN compiler has been used with emphasis put on new features of the new version from the point of view of the TRIGA core analysis. The code has been then used in practical calculations of group- constants in seven energy groups for various parts of the TRIGA reactor lattice. The effect of new features and possibilities in the code were examined. The new version writes the group cross section library on a logical unit. It may be mentioned that in its standard form the WIMS-D/4 code used to homogenize cell averaged scattering matrices but to generate cross sections for the non-fuel zones the scattering matrices are required for each region/materials of the cell. The scattering matrices $\sum S$ of the averaged homogenized cell complicates some basic phenomenon of reactor Physics calculations. The diffusion coefficient D is defined

$$D = 1/[3(\sum t - \sum s \mu o)] \quad (1)$$

Where,

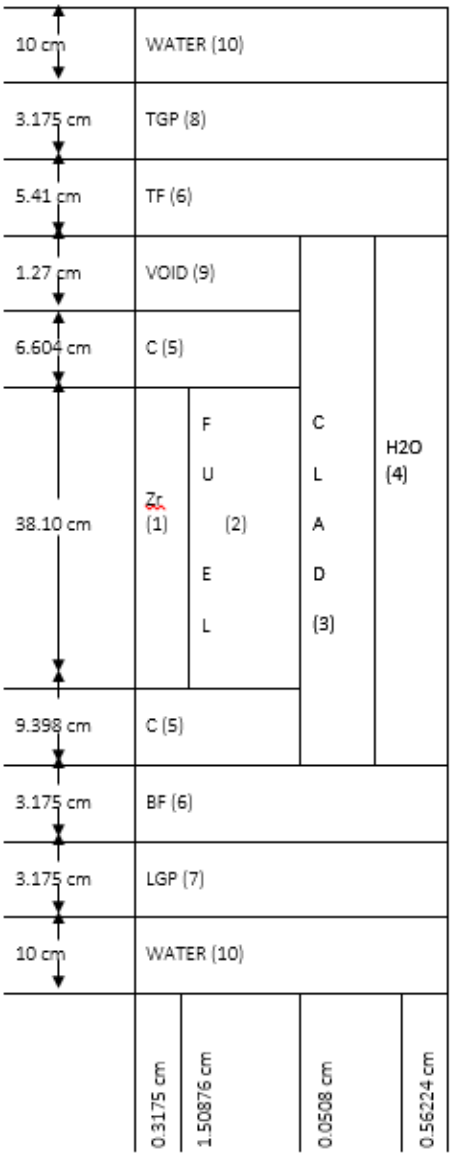
$$\sum t = \sum a + \sum s \quad (2)$$

μo is the average cosine of the neutron scattering

$$\sum tr = (\sum t - \sum s \mu o)$$
$$= \sum s (1- \mu o) + \sum a \quad (3)$$

The set of thermal neutron scattering cross-section data thus must satisfy the detailed balance in equation (3). But while producing group constants with the standard WIMS-D/4 we obtained set of $\sum a$ and $\sum s$ values which becomes greater than $\sum tr$ which cannot be true. A detail investigation showed that WIMSD output writes $\sum g\text{-}g'$ transfer matrices that are cell averaged, although it calculates the region/material dependent scattering cross section internally on a scratch. WIMSD-5B writes the region/material dependent cross sections in the output. In the cell averaging process the group-to-group scattering matrices yielded the error that we had encountered in WIMS-D/4. In the WIMSD-5B this error has been corrected, and it is possible to get scattering cross section $\sum s$ by material/regions. A comprehensive analysis of the core taking into account fuel and non-fuel regions will need a number

of cells which can physically represent total simulation of the core. Various cell and super cell models were tested and finalized to generate cross section data for the fuel and non-fuel regions. It has been used as the basis for the WIMS runs to create homogenized few group cross sections to describe the fuel region. At the center of the fuel meat, there is a zirconium rod with a diameter of 0.4572 cm. This rod is needed during the manufacturing process, the fuel meat (a mixture of uranium and zirconium hydride) surrounds the zirconium rod with an outer diameter of 3.48234 cm. The fuel meat is enclosed by a stainless steel clad with thickness of 0.0508 cm (0.02 inch). The hydrogen in the ZrH is used as an in-fuel moderator that can dramatically shorten the time constant for moderator temperature reactivity feedback. This is one of the factors that give the TRIGA fuel a large prompt negative temperature coefficient of reactivity. The element erbium is added to the fuel as a burnable poison. The isotope Er-167 has a large resonance capture peak at ~ 0.4eV. This enables erbium to absorb neutrons and, therefore, reduce the neutron multiplication factor of the fuel matrix. In addition, erbium has a large enough cross section that depletes away over the lifetime of the fuel. Through trial and error finally three models were accepted for the respective cell description. These are CLUSTER, PRIZE, and CELL with NPLATE option. For LEU Fuel several models were tried to investigate the effects of the group constants on the integral parameters, and it was found that CLUSTER model which is a very close representation of the physical situation, yields the best set of constants. From the output of this WIMS run the cell averaged cross-sections of the four-region central cell (i.e. zircon-rod, fuel meat, clad, and coolant) were weighted and mixed to produce the group constants for the fuel cell. It may be mentioned here (as we will see later) that in the simulation of the TRIGA core we represented a fuel cylinder cut with an equivalent rectangle which includes the surrounding.



Notes:

1. Zr
2. FUEL,
3. CLAD (SS)
4. IN-CORE WATER,
5. GRAPHITE
6. (25% WATER + 75 % SS BY VOLUME)
7. (2% SS + 32.75% WATER + 65.25% AL BY VOLUME)
8. (2 % SS + 98 % AL BY VOLUME)
9. VOID
10. WATER.

Figure 01: The PRIZE model used in WIMSD-5B for the Fuel rod

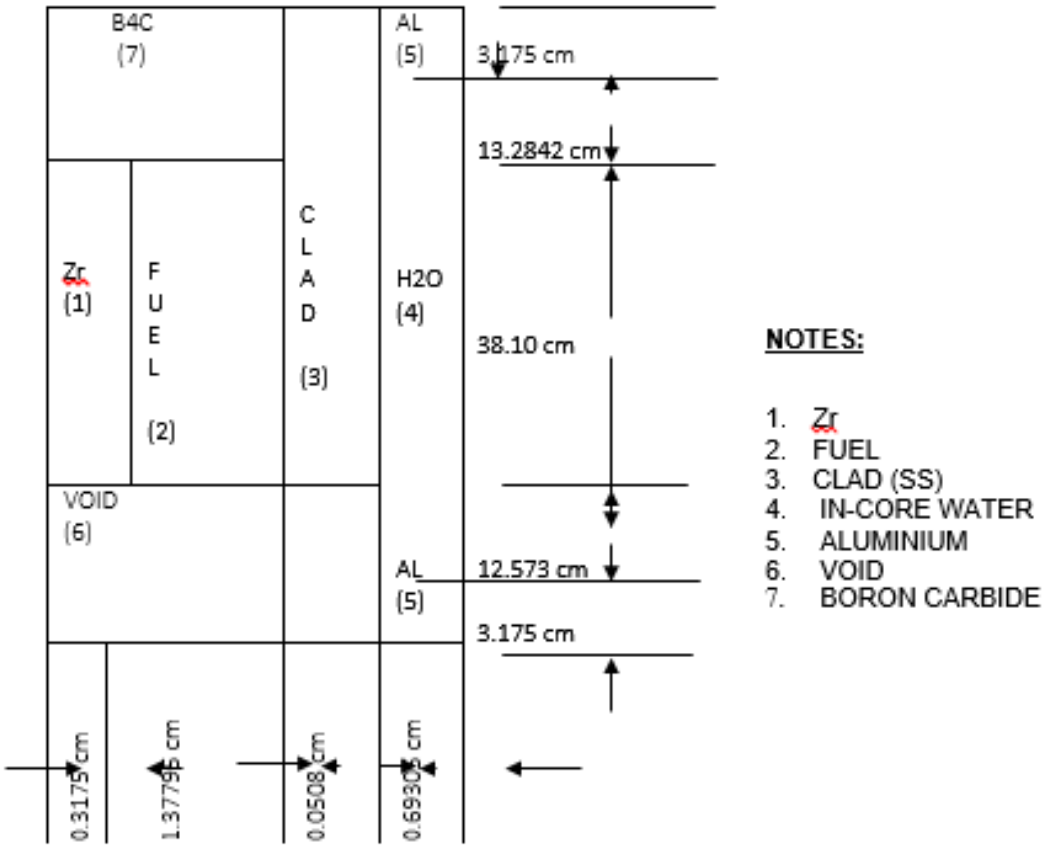


Figure 02: The PRIZE model used in WIMSD-5B for the Fuel follower

Water around the fuel rod, the model is the closest one to reality both from physical and calculated point of view. The CLUSTER model was also found to be suitable for Central Thimble, Pneumatic Transfer Tube, Graphite Dummy Rods, empty source locations, and several others. The performance of the cluster model was found to be very poor for the LEU fuel followers i.e., control rods. In this case, the presence of boron carbide introduced irregularities in the flux distribution. Specially, in the interface between fuel meat and boron carbide the flux gradient influence is significant. To take into account this sharp gradient in the flux the PRIZE option was found to yield the most suitable constants. Figures show the model used in PRIZE run for fuel and control rods respectively. For the same reason the PRIZE model was found appropriate for the graphite reflector above and below the fuel meat. Some other materials having sharp flux gradient also needed PRIZE option for adequate good results. For the materials in the periphery of the core, such as, graphite reflector, lead, graphite-water mixture, and water outside the core the CELL with NPLATE option were found most suitable. Independent runs of the WIMSD-5B code prepared these sub-sets. These sub-sets of the group constants generated at zero burn up in CITATION format. A 7-group structure was used for all the WIMS runs. After careful analysis seven-group structure collapsed from the 69-group WIMS structure were found to be optimum. Energy ranges of the 7- group structure are shown in the table below.

Table 01: Seven-energy-group structure.

Group	Upper energy boundary	Lower energy boundary
Group 1	10.0 MeV	0.500 MeV
Group 2	500.000 keV	9.118 keV
Group 3	9.118 keV	1.123 eV
Group 4	1.123 eV	0.625 eV
Group 5	0.625 eV	0.140 eV
Group 6	0.140 eV	0.050 eV
Group 7	0.050 eV	0.000 eV

In TRIGA fuel, the burnable poison, Er-167, is a major contributor to the resonance absorption. The mesh points were chosen such that they give best results considering the physical geometry of the cell. Sequence 1 (DSN) were used in most cases and in some cases specially, in the PRIZE runs, Sequence 2 (PERSEUS) were used. Thus the generated TRIGA library that contains diffusion coefficient D, absorption cross section $\sum a$, transfer cross section $\sum s$, and production cross section $\nu\sum f$ were customized to interface with CITATION. The models and assumptions used in preparation of the macroscopic cross-section library were investigated in detail. Special emphasis was put on spectral effects. Additional calculations of macroscopic cross-sections for the basic fuel cell in asymptotic spectrum (infinite lattice assumption) were carried out using 69 energy groups with condensation procedures existing in the new WIMS version. The influence of so called 'spectrum indices' choice in WIMS input on final results was tested. It has been established that the model applied by the new version has given correct sets of cross-sections for the basic fuel zone. The same applies to the sets of cross-sections for Central Thimble, Central Thimble at TGP, Graphite dummy elements, Source location at TGP & BGP. For materials to which the asymptotic spectrum assumption could not be applied, the influence of the direct neighborhood on the spectrum and hence the macroscopic cross-sections has been checked. In particular, it has been recommended to apply a 2-dimensional (r, z) PRIZE model of WIMSD to obtain cross-sections for non-fuel planes (Fuel Follower, Pneumatic Tube, Transient Rod, Void in the Fuel Rod, Top SS Fittings, Graphite reflector above fuel meat, Boron Carbide, Top grid plate at fuel & Graphite Dummy element, Top grid plate at control rod position, Void above BGP of the control rod, Control rod at BGP, Bottom grid plate, Bottom fittings). The number of cross-section sets for source locations could be reduced due to negligible differences between cross-sections calculated for bottom and top grid plate positions with the same material composition. The Fuel Followers (FF) were treated especially as it contains control rod having boron carbide as absorbing material. In FF in-core zone has been divided into three sub-zones with the interfaces chosen to ensure the effect of spectral influence of the boron carbide absorber situated above the core level. Separate calculation has been carried out for the central thimble surrounded by two rings B and C; ring B with graphite dummies and ring C with 12 fuel elements. The three sets of 7-group cross sections have been picked up for the central thimble, for ring B with graphite dummies, and for ring C composed of fuel elements. Up to now the 'moxon iron' data from WIMS library had been used for stainless steel canning and fuel element fittings. The influence of applying the stainless steel composition was tested using the full SS-304 composition. Also a change of isotope applied as iron in the WIMS library was performed. The detailed analysis has led to a final conclusion on using the fuel-follower cross sections from the 69-group PRIZE calculation with three sub zones in fuel followers. The whole core simulation using the TRIGAP package is based on 1-dimensional diffusion theory. This approach is not sufficient for a full core analysis aiming at modification of reactor configuration. The more detailed multidimensional analysis is needed to establish neutronic and safety parameters of the TRIGA core. Global calculation encompasses the whole core simulation, for this analysis the CITATION code was used in the present work. This work is done using the libraries JENDL-3.3 and JEF-2.2. Simulation of TRIGA core has been attempted in X, Y, Z geometry. The geometry of the core consists of concentric layers of hexagons designated by A, B, C, D, E, F, G with a equidistant pin rod array of hexagonal symmetry. The actual geometry of the grid plate has permitted the concentric hexagonal division of the core. A detailed representation of a horizontal cross section of the TRIGA reactor core has been used with exact positioning of fuel & control elements, graphite dummy elements, source location, and central thimble, followed by surrounding graphite reflector, lead and water. The surrounding graphite reflector, lead, and water shield were also taken into account by equivalent rectangular cells necessary to represent the exact geometry. The geometry of the Lazy Susan housed in the outer core graphite reflector assembly was also taken into account. The beam ports have been inserted in the reflector assembly. In each rectangular cell three mesh points in X-direction and two in Y-direction i.e. six points in a single cell were considered. In the whole XY- simulation 49 meshes in X-direction (column) and 53 meshes in Y-direction (row) have been defined. In the Z-direction (axial) 41 mesh points have

been considered. The XY simulation also shows the other non-fuels included in the calculation. A detail treatment was followed in axial direction (Z-direction). It may be mentioned that the sub-regions of the active fuel near the top and bottom graphite reflector were treated specially with more mesh points having smaller mesh sizes to take into account the fine structure of the flux gradient in the fuel-reflector interface region.

3. RESULT & DISCUSSION

3.1. Flux Distribution

A comprehensive analysis of the various fluxes, and the flux-distributions in the X, Y, Z directions have been performed. The fluxes were normalized to 3 MW, the thermal (groups 5, 6 & 7) and epi-thermal (groups 3 & 4) fluxes have been compared with the experimental thermal and epi-thermal fluxes. It may be mentioned here that the experiments were performed by activation method with gold foils. The thermal and epi-thermal fluxes were determined by activating gold foils with and without cadmium. Comparisons of the calculated keff and flux values with the experimental results are shown in the table.

Table 02: Comparison of the calculated keff values using the libraries JENDL-3.3 & JEF-2.2 with the experimental results.

Library	Multiplication factors, keff		Difference = [(C-E)/E] x 100*
	Experimental	Theoretical	
JENDL-3.3	1.077459	1.080333	(0.266%)
JEF-2.2		1.077981	(0.048%)

*C = calculated value or theoretical value and E = experimental value

Table shows that the calculated values of Effective multiplication factor (keff) using the data library JEF-2.2 is better than that of data library JENDL-3.3. However, both the results are in good agreement with the experimental result.

Table 03: Comparison of the calculated thermal flux values using the libraries JENDL-3.3 & JEF-2.2 with the experimental results.

Library	Maximum Thermal Flux (x1013)		Difference = [(C-E)/E] x 100*
	Calc.	Expt.	
JENDL-3.3	6.2280	5.56	(12.01%)
JEF-2.2	6.260		(12.58%)

Table 04: Comparisons of the calculated epi-thermal flux values using the libraries JENDL-3.3 & JEF-2.2 with the experimental results.

Library	Maximum Epi-thermal Flux (x1013)		Difference = [(C-E)/E] x 100*
	Calc.	Expt.	
JENDL-3.3	2.13	1.71	(24.56%)
JEF-2.2	2.22		(29.28%)

Table indicates that the calculated values of thermal and epi-thermal fluxes using the data library JENDL-3.3 are better than those of the data library JEF-2.2. The variations of the results are due to the difference of the libraries. The seven group fluxes by CITATION have been investigated in several locations of interest in the core, both radially and axially. The graphical representation of the flux distribution along X, Y and Z directions are shown in the figure. The axial flux is compared with the experimental values.

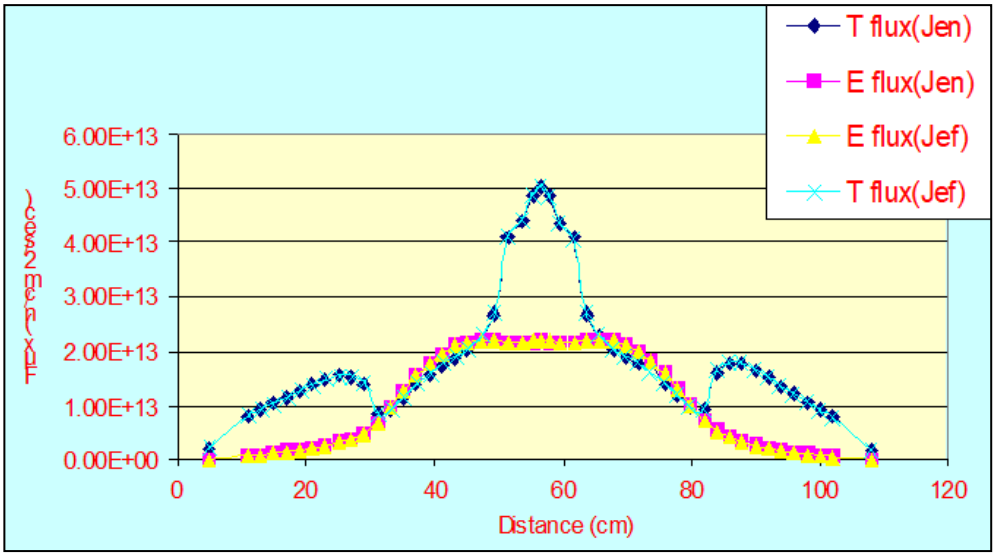


Figure 03: Flux distribution along X-direction

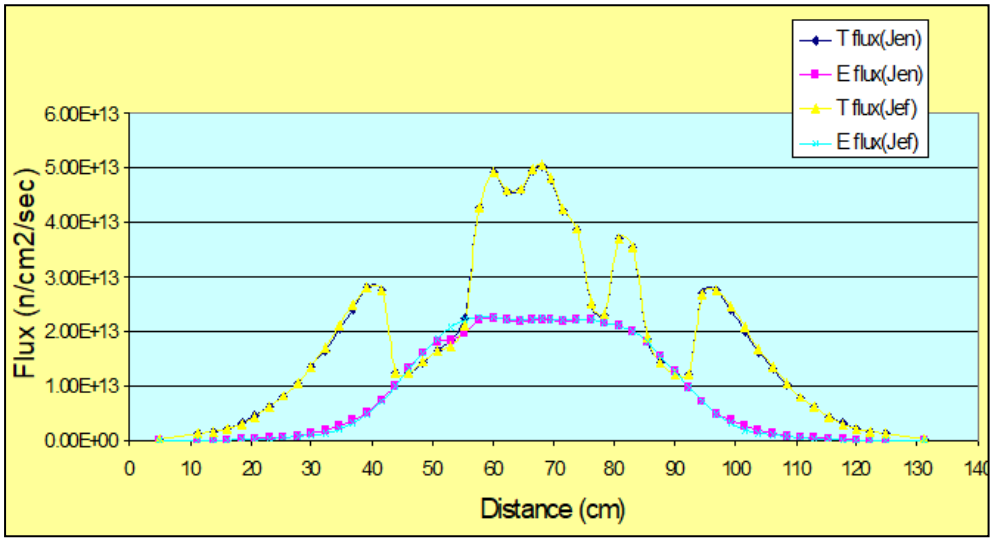


Figure 04: Flux distribution along Y-direction

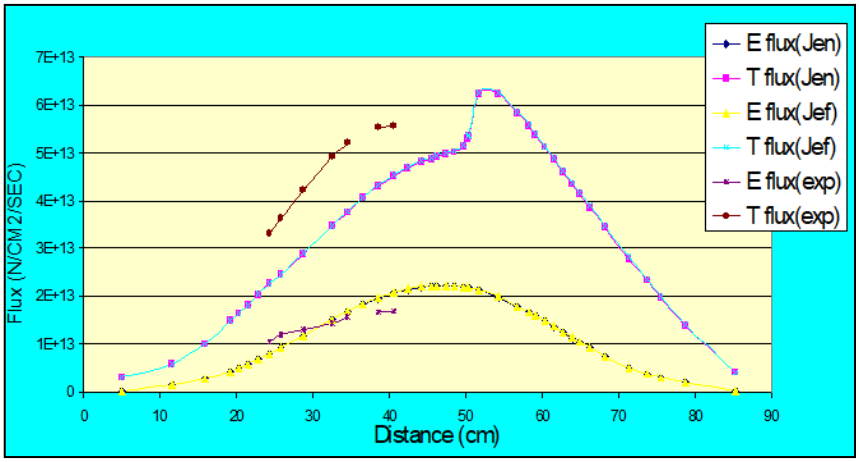


Figure 05: Axial flux distribution

Figures describe that the thermal flux distribution along X, Y and Z direction for both JENDL-3.3 and JEF-2.2 are in good agreement and above description is also true for epi-thermal flux distribution. The axial distribution profile shows that the agreement of the thermal and epi-thermal fluxes with experiment is reasonably good except at a few points near the core mid-plane. It appears that the experimental value at the axial mid-point of the CT is over predicted.

3.2. Power Distributions

Power distribution is very important for the safety of a nuclear reactor. It is also used for the thermal hydraulic analysis of the core. This is why, a comprehensive analysis of the various powers and the power distributions in the X, Y, Z directions have been performed. The total power produced with in the core was calculated through CITATION & PPCT using JENDL-3.3 & JEF-2.2. The maximum power production of 5.5619x 104 kw is observed with in the fuel element designated by C4 by using JENDL-3.3 and JEF-2.2 data libraries and is assumed the hottest rod in the TRIGA core. The axial (z-plane) power distribution of the core containing the Hot spot i.e., the point at which maximum power density occurs, is shown in the figure. The graphical representation of the power distribution along X, Y and Z directions are shown in the figure.

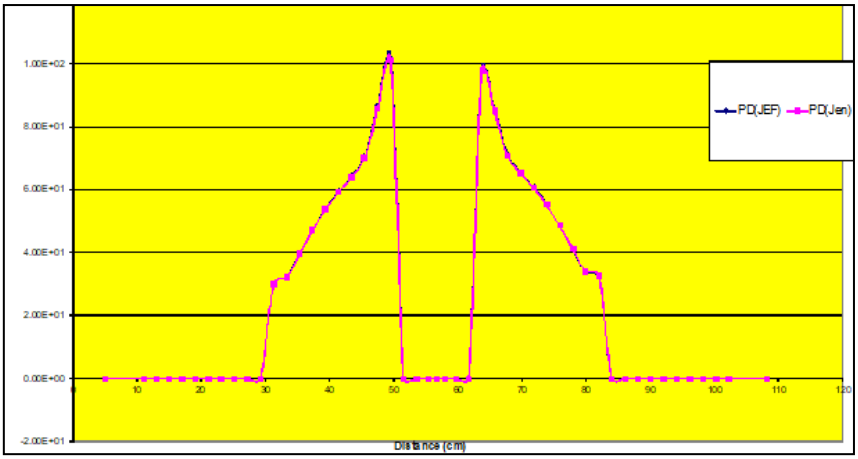


Figure 06: Power density distribution in X-plane

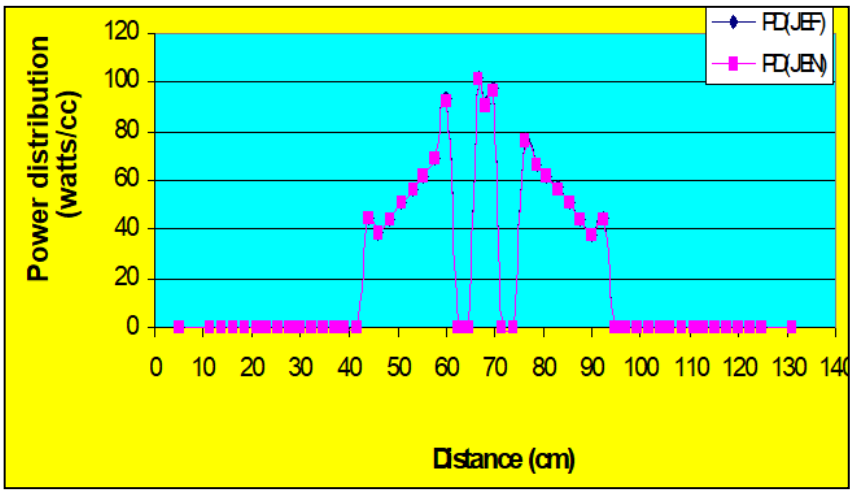


Figure 07: Power density distribution in Y-plane

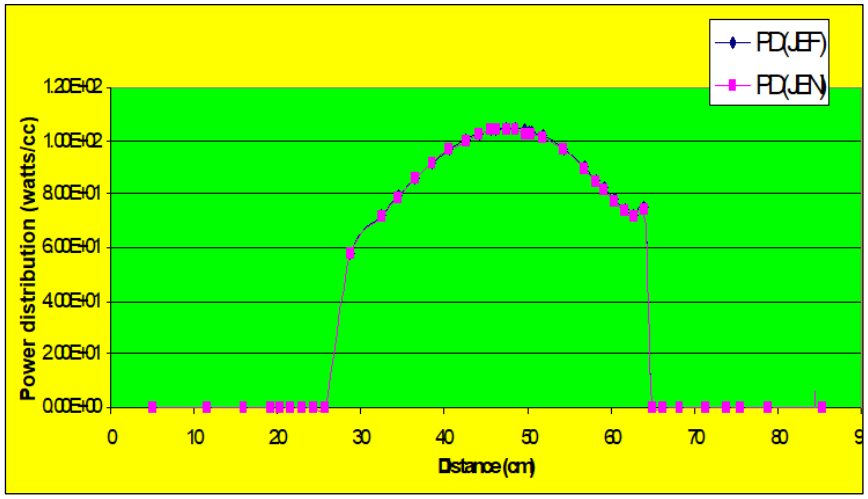


Figure 08: Axial Power density distribution

3.3. Power Peaking Analysis

The peaking factors are the most important safety parameter of a thermal reactor. Hence, it was studied very carefully and thoroughly. The power peaking factors have been defined and analysed in terms of the following parameters, hot rod power peaking factor - fHR : It is the maximum of the 'rod power factor', which is defined as the power generation in a fuel rod relative to the core average power generation i.e. Prod / $\bar{P}$core

Therefore hot rod factor fHR = (Prod / $\bar{P}$core) max

Axial power peaking factor - fZ : The axial power peaking factor fz is defined as the axial peak to average power ratio.

i. e. fZ = (P/ $\bar{P}$) axial

Radial power peaking factor - fR : Radial peaking factor fR is defined as the peak-to-average power ratio on a radial plane with in a fuel rod.

i. e. fR = = (Prod / $\overline{P}$rod) radial

The maximum power density at the hot spot was 1.0419 x 102 and 1.0468 x 102 watt/cc by using data libraries JENDL-3.3 and JEF-2.2 respectively. The various peaking factors are compared with those in the original SAR in table.

Table 04: Comparison among the calculated values of power peaking factors using JENDL-3.3 and JEF-2.2 with the SAR values

Type of peaking	Calculated value using JENDL-3.3	Calculated value using JEF-2.2	SAR value
fHR	1.79	1.79	1.70
fZ	1.4122	1.4184	1.25
fR	2.0526	2.0623	2.65
fT (Total peaking factor)	5.1932	5.2407	5.6313

Table shows that the hot rod factor fHR and the axial power peaking factor fz calculated by CITATION are relatively higher than those reported value in the original SAR. The radial power peaking factor fR calculated by CITATION is relatively lower than original SAR value. The calculated total peaking factor shows fairly good agreement with SAR and is more conservative than that in the SAR.

4. CONCLUSION

The computational methods, tools and techniques, customisation of cross section libraries and a lot of associated utilities have been standardised and established/validated for the overall core analysis. The chain of NJOY99 - WIMSD-5B - CITATION codes forms a powerful technology for the overall analysis and design of TRIGA research reactors. The calculated values of effective multiplication factor (keff) using the data library JEF-2.2 is better than that of data library JENDL-3.3. However, both the results are in good agreement with the experimental result. The agreement of the thermal and epi-thermal fluxes with the experiment is reasonably good except at a few points. The calculated total peaking factor shows fairly good agreement with SAR and is more conservative than that in the SAR. The calculated results show no significant differences between JENDL-3.3 and JEF-2.2 libraries. Hence, both the libraries JENDL-3.3 and JEF-2.2 are sufficiently reliable for the criticality and Safety parameters studies of thermal reactor calculations. The methodologies and the strategy established in this study are being used for thermal hydraulic calculations of the 3 MW TRIGA Mark-II research reactors.

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