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Analysis of Power Peaking Factors of 3 MW Triga Mark-Ii Research Reactor Using the Deterministic Diffusion Code SRAC-Citation

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Abstract: The aim of this paper is to analyze the power peaking factors of 3 MW TRIGA Mark-II research reactor based on diffusion method. Power peaking factors are necessary for safe operation of nuclear reactor and they are one of the most important core safety parameters of nuclear reactor. The most important power peaking factors are the hot rod power peaking factor (f_{HR}), the axial power peaking factor (f_Z), the radial power peaking factor (f_R) and the total power peaking factor (f_T). These factors are calculated using the 3-D diffusion code SRAC-CITATION of the comprehensive neutronics calculation code system SRAC2006 based on the evaluated nuclear data libraries JENDL-3.3 and ENDF/B-VII.0 respectively. The calculated results are compared to the available safety analysis report (SAR) values of 3 MW TRIGA Mark-II reactor by General Atomic as well as the MCNP4C results based on the evaluated nuclear data library ENDL/B-VI. It was found that the calculated results are well consistent between the said libraries as well as the SAR values and the MCNP4C results respectively. Therefore, this analysis will be important to improve the power peaking factors data as the core safety parameters of 3 MW TRIGA Mark-II research reactor, AERE, Dhaka, Bangladesh for its safe operation.
Keywords – Power Peaking Factors, 3 MW TRIGA Reactor, SRAC-CITATION code, SRAC-PIJ Code.

I. INTRODUCTION

Generally, the monitoring of the core power distribution in a nuclear reactor is a prerequisite for the safe operation of nuclear reactor to ensure that various safety limits imposed on the fuel pellets and fuel clad barriers, such as the local power density and the departure from the nucleate boiling ratio (DNBR) are not violated during the reactor's operation. Knowing more about the local power density (LPD) at the hot spot of a nuclear reactor core can provide more important information than knowledge of the LPD at any other position in the reactor core. The LPD at the hot spot needs to be estimated accurately in order to prevent the fuel rod from melting in a nuclear reactor. The LPD is directly related to the power peaking factor and also fuel temperature at hot spot.

As fuel temperature is one of the most important limiting conditions for nuclear reactor operation. It depends on the reactor design, thermal-hydraulics properties and on the power density released in fuel. Power density distribution depends on core configuration and loading pattern. Fuel power density in the reactor core must be limited due to temperature, thermal-hydraulics and mechanical design limitations. The peak adiabatic temperature in TRIGA LEU fuel rod [1] is limited to 1150^0 C during transient condition by the internal pressure due to dissociation of Hydrogen in Zirconium-Hydride at high temperature. It is the responsibility of the reactor operator to keep temperature and consequently the maximum power density within the limits prescribed by the safety analysis report.

In this study, power distribution calculations are performed in diffusion approximation assuming unit cell homogenization using the 3-D diffusion code SRAC-CITATION [2]. Each fuel rod is homogenized with water and other components of the unit cell. Also the power density distribution is smeared over the entire unit cell volume. All of the fuel rods (each containing 20 wt. % of uranium) are 19.7 % enriched and fresh (zero burn-up). The said most important power peaking factors of 3 MW TRIGA Mark-II research reactor are calculated based on the evaluated nuclear data libraries JENDL-3.3 [3] and ENDF/B-VII.0 [4] respectively. So this analysis will assist to improve the power peaking factors data as core safety parameters for reactor operator group of 3 MW TRIGA Mark-II reactor and also its safe operation.

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II. CALCULATION TECHNIQUES

A. Reactor Simulation Codes

Two reactor engineering codes of the comprehensive neutronics calculation code system SRAC2006 [5] were used to perform this analysis and these are (i) the collision probability method lattice transport code SRAC-PIJ [6] was used for the generation of group constants or cross-sections data sets for various core regions of TRIGA reactor and (ii) the diffusion approach code SRAC-CITATION was used to perform global core calculations of TRIGA Mark-II reactor. The calculation scheme of the SRAC2006 code system is shown in Fig. 1.

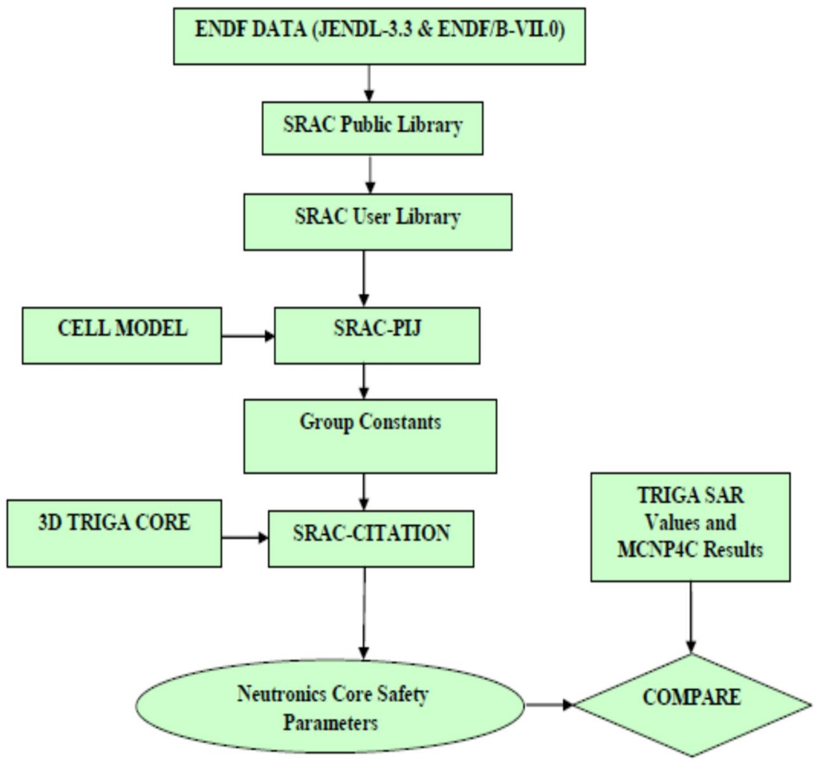

Fig. 1. Calculation scheme of the SRAC2006 code system

B. Simulation Methodology

The simulation methodology was developed in two steps (i) by the SRAC-PIJ model and (ii) by the SRAC-CITATION model. In case of SRAC-PIJ model, cell calculation of the LEU (Low Enriched Uranium) fuel rod was performed by the lattice physics transport code SRAC-PIJ with 1-D hexagonal model using geometry type IGT=6 (hexagonal cell).

For simulating actual spectrum, fuel cell was divided into several annular rings. Same model was also used for fuel part of the control rod. The cross section obtained from this model was used in SRAC-CITATION to model the 3-D TRIGA fresh core. Graphite dummy element, control rod element, pneumatic transfer tube, transient rod with boron carbide part and the transient rod with air follower part were modeled with the geometry type IGT=12 (hexagonal assembly with asymmetric pin rods). In this model the rod of interest was in the center surrounded by six fuel rods. This model was employed to simulate actual situation in the TRIGA LEU core so that reliable cross sections data could be generated.

Central thimble (CT) cell was modeled using geometry option 12 where the central rod, CT was surrounded by six graphite dummy elements and twelve fuel rods in the first and second layer of the model respectively to simulate the surrounding of the CT in the TRIGA core. The effective macroscopic cross section for the B-ring used in the global calculation was obtained from the CT cell model. The PEACO routine [7] was selected for the calculation of effective resonance cross section in the resonance energy range. PEACO solves a multi-region cell problem by the collision probability method using an almost continuous (hyper-fine) energy group structure for the resonance energy range. The interaction of resonances can be accurately treated by the PEACO routine. For the material outside the active core region such as top and bottom fitting, grid plate, top and bottom reflector, lazy Susan, graphite

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blanket and lead shielding, asymptotically collapsed cross sections were used in the CITATION.

The SRAC-PIJ code used the 107 energy group (73 fast and 34 thermal) for both nuclear data libraries JENDL-3.3 and ENDF/B-VII.0 respectively. The fast (73) group was divided into 4 energy groups and the thermal (34) group was also divided into 3 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 I.

TABLE I

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

Energy Group no.	Energy (eV)		Flux type
	Upper	Lower	
1	10.000E+06	8.2085E+05	Fast
2	8.2085E+05	9.1188E+03	
3	9.1188E+03	8.3153E+00	Epithermal
4	8.3153E+00	4.6912E-01	
5	4.6912E-01	2.1878E-01	Thermal
6	2.1878E-01	3.7813E-02	
7	3.7813E-02	1.0000E-05	

Special consideration was made in development of energy group structure. First group was based on above threshold fission of ^{238}U and no delayed production. Second group was based on the average energy of delayed neutron group produced. Groups 1 and 2 were considered as fast energy group. Groups 3 and 4 were considered as epithermal group. Groups 5, 6 and 7 were considered as thermal energy group. Thermal cut-off energy in calculation was 0.46912 eV while in measurement thermal cut-off energy of Cadmium with gold foil was 0.41 eV. This group structure was expected to yield better results.

In case of SRAC-CITATION model, the 3-D neutron diffusion code SRAC-CITATION was used to perform various core parameters (like k_{eff}, power peaking factors, neutron fluxes and adjoint fluxes etc,) of the 3 MW TRIGA Mark-II reactor by solving the time-independent neutron diffusion equation. The TRIGA LEU core was modeled in X, Y, Z, geometry by this code as shown in Fig. 2. The present core of the reactor consists of a total of 100 fuel elements (including five fueled follower control rods), six control rods, 18 graphite dummy elements, one Central Thimble and one pneumatic transfer system irradiation terminus. The geometry of the core consists of concentric layers of hexagons designated by A, B, C, D, E, F and G with an equidistant rod array of hexagonal symmetry.

The actual geometry of the grid plate was allowed the concentric hexagonal divisions of the core. In the concentric hexagons rods were mounted. Number of rods in the hexagonal array was multiple of six. A detailed representation of a horizontal cross section of the TRIGA reactor core was used with the exact positioning of fuel and 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 to represent the exact geometry. The geometry of the Lazy Susan housed in the outer core graphite reflector assembly was simulated. The beam ports were inserted in the reflector assembly.

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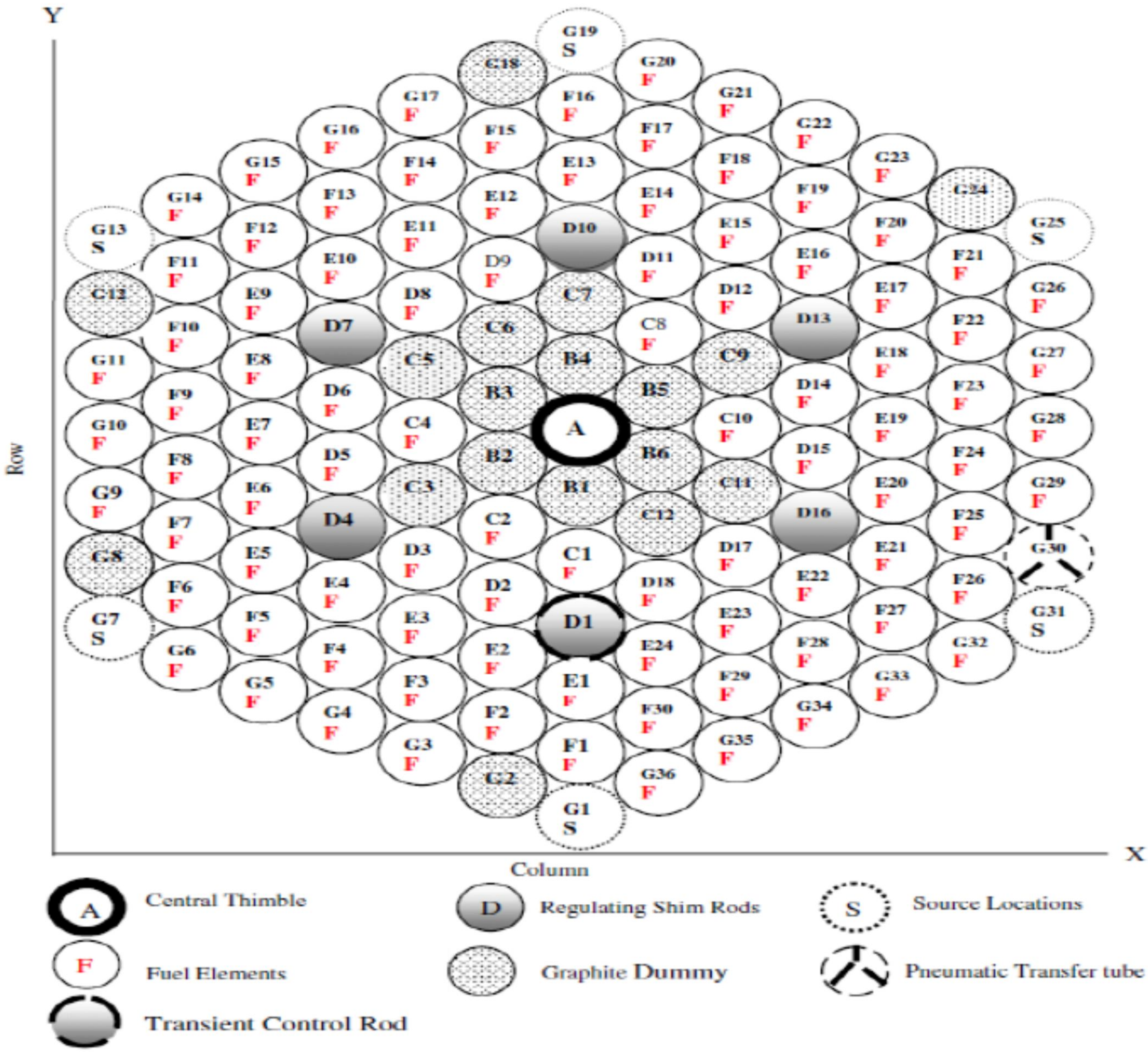

Fig. 2: Present core configuration of 3 MW TRIGA Mark II research reactor

In each hexagonal fuel cell four mesh points in X-direction and four in Y-direction i.e. sixteen points in a single fuel cell were considered and mesh point was different in non-fuel cell. In the whole XY- simulation 104 meshes in X-direction (column) and 104 meshes in Y-direction (row) were defined. In the Z-direction (axial) 105 mesh points were considered. The XY simulation also shows the other non-fuels included in the calculation.

III. POWER PEAKING FACTORS

Power peaking factors are necessary to evaluate accurately for safe reactor operation in order to reflect the safety condition of nuclear reactor. Since power peaking factors are the bridge between the nutronics and thermal-hydraulics analysis of the nuclear reactor core. Because of they describe maximum power released locally in the core and consequently maximum fuel temperature was generated in the fuel rod (at hot spot) of the core. The most important power peaking factors are usually introduced in TRIGA

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reactor at steady state and pulse operation [8]:

A. Hot rod power peaking factor (f_{HR})
B. Axial power peaking factor (f_Z)
C. Radial power peaking factor (f_R) and
D. Total power peaking factor (f_T) = $f_{HR} \times f_Z \times f_R$

They determine maximum total power released by one fuel element as well as its axial peaking value and radial peaking value which are used as parameters in thermal hydraulic analysis for the calculation of maximum fuel temperature and DNBR (departure from nuclear boiling ratio). They are directly applicable to TRIGA reactor for its safe operation and safety analysis.

A. Hot rod power peaking factor (f_{HR})

It is defined as the ratio between the maximum power released by one fuel rod, (P_{rod}) max and the average fuel rod power in the core, P_{core}:

$$\boldsymbol{f_{HR}} = \frac{(\boldsymbol{P_{rod}})\boldsymbol{max}}{\boldsymbol{P_{core}}} \qquad (1)$$

According to this definition, $\boldsymbol{P_{core}} = \frac{\boldsymbol{P}}{\boldsymbol{N_{el}}}$ (2)

Where, P is the total thermal power (e.g. 3 MW for TRIGA) and N_{el} is the number of fuel element in the core. Taking into account that all types of fuel elements have the same volume of fissionable material, the definition of f_{HR} applies also to the ratio between the average power density $\bar{P}_{rod}$ of the hot rod and the core average power density, $\bar{P}_{core}$.

$$\boldsymbol{f_{HR}} = \frac{(\bar{\boldsymbol{P}}\boldsymbol{rod})\boldsymbol{max}}{\bar{\boldsymbol{P}}\boldsymbol{core}} \qquad (3)$$

This definition (3) is more appropriate for pulse analysis while definition (1) is more convenient for steady state condition. It (f_{HR}) determines the power generation of the hottest fuel rod in the core and it is used in the thermal analysis for determination of the maximum fuel temperature.

B. Axial power peaking factor (f_Z)

It is defined as the ratio between maximum and average axial power density, $(P_Z)max$ and $\bar{P}_Z$ respectively:

$$\boldsymbol{f_Z} = \frac{(\boldsymbol{P_Z})\boldsymbol{max}}{\bar{\boldsymbol{P}}_{\boldsymbol{Z}}} \qquad (4)$$

In principle f_Z may vary from fuel element to fuel element. But the differences between axial power distributions in different fuel elements in the same core are small because TRIGA reactor is very small and axial power distributions are hard. The axial power distribution is relatively independent of fuel temperature or radial position in the core. The shape of the axial power distribution changes significantly adjacent to a partially inserted control rod, but the peak power value is essentially equal to the unloaded value.

C. Radial power peaking factor (f_R)

It is defined as the ratio between maximum and average power density P(x,y,z) in radials integrated in a particular point (x,y) over entire height (h) of the fuel rod:

$$\left(\frac{1}{h}\int dz\, p(x,y,z)_h\right)max = \left(P_r(x,y)\right)max \qquad (5)$$

and its average $\left(\frac{1}{h}\int dz\, p(x,y,z)_h\right)av = \left(P_r(x,y)\right)av$ (6)

$$f_R = \frac{(P_r(x,y))max}{(P_r(x,y))av} \qquad (7)$$

In the fuel element radial power distribution is important for calculations of fuel and cladding temperature distributions. Power density is approximately proportional to thermal flux distribution which reaches its maximum in water around the fuel element and decreases in the fuel element due to much higher absorption in fuel than water. The radial power distribution within the fuel element has a small effect on the peak temperature due to its distribution is assumed to be constant (independent of radius) in temperature calculations.

D. The total power peaking factor (f_T)

It is defined as the product of hot rod power peaking factor (f_{HR}), axial power peaking factor (f_Z) and radial power peaking factor (f_R):

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$$\boldsymbol{f_T} = \boldsymbol{f_{HR}} \times \boldsymbol{f_Z} \times \boldsymbol{f_R} \quad (8)$$

It is used to calculate the peak fuel temperature under adiabatic condition where temperature distribution is the same as power distribution in the core and it also implies that the peak fuel temperature will be value of f_T times higher than the average fuel core temperature.

IV. RESULTS AND DISCUSSIONS

The calculated results of the power peaking factors of 3 MW TRIGA Mark II reactor core are summarized in Table II and Table III and these parameters are compared to the safety analysis report values [1] as well as the MCNP4C results [9] respectively.

Table II

Shows the Comparison of the Calculated Results of f_{HR}, f_Z, f_R, and f_T by SRAC-CITATION Code with SAR Values [1] and MCNP4C Results [9] Based on JENDL-3.3 Library

Power Peaking Factors	SRAC-CITATION Code (JENDL-3.3)	SAR* Values	MCNP4C Code* (ENDF/B-VI)
f_{HR}	1.861 (9.47 %)#	1.700	1.854 (9.06 %)#
f_Z	1.219 (-2.48 %)	1.250	1.210 (-3.20 %)
f_R	2.558 (-3.47%)	2.650	2.553 (-3.66 %)
f_T	5.803 (3.06%)	5.631	5.727 (1.71 %)

Error (%) $= \frac{\text{Calculated Value} - \text{Safety Analysis Report(SAR) Value}}{\text{safety analysis report value}} \times 100$ and * Reference

Table III

Shows the Comparison of the Calculated Results of f_{HR}, f_Z, f_R, and f_T by SRAC-CITATION Code with SAR Values [1] and MCNP4C Results [9] Based on ENDF/B-VII.0 Library

Power Peaking Factors	SRAC-CITATION Code (ENDF/B-VII.0)	SAR* Values	MCNP4C Code* (ENDF/B-VI)
f_{HR}	1.853 (9.00 %)#	1.700	1.854 (9.06 %)#
f_Z	1.220 (-2.40 %)	1.250	1.210 (-3.20 %)
f_R	2.563 (-3.28 %)	2.650	2.553 (-3.66 %)
f_T	5.794 (2.50%)	5.631	5.727 (1.71 %)

Error (%) $= \frac{\text{Calculated Value} - \text{Safety Analysis Report(SAR) Value}}{\text{safety analysis report value}} \times 100$ and * Reference

In case of f_{HR}, the calculated value of f_{HR} obtained from the SRAC-CITATION based on JENDL-3.3 library is 9.47 % higher whereas the value obtained from the ENDF/B-VII.0 is 9.0 % higher than the safety analysis report (SAR) value. Both percentage errors lie within the % uncertainty limit while this error is 9.06 % in MCNP4C based on ENDF/B-VI.0 library. In addition, the hot spot was found physically at the fuel position C8 (in the C ring of the TRIGA core) for both libraries with a maximum power density of 1.7014E+02 Watt/cc (JENDL-3.3) and 1.6971E+02 Watt/cc (ENDF/B-VII.0) respectively.

In case of f_Z, the maximum % error is -2.48 % between the said libraries while this value is -3.20 % in MCNP4C. In case of f_R, the maximum % error is -3.47 % between the said libraries while this value is -3.66 % in MCNP4C. In case of f_T, the maximum % error is -3.06 % between the said libraries while this value is -1.71 % in MCNP4C. Hence, the calculated power peaking factors lie within the % error limit (10 %) and the SRAC-CITATION code predicted the power peaking factors reasonably well with the MCNP4C results as well as the safety analysis report (SAR) values.

In addition, the total peaking factor implies that under adiabatic condition the peak heat generation in the fuel element can be almost six times higher than the core average heat generation. It is found that the hot-spot and the peaking factors are function of core configuration and must be calculated from case to case. As it is not clear what was the actual core configuration for which the projected values in the safety analysis (SAR) were calculated it is difficult to make serious comparison. It was observed that $\boldsymbol{f_{HR}}$ is increased in all practical cases if central irradiation channel is used.

V. CONCLUSIONS

This analysis evaluates the power peaking factors of 3 MW TRIGA Mark II research reactor based on the diffusion approach. By comparing the calculated results with safety analysis report (SAR) values [1] as well as earlier published MCNP4C results [9] as numerical benchmarks it was found that the calculated results of power peaking factors show a good agreement between JENDL-3.3 and ENDF/B-VII.0 libraries as well as the SAR values and the MCNP4C results respectively, which indicates that the calculation techniques the SRAC-PIJ code and the SRAC-CITATION code of the comprehensive neutronics calculation code system SRAC2006 are reliable for analysis of power peaking factors of TRIGA Mark-II research reactor. Accurate calculations of power peaking factors are very essential for safe reactor operation as they determine different kinds of safety measures. Therefore, this analysis will be essential to improve the power peaking factors data as core safety parameters data for safe operation of 3 MW TRIGA Mark-II research reactor at AERE, Savar, Dhaka, Bangladesh.

VI. ACKNOWLEDGEMENT

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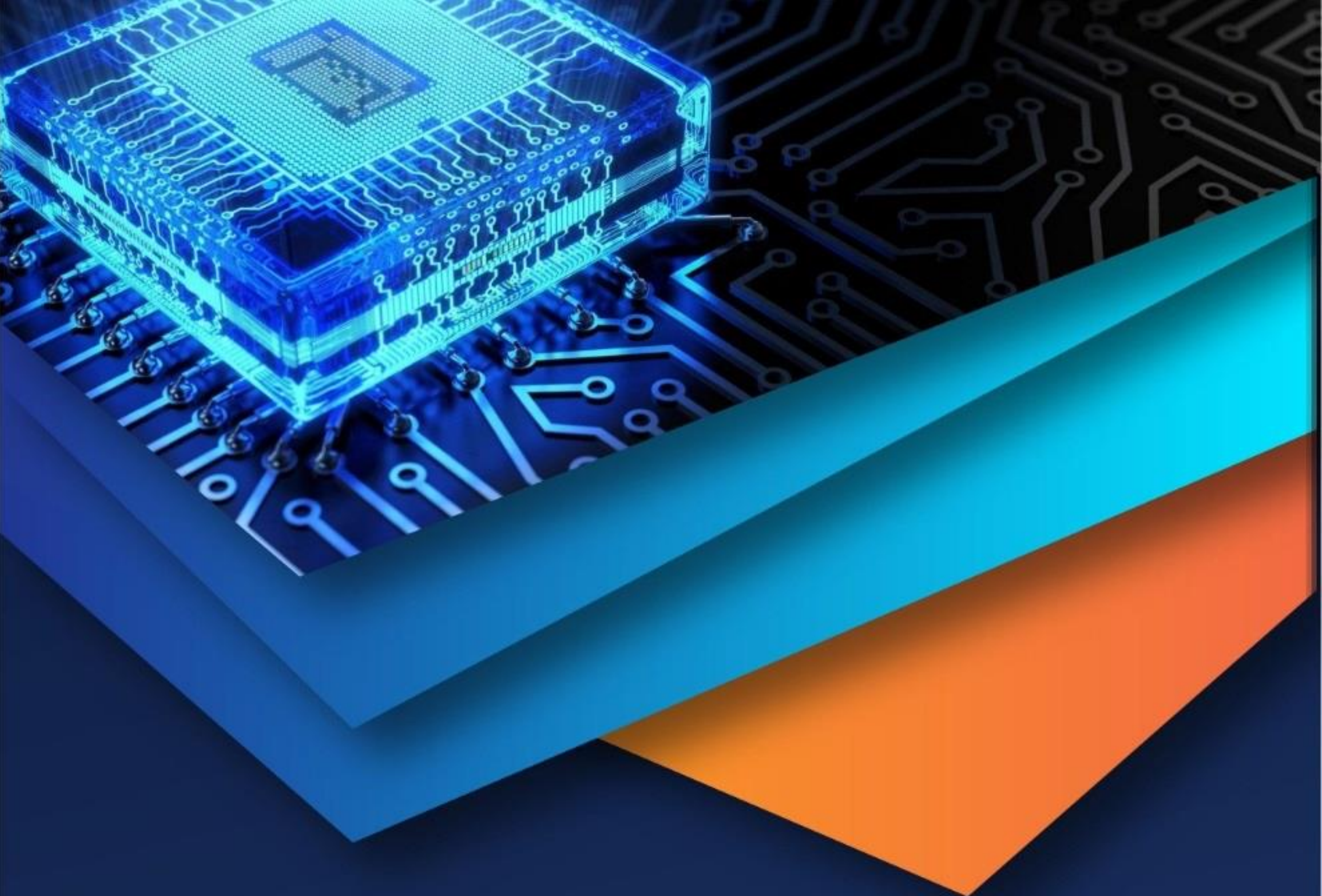

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