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POWER AND NEUTRON FLUX CALCULATION FOR THE PUSPATI TRIGA REACTOR USING MCNP

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ABSTRACT

The Malaysian 1 MW TRIGA MARK II research reactor at Malaysian Nuclear Agency is designed to effectively implement the various fields of basic nuclear research, manpower training, and production of radioisotopes for their use in agriculture, industry, and medicine. This study deals with the calculation of neutron flux and power distribution in PUSPATI TRIGA REACTOR (RTP) 14^{th} core configuration. The 3-D continuous energy Monte Carlo code MCNP was used to develop a versatile and accurate full model of the TRIGA core and fuels. The model represents in detailed all components of the core with literally no physical approximation. Continuous energy cross-section data from the more recent nuclear data as well as S (α, β) thermal neutron scattering functions distributed with the MCNP code were used. Results of calculations are analyzed and discussed.

Keywords: MCNP, neutron distribution, peaking factor, RTP

INTRODUCTION

The Malaysian 1MW PUSPATI TRIGA Reactor (RTP) was designed to effectively implement the various fields of basic nuclear research and education. It incorporates facilities for advanced neutron and gamma radiation studies as well as for isotope production, sample activation, and student training. RTP has reached its first criticality on 28 Jun 1982. It uses standard TRIGA $UZrH_{1.6}$ fuel of 8.5wt%, 12wt% and 20wt% with 20% of U-235 enrichment. It has cylindrical core arrangement and surrounded with graphite reflector and cooled by natural convection. The top and bottom grid plate was from Al-6061 type. The RTP has 4 control rods which are made up of boron carbide. Three of them are from fuel follower type and the other is air follower. The fuel follower control rods (FFCR) was installed with 8.5wt% $UZrH_{1.6}$ and B_4C absorber on top of the fuel section.

The RTP is used mainly for beam experiments, samples analyses, education and trainings. A three-dimensional model of the 14^{th} core configuration was elaborated using the Monte Carlo code MCNP to investigate the in-core neutron flux and power distribution. This information is crucial since it is important neutronics parameters that contributed to the performance and optimum utilization of RTP. Neutron flux and power distribution were expected to be much more dependent on the fuel loading pattern and location in the core. The MNCP code was chosen because of its general modeling capability and continuous energy cross-sections. The latter is the most significant because this eliminates the need for collapsing multi-group cross-sections for reactor calculations.

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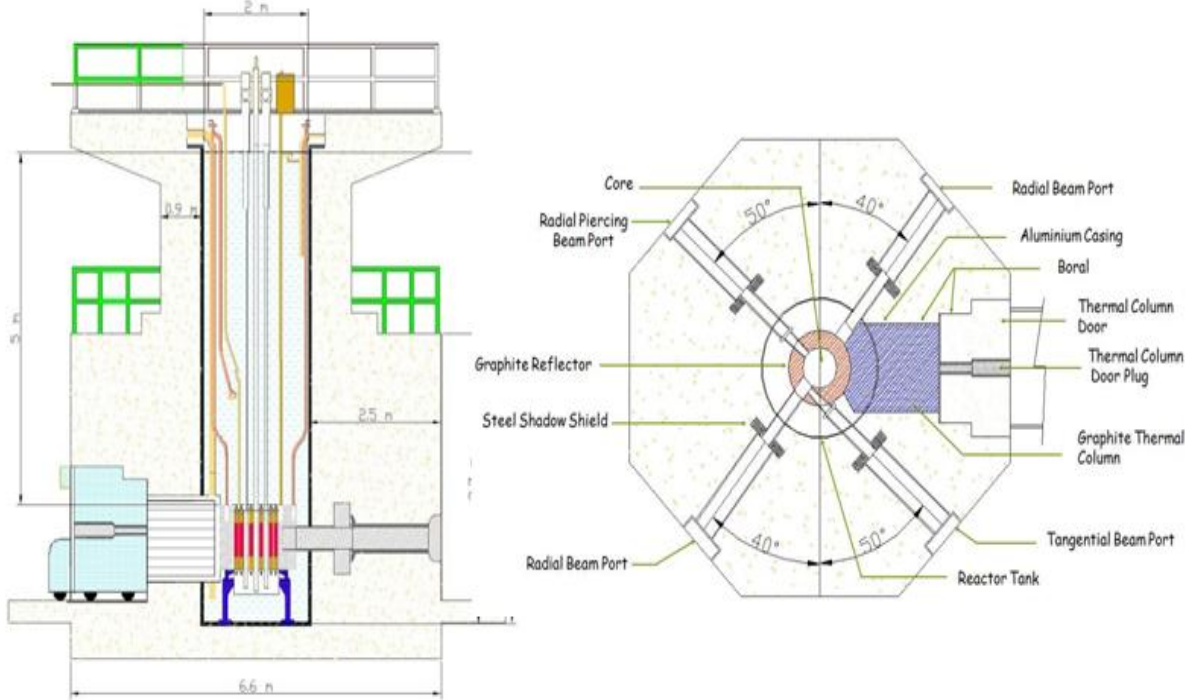

Fig. 1: Side and top view of RTP

MCNP MODEL

In MCNP the most common way to calculate the neutron flux and power distribution in a reactor core is through the use of KCODE option. It is important to note that all the standard MCNP tallies can be made during a criticality calculation. All outputs are normalized to reactor power. The 14^{th} core consists of 112 fuel elements (86 8.5 wt% fuel type, 16 12 wt% fuel type and 10 20 wt% fuel type), 4 control rods, 11 graphite elements and central thimble. The cross-sectional view of the 14^{th} core configuration of the reactor is shown in Fig. 2a and 2b.

Elements are arranged in seven circular rings and the spaces between the fuel rods are filled with water that acts as coolant and moderator. The reactor was modeled in full 3-D detail to minimize the number of approximations. The fuel elements were modeled explicitly specifying the detailed structure of the rod (cladding, air gap, top and bottom graphite, moly disk) to eliminate any homogenization effects. The control rods were explicitly modeled along the active length containing three vertical sections of boron carbide, fuel follower, and void region. The central thimble was considered to be filled with water. The graphite dummy elements are of the same general dimensions and construction as the fuel-moderator elements, except these elements are filled entirely with graphite.

The $S(\alpha, \beta)$ thermal scattering cross-sections of bound nuclei (i.e. H in H_2O, C in graphite, Zr in ZrH, and H in HZr) were taken directly from the standard MCNP cross-section library respecting the operation conditions of the TRIGA reactor. These thermal scattering data are essential to accurately model the neutron interactions at energies below 4eV. The ENDF-VI cross section data set was used to calculate all the isotopes and calculation were run at MCNP default room temperature. For axial and radial neutron flux calculation, tally cells were created and arranged at several locations in the core.

From	:	24-March-2009
Until	:	
Operating Hour	:	
Energy Released	:	
Total FE - 8.5%	=	86
- 12%	=	16
- 20%	=	10

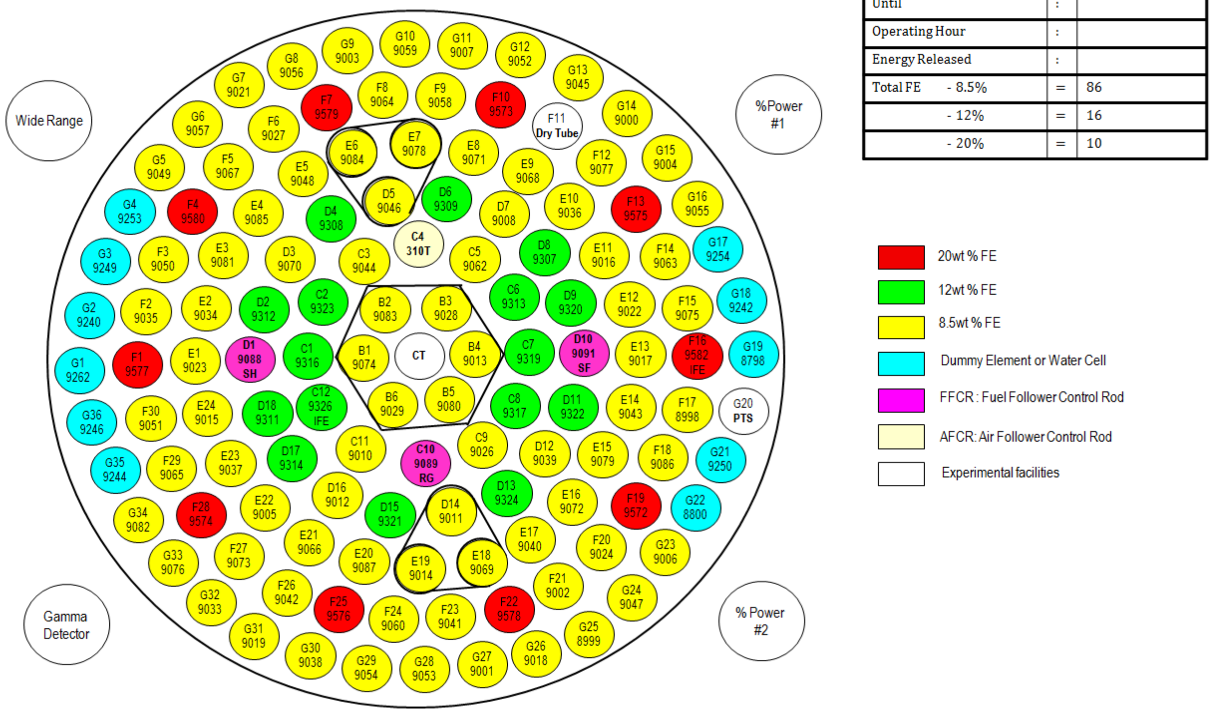

Fig. 2a: RTP Core-14 configuration

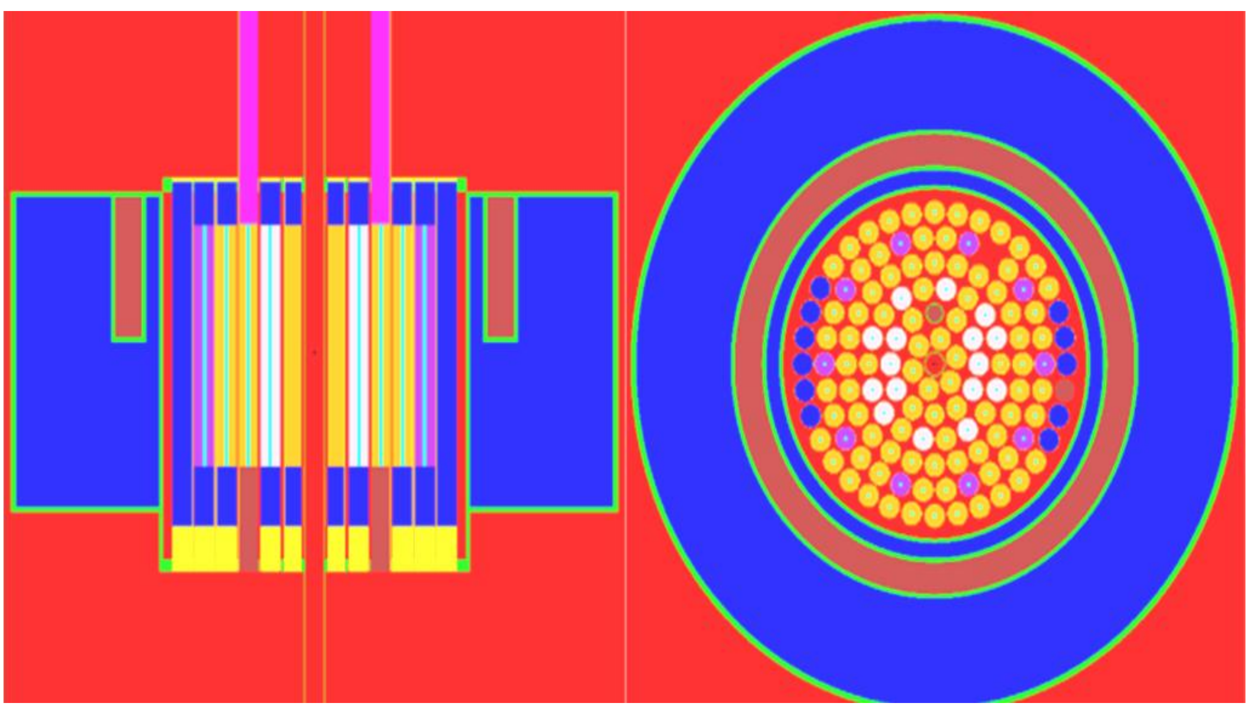

Fig. 2b: Side and top view of MCNP core model for Core-14

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NEUTRON FLUX DISTRIBUTION

Since the MCNP results are normalized to one source neutron, the result has to be properly scaled. The F4 tally result was normalized to reactor power to get the actual flux using the following equation;

$$\Phi_{total} = \Phi_{F4} \times (Power\ Level \times \upsilon) \div (Q_{value} \times k_{eff}) \qquad \text{(Equation 1)}$$

Φ is neutron flux, υ is average neutron emitted per fission and Q is recoverable energy per fission. A small cell (1cm x 1cm) created for F4:N tally for neutron flux calculation with neutron energy range; from 0 to 0.05e-6MeV, 0.14e-6MeV, 0.5e-6MeV, 1.25e-6MeV, 9.1e-3MeV, 1MeV, 10MeV. Thermal neutron energy considered to be all energy below 0.5eV, this is the energy cut-off for Cadmium used in neutron flux measurement, thus to ease the comparison of MCNP calculation with measured one. The k_{eff} predicted by MCNP was 1.12728 ±0.00071 which is higher than the measured one which is 1.091036, this is expected since the MCNP does not taking into account the fuel burnup.

The MCNP predicted thermal flux values seem to be higher than the experimentally determined values. One possible explanation for this behavior is error in physical model or the temperature effect, which was neglected in MCNP calculation – this would result in a different power distribution than during the experiment. The magnitude of this effect is not known but is believed to be small. Even with this consideration, the agreement between the MCNP predicted values and the experimentally determined values is fairly good, see table 1. A big discrepancy for the RR Lower calculation by MCNP may be due to the effect of single cell F4 tally for the lower part of RR, where the thermal flux is averaged over the whole volume region. While the activation foils method for thermal flux measurement only take a very small volume or nearly point like detector in the RR Lower.

Table 1: Thermal neutron flux in some irradiation facility at 750kW reactor power for Core-14

	Central Thimble **$\times 10^{13}$ neutron/cm^2.s**	**PTS** **$\times 10^{12}$ neutron/cm^2.s**	**RR Lower** **$\times 10^{12}$ neutron/cm^2.s**
MCNP	1.897 ± 0.019	5.025 ± 0.029	2.948 ± 0.012
Experiment	1.777 ± 0.031	4.813 ± 2.447	1.494 ± 0.196

The neutron flux distribution inside the core was found to be dependent on the different elements in it. In a typical TRIGA reactor it is expected that, maximum neutron flux will be distributed towards core center, see fig. 3. Maximum thermal neutron flux was found at the Central Thimble location. Since this location was filled with water moderator, fast neutron are thermalized, thus increases the thermal neutron flux in it. This is the reason why the fast neutron radial distribution drop at the center of the core, see fig. 4. At the fuel region from B ring to G ring, fast neutron flux relatively exceeds the thermal flux due to more thermal absorption in fuel.

At the edge of the core the thermal flux relatively higher than the fast flux because of the graphite reflector effect. Not only the thermal neutron reflected back to the core, but thermalization of fast neutrons was also taking place inside the reflector. This will also cause the flattening of the thermal neutron flux distribution see Fig. 3 and Fig. 4.

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From the axial point of view, at the Central Thimble location, as expected, the maximum flux was at the center or middle plane of the active fuel core region, see Fig.5. Flux distribution decreased symmetrically towards the upside and downside of the Central Thimble. Axial flux distribution at the fuel region, see Fig. 6, was a little bit different from how its look like at the Central Thimble where at the top and bottom of the fuel, the thermal neutron flux distribution tend to swelled. This is believed to be the effect of graphite reflector at the top and bottom of the fuel rod. Whole core thermal and fast flux distribution is shown in Fig.7 and Fig.8.

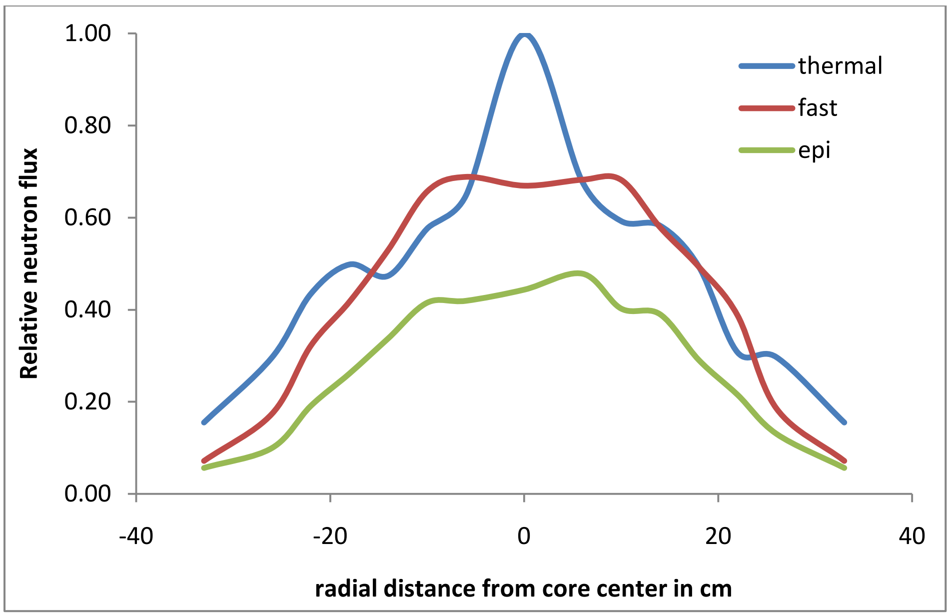

Fig. 3: Radial flux distribution

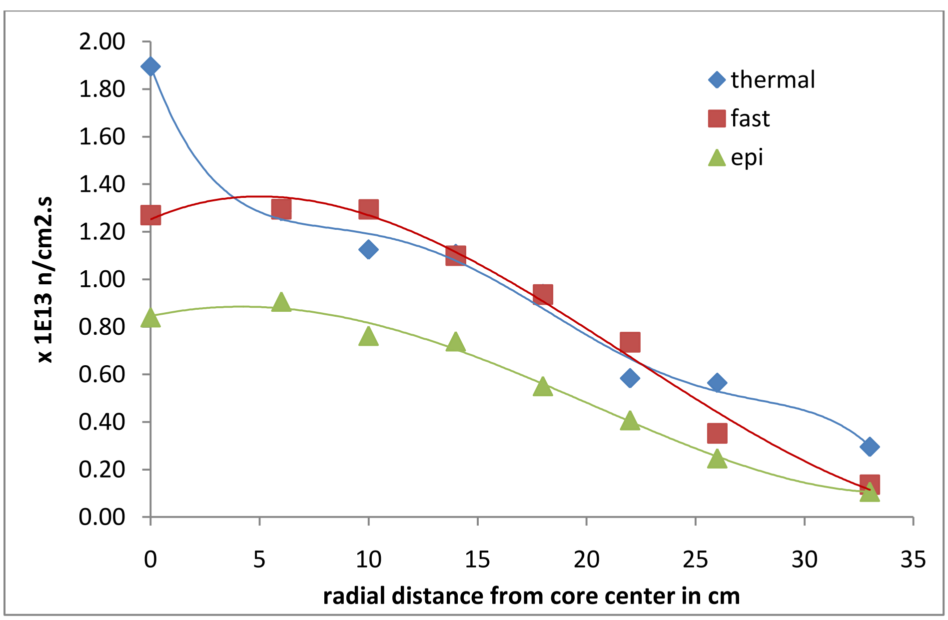

Fig. 4: Neutron flux value at 750kW

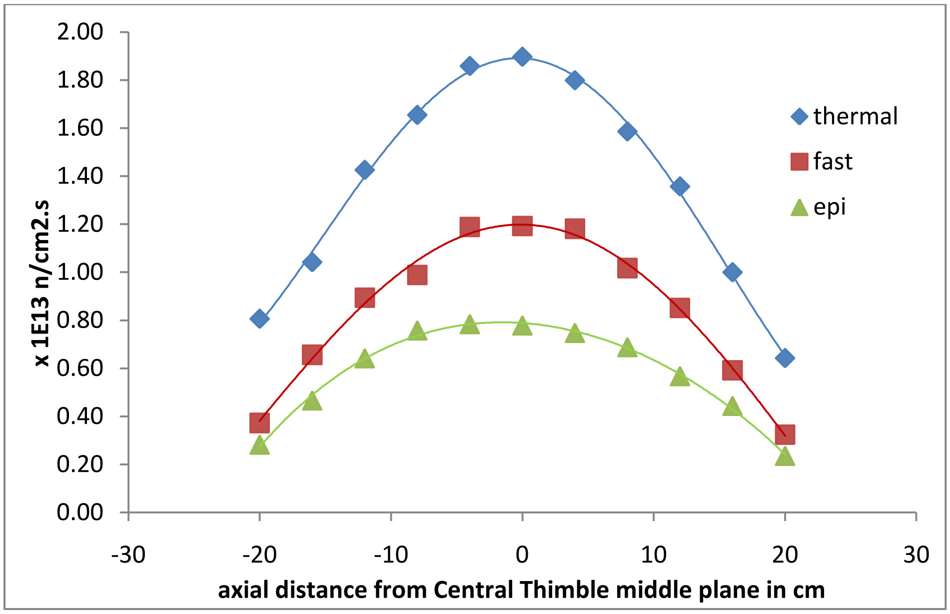

Fig. 5: Neutron flux value in Central Thimble at 750kW

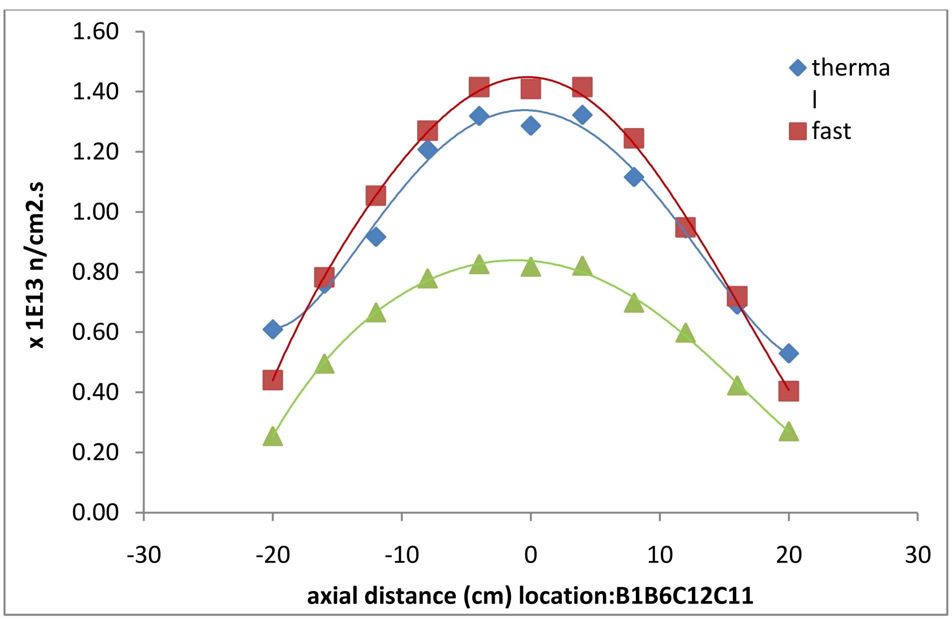

Fig. 6: Neutron flux value between fuel locations at 750kW

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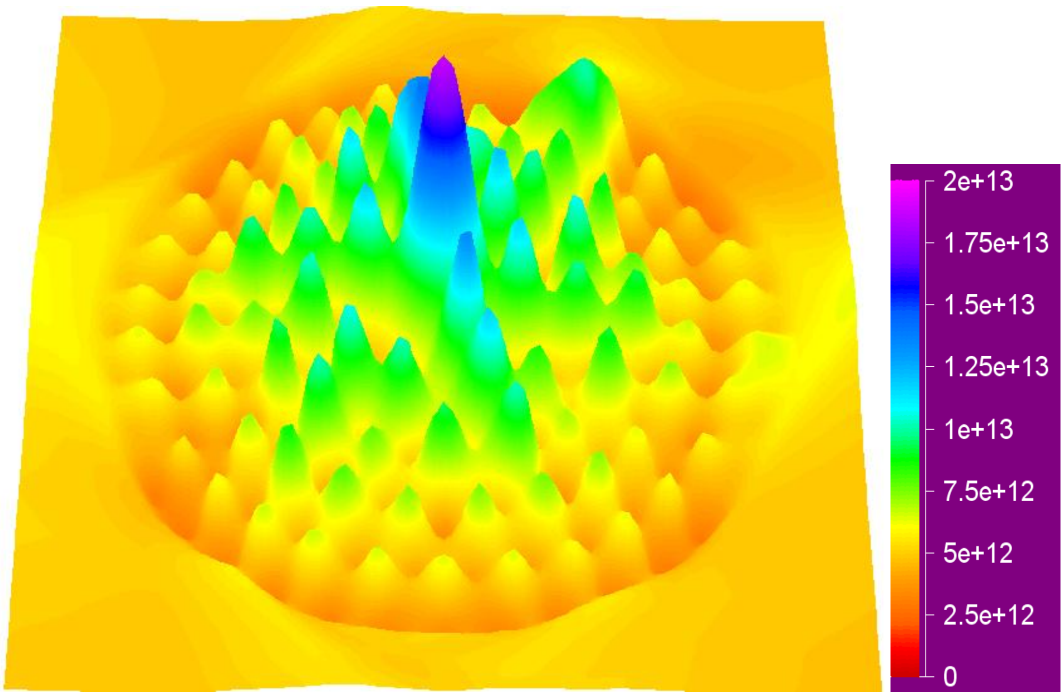

Fig. 7: Thermal neutron flux distribution ($n.cm^{-2}.s^{-1}$)

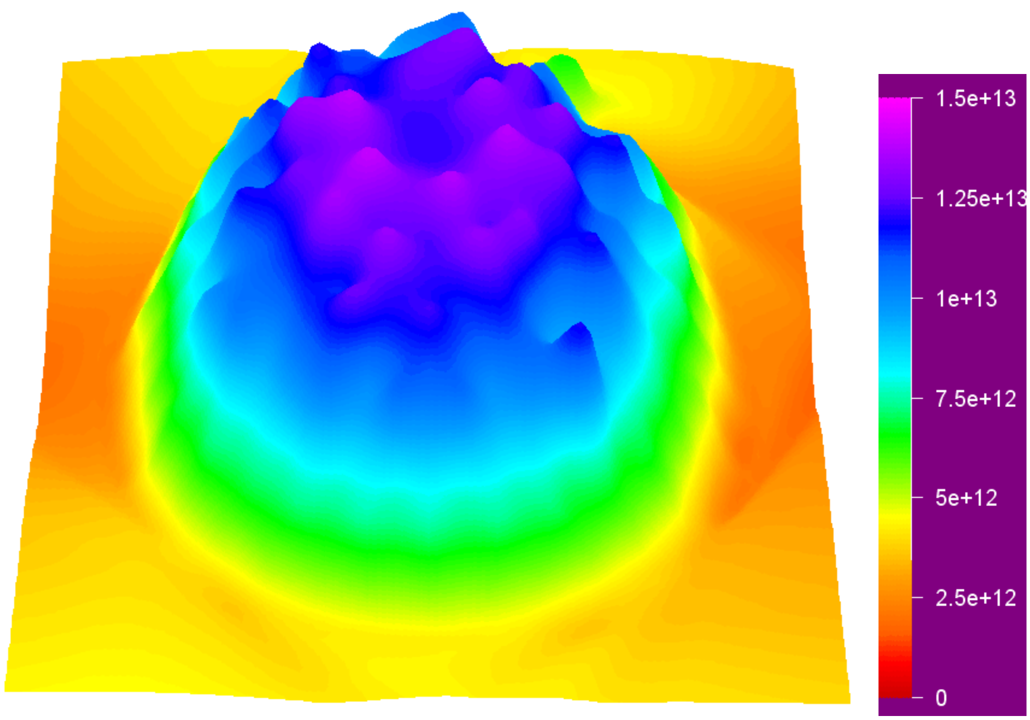

Fig. 8: Fast neutron flux distribution ($n.cm^{-2}.s^{-1}$)

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POWER DISTRIBUTION

Power distribution is better explained in power peaking factor. The elaborated MCNP model of the TRIGA reactor was used for the calculation of three power peaking parameters; (a) Hot rod power peaking factor f_{HR}, (b) Axial power peaking factor f_Z and (c) Radial power peaking factor f_R. These three factors are important for steady state operation; they determine the maximum total power released by one fuel element as well as its axial and radial peaking values, which are used as parameters in thermal-hydraulic analysis. The hot rod power peaking factor is defined as the ratio between the maximum power released by one fuel rod P_{Rod} and the average power per element in the core, P_{core}

$$f_{HR} = (P_{rod})_{max} / P_{core} \qquad \text{(Equation 2)}$$

and

$$P_{core} = P / N_{EL} \qquad \text{(Equation 3)}$$

Where P is the total power, which is 1 MW for Core-14. The term N_{EL} is the number of fuel elements in Core-14 configuration, which is considered to be 112 (109 fuel elements+3 fuel follower control rods). The axial power peaking factor f_Z is defined as peak $p(z)$ to average P_z axial power density in the fuel element.

$$f_z = p(z)_{max} / P_z \qquad \text{(Equation 4)}$$

The radial power peaking factor f_R is defined as peak $p(r)$ to average P_r radial power density in the core:

$$f_r = p(r)_{max} / P_r \qquad \text{(Equation 5)}$$

In this calculation the radial power peaking factor f_R is the same with the hot rod power peaking factor, because radial power density $p(r)$ considered is only average for the whole core not specifically inside the fuel element. It is assumed that the power density is directly proportional to the fission density. The axial power peaking factor f_Z was calculated by dividing axially the fuel element rod into 11 volumes and calculating the fission density integrated over each of the volumes; then the axial power peaking factor was calculated by dividing the maximum axial fission density to the average axial fission density according to Eq.4. F7 tally were used to calculate fission power in each volume. In order to obtain the core power distribution, power released in every fuel meat volume was also calculated using F7 tally.

Fig. 9 gives the maximum hot rod and Fig.10 shows detail power distribution calculated using MCNP for fuels and fuel follower elements in each ring. The maximum power produced in the hottest fuel element is found to be 12.56 kW in C-ring (C-12 12 wt% fuel type) and the calculated hot rod power peaking factors are found to be 1.41.

In general, typical TRIGA reactor having the characteristic where the maximum power distributed towards the center of the core. In a mixed core like the RTP Core-14, hot rod peaking factor were also depend on the fuel type and location. In F ring, there were 20wt% fuel types and some location in C ring filled with 12wt% fuel elements; that is why we can see power peaked at these locations. Non-fuel element like the graphite rods dose not contribute in fission power but the prompt-gamma and neutron heating to these elements are considered in MCNP although the magnitude were rather small compare to fuel elements.

Fig.11 shows the axial power distribution within the fuel meat of the fuel elements in ring B to ring F. This figure shows an almost analytical chopped cosine shape of axial power distribution with a peak-to-average value of ~1.3 obtained at the middle point of the fuel elements, which is equivalent also to the axial mid-plane of the core. This value represent typical characteristic of TRIGA fuel in core.

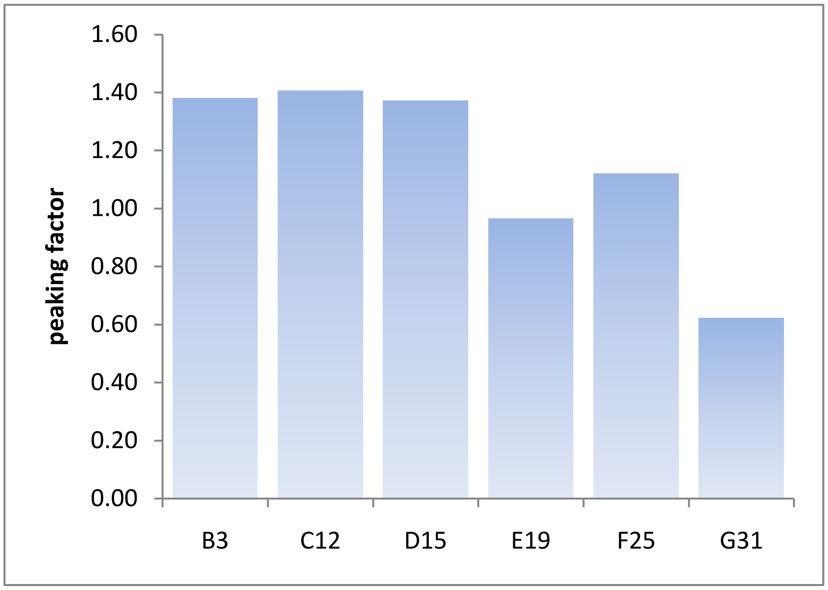

Fig. 9: MCNP maximum hot rod power peaking factor of each ring for Core-14

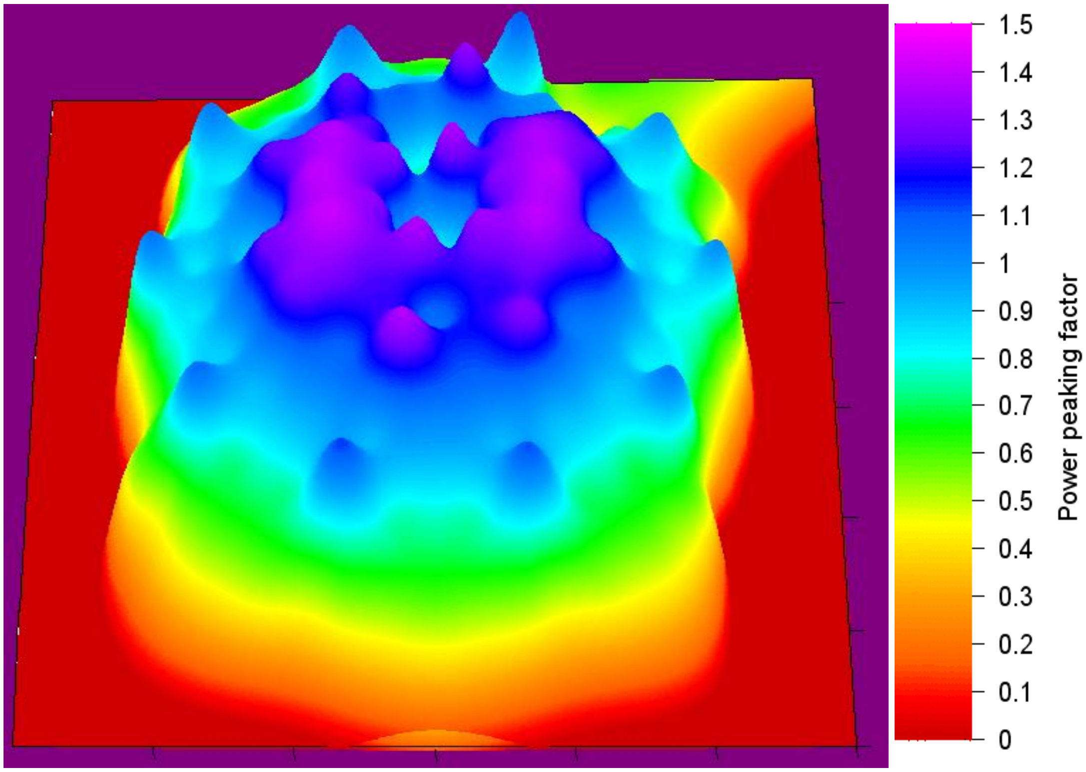

Fig. 10: MCNP calculated power distribution for Core-14

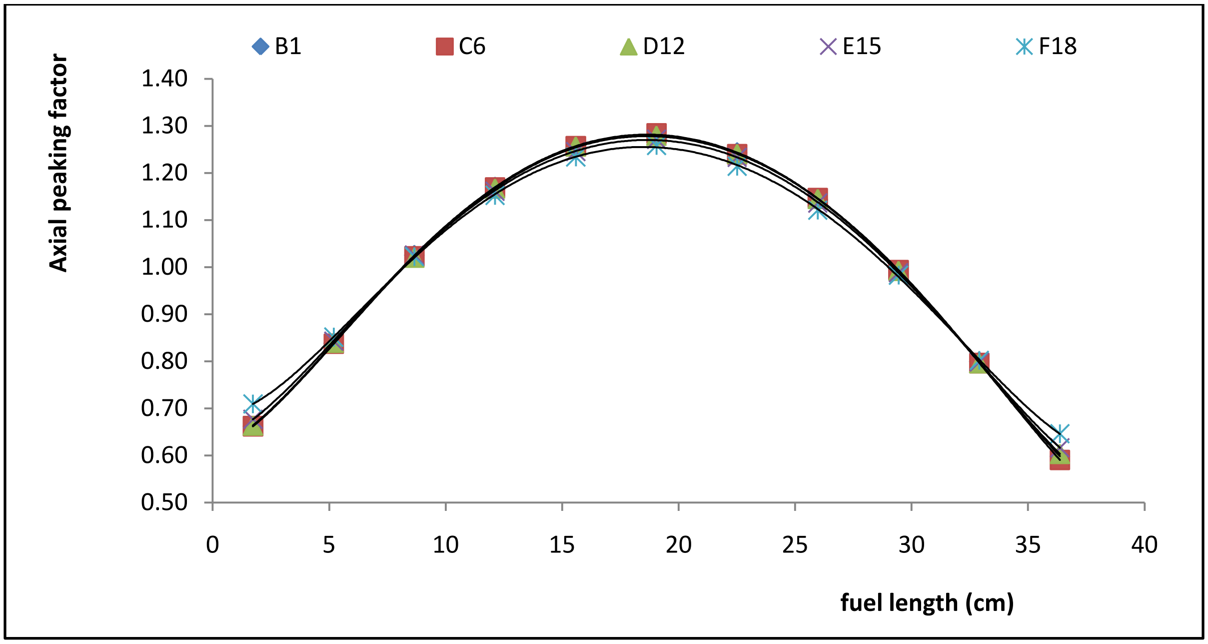

Fig. 11: MCNP calculated axial power peaking factor of fuel elements for Core-14

It can be summarized that the RTP core-14 configuration produce radial power peaking factor below 1.6 and the maximum power for fuel does not exceed 22kw which is both are among the design requirements and operational limitations.

CONCLUSION

The evaluation of neutron flux and power distribution in RTP 14th core configuration was performed by a three-dimensional continuous energy Monte Carlo code MCNP. Radial and axial neutron flux distribution was much more depends on the fuel loading pattern and location in the core. This result also shows the reliability of MCNP code both for design verification and reactor core parameters calculation.

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