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NEUTRON FLUX CHARACTERISATION OF IRRADIATION FACILITIES IN RTP

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ABSTRACT

The Malaysian's PUSPATI TRIGA Reactor (RTP) achieved its initial criticality on June 28, 1982. The reactor is designed to effectively implement various fields of basic nuclear research, manpower training, and production of radioisotopes. Several past activities on neutronics modelling development and validation of the RTP were carried out using Monte Carlo Code MCNP. In this work, the developed model was used to characterise in-core and beam-ports irradiation facilities of the reactor. The thermal and fast neutron flux distributions in these facilities were determined using MCNP mesh tally method. It was found that the flux as well as its spectral characteristics depended very much on the position of the irradiation facility in the reactor core or in the beam-ports. The maximum neutron flux was found to be in the Central Thimble facility with 1.98E13 nv of thermal neutron. The thermal-to-total flux ratio varies significantly from 0.41 for the in-core facility, 0.58 in the reflector and up to 0.88 in the beam-ports.

Keywords: Fuel elements, neutron flux, MCNP, RTP, TRIGA

INTRODUCTION

Determination of neutron flux characteristics in irradiation facility is important to support various experiments that are performed in a research reactor. So far, measurements technique was used to determine neutron flux in irradiation facilities of the reactor core of Malaysian PUSPATI TRIGA (RTP). In addition to these measurements, transport calculations can provide useful information of irradiation field characteristics (Jeraj et al., 2001) in the reactor. Several codes have been used to perform neutronics analysis of the RTP core including 2D diffusion code, TRIGLAV. However, due to the complexity in geometry and configuration of reactor, only Monte Carlo transport calculations are accurate enough for reliable characterisation of the irradiation facilities. The Monte Carlo transport calculations have also several advantages over the measurements since they enable a very fine spectral description (not limited to only few groups) (Jeraj et al., 2001).

Neutron flux (Φ) is defined as, $\boldsymbol{\Phi} = \boldsymbol{nv}$, where: $\boldsymbol{\Phi}$ = neutron flux (neutrons cm^{-2} s^{-1}), $\boldsymbol{n}$ = neutron density (neutrons cm^{-3}), and $\boldsymbol{v}$ = neutron velocity (cm s^{-1}). Neutrons are generally classified according to their energy. In this work, neutrons with energies less than 1.0 eV are classified as thermal neutrons. Neutrons with energies between 1.0 eV and 0.1 MeV are classified as epi-thermal neutrons. Neutrons with energies greater than 0.1 MeV are classified as fast neutrons. All neutrons, at the time of their birth (from a fission event) are fast. Fast neutrons lose their energy by colliding elastically with atoms within their environment. These neutrons eventually lose kinetic energy to a point where they are in thermal equilibrium with the surrounding gas molecules and are then considered to be "thermalized." Once "thermalized," these neutrons have a high likelihood of being captured by absorbing nuclei. If the neutrons are absorbed by the reactor fuel, additional fission events shall occur (Ashbaker, 2005). The evaluation of neutron flux and energy spectrum in irradiation facility is essential for determination of its irradiation characteristics.

REACTOR CORE DESCRIPTION

RTP has reached its first criticality on 28 Jun 1982. It loaded standard TRIGA $UZrH_{1.6}$ fuels with 8.5 wt %, 12 wt % and 20 wt % content of uranium fuel. Enrichment of ^{235}U in uranium fuel is 19.9 %. It is of cylindrical core surrounded with graphite reflector and cooled by natural convection. Both top grid plate and bottom grid plate are made of Al-6061. RTP has 4 control rods which are made of boron carbide. Three of them are of fuel follower type and the other is of air follower type. The fuel follower control rods (FFCR) consist of 8.5 wt % of uranium and B_4C absorber on top part of the fuel section. The reactor utilizes hydride fuel, made of homogeneous mixture of uranium and zirconium hydride ($UZrH_{1.6}$), specification of the fuels is given in Table 1. Fuel elements are arranged in seven circular rings in the core and the spaces between the fuel elements are filled with water that functions as coolant and moderator (Rabir et al., 2017). The fuels arrangement is shown in Figure 1. They are several irradiation facilities in the core and outside the core including four beam ports and the Thermal Column. Among the in-core facilities are Dry Tube, Central Thimble, Delayed Neutron Activation and the Rotary Rack. Of the four existing beam ports, two are already used for Small Angle Neutron Scattering (SANS) and neutron radiography (NR). In this work, in-core and beam-ports neutron fluxes were determined using Monte Carlo Code, MCNP, to model the RTP core. Detail on the MCNP model development and validation of the core can be found in reference (Rabir et al., 2016a; Rabir et al., 2016b; Rabir et al., 2016c; Rabir et al., 2017).

Table 1: Details specification of TRIGA fuel elements and control rods

	Fuel Element			**Fuel Follower Control Rod**
Geometrical data				
Outer radius of Zr rod (cm)	0.3175			0.3175
Outer radius of fuel (cm)	1.765			1.665
Air gap thickness (cm)	0.05			0.05
Cladding thickness (cm)	0.05			0.05
Fuel composition				
Uranium (wt. %)	8.5	12	20	8.5
Enrichment (wt. %)	19.9			19.7
H:Zr ratio	1.6			1.6
Absorber				B_4C

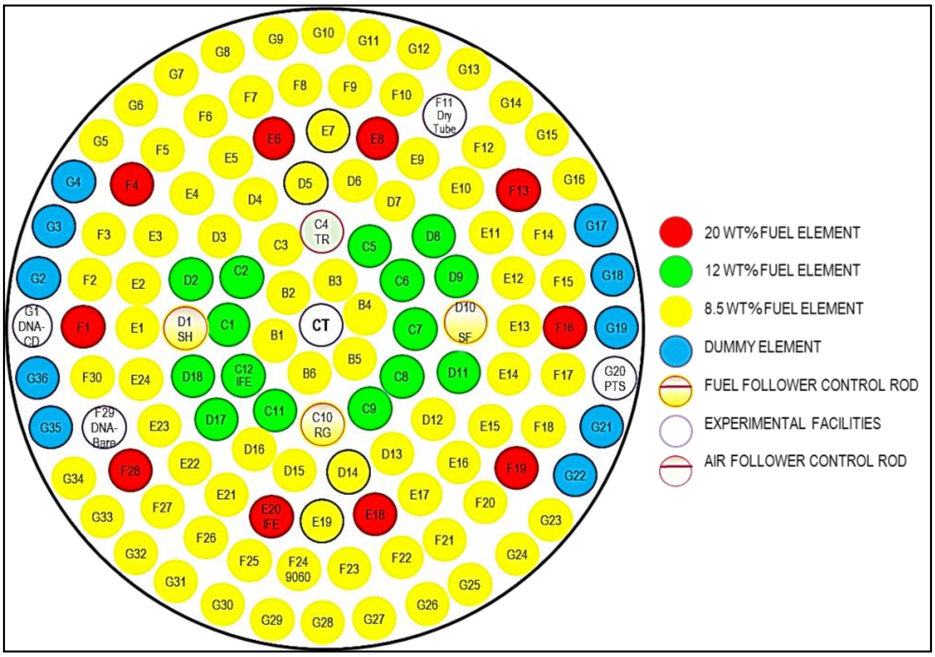

Figure 1: RTP core configuration

MATERIALS AND METHODS

The calculations reported in this paper were performed using MCNPX version 2.70 with the ENDF/B-VII.0 cross-section library. The MCNP calculations were performed in the ‘‘kcode’’ mode. MCNP model for core configuration Core-15 is shown in Figure 2 and Figure 3. The simulations were carried out at reactor thermal power of 750 KW. Calculation of neutron flux within radial distribution of the core was done by using the superimposed mesh tally card, which allows the user to define a mesh tally superimposed over the problem geometry. The flux distribution was calculated in the following approach; numbers of meshes are 600 × 600 × 1 cells, in which the calculated neutron flux was superimposed over the reactor core. Geometry of each cell is (length × width × height) 0.1 cm × 0.1 cm × 10.0 cm, so that the mesh results were averaged over the 10.0 cm of fuel height at the middle plan of the active core (where axial peaking factor are maximum) (Rabir et al., 2016a). The axial map of thermal neutron flux is also calculated (from the bottom of the reactor tank to around 2 meters above the core) but with larger mesh cell of 5.0 cm x 5.0 cm x 5.0 cm.

The neutron flux values, utilizing 111 energy groups, were also calculated by means of the track length estimator, or in MCNP terminology, the F4 tally, were implemented in several in-core irradiation facilities including the Rotary Rack and beam-ports. With F4 tally, the values of neutron fluxes are averaged over the selected cell volume. The mesh tally cell volume is relatively smaller than F4 tally volume; so that the results of neutron flux using mesh tally is expected to be a slight higher. F2 tally was also used for calculation of surface flux inside neutron beam-port number 1, 2, 3 and 4 (BP1, BP2, BP3 and BP4). Tallies for neutron surface flux at every 5 cm distance inside the beam ports from the reflector are shown in Figure 3.

The MCNP model of the core used in this work considered the individual fuel burn-up as explained in reference (Rabir et al., 2017), thus the results are updated where as in reference (Rabir et al., 2016a) where the burn-up of fuel was neglected. The thermal-to-total flux ratio is the parameter used in this paper so that the thermal neutron population and thermalisation of fast neutron in the irradiation facilities can be determined.

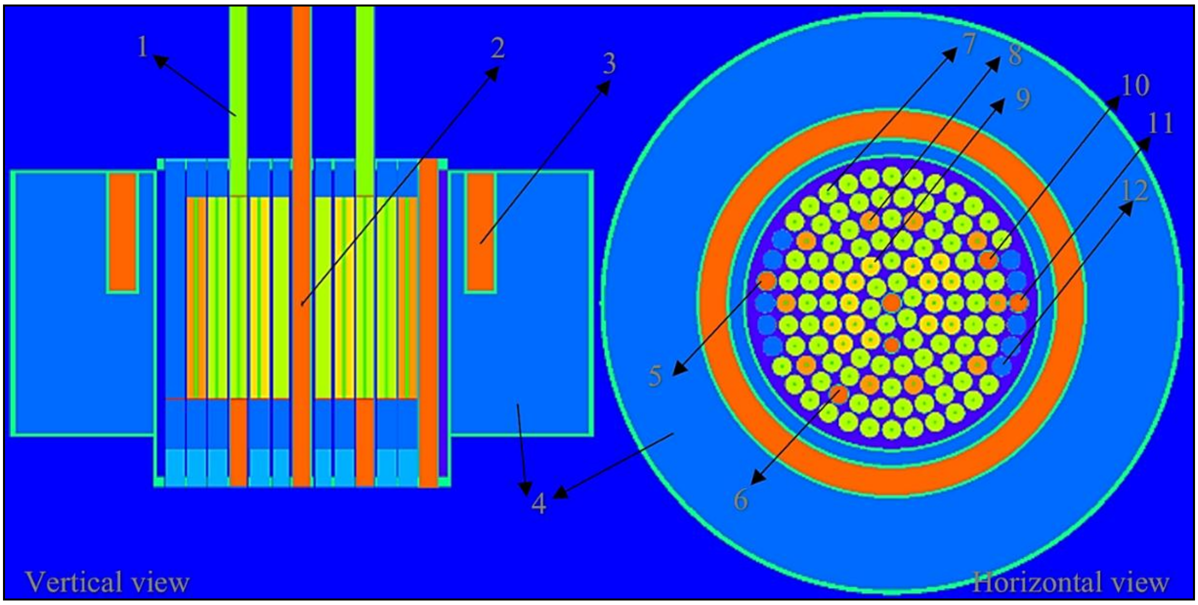

Figure 2: Cross-sectional view of RTP Core-15: 1- Fuel follower control rod, 2- Central Thimble, 3- Rotary Rack (RR), 4- Graphite reflector, 5- Pneumatic Transfer System (PTS), 6- Dry Tube, 7- Fuel rod(8.5 U wt.%), 8- Fuel rod(12 U wt.%), 9- Fuel rod (20 U wt.%) , 10- DNA-Bare, 12- Dummy fuel (graphite rod)

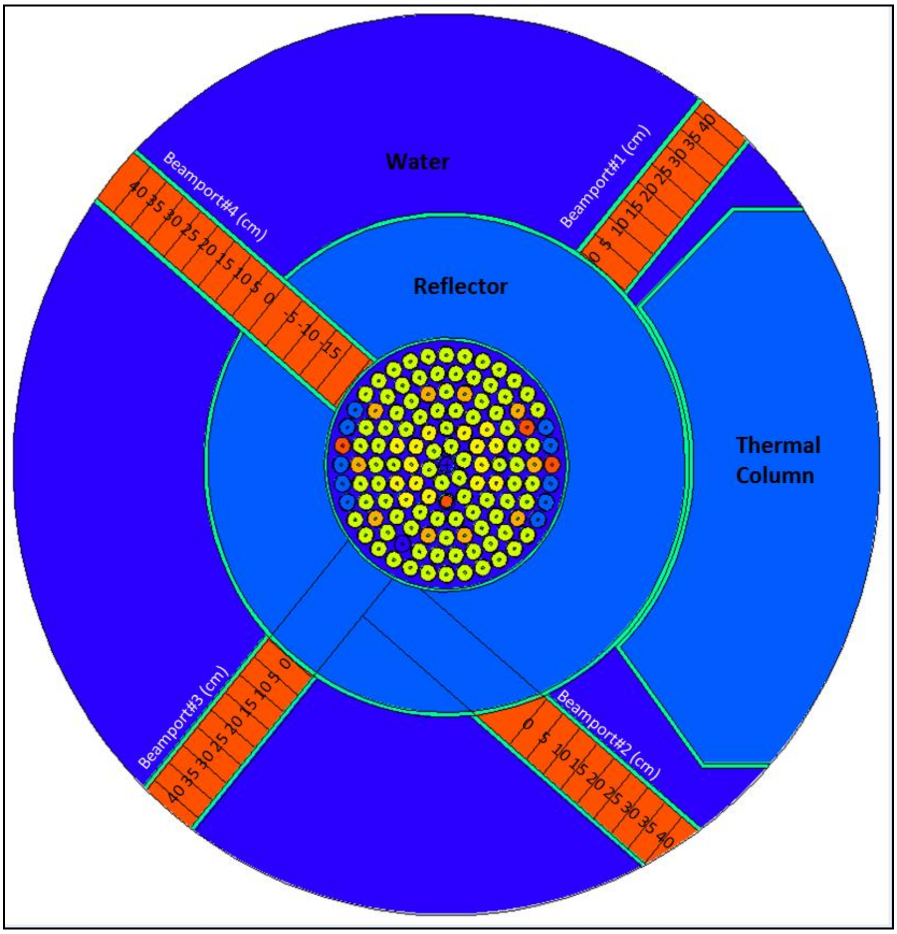

Figure 3: Cross-sectional view of RTP Core-15 MCNP model showing the beam-ports, thermal Column and graphite reflector. Beam-ports are sectioned into smaller volume each with 5 cm length for surface flux tally calculation

Since the MCNP results are normalized to one source neutron, for steady-state thermal power of a critical system, the following scaling factor in units of fission neutrons per unit time (total neutrons, N_{tot}) as shown in equation (1), was used (Rabir et al., 2016a):

$$N_{tot}[n/s] = \frac{\left(P_{core}[watt] \times \nu[\frac{n}{fission}]\right)}{1.6E-13\left[watt.\frac{s}{MeV}\right] \times 200[\frac{MeV}{fission}]} \qquad (1)$$

P_{core} is reactor power in watt and ν is number of neutron emitted per fission. MCNP tally results were normalized to reactor power using equation (2):

$$\Phi_{[\frac{n}{cm^2.s}]} = \phi[\frac{\#}{cm^2}] \times (N_{tot}) \times \left(\frac{1}{k\_eff}\right) \qquad (2)$$

Where, ϕ is the flux tally output.

RESULTS AND DISCUSSION

Neutron Flux Distribution

Radial distribution of thermal neutron flux for RTP Core-15 configuration is presented in Figure 4. Maximum thermal neutron flux was found at the centre of the core in the Central Thimble irradiation facility. Variation in local flux is due to the heterogeneities in the core such as water gaps and differences in uranium concentration. Large depressions of thermal neutron flux inside the fuel elements at several locations in the F ring are clearly seen as they were filled with 20 wt. %. Thermal fluxes peaking within the area of water gaps between the fuel elements are also recognizable. This is due to thermalisation of fast neutrons within the water region between the fuels. High fast neutron flux within the fuel, as shown in Figure 5, is before thermalisation process, while the reduction of thermal neutron is due to absorption by the fuel. It can be observed that the peaking is extremely localized within water gaps and different fuel elements influence only its first neighbors (Snoj and Ravnik, 2008).

At the edge of the core, not only the thermal neutron is reflected back to the core, but thermalisation of fast neutrons was also taking place inside the reflector. This will cause the flattening of the thermal neutron flux distribution. Figure 6 shows the axial neutron flux distribution of the core. The calculation of axial neutron flux distribution started from the bottom of RTP tank up to around 2 meters above the core. Thermal neutron flux at a few centimeters from the reflector was around 10^{11} *nv* and it reduces to around 10^9 *nv* at 20 cm distance outside the reflector. Within the coolant regions, at 1 meter above the core, the thermal neutron flux calculated is around 10^6 *nv*.

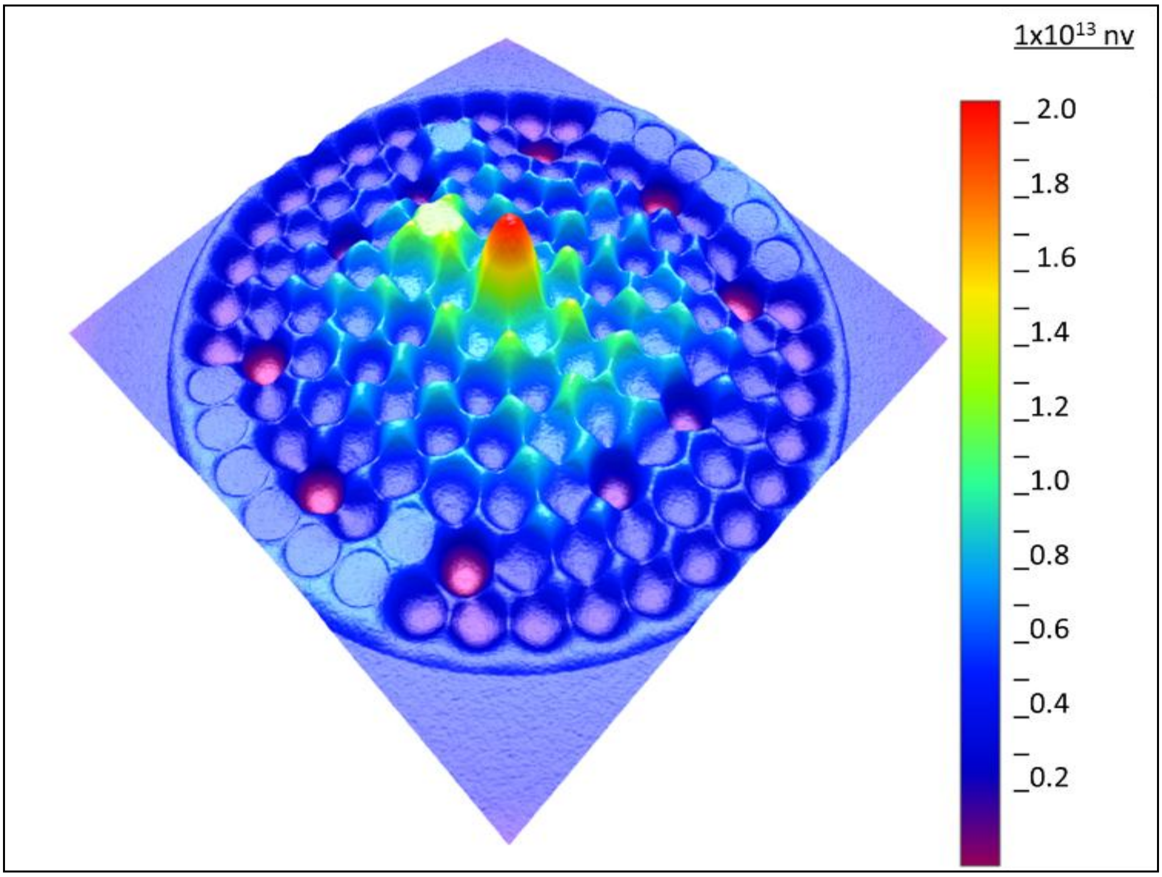

Figure 4: In-core radial thermal neutron flux distribution at 750 kW thermal powers

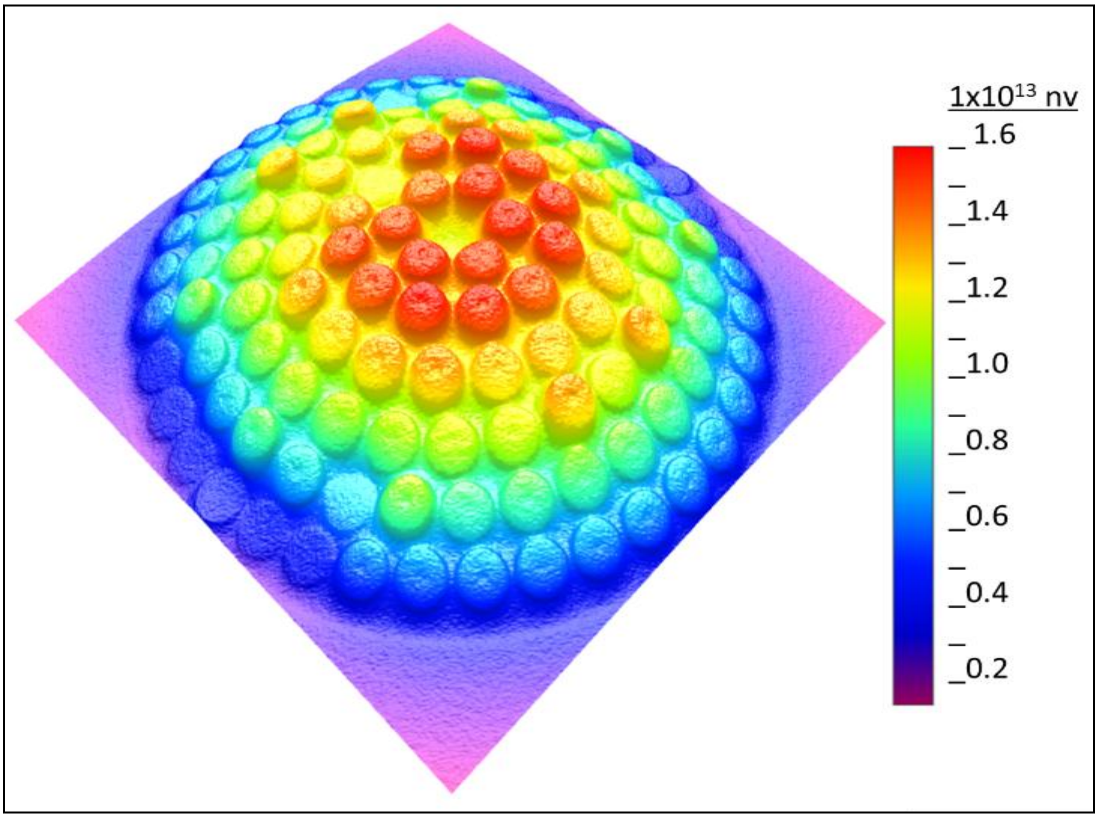

Figure 5: In-core radial fast neutron flux distribution at 750 kW thermal powers

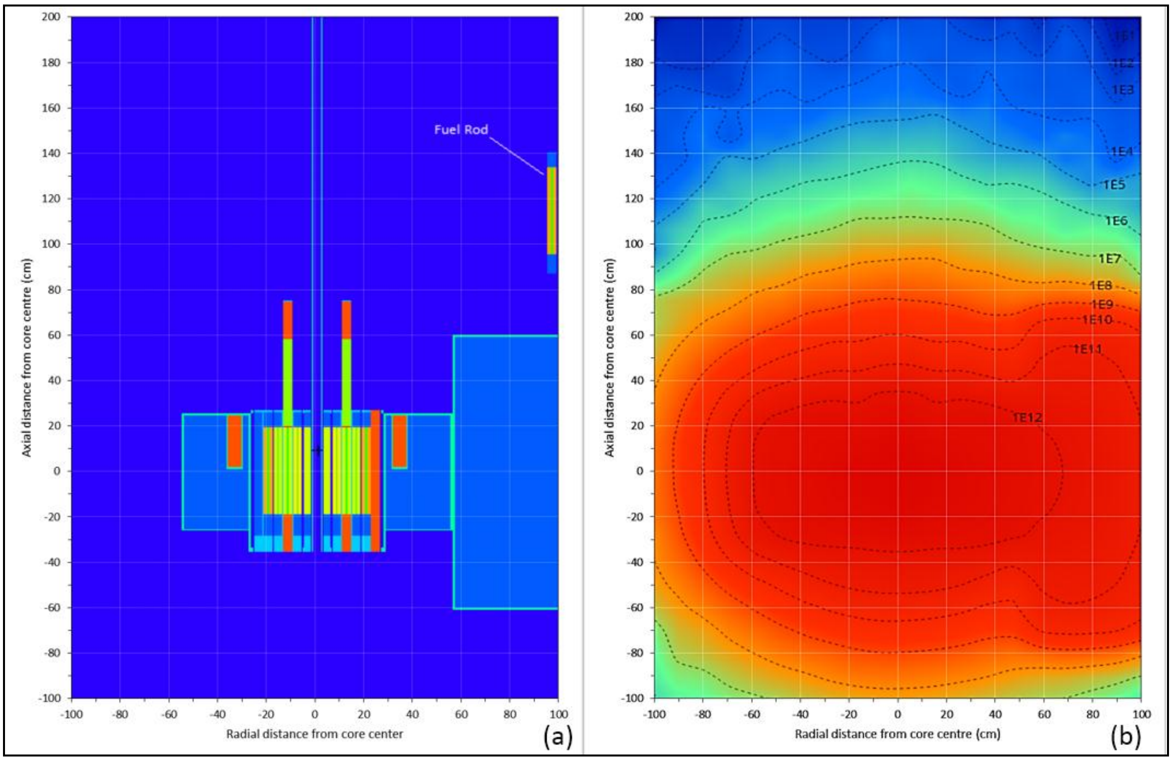

Figure 6: (a). Geometry of RTP core model consisting of reactor tank contains the core, fuel rods, fuels racks inside the water coolant moderator used in the simulation of axial thermal neutron flux distribution at 750 kW thermal power. (b) The axial distribution of thermal neutrons flux

Neutron Flux in In-Core Irradiation Facilities

Calculated neutron flux spectrum for in-core irradiation facilities is presented in Figure 7. Table 2 presents the calculated thermal, epithermal and fast neutron fluxes for in-core irradiation facilities. Thermal neutrons are in the 1/*v* region (typically neutron below 1 eV) where absorption cross-section increases as the neutron energy decreases. Nearly most of the facilities are favored due to their thermal neutron flux. Higher thermal neutron flux means increased reaction rate and lower irradiation time and thus, more efficient exposure and analysis can be done.

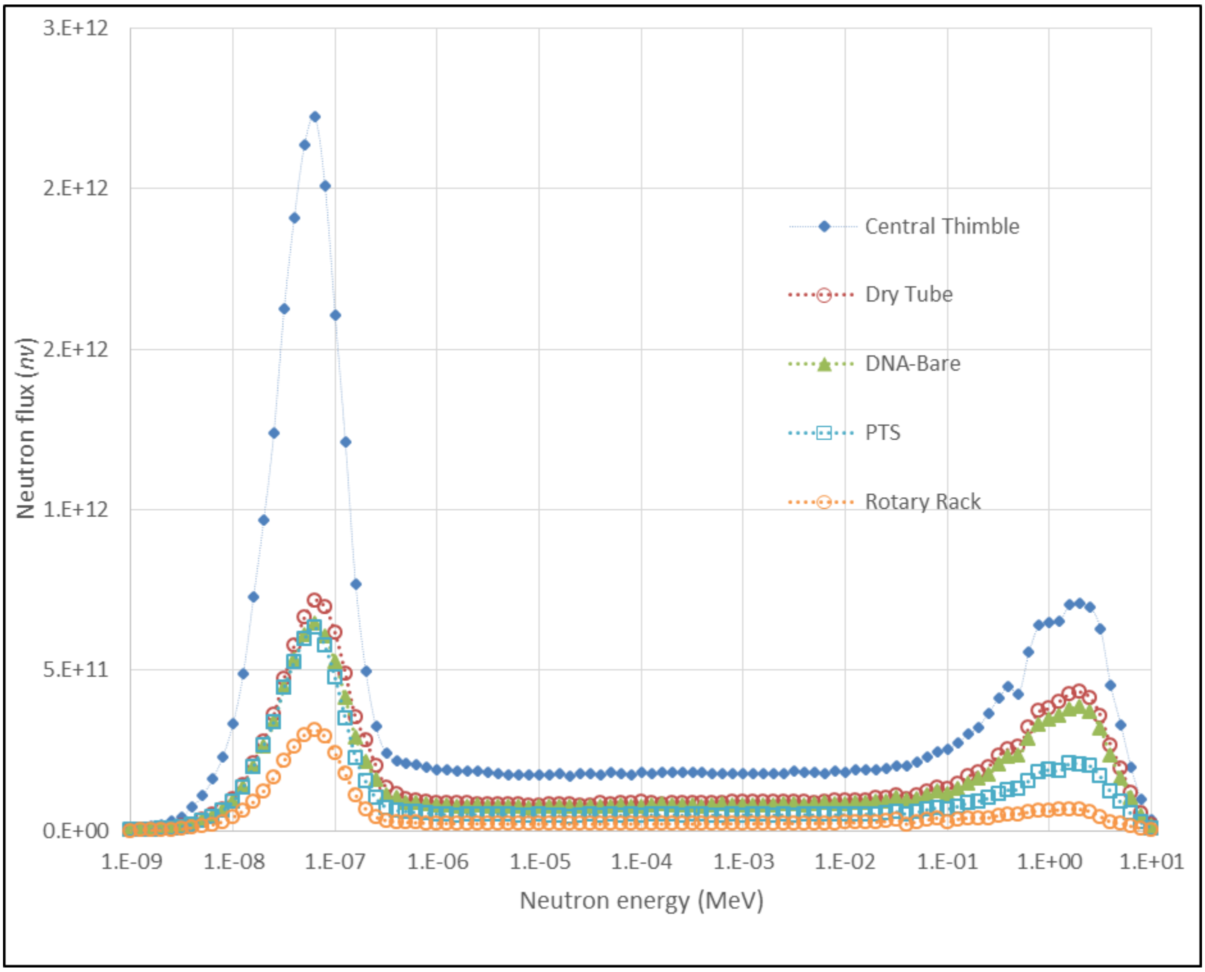

Figure 7: Neutron spectrum in-core irradiation facilities at 750 kW thermal powers

The maximum thermal neutron flux is in the Central Thimble and reduces to only 13.9% in the Rotary Rack, but with higher thermal-to-total flux ratio about 0.58 as is in the Rotary Rack. This is due to the thermalisation process with no production of fast neutron inside the graphite reflector. In the core, while fast neutrons are slowing down in the water region, it is continuously being produced by the fission reactions inside the fuel. PTS facility is surrounded by graphite and Central Thimble is filled with water as moderator, while Dry Tube and DNA-Bare are empty tubes, containing air only. Therefore, less moderation process in the latter in comparison to Central Thimble and PTS, so that the fast neutron fluxes are higher in these facilities.

Table 2: In-core neutron flux at 750 kW thermal powers

Neutron Flux (*nv*)	**Central Thimble**	**Dry Tube**	**DNA-Bare**	**PTS**	**Rotary Rack**
Total Flux	3.82E+13	1.67E+13	1.49E+13	1.07E+13	4.76E+12
Thermal Flux	1.98E+13	6.90E+12	6.12E+12	5.61E+12	2.75E+12
Epithermal Flux	9.22E+12	4.51E+12	4.05E+12	2.51E+12	1.15E+12
Fast Flux	9.15E+12	5.33E+12	4.73E+12	2.58E+12	8.61E+11
Thermal/Total Flux Ratio	0.52	0.41	0.41	0.52	0.58

Neutron Flux in Beam-ports

Figure 8 shows the thermal neutron flux inside each beam-port starting from outer surface of the core up to 40 cm length. The radial piercing beam-port number 4 (BP4) shows the highest flux while the tangential beam-port (BP2) has the lowest flux. As shown in Table 3, BP4 has the highest fast-to-thermal flux ratio. Since BP4 starts from the core and piercing through the graphite reflector, less thermalisation process occurs as compared to the others. Most of the fast neutron produce inside the fuel will have to travel radially through the relatively large graphite structure surrounding the core and being thermalised before entering BP1, BP2, and BP3.

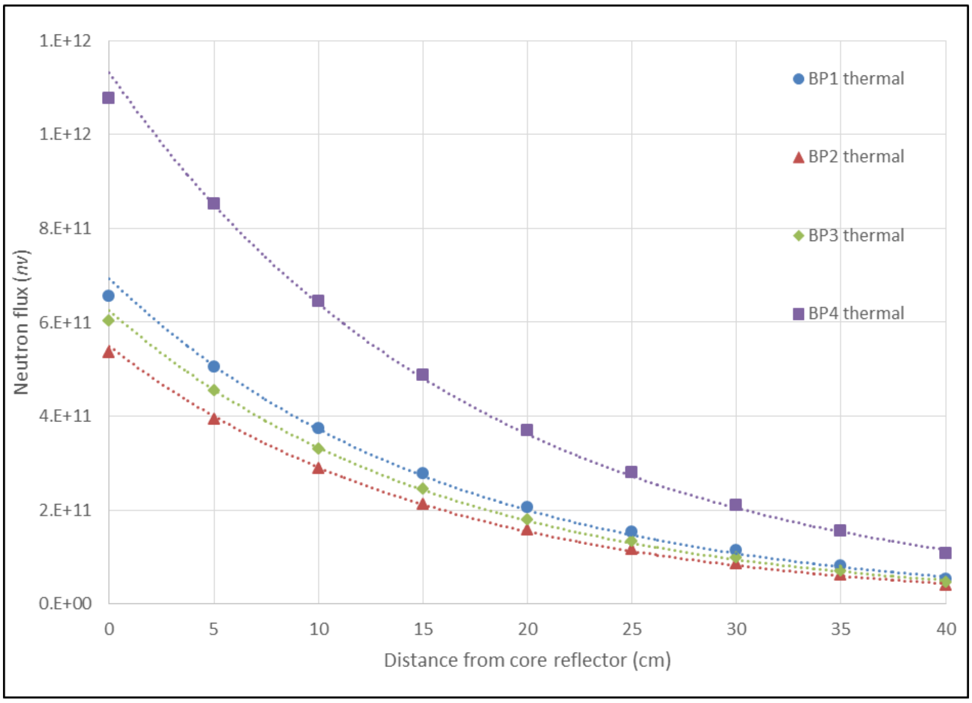

Figure 8 (a): Thermal neutron flux in BP1, BP2, BP3 and BP4 at 750 kW thermal powers

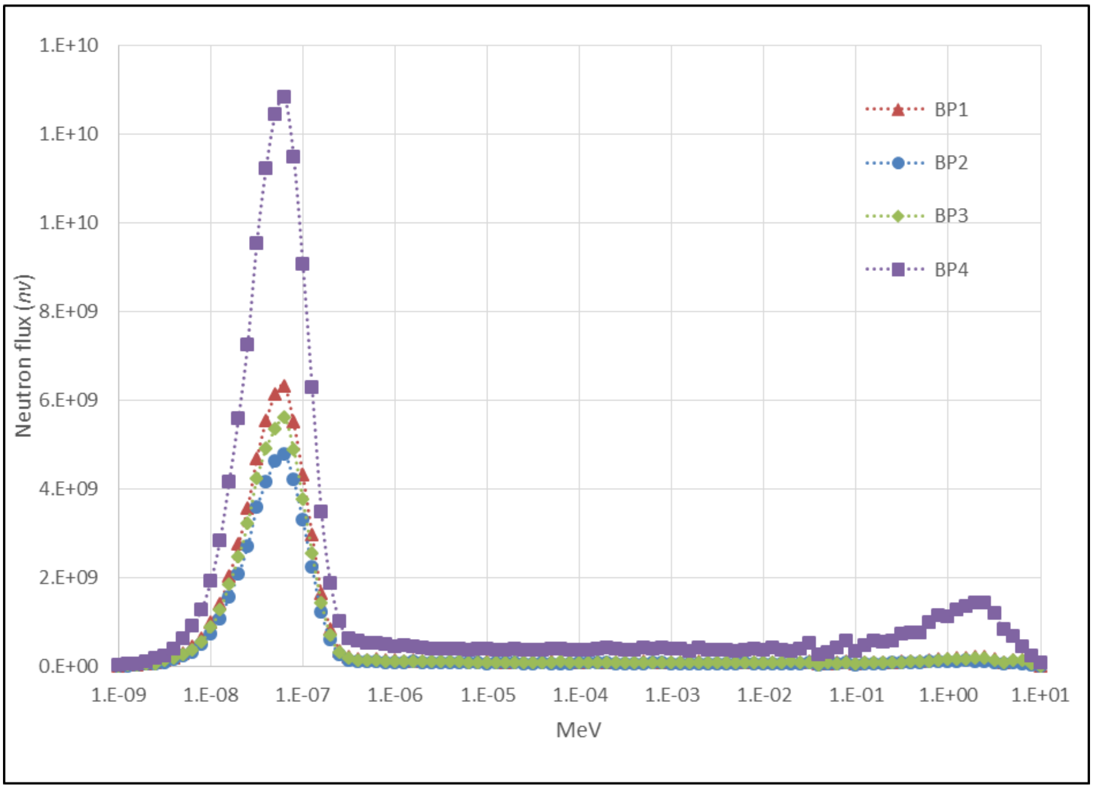

Figure 8 (b): Neutron spectrum at 40 cm distance from reflector at 750 kW thermal powers

Table 3: Neutron flux for BP1, BP2, BP3 and BP4 at 40 cm distance from reflector at 750 kW thermal powers

Neutron Flux (*nv*)	**BP1**	**BP2**	**BP3**	**BP4**
Total Flux	5.94E+10	4.45E+10	5.27E+10	1.44E+11
Thermal Flux	5.18E+10	3.92E+10	4.59E+10	1.08E+11
Epithermal Flux	5.01E+09	3.62E+09	4.41E+09	1.96E+10
Fast Flux	2.58E+09	1.64E+09	2.36E+09	1.70E+10
Thermal/Total Flux Ratio	0.87	0.88	0.87	0.75

CONCLUSIONS

The neutron flux and spectrum of the RTP core and irradiation facilities were determined by simulation method using a three-dimensional continuous energy Monte Carlo code MCNP. The in-core facilities, thermal and fast neutron flux distributions were estimated using MCNP mesh tally method. The neutron flux mapping obtained revealed the heterogeneous configuration of the core. It was found that the flux as well as its spectral characteristics depended very much on the position and location of the irradiation facility and its neighboring elements such as water, fuel or graphite rod in the core. The neutron flux characteristic in the beam-ports differed in how the beam tubes were arranged, with the piercing beam-port having the highest neutron flux and also with highest fast-to-thermal flux ratio. The maximum neutron flux was found to be in the Central Thimble facility with 1.98E13 *nv* of thermal neutron flux. The thermal-to-total flux ratio varies significantly from 0.41 for in-core facility, 0.58 in the reflector and up to 0.88 in the beam-ports.

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