Measurement and simulation of thermal neutron flux distribution in the RTP core

Mohamad Hairie B. Rabir¹, Abi Muttaqin B. Jalal Bayar¹, Na’im Syauqi B. Hamzah¹, Muhammad Khairul Ariff B. Mustafa¹, Julia Bt. Abdul Karim¹, Muhammad Rawi B. Mohamed Zin¹, Yahya B. Ismail¹, Mohd Huzair B. Hussain¹, Mat Zin B. Mat Husin¹, Roslan B. Md Dan¹, Ahmad Razali B. Ismail¹, Nurfaizila Bt. Husain¹, Zareen Khan B. Abdul Jalil Khan¹, Shaiful Rizaide B. Mohd Yakin¹Moadh Fauzi B. Saad¹, Zarina Bt. Masood¹

¹Reactor Technology Centre, Technical Support Division, Malaysian Nuclear Agency, Bangi, 43000 Kajang, Selangor, Malaysia

m_hairie@nuclearmalaysia.gov.my

Abstract. The in-core thermal neutron flux distribution was determined using measurement and simulation methods for the Malaysian’s PUSPATI TRIGA Reactor (RTP). In this work, online thermal neutron flux measurement using Self Powered Neutron Detector (SPND) has been performed to verify and validate the computational methods for neutron flux calculation in RTP calculations. The experimental results were used as a validation to the calculations performed with Monte Carlo code MCNP. The detail in-core neutron flux distributions were estimated using MCNP mesh tally method. The neutron flux mapping obtained revealed the heterogeneous configuration of the core. Based on the measurement and simulation, the thermal flux profile peaked at the centre of the core and gradually decreased towards the outer side of the core. The results show a good agreement (relatively) between calculation and measurement where both show the same radial thermal flux profile inside the core: MCNP model over estimation with maximum discrepancy around 20% higher compared to SPND measurement. As our model also predicts well the neutron flux distribution in the core it can be used for the characterization of the full core, that is neutron flux and spectra calculation, dose rate calculations, reaction rate calculations, etc.

1. 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 with excess reactivity of $ 0.15. The cross-sectional view of Core-15, the latest core configuration is shown in figure 1. It has annular core surrounded by graphite reflector and cooled by natural convection. 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 core consists of 8.5 wt. %, 12 wt. % and 20 wt. % fuel elements, 4 control rods, some graphite elements and central thimble. RTP uses standard TRIGA UZrH₁₆ fuel with 20 % of U-235 enrichment. The fuel dimension is shown in figure 2. 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) made up of 8.5 wt. % UZrH$_{1.6}$ and B$_4$C absorber on top of the fuel section.

![Figure 1](image1.png)

**Figure 1.** RTP core loaded with fuels, control rods, irradiation channels and dummy rods

![Figure 2](image2.png)

**Figure 2.** RTP fuel dimension

Most reactors use thermal neutrons to sustain the fission chain reaction. These so called thermal reactors contain neutron moderator that slows down fast neutrons to thermal equilibrium with the atoms (E < 1 eV) in the system. Neutron flux ($\Phi$) is to consider it to be the total path length covered by all neutrons in one cubic centimeter during one second. Mathematically, this is the equation,

$$\Phi = nV\phi$$  \hspace{1cm} (1)

where,
\[\phi: \text{neutron flux (neutrons cm}^{-2}\text{s}^{-1})\],
\[n: \text{neutron density (neutrons cm}^{-3})\] and.
\[V: \text{neutron velocity (cm s}^{-1})\].
As a very good approximation, the distribution of neutrons inside a nuclear reactor core can be determined using the two group diffusion equations as shown in figure 3. This flux distribution is valid in a homogeneous core. In reality, the distribution of neutrons is greatly affected by different elements in the core and reflector. Thus, to predict the detail neutron flux distribution in real reactor core arrangement, we rely on measurements or the use of Monte Carlo neutron transport method.

![Diagram](image.png)

**Figure 3.** Fast and thermal neutron distribution in core and reflector estimated using the two group diffusion equation.

The data and information on the in-core neutron flux distribution is crucial in research reactors. It is an input variable for many experiments and important for determining the distribution of in-core power. In this work, the radial flux density distribution of the thermal neutrons is measured in RTP core. The used method, i.e. on-line self-powered neutron detector, is one of the most important methods in nuclear technology. The results will then be compared to the Monte Carlo neutron transport code model of the core. This measurement plays a vital role as part of RTP neutronics properties analysis to support the improvement of in-core management strategy. A continuous improvement of core and fuel management strategy is crucial element needed for better utilization and to ensure safe operation of RTP.

2. Method

Self-powered neutron detector (SPND) is a unique type of neutron detector that is widely applied for in-core flux measurement. These devices incorporate a material chosen for its relatively high cross section for neutron capture leading to subsequent beta or gamma decay. In its simplest form, the detector operates on the basis of directly measuring the beta decay current following capture of the neutrons. There is no external bias voltage needed to be applied to the detector, hence the name self-powered. SPND is small in size, low cost, and relatively simple electronics required in conjunction with their use.

The SPND used in RTP consists of 3 mechanical components which is a coaxial cable consisting of an inner electrode (emitter), surrounded by insulator and an outer electrode (collector), as shown in figure 4. The emitter or beta source material absorbs neutrons and emits beta particles. An insulator electrically isolates the beta source material from the collector while collector absorbs the emitted beta particles as illustrated in figure 5. Vital material in this experiment is vanadium ($^{51}\text{V}$) as an emitter, which has a neutron-beta interaction with thermal neutron cross-section of 5 barns ($5 \times 10^{-24} \text{ cm}^2$) featuring a $1/v$ characteristic without resonances in the energy range of thermal/epithermal neutrons. Vanadium will react with thermal neutron and produce $^{52}\text{V}$ which is in excited state, based on $^{51}\text{V}(n,\gamma)^{52}\text{V}$ reaction. The $^{52}\text{V}$ then decay with a half-life of about 3.76 minutes to a stable nuclide which is chromium ($^{52}\text{Cr}$) with the emission of a beta particle. Due to this decay time, the SPND signal needs to be corrected using the following formalism.
2.1. SPND signal

Due to decay time of emitter, SPND signal needs to be corrected. The basic equation of the population of $^{52}\text{V}$ is the following:

$$\frac{dN(t)}{dt} = \sigma N_0 \varphi(t) - \frac{N(t)}{\tau}$$

(2)

with,

$\varphi$: neutron flux,

$N_0$: atom density of $^{51}\text{V}$,

$\sigma$: neutron absorption cross section of $^{51}\text{V}$ to produce $^{52}\text{V}$ ($=4.9b$),

$\tau$: characteristic time decay of $^{52}\text{V}$,

and $\tau$, is given by:

$$\tau = \frac{T_{1/2}}{\ln(2)}$$

$$= \frac{3.75 \text{ min}}{0.693} = 5.41 \text{ min}.$$  

(3)

Signal response correction via digital compensation formalism is given by:

$$\varphi_i = C I_i$$

$$\varphi_{i+1} = C \left[ I_{i+1} - \exp\left(-\frac{\Delta}{\tau}\right) I_i \right] + \exp\left(-\frac{\Delta}{M\tau}\right) \varphi_i$$

(4)

with $\Delta$ as the sampling time and $C$ as a normalisation constant and, is given by:

$$C = \frac{1 - \exp\left(-\frac{\Delta}{M\tau}\right)}{1 - \exp\left(-\frac{\Delta}{\tau}\right)}$$

$$M = \frac{1}{1 + \frac{K_d}{K_p}} = \frac{K_p}{K_p + K_d}$$

(5)
Here, $K_d$ is the probability that a $^{52}$V decay produces an electron contributing to the current and, $K_p$ as the the probability that a neutron captured by $^{52}$V decay leads instantaneously to an electron contributing to the current.

**Figure 6.** Example of corrected response

2.2. Neutron source
The MCNP tally results are relative, normalised to one source particle. It needs to be scaled to a desired fission source (power) level to get absolute value (flux, reaction rate, fission density, etc). Typical source strength for 1MW power, $S$ and scaling method is given by:

$$S = \frac{P \times \nu}{Q}$$  \hfill (6)

where,

- $P$: reactor power (watt),
- $\nu$: neutrons emitted per fission for U-235,
- $Q$: energy released per fission (watt.s)

The scaled flux $\Phi$ (n.cm$^{-2}$s$^{-1}$), is given by:

$$\Phi = \phi_{F4} \times S \times \frac{1}{k_{\text{eff}}}$$  \hfill (7)

where,

- $\phi_{F4}$: MCNP F4 tally output (cm$^{-2}$)
- $S$: source strength (n.s$^{-1}$).

Figure 7 shows the measurement set-up where each SPND is attached to a holder made of an Aluminium hollow rod. The holder then can be fixed to a flux hole on the reactor core top grid plate. The location of flux holes used in this measurement is shown in figure 8 and 9 respectively.
The Monte Carlo N-Particle Code (MCNP) with ENDF-VII cross section data was used to simulate the thermal neutron flux distribution in the core. There are several works mentioned in reference [1], [2] and [3] on the development of RTP MCNP core calculation. Detail of MCNP calculation of RTP can be found from these references. The RTP core neutronics analysis using MCNP code for current configuration involved burnup and nuclide inventory data. TRIGLAV code produced individual fuel burnup in MWD and then the actinides and fission product build-up calculation performed using MCNPX BURNUP card. The model then was used to re-create thermal neutron flux measurement at the same location of the SPNDs as shown in figure 8 and figure 9.
3. Results

Figure 10 shows the measurement results for CT and E1 location at different core power level. Reactor power increased gradually from 15 W to 50 kW, 100 kW, 250 kW, 500 kW, 750 kW and lastly 1 MW. The delayed response with a characteristic half-life value of 3.75 minutes from SPND signal needs to be corrected using special digital compensation method to get actual flux curve as shown in section 2.1. There are several literatures [4][5] on the formulation for the SPND digital compensation method; in this work, we used the provided formulation in reference [6]. The response correction was performed for every SPND signals at every measurement location in the core.

Figure 10. Gradual increase of thermal neutron flux measurement at Central Thimble (CT) and E1 location for gradual increase of core power level from zero power to maximum 1 MW.
Figure 11 shows the spatial thermal neutron flux for the half side of the core. Based on these results, we can see the trend where maximum flux is in the centre and gradually reduced towards the edge due to leakage. The local flux fluctuation is observable heterogeneous core trend. The radial thermal flux profile was found to be non-cosine due to heterogeneity of the core and variation in fuel element types and burnup. The significant peak at the core centre is also because of it has larger water volume compare to other location of measurements in B, C, D, E, F or G ring. Hence more moderation process occur which increased thermal neutron flux at the core centre. The effect of graphite rods on the core flux shape are also noticeable; SPND measurement at the core edge where it was surrounded by graphite elements, cause the flattening of thermal neutron flux.

![Figure 11. Thermal neutron flux trend normalized to maximum flux value of MCNP simulation and SPND measurement.](image1)

Figure 11 also shows the thermal neutron flux radial profile comparison between measured and calculated from MCNP simulation. The thermal neutron flux calculation outputs from MCNP are normalized to reactor power using the method mentioned in section 2.2. MCNP simulation overestimate measured flux at average discrepancy around 10% but the trend for both results are almost the same. There are several factors of this discrepancy and one of them is the linear scaling of power to flux in MCNP. In actual phenomena, neutron flux does not increase linearly with power due to the strong negative temperature feedback as shown in measurement curve in figure 12. Measured fuel temperature at full power level was around 300 °C to 400 °C.

The radial thermal neutron flux distribution (integrated over axial active region) simulation results using RTP MCNP model is shown in figure 13. Notice that large depressions of thermal flux inside the fuel elements, especially the several locations in the F-ring filled with 20 wt. % fuel type. Local thermal flux peaks inside the water gaps between the fuel elements are also recognizable. The calculated excess reactivity is $5.20$ while measured data at the beginning of the 15th core configuration is $5.18$. 
Figure 12. Thermal neutron flux in CT at different reactor power level. Comparison between MCNP simulation linear curve and SPND measured neutron flux with polynomial curve.

Figure 13. MCNP simulation of thermal neutron flux radial distribution at 1 MW core power level.
4. Conclusions
Based on the measurement, the thermal neutron flux profile peaked at the centre of the core and gradually decreased towards the outer side of the core. This distribution resulted shows a clear heterogeneous core effect that could not be predicted using neutron diffusion method. Thus a more advanced code; MCNP Monte Carlo simulation was needed due its ability to predict complex core arrangement. Comparison between MCNP code and SPND measurement results showed a good agreement (relatively), where both methods showed the same radial thermal flux trend inside the core. MCNP simulation results over estimation was ~10% higher compare to the SPND measurement. This was due to error in model simplification, cross section data (ENDFVII), feedback phenomena, uncertainty in power and actual measurement setup. As our model also predicts well the neutron flux distribution in the core it can be used for the characterization of the full core, that is neutron flux and spectra calculation, dose rate calculations, reaction rate calculations, etc.

5. References
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