ONLINE IN-CORE THERMAL NEUTRON FLUX MEASUREMENT FOR THE VALIDATION OF COMPUTATIONAL METHODS

Mohamad Hairie B. Rabir, *Dr Muhammad Rawi B. Mohamed Zin, Yahya B. Ismail, Mohd Huzair B. Hussain, Mohammad Raffiq B. Abdul Rahim, Mat Zin B. Mat Husin, Roslan B. Md Dan, Ahmad Razali B. Ismail, Nurfazila Bt. Husain, Zareen Khan B. Abdul Jalil Khan, Mark Dennis Anak usang, Abi Muttaqin B. Jalal Bayar, Na'im Syauqi B. Hamzah, Muhammad Khairul Ariff B. Mustafa, Shaiful Rizaide B. Mohd Yakin, Mohamad Fauzi B. Saad, Datin Zarina Bt. Masood*

Reactor technology Centre, Technical Support Division, Malaysian Nuclear Agency, Bangi, 43000 Kajang, Selangor, Malaysia.

m_hairie@nuclearmalaysia.gov.my,+603-89112000 ext: 2327 / 6133

ABSTRACT

In order to verify and validate the computational methods for neutron flux calculation in RTP calculations, a series of thermal neutron flux measurement has been performed. The Self Powered Neutron Detector (SPND) was used to measure thermal neutron flux to verify the calculated neutron flux distribution in the TRIGA reactor. Measurements results obtained online for different power level of the reactor. The experimental results were compared to the calculations performed with Monte Carlo code MCNP using detailed geometrical model of the reactor. The calculated and measured thermal neutron flux in the core are in very good agreement indicating that the material and geometrical properties of the reactor core are modelled well. In conclusion one can state that our computational model describes very well the neutron flux distribution in the reactor core. Since the computational model properly describes the reactor core it can be used for calculations of reactor core parameters and for optimization of RTP utilization.

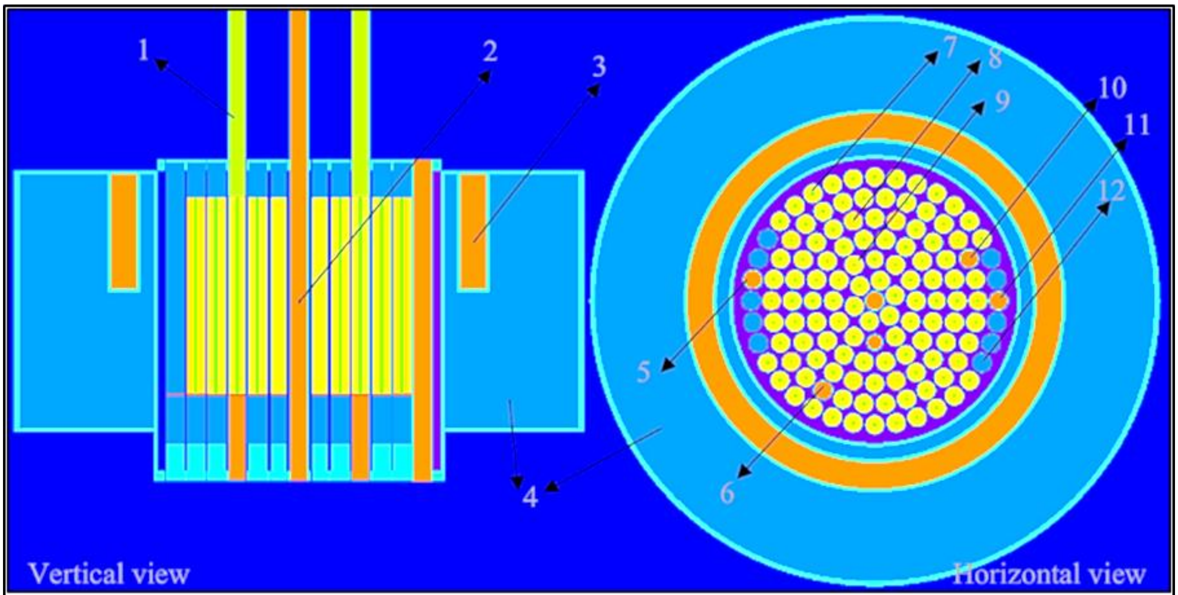
Keywords: SPND, MCNPX, RTP, Neutronics calculation

Area of research: Nuclear Engineering, Reactor Physics, Neutronics Analysis

INTRODUCTION TO RTP

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. It uses standard TRIGA $UZrH_{1.6}$ fuel of 8.5 wt. %, 12 wt. % and 20 wt. % with 20 % of U-235 enrichment. It has annular core surrounded by graphite reflector and cooled by natural convection. The top and bottom grid plate is made of Al-6061. 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_4C absorber on top of the fuel section. RTP used mainly for beam experiments, samples analyses, education and trainings. The reactor utilizes hydride fuel which is a homogeneous mixture of uranium and zirconium hydride (UZrH). The ZrH is used as main moderator. The core consists of 8.5, 12 and 20wt. % fuel elements, 4 control rods, some graphite elements and central thimble. The cross-sectional view of Core-15, the latest core configuration is shown in Figure 1.

Elements are arranged in seven circular rings and the spaces between the fuel rods are filled with water that acts as coolant and moderator.



Cross-sectional view of RTP Core-15 MCNP model:
1- Fuel follower control rod, 2- CT, 3- RR, 4- Graphite reflector, 5- PTS, 6- DT, 7- 8.5 U wt.% fuel rod, 8- 12 U wt.% fuel rod, 9- 20 U wt.% fuel rod, 10- DNA-Bare, 11- DNA-Cd, 12- Dummy fuel (graphite rod)

Figure 1, Side and top view of RTP Core-15 configuration.

RTP core neutronics analysis have been done using deterministic and Monte Carlo code. Monte Carlo methods have gained interest due to the ability to more accurately model complex 3-dimesional geometries. TRIGLAV is a deterministic code, based on diffusion approximation of transport equation, which uses WIMSD program to calculate a unit cell averaged cross section data. TRIGLAV program package is developed for reactor calculations of mixed cores in TRIGA Mark II research reactor. It can be applied for fuel element burnup calculations, for power and flux distributions calculations and for critically predictions. The main aim of this work is to perform in-core thermal neutron flux measurement using Self-Powered Neutron Detector (SPND) to obtain radial flux profile and to validate the developed MCNP model of the PUSPATI TRIGA Reactor‘s 15th core configuration.

NEUTRON FLUX

Neutron flux ($\boldsymbol{\phi}$) is to consider it to be the total path length covered by all neutrons in one cubic centimetre during one second. Mathematically, this is the equation, $\boldsymbol{\phi} = \boldsymbol{nv}$, 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}). In order to ensure predictable temperatures and uniform depletion of the fuel installed in a reactor, numerous measures are taken to provide an even distribution of flux throughout the power producing section of the reactor. This shaping, or flattening, of the neutron flux is normally achieved through the use of reflectors that affect the flux profile across the core. For any nuclear reactor, knowledge on the spatial neutron flux density distribution is of major interest: in research reactors, it is an input variable for many experiments (e.g. as the source strength for irradiation experiments), in power reactors, it is important for determining the distribution of heat sources. In this work, the radial flux density distribution of the thermal neutrons is measured. The used method, i.e. on-line self-powered neutron detector, is one of the most important methods in nuclear technology.

ON-LINE IN-CORE NEUTRON FLUX MEASUREMENT

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. This current should then be proportional to the rate at which neutrons are captured in the detector. Because the beta decay current is measured directly, no external bias voltage need be applied to the detector, hence the name self-powered. Compared with other neutron sensors, self-powered detectors have the advantages of small size, low cost, and the relatively simple electronics required in conjunction with their use.

The SPND used in RTP consist of 3 mechanical components which is a coaxial cable consisting of an inner electrode (emitter), surrounded by insulator and an outer electrode (collector), figure 2. The emitter or beta source material absorbs neutrons and emits beta particles. Insulator, electrically isolate the beta source material from the collector while collector absorbs the emitted beta particles. Vital material in this experiment is vanadium (^{51}V) as an emitter, which has a neutron-beta interaction with thermal neutron cross-section of 5 barns ($5 \times 10^{-24}\ cm^2$) featuring a 1/v characteristic without resonances in the energy range of thermal/epithermal neutrons. Vanadium will react with thermal neutron and produce ^{52}V which is in excited state, based on ^{51}V(n,ɣ)^{52}V reaction. The ^{52}V then decay with a half-life of about 3.76 minutes to a stable nuclide which is chromium (^{52}Cr) with the emission of a beta particle. Figure 3 showing the measurement set-up where each SPND attached to a holder made of an aluminium hollow rod. The holder then can be fixed to a flux hole on the reactor's top grid plate. The location of flux holes used in this measurement is shown in figure 4 and 5 respectively. Other details on the SPND measurement in RTP can be found in reference 5.

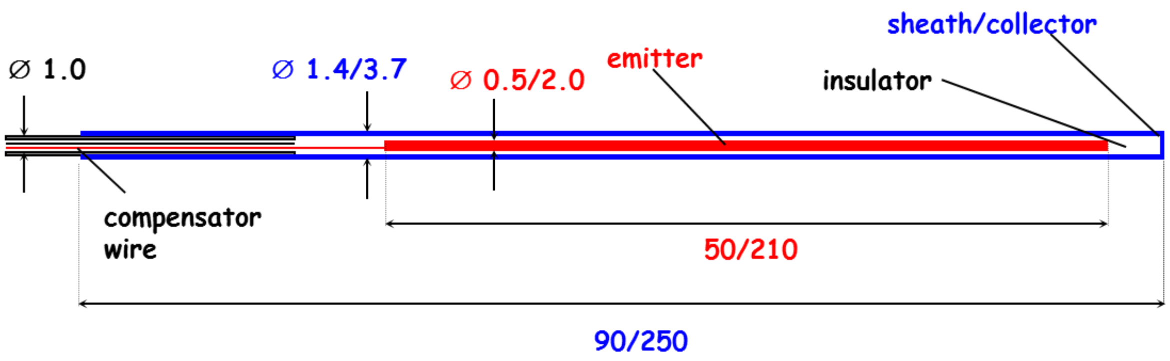


Figure 2, Schematic diagram of sensitive region of SPND used in RTP.

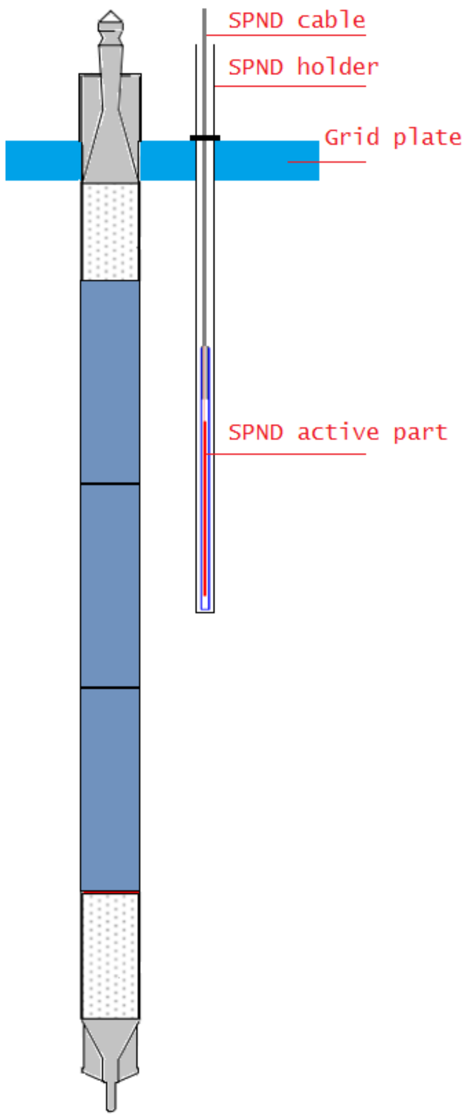


Figure 3, Schematic diagram of RTP fuel with SPND inserted in a holder which left hanging on the top of core grid plate where the axial position can be adjust.

RTP CORE MCNP MODEL

In previous work, RTP MCNP model with fresh fuel was developed and validated against experimental data. However, a core model with burned fuel is needed in order to obtain accurate analysis for current and future core parameters. RTP core neutronics analysis using MCNP code for current configuration will involve burnup and nuclide inventory data. The MCNP model includes actinides and fission products where the determination of fuel burnup will based on TRIGLAV calculation. Based on the core burnup, the burnup value for each individual fuel rod can be determined. TRIGLAV output produced individual fuel burnup in MWD. Actinides and fission product build-up calculation will be done using MCNPX. Each fuel with specific burnup value will be matched to its actinides and fission products inventory, and used in material input deck of the MCNP core model.

Detail information on the development of the model can be found in reference 5, 6 and 7. This model then will be used to calculate thermal neutron flux at the same location of SPND measurement, core power level and with the same control rod position during the measurement as shown in figure 4 and 5. The radial thermal neutron flux distribution simulation results using RTP MCNP model is shown in figure 6. Notice that large depressions of thermal flux inside the fuel elements, especially the several locations in the F ring filled with 20wt. % 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 15^{th} core configuration is \$5.18.

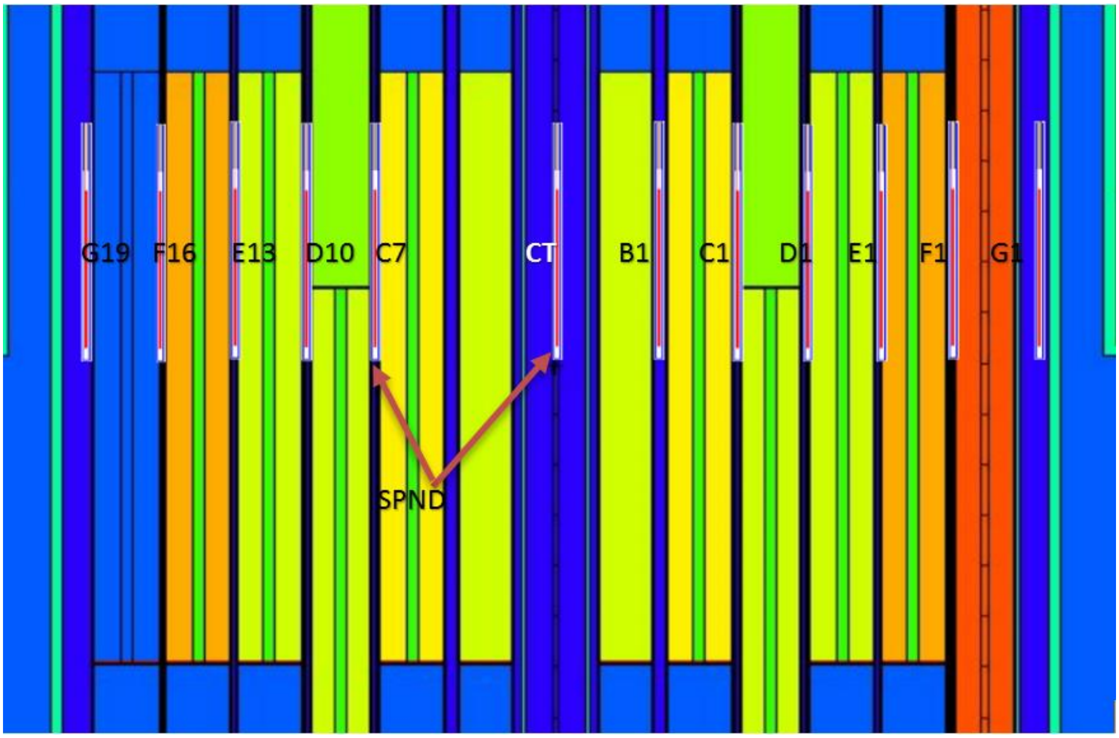


Figure 4, Side view of RTP core model in with actual control rod position during the measurement.

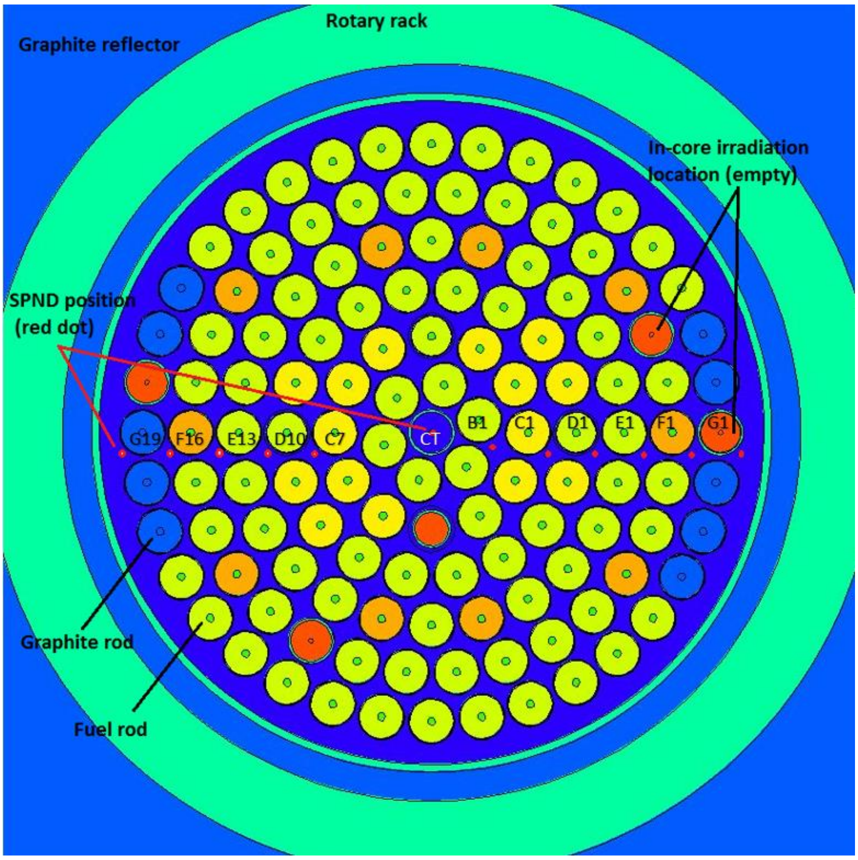


Figure 5, Top view of RTP core showing SPND position.

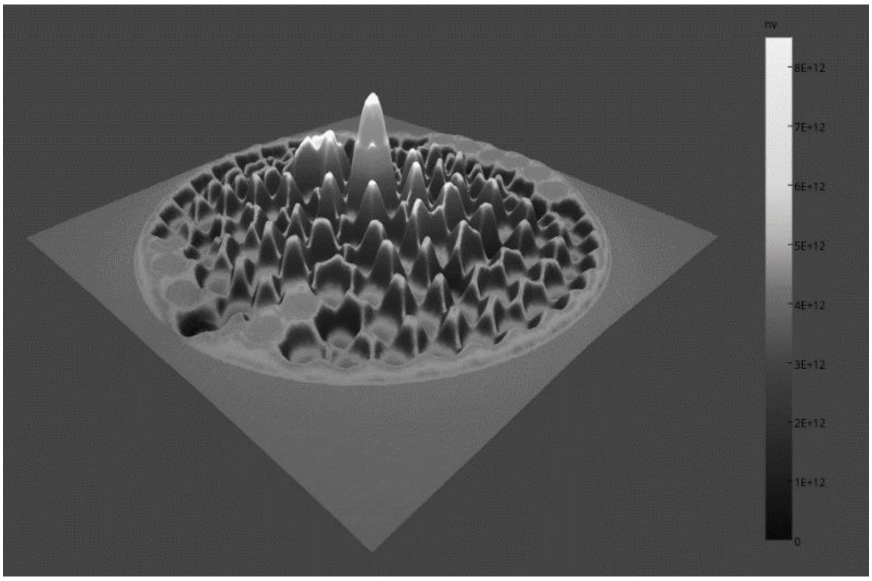


Figure 6, Thermal neutron flux distribution inside RTP core predicted by MCNP simulation where the maximum flux located at the centre of the core (in Central Thimble irradiation facility). The maximum thermal flux value is around $8.7\text{x}10^{12}$ n.cm^2.s^{-1} at 500 kW core power level.

RESULTS

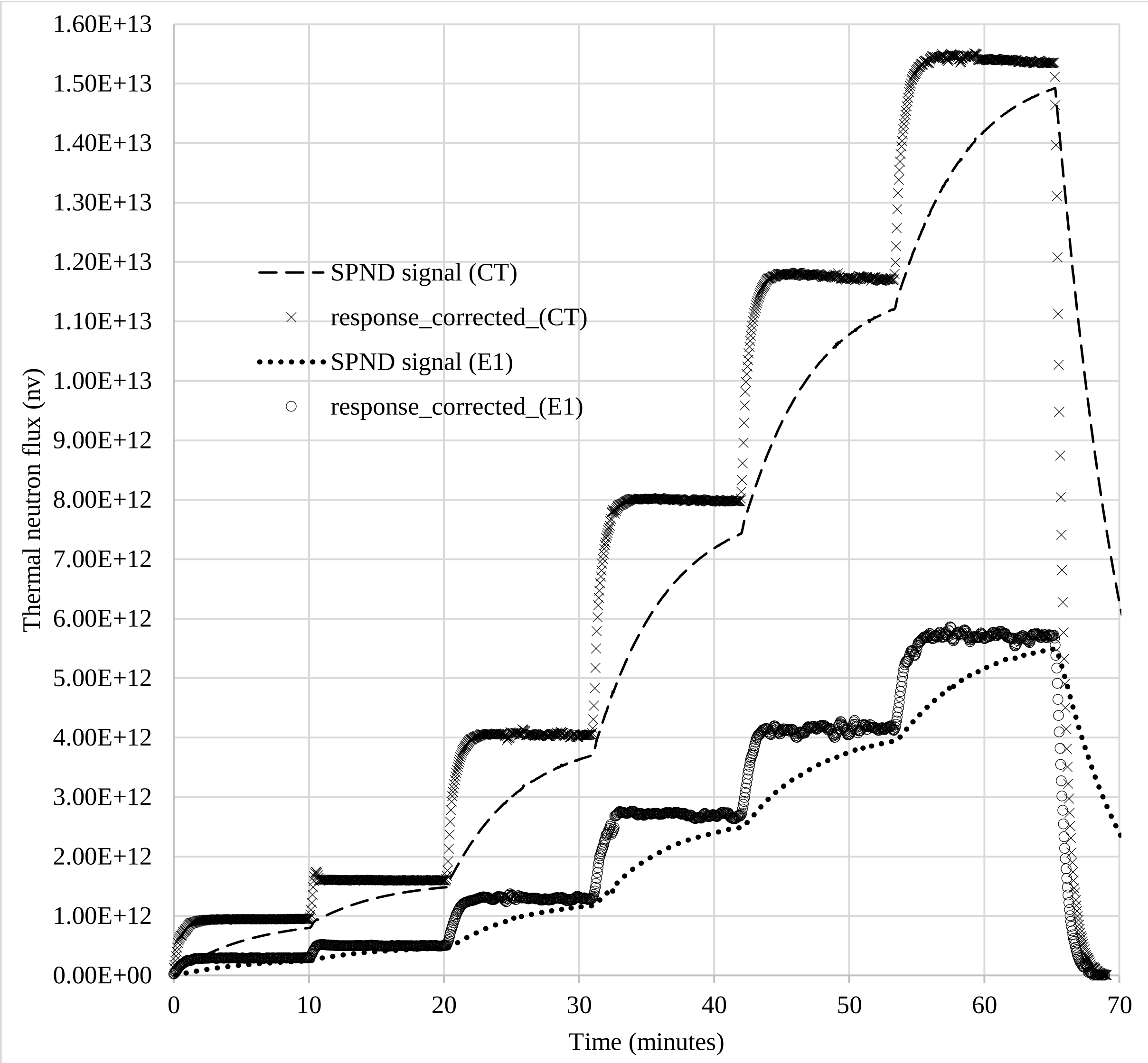


Figure 7, Gradual increase of thermal neutron flux measurement at Central Thimble (CT) and E1 for gradual increase of core power level from zero power to maximum 1 MW.

Figure 7 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 need to be corrected using special digital compensation method to get actual flux curve. There are several literature (reference 8 and 9) on the formulation for the SPND digital compensation method, we used the provided formulation in reference 2. The response correction was performed for every SPND signals at every measurement location in the core. This results give us information on spatial thermal neutron flux value as shown in figure 8. Figure 8 also shows the thermal neutron flux radial profile comparison between measured and calculated from MCNP simulation. MCNP simulation overestimate measured flux at average discrepancy around 11% but the trend for both results are almost the same.

The thermal neutron flux calculation outputs from MCNP are normalized to reactor power using the method mentioned in reference 6 and 7. The radial thermal flux profile was found to be not symmetrical due to heterogeneity of the core and variation in fuel element burnup. The significant peak at the core centre is 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 where at G19, the SPND surrounded by more graphite elements including the reflector compare to SPND at G1 location which cause higher thermal neutron flux.

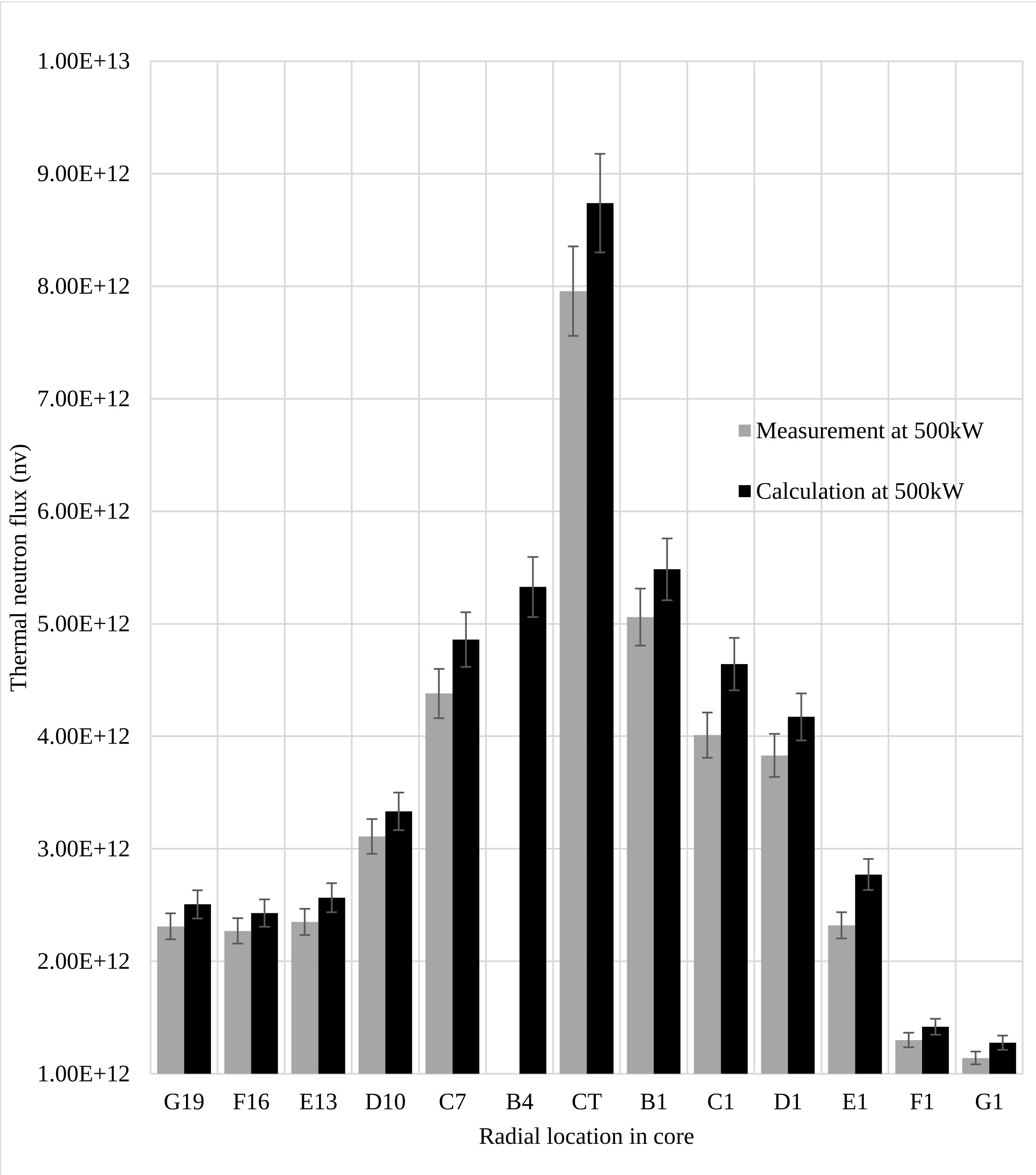


Figure 8, Comparison between calculated and measured in-core thermal neutron flux.

CONCLUSION

Based on the measurement, the thermal flux profile peaked at the centre of the core and gradually decreased towards the outer side of the core. Verification of the neutron flux distribution was performed by comparing the MCNPX simulation and online SPND measurement. 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. Moreover it can also be used for the calculation of various core parameters, such as power peaking factors and kinetic parameters of the reactor core.

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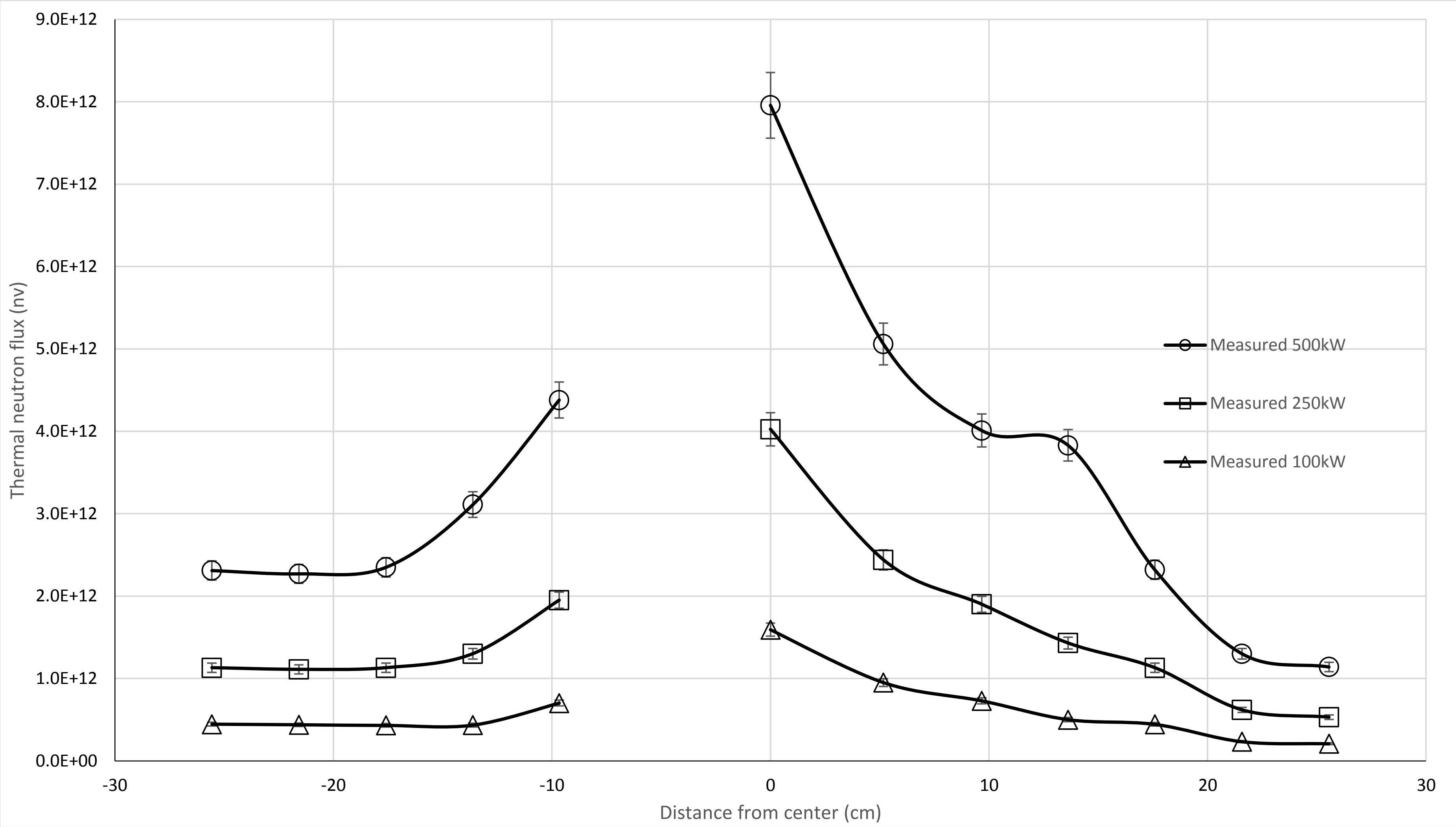
APPENDIX



Measurement results of in-core thermal flux profile at 100 kW, 250 kW and 500 kW