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TECHNICAL MATERIAL

Gamma Dose Rate Calculation for RTP TRIGA Reactor Using MCNP

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The Malaysian 1 MW TRIGA MARK II research reactor at Malaysian Nuclear Agency achieved initial criticality on June 28, 1982. The reactor 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 gamma dose rate on the water pool top and outer surface of concrete shielding of the TRIGA reactor during steady state operation. The 3-D continuous energy Monte Carlo code MCNP was used to develop a versatile and accurate full model of the TRIGA core with pool water and concrete shielding and compared with the measured data available in the safety analysis report (SAR) of the reactor. The model represents in detailed all components of the reactor 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, dose rate, shielding, RTP

I. 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 June 1982. It uses standard TRIGA UZrH1.6 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. Top and bottom grid plates are made of Al-6061 type. RTP has 4 control rods which are made of boron carbide. Three of them are fuel follower type and the other is air follower. Fuel follower control rods (FFCR) are installed with 8.5wt% $UZrH_{1.6}$ and B_4C absorber on top of the fuel section. RTP is used mainly for beam experiments, samples analyses, education and trainings. **Figure 1** shows the cross sectional view of the reactor.

As part of safety improvement in the reactor, it is crucial to simulate the shielding capability of the reactor and to verify the measured data available in the safety analysis report of RTP. It is one of the calculation development activities for the reactor and for the upcoming upgrading project. For this purpose, a three-dimensional model of the reactor with shielding was elaborated using the Monte Carlo code MCNP to investigate the gamma dose rate on the top of water pool and on the outer surface of concrete shielding during steady state operation.

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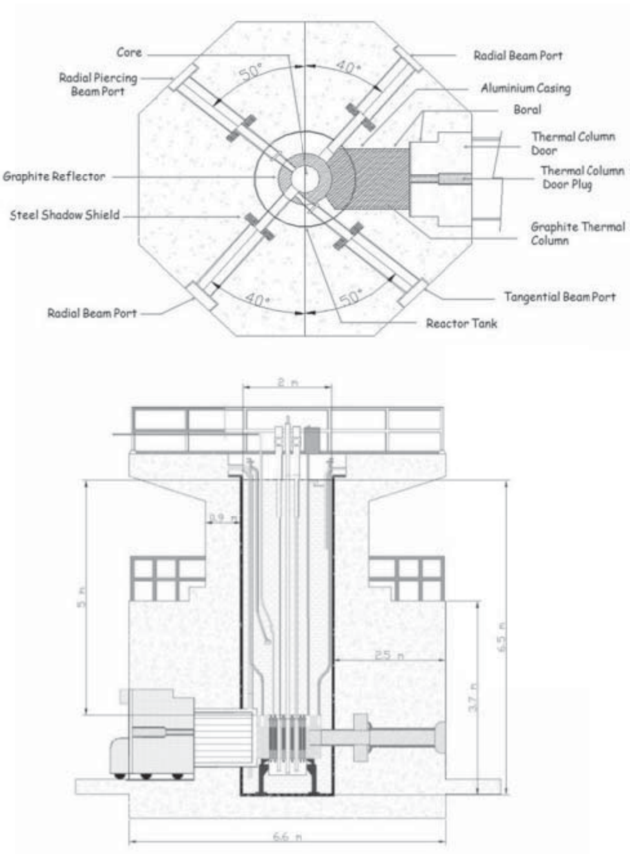

Fig. 1 Side and top view of RTP.

II. Method

Evaluation of shielding capability of water above reactor core and its biological concrete shielding during steady state operation is one of the most important aspects of a TRIGA research reactor. In our MCNP calculation of gamma dose rate during operation was through the use of KCODE card with f6 and f4 tallies along with DE and DF cards for energy to dose conversion. 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 current, 14th core consists of 112 fuel elements, 4 control rods, 11 graphite elements and central thimble. The cross-sectional views of core and shielding of the reactor are shown in **Figure 2** and **Figure 3** respectively. Fuel elements are arranged in seven circular rings and the spaces between the fuel rods are filled with water that acts as coolant and moderator. Concrete shielding composition of the reactor is shown in **Table 1**.

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 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 graphite dummy elements are of the same general dimensions and construction as the fuel-moderator elements, excluding that these elements are filled entirely with graphite. Ring shape tally cells were created in the concrete shielding to calculate the gamma dose rate trend towards the outer surface of the concrete. Several tally disk cells were also created above the core until the pool surface for the same purpose.

Table 1 Concrete shielding composition.

Element	Material ID	(number of atoms) /(barn*cm)
H-1	1001	0.00786
O-16	8016	0.0439
Al-27	13027	0.00239
Ca-natural	20000	0.00292
Na-23	11023	0.00105
Si-natural	14000	0.0158
Fe-natural	26000	0.00031
Mg-natural	12000	0.00014
K-natural	19000	0.00069

The dose rate calculated directly using the f4 tally for comparison purpose with the f6 tally results. The f6 tally results are given in units of MeV/g. It can be modified by a constant multiplier (1 MeV/g = 1.602×10^{-8} rad) to get dose values in units of rad-in-air and then converted to Sv/hr (1rad = 0.01Sv for gamma radiation). The f4 tally was used with DE and DF cards for energy to dose conversion using ICRP 74 for photon which was taken from reference 5. Both tallies which are in kcode criticality calculation need to be scaled to the number of fission neutrons of $7.59\text{x}10^{16}$ n/s for 1MW operation with an assumtion of 200MeV energy/fission. Another set of calculation was done using SDEF card with a cylindrical volume source. The active region in the reactor core was homogenised into a cylindrical volume and the gamma ray spectrum was taken from criticality calculation. The f6 tally used in SDEF source and scaled with the gamma flux value.

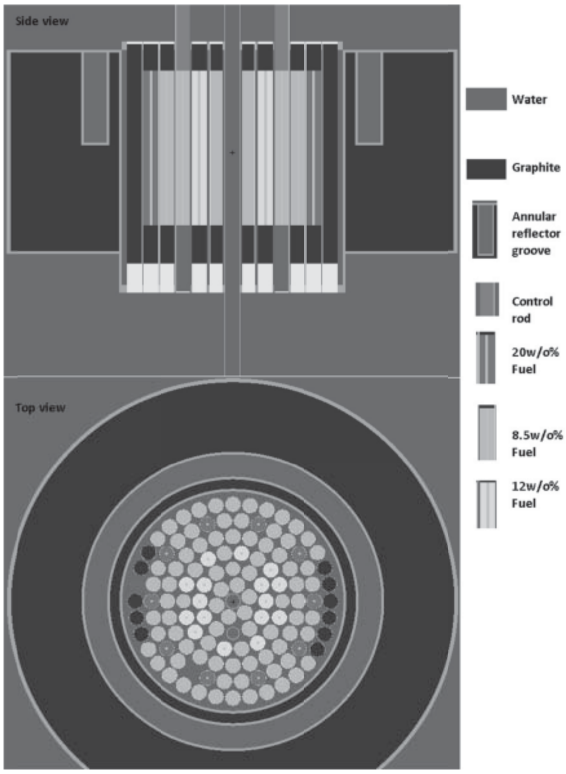

Fig. 2 Side and top views of MCNP core model.

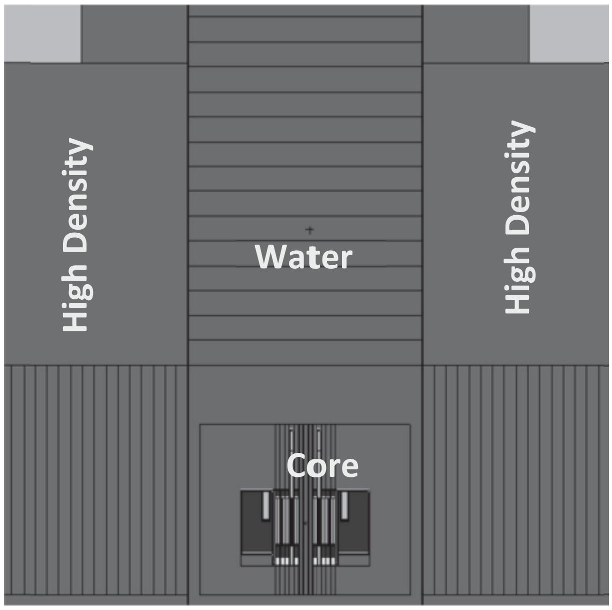

Fig. 3 MCNP model for the reactor with tally cells above cor and arround the core.

III. Results and Discussion

Figure 4 and **Figure 5** show the gamma dose rate calculation results in water coolant above the core and concrete shielding for 1MW thermal power respectively.

Two calculation set using f4 and f6 tally were done in kcode and another calculation using sdef with gamma spectrum can be seen in **Figure 6**. The unit of f6 tally result was in Mev/g while f4 tally was in gamma/cm^2; both use different method to convert MCNP result to dose rate in Sv/hr. Statistical errors were kept below 5% by modifying the number of histories in kcode calculation with parallel run time of approximately ~14 days, while for sdef calculation the run time only takes a few days in order to obtain relatively small statistical error.

The dose rate predicted by MCNP on the water pool surface seems to be lower than the measured data in the Safety Analysis Report (SAR) of PUSPATI TRIGA Reactor (1983) as can be seen in Table 2. One possible explanation for this behavior is the contribution of dose from other source especially water coolant activity and activation products were neglected in MCNP.

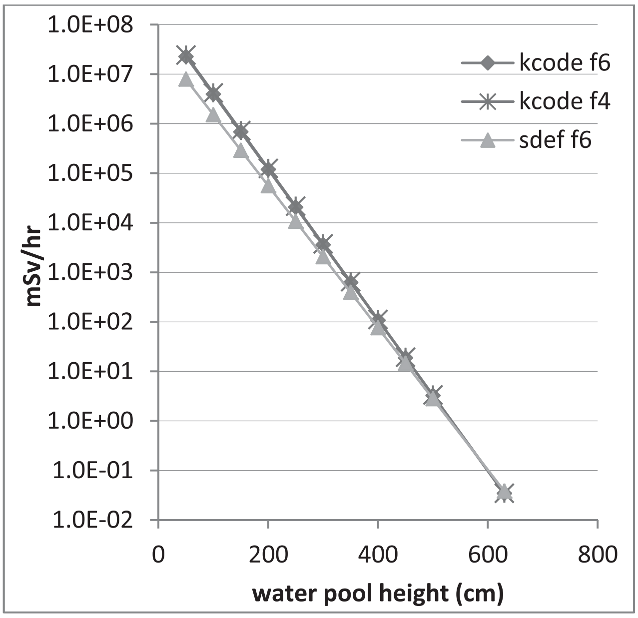

Fig. 4 Dose rate in water coolant at 1MW (semi-log scale).

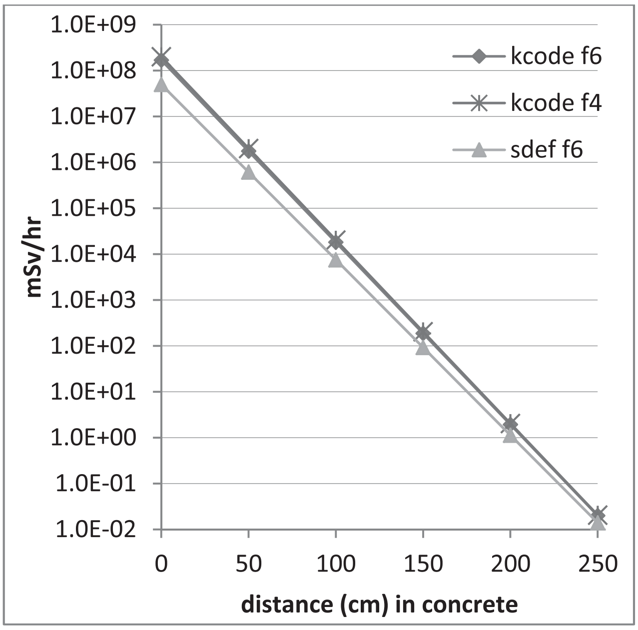

Fig. 5 Dose rate in concrete at 1MW (semi-log scale).

The magnitude of this effect is not known but it is believed to be significant. The calculated dose rate in concrete shielding was also smaller compared to the SAR value, but still in a good agreement. **Figure 6** shows the calculation results of gamma ray spectrum in the core at 1MW operation. The prompt fission gamma ray with energy 0.5MeV was found to have the maximum flux value of 8.70×10^{15}photons/s.

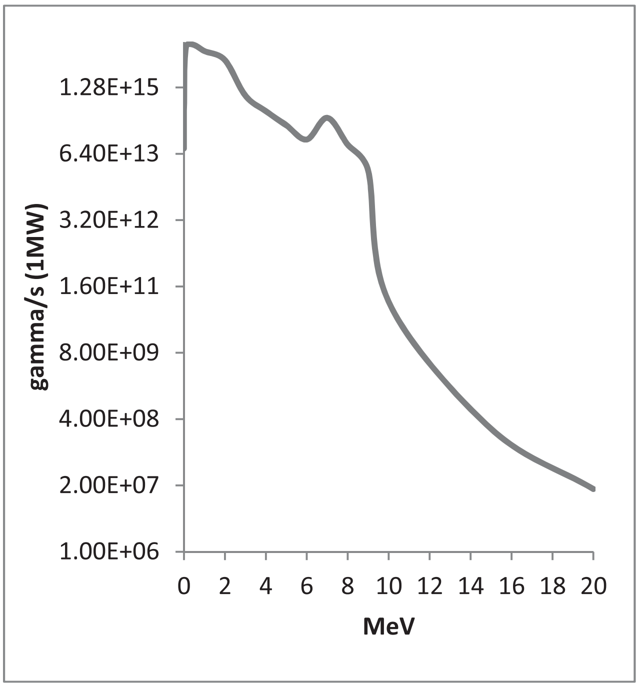

Fig. 6 Gamma ray spectrum (log-linear).

Table 2 MCNP dose rate and measured data in SAR.

Power level at 1MW	**SAR** (mSv/hr)	**MCNP** (mSv/hr)
Reactor pool top	5.50 ± 0.50	5.02
Concrete shielding outer surface	0.05 ± 0.05	0.02

IV. Conclusion

The evaluation of gamma dose rate on the reactor water pool top and the outer surface of concrete shielding during operation for RTP Triga Reactor was performed by three-dimensional continuous energy Monte Carlo code MCNP. MCNP calculation was found to be lower than the experimental value stated in SAR, but both values are still in agreement. It is understood that, the dose rate originated from activation product was neglected in MCNP calculation. This result shows that the reliability of MCNP code both for design and verification of reactor shielding, but the result must be assessed with great care due to some limitation of the code.

Acknowledgement

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