International Conference
Nuclear Energy for New Europe 2009
Bled / Slovenia / September 14-17

Modeling a TRIGA Mark II Biological Shield using MCNP5

R. Khan , S. Karimzadeh, H. Böck, M.Villa
Vienna University of Technology/Atominstitute
Stadionallee 2, A-1020, Vienna, Austria
nrustam@at.ac.at , sam.karimzadeh@ati.ac.at , boeck@at.ac.at , mvilla@ati.ac.at

ABSTRACT

The 250kW TRIGA Mark II research reactor, Vienna, is equipped with experimental facilities inside and outside the reactor core. Outside the core, there are four beam tubes (three radial and one tangential beam tube), a thermal column and a neutron radiography collimator, supplying neutrons for irradiation experiments. These experiments include neutron radiography, neutron tomography, neutron activation and academic training purposes. This paper describes calculations of the neutron flux density in the thermal column and in one of the selected beam tubes (i.e. BT-A), using the MCNP5 neutron transport code. To validate these calculations with experiments, neutron flux measurements using the gold foil activation method were performed on 13 selected positions in the thermal column. After comparison between calculations and measurements on thermal column, this paper also presents the thermal flux distribution in BT-A calculated by MCNP5. For this task, the already developed MCNP model for the TRIGA core was extended to the biological concrete shield including the four irradiation beam tubes and the above mentioned irradiation facilities. To save computational costs and incorporate the accurate and complete information for the individual MC particle tracks, the surface source writing capability of MCNP was applied to the TRIGA shielding model. The variance reduction techniques were also applied to improve the statistics of the problem and to save computational efforts.

1 INTRODUCTION

The TRIGA Mark II research reactor in Vienna has been operating since March 1962 with a thermal power of 250 kW. Its core is a mixed core of three types of fuel elements [1]. This reactor is the only operating research reactor in Austria. Since its first operation, this reactor has served five main purposes: (1) research and training in nuclear technology fields, (2) radio-isotope production for in-house research, (3) neutron scattering experiments to study neutron- and solid-state physics, (4) neutron radiography/tomography and (5) neutron activation analysis. The first two research areas utilize several irradiation facilities located inside the reactor core while the last three use facilities positioned outside the reactor core.

To satisfy its experimental irradiation requirements outside the reactor core, this reactor has a thermal and a dry irradiation room, four beam tubes (A, B, C and D). Three of these beam tubes (A, B, and C) are radial while the fourth one is tangential related to the reactor core. The top view of these experimental facilities can be seen in figure 1.

The thermal column of the TRIGA Mark II reactor is used to produce thermal neutrons for special irradiation experiments. It is basically a large, Boral lined, graphite-filled aluminium container. Its outside dimensions are 1.2 x 1.2 metres in cross section by approximately 1.6 metres in depth [2].

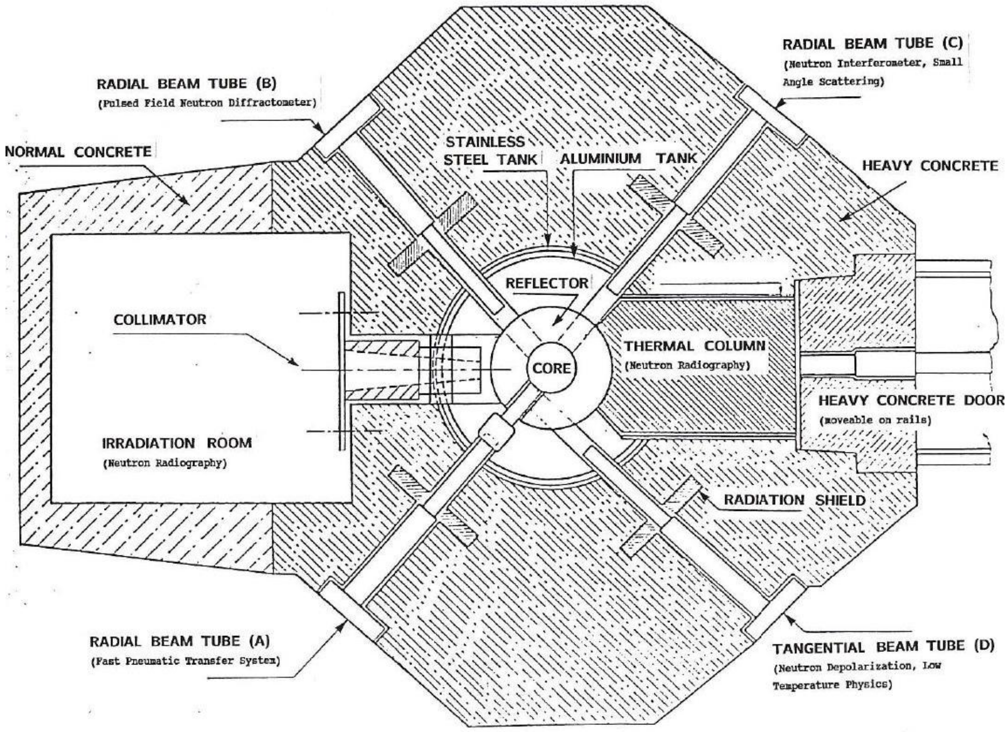

Figure 1: Top view of the TRIGA out-core irradiation facilities [2]

The thermal column liner is a seal-welded container fabricated from a 12.7 mm thick aluminium plate. The outer portion is embedded in the concrete shield and the inner portion is welded to the aluminium tank. The exterior surfaces of the thermal column, which are in contact with concrete, are coated with plastic for corrosion protection. The portion welded to the aluminium tank extends to the graphite reflector and matches the contour of the reflector over a 100-degree angle. The horizontal centreline coincides with the active core lattice. In vertical plane, the column extends approximately 33 cm above and below the reflector, with the central lines of the column and the reflector coinciding [2].

The aluminium container is opened toward the reactor room. Blocks of AGOT nuclear grade graphite occupy the entire void. The dimensions of each block are approximately 10.2 x 10.2 cm in cross-section and 127 cm in length.

The four beam tubes penetrate the concrete shield and the aluminium tank and pass through the reactor tank water to the reflector. These tubes provide neutron beams and gamma radiation for a variety of experiments. These tubes also provide the irradiation facilities for large specimens (up to 15 cm) in a region close to the core. Three of the beam tubes are oriented radially with respect to the centre of the core while the fourth tube is tangential to the outer edge of the core. Two of the radial tubes terminate at the outer edge of the reflector assembly but are aligned with the cylindrical void in the reflector graphite. The third (i.e. BT-A) tube, specifically developed for neutron activation, penetrates into the graphite reflector and terminates at the inner surface of the reflector, just at the outer edge of the core. In order that the beam-tube voids in the reflector graphite pass beneath the rotary specimen rack, their horizontal centrelines are located 7 cm below the centreline of the core [2].

The neutron flux calculations were performed in this tube (BT-A) using MCNP5. A barite concrete cylinder (length 75 cm, diameter 14 cm) was inserted into this BT-A. By applying the MCNP5 model, the thermal flux at each 12 cm was calculated and presented in this paper.

2 MCNP MODELLING

An detailed MCNP model of the TRIGA Mark II reactor core was developed at the Atominstitute by using a 3-D continuous-energy Monte Carlo code [3]. That model included all components of the core, i.e. all three types of fresh fuel elements, three control rod elements, graphite reflector elements, surrounding annular reflector with the rotary specimen rack well and reactor tank.

MCNP5 uses point-wise cross-section data. For neutrons, all reactions given in a particular cross-section evaluation (such as ENDF/B-VI) are accounted for. Thermal neutrons are described by both the free gas and S (α,β) models [3]. The continuous energy cross-section data and $S(\alpha,\beta)$ scattering functions from ENDF/B-VI.5 library were used. To perform the calculations outside the TRIGA core (e.g. on thermal column and in BT-A), this already developed model was extended to the biological shielding of the reactor with all four beam tubes, collimator and thermal column as described in figure 2.

To avoid computational costs and to incorporate the accurate and complete information for the individual MC particle tracks, the MCNP5 computer code provides a SSW (surface source writing) capability to generate a WSSA-format file [4]. This source-file contains all required messages of individual particle tracks crossing a given surface. This capability was employed to create WSSA-file on the outer surface of the core. The MCNP model with and without core (surface source capability), all four beam tubes, the thermal and the neutron radiography collimator is shown in figure 2.

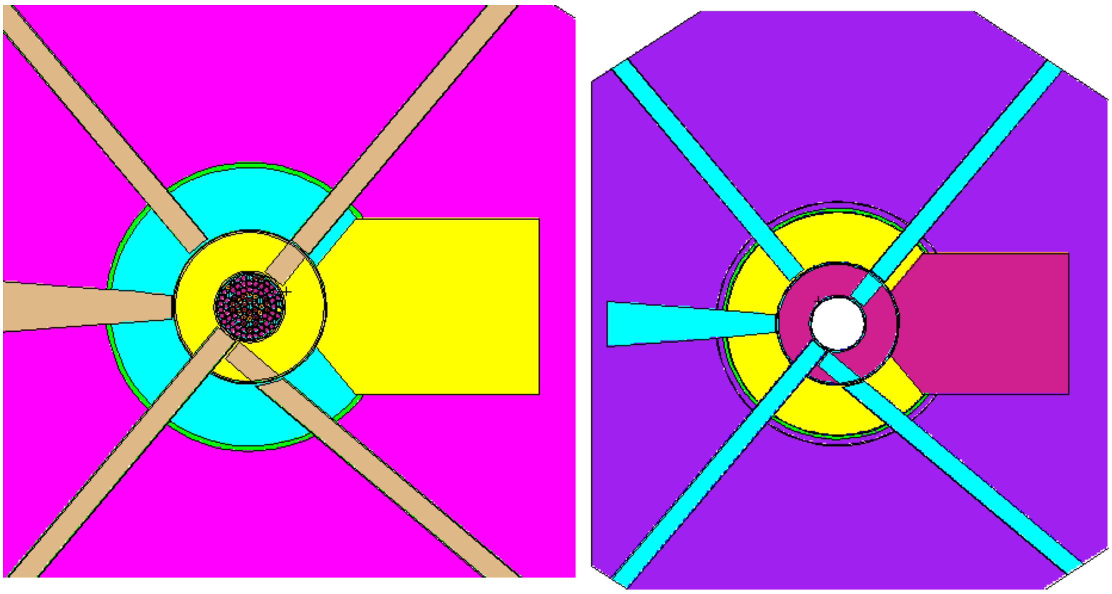

Figure 2: Top view of the extended model MCNP of the TRIGA Mark II with and without the reactor core

2.1 Thermal column Model

To calculate the thermal column flux density, superimposed mesh tally was invoked into the model by FMESH card using appropriate settings of available geometrical and other options. The FMESH card allows the user to define a mesh tally superimposed over the problem geometry. Results are written to a separate output file, with the default name MESHTAL. By default, the mesh tally calculates the track length estimate of the particle flux, averaged over a mesh cell, in units of particles/cm^2, and is normalized to “per starting particle”, except in

KCODE criticality calculations. By this card, the results are written to a separate output file, with the default name MESHTAL. The mesh tally results (i.e. the track length estimation of the particle flux), are averaged over a mesh cell in units of particles/cm^2.

For thermal flux analysis, FMESH was applied to one face of the thermal column where measurements were performed at 13 different locations. The coarse mesh for each location was selected so that one graphite block is represented by one tally mesh. The MCNP5 YZ-view of the applied mesh tally and actual view of the thermal column can be seen in figure 3.

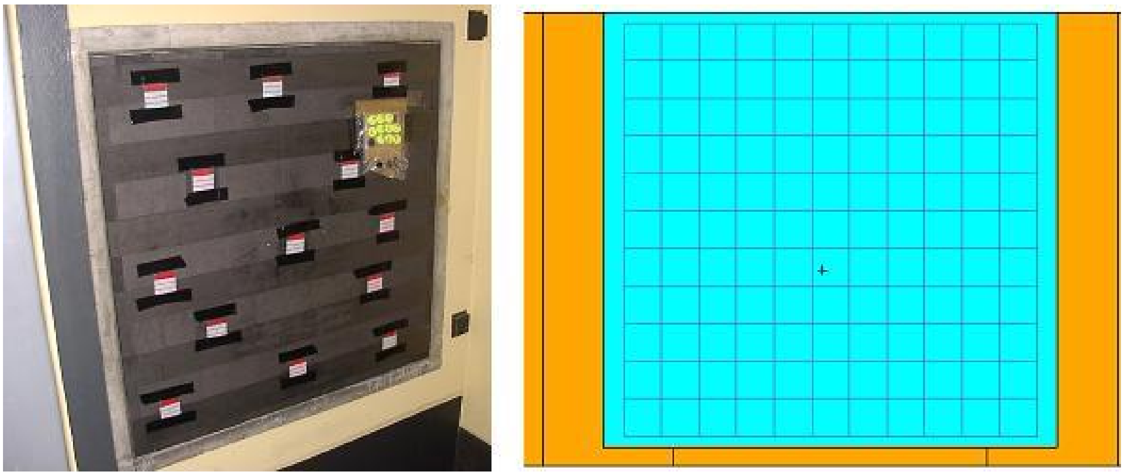

Figure 3: TRIGA Mark II- actual [8] and MCNP5 mesh tally plot (YZ-view) of the thermal column

2.2 Beam Tube model

The thermal flux measuring experiments have been planned in near the future (December 2009). According to these planned experiments, the thermal flux density in the BT-A region will be measured and calculated. This paper presents the MCNP model of BT-A. The combination of neutron current integrated over surface tally i.e. F1 with “Sd”, “Sn” and “En” cards was applied. This combination computes the results into probability of single neutron interaction per square centimetre (ns/cm$^{2)}$. These results then need to be normalized to some experimental values just to convert the F1-result into a physical quantity (i.e. neutron flux density). The F1-tally represented by the Monte Carlo weight crossing a surface in specified bins according to the experimental positions in BT-A was applied to compute the thermal flux density distribution in BT-A. The MCNP XY-view of the BT-A simulation has been shown in figure 4. These results were then normalized to the experimental results of the central block of the thermal column.

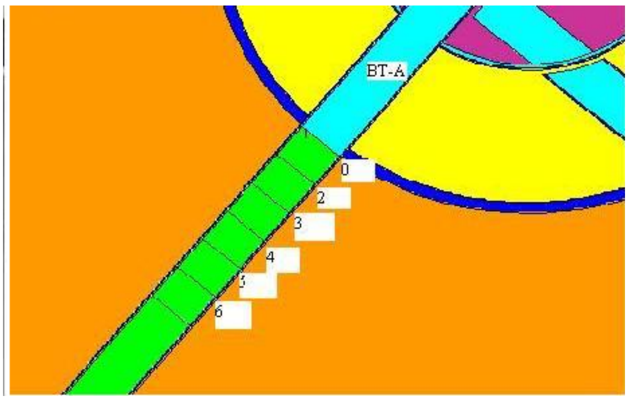

Figure 4: MCNP (YZ-view) of beam tube A

This model uses barite concrete in BT-A and magnetite concrete for the biological TRIGA shielding. The MCNP material compositions for both concretes have been described in table 1[6]. For each element its ZAID, weight (wt.) fraction and atom (at.) fraction have been described in table 1.

Table 1: Barite and Magnetite material composition employed in MCNP model

MCNP material ID		**Barite Concrete density 3.35 g.cm^{-3}**		**Magnetite Concrete density 3.53 g.cm^{-3}**	
Element	ZAID	wt. fraction	at. fraction	wt. fraction	at. fraction
H	1000	0.003585	0.109602	0.003000	0.083258
O	8016	0.311622	0.600193	0.320000	0.559481
Mg	12000	0.001195	0.001515	0.006000	0.006905
Al	13027	0.004183	0.004777	0.029000	0.030066
Si	14000	0.010457	0.011473	0.035000	0.034860
S	16000	0.107858	0.103654	0.010700	0.009335
Ca	20000	0.050194	0.038593	0.007000	0.004886
Fe	26000	0.047505	0.026213	0.205000	0.252957
Ba	56000	0.463400	0.103984	-	-
P	15031	-	-	0.001700	0.001535
Ti	22000	-	-	0.028000	0.016363
Mn	25055	-	-	0.000700	0.000356

3 EXPERIMENTS

3.1 Thermal Column

The thermal neutron flux measurement was performed at the thermal column of the TRIGA research reactor by using the gold foil method [8]. Bare and cadmium-covered gold foils with a diameter of 5 mm and an average weight of 0.0084 g were used for this experiment.

For this purpose the concrete door of the thermal column was opened and the foil detectors were placed in the thermal column (13 positions). The air gap between the thermal column and the concrete door was about 2 cm. The reactor operated at 100 kW, and for the second measurements at 250 kW maximum power, the gold foils (bare and Cd-covered) were irradiated for a duration of 10 min. After reactor-shutdown, the gold foils were collected and the activities were measured using a 4π beta counter system. The positions for measurements are shown in figure 3.

4 RESULTS AND DISCUSSIONS

4.1 Thermal column

The mesh tally was applied to (11x11) matrix of the thermal column excluding the upper most and lower most graphite layers. The thermal flux was measured on (2x2), (2x6), (2x10), (4x3), (4x8), (6x2), (6x6), (6x10), (8x3), (8x9), (10x2), (10x6) and (10x10) positions and compared with corresponding MCNP results. This comparison is shown in table 2 and figure 5.

Table 2: Comparison of experimental thermal flux with MCNP5 results in ($cm^{-2}s^{-1}$)

Meas. position	Exp. thermal flux	Cal. thermal flux	Cal./Exp.
(2,2)	1.6932E+07	2.0755E+07	1.226
(2,6)	1.9255E+07	3.7084E+07	1.926
(2,10)	1.5848E+07	2.0693E+07	1.306
(4,3)	4.5127E+07	5.0593E+07	1.121
(4,8)	5.6064E+07	6.0810E+07	1.085
(6,2)	2.9609E+07	4.4650E+07	1.508
(6,6)	8.3050E+07	8.3050E+07	1.000
(6,10)	4.5768E+07	4.4126E+07	0.964
(8,3)	4.4985E+07	5.2957E+07	1.177
(8,9)	4.4196E+07	6.3907E+07	1.446
(10,2)	2.0561E+07	2.2168E+07	1.078
(10,6)	3.2400E+07	4.2623E+07	1.315
(10,10)	1.6723E+07	2.2406E+07	1.340

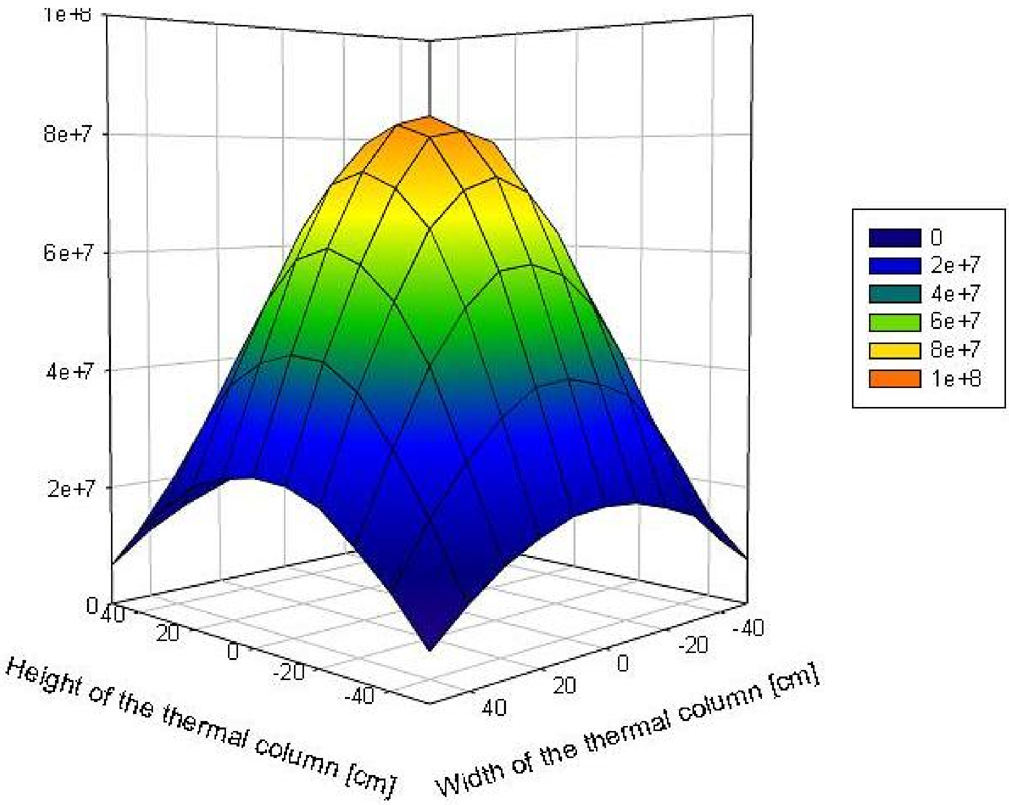

Figure 5: MCNP5 thermal flux ($cm^{-2}.sec^{-1}$) distribution in thermal column

Figure 5 reflects the experimental results. The thermal flux density has maximum value at its centre while decreases radial direction. The MCNP5 model uses nuclear grade graphite of 1.6 $g.cm^{-3}$ while the actual density may vary from 1.6 $g.cm^{-3}$. The differences may also be because of the different structure of the thermal column in the modelling and in the experiments. The MCNP5 results can be improved by the confirmed density of the graphite and block structure of the thermal column.

4.2 Beam Tube-A

The material composition of the TRIGA biological shielding is under investigation. However the model is also applied to BT-A, using a standard magnetite concrete composition [6], and figure 6 presents these results for the thermal flux density. It has been planned that the model will be rerun using the actual material composition for the biological shield when it is available. For the validation of the model in this BT-A region these updated results will be compared with those of the experimental observations.

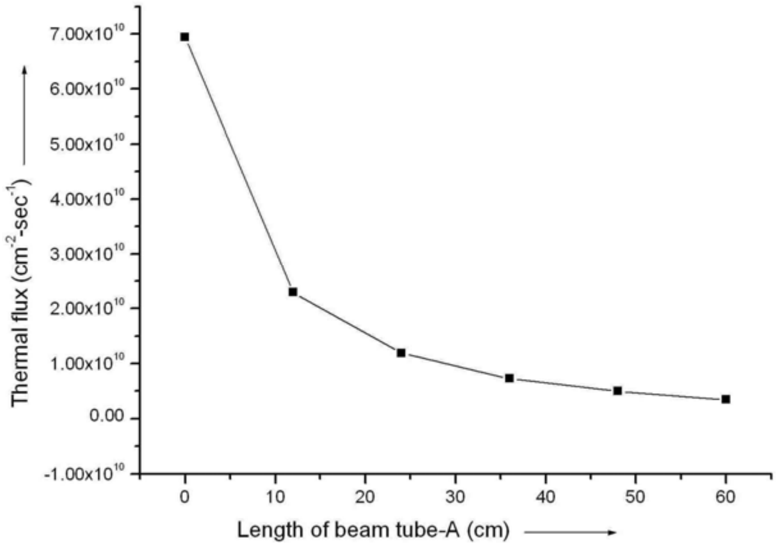

Figure 6: Calculated thermal flux along the length of BT-A using barite concrete barrel

CONCLUSION

To calculate the shielding and other nuclear engineering related problems outside the reactor core, an already developed MCNP model of TRIGA core was extended to the thermal column, the biological shield including the radiography collimator and all four beam tubes. The results of this extended model for thermal column have been compared with the experimental observations, and yielded fair agreement. Whereas for the BT-A region, standard magnetite concrete composition of biological shield was used in the model. The actual concrete samples are under investigation and this composition is planned to be employed to recalculate thermal flux in the BT-A region. To confirm this thermal flux distribution in the BT-A, a neutron measuring experiment will be performed. After the confirmation of results in the BT-A region, this extended model will be applied for other TRIGA experimental facilities.

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