# Reactor-physical calculations using an MCAM based MCNP model of the Training Reactor of Budapest University of Technology and Economics

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*A new MCNP model of the Training Reactor of BME was developed using SuperMC/MCAM, a novel CAD to MCNP converter code and calculations were made to present the independent working capabilities of the Vietnamese participants of a course called: Application of Monte Carlo methods in reactor physics, which was held at BME NTI. The following calculations were carried out: $k_{eff}$ calculations and finding the critical rod positions, reactivity worth calculations of the control rods, $\beta_{eff}$ calculations, vertical and horizontal neutron flux distribution calculations, determination of water level and water temperature dependence of $k_{eff}$. Comparison of the calculated results with experimental data was also done.*

## Introduction

In 2011, the Vietnamese and the Hungarian government signed an agreement about cooperation in nuclear education and training. According to this agreement, in 2016 a group of Vietnamese experts (6 people) participated in a training as students at the Institute of Nuclear Techniques (NTI), Budapest University of Technology and Economics (BME) in Hungary. The title of the project was "Application of Monte Carlo methods in reactor physics" which covered hands-on trainings in the following topics: fundamentals of reactor physics, fundamentals of modelling using CAD drawings and the evaluation of the results. Besides the training, participants also spent two weeks at the Paks Nuclear Power Plant (NPP) to get personal experiences about the application of nuclear techniques in an industrial scale facility. The 12 weeks training ended up with a project work and a project report which demonstrates the independent working capabilities of the students. This article covers the details of this project work. The project tasks were the following: building the Monte Carlo N-Particle code (MCNP) [1] model of the Training Reactor of BME based on SuperMC (MCAM) software [2-3]- developed by FDS Team, China - then calculating $k_{eff}$ and flux distribution with different control rod positions and with various water levels and temperatures of the reactor core.

The pool-type nuclear reactor of the university - Training Reactor (TR) of BME - is in operation since 1971. The core is built of EK-10 fuel assemblies with 10% enrichment. The maximum thermal power is 100 kW, the maximum thermal neutron flux is $2.7 \times 10^{12}$ n/cm$^2$s. The main purpose of the facility is to support education in nuclear engineering and physics; however, extensive research work is carried out as well. Neutron and gamma irradiation can be performed using 20 vertical irradiation channels, 5 horizontal beam tubes, two pneumatic rabbit systems and a large irradiation tunnel. The facility has physical and radiochemical laboratories and a hot cell too [4].

For simulating a nuclear reactor a wide variety of software is available including: SuperMC, Serpent [5], Scale [6], MCNP, etc. In this work, MCNP6 (version 1.0) was used for calculations. MCNP is a very powerful and versatile tool for particle transport calculations. It can be used for transport calculations of neutrons, photons, electrons and many other

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particles. Transport of neutrons is of special interest for a reactor physicist. MCNP code can be used for: criticality calculations, activation calculations, particle fluence and spectra calculations, burn-up and shielding calculations, etc. For MCNP calculations precise material composition and geometrical information are needed as well as the precise set up of quantities of interest (hereinafter tallies). Nevertheless, to build a precise and also complicated model, it requires a lot of time to develop the model. To handle the problem of the increased development time, MCAM interface software has been used which can convert complex CAD models into MCNP input. With MCAM, we significantly diminish the need of manpower and augment the reliability of calculation model. CAD models are possible to be converted automatically together with physical properties into MCNP input, but the geometrical restrictions of MCNP should be taken into account.

# MCNP Model

The MCNP model of the TR was built and converted by using MCAM, version 2.3.5. The reactor model was created in user interface of MCAM using basic geometric volumes such as: block, cylinder, etc. Fuel elements and other repeated structures, with same geometry, were created by duplicating entities, using "copy" or "rectangle array" function. In the reactor model, the moderator is water and the horizontal reflector material is graphite. The reactor core consists of 24 fuel assemblies with (9 types of different configuration) 369 fuel elements, 2 control rods, 2 safety rods and some graphite elements. Fuel element is $UO_2$ in a Mg matrix, with 10% enrichment and 5.45719 g/cm$^3$ density, which consists 8 g of $^{235}U$, 71.936 g of $^{238}U$, 63.6 mg of $^{234}U$, 13.029 g of Mg, 0.37 g C and 0.42 g of $O_2$ (not included in the $UO_2$). 6 ppm natural boron was also included in the fuel. For every calculations the ENDF/B-VII cross-section libraries were used. After completing the MCAM model, the whole reactor was converted into an MCNP input file.

Fig.1 shows the 3D model of the TR created by MCAM with two cross-sectional views of the reactor core. The height of outer water is 5.7 m, the thickness of heavy concrete is 1.1 m and the thickness of normal concrete is 0.9 m. The cross-sectional view of reactor core, fuel assembly, fuel element and control rods are shown in Fig. 2. The MCNP model of the TR was used to estimate some reactor physics parameters:

1.) The $k_{eff}$ factor was calculated with different rod positions, with different water levels and with different temperatures of the water. The change in water temperature was taken into consideration by only adjusting the density of water,

2.) The neutron flux distribution was calculated vertically in the core at E6 position and air pins at D5, G5. A horizontal flux distribution was also calculated in the midplane of the core through the "E" column. The neutron flux distributions were calculated by determining the flux averaged over a cell (tally F4) with segmentation cards.

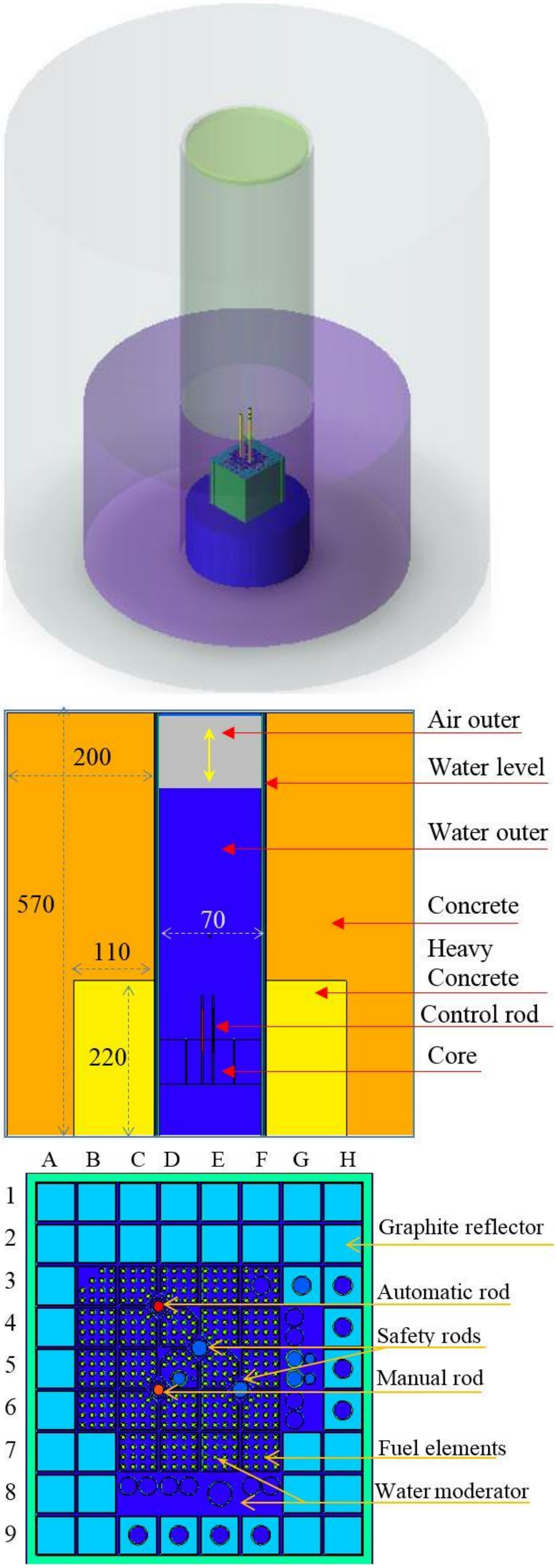


*Fig. 1: 3D model and cross sections of the TR (measure in cm)*

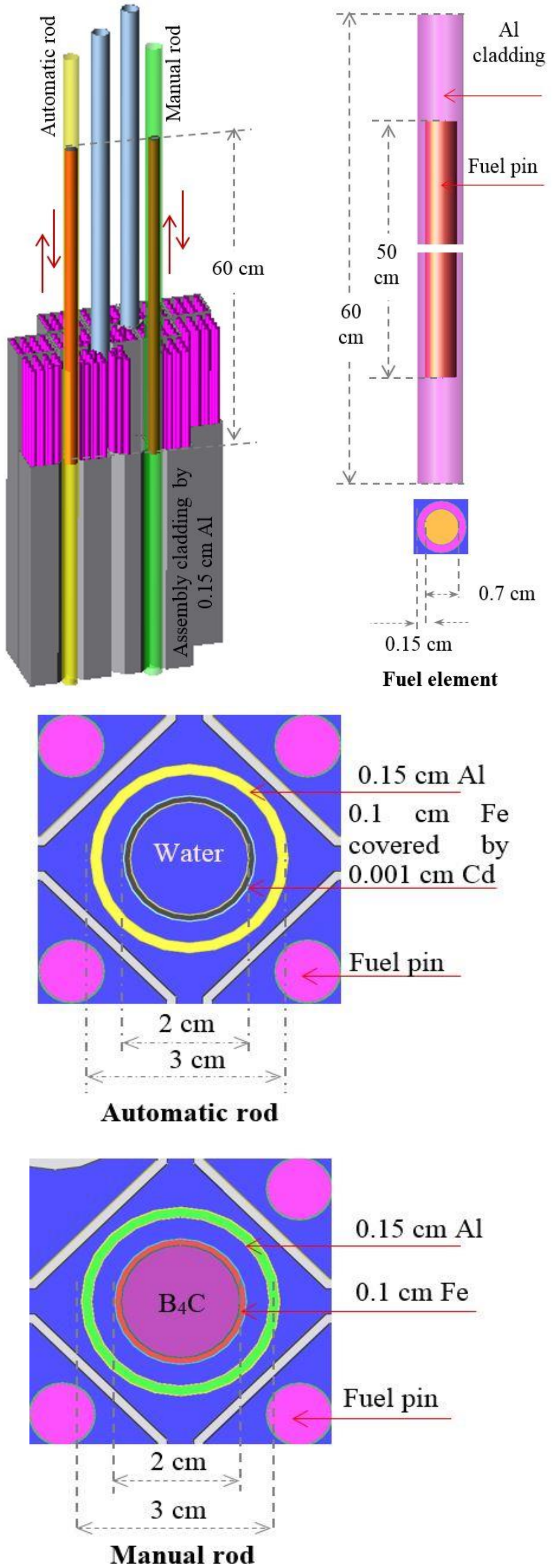


*Fig. 2: Cross-sectional diagram of a fuel element, control rods, and an exploded view of the core.*

# Results and discussions

## *Criticality Simulation*

We have found the critical state of the reactor by changing the control rod positions. When automatic and manual control rods were both in positions of 42 cm, simulation resulted that the reactor reached critical state, with total effective multiplication factor $k_{eff}$ = 1.00276 ± 0.00029. This critical state is regarded as a standard state for other calculations in this paper. As it is well known, the delayed neutron fraction ($\beta_{eff}$) has an important role for reactor operation. Therefore, we also calculated $\beta_{eff}$ by using KOPTS card in MCNP6 with ENDF/VII library in this case. The MCNP calculations were run with 4,700,000 active histories and value of $\beta_{eff}$ was 0.00705 ± 0.00035. It differs from the nominal value (0.00798) about 11.65%, we think the reason of the difference is the different material composition and the simplifications in the geometry of our MCNP model.

In order to benchmark our model, we simulated the reactor in two cases:

1.) both manual and automatic control rod are inserted into the core;

2.) both manual and automatic control rod are pulled out the core. The obtained values of $k_{eff}$ and reactivity are shown in Table 1. The total control rod reactivity worth: difference between the excess reactivity and the minimum reactivity when all control rods are fully inserted, is:

$$\Delta\rho = \rho_1 - \rho_2 = 0.03311 \quad (1)$$

*Table 1. $k_{eff}$ value with all control rods in the core and without control rods*

| | *1- all control rods pulled out* | *2- all control rods inserted* |
|---|---|---|
| $k_{eff}$ | 1.01084 ± 0.00037 | 0.9781 ± 0.00038 |
| $\rho$ | 0.01072 (~1.5$) | -0,02239 |

The reactivity of the reactor is approximately 1.5$ which means our calculation is in a good agreement with the results of the 1980's measurement which was done with a relatively fresh core. The dependence of $k_{eff}$ on water levels was also analysed, the results are shown in Fig 3. The reactor reached criticality (by applying critical rod positions) when the water level was larger than 60 cm (from the bottom of fuel rod). Besides, if the water level is more than 1 m, the values of $k_{eff}$ will saturate. These results are in good agreement with the expectations because after a few diffusion length distances the presence of water does not effect on the reactor [7].

As it is expected, the water temperature and thus the density change affects reactivity. As the temperature of water increases the density of it decreases, thus the reactivity will decline at average of -0.992 cent/°C, if we assume that the value of $k_{eff}$ depends linearly on temperature (see Fig 4).

## *Neutron Flux distribution in the core*

The vertical neutron flux distribution was calculated at E6 position because in this position measured data are also available, therefore we can directly compare our results. Calculated data are obtained in the fuel material, in the cladding and in a Dy-Al wire which was inserted to the top left water pin of E6 (see Fig. 5, 6, 7 respectively) to reproduce the measurement conditions (see Fig. 8.).

 

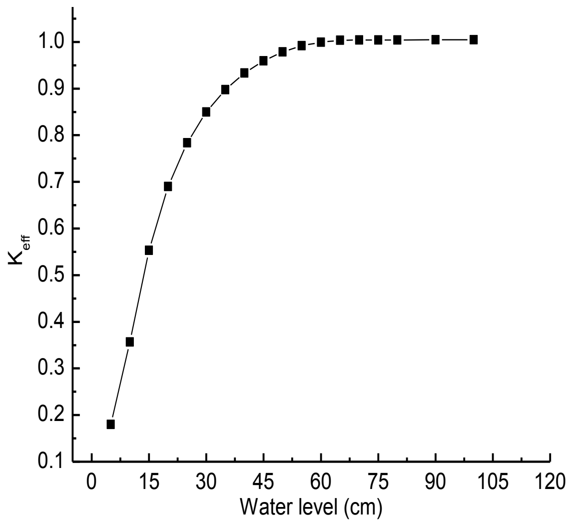


*Fig.3: The dependence of $k_{eff}$ on water level*

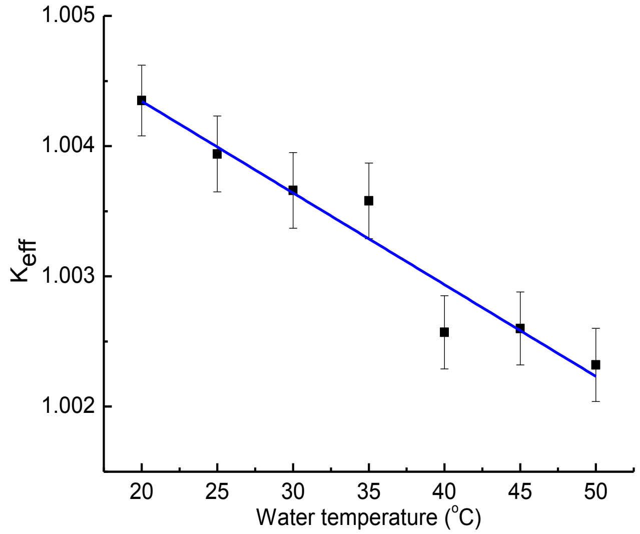


*Fig.4: The dependence of $k_{eff}$ on water temperature*

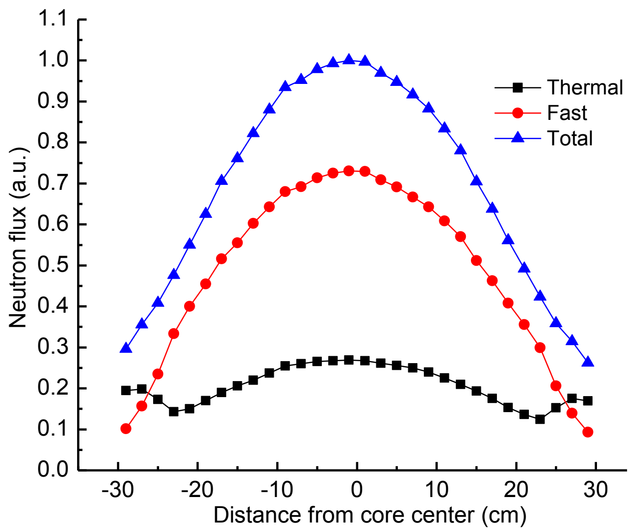


*Fig.5: Vertical flux distribution in fuel cladding*

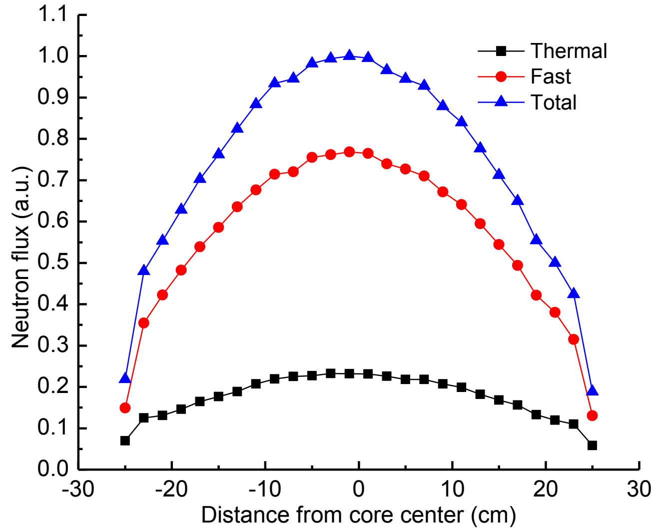


*Fig.6: Vertical flux distribution in fuel rod*

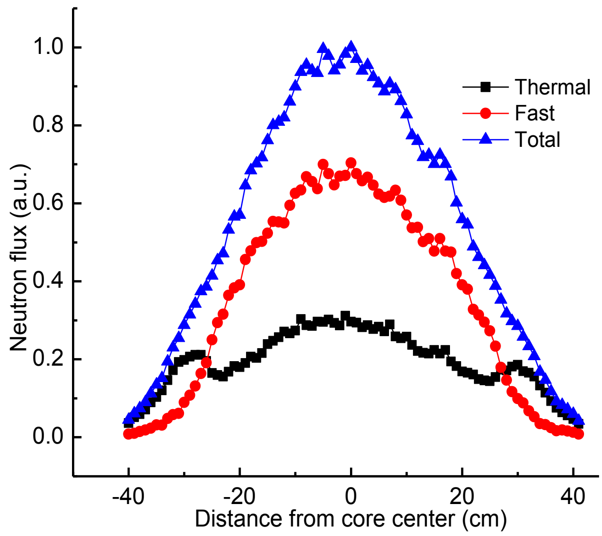


*Fig.7. Neutron flux distribution in Dy-Al wire*

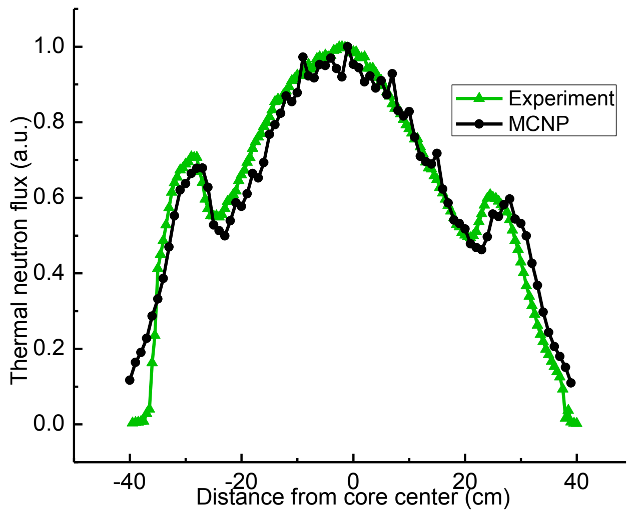


*Fig.8. Vertical thermal flux distribution in Dy-Al wire, experimental and simulated results.*

The peaks of the thermal neutron flux are also called reflector peaks. These peaks can be clearly seen in the bottom and the top of the reactor (see Fig. 7, 8). The reflector peaks appear in the water due to the thermalization of the escaping fast neutrons from the core. By reducing neutron leakage, the reflector increases $k_{eff}$ and reduces the amount of fuel necessary to make the reactor critical. The reflector peaks at the bottom and the top of reactor are not the same which can be explained by presence of the control rods above of the core in the vicinity of the Dy-Al wire. One can see that the vertical neutron flux distribution in the fuel pin and in the cladding have the shape of a cosine, which is in a good agreement with the reactor physical expectations of a slab shaped reactor [7]. The maximum values of neutron flux at E6 position show in Table 2.

*Table 2. The maximum values of neutron flux at E6 position*

| | Thermal neutron ($cm^{-2}.s^{-1}$) | Error (%) | Fast neutron ($cm^{-2}.s^{-1}$) | Error (%) | Total ($cm^{-2}.s^{-1}$) | Error (%) |
|---|---|---|---|---|---|---|
| In fuel | $1.60\times10^{12}$ | 2.92 | $5.32\times10^{12}$ | 1.71 | $6.92\times10^{12}$ | 1.30 |
| In cladding | $1.31\times10^{12}$ | 3.01 | $3.54\times10^{12}$ | 2.11 | $4.85\times10^{12}$ | 1.92 |
| In Dy-Al wire | $1.85\times10^{12}$ | 2.8 | $4.18\times10^{12}$ | 2.02 | $5.95\times10^{12}$ | 1.66 |

Also, we have calculated vertical neutron flux distribution in the pneumatic systems at D5 and G5 position (see Fig.9, 10 respectively). The calculated (C) maximum values of thermal neutron flux at D5 and G5 respectively are $2.88\times10^{12}$ n/cm²/s and $1.76\times10^{12}$ n/cm²/s. In the measurements (M) the value of the thermal neutron flux in these positions are $3.338\times10^{12}$ n/cm²/s and $2.51\times10^{12}$ n/cm²/s respectively and their standard deviation is 3.2% and 3.4% respectively which shows a slight underestimation. $C/M_{D5}$: 0.862, $C/M_{G5}$: 0.701

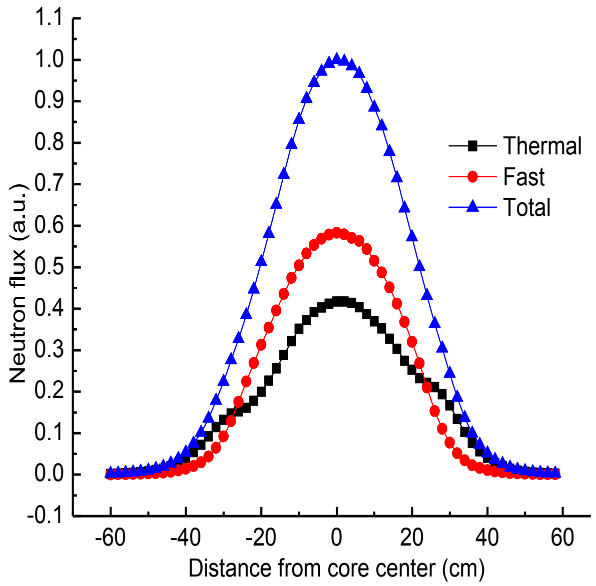


*Fig.9: Vertical flux distribution in air pin at D5*

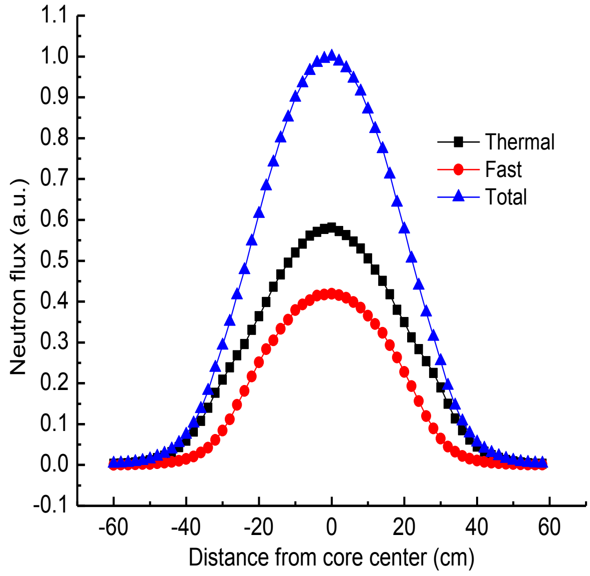


*Fig.10: Vertical flux distribution in air pin at G5*

Horizontal flux distributions in the midplane of the core through the column "E" (second fuel pin column) are shown in Fig.11 and Fig.12. Because of arrangement of the fuel pins in the core, the distributions are not perfectly cosine shaped.

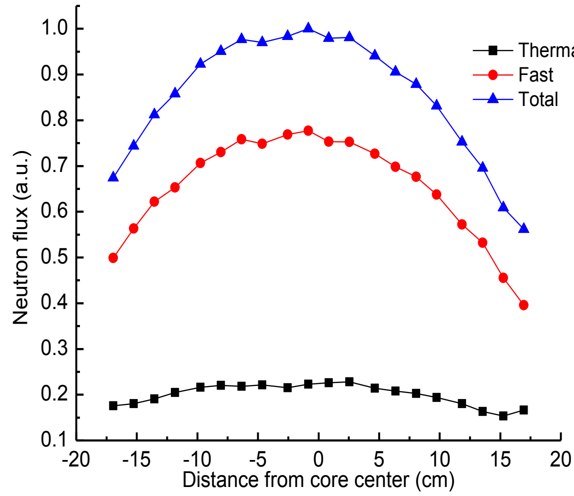


*Fig.11: Horizontal flux Distribution in the midplane of the core in the cladding, through the column "E"*

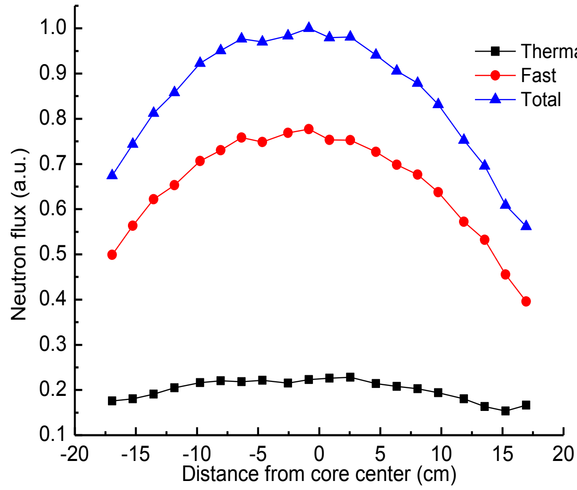


*Fig.12: Horizontal flux Distribution in the midplane of the core in the fuel, through the column "E"*

## Conclusion

A new MCNP model of the Training Reactor of BME was developed using MCAM, a novel CAD, to MCNP converter code and calculations were made to present the independent working capabilities of the Vietnamese participants of a course called: Application of Monte Carlo methods in reactor physics. The following simulations were carried out: $k_{eff}$ calculations and finding the critical rod positions, reactivity worth calculations of the control rods, $\beta_{eff}$ calculations, vertical and horizontal neutron flux distribution calculations, determination of water level and water temperature dependence of $k_{eff}$, comparison of the calculated results with experimental data which resulted in good agreement.

## References


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