MONTE CARLO SIMULATION OF THE 3 MW TRIGA MARK II BENCHMARK EXPERIMENTS

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ABSTRACT

This study deals with the neutronic analysis of the current core configuration of a 3 MW TRIGA MARK II research reactor at Atomic Energy Research Establishment (AERE), Savar, Dhaka, Bangladesh and validation of the results by benchmarking with the experimental, operational and available Final Safety Analysis Report (FSAR) values. The three-dimensional continuous-energy Monte Carlo code MCNP4C was used to develop a versatile and accurate full-core model of the TRIGA core. The model represents in detail all components of the core with literally no physical approximation. All fresh fuel and control elements as well as the vicinity of the core were precisely described. Continuous energy cross-section data from ENDF/B-VI and ENDF/B-V and S(α,β) scattering functions from the ENDF/B-IV library were used. The validation of the model against benchmark experimental results is presented. The MCNP predictions and the experimentally determined values are found to be in very good agreement, which indicates that the Monte Carlo model is correctly simulating the TRIGA reactor.

INTRODUCTION

A 3 MW TRIGA MARK II research reactor was commissioned at the Atomic Energy Research Establishment, Savar, Dhaka in 1986 and it went critical on 14th September, 1986. An increasing interest for reactor calculations by research reactor user has been observed in recent years. A variety of advanced and sophisticated computer codes for research reactor calculations have been developed or adapted for small and personal computers. We have many available experimental data, reactor operational data and the experience of the investigators. The diffusion theory model using multigroup cross section libraries analyzed some of the reactor experimental data. [1] In most cases, it was not possible to make valid comparisons because of the inherent difficulties in using computer codes to model experimental or operational conditions. Examples of such conditions are the effects of depletion, fission product poisoning, temperature effects and control rod positions. Furthermore, the differences between the codes and the operational data are compounded by the various geometric and analytical approximations commonly associated with the

deterministic codes e.g. one dimensional, two dimensional, infinite medium, homogenization and multigroup cross section treatment etc. These differences in some cases have been significant and it cannot be determined whether the discrepancies are a result of the differences in the models or the limitations of the deterministic codes. So, application of these codes for criticality analysis require qualification from other independent codes which to some extent are free from the above mentioned shortcomings to determine the range of their validity. This is especially important for application that has a direct impact on reactor safety and operation. Because of this need for independent assessment, the Monte Carlo technique can be beneficial. For the purpose of modeling the TRIGA MARK II reactor, the general purpose 3-D *M*onte *C*arlo *N*-*P*article code MCNP4C [2] was chosen because of its general geometry modeling capability, correct representation of transport effects and continuous energy cross sections. The later is the most significant because this eliminates the need for collapsing multigroup cross sections for the reactor model and in-core-experiments.

The scope of this study is to present the Monte Carlo computer code MCNP4C simulation of the TRIGA MARK II benchmark experiment. To reduce possible systematic errors due to inexact geometry simulation, a very thorough 3- dimensional model of the TRIGA reactor was developed. All fresh fuel, control rod, and other elements (e.g., source elements, graphite-loaded elements) models were prepared in order to be able to simulate any possible fresh core. The MCNP4C input was prepared in such a way that a very quick setup of any desired core configuration with an adequate position of all control rods is possible. Continuous energy cross-section data from ENDF/B-VI and ENDF/B-V and S(α,β) scattering functions from the ENDF/B-IV library were used in the calculations.

An essential aspect of developing an accurate reactor physics model is validation. The accuracy of both the neutron transport physics as represented in MCNP and the user-defined model must be assessed. However, even though MCNP has been proven to simulate the physical interactions correctly, that does not mean that the model of TRIGA will provide accurate answers. Therefore, to build confidence, all the neutronic parameters including effective multiplication factor, in-core and ex-core neutron flux and benchmarking of reactivity experiments were performed for the fresh core with the MCNP4C modeling of TRIGA reactor to supplement and compare MCNP predicted values with the previously obtained results from diffusion theory codes and experiments.

MCNP MODELING OF TRIGA

The TRIGA core consists of 100 fuel elements arranged in a concentric hexagonal array within the core shroud. The reactor is a light water cooled, graphite-reflected one, designed for continuous operation at a steady-state power level of 3000 kW (thermal), and for routine pulsing with reactivity insertions of up to 1.4% δk/k ($ 2.00). The reactor core and reflector assembly is located at the bottom of a 2 m diameter aluminum tank 8.2 m deep (Fig. 1). Approximately 6.4 m of water above the core provides vertical shielding. The reactor core consists of a lattice of fuel-moderator elements, graphite dummy elements (optional), and control rods. Figure 2 shows the cross sectional view of the present core arrangement of the reactor, which was achieved on October 9, 1986 during reactor start-up at full power operation. In the present core configuration, there are 95 fuel-moderator elements, 18 graphite dummy elements, 5 control rods, 1 transient control rod and 1 pneumatic transfer

tube and 6 source locations. Elements are arranged in seven concentric rings: A, B, C, D, E, F, and G, having 1, 6, 12, 18, 24, 30, and 36 locations, respectively. Each location corresponds to a hole in the aluminum upper grid plate of the reactor. Six more holes near the G-ring of the grid plate locate and provide supports for the neutron source holder at alternate positions. The spaces between the rods are filled with waters that act as coolant and moderator.

The TRIGA MARK II research reactor was modeled in full three-dimensional detail to minimize the number of approximations. The repeated structure capability of MCNP was used to create a full core, three-dimensional model of TRIGA. [2] The primary tool used in the full core model is the fill array matrix specification. This tool makes filling of the elements (fuel rods, control rods, graphite dummy rods, transient rod, pneumatic transfer tube and central thimble) in the TRIGA core lattice structure. This fill array is a 17 X 17 elements square matrix that designates the universe that fill the various lattice positions in the reactor tank. In case of hexagonal lattice geometry, the radius of the equivalent unit cell in terms of the lattice pitch (i.e., the distance between fuel pin centerlines in the lattice) was considered to be 0.525 times of the pitch. [3] The universe is defined by the integers on the fill specification on a cell card. The TRIGA MARK II lattice can be represented as a hexagonal prism, solids with eight faces, with a 0 dimension in the third axis. [In the fill matrix, 3 designates the central thimble, 1 - fuel rods, 2 - control rods, 4 - graphite dummy elements, 5 - transient rod, 6 - pneumatic tube, 8 – source locations and 7 - water locations.]

The fuel elements were modeled explicitly specifying the detailed structure of the rod (Fig. 3) to eliminate any homogenization effects. The power level of the TRIGA reactor is controlled with six control rods: a regulating rod, four shim/safety rods, and a transient rod. The control rods were explicitly modeled along the active length containing 3 vertical sections of boron carbide, fuel follower and a void region with the exception that the fuel follower section of the transient rod was replaced through a void region. The control rods travel vertically a distance of approximately 38.1 cm between their fully withdrawn and inserted positions, as shown in Fig. 4. The safety/transient rod is a sealed, 94.996 cm long by 3.18 cm diameter aluminum tube containing solid boron carbide as a neutron absorber. Below the absorber is an air-filled follower section. The absorber section is 38.1 cm long and the follower is 55.753 cm long. All the control rods were explicitly modeled along the active length with the exception of the drive mechanism. The central thimble was considered to be filled with water in the model and the pneumatic tube was assumed to be void. The graphite dummy elements are of the same general dimensions and construction as the fuel-moderator elements, except these elements are filled entirely with graphite. Six source locations in the upper grid plate (1.58 cm diameter) were also modeled. The model was extended up to 53.34 cm radially containing the graphite reflector and lead shield and approximately 110 cm above and below the core centerline, which was more than sufficient to account for the neutron returning from the H_2O coolant above and below the core (Fig. 5). An annular well on the inside diameter in the top of the graphite reflector that provides for the rotary specimen rack was also modeled. This well is lined with aluminum and is an integral part of the aluminum reflector housing. The outer wall of the reflector housing extends 0.8 m above the top of the core that serves as upper plenum was also modeled. A plenum welded to the bottom of the reflector was also modeled. The radial and tangential beam ports were also modeled (Fig. 6). An aluminum tube that forms a housing for the piercing radial beam port pierces the graphite and the outer surface of

the aluminum can. Two additional holes penetrate the graphite, one for a radial beam tube and one for the tangential beam tube. These holes do not pierce the aluminum can, and their respective beam tubes terminate in line with them, just outside the reflector can. Modeling of the reactor extended up to the reactor tank consists of an aluminum vessel (0.635 cm thick) installed in the reactor shield structure. It has an inside diameter of approximately 2 m. Thus, it has been possible to describe the geometry of the TRIGA reactor explicitly without resorting to any approximation at all. A summary of the principal design parameters and material composition data for the TRIGA MARK II research reactor is given in Table I. All geometric and material data are taken from the fabrication and shipment documentation, provided by the reactor manufacturer General Atomics.

The primary coolant of TRIGA is non-pressurized and experiences a room temperature of ~ 27^{o}C. Based on these operating conditions, most of the continuous energy neutron interaction data from the ENDF/B-VI cross-section library for 300K evaluations were used in the MCNP calculation. [4] Some data from ENDF/B-VI evaluation were not available: the neutron cross-section data for ^{nat}Zr, ^{nat}Mo, ^{nat}Cr, ^{nat}Fe, ^{nat}Ni, ^{nat}Si, and ^{nat}Mg were taken from the ENDF/B-V library. [5] There are two standard methods in MCNP for treating thermal neutron interactions: the free gas thermal treatment and the S(α,β) thermal library available for only a few low-z materials at certain temperatures. Full S(α,β) treatment for Zr in ZrH, H in ZrH, light water and graphite at 300^{o}K were used in order to accurately simulates the operational reactor at room temperature. This special scattering law data files, processed from ENDF-B/IV data, was obtained from the standard set that accompanied the MCNP4C code. [5,6] The slow neutron scattering cross section data used to account for the molecular binding effects of the light water, H/Zr and Zr/H were also evaluated at 300K. These thermal scattering data are essential to accurately model the neutron interactions at energies below ~ 4eV.

RESULTS AND DISCUSSIONS

The neutronic analysis of the 3 MW TRIGA MARK II benchmark experiments at AERE, Savar was performed by the three-dimensional continuous energy Monte Carlo code MCNP4C. All the neutronic parameters including effective multiplication factor, in-core and ex-core neutron flux and benchmarking of reactivity experiments were performed for the fresh core with the MCNP4C modeling of the TRIGA reactor.

A. Criticality Calculation

Nuclear criticality, the ability to sustain a chain reaction by fission neutrons, is characterized by k_{eff}, the eigenvalue to the neutron transport equation. The calculations of the effective multiplication factor (k_{eff}) in the eigenvalue problem for the fresh-core were performed with the MCNP4C code. This was done because the core multiplication factor is an integral quantity and criticality calculation is easy to perform. Also, any gross errors in the modeling should have been immediately apparent. The combined average of the absorption/collision/track-length estimator is quoted as the k_{eff} value in MCNP. The calculations were performed for the core both with and without control rods. The control rods in the former were in the critical positions and in the latter were completely withdrawn like an excess reactivity measurement. The initial critical core configuration (k_{eff} equal to 1.0) was

obtained with critical rod height of all control rod bank positions to 37.1309% equivalent to a length of 14.146875 cm, i.e., all control rods were 23.953125 cm inserted to the active core. A satisfactory stabilization of the fission source was achieved by following 3000 neutrons per cycle in 1000, 2000, and 3000 cycles, respectively, and with a cumulative CPU time of ~ 595.95 min in a 733 MHz Pentium III PC. In order to decrease statistical error estimates, initial numbers of 25 cycles were skipped for each case. The estimated statistical error (1σ) was reduced below 0.03% upon 3000 cycles of iteration on a nominal source size of 3,000 particles per cycle. The comparison of the combined collision, absorption, and track length k_{eff} estimator between the experimental value and the MCNP calculation is shown in Table II.

The MCNP calculated value underestimated 0.2510%Δk/k for control rods critical positions and overestimated 0.0966%Δk/k for all control rods withdrawn positions. Table II also includes a comparison of two effective multiplication factors: the MCNP predicated value and the CITATION predicated value [1] for the initial critical experiments in condition of all control rods withdrawn. This result is very encouraging and seems to indicate that the MCNP model of the TRIGA core is correct.

B. Benchmark Analysis of Neutron Flux

Core multiplication factors, however, are still global quantities and not local quantities. To further assess the accuracy of the MCNP model of the TRIGA reactor, it is essential to make some comparisons to local quantities such as neutron flux measurements. The knowledge of neutron environment in the reactor core is essential for effective utilization of the research reactor. A comparison was made between the experimentally measured neutron flux distribution at the water filled central thimble of the reactor with the MCNP calculations.

B.1. Experimental Measurements

The neutron flux distributions at different positions in the core of the 3 MW TRIGA MARK II research reactor was investigated in 1987 using the foil activation measurement technique. [7] The flux measurements were performed by irradiating gold foils for slow neutron reaction: $^{197}\mathrm{Au}(n,\gamma)^{198}\mathrm{Au}$. Since the neutron in a reactor cover a very wide range—thermal to well over 10 MeV, so the bare gold foil is activated both by the thermal and epithermal neutrons during irradiation inside the reactor core. This is eliminated by using cadmium filters since cadmium has a high absorption cross section for neutron below 0.41eV. Activity induced by thermal neutrons is simple to determine by irradiating two uniform detectors, one bare and the other covered with cadmium. Activation due to thermal flux could, therefore, be derived by simply subtracting the Cd-covered values from those of the bare foils. All the measurements were performed at the reactor power level of 50 watts and then the results were extrapolated to 3 MW.

B.2. MCNP Calculations

The neutron flux, normalized to 3 MW (thermal), was calculated in MCNP using two methods. The first method tallied the integrated neutron flux above the appropriate energy in the irradiated volume. The second method tallied the appropriate reaction rates using

dosimetry cross-section tables available in MCNP. The reaction rates were then divided by the appropriate cross section to yield the flux. If the cross section were correct, then the results from the different MCNP methods should be equivalent.

Equation 1, as follows, is a mathematical representation of the neutron flux calculated directly with MCNP within the energy range between E_1 to E_2; Eqs. 2 and 3 mathematically depict the method used in MCNP to simulate the experiment in which information was obtained:

$$\Phi_g = \int_{E_1}^{E_2} \Phi(E)dE \qquad (1)$$

$$R^j = \int_{E_1}^{E_2} \Phi(E)\sigma^j(E)dE \qquad (2)$$

and

$$\Phi^j = R^j \Big/ \sigma_{exp}{}^j \qquad (3)$$

where Φ_g = neutron flux above E_g which is the threshold energy of interest

R^j = reaction rate for reaction j

$\sigma^j(E)$ = cross section for reaction j

Φ^j = neutron flux above the threshold energy calculated using reaction j

$\sigma_{exp}{}^j$ = threshold reaction cross section for reaction j calculated and used by the experimenter

If the reaction rate R^j calculated with MCNP and the threshold cross section are correct, then both fluxes, Φ_g and Φ^j should be equivalent. Any differences between these values could be due to an error in the calculation of the threshold cross section performed by the experimenter or an error in the Monte Carlo simulation. An agreement between these values would indicate that the Monte Carlo simulation of the TRIGA reactor is correct.

Note that the flux measurements were performed by irradiating gold foils for slow neutron reaction: $^{197}\mathrm{Au}(n,\gamma)^{198}\mathrm{Au}$. The distribution profile of the $^{197}\mathrm{Au}(n,\gamma)^{198}\mathrm{Au}$ reaction rate is quite different from those of threshold reactions. This is simply because the most sensitive energy for (n,γ) reaction, in general, is at the resonance or the thermal energy. On the other hand, the threshold type reaction show almost same trend of the decreasing distribution profile. The ratios of reaction with higher threshold energy decrease more rapidly than those with lower threshold energies. That is why Φ_g, neutron flux within a certain range of energy, instead of Φ^j was compared with the experiment. Because Φ_g is calculated directly in MCNP using a whole range of pointwise cross section data. Here the thermal energy range was chosen from 0 to 0.41 eV and epithermal energy range was 0.41 eV to 9.118 keV.

Instead, the comparison between the experiment and the MCNP calculation for the experimental simulation of flux measurement was made in a little bit different manner. The saturated activities for the bare and Cd-covered gold foil reactions were taken from the experimental values and were then compared with those calculated by MCNP. In this way no threshold reaction cross section is required. An agreement between these values would indicate that both fluxes, Φ_g and Φ^j are equivalent and consequently the Monte Carlo simulation of the TRIGA reactor is correct.

The saturated activity for foil reactions were calculated by MCNP4C for bare and Cd-covered gold foils in the central thimble and were compared with the experimental values. The FMn card was used to calculate any flux tally of the form $\int \varphi(E)dE$ into $C\int R_m(E)\varphi(E)dE$, where $R(E)$ is an operator of additive and/or multiplicative response functions from the MCNP cross section libraries or specially designated quantities. The constant *C* is any arbitrary scalar quantity that can be used for normalization and *m* is the material number. From the ENDF standard reaction list, the special reaction used for this purpose for ^{197}Au is: 102 that stands for (n,γ) reaction. The F4 neutron tally from MCNP using a multiplier set, the track length estimate of the average fluence (reactions/source neutron/nucleus), were calculated at different positions of the water filled central thimble vertically from 28 cm below the core centerline to 28 cm above for gold atom with an atomic fraction of 100%. This average fluence multiplied by the source strength (neutron/sec) due to 3 MW reactor power yields reactions/nucleus/sec (or DPS/nucleus) that is the saturated activity at a particular position in the CT. Similar method was also used to calculate the saturated activity for the Cd-covered gold foil. This time the F4 neutron tally, the track length estimate of the average fluence (reactions/source neutron/nucleus), were calculated using attenuator set option of MCNP in addition to the multiplier set option.

B.3. Comparison with MCNP Calculation (Direct Method)

The results of the experimental thermal and epithermal flux measurements and the MCNP calculated values (direct method) in the different positions of central thimble is shown graphically in Fig. 7. All the flux values are extrapolated to 3 MW. The maximum thermal flux i.e. the peak is in the core center and the axial profile is generally symmetrical about this peak. The peak thermal neutron flux calculated by MCNP is under predicting the experimental one by about 13.6%. As can be seen from Fig. 1, the experimental thermal flux at core center is 7 times more than the top and 5.3 times more than the bottom of the core, whereas those for the MCNP calculation are 9.6 and 9.7, respectively. It is also observed that the thermal flux at the core center are 4.40 and 4.03 times more than the epithermal flux for the experiment and MCNP calculation, respectively. It is also found that the MCNP predicted values for the epithermal neutron flux agree more closely with experimental values.

B.4. Comparison with Foil Activation Results

In order to do the calculation of saturated activity by MCNP, the following method was adopted. First the *Q-value* in the fission process and the fission neutrons released per fission (ν) were determined. The fission *Q-value* in a fission reaction with ^{235}U fissile

material is found to be 181.22 MeV/fission. The average (total) numbers of neutrons ν liberated for each neutrons absorbed in a fission reaction with ^{235}U fissile nucleus is observed as 2.50. Now for the normalization of a criticality calculation by the steady-state power level of a reactor, the following conversion is used: [2]

$$\left(\frac{1\,joule/sec}{watt}\right)\left(\frac{1MeV}{1.602x10^{-13}\,joules}\right)\left(\frac{1}{Q-value}\right)$$
$$= 3.44454 \times 10^{10}\ fission/watt\text{-}sec$$

where the calculated *Q-value* in the fission process is 181.22 MeV/fission.

Therefore, to produce 3 MW of power, one needs

$$= 1.0334 \times 10^{17}\ fission/sec.$$

So the source strength of this power level becomes

$$= 1.0334 \times 10^{17} \times \nu$$
$$= 2.5835 \times 10^{17}\ neutrons/sec,$$

where ν is 2.50 *source neutrons/fission*.

The average fluence for bare and Cd-covered gold foils, calculated by FMn option of MCNP, multiplied by this source strength (neutron/sec) due to 3 MW reactor power yields reactions/nucleus/sec (or DPS/nucleus) termed as the saturated activity. The results of the saturated activity for bare and Cd-covered gold foil reactions in the different positions of CT is shown graphically in Fig. 8. All the values are extrapolated to 3 MW. The maximum saturated activity i.e. the peak is in the core center and the axial profile is generally symmetrical about this peak. The peak saturated activity for thermal neutron calculated by MCNP is under predicting the experimental one by about 6.9%. It justifies the use of MCNP simulation of experiment through foil activation method for flux comparison. The MCNP and experimental curves (Fig. 8) are much closer than those for direct MCNP calculation of neutron flux (Fig. 7).

C. Benchmark Analysis of Reactivity Experiments

The benchmark analysis of the reactivity experiments in the 3 MW TRIGA MARK II research reactor core was also performed by the three-dimensional continuous-energy Monte Carlo code MCNP4C. The reactivity worth and integral reactivity curves of the control rods as well as the excess reactivity and shutdown margin were used in the validation process of the physical model. The experimental data that were used to benchmark the MCNP4C TRIGA LEU core model was the initial control rod calibration and shutdown margin measurements performed on Oct. 9, 1986 during reactor start-up at full power operation. [8]

C.1. Control Rod Worth

Adequate treatment of control rods is very important in the simulation of any specific core configuration. Even small deviations of the model could eventually lead to large

systematic errors of the calculated k_{eff}. The positive period method (rod-exchange method) of measuring the control rod worths was used in the experiment. [8] Following this method, one control rod is measured in the presence of another rod, used for compensating the excess reactivity. Because faithful simulation of the experiment would be very time consuming, we decided to simulate measurement in a way similar to the rod-insertion method. Although both methods differ significantly from the experimental point of view, they should both give the same results, if used correctly. [9] The major experimental inconvenience of the rod-insertion method—the fall of the neutron flux for several orders of magnitude during the measurement—can be circumvented in MCNP4C by using selected estimators of k_{eff}. Thus, no corrections needed to be applied on the calculated results.

The core arrangement used for the reactivity calculation is shown in Fig. 2. All the six control rods are positioned at different locations of D-ring. We started the simulation with all control rods completely withdrawn, calculating the $k_{eff,0}$ of the core (1.07850 ± 0.00028). Then one of the control rods was inserted to a certain position, calculating a new k_{eff}. To calculate values of reactivity associated with each of these k_{eff} values, the following definition of reactivity was used:

$$\rho = \frac{k_{eff} - 1}{k_{eff}\beta} \text{ (in dollar, \$)} \qquad (4)$$

where β is the effective delayed neutron fraction for U-ZrH type fuel with a value of 0.007 that is used to convert the unit from $\Delta k/k$ to dollar or cent. [10]

Therefore, the control rod worth for that position was determined by comparing k_{eff} and $k_{eff,0}$ as denoted in Eq. 5 represented by reactivity ρ. The error propagated is deduced from Eq. 6. [11]

$$\begin{aligned}\rho &= \rho_0 - \rho_1 \\ &= \left[\left(1 - 1/k_{ff,0}\right) - \left(1 - 1/k_{eff}\right)\right] x \frac{1}{\beta} \\ &= \left(\frac{1}{k_{eff}} - \frac{1}{k_{eff,0}}\right) x \frac{1}{\beta} \qquad (5)\end{aligned}$$

$$\Delta\rho = \left(\left(\frac{\Delta k_{eff}}{k_{eff}}\right)^2 + \left(\frac{\Delta k_{eff,0}}{k_{eff,0}}\right)^2\right)^{1/2} x \frac{1}{\beta} \qquad (6)$$

where Δk_{eff} and $\Delta k_{eff,0}$ show the fractional statistical error estimates for k_{eff} and $k_{eff,0}$ respectively.

A control rod, when inserted into the core, causes severe flux distribution deformation. As a consequence, more "continue" MCNP4C runs were needed to achieve stable spatial source distribution for a certain configuration. Total control rod worths together with experimental data are summarized in Table III. Integral reactivity curves for all six control rods were reproduced in this way. Figure 9 shows the integral curve comparison between the MCNP predictions and the experiments for all six-control rods. The worths were

calculated for all the rods in 3.81-cm step increments. The MCNP and the experimental curves are very smooth. However, the MCNP predicted values seem to be consistently lower than the experimentally determined values except for transient rod. One possible explanation for this behavior is the lack of modeling xenon - which would result in a different power distribution during the experiment than was predicted with MCNP. The magnitude of this effect is not known but is believed to be small. Even with this consideration, the agreement between the MCNP predicted values and the experimentally determined values are fairly good. Referring to the control rod worths for the TRIGA core summarized in Table III, there are generally good agreement with actual measured control rod worths, with the difference between calculated versus experimental rod worths ranged from –3.34% to +8.42%. The results for each rod are compared with the experimental results. The calculated control rod reactivity worths are consistent with the experiments within the estimated error of 10%. [12]

C.2. Critical Rod Height

Using the TRIGA model, the next test of the MCNP4C simulation was to calculate the critical rod height (CRH) and compare this predicted CRH with the actual CRH for the fresh core with no fuel burnup. This calculation involved several criticality case run at different control rod bank positions to determine the CRH for which k_{eff} was closest to 1.0. For the experimental determination of CRH, the integral control rod worth curves were used. The initial critical core configuration (k_{eff} equal to 1.0) was obtained with CRH of all control rod bank positions to 37.1309% equivalent to a length of 14.146875 cm, i.e., all control rods were 23.953125 cm inserted to the active core. In the MCNP4C model, all the control rod bank positions were adjusted until a value of k_{eff} approximately equal to 1.0 was obtained. After several runs at different control rod bank positions, a critical core configuration was finally obtained, with a value of k_{eff} equal to 1.00015 ± 0.00028. This corresponded to all control rod bank positions of 37.2315% equivalent to 14.1852 cm i.e., all control rods were 23.9148 cm inserted to the active core. This result shows excellent agreement for the CRH with the experiment within an estimated error of ~ 0.27%.

C.3. Excess Reactivity Analysis

The excess reactivity was experimentally determined from control rod worths critical positions and their calibration curves, measured by the period method (rod-exchange method). Then the individual values of excess reactivity associated with each control rod above its CRH position were combined to obtain the total core excess reactivity at 300K. To be consistent with the experimental determination of excess reactivity, a similar method was used using the TRIGA model. For each control rod, k_{eff} was calculated with the given control rod at its CRH position and all of the other control rods fully withdrawn. The excess reactivity of each control rod was then calculated using the following definition

$$\rho_{\text{excess}} = \rho_{\text{out}} - \rho_{\text{crit}} \qquad (7)$$

where ρ_{excess} = excess reactivity associated with a given control rod

ρ_{out} = reactivity associated with all control rods fully withdrawn position

ρ_{crit} = reactivity associated with given control rod at its CRH and all other rods withdrawn position

Results of the MCNP calculated excess reactivity worths for all six control rods along with those of experimental values are presented in Table IV. The MCNP estimation from control rod critical positions was obtained combining calculated integral reactivity curves and critical positions. Estimation was done for each rod separately, and the total value is also given in Table IV. As can be observed, the agreement between the experimental results and the results of the Monte Carlo simulation again fall well within the range of +2.44% to -10.73%. The inconsistency between the MCNP calculated value and the experiment could be settled by the assumption of the experimental error about 10%.

C.4. Shutdown Margin

This specification applies to the reactivity margin by which the reactor core will be considered shut down when the reactor is shutdown in a cold, xenon free condition. In the case of shutdown margin, the following definition was used in both the experimental case and in the case of the MCNP4C calculation for the shutdown margin of the core

$$\rho_{sdm} = \rho_{rods} - \rho_{excess} - \rho_{shim\ III} \qquad (8)$$

where

ρ_{sdm} = shutdown margin of the core
ρ_{rods} = total control rod worths
ρ_{excess} = excess reactivity of the core
$\rho_{shim\ III}$ = most reactive rod worth (shim III)

Equation 8 is based on the definition of shutdown margin in the Technical Specifications of the reactor. [10] Based on this definition of shutdown margin, a value of 3.36 $ was calculated based on experimental measurements, while a value of 3.2521 $ was calculated based on the TRIGA core model. The difference between the calculated and experimental values of the shutdown margin (3.21%) is mainly due to the difference in the values of excess reactivity in each case, which has already been discussed. This value of the shut down margin assures that the reactor can be shutdown from any operating condition with the assumption that the highest worth control rod remains fully withdrawn (stuck rod condition).

CONCLUSIONS

MCNP has been used to develop a versatile and accurate reactor physics model of the TRIGA MARK II research reactor. To minimize errors due to an inexact geometry model, the reactor was very thoroughly modeled. All fresh fuel and control elements as well as the vicinity of the core were precisely simulated. MCNP4C input was prepared in such a way that any desired core configuration could be simulated easily. The consistency and accuracy of the MCNP4C model of the TRIGA reactor core was established by comparing calculations to the experimental results of the benchmark experiments. The MCNP calculated values of the multiplication factor are consistent with the experimental data, although the MCNP calculated value underestimated 0.2510%Δk/k for control rods at critical positions and overestimated 0.0966%Δk/k for all control rods withdrawn positions. The reactivity curves calculated by MCNP were also consistent with the experiments within the estimated error of 10%. The excess reactivity of the core was determined by using estimation from control rod

critical positions. All in all, it can be concluded that the model of the TRIGA MARK II research reactor is precise enough to reproduce for analyzing benchmark experiments. The MCNP calculation establishes that the Monte Carlo simulation might also be used as reference with confidence for further core configuration studies.

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Table I: Principal Design Parameters and Material Composition Data of the TRIGA MARK II Research Reactor used for Neutronic Analysis

Geometrical Data	Fuel element	Fuel in Fuel-follower
Radius of Zirconium rod (cm)	0.3175	0.3175
Radius of Fuel (cm)	1.82245	1.69164
Gap thickness (cm)	0.00381	0.00381
Cladding thickness (cm)	0.0508	0.0508
Material Composition	Fuel element	Fuel in Fuel-follower
Mass of U-Er-ZrH (g)	2466.096842	2089.809999
Mass of uranium (g)	493.2193684	417.962
Mass of ^{235}U (g)	97.65684211	82.756
Uranium (wt%)	20	20
Enrichment (%)	19.7998798	19.7995
Erbium (wt%)	0.47	0.47
Mass of erbium (g)	11.59065513	9.822106962
Mass of ^{166}Er (g)	3.871278783	3.280583698
Mass of ^{167}Er (g)	2.66190984	2.255745057
Hydrogen-zirconium atom ratio	1.6	1.6
Composition	Control rod absorber	
Absorber	B_4C	
Natural boron (wt%)	80	

Atomic Number Density (x 10^{24} atom/cm^3)			
Material (density: g/cm^3)	Nuclide	Fuel element	Fuel in Fuel-follower
Fuel Meat (U-Er-ZrH)	^{235}U	$6.491295097 \times 10^{-4}$	$6.416713999 \times 10^{-4}$
	^{238}U	$2.596110832 \times 10^{-3}$	$2.566282051 \times 10^{-3}$
	^{1}H	$5.281039106 \times 10^{-2}$	$5.220361432 \times 10^{-2}$
	^{nat}Zr	$3.300649442 \times 10^{-2}$	$3.262725913 \times 10^{-2}$
	^{166}Er	$3.645075053 \times 10^{-5}$	$3.603194166 \times 10^{-5}$
	^{167}Er	$2.491330402 \times 10^{-5}$	$2.462705703 \times 10^{-5}$
Zr-rod (6.52)	^{nat}Zr	4.3044407×10^{-2}	
Mo-ring (10.2)	^{nat}Mo	6.4024×10^{-2}	
Graphite (1.65)	^{12}C	8.2730×10^{-2}	
Stainless Steel Clad (SS-304: 7.889)	^{nat}Cr	1.7381×10^{-2}	
	^{55}Mn	1.7124×10^{-3}	
	^{nat}Fe	5.9195×10^{-2}	
	^{nat}Ni	7.5996×10^{-3}	
Control rod (2.465)	^{10}B	2.1386×10^{-2}	
	^{11}B	8.6082×10^{-2}	
	^{12}C	2.6867×10^{-2}	
Reflector (1.65)	^{12}C	8.2730×10^{-2}	
Lead Shield (11.3405)	^{207}Pb	3.2960×10^{-2}	
Grid Plate/Guide Tube/ Center Thimble/ Aluminum Tank/Lazy Susan/Reflector Housing (Anodized Aluminum: 2.7)	^{nat}Si	2.3150×10^{-4}	
	^{nat}Fe	2.0337×10^{-4}	
	^{55}Mn	4.4380×10^{-4}	
	^{nat}Mg	5.3490×10^{-4}	
	^{nat}Cr	4.6890×10^{-5}	
	^{27}Al	5.8830×10^{-2}	
Coolant (0.99644) (~27^{o}C)	^{1}H	6.6678665×10^{-2}	
	^{16}O	3.3339332×10^{-2}	

Table II: Comparison of the MCNP Criticality Calculations to the Experiment at Different Control Rod Positions

Method	Control Rods Position	Core Multiplication Factor, k_{eff}	
		Experiment	Calculation
MCNP4C	Critical	1.00	0.99749 ± 0.00029
	Withdrawn	1.077459	1.07850 ± 0.00028
CITATION	Withdrawn	1.077459	1.0755

Table III: Comparison between the MCNP Calculated Control Rod Worths of the TRIGA Core to the Experiment

Control Rod Worth ($)	Calculation (C) (k_{eff})	Experiment (E)	C/E
SAFETY	2.6163 ± 0.0530 (1.05761 ± 0.00028)	2.73 ± 0.10	0.9584 ± 0.0419
SHIM I	2.8031 ± 0.0521 (1.05615 ± 0.00027)	3.06 ± 0.10	0.9160 ± 0.0376
SHIM II	2.7455 ± 0.0536 (1.05660 ± 0.00029)	2.82 ± 0.10	0.9736 ± 0.0405
SHIM III	2.8684 ± 0.0526 (1.05564 ± 0.00028)	3.12 ± 0.10	0.9194 ± 0.0370
REGULATING	2.6738 ± 0.0536 (1.05716 ± 0.00029)	2.78 ± 0.10	0.9618 ± 0.0412
TRANSIENT	2.3145 ± 0.0545 (1.05998 ± 0.00030)	2.24 ± 0.10	1.0333 ± 0.0505
TOTAL WORTH	**16.0216**	**16.75**	**0.9565**

Table IV: Comparison between the MCNP Calculated Excess Reactivity of the TRIGA Core to the Experiment

Control Rod Worth ($)	Calculation (C) (k_{eff})	Experiment (E)	C/E
SAFETY	1.6488 ± 0.0537 (1.06524 ± 0.00029)	1.79 ± 0.10	0.9211 ± 0.0647
SHIM I	1.7887 ± 0.0433 (1.06413 ± 0.00028)	1.89 ± 0.10	0.9464 ± 0.0582
SHIM II	1.7320 ± 0.0547 (1.06458 ± 0.00030)	1.81 ± 0.10	0.9569 ± 0.0636
SHIM III	1.8443 ± 0.0538 (1.06369 ± 0.00029)	1.97 ± 0.10	0.9362 ± 0.0586
REGULATING	1.6589 ± 0.0518 (1.06516 ± 0.00027)	1.70 ± 0.10	0.9758 ± 0.0666
TRANSIENT	1.2284 ± 0.0556 (1.06859 ± 0.00031)	1.11 ± 0.10	1.1067 ± 0.1008
TOTAL WORTH	**9.9011**	**10.27**	**0.9641**

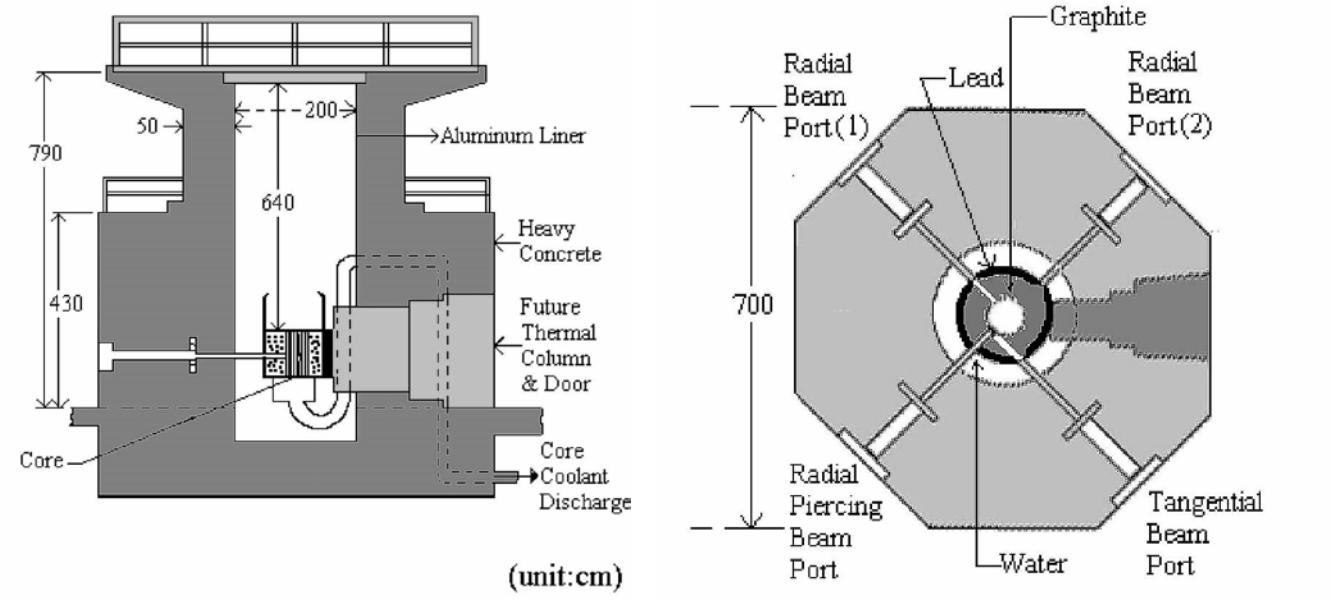

Fig. 1. Cutaway view of TRIGA reactor.

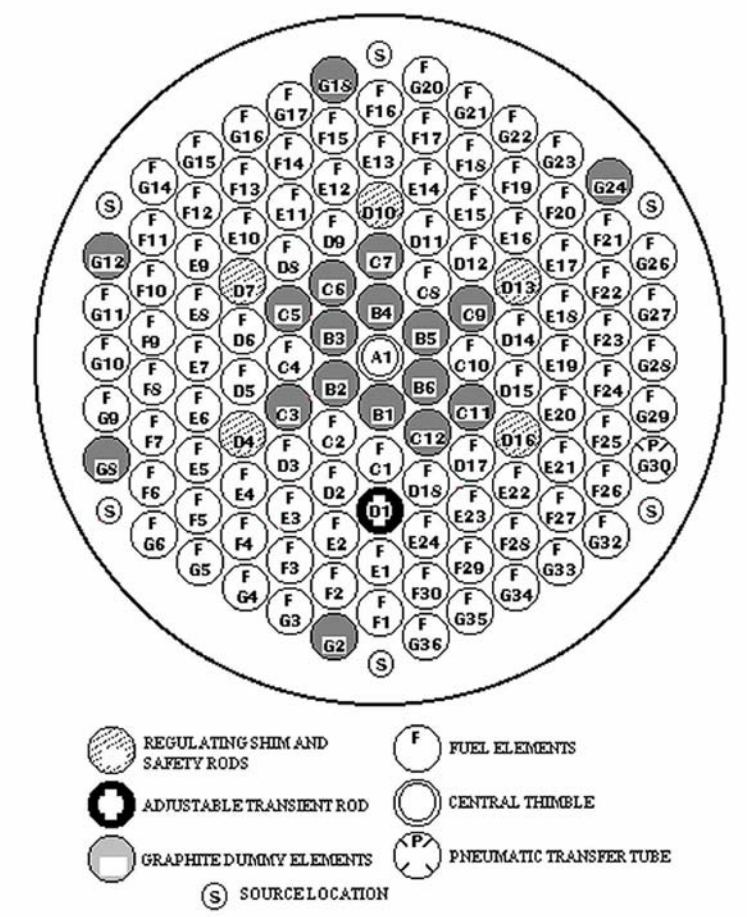

Fig. 2. Final core arrangement of the TRIGA reactor.

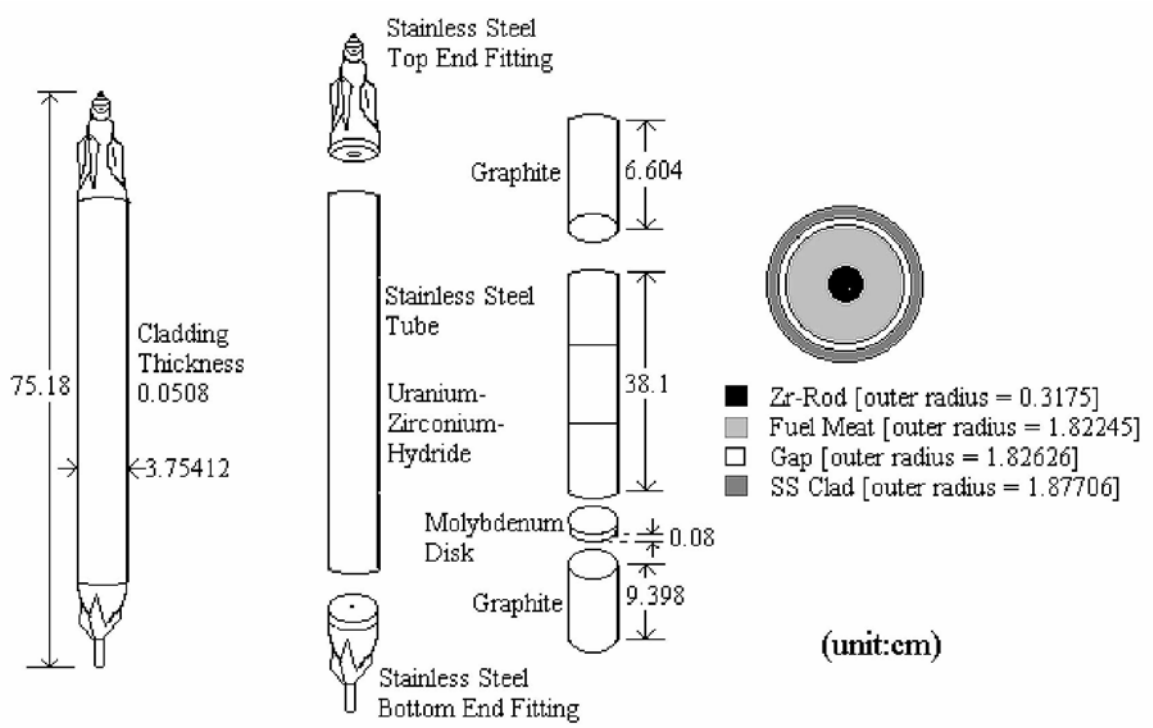

Fig.3. TRIGA stainless-steel-clad fuel element with triflute and fittings.

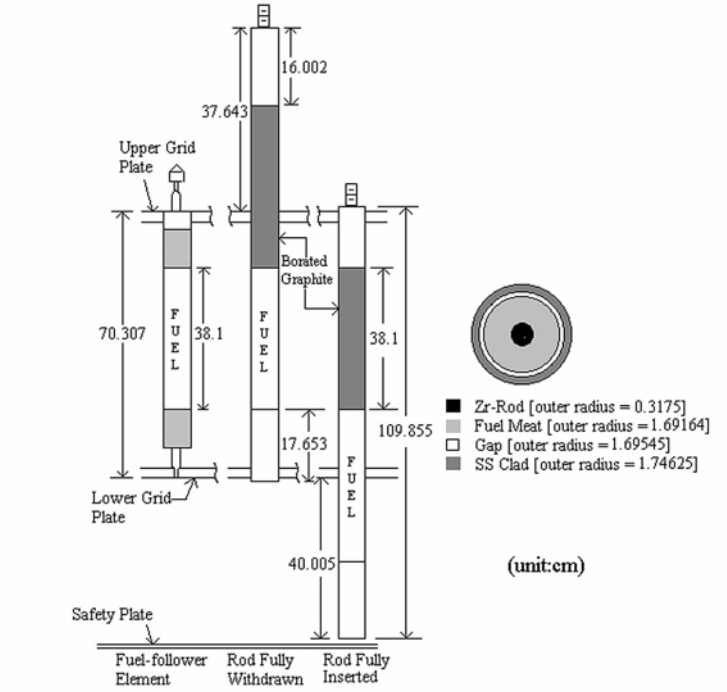

Fig.4. Fuel-follower control rod - withdrawn and inserted position.

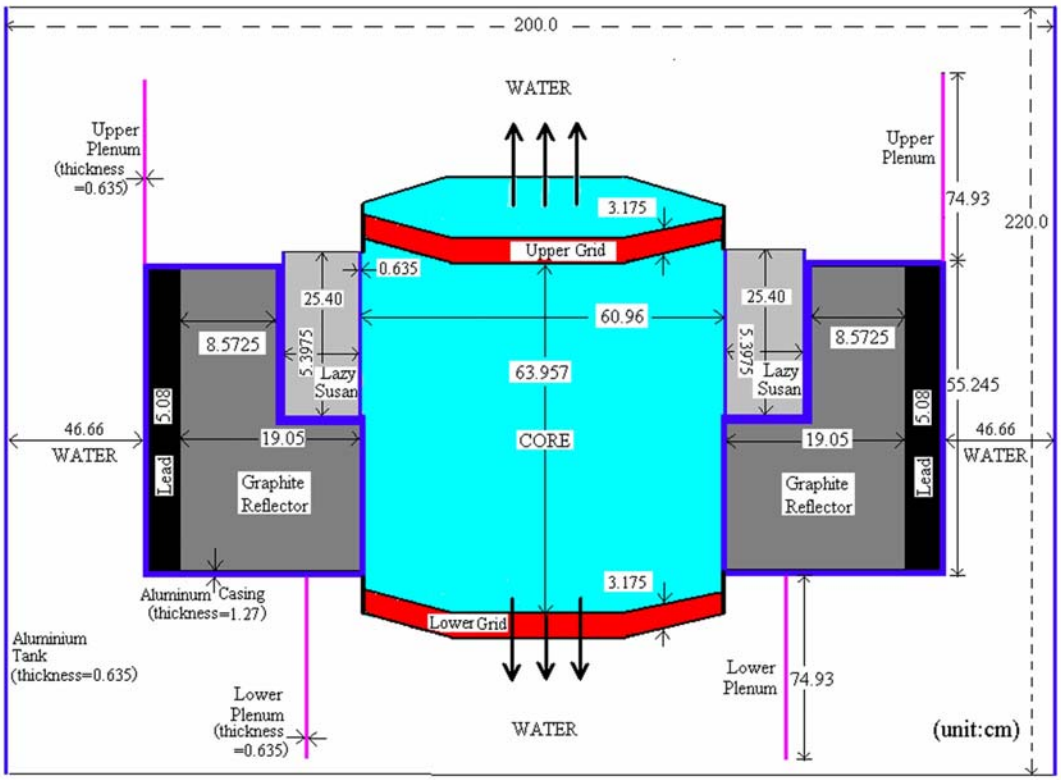

Fig.5. MCNP modeling of TRIGA MARK II research reactor.

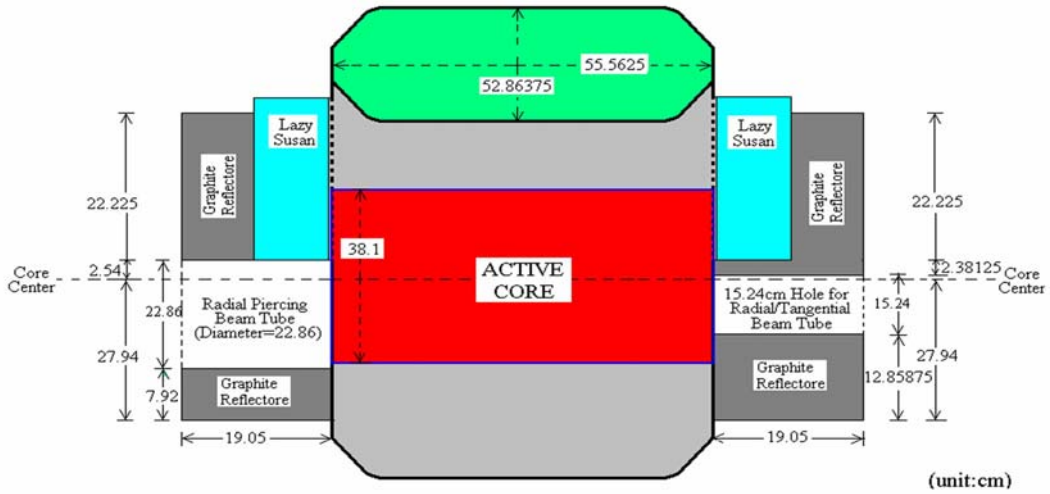

Fig.6. Cutaway view of TRIGA active core including rotary specimen rack Lazy Susan and beam ports.