Session 13-1

MONTE CARLO ANALYSIS OF MUSASHI TRIGA MARK II REACTOR CORE

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ABSTRACT

The analysis of the TRIGA-II core at the Musashi Institute of Technology Research Reactor (Musashi reactor, 100 kW) was performed by the three-dimensional continuous-energy Monte Carlo code (MCNP4A). Effective multiplication factors (k_{eff}) for the several fuel-loading patterns including the initial core criticality experiment, the fuel element and control rod reactivity worth as well as the neutron flux measurements were used in the validation process of the physical model and neutron cross section data from the ENDF/B-V evaluation. The calculated k_{eff} overestimated the experimental data by about 1.0 $\%\Delta k/k$ for both the initial core and the several fuel-loading arrangements. The calculated reactivity worths of control rod and fuel element agree well the measured ones within the uncertainties. The comparison of neutron flux distribution was consistent with the experimental ones which were measured by activation methods at the sample irradiation tubes. All in all, the agreement between the MCNP predictions and the experimentally determined values is good, which indicates that the Monte Carlo model is enough to simulate the Musashi TRIGA-II reactor core.

1. INTRODUCTION

The Musashi Institute of Technology Research Reactor (Musashi reactor) is a 20 % enriched TRIGA-type-fueled, 100 kW multipurpose research reactor that was designed to provide both in-core and out-of-core irradiation facilities. The Musashi reactor had many different uses including BNCT projects[1], a wide range of neutron activation studies, an application of neutron radiography and other neutron beam experiments[2] till the end of 1989 when unfortunately water leaking trouble was occurred in the reactor tank. For installing a new tank and restarting the reactor an adequate design of a core and irradiation facilities is inevitable. It is also important to develop an accurate three-dimensional reactor physics model of the TRIGA type reactor core. The MCNP Monte Carlo code (MCNP4A)[3] was chosen because of its general geometry modeling capability, correct representation of transport effects and continuous-energy cross sections.

The main requirement for the reliable use of a particle transport computer code is its validation on a benchmark experiment. There are two main objectives of such verification. The first is to check the consistency of physical models and data used in a transport code, and the second is to determine systematic errors made by approximate simulation of the experiment. To put full confidence in the model, previously obtained experimental data were used to compare MCNP calculated values. These data were obtained in 1985, after replacing the old fuel elements of aluminum cladding with new ones of stainless-steel cladding to keep a more safe operation[4]. The experimental data used are as follows:
1) criticality measurements including the initial core and the core for several fuel-loading pattern,
2) reactivity measurements including fuel element reactivity worth distributions and control rod worths, and 3) neutron flux measurements in the sample irradiation tubes.

2. MCNP CALCULATION

The Musashi reactor core was modeled using the three-dimensional detail to reduce possible systematic errors due to inexact geometry simulation. Therefore, the fuel elements were explicitly modeled to eliminate any homogenization effects. A fuel element consists of meat, graphite reflectors and stainless steel end fixtures. The meat is a solid, homogenized mixture of U and $ZrH_{1.6}$ containing 8.45% U enriched to 20% ^{235}U. The fuel element is 3.75 cm in diameter with a total length of 76 cm, and is clad in 0.05 cm thick stainless steel canning. **Figure 1** is a cross-sectional view of the Musashi reactor core as modeled. The reactor core consists of a lattice of fuel elements, graphite dummy elements, control rods and irradiation tubes. These components are located at ninety holes reserved for them on the grid plate, distributed in five circular rings (B to F ring) from inner to outer positions. The reflector is a ring-shaped graphite block rising slightly higher than 55 cm, with an inside diameter of 45 cm and radially surrounding the core to a thickness of 30 cm. Provision for the isotope production facility (irradiation pit) is made in the form of ring-shaped well (25 cm in deep and 8 cm in thickness) in the graphite that is open toward the tank top.

The control rods (safety, shim and regulating rods) were made of boron carbide powder (B_4C). The safety and shim rods are 3.0 cm in diameter and the regulating rod 2.0 cm in diameter, respectively, which are clad in 0.2 cm thick aluminum. The active lengths of the control rods are 48 cm that extend well above the core in the fully inserted position. The control rods were also explicitly modeled along the active length with the exception of the drive mechanisms.

Three of the beam tubes (A, B and C) are oriented radially with respect to the center of the core, and the fourth tube (D) is tangential to the outer edge of the core. Two of the radial tubes (B and C) terminates at the outer edge of the reflector assembly; one (C) is aligned with a cylindrical void in the reflector graphite. The third radial tube (A) penetrates into the graphite reflector and terminates at the inner surface of the reflector assembly, just at the outer edge of the core as shown in Fig. 1 . Their horizontal centerlines are located 7 cm below the centerline of the core. The inner diameter of two beam tubes penetrating through the reflector graphite is 15 cm.

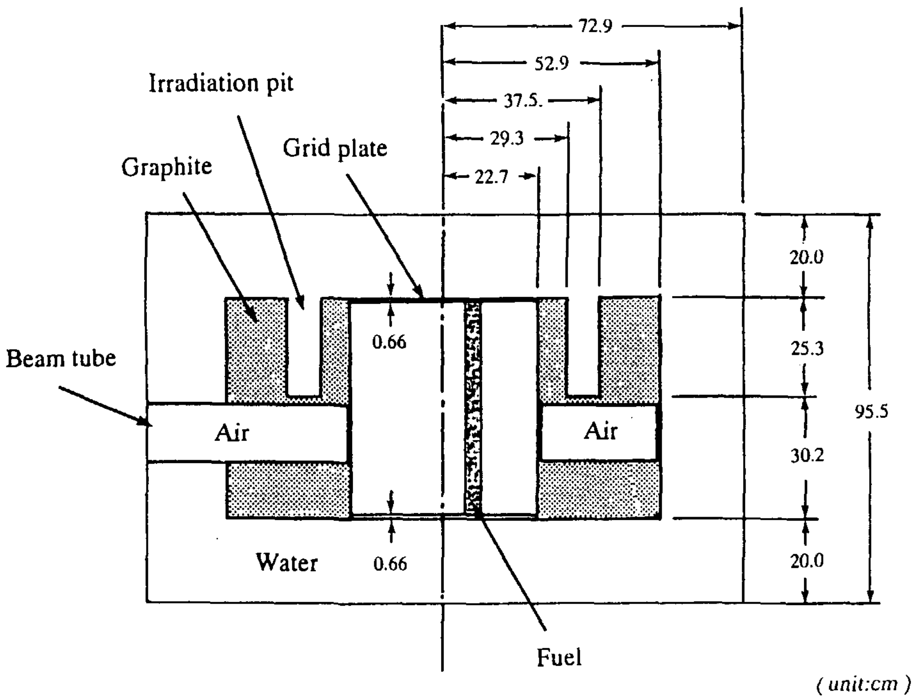

Fig. 1 Model of core and reflector arrangements used for the MCNP criticality calculations.

The MCNP model was extended to 20 cm beyond the end of the graphite reflector and approximately 20 cm above the core, which were more than sufficient to account for the neutron returning from the H_2O coolant. The primary coolant was non-pressurized and the cooling was natural convection. The coolant temperature was $28 \pm 1^{\circ}C$ throughout the criticality experiments which carried out at zero power (maximum 1W). Based on these operating conditions, all of the cross-sections used in the MCNP model were 300 K evaluations, usually from the ENDF/B-V cross-section data[5]. The slow neutron scattering law $S(\alpha, \beta)$ used to account for the molecular binding effects of the light water, graphite, and hydrogen and zirconium in ZrH were also evaluated at 300K. These thermal scattering data are essential to model accurately the neutron interactions at energies below ~ 1 eV.

The calculations of the effective multiplication factor (k_{eff}) in the eigenvalue problem were performed with the "KCODE" option in MCNP4A code. The initial source distribution for the k_{eff} calculations was given on the fuel meat points. There was 1 source point (1 cm part for radial direction from the meat center) in each of the fuel element in this model which placed on the "ksrc" card in the input file. 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 calculation of the reactivity worth such as control rod and fuel element was carried out by comparison with the condition between the insertion and the withdrawn. The estimated statistical error (1 σ) was reduced below 0.1% upon 300 cycles of iteration on a nominal source size of 3,000 particles per cycle. The calculation of neutron flux in the sample irradiation tube was performed by using "Tally" card (f:4) in the input file which was able to calculate average flux over the cell volume.

3. RSULTS AND DISCUSSION

3.1. Criticality

The first comparison to the experimental data was done for the fresh-core multiplication factor. An MCNP pictorial representation of the initial fuel loading is presented in **Fig. 2a**. The comparison of k_{eff} between the experimental value and the MCMP calculated one is shown in **Table 1**. The calculated value overestimated about 1 $\%\Delta k/k$ for both all control rod withdrawn and the control rod critical positions. Table 1 also includes a comparison of three effective multiplication factors: the MCNP predicted value, the KENO predicted value[6] and the CITATION predicted value[7] 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 Musashi reactor core is correct. The foremost reason for a discrepancy could be an error in the model (i.e., physical representation of the reactor).

After the initial critical, the excess reactivity adjustment was performed because an additional reactivity was need for the actual reactor operation. The fuel or graphite dummy element was simply added only to the F-ring in the initial critical core with 66 elements. A desirable value of excess

Table 1. Comparison of the MCNP criticality calculations to the experiment for the initial core.

Initial core (66 elements)	Multiplication factor (k_{eff}) Calculation	Experiment	Control rods positions
MCNP	1.0086 ± 0.0010	1.00	Critical positions
	1.0124 ± 0.0009	1.0018	Up (withdrawn)
KENO	1.0169 ± 0.0011	1.0018	Up (withdrawn)
CITATION	1.0609	1.0018	Up (withdrawn)

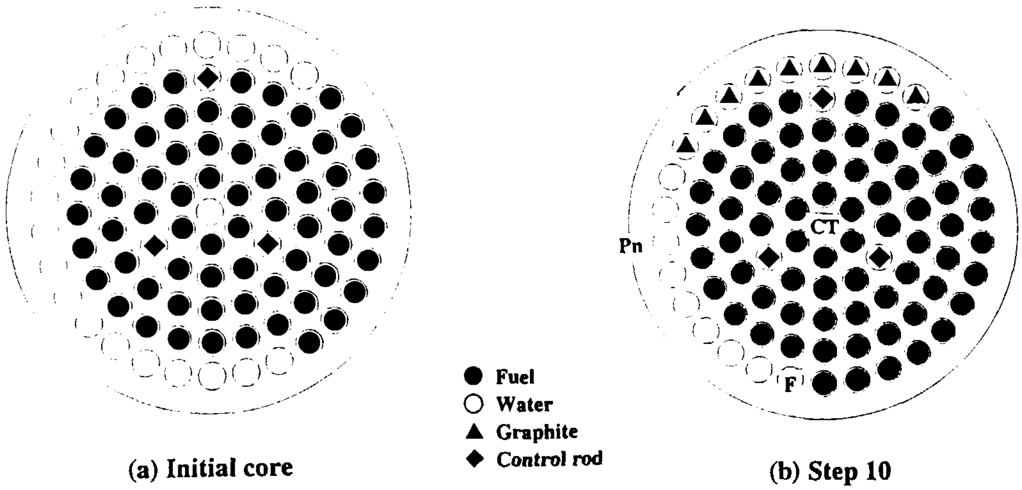

Fig. 2 Typical core configurations used for the comparison. (a) initial core loading (b) step 10

reactivity which was as close as possible to required excess reactivity (1.6 $\%\Delta k/k$) was achieved at step 10 as shown in **Fig. 2b**. The measurement of excess reactivity was executed by using the period method. The MCNP calculated values are compared with the experimental data in **Fig. 3.** The excess reactivity increased gradually every step. The values indicated in black symbol mark (●) are fitted by a least squares technique. The values indicated in white symbol mark (○) should become 1.0 since those are obtained at the critical condition. The MCNP calculated values overestimated the experimental ones by about 1.0 $\%\Delta k/k$. This comparison shows that a fuel-loading arrangement at F-ring could be modeled in MCNP without many approximations.

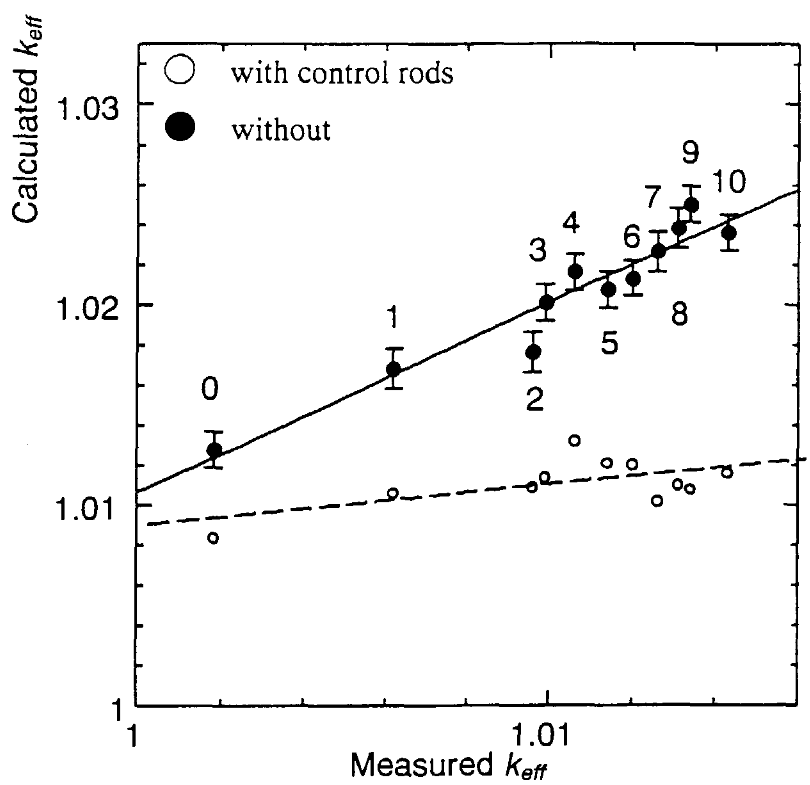

Fig. 3 Comparison of the MCNP calculated values to the experimental data from step 1 to step 10 under the conditions of all rod-withdrawn (symbol ●). The MCNP prediction values under the condition of critical rod positions are shown in symbol ○ at each step. Step number is denoted in numerical.

Table 2. Comparison of the MCNP calculations to the experimental data for the fuel element reactivity worth distributions.

Fuel insert position	B4	C7	D11	E15	F19
Experiment ($\%\Delta k/k$)	1.20	0.91	0.72	0.48	0.28
Calculation ($\%\Delta k/k$)	0.95±0.12	0.76±0.11	0.62±0.12	0.45±0.11	0.30±0.11

3.2. Reactivity worth

The fuel element reactivity worth (substitution reactivity of fuel and water) distributions were calculated and compared with the experimental data. The fuel was simply replaced to the water at B4, C7, D11, E15 and F19 positions in the final adjusting configuration. **Table 2** shows the comparison between the MCNP calculated values and the experimental data. It is generally recognized that the closer a insertion reactivity of fuel gets to the center of a core, the larger it becomes. However, the discrepancy between the calculation value and the experimental data gradually increased as the insert position was closer to the center of core. The experimental error should be considered in the referred benchmark data which is not estimated[4)]. The inconsistency between the MCNP calculated value and the experimental ones for the B and C rings could be settled by the assumption of the experimental error about 10 %.

The control rod worth was also calculated and compared with the experimental data. The calculation was started with all control rods completely withdrawn, calculating the $k_{eff,0}$ of the core. Then one of the control rods was inserted to full position, calculating a new k_{eff}. The control rod worth was determined by comparing $k_{eff,0}$ and k_{eff} (A in Table 3). Another calculation was also performed with the same critical rod position as the experiment (B in Table 3). The experiment was performed by the rod drop method and rod-exchange method was used. **Table 3** also shows the comparison between the MCNP calculated values and the experimental data. The calculated values were fairly good except the safety rod worth, though the safety and shim rods had same geometry and material, and also symmetrically positioned at B-ring as shown in Fig. 2b.

3.3 Neutron flux in the sample irradiation tubes

The MCNP code was used to estimate the thermal and fast neutron fluxes for comparison the experimental data. Those were obtained at three irradiation tubes; central thimble (CT), F-ring (F) and pneumatic tube (Pn) as depicted in Fig. 2b, provided for isotope production. Water is fulfilled at the CT and F and air at the Pn. Table 4 shows the comparison between the calculation and the experiment.

Table 3. Comparison of the MCNP calculations to the experimental data for the control rod worth.

Control rod	Calculation (cent)		Experiment (cent)
	A	B	
Safety	285±26	316±24	385
Shim	281±24	321±26	303
Regulating	71±26	100±24	84

Table 4. Comparison of the MCNP calculations to the experimental data for the neutron flux.

	Calculation (unit: $ncm^{-2}s^{-1}$)		Experiment (unit: $ncm^{-2}s^{-1}$)	
Position	Thermal (<0.4 eV)	Fast (>1 MeV)	Thermal (<0.4 eV)	Fast (> 1 MeV)
CT	3.1×10^{12}	7.8×10^{11}	$(3.2 \pm 0.1) \times 10^{12}$	$(6.8 \pm 0.4) \times 10^{11}$
Pn	1.3×10^{12}	4.2×10^{12}	$(1.4 \pm 0.1) \times 10^{12}$	$(2.5 \pm 0.4) \times 10^{12}$
F	1.7×10^{12}	3.5×10^{12}	$(1.8 \pm 0.1) \times 10^{12}$	$(3.9 \pm 0.4) \times 10^{12}$

The experiment was performed by activation method of gold and indium foils by use of $^{197}Au(n,\gamma)^{198}Au$ and $^{115}In(n,n')^{115m}In$ reactions for thermal and fast neutron flux measurements, respectively. The counting error only was considered in the experiment. The statistical error estimate of the calculated neutron flux was less 1%. The thermal neutron flux of MCNP predicted value is consistent with the experimental one. The fast neutron flux is slight larger than the experiment one which could be depend on the threshold energy of cross section.

4. CONCLUSION

The simulation of the TRIGA-II benchmark experiment at the Musashi Institute of Technology research reactor was performed by the three-dimensional continuous-energy Monte Carlo code (MCNP4A). The MCNP calculated values of the multiplication factor are consistent with the experimental data for the initial critical experiment and for the simple core of several fuel-loading arrangements, although the calculated values are about 1.0 $\%\Delta k/k$ overestimated the experimental values. As to the reactivity worth of control rod and fuel element, the MCNP calculated values agree well with the measured ones within uncertainties except for the inner-ring which underestimated the experimental data. The neutron flux comparison is also good. All in all, it can be concluded that our model of the TRIGA-II core is precise enough to reproduce criticality experiment, control rod and fuel element reactivity worths as well as neutron flux distribution.

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