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EXPERIMENTAL AND MCNP5 BASED EVALUATION OF NEUTRON AND GAMMA FLUX IN THE IRRADIATION PORTS OF THE UNIVERSITY OF UTAH RESEARCH REACTOR

by

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Scientific paper
DOI: 10.2298/NTRP1203222N

Neutron and gamma flux environment of various irradiation ports in the University of Utah training, research, isotope production, general atomics reactor were experimentally assessed and fully modeled using the MCNP5 code. The experimental measurements were based on the cadmium ratio in the irradiation ports of the reactor, flux profiling using nickel wire, and gamma dose measurements using thermo luminescence dosimeter. Full 3-D MCNP5 reactor model was developed to obtain the neutron flux distributions of the entire reactor core and to compare it with the measured flux focusing at the irradiation ports. Integration of all these analysis provided the updated comprehensive neutron-gamma flux maps of the existing irradiation facilities of the University of Utah TRIGA reactor.

Key words: neutron flux, TRIGA reactor, MCNP5 code, irradiation port, gamma dose

INTRODUCTION

Neutron energy spectrum and neutron and gamma flux must be well known in order to accurately quantify the experiments performed at any research reactor [1]. The objective of this research was to develop an MCNP5 model and validate it by developing numerous experiments at training, research, isotope production, general atomics (TRIGA) reactor at the university of utah (UUTR) in profiling neutron flux and gamma dose of irradiation ports – the thermal irradiator (TI) and the fast neutron irradiation facility (FNIF). Although many experiments have been conducted in past decades, in order to characterize the irradiation environment of the UUTR, the quantification of the energy spectrum and neutron flux in the FNIF was never performed, while the TI flux mapping was not tested for over a decade.

The University of Utah TRIGA nuclear research reactor operates at 100 kW (thermal). The reactor core is a heterogeneous assembly of standard TRIGA fuel elements, deuterium oxide reflector elements, and three neutron absorbing control rods containing boron carbide all supported in a hexagonal lattice by an aluminum grid structure. The cylindrical stainless steel and aluminum-clad fuel elements contain solid, homogeneously dispersed uranium-zirconium hydride enriched in less than 20% in the fissile isotope of ^{235}U [1]. The predominant utilization of the UUTR is as a copious neutron source for activation analysis, nuclear research, and training. The accuracy and assessment of experimental uncertainties depends on the accuracy of the neutron spectrum characterization [2].

Samples may be irradiated at several positions within the UUTR, and there are two primary extra-core facilities for sample irradiations – the TI and FNIF. The TI consists of a tank filled with D_2O and provides an isotropic thermal neutron environment that is particularly well suited for neutron activation analysis and basic nuclear medicine related experiments. Whereas the FNIF consists of a lead tank where neutron moderation is minimal [1].

One of the present-day methods to experimentally quantify the thermal neutron flux is to compare the quantity known as the cadmium ratio, which is calculated via the measurement of the activity of gold foils through the $^{197}Au(n, \gamma)\ ^{198}Au$ reaction [3]. These measurements were utilized to create a topical flux map of the UUTR and analyze the various irradiation ports. MCNP5 was used to compute the cadmium ratio of the entire reactor core and compare against the measurements for this validation study [4].

Nickel mapping is another experimental technique used to quantify thermal and fast neutron flux. Comparison of ^{65}Ni via (n, γ) reaction and ^{58}Ni via (n, p) reaction, create a flux distribution map over a specific reactor core area [5]. This was conducted within the FNIF to obtain a vertical map, showing how flux varied with depth of the FNIF. By using thermo luminescence dosimeters (TLD) within the FNIF, the gamma induced irradiation environment was mea-

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sured [6, 7]. Using these methods in conjunction with each other, it was possible to create required understanding of the radiation environment at the UUTR's irradiation ports.

METHODS AND DESCRIPTION

Cadmium ratio

Experimental discrimination between thermal and fast neutron spectra is possible using cadmium foils. The cadmium ratio is defined as the ratio of the activity of a bare detector, gold foil, to the activity of the same but covered detector, *i. e*. a cadmium-covered foil [8]. The bare irradiated gold foil is exposed to neutrons of all energies. The cadmium-covered gold foil is activated mostly with fast neutrons. If A_0 and A_{Cd} represent the activities for bare and cadmium covered gold foils, respectively, then the difference between them corresponds to the activity of the bare gold foil activated by thermal neutrons, which are absent in the gold foil covered with cadmium foil. This is because nearly all neutrons below the cadmium cut-off energy, which ranges from about 0.4-0.6 eV, are absorbed by a cadmium foil, while nearly all neutrons above this energy passing through the cadmium [3]. The cadmium ratio R_{Cd} is, therefore, determined as follows

$$R_{Cd} \quad \frac{A_0}{A_{Cd}} \tag{1}$$

The experiment performed at the UUTR provided the data to assess the cadmium ratios in TI and FNIF. Great attention was paid to appropriate geometrical positioning of the gold foils in ensuring the two foils were equally placed in respect to the reactor core center. Since the vertical flux distribution in the TI is Gaussian, one gold foil was placed just above the vertical center and the other just below the vertical center and irradiated at the same time thus ensuring identical exposure to neutron flux. Both foils were irradiated for 5 minutes at a power of 30 1.2 kW; the foils were left to decay for three days before their activity was measured.

In the FNIF, very small (surface area of 1 mm^2), thin (0.0024 mm thickness) square gold foils were irradiated in exactly the same environment under the same flux conditions. Foils were irradiated at the point of maximum flux in the FNIF for 15 minutes at power level of 10 0.4 kW; the foils were then left to decay for three days before their activity was measured.

Measurements and instrumentation

^{198}Au emits 411.8 keV gamma rays after being irradiated. Using a high purity germanium (HPGe) detector, a very narrow gamma energy window, 411.7-411.9 keV, was used to identify the activity of the gold foils. By using a variance window, more accurate results were achieved which accounted for a small discrepancy in emitted gamma ray energy. Figure 1 shows two spectra acquired from the HPGe detector. From these spectra it was obtained that the number of detected gamma rays with energy 411.7-411.9 keV were about (4.18 0.006) 10^5 for the bare gold and (1.01 0.003) 10^5 for the cadmium-covered detector in the TI. These values were then used in eq. (1) to obtain an experimental cadmium ratio of 4.14 0.015 in the TI. Similarly, using the HPGe detector the activity of the bare gold in the FNIF was obtained to be (4.22 0.006) 10^5 Bq and (2.10 0.005) 10^5 Bq for the cadmium-covered gold detector, resulting in an experimental cadmium ratio of 2.01 0.005.

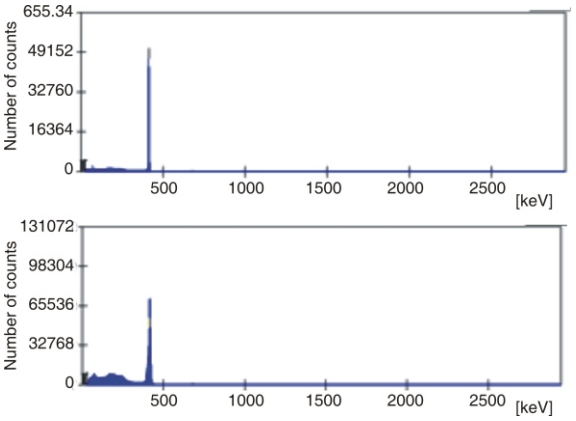

Figure 1. Gamma spectroscopy of bare gold foil (top) and cadmium-covered foil (bottom) irradiated in the UUTR TI

MCNP5 simulation to determine cadmium ratio in UUTR TI port

An exact MCNP5 model of the UUTR core – including the fuel, and other materials present in the core – was used to model the gold foil TI experiment exactly as was physically conducted in order to accurately compare with the measured values and validate the model [2]. The ENDF-7 cross-section data library was selected and the simulation was run in (n, p) mode which accounts for neutrons and gamma rays. Table 1 shows the result from this MCNP5 simulation. Two billion particles were simulated and the cadmium ratio was determined based on eq. (1). The errors are relative standard deviations as obtained in the MCNP5 code and are relatively high even with a very large number of particles. Based on the analysis of the error trends for different numbers of particles, the two billion particles provided close to a constant error value. Further increase in number of particles will provide a slight reduction in the MCNP5 reported errors. A re-

Table 1. MCNP5 cadmium ratio in TI

History (million)	CPU time (minutes)	Activity [Bq]	Error	Cadmium ratio
2.000	4.927	7.289 10^4 (bare)	0.4749	4.025 2.763
		1.811 104 (covered)	0.4958	

duction of the computational error can be optimized by increasing the number of particles and also by adopting a variance reduction method such as energy and time cutoffs, as well as weight cutoff [9]. However, because of the good agreement with the experimental measurements, further optimization of the MCNP5 error reduction vs long CPU time is not found necessary.

Computational methods, particularly full three-dimensional models of the UUTR using Monte Carlo methods are very powerful in providing information on detailed shape of the neutron spectrum and flux, but the results require careful validation to eliminate modeling errors, biases due to approximations in the methods and uncertainties in nuclear data [2]. Thus, the spectra are validated by comparing to measured values.

Figures 2-6 show spatial neutron flux distribution in the UUTR for different neutron energies based on the MCNP5 model. By using a mesh tally in MCNP5 for the entire reactor core, a plot was created in MATLAB for each energy region [10-12]. This shows how drastically the neutron flux changes at different positions within the reactor core for different neutron energies.

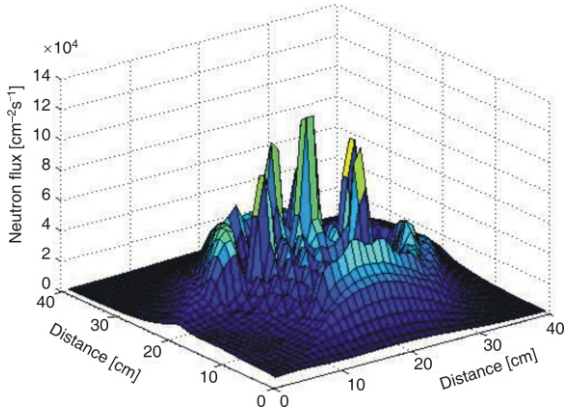

Figure 2. UUTR flux map for neutron energies 0-0.025 eV

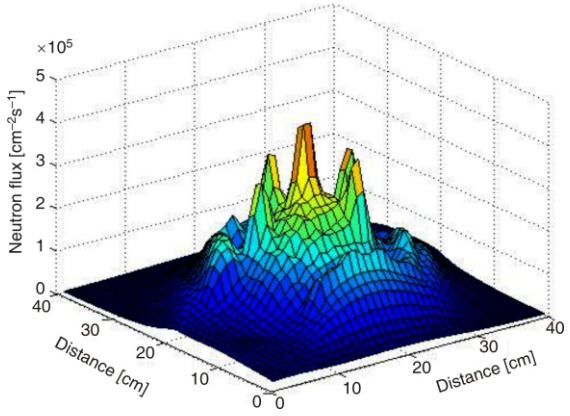

Figure 3. UUTR flux map for neutron energies 0.025-0.5 eV

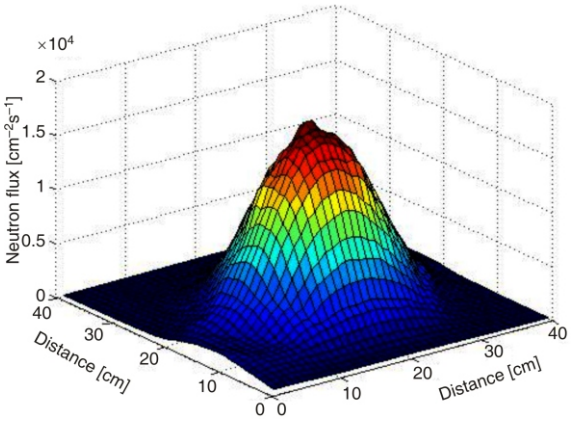

Figure 4. UUTR flux map for neutron energies 0.5 eV-1.0 eV

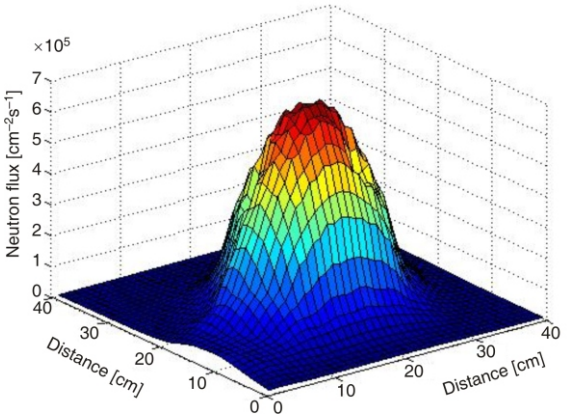

Figure 5. UUTR flux map for neutron energies 1.0 eV- 20 MeV

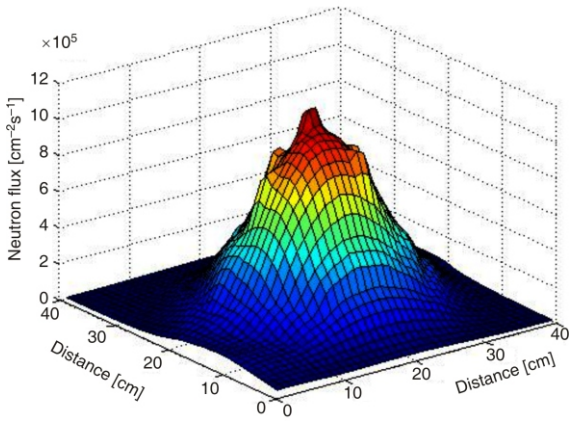

Figure 6. UUTR flux map for total neutron energy

Figure 7 shows a map of the MCNP5 based cadmium ratio for the entire reactor. Mesh tally matrices of the reactor core were utilized to take the ratio of fast neutrons to neutrons of all energies [10-12]. It was known that the cadmium cut-off energy varies 0.4-0.6 eV. Figure 7 was created assuming that the cut-off energy is 0.5 eV. Based on this figure, the cadmium ratio was approximately determined to be 4 ± 0.3 in the TI, and 2 0.2 in the FNIF where the color bar represents the cadmium ratio. These values agree well with the experimental cadmium ratio for both irradiation ports as described previously and are thus validated.

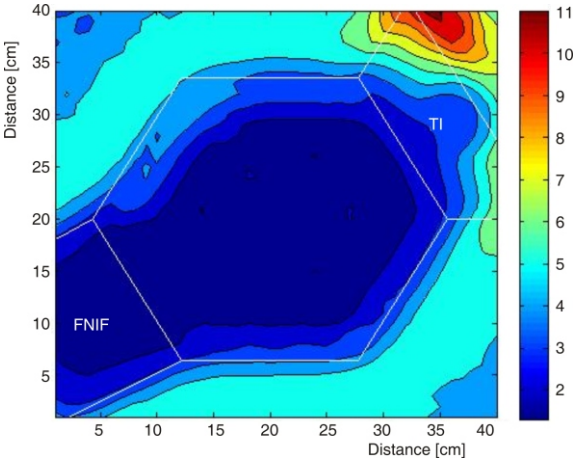

Figure 7. UUTR cadmium ratio flux map indicating cadmium ratio of 4.0 in the TI and 2.0 in the FNIF

Nickel flux profiling

Another method used to obtain neutron flux profile in UUTR irradiation ports, was based on irradiation of nickel foils. Neutron irradiation of metallic nickel results in the production of two radioisotopes: ^{65}Ni via (n, γ) reaction and ^{58}Ni via (n, p) reaction. In the experiment, fifteen nickel foils were arranged 2.54 cm apart to create a grid and were irradiated in the FNIF for 15 minutes at the UUTR power of 10 ± 0.4 kW. The dashed line in fig. 10 shows the area of maximum flux where the grid was formed in the FNIF. An HPGe detector was used to detect these radionuclides in the irradiated nickel foils. The radiative capture microscopic cross section for ^{65}Ni is small for the fast and epithermal neutron energies. Thus, the detection of ^{65}Ni represents the thermal flux received by the nickel foil [7]. By using an interference correction report generated by and HPGe detector, the weight mean activity [Bqg^{-1}] of each radionuclide was compared for each foil. Thus, fast flux and thermal flux maps can be produced. Figures 8 and 9 show thermal and fast spatial neutron flux distribution in the FNIF expressed in neutrons per centimeter squared per second and represented as a percentage of the maximum flux. Compared to figs. 3-7, based

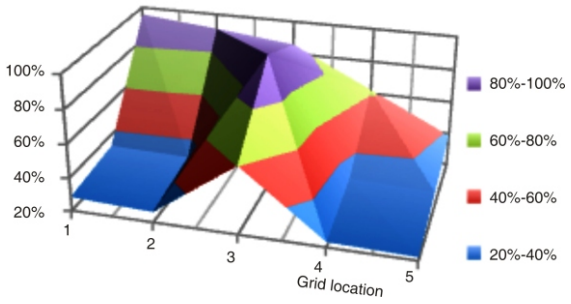

Figure 8. UUTR FNIF thermal flux distribution based on nickel wire experiment with 2.54 cm spacing

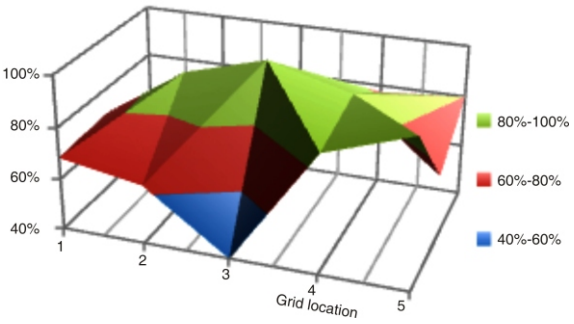

Figure 9. UUTR FNIF fast flux distribution based on nickel wire experiment with 2.54 cm spacing

on MCNP5, figs. 8 and 9 show the flux details inside of the FNIF and how it varied with the FNIF depth.

The neutron flux map was also measured experimentally using 84 sulfur pellets (2 0.0002 grams of mass each) providing the contour lines as depicted in fig. 10. These 84 pellets were located inside the FNIF in a way to assure the grid mapping, in other words seven of the pellets were placed across along the grid row, and twelve along the grid columns 2.54 cm apart. The sulfur pellets were irradiated for 15 minutes inside of the FNIF at the reactor power of 1 0.04 kW. This experiment was repeated three times in order to minimize the statistical error. When ^{32}S reacts with neutrons of energy greater than 2 MeV, ^{32}P that is a pure beta-emitter, is formed. This beta particle coming from ^{32}P has a maximum energy of 1.709 MeV and is measured using a Geiger-Muller detector to determine neutron flux contour lines [13-15]. This neutron flux map does not provide any information about neutron energy distribution or thermal neutron flux, but it provides information about fast neutron flux distribution. As can be seen from fig. 10, the peak thermal and fast flux occurred at the same location that is slightly above the axial center of the FNIF in the reactor.

Dosimetry measurements

A complete understanding of the FNIF radiation environment was achieved by measuring in addition, the gamma dose as a function of equivalent neutron

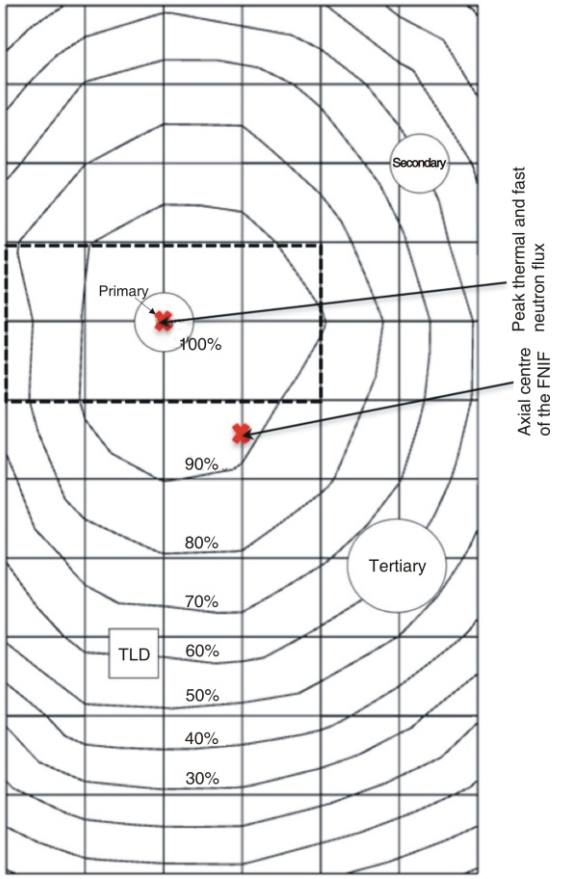

Figure 10. Experimentally determined flux map for the FNIF (using sulfur pellets) showing peak flux location; dashed line refers to the area used for dosimetry measurements

fluence. The primary source of gamma radiation in the FNIF environment is from prompt fission and delayed decay gamma rays coming out form the reactor core. The gamma energy spectrum in the FNIF is modified by background sources of gamma radiation near and surrounding the irradiator *i. e*. pool water and aluminum. Compton-scattering gamma rays and photons produced via neutron capture reactions constitute the majority of the gamma dose within the irradiation port [1]. Thus, the gamma energy spectrum in the irradiation port was expected to be a continuum related to the prompt gamma spectrum upon which are superimposed peaks associated with the capture and decay gamma rays [1].

Thermo luminescence dosimeters (TLD) were placed at the point of 80% maximum flux in the FNIF and irradiated at power of 1 0.04 kW for 15 minutes and at 10 1.2 kW for 15 minutes. A Harshaw TLD System Model 3500 Manual TL Reader was used to measure the gamma dose received by each dosimeter from the FNIF. The dose values in FNIF are shown in tab. 2. The TLD reader subjects the dosimeter to a precise heating cycle to stimulate TL emission and a photomultiplier tube converts the TL photons into electrical current to yield a total collected charge [1].

Table 2. TLD gamma dose values as measured in FNIF

Gamma dose [Sv]	Error [Sv]	Neutron fluence [cm^{-2}]	Reactor power	Dose equivalent per unit fluence [$Sv\,cm^2$]
4.55	0.25	$1.33\ 10^{12}$	1 kW	$3.42\ 10^{-12}$
6.37	0.11	$1.78\ 10^{12}$	1 kW	$3.58\ 10^{-12}$
5.63	0.14	$1.78\ 10^{12}$	1 kW	$3.16\ 10^{-12}$
5.85	0.19	$1.78\ 10^{12}$	1 kW	$3.29\ 10^{-12}$
5.70	0.25	$1.78\ 10^{12}$	1 kW	$3.20\ 10^{-12}$
7.26	0.58	$1.95\ 10^{12}$	1 kW	$3.72\ 10^{-12}$
6.09	0.12	$1.74\ 10^{12}$	1 kW	$3.50\ 10^{-12}$
6.38	0.27	$1.74\ 10^{12}$	1 kW	$3.67\ 10^{-12}$
1.74	0.90	$5.09\ 10^{11}$	1 kW	$3.42\ 10^{-12}$
Average dose equivalent per unit fluence at 1 kW				$3.44\ 10^{-12}$
310	18.0	$4.81\ 10^{13}$	10 kW	$6.44\ 10^{-12}$

The response is compared to the response of identical dosimeters that have been calibrated in a NIST-traceable gamma source to determine the absorbed dose received during the irradiation. The TLD reader does not provide any error from the machine; standard error was calculated for the single measurement using previous benchmark study conducted at UUTR which observed that TLD error remained constant at around 5% of the mean except when the gamma dose approached the lower limit for the TLD (0.01 Sv) the error rose almost exponentially reaching 80% [6].

TLD irradiation was repeated several times for a UUTR power of 1 kW to ensure accuracy. In all experiments, the dosimeters were used only once. Table 2 also shows the correlation of gamma dose with varying neutron fluence. The average dose equivalent per unit fluence was $3.44\ 10^{-12}$ Sv cm^2 per neutron for 1 kW UUTR power and $6.44\ 10^{-12}$ Sv cm^2 per neutron for 10 kW power. As expected, the gamma dose received by the TLD increases linearly with increasing reactor power and can be extrapolated for larger neutron fluence.

CONCLUSIONS

The neutron flux in reactor cores depends on geometrical location and neutron energy; since it determines the reaction rate, it is very important to have accurate maps of thermal and fast neutron fluxes and gamma dose distributions in assuring the highest possible accuracy of the experiments [16]. The cadmium ratios in TI and FNIF of the UUTR were experimentally measured and also estimated based on the detailed MCNP5 simulation. The experiment and MCNP5 simulation have yielded results of about 4.1 for the cadmium ratio in the TI and about 2.0 in the FNIF. The error from MCNP5 simulation was relatively large even with a very large number of particles.

This computational error can be further optimized by increasing the number of particles further, and also by adopting a variance reduction method such as energy and time cutoffs, and weight cutoff. However, because of the good agreement with the experimental measurements, further optimization of the MCNP5 error reduction vs long CPU time is not found necessary, and thus remained for future studies.

In the FNIF there were twice as many epithermal/fast neutrons than thermal neutrons. A three-dimensional cadmium ratio map was created using MCNP5 and easily used (since it was experimentally validated) to predict the cadmium ratios at other available locations in the reactor core.

The measured axial flux distributions were in good agreement with the calculated value and gave a better understanding of the irradiation environment within the FNIF. The gamma dose measurements were recorded for varying fluence and it was found that gamma dose linearly increases with increasing neutron fluence, which can be extrapolated for larger neutron fluence. The average dose equivalent per unit fluence was 3.44 10^{-12} Sv cm^2 per neutron for 1 kW UUTR power and 6.44 10^{-12} Sv cm^2 per neutron for 10 kW power.

Various methods were utilized to evaluate the irradiation environments in the University of Utah TRIGA including cadmium ratio experimental data, MCNP5 simulations, nickel flux profiling, and TLD measurements. Using these methods in conjunction with each other created a complete understanding of the neutron & gamma irradiation environment present in the reactor.

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Received on August 12, 2012
Accepted on September 5, 2012

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ЕКСПЕРИМЕНТАЛНА ПОТВРДА И MCNP5 ЕВАЛУАЦИЈА НЕУТРОНСКОГ И ГАМА ФЛУКСА У ИРАДИЈАЦИОНИМ КАСЕТАМА ИСТРАЖИВАЧКОГ РЕАКТОРА НА УНИВЕРЗИТЕТУ У ЈУТИ

Неутронски и гама флуксеви у ирадијационим касетама у истраживачком реактору ТРИГА на Универзитету у Јути експериментално су измерени и моделовани употребом MCNP5 програма. Експеримент је заснован на мерењу кадмијум коефицијента у ирадијационим касетама у реактору, а неутронски флукс је измерен коришћењем никла, док је гама доза измерена термолуминисцентним дозиметром. Комплетан тродимензиони модел реактора развијен је према MCNP5 програму са циљем да се добије расподела неутронског флукса у целом реактору укључујући ирадијационе касете. Рачунарски одређен флукс упоређен је са мерењима. На основу експерименталних и нумерички одређених вредности неутронског и гама флукса добијене су детаљне мапе флукса у ирадијационим касетама истраживачког реактора на Универзитету у Јути од интереса за будуће експерименте.

Кључне речи: неутронски флукс, ТРИГА реактор, MCNP5 код, ирадијациона касета, гама доза