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Development and Validation of a Serpent-2 Model for the Former 3 MW TRIGA Core Configuration of the Philippine Research Reactor-1

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The 3 MW TRIGA (Training, Research, Isotopes, General Atomics) Philippine Research Reactor-1 (PRR-1) at the DOST-PNRI achieved its first criticality on 08 Mar 1988 after its successful upgrade from a plate-type reactor. However, due to unresolved technical problems discovered weeks after the upgrade, the PRR-1 was considered inoperable and has been in shutdown status since then. The slightly irradiated TRIGA fuel rods of the PRR-1 are currently in an interim storage tank and are planned to be utilized in a subcritical reactor assembly. As part of the project to reuse the fuels, simulation models for both present and proposed configurations are important. In this work, we present the complete model of the former configuration of PRR-1 with 115 TRIGA fuel rods developed with the Serpent Monte Carlo code version 2 for simulation of criticality and neutronic analysis. The model of the TRIGA fuel rods was validated in the fresh fuel configuration through the benchmark analysis described in the 1988 reactor criticality report. The effective multiplication factors from the Serpent-2 simulation (k_{eff} = 1.0690 ± 0.0012) and measured value of 1.0661 have been found to agree with a deviation of 259 pcm. Neutron flux and fission power distribution simulations using the same reactor configuration were also presented to serve as reference for future burn up calculations and fuel characterization.

Keywords: flux distribution, Monte Carlo, Serpent 2, TRIGA research reactor

INTRODUCTION

Research reactors (RR) have been the driving force behind innovation and growth of nuclear science and technology. RRs are a type of nuclear reactors with the main purpose of generating neutrons for different applications. The PRR-1 is the sole RR in the Philippines and is housed at the DOST-PNRI. It was used for a wide array of scientific activities covering fundamental research in reactor physics, nuclear physics, nuclear chemistry, radiobiology, radioisotope production, neutron activation analysis, material testing, and training during its operation from 1963 to 1983 (Dela Rosa and Aleta 1992). The original 1-MW reactor utilized fuel assemblies with plate-type fuel made of uranium-aluminum alloys and was converted into

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a 3-MW reactor with TRIGA fuel rods, replacing each plate-type fuel assembly with a cluster of four rods. The upgraded PRR-1 achieved its first criticality on March 1988. However, a few weeks after the conversion, the PRR-1 was considered inoperable due to unresolved technical problems and has been in shutdown status since then (FNCA-RWM 2008).

The inoperability of the PRR-1 has left a challenge in preserving nuclear science and engineering knowledge and expertise in the country. To address this problem, the DOST-PNRI decided in 2014 to utilize the TRIGA fuel rods in a subcritical configuration for training, education, and research. Capacity building, which include personnel training and infrastructure development as well as research and development projects, have been initiated in support of the utilization plan. This would develop the expertise of researchers and increase the research outputs in nuclear science and engineering. Monte Carlo-based simulation of various fuel configurations has been performed in preparation for the design of a subcritical reactor assembly that uses the PRR-1 TRIGA fuel (Asuncion-Astronomo *et al.* 2019). The specifications of the fuel rods and the PRR-1 configuration are both documented and can be validated with recorded measurements.

In this paper, we developed a three-dimensional model of the PRR-1 core with TRIGA fuel rods, which was validated by using the reactivity values reported in the PRR-1 TRIGA conversion reports. The model includes details of essential core components that can influence the evaluation of neutron flux and fission energy distribution within the core. It was implemented using Serpent-2 Monte Carlo continuous-energy radiation transport code, developed by VTT Technical Research Center of Finland, Ltd. (Leppänen 2015). The code is becoming a popular and validated tool for TRIGA RR simulations (Ćalić *et al.* 2016, Viitanen and Lappänen 2016, Castagna *et al.* 2018). The simulation results include the distribution of neutron flux and fission power, the effective neutron multiplication factor (k_{eff}), and kinetic parameters of the core configuration. These were subsequently compared with the analysis report of the TRIGA PRR-1 control blades/rods calibration data performed in 1988, for validation of the simulation model.

THEORY

RRs are mainly utilized as a source of neutrons and thus, the determination of the neutron population in the system is important. The effective multiplication factor (k_{eff}) is a parameter used to quantify the multiplication of neutrons in the reactor. The k_{eff} value is a measure of the change in the fission neutron population from one generation to the subsequent generation and is defined in Equation 1.

$$k_{eff} = \frac{\text{Number of neutrons in one generation}}{\text{Number of neutrons in preceeding generation}} \quad (1)$$

A reactor with k_{eff} = 1 has an approximately constant number of neutrons and is in a critical state. When k_{eff} < 1, the reactor system is in subcritical state, while it is supercritical when k_{eff} > 1. The deviation of the multiplication factor k_{eff} from the critical state or the critical value of 1 is described using an important parameter in reactor physics known as reactivity (ρ), which is described in Equation 2.

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \quad (2)$$

ρ is an important dimensionless parameter for safety. It is often quoted in terms of pcm which is simply equal to $\rho \times 10^5$. While in RRs, it can be convenient to normalize ρ with respect to the effective delayed neutron fraction (β_{eff}), which expresses the reactivity in terms of the unit dollar ($) as shown in Equation 3. The β_{eff} value is the ratio of neutrons that appear as delayed neutrons over all neutrons released after fission (Duderstadt and Hamilton 1976). It is one of the important kinetic parameters reported in the safety analysis report of the PRR-1, along with the prompt neutron lifetime – denoted by l_p – which is the average time between the emission of prompt neutrons and their absorption in a reactor. If all control elements and poisons were instantaneously removed from the reactor, the reactivity at this state is termed as excess reactivity.

$$\$ = \frac{\rho}{\beta_{eff}} \quad (3)$$

MATERIALS AND METHODS

Serpent-2 Model of the TRIGA Fuel Rods

The PRR-1 TRIGA fuel rod is 75.18 cm long and 3.07 cm in diameter, as shown in Figure 1. The TRIGA fuel-moderator elements (FE) comprise of low enriched uranium and erbium homogenously combined with a zirconium hydride moderator. The active section of the TRIGA fuel rod containing the FEs is about 50.8 cm long, has a 2.97 cm diameter, and contains approximately 20 wt% uranium enriched to 19.7% U-235. Mixed with UZrH as burnable poison is 0.45 wt % erbium and the H/Zr ratio of the FE is about 1.6. Table 1 summarizes the atom fractions of FEs as used in the materials composition of the Serpent-2 simulation. The active fuel section with the top and bottom graphite pieces were contained in a 0.05 cm thick Incoloy-800 cladding with welded fittings

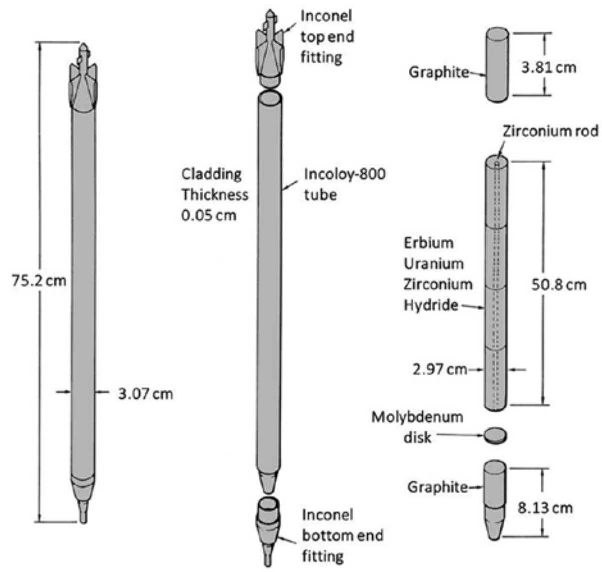

Figure 1. PRR-1 TRIGA Incoloy-800 clad fuel-moderator.

Table 1. Composition of UErZrH alloy as simulated in Serpent-2.

Elements	Composition of isotopes (in atom fractions)
Erbium	^{162}Er (1.615552E–06); ^{164}Er (1.860791E–05); ^{166}Er (3.893945E–04); ^{167}Er (2.657990E–04); ^{168}Er (3.135566E–04); ^{170}Er (1.732941E–04)
Hydrogen	^{1}H (5.922097E–01); ^{2}H (6.811195E–05)
Uranium	^{235}U (7.247425E–03); ^{238}U (2.913886E–02)
Zirconium	^{90}Zr (1.904543E–01); ^{91}Zr (4.153348E–02); ^{92}Zr (6.348478E–02); ^{94}Zr (6.433618E–02); ^{96}Zr (1.036486E–02)

made of Inconel-600. The fuel element was separated from the bottom graphite reflector by a molybdenum disk. The simulation model was in fresh fuel configuration; thus, any contamination of fission products was not considered.

Serpent-2 Model of the TRIGA Reactor Core

The TRIGA PRR-1 has a total of 115 fuel rods that are inserted into rectangular hollow shrouds made of aluminum; each shroud can accommodate up to four fuel rods. A detailed model of the reactor core is shown in Figure 2, which consists of the four-rod clusters modeled individually by macrobody geometry. The three in-core irradiation tubes, two guide tubes, control blade shrouds, uranium irradiation, and two dry pipes were also included in locations indicated in Figure 2. Surrounding the fuel rods were graphite blocks that act as reflectors to increase the amount of fission.

Criticality Calculations

The k_{eff} of the PRR-1 TRIGA reactor was simulated using Serpent version 2. Accompanying the simulation results of k_{eff}, the results of kinetic parameters β_{eff} and l_p were also reported in this work, as provided by Serpent. The neutron cross-sections as well as the thermal scattering cross-sections, which are important to simulate the moderating properties of zirconium hydride and water, were taken from the ENDF/B-VII.I nuclear data library. The simulation required an initial guess for k_{eff}, which was set to unity. A total of 2000 source neutrons per cycle was used in the simulation. The first 1000 cycles of the simulation were discarded to allow convergence of fission points. This is also to ensure that the result was not influenced by a poor initial guess. Results from the succeeding 500 cycles were recorded. The initial source points were randomly selected inside the fissile cells in the geometry and no source input was needed. Criticality calculation, which will determine k_{eff}, was done for a reactor configuration where the control rods and control blades were taken out or fully withdrawn from the core.

Neutron Flux and Fission Power

The Serpent has a built-in capability to visualize the fission power and thermal flux distributions in a single graphics file using the mesh plotter. The code calculates the reaction rates and projects the data according to the specified plot plane orientation. In this work, the total thermal flux and fission power distribution are mapped on horizontal and vertical planes.

RESULTS AND DISCUSSION

Criticality Calculation

The nuclear criticality, or the ability to sustain a chain reaction by fission neutrons, is characterized by the k_{eff}. The calibration of the reactor control element that was performed in 1988 determined the excess reactivity of PRR-1 to be 8.73 \$. Serpent-2 calculations of the k_{eff} for the fresh core were performed with control elements fully withdrawn from the core. Since the value of k_{eff} was not recorded in previous reports, the value of 1.0661 was obtained from the given excess reactivity value using

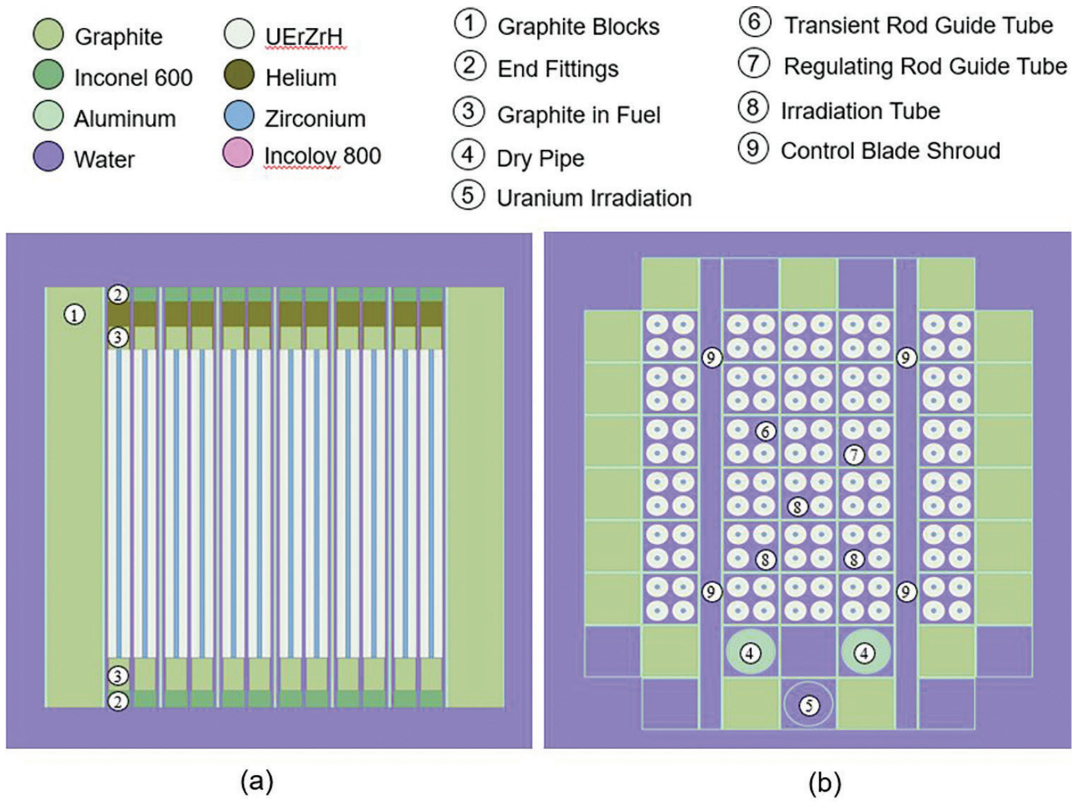

Figure 2. The (a) vertical and (b) horizontal section of the Serpent reactor model.

Equations 2 and 3 and the fraction of delayed neutrons (β_{eff}) declared by the manufacturer as 0.0071.

The comparison between the values reported in the safety analysis report and Serpent-2 simulated values, including the propagation of the simulated error, are tabulated in Table 2 for the k_{eff}, the excess reactivity, and the kinetic parameters. The simulated values are found to have a deviation of 259 pcm using the ENDF/B-VII.I nuclear data library. The discrepancy, nonetheless, remains quite reasonable and may be attributed to minor simplifications implemented in the geometry of the model and because of the neutron data libraries. Additionally, the simulated kinetic parameters are found to agree with the given data in the safety analysis report. These results indicate the validity of the TRIGA PRR-1 model used in the simulation.

Neutron Flux and Fission Power

In the first part of the study, the Serpent-2 model of the TRIGA PRR-1 was validated using the k_{eff} calculated from the excess reactivity, $ that is documented in the PRR-1 safety analysis report. As a possible application of the reactor model, it was used for the determination of thermal flux and fission power distribution in the former PRR-1 core configuration. These factors are important for the estimation of the reactor core power and temperature distribution in the core. Moreover, these factors provide information that can help characterize the burn-up of each fuel elements, which is essential for the reuse of the TRIGA fuels.

The neutron thermal flux and fission power distribution in the core can be observed in color schemes, as shown in Figure 3. The hot shades of red and yellow represent relative fission power while the cold shades of blue represent relative thermal flux (flux below 0.625 eV).

Table 2. Comparison of k_{eff}, β_{eff}, l_p, and excess reactivity, $, between the measurement reported in the safety analysis report and the Serpent-2 simulation.

	k_{eff}	β_{eff}	l_p, µs	Excess reactivity, $
Measured	1.0661	0.0071	42.0	8.7
Simulated	1.0690 ± 0.0012	0.0073 ± 0.0004	42.6 ± 0.3	9.0 ± 0.6

Fission power is highest at the center of the core. On the other hand, lighter blue color – signifying higher neutron thermal flux – is seen in regions where there are no thermal neutron absorbers.

The graphical representation of the total thermal neutron flux distribution of the reactor core is shown in Figure 4. The thermal flux is not symmetric in the horizontal section, as shown in Figure 4a since fuel rods were not symmetrically positioned. There are empty slots in the four-rod aluminum cluster, which are designated for irradiation channels and control rod slots. Relatively high distribution of thermal flux can be found in the regions with no fuel rods. The increase in thermal flux in these regions is due to slowing down of fast neutrons, which escaped from the adjacent fuel rods. Thermal neutrons are not absorbed quickly due to a much smaller absorption cross-section in these regions since there are no absorbing fuel rods. The increase of thermal neutron flux along the reflector demonstrates the efficiency of the reflecting graphite surrounding the fuel rods. It can also be observed that the peak of thermal flux in the vertical section of the reactor model is at the center of the core. Similarly, the slight increase in the thermal neutron flux along the vertical section shown in Figure 4b is due to the graphite reflector in the fuel rod.

CONCLUSION

The Serpent model of the TRIGA PRR-1 in which all control elements were withdrawn from the core has been validated using the experimental data for k_{eff} obtained from PRR-1 conversion reports. The parameters used in the fuel model can be employed in the design of an intended subcritical reactor assembly. The figures of neutron flux and fission power distributions show the areas where the fuel burn-up is relatively higher, which will be significant in future burn up calculations and fuel utilizations.

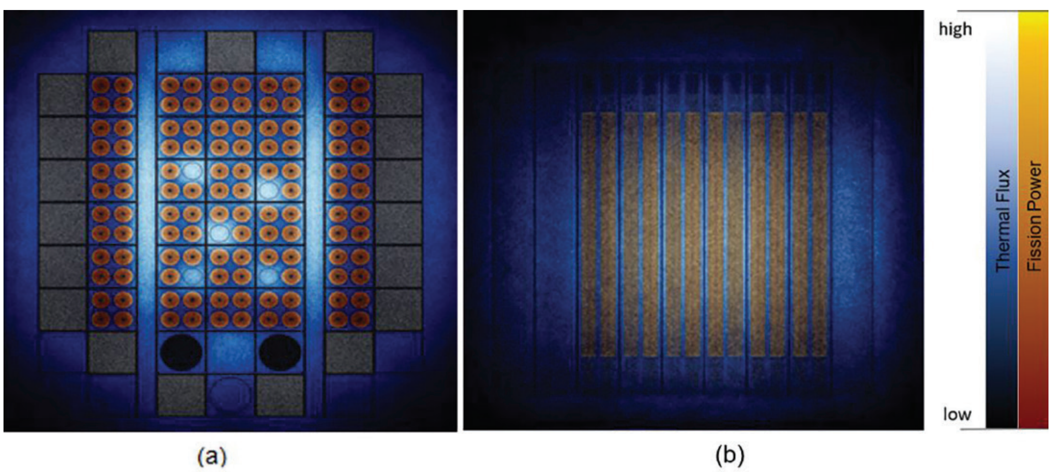

Figure 3. Total fission power and neutron thermal flux distribution in the core of TRIGA PRR-1 simulated in Serpent; (a) horizontal section and (b) vertical section of the reactor model.

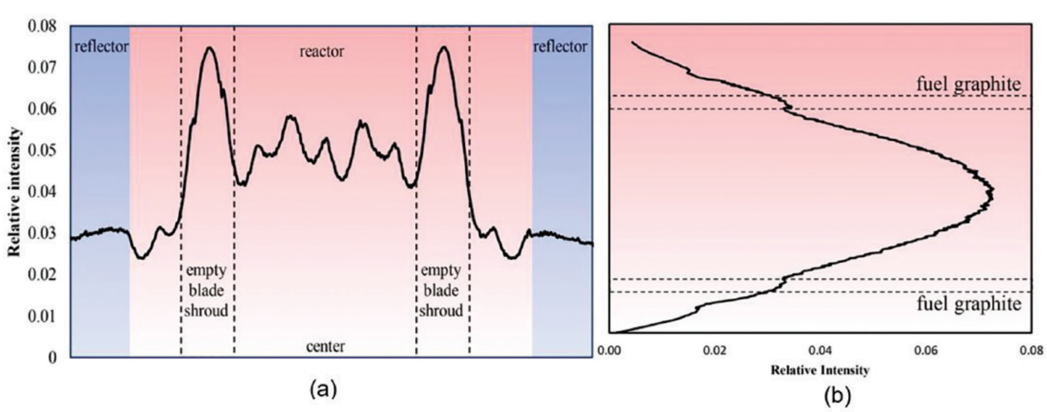

Figure 4. Graphical representation of total neutron thermal flux distribution: (a) horizontal section and (b) vertical section of the reactor model.

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