Fuel burnup analysis for Thai research reactor by using MCNPX computer code

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Abstract. This paper presents the fuel burnup analysis of the Thai research reactor (TRR-1/M1), TRIGA Mark-III, operated by Thailand Institute of Nuclear Technology (TINT) in Bangkok, Thailand. The modelling software used in this analysis is MCNPX (MCNP eXtended) version 2.6.0, a Fortran90 Monte Carlo radiation transport computer code. The analysis results will cover the core excess reactivity, neutron fluxes at the irradiation positions and neutron detector tubes, power distribution, fuel burnup, and fission products based on fuel cycle of first reactor core arrangement.

1. Introduction

One piece of crucial information for regulatory purpose is the burnup data of the regulated reactor. The burnup data of each fuel element will reflect reactivity, neutron flux, power distribution, and fission products. Each fuel reshuffling will be permitted by the regulator only when such reshuffling will abide the operating limit conditions, in particular excess reactivity, thermal power, and fuel temperature. In addition, the fission products generated present some concern regarding activity level when moving fuel elements from one place to another within the reactor pool or transferring fuel elements to a storage place. Besides safety concern, the regulator also focuses on the remaining quantity of U-235 and generated quantity of plutonium for the safeguards purpose.

Several tools exist in calculating fuel burnup. One popular and reliable software tool is MCNPX (Monte Carlo N-Particle eXtended Transport Code) developed by Los Alamos National Laboratory [1]. It is a Fortran90 Monte Carlo radiation transport computer code that models the interaction of radiation with matter, capable of burnup analysis.

This paper aims to present the burnup analysis of the Thai research reactor (TRR-1/M1), TRIGA Mark-III, at Thailand Institute of Nuclear Technology (TINT) in Bangkok, Thailand. Specifically, only the first core arrangement of fuel elements is analyzed. The results will further serve as an input for the calculation of the subsequent core arrangements to benchmark with the actual operating data and to evaluate radiation safety as well as to assess nuclear security and safeguards.

2. The TRR-1/M1 Reactor

2.1. The Description of TRR-1/M1 Reactor

The TRR-1/M1 reactor is an open pool type TRIGA Mark-III research reactor, moderated and cooled by light water. The TRR-1/M1 reactor was converted from an MTR-type reactor (TRR-1) into rod types in 1975 and it has been operated for the modified type reactor (TRIGA Mark-III) since 1977.
The TRR-1/M1 reactor was designed for operation at a maximum steady state thermal power of 2000 kW. However, in 2012, the maximum thermal power is limited at 1300 kW for operation. The perspective view of the TRR-1/M1 reactor is shown in Figure 1.

The fuel for the TRR-1/M1 reactor in the first core loading is the solid, homogeneous mixture of uranium-zirconium hydride (UZrH) alloy with Uranium-Zirconium atomic ratio of 1.6 to 1.7. The power level of the reactor is controlled with five control rods; a safety rod, a regulating rod, two shim rods, and a transient rod. The safety rod, regulating rod, and shim rods are fuel follower control rods. The neutron absorber for all control rods is boron carbide in a solid form. The cross-section view of the reactor core configuration is shown in Figure 2.

The reactor core has a cylindrical configuration. There are 121 locations in the core that can be filled by fuel elements, control rods, neutron source, irradiation tubes, or neutron detectors. Elements are arranged in seven concentric rings in hexagonal geometry and spaces between the rods are filled with light water as coolant and moderator. The number of fuel elements in the core varies for each core loading. As mentioned above, each fuel element in the first core loading (core number 1) has about 38 grams of uranium (about 8.5 weight percent) with 20% U-235 enrichment. However, in the subsequent core loading, fuel elements made of uranium-erbium-zirconium-hydride (UErZrH) alloy are added. Each newly added fuel element has about 98 grams of uranium (about 20 weight percent) with 20% U-235 enrichment.

![Figure 1. Perspective view of the TRR-1/M1 reactor.](image1)

![Figure 2. Cross-section view of the TRR-1/M1 reactor core.](image2)

2.2. Modelling of TRR-1/M1

The reactor was modeled in 3D with non-repeated structures in order to separate burnup calculation for each fuel element. The core number 1 model consisted of 97 fuel elements, four fuel follower control rods, a transient control rod, a central thimble irradiation tube, and three neutron detector tubes. The fuel elements were explicitly modeled non-homogeneously with the sections of fuel (UZrH), zirconium rod through the center of active fuel section, top and bottom graphite reflectors, molybdenum disk, top and bottom stainless steel end fittings, and stainless steel cladding. The control rods were modeled with the sections of boron carbide, fuel follower with zirconium rod through the center of active fuel section and void region. In the model, the central thimble was filled with water, and the three neutron detector tubes were void.
In the model, the UZrH density was intentionally varied in each fuel element in order to reflect the actual fuel measurement data from the manufacturer. All the geometric and material data were taken from the safety analysis report (SAR) prepared by Thailand Institute of Nuclear Technology in 2012 [2]. The physical properties of fuel and fuel follower elements are shown in Table 1.

| Physical property                      | Fuel element | Fuel follower control rod |
|----------------------------------------|--------------|---------------------------|
| Uranium content (wt%)                  | 8.5, 20      | 8.5, 20                   |
| Hydrogen/zirconium ratio               | 1.6          | 1.6                       |
| Erbium content (wt%)                   | 0.0, 0.5     | 0.0, 0.5                  |
| Enrichment of U235 (wt%)               | 20           | 20                        |
| Outer diameter (cm)                    | 3.734        | 3.429                     |
| Fuel diameter (cm)                     | 3.632        | 3.327                     |
| Fuel length (cm)                       | 38.1         | 38.1                      |
| Diameter of zirconium rod (cm)         | 0.635        | 0.635                     |
| Cladding thickness (cm)                | 0.051        | 0.051                     |
| Absorber diameter (cm)                 | -            | 3.327                     |
| Absorber length (cm)                   | -            | 38.1                      |

In the model, the burnup calculation for the core number 1 was based on the reactor operation on weekdays at 1 MW, the average reactor power at steady state. Also in the model, the reactor was assumed to be operated continuously 24 hours for 5 days (1 MWD for each operating day) and to be shut down for 2 days for each week. Since the control rod position could not be adjusted real time in the MCNPX model, the control rod was simply fixed at 29.53 cm out of 38.1 cm (77.5% of fully withdrawn position), which was the maximum position used in the actual reactor operation for core number 1, taken from Thai research reactor (TRR-1/M1) operation log book volumes 25-27 (1977-1979). The end of cycle for the core number 1 had the accumulated burnup of 61.22 MWD before shutting down for maintenance and fuel reshuffling for the core number 2.

In the model using MCNPX, the average UZrH fuel temperature was assumed to be at 600 K and the ENDF/B-VI.2 cross section libraries were used. The temperature limitation in MCNPX data library for the structural materials such as graphite reflector and stainless steel are only available at 293.6 K, 600 K, 900 K, 1200 K, and 2500 K. The structural materials were then assumed to be at 293.6 K because the temperatures of those structures must be below the fuel temperature. Zirconium and hydrogen in the ZrH fuel were assumed to be at 600 K and the S(α,β) thermal scattering cross section in the ENDF5 libraries was used. However, hydrogen in H2O and carbon in graphite were assumed to be at 300 K and the same ENDF5 cross section libraries were used. In the model using MCNPX, the calculations were performed to find effective multiplication factor (k<sub>eff</sub>), radial power peaking distribution, axial neutron flux in central thimble, axial neutron flux in neutron detector tubes, uranium depletion for each fuel elements and fission products. The axial and radial views of MCNPX model of the TRR-1/M1 of TINT are shown in Figure 3 and Figure 4 respectively.

The computing tool used in this calculation is a personal computer with Intel® core™ 2 Quad CPU Q8300 2.50 GHz, 4GB RAM under WINDOWS server® 2008 enterprise operating system. The calculation was performed with 115 cycles on a nominal source size of 10,000 particles per cycle. To avoid data fluctuations during the initial cycles, the first 15 cycles were skipped.
3. Result and discussion

3.1. Reactivity and material concentration

The core excess reactivity at the beginning of cycle (BOC) using MCNPX is calculated to be $6.79 (k_{\text{eff}} = 1.04989 \pm 0.00068)$. The comparisons for the core excess reactivity at the beginning of cycle (BOC) are shown in Table 2. Table 2 also compares the result from an unpublished work using SCALE [3]. After a few days, a sharp reactivity loss of $4.31 (k_{\text{eff}} = 1.03111 \pm 0.00075)$ appears due to the buildup of neutron poison, mainly xenon-135. Xenon-135 reaches equilibrium after seven operating days.

| Table 2. The core excess reactivity at the beginning of cycle (BOC) |
|---------------------------------------------------------------|
| | Actual data | MCNPX | SCALE |
| Reactivity ($\sigma$) | $7.43$ | $6.79$ | $7.17$ |
| $k_{\text{eff}}$ | 1.05486 | 1.04989 | 1.05287 |
| $\sigma$ | - | 0.00068 | 0.00061 |

At the end of cycle (EOC) with the burnup of 61.22 MWD, the core excess reactivity is calculated to be about $2.50 (k_{\text{eff}} = 1.01781 \pm 0.00074)$. The core excess reactivity as a function of operating time is shown in Figure 5 and the Xenon buildup as a function of operating time is shown in Figure 6. From the BOC to the EOC, uranium-235 is burned while Plutonium-239 is steadily produced by capture reaction on uranium-238. The uranium, plutonium, caesium and samarium concentrations as a function of operating time are presented in Figure 7 to Figure 10, respectively.
Figure 5. Core excess reactivity as a function of operating time.

Figure 6. Xenon build-up as a function of operating time.

Figure 7. Uranium concentration as a function of operating time.

Figure 8. Plutonium concentration as a function of operating time.

Figure 9. Caesium concentration as a function of operating time.

Figure 10. Samarium concentration as a function of operating time.
3.2. Power distribution and U-235 depletion

For both the beginning of cycle (BOC) and end of cycle (EOC), the maximum power factors of fuel are found to be in the B ring due to higher thermalization of neutrons in central region of the reactor core. At the BOC, the maximum power factor is 1.70 at the B2 position and the minimum is 0.54 at the G30 position. At the EOC, the maximum power factor is 1.66 at the B5 position and the minimum is 0.55 at the G30 position, which is the same position as in the BOC. The difference between the maximum and minimum power factor positions is about 70% for both the BOC and the EOC. The maximum power factors of the EOC and BOC differ by about 2%. The radial power factor distribution of fuel and fuel follower elements in the reactor core at the BOC is shown in Figure 11.

The individual %U-235 depletion of fuel and fuel flower elements in the reactor core at EOC (61.22 MWD) is shown in Figure 12. The maximum fuel depletion of around 3.27% (1.24 grams of U-235) occurs in the B ring. From the centre to the rim of the reactor core, the U-235 depletion decreases and reaches the minimum value of 1.07% (0.41 grams of U-235) at the G30 position. The average burnup of the core is about 2.00% or roughly 76 grams of U-235.

![Figure 11. Radial power factor distribution at BOC of reactor core](image1)

![Figure 12. %U-235 depletion of fuel and fuel flower elements at EOC of reactor core](image2)

3.3. Neutron flux

The axial neutron fluxes distributions in the reactor core at the central thimble (CT) and at the three neutron detector tubes (at the E06, E08, and E19 positions) are shown in Figure 13 and Figure 14, respectively.

For the central thimble (CT) at the A1 position of the reactor core, the neutron flux distributions at the BOC and EOC are in parabolic shape. The maximum neutron fluxes at both the BOC and EOC occur at the middle of the CT due to high neutron density in the active region of the reactor core. At the middle of the CT, neutron fluxes at the BOC are little higher than neutron fluxes at the EOC by about 2% to 5%. Thermal neutron fluxes at the BOC and EOC are 7.01x10^{12} n/cm^2-s and 6.86x10^{12} n/cm^2-s, respectively. Epithermal neutron fluxes at the BOC and EOC are 3.09x10^{13} n/cm^2-s and 2.99x10^{13} n/cm^2-s, respectively. And fast neutron fluxes at the BOC and EOC are 1.12x10^{13} n/cm^2-s and 1.06x10^{13} n/cm^2-s, respectively.

The neutron fluxes of the three neutron detector tubes (ND) at the E06, E08, and E19 positions of the reactor core represent the total energy range (0.0-15.0 MeV) unlike the neutron fluxes at the CT, which represent the thermal, epithermal and fast neutron energy ranges. This is due to the nature of a
neutron detector that measures neutrons for the whole energy range. The neutron flux distributions of neutron detector tubes at the BOC and EOC are also in similar parabolic shape as the neutron flux distributions of the CT. The maximum neutron flux at the EOC is nearly the same as the maximum neutron flux at the BOC, differing by about 1%. The average neutron flux at the E19 position appears higher than the average neutron fluxes at the E06 and E08 positions by about 12%. This is mainly because the E19 position is surrounded by more fuel elements. At the EOC, the maximum neutron fluxes at the E06, E08 and E19 positions are $4.09 \times 10^{13}$, $4.08 \times 10^{13}$, $4.60 \times 10^{13}$ n/cm$^2$-s, respectively.

![Figure 13](image1.png)  ![Figure 14](image2.png)

**Figure 13.** Thermal, epithermal and fast neutron fluxes distribution at central thimble (CT)  
**Figure 14.** Total neutron fluxes distribution at neutron detector tubes (ND)

4. Conclusion
This paper presents a preliminary analysis of fuel burnup of the TRR-1/M1 Thai research reactor. The analysis covers the calculation of core excess reactivity, neutron flux, power distribution and fission products. The results show that even though reactivity decreases greatly at the EOC, the power and neutron flux distributions at the EOC are less than at the BOC by only about 2%. Also, the neutron flux distributions of the CT and ND at the BOC and EOC differ by only about 1-5%. This means the reactor core behaves almost the same throughout the operation from the BOC to the EOC. The analysis results such as depletion of uranium and production of fission products will be considered and served as the input for the burnup analysis of the next core loading. In addition, the results from this paper will be benchmarked in the future with the actual experiment results or the calculation results from other calculation software.

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