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Fuel burnup analysis and fuel management for TRIGA Mark III research reactor

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Abstract. The fundamental advantage in using Monte Carlo methods for burnup calculations is to formulate an effective optimal fuel management strategy for the TRR-1/M1 research reactor. The core management study has been performed by utilizing the essentially parameters including multiplication factor, power peaking, neutron flux and burnup calculation based on the Monte Carlo calculation. The fuel element burnup was calculated after reshuffling the reactor core. The fuel cycle length and core parameters such as core excess reactivity, neutron flux, axial and radial power factors and other parameters are determined. The core excess reactivity was calculated as a function of burnup. The maximum excess reactivity shall not exceed 6.3% Δk/k. The maximum fuel temperature shall not exceed 930 °C during steady-state operation. Typically, a core loading operated with the maximum burnup between 100 to 200 MWD depending on the utilization requirements. The thermal neutron flux in the irradiation positions is within the order of 10^{11} - 10^{13} n/cm^2-sec. The study gives valuable results into the behaviour of the TRR-1/M1 research reactor and will ensure optimized utilization and operation of the reactor during its life time. It will establish the strategic planning for fuel management in the reshuffling and reloading schemes patterns and its safe implementation in the future.

1. Introduction

The 2 MW TRIGA MARK III research reactor referring as TRR-1/M1 was commissioned at the Thailand Institute of Nuclear Technology. The reactor was designed to implement several fields of basic nuclear research, gem stone irradiation, educational and training and production of radioisotopes for its uses in agriculture, industry and medicine. The reactor is a light water cooled, designed for operation at a steady-state power level of 1.2 MW. An outstanding feature of the TRIGA reactor is its inherent safety feature resulting from the large prompt negative temperature coefficient of reactivity of its UZrH fuel-moderator material. The objective of the study is to determine an optimal fuel management strategy for most effective utilization using burnup calculation from fuel elements loaded in the initially in the TRIGA core. The basic objectives of reactor core management are to achieve in high fuel utilization and safe and efficient rated power operation. This requires strategic planning in fuel reshuffling and reloading fuel pattern with the safe implementation in the operational limits and conditions. Presently, it is important to know the individual fuel burnup for the reshuffling and design core pattern of the TRR-1/M1 reactor to ensure optimum utilization of fuel. Therefore, it can contribute to safe and economic use of the TRR-1/M1 reactor. The core management and burnup study has been performed by utilizing four basic types of information: criticality (k_{eff}), power peaking, neutron flux and

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power distributions and fuel element burnup calculations. These involve the relationship of core as a function of burnup. The fuel depletion and burnup calculations using MCNPX were used for core management. Several researches studied the fuel burnup calculation using different computer codes such as MCNPX [1], WIMS-D5 [2], TRIGAP2 [3], MONTEBURNs [4], and BUCAL1 [5]. They had performed burnup analysis for several reactor core benchmark and many reactor parameters had been calculated. Various patterns for distribution of fuel rods have been considered in order to achieve high fuel burnup and reasonably flat power distribution according to the safety limit in operation of the reactor. The MCNPX computer code has been used to determine of individual fuel element burnup, calculate of core lifetime and formulate of optimal fuel management strategy.

2. Computational Methods

2.1 MCNPX 2.6.0 Computer Code

The TRR-1/M1core consists of fuel elements arranged in a concentric hexagonal array within the core shroud. Elements are arranged in seven concentric rings and the spaces between the rods are filled with water. The lattice was modeled as a hexagonal prism, solids with eight faces (Figure 1). The fuel elements were modeled with homogeneous mixture of uranium-zirconium hydride alloy with the uranium-to-zirconium atom ratio of 1.6 to 1.7. The uranium in the uranium-zirconium hydride mixture is enriched to approximately 20% U-235. There are two types of fuel elements loaded in TRR-1/M1 core including 8.5% uranium by weight type and 20% uranium by weight type. The 20 wt% fuel element is a mixture of uranium-erbium-zirconium-hydride (UErZrH) alloy containing approximately 0.5 wt% erbium. The core configuration in Figure 1 consists of 20 wt% fuel elements and 8.5 wt% fuel elements including FFCRs, 3 locations of neutron detectors (position) and 10 locations of in-core irradiation facilities (including pneumatic transfer system) for in-core utilization. The burnup calculations were performed using MCNPX which is a general purpose Monte Carlo radiation transport code designed to track many particle types over broad ranges of energies and comparing with MVP-BURN computer program which is continuous energy and multigroup Monte Carlo method developed by Japan Atomic Energy Agency (JAEA).

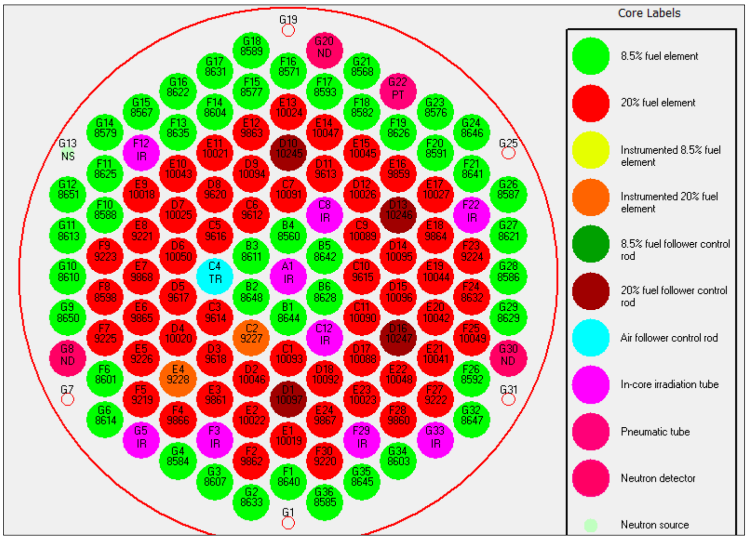

Figure 1. Fuels and FFCRs with 20 wt%. and 8.5wt%. fuel elements.

The fuel element is approximately 3.73 cm in diameter and 73.15 cm in overall length and the active part of the fuel element is 38.1 cm long. The power level of the reactor is controlled with five control rods: The TRR-1/M1 uses five control rods; a safety rod, a regulating rod, two shim rods and a safety-transient rod. The regulating, shim, and safety rods are sealed 304 stainless steel tubes approximately

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109 cm long by 3.43 cm in diameter in which the uppermost 16.5 cm section is an air void and the next 38.1 cm is the neutron absorber (boron carbide in solid form). The control rods were explicitly modeled along the active length containing three vertical sections of boron carbide, fuel follower, and void region. The central thimble was considered to be filled with water and the pneumatic tube was considered to be void. The fuel elements figures are presented in Figure 2a and 2b.

(a) (b)

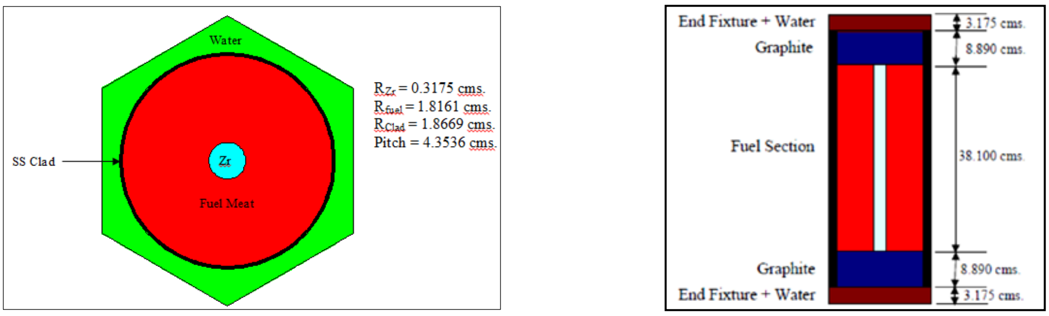

Figure 2. Fuel element (horizontal Cross Section (a) and vertical cross section (b)).

2.2 Burnup Calculation

MCNPX 2.6.0 [6] includes many new capabilities, particularly in the areas of transmutation, burnup [7] and delayed particle production [8]. CINDER data library is required for burnup and depletion calculations [9]. Neutron transmutation, fission, and radioactive decay are included in the modeling of the production and removal terms for each isotope of interest. For a fueled region, neutron transmutation, fuel depletion, fission-product poisoning, actinide generation, burnable poison loading and depletion effects are included in the calculation. The MCNPX 2.6.0 uses the fourth order Rung Kutta method [10] with the predictor–corrector approach for the resolution of the depletion equation. Using MCNPX can calculate for standard burnup calculation, burnup calculation followed by a space of time of cooling, burnup calculation with shuffling of fueled regions and burnup calculation with reloading new fresh fuels. Two groups of nuclides under consideration are: actinides that contain heavy metal nuclides and their decay daughters; and fission products produced by fissions and their decay/capture daughters.

3. Determination of Core Lifetime

The burnup throughout an expected life cycle of the reference core is analyzed. The analysis was performed by MCNPX which uses predictor-corrector burnup methodology. For this analysis, each fuel element was designated as a burnup zone to accurately representing the variation in burnup rate for each fuel element. The burnup calculation was done at the full power of 1.2 MW. The core excess reactivity was calculated as a function of burnup. The core was burnt without changing the loading pattern. Typically, a core loading would be operated with the maximum burnup of between 100 MWD and 200 MWD depending on the utilization requirements and other appropriate schedules. In the analysis, the core is operated in one cycle of 200 MWD. The core excess reactivity of the core loading as a function of burnup is shown in Figure 3. As it was calculated by MCNPX, the begin of cycle (BOC) excess reactivity is 7.07% Δk/k and it drops to 5.12% Δk/k shortly after the burnup. This sharp core excess reactivity drop is caused by the buildup of ^{133}Xe and ^{149}Sm in the fuel and the reactivity loss due to heating of fuel. It is also observed that the burnup rate of the first 100 MWD is higher than that of the second half.

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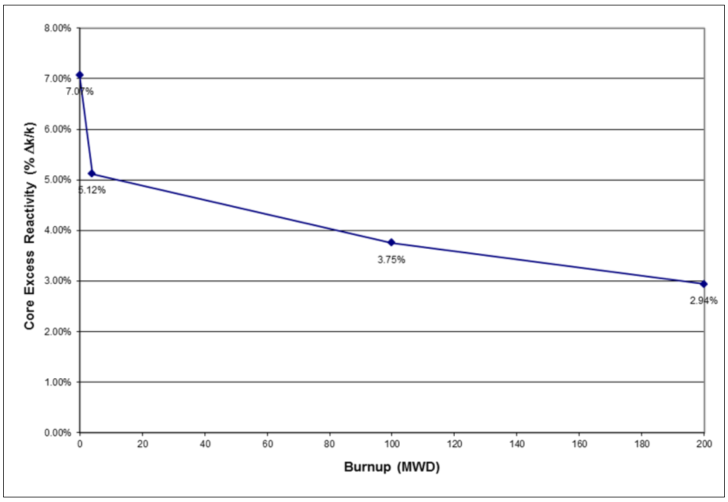

Figure 3. Core excess reactivity in relation to burnup (MWD) at Xe and Sm equilibrium.

At the initial burnup time a sharp loss of reactivity because of the build-up of ^{133}Xe and ^{149}Sm. The concentrations of ^{133}Xe and ^{149}Sm fission products poisons influence the reactivity and eventually reach equilibrium for ^{133}Xe and ^{149}Sm as presented in Figure 4. The analysis calculated the Xenon reactivity feedback of 1.95% Δk/k at full power. After Xenon equilibrium, the reactivity loss of the TRR-1/M1 is found to be 2.18% Δk/k.

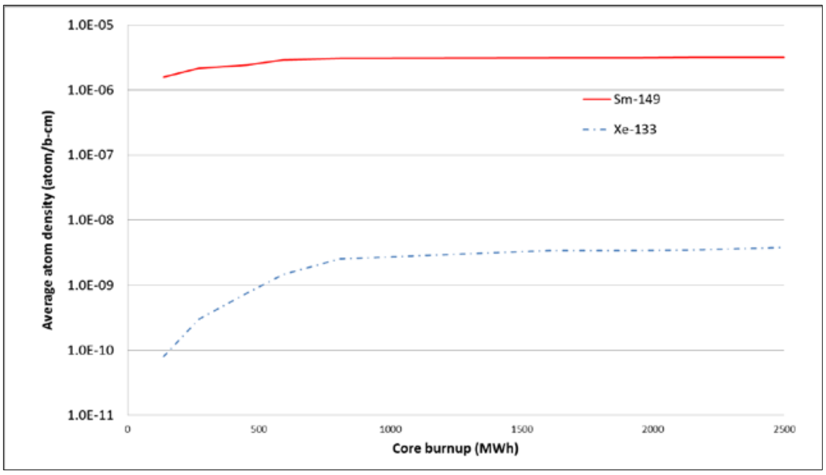

Figure 4. ^{133}Xe and ^{149}Sm buildup as a function of core burnup.

The relative radial power factor of the fuel and fuel follower elements are presented in Figure 5. The calculations are performed from BOC to end of cycle (EOC). Radially, it seems that both BOC and EOC distributions, the maximum power factors are found in B ring because of the relatively higher thermalization of neutrons in the central region of the core. At BOC, for each ring, the power factors have a cosines shape that peaks at the fuel elements further than the control rods, such as for B ring the maximum is found to be 2.03 at B6 and for C ring the maximum is found to be 1.84 at C9. The amplitude of the shape is higher in rings B to D due to the influence of the control rods which are withdrawn with 25.00 cm. At EOC, the power factors values of rings B to D decrease due to high burnup values at this region of the core and the cosines shape disappear partially resulting from the new control rods positions that became roughly fully withdrawn approximately 28.58 cm. EOC/BOC ratios show that, the power factors of fuel elements nearest to CRs become higher than in BOC due to the neutron flux change while withdrawing the CRs.

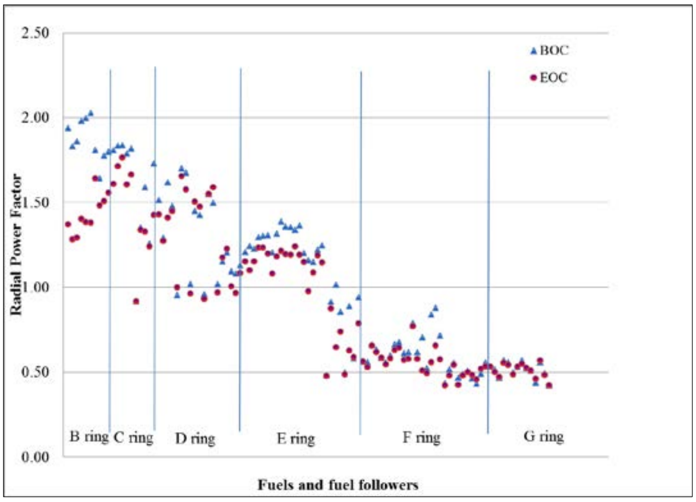

Figure 5. Radial power factors distribution within the fuel and fuel follower elements of TRR-1/M1.

Figure 6 and 7 present the axial thermal fluxes versus burnup at the central thimble (CT) and G33 irradiation positions of TRR-1/M1 reactor, respectively. For CT, it shown that the thermal flux is flattening as a function of burnup resulting in a decrease of the peak value between BOC and EOC, because of the high burnup values at the core center and the CRs positions. The peak value shift upward from 34 cm to 30 cm above the core center line. The calculation shown that the thermal neutron flux peaks at the central thimble and decrease in D ring and continue to fall up to F and G ring.

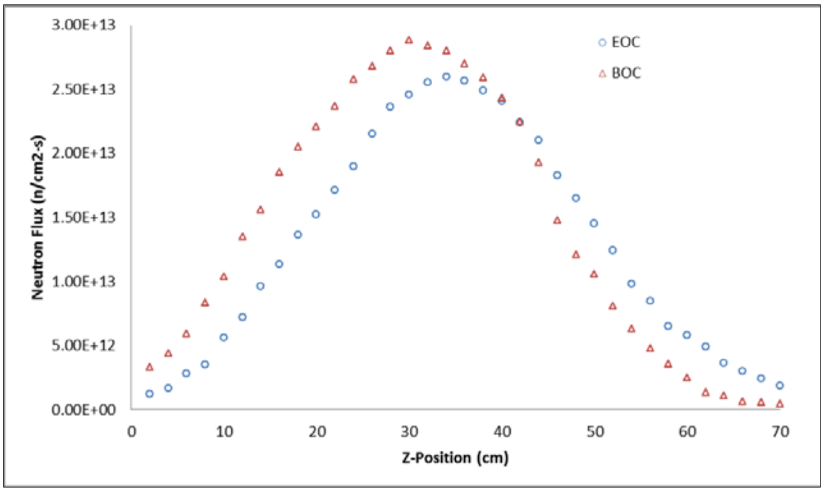

Figure 6. Thermal neutron flux at the BOC and EOC at CT position of TRR-1/M1.

For G33 irradiation position, the thermal flux is flattening with increasing of the peak value at EOC. This is because of more of the fissile material (^{235}U) is depleted at the inner region of the core (B and C rings), more power must be supplied by the fuel elements located at the core periphery since the total power is the same (1.2 MW), therefore, a small increase of the thermal flux at the outer part of the core as shown in Figure 7.

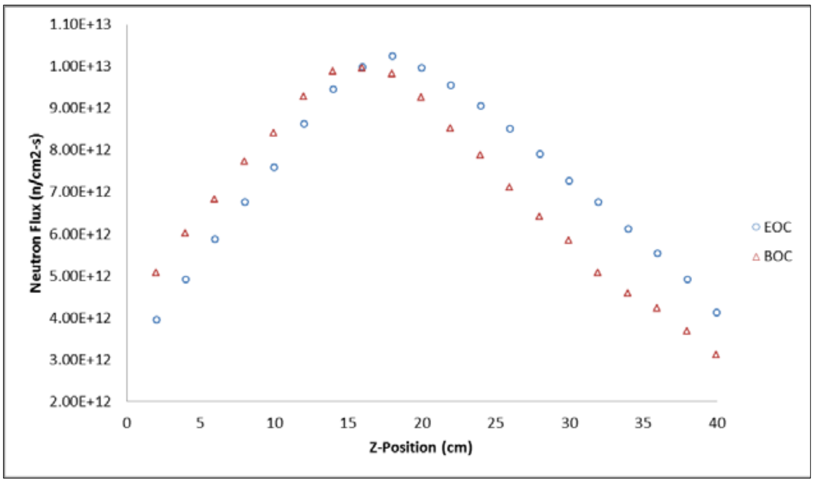

Figure 7. Thermal neutron flux at the BOC and EOC at G33 irradiation position of TRR-1/M1.

4. Comparison between MCNPX and MVP

The individual fuel burnup for selected standard fuel element (fuel number 8558) was compared by the MCNPX and MVP code. The cross section library for MVP calculation was derived from JENDL3.3 [11]. The calculated individual burnup (% ^{235}U burn) at the end of core in different core number during the years of 1977–2018 (total burnup was 2211.40 MWD) is shown in Figure 8. A good agreement was less than ±8% difference which was observed between the MCNPX and MVP. It provides the confidence for MCNPX and MVP calculations.

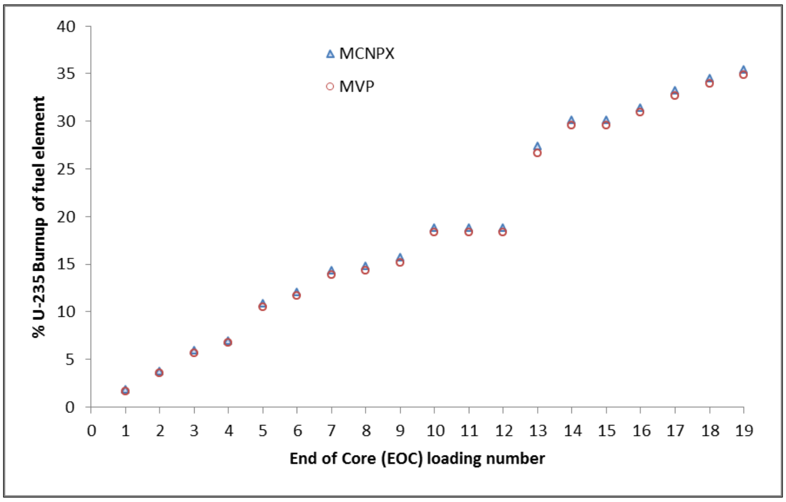

Figure 8. Comparison of the individual TRIGA fuel burnup (%^{235}U) calculated by MCNPX and MVP at EOC.

5. Conclusion

The burnup calculation of TRR-1/M1 fuel elements was performed using MCNPX to study the burnup dependent neutronic parameters for the 1.2 MW TRR-1/M1 reactor. The fuel management strategy was planned according to the burnup calculation. The plan was that the fuel elements were shuffled every year after 10-15% burnup of initial ^{235}U which will increase the core life time after fuel rearranged at a certain period of operation. Particularly the highest burnt elements were replaced by relatively least burnt elements in the core periphery. The optimal in core fuel management strategy of TRR-1/M1 will contribute the safe operation and better utilization of TRR-1/M1 fuel. The specific parameters included excess reactivity, spatial flux, k_{eff}, Xe and Sm equilibrium and radial power density distribution. The radial power distribution is strongly affected by burnup due the CR's positions to compensate the loss of reactivity. Analysis of the thermal neutron fluxes at CT and G33 irradiation positions versus burnup shows that the thermal flux in CT position can be decrease due to burnup and it can be increase slightly in G33 due to the more power which supplied by fuel elements in the core periphery in order to reach the full power at the end of life. The results from the MCNPX and MVP computer code [12. 13] were found to be in good agreement; therefore, the simulation model can be used as reference with confidence for TRR-1/M1 core configuration.

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