Calculation of neutronic parameter design for MTR reactor core using safety control rods

T Surbakti*, S Pinem, L Suparliana and F Yogatama

Center for Nuclear Reactor Technology and Safety, BATAN, Kawasan PUSPIPTEK Gd. No. 80 Serpong, Tangerang Selatan, 15310 Indonesia

*tukiran@batan.go.id

Abstract. Indonesia has three research reactors that are old, so it needs to be considered designing new research reactor. This new research reactor uses compact core, MTR type, low enrichment of 235U and a new fuel, U7Mo/Al. The advantage of using this new fuel is that it can be fabricated with high density up to 16 gU/cm3. However, on this research a density of uranium was varied from 4.5 -8.3 gU/cm3. The reactor core comprises 16 fuels and 4 control rods inserted into a 5 x 5 grids. Inside the core there are irradiation facilities comprising 4 holes at the edge and 1 in the middle of the core for radioisotope production. The reactor core is also equipped with 2 safety control rods to increase the safety of the reactor operation for one cycle. This research gets an optimal core configuration with maximum radioisotope production. Analysis was performed by calculating using the WIMSD-5B and Batan-FUEL computer programs. Optimal core design a cycle length of not less than 20 days and the reactor can be operated safely with a power of 50 MW producing a thermal neutron flux greater than 1.0x1015 n/cm2s. The calculation results show that for fuel loading a density of 6.52 gU /cm3 will obtain a thermal neutron flux of 1.03 x1015 n/cm2s. The cycle length can reach 32 days. It meets safety and acceptance criteria; however, it still requires a more negative reactivity of the control rod for safe operation at 50 MW compared to other densities.

1. Introduction

One of the main tasks of BATAN is to study the technology of nuclear reactors in the world. The technology of nuclear reactors, research reactors have been proven to welfare people by producing radioisotopes. Nuclear reactor technology continues to develop from generation to a new generation. Research reactor technology in the world has examined the possibility of using uranium molybdenum fuel. South Korea is building the KJJR research reactor that using high-density uranium molybdenum fuel [1]. The advantage of using this fuel is that it can be fabricated with high density up to 16 gU/cm3 [2]. Batan has planned to design a new research reactor to support the increasing radioisotope production program in Indonesia. The research reactor to be designed is a compact core, using U7Mo/Al high-density uranium molybdenum fuel. It has carried some conceptual design calculations out starting from the configuration of the core, the height of the core and the amount of fuel and reactor power. At present the MTR research reactor designed by Batan is using compact core with 16 fuels (FE), 4 control rods (CE), and 2 safety control rods outside the core. The type of fuel used is U7Mo / Al with densities varying from 4.3 gU/cm3 - 8.3 gU/cm3 [3]. The aim of this research is to obtain an optimal reactor core configuration without violation safety limits. Optimal core configurations must meet the acceptance criteria of a thermal neutron flux greater than 1x1015 n/cm2s and a minimum cycle length of 20 days at
50 MW power [4]. Analysis of the neutronic parameters calculation of the MTR reactor using 2 safety control rods was carried out using the WIMSD-5B computer program with nuclear data from ENDFBVII.0 [5] and the core calculation using the Batan-FUEL [6,7] computer program. Conceptual core design of the MTR type reactor with addition of 2 safety control rods outside of the core will improve the safety.

2. Fuel description
High-density MTR research reactor fuel is a plate type which is the same size as RSG-GAS core fuel. One fuel assembly consists of 21 plates containing uranium molybdenum, while the control rod assembly consists of 15 plates in which 3 sides of the left and right plates are used for absorbent plates with AgInCd material. 16 fuel assemblies and 4 control rod assemblies are put into the reactor core with a 5 x 5 grids. The type of fuel material used on the MTR core in this study is U7Mo / Al with variations in density as follows: 1. 450 g \(^{235}\text{U}\) or the uranium density of 4.57 gU/ cm\(^3\); 2. 550 g \(^{235}\text{U}\) or the uranium density of 6.52 gU / cm\(^3\); 3. 700 g \(^{235}\text{U}\) or the meat uranium density of 8.30 gU / cm\(^3\).

3. Methodology
To analyze the core design of the MTR reactor, the neutronic parameter calculation is done by means of 2 parts calculation. First, the calculation of core material cells using the WIMSD program. The goal is to obtain the macroscopic constant needed for core calculations. The WIMSD (Winfrith Improved Multi-group Scheme) program package is a cell calculation. This program package was developed initially by AEE Winfrith. Cell units that can be handled by WIMSD [8] basically consist of 3 or 4 regions, namely fuel (1), can or cladding (2), coolant (3) and moderator (4) which are represented in slab. Then, the second step is to calculate the core parameter using the diffusion method with the Batan-FUEL program, to get the neutronic parameters of the MTR reactor core.

3.1. Cell calculation
The WIMSD-5B code has been written to provide a comprehensive scheme of reactor lattice cell calculations applicable to a wide range reactor types including both thermal and fast reactors. The output of this code provides detailed reaction rates for the lattice and eigenvalues for cases where a simple buckling mode is applicable; alternatively, average cell constants, corrected for leakage where required, are provided for use in overall reactor calculation. The structure of the neutron energy in the WIMSD-5B code is 10 MeV, 0.821 MeV, 5.531 keV, 0.625 eV and 1 × 10\(^{-5}\) eV [9]. The composition and width representing the cells are defined in the material and spectrum data adjusted to the four regions and this has been done by the WIMSD program package. If the user does not want the material to be weighted by the existing spectrum, then it is given a negative sign so that neglect occurs. This is especially necessary when dealing with strong absorbent cells. By using the shape of the spectrum cell calculations are performed. This concept indicates the WIMD program package uses tricks to get accurate results. Cell calculation results include: flux in many groups (multi-group) for 3 or 4 regions representing cells, k-inf unit cell, and macroscopic cross-sections in few groups for all materials.

3.2. Core calculations
Core calculations are performed to obtain neutronic parameters with the Batan- Fuel program which consists of several programs such as Batan-2DIFF and Batan-3DIFF and Batan- Equil. Flow diagram of neutronic parameter calculation can be seen in Figure 1. The results of the WIMSD-5B calculation in the form of macroscopic constants are used for core calculation to obtain neutronic parameters with core condition of fresh fuel and equilibrium core conditions. Table 1 is the material and size of the fuel and the control rods used in the calculation of WIMSD and Batan-FUEL. For core calculations, the equilibrium core, where after operating one cycle only a part of the fuel is replaced by a new one. It is needed a fuel management calculation to shift the fuel in the core.
Table 1. Fuel material data for MTR reactor core [10].

| Parameter                                | Values                          |
|------------------------------------------|---------------------------------|
| Core grid dimension (cm)                 | $7.71 \times 81 \times 70$     |
| Fuel plate thickness (cm)                | 0.13                            |
| Channel width of Coolant (cm)            | 0.255                           |
| Plates for fuel element number           | 21                              |
| Plates for control element number        | 15                              |
| Material of cladding                     | AlMg2                           |
| Material of edge plate                   | AlMg1                           |
| Fuel cladding thickness (cm)             | 0.038                           |
| Dimension of active zone (meat), cm      | $0.054 \times 6.275 \times 70$  |
| Material of fuel                         | U7Mo-Al                         |
| Uranium fuel loading (gram)              | 360, 390, 450, 550, 700         |
| Material of absorber                     | Ag-In-Cd                        |
| Absorber thickness (cm)                  | 0.338                           |
| Material of absorber cladding           | SS-321                          |
| Absorber cladding thickness (cm)         | 0.085                           |

Figure 1. Flowchart of core calculation for Batan-FUEL code [11].
4. Result and discussion

There are 5 core configurations for equilibrium core namely, core configuration with density of fuel 3.66 gU/cm$^3$; 3.96 gU/cm$^3$; 4.75 gU/cm$^3$; 6.53 gU/cm$^3$ and 8.39 gU/cm$^3$. The calculation results can be seen in Figure 2. The Figure showed that the integral reactivity control rods worth for all core configurations can cover the excess reactivity. The control rod reactivity profile forms an S curve, with the maximum gradient in the middle. The s-shaped of integral reactivity control rod showed that the biggest absorber has in the middle of the control rods, where the highest neutron flux. The curves of the individual control rods are the same shape as the total control rods. It can be seen from the figure that the amount of uranium in the fuel also greatly influences the total reactivity value of the control rod. The results of calculation for equilibrium core can be seen in Table 2. From this Table, it can be seen that the excess reactivity of the core rises according to the amount of uranium in the core. Stuck rod condition for all core configurations have met the safety limit (-0.5 %). The maximum of PPF (power peaking factor) value is still lower than safety limit (1.4). Addition of 2 safety control rods outside the core at a density of 4.57 gU/cm$^3$; 6.52 gU/cm$^3$ and 8.30 gU/cm$^3$ can meet the safety limits and acceptance criteria, i.e. the operating cycle length is more than 21 days.

However, the thermal neutron flux at CIP does not meet the acceptance criteria for the core configuration at a density of 8.30 gU/cm$^3$. The maximum burn-up discharged on the equilibrium core obtained 53.67% where it is still far from the safety limit of 60%. Table 3 shows the results of neutron flux calculations at each irradiation facility (IP). The calculation results state that for core configurations with densities of 4.57 gU/cm$^3$ and 6.52 gU/cm$^3$ can produce thermal neutrons greater than $1.0 \times 10^{15}$ n/cm$^2$s in the central irradiation facility and this meets the acceptance criteria. The greater the amount of uranium in the fuel, the smaller it produces thermal neutron flux but the longer the cycle length of operations. From the results of this analysis the most optimal core configuration is the core which has a density of 6.57 gU/cm$^3$, because the core is the best and meets the safety limits and acceptance criteria.
Table 2. Neutronic parameter for equilibrium core.

| No. | Reactivity balance                       | Mass per fuel |
|-----|------------------------------------------|---------------|
|     |                                          | NO SCR       | 2 SCRs       | 2 SCRs       |
| 1.  | Massa (gram) and class burn-up           | 360          | 450          | 550          | 700          |
|     |                                          | 2/8          | 2/8          | 2/8          | 1/4          |
| 2.  | Power (MW)                               | 50           | 50           | 50           | 50           |
| 3.  | Density (gram/cm³)                       | 3.66         | 3.96         | 4.57         | 6.52         | 8.30         |
| 4.  | One cycle reactivity (%Δk/k)             | 4.58         | 5.41         | 6.25         | 7.50         | 6.70         |
| 5.  | Xenon reactivity (%Δk/k)                 | 3.94         | 3.95         | 3.96         | 3.98         | 4.01         |
| 6.  | Samarium reactivity (%Δk/k)              | 0.31         | 0.32         | 0.34         | 0.38         | 0.30         |
| 7.  | Hot to cold reactivity (%Δk/k)           | 0.43         | 0.42         | 0.40         | 0.38         | 0.39         |
| 8.  | Excess reactivity (%Δk/k)                | 10.72        | 11.32        | 11.33        | 13.54        | 12.54        |
| 9.  | Stuck rod condition (%Δk/k)              | -2.61        | -2.59        | -2.63        | -1.10        | -1.86        |
| 10. | Control rods worth reactivity (%Δk/k)    | -26.41       | -25.60       | -28.50       | -26.31       | -24.87       |
| 11. | Shutdown margin reactivity (%Δk/k)       | -15.69       | -14.28       | -16.17       | -12.78       | -12.33       |
| 12. | Cycle length (days)                      | 11           | 15           | 21           | 32           | 24           |
| 13. | Maximum burn-up (%)                      | 29.91        | 37.05        | 44.20        | 53.67        | 44.41        |
| 14. | Average radial PPF                       | 1.22         | 1.22         | 1.24         | 1.27         | 1.19         |
| 15. | Stuck rod condition (%)                  | -3.91        | -2.92        | -3.85        | -1.59        | -2.07        |
| 16. | Average power density (W/cc)             | 635          | 635          | 635          | 635          | 635          |

Table 3. Neutron fluxes at the IPs for equilibrium core.

| Group of neutron energy       | Max. neutron flux $10^{15}$ (neutron/cm² s⁻¹) |
|-------------------------------|-----------------------------------------------|
|                               | 450 g       | 550 g       | 700 g       |
| Fast neutron $> 0.821$ MeV    | 0.283       | 0.279       | 0.267       |
| Epithermal neutron, $0.625$ eV $< E < 0.821$ MeV | 0.560       | 0.551       | 0.533       |
| Thermal neutron, $< 0.625$ eV  | 1.072       | 1.029       | 0.986       |
|                               | CIP          | IP          |
| Fast neutron $> 0.821$ MeV    | 0.125       | 0.124       | 0.124       |
| Epithermal neutron, $0.625$ eV $< E < 0.821$ MeV | 0.302       | 0.298       | 0.299       |
| Thermal neutron, $< 0.625$ eV  | 0.579       | 0.553       | 0.541       |

Figure 3. Thermal neutron flux for equilibrium core.
Distribution of thermal neutron flux at CIP (center for Irradiation Position) can be seen in Figure 3. This Figure shows the profile of the thermal neutron flux for the equilibrium core. Thermal neutron is the highest in CIP because in this irradiation facility the most cooling water is found.

5. Conclusion
For the calculation of equilibrium core, it can be concluded that for the most optimal core configuration is the core configuration with a density of 6.52 gU/cm$^3$. All safety parameters are not violated and the acceptance criteria can also be met and the maximum fuel burn-up is 53.67%. This maximum fuel burn-up is close to the maximum RSG-GAS core burn-up of 59.56%.

Acknowledgements
We thank the Head of PTKRN and the Head of BFTR for all their input on this research so that it can be completed on time. We also thank the INSINAS 2019 Menristekdikti for funding this research.

References
[1] Yim J S et al 2017 On-going Status of KJRR Fuel (U7Mo) Qualification INL/Con-17-4124
[2] Chusova I A, Shelegova A S, and Kochnovb O Y 2016 Design Features of Water-cooled Research Reactors Nuclear Energy and Technology 2 4 287–293
[3] Guohai Wei, Songbai Han, Hongli Wang, Lijie Hao, Meimei Wu, Linfeng He, Yu Wang, Yuntao Liu, Kai Sun, Dongfeng Chen 2013 Design of the testing set-up for a nuclear fuel rod by neutron radiography at CARR Physics Procedia 43 307 – 313
[4] Raheleh K 2014 Sensitivity Analyses of the Use of Different Neutron Absorbers on the Main Safety Core Parameters in MTR Type Research Reactor Nucl. Eng. Technology 46 4 513–520
[5] Smith D L 2011 Evaluated Nuclear Data Covariances: The Journey From ENDF/B-VII.0 to ENDF/B-VII.1 Nucl. Data Sheets 112 12 3037-3057
[6] Pinem S, Sembiring TM, Surbakti T 2018 Core conversion design study of TRIGA Mark 2000 Bandung using MTR plate-type fuel element International Journal of Nuclear Energy Science and Technology 12 3 222-238
[7] Surbakti T, Pinem S, Sembiring T M, Hamzah A, Nabeshima K 2019 Calculation of Control Rods Reactivity Worth of RSG-GAS First Core Using Deterministic and Monte Carlo Methods Atom Indonesia 45 2 69-79
[8] Hastutti E P, Surbakti T, Widodo S, Sudarmono 2018 Abnormal control rod withdrawal analysis for innovative research reactor using PARET-ANL codes Kerntechnik 83 2 96-105
[9] Surbakti T, Pinem S, and Suparlina L 2018 Dynamic Analysis on the Safety Criteria of the Conceptual Core Design in MTR-type Research Reactor. Journal Atom Indonesia 44 2 89-98
[10] Liem P H, Surbakti T and Donny S 2018 Kinetics parameters evaluation on the first core of the RSG GAS (MPR-30) using continuous energy Monte Carlo method. Progress in Nuclear Energy 109 196-203
[11] Pinem S, Liem P H, Tagor M S and Surbakti T 2016 Fuel element burnup measurements for the equilibrium LEU silicide RSG GAS (MPR-30) core under a new fuel management strategy. Annals of Nuclear Energy 98 211-217