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DESIGN OF NEUTRONIC PARAMETERS OF MTR REACTOR USING WIMSD-5B/BATAN-FUEL CODES

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ABSTRACT

BATAN has three aging research reactors, so it is necessary to design a new, more modern MTR type reactor using high-density, low enrichment uranium molybdenum fuel. The thermal neutron flux at the irradiation position is an important concern in the design of research reactors. This analysis is performed using standard computer codes WIMSD-5B and Batan-FUEL. The purpose of this study is to analyze the effect of the core configuration with safety control rods and neutronic parameters using the diffusion method calculation. The reactor core consists of 16 fuel elements and four control rods placed in the 5 x 5 position of the grid plate and is loaded the reflector elements outside the core. The cycle length is also a concern, not less than 20 days, and the reactor can be operated safely with a power of 50 MW. The calculation results show that for the highest fuel loading, which is 450 grams of U7Mo/Al fuel with D2O as a reflector, it will provide the lowest thermal neutron flux at the center of the core irradiation position, namely 1.0 x1015 n/cm2s. The core fuel cycle length will be up to 39 days, meeting the expected acceptance and safety criteria.

Keywords: thermal neutron flux, a research reactor, U7Mo/Al fuel, WIMSD-5B, Batan-FUEL

INTRODUCTION

Nuclear technology is very beneficial for humans. One of the uses of nuclear technology is in a research reactor. Research reactors are used for the production of radioisotopes for health, agricultural, industrial, and research purposes. Indonesia has three research reactors, two research reactors are the TRIGA types, and one is the MTR (Material Testing Reactor) type, RSG-GAS. The TRIGA type reactor also no longer produces its fuel, while the RSG-GAS research reactor uses uranium silicide fuel. In the future, uranium silicide fuels will no longer be used because of the very limited density that can be contained in a fuel assembly [1]. The world's research reactors will switch to a new fuel, namely higher density uranium molybdenum (U7Mo/Al).

RSG-GAS also uses a wide core, and modern research reactors tend to use a compact core for the purpose of obtaining high thermal flux neutron to irradiate radioisotope, which requires less time. To anticipate these problems, it is necessary to design a new research reactor core, in which the fuel, core configuration, and facilities adjust to future needs. To design a new research reactor, there are several steps that need to be done, such as studying the function, purpose, and benefits of the reactor, as well as conceptual design, basic design, and detailed design.

The initial conceptual design of the Material Testing Reactor (MTR) from the neutronic aspect has been carried out by T. Surbakti et al., [2] Where the main characteristics of the core configuration are as defined as follows: The RRI research reactor has a nominal power of 20 MW, uses a uranium-molybdenum alloy, U9Mo/Al fuel with a geometry that adopts RSG-GAS fuel. The 5 × 5 core grid configuration consisting of 20 fuel elements and 5 irradiation positions is capable of producing a thermal neutron flux in the amount of 2.87×1014 n/cm^{2}s. However, this neutron flux value is lower than stated as the criteria for acceptance of an MTR type reactor where the reactor must have a minimal thermal neutron flux in the irradiation position and in the reflector region at least 1.0 × 1015 and 0.4 × 1015 n/cm^{2}s [3].

The second conceptual design was proposed by I. Kuntoro et al., [4], where the core grid is the same, but the fuel level and reactor power are different. To meet the acceptance criteria, the core is designed with a fuel element with a height of 70 cm, a power level of 50 MW and other sizes the same as the previously designed reactor grids. The core configuration is determined by the number and position of the fuel element and the position of the irradiation. After finding the optimal core configuration, it fulfills the acceptance criteria that the neutron flux value at the center of the core is not less than 1.0x1015 n/cm^{2}s. However, it is necessary to analyze the alternative MTR core design.

The third conceptual design was proposed by T. Surbakti et al., [5], Where the height of the core is 85 cm. The alternative design of the RRI core of this study is to use MTR fuel with variations in the mass loading of the fuel, and the purpose of this study is to obtain the optimal configuration of the reactor core, which has a thermal neutron flux in the middle of the core 1.0 × 1015 n/cm^{2}s at a power of 50 MW with fuel as high as 70 cm. Plate type fuel design with two safety control rods as an addition to the outside of the core for increased safety. The design of the reactor core used is based on balanced core fuel management, while the design based on the core with single-use fuel (fresh fuel core) has never been done.

Based on previous research, the 5 × 5 core configuration with the amount of fuel 16 and control rod 4 is the most efficient fuel. The calculation of the core design was carried out using WIMSD-5B and BATAN-FUEL. These codes are used because they have been verified and validated using a similar reactor core, namely, RSG-GAS [6]. These codes are fast in calculations and simple in the model, and user friendly. Neutron flux and reactivity calculations have been performed using the BATAN-FUEL code [7]. Several core models have been carried out in previous studies [8]. For this year, the research will focus on the design, estimation of neutronic parameters using a single exhaust core configuration with two safety control rods, and U7Mo/Al fuel. The core configuration is a 5 x 5 grids and 70 cm high. The initial stage of research has been carried out to design research reactors in several countries in the world using high-density uranium molybdenum fuel, such as JMTR in Japan, CARR in China, MPRR-30 in India, HFR in the Netherlands and OPAL in Australia, KJRR in Korea [9]. This research reactor uses high-density silicide fuel, and only KJRR uses new fuel (U7Mo/Al). This research reactor uses high-density silicide fuel, and only KJRR uses the new U7Mo/Al fuel, which is being studied. The density of uranium in U7Mo/Al fuel can reach densities of up to 16 gU/cm^3 [9]. This research also conducted a study on the core design of MTR research reactors in Indonesia using U7Mo-Al fuel with a load of 360 - 450 grams [10]. This paper presents the results of research on the conceptual design of a new research reactor alternative from the neutronic aspect using new molybdenum fuel. Conceptual design is the first step towards obtaining a comprehensive concept design. The purpose of this research is to obtain an optimal research reactor core configuration design with the criteria of having a thermal neutron flux of at least 1.0×1015 n/cm^2s at a power of 50 MW with a core height of 70 cm. The length of one operating cycle must be more than 20 days and uses fuel more efficiently. In this conceptual design, the safety criteria applied to control the reactivity are one control rod stuck, the reactor remains sub-critical, even the control rod which has the greatest reactivity fails to enter the core.

RESEARCH METHODOLOGY

The analysis of the neutronic safety aspects of the MTR reactor core with high density aims to determine the optimal core configuration to meet the safety margins and acceptance criteria for the neutronic parameters. In this analysis, it is also determined the maximum discharge burn up at the end of each operating cycle. From this calculation, the maximum irradiation time or cycle length of the core can be determined. The safety analysis from the aspect of neutronic parameters was carried out with several calculations using the WIMSD-5B and Batan-FUEL computer codes. The WIMSD-5B code is used to perform cell calculations to obtain macroscopic cross-section constants, such as neutron diffusion, sigma absorption, and sigma fission for the U7Mo-Al fuel plate. U7Mo-Al fuel with a uranium mass between 360 - 450 grams is used in this calculation. The fuel macroscopic cross-section constant U7Mo-Al is a function of the fuel burn up in the 17 steps. The method used in this calculation is the multi-slab available in the WIMSD-5B code. The cross-section obtained is used by the core geometry as input for the calculation of the core in the Batan-FUEL code to determine changes in core reactivity, the distribution of the radial and axial peak power factors, and the neutron flux. One of the modules in the Batan-FUEL program is Batan-2DIFF to calculate core reactivity. The calculation of the neutronic parameters for the conceptual design of MTR alternative core requires several steps, namely: performs cell and core calculations. The core configuration shown in FIGURE 1 contains the fuel, as shown in FIGURE 2, and the control rod in FIGURE 3. The fuel geometry data are shown in TABLE 1. Generation of the macroscopic cross-section of fuel and core material was used by the WIMSD-5B code With nuclear data from ENDF / B-VII.1 [11]. The calculation is done by finding core neutronic parameter with the new U7Mo/Al fuel material. This cross-sectional generation was carried out using the neutron transport method. Meanwhile, an alternative conceptual core design was obtained using the 3-D neutron diffusion method, Batan-FUEL code.

CELL CALCULATION

In this study, the WIMSD-5B code was used to generate all material diffusion constants in the core in 4 (four) neutron energy groups. The energy limits for the neutrons are 10 MeV, 0.821 MeV, 5.531 keV, 0.625 eV, and 1×10^{-5} eV [12]. Specifically for the fuel generator group constants generated as a function of the 235U mass in the core (360-450 grams), temperature (cold and hot), and conditions Xe (free and equilibrium). The generation of the neutron diffusion constant group is under ambient conditions (20^{o}C) and as a function of burn-up. The multi-slab input model for fuel available in the WIMSD-5B program can be seen in FIGURE 4.

TABLE 1. Geometry data for fuel assembly of the MTR reactor [13]

Parameter (unit)	Values
Dimensions of standard fuel and control element, mm	77.1 × 81 × 700
Fuel plate thickness, mm	1.3
Cooling channel width, mm	2.55
Number of fuel plates element	21
Number of control plates element	15
Cladding material	AlMg2
Side plate Material	AlMg1
Fuel cladding thickness, mm	0,38
Dimension of active zone (*meat*), mm	0.54 × 62.75 × 700
Fuel material	U7Mo-Al
Loading mass ^{235}U, gram	360, 400, and 450
Absorber material	Ag-In-Cd
Absorber thickness, mm	3.38
Absorber cladding material	SS-321
Absorber cladding thickness, mm	0.85

CORE CALCULATION

Core calculations are carried out to achieve a fresh core with the MTR reactor. The core calculation is done using the Batan-FUEL code. The flow chart for the computation of the core using the Batan-FUEL code is shown in FIGURE 5. The core is modeled in two-dimensional X-Y geometry. In this calculation, the reactor power is assumed to be fixed at 50 MW, and the effect of the core configuration on the neutron flux and the length of the operating cycle is determined. Because the cycle length is a parameter that affects core reactivity, so core calculations are carried out for cycle lengths that meet safety criteria, namely, the criteria for the largest control rod stuck in one reactor is still sub-critical. It is called a stuck rod condition, if the control rod, which has the greatest reactivity value, does not make it into the core, then the reactor core must be in a sub-critical condition. If the criterion for the stuck-rod condition is not met, it is possible to use safety control rods on the core. For peak power factor (PPF), if the maximum PPF value exceeds the set limit, a reconfiguration is carried out where the position and location of the fuel are rearranged in the core so that the maximum PPF value is reduced. Otherwise, the core configuration cannot be designed safely. Core calculations are carried out for each level of the core and mass loading of uranium using a D_2O reflector.

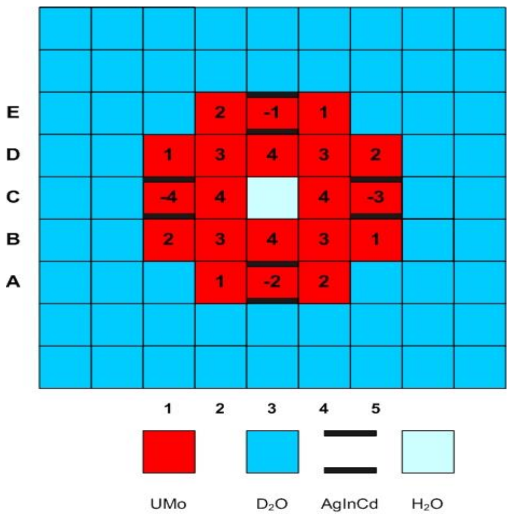

FIGURE 1. MTR core configuration [15]

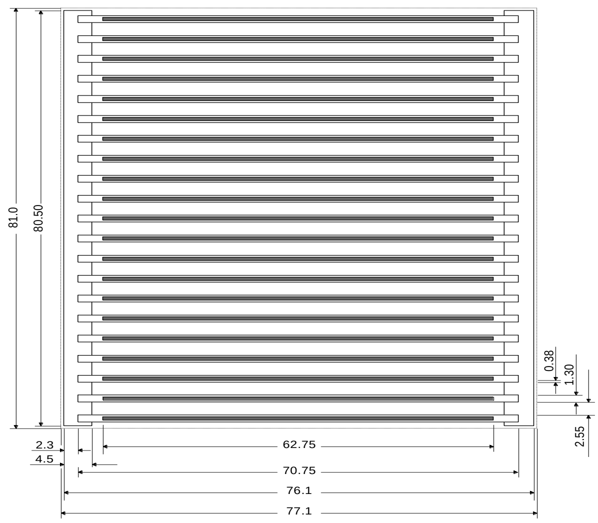

FIGURE 2. MTR fuel element [16]

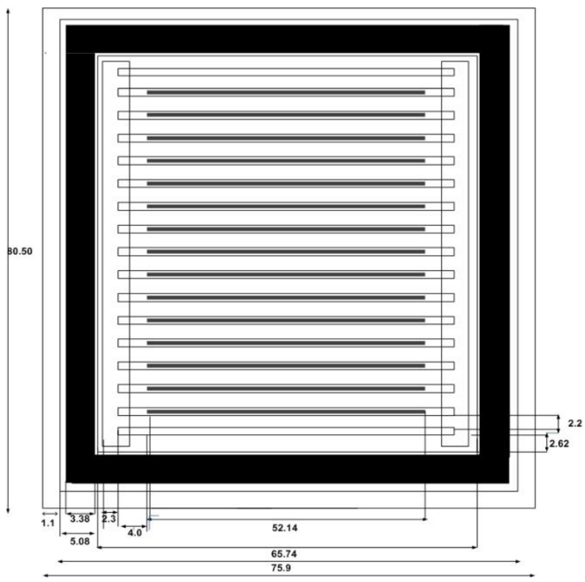

FIGURE 3. MTR control element [17]

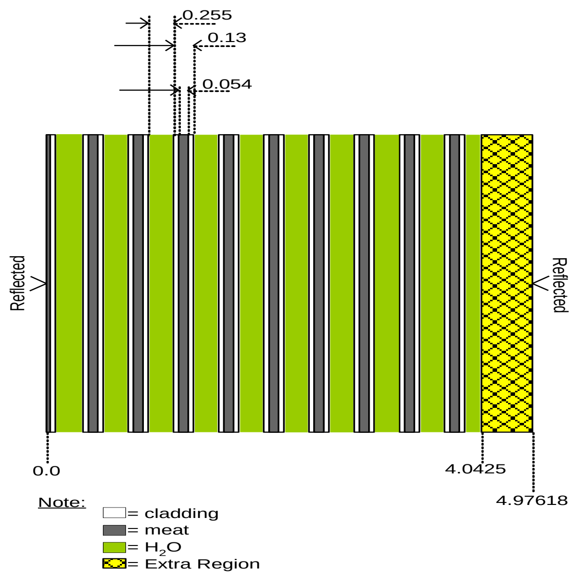

FIGURE 4. Cell model for U7Mo-Al fuel [18]

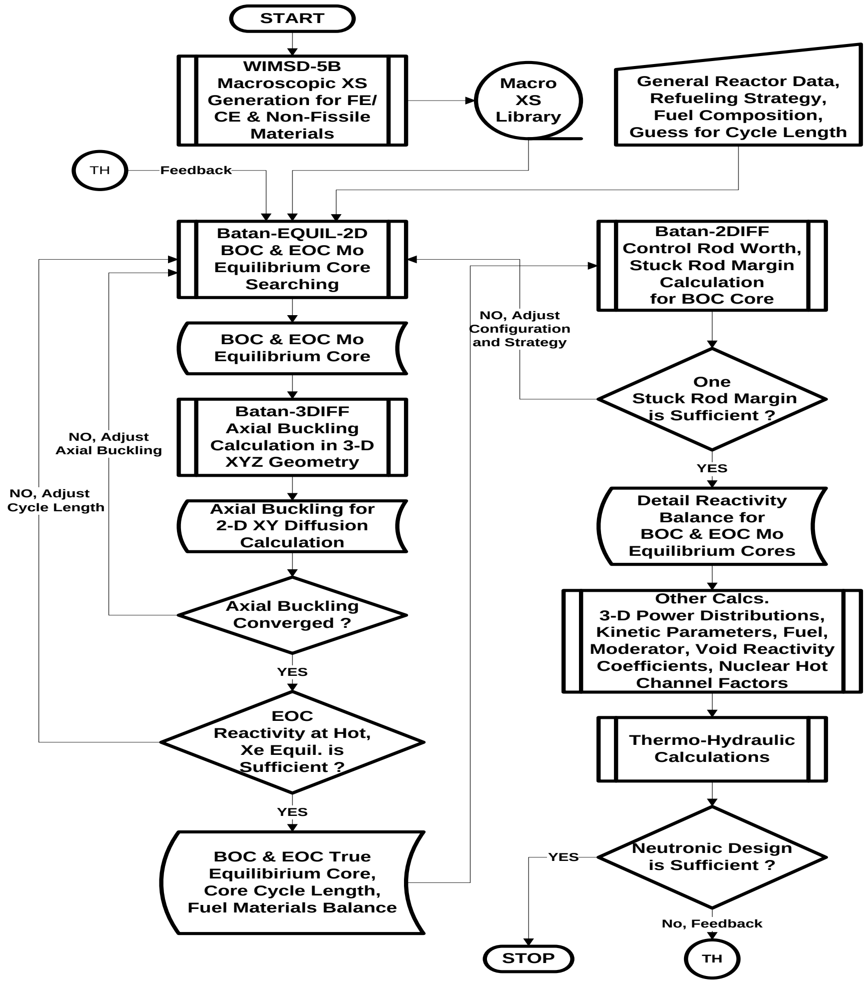

FIGURE 5. Flow chart for core calculation using Batan-FUELcode [19]

RESULT AND DISCUSSION

The calculation results of the neutronic design parameter for the MTR reactor with the core configuration of the fresh fuel using WIMSD5B/Batan-FUEL codes are shown in TABLE 2. The configuration of the fresh fuel core is shown in FIGURE 1. The core configuration of the 5 × 5 grids is the same as the CARR (China) core. It should be noted that if the minimum shutdown margin or shutdown reactivity is positive (+), the neutronic parameters of the results are not noted. The calculations are not shown in TABLE 2. The 5 × 5 core configuration consists of 16 standard fuel elements (FE) and 4 fuel control elements (CE) and four irradiation positions in the core are indicated by (IP1, IP2, IP3, and IP4) and the position the central (middle) irradiation in the core is IP0. The reflector used in this analysis is D_2O. The calculation result in TABLE 2 showed that the reactor core with fuel load 360 and 400 grams does not require a safety control rod. The core with a fuel load of 450 grams, it has required the addition of two (2) safety control rods, SCR1 and SCR2, outside the core. A safety control rod (SCR) is a control system or equipment which is the same size and type as a control rod but different positions and functions. Safety control rods are used in 2 positions outside the active core to balance the neutron flux. The function of the safety control rod is only for safety and not to shutdown the reactor, to regulate the power and reactor core reactivity.

In TABLE 2, it can also be seen that from the aspect of reactivity control, a core with a D2O reflector and a fuel load of 400 grams has a minimum shutdown margin value (stuck rod condition) compared to other core. This is due to the reactivity of the core in this core configuration without the safety rod. TABLE 2 also shows that the safety criteria for the minimum shutdown margin for all cores meet the safety criteria of - 2% Δk/k. The most important parameter for the safety of reactor operation is the stuck-rod condition. The minimum value must be - 0.5% Δk/k

The cycle length of this core is in accordance with the acceptance criteria, which is more than 20 days. This is due to a shift in the neutron spectrum that gets harder due to an increase in the level of 235U, which corresponds to the D_2O reflector. As a result, the control rod's absorption ability for thermal neutrons is reduced. From the aspect of reactor operation, TABLE 3 shows that the cycle length is the largest for the core with a fuel mass of 450 grams, but the shutdown margin is smaller than for the core with a fuel mass of 400 grams with a cycle length of 30 days—the FPD value changes due to the 235U mass of the fuel and the type of reflector. Therefore the main factor for determining the maximum value of FPD depends on the distribution of burn up around the FE and CE (IP) irradiation positions. The maximum radial FPD value for all cores is less than the value of the safety criteria 1.4. This indicates that the proposed design can meet safety requirements. The maximum burn up at the end of each operating cycle for all cores is less than the safety margin of 70%. The discharged burn up for fresh core in this analysis is 34.52%. The best neutronic parameter from an economic point of view is cycle length, but safety has met. The longer the fuel in the reactor core is economically very profitable.

TABLE 2. Neutronic parameters of fresh MTR core

No	Parameter	360 grams No SCR	400 grams No SCR	450 grams 2 SCRs
1	Power (MW)	50	50	50
2	Uranium density (gram/cm^3)	3.10	4.10	4.60
3	One cycle reactivity (%Δk/k)	6.80	7.86	8.16
4	Xenon reactivity (%Δk/k)	3.79	3.86	3.87
5	Samarium reactivity (%Δk/k)	0.03	0.03	0.03
6	Hot-cold reactivity (%Δk/k)	0.43	0.42	0.41
7	Core excess reactivity (%Δk/k)	15.61	16.40	17.65
8	Shutdown margin reactivity (%Δk/k)	-7.62	-5.52	-6.23
9	Total control rod reactivity (%Δk/k)	-22.23	-21.92	-23.88
10	Stuck rod condition (%Δk/k)	-1.97	-1.88	-1.63
11	Cycle length (days)	23	30	39
12.	Max. discharged burn up (%)	26.50	30.61	34.52
13	Average radial PPF	1.19	1.18	1.17
14	Average power density (W/cc)	635	635	635

FIGURE 6 shows the thermal neutron flux in the fresh core. Three cores whose thermal neutron flux met the acceptance criteria were more than 1.0 x 1015 n/cm2s in the middle of the core—the greater the amount of uranium in the core, the smaller the thermal neutron flux.

However, the length of the cycle is getting bigger. It is necessary to optimize which one to choose, whether the thermal neutron flux or the cycle length.

At FIGURE 7, it is shown that the reactivity of the control rods produced by the 3 fresh MTR cores. The reactivity changes when the control rod moves from one initial position to a certain position. It is called the integral control rod value for a given displacement distance, and the rate of change is the differential reactivity value. The control rod is inserted gradually from top to bottom of the core when the initial conditions of the reactor are critical. This simulation assumes that the integral value is zero when the control rod is fully drawn. The calculation steps are 10 cm, 20 cm, 30 cm, 40 cm, 50 cm, 60 cm and 70 cm. The result of the integral value is shown in FIGURE 7, and the control rod has just been inserted into the core. The resulting curve is exactly symmetrical due to the symmetric distribution of the fuel in the simulation. The differential values differ when the control rods are above and below the core. The greatest reactivity value of the control rod is in the middle because the high neutron flux is also designed to be in the middle.

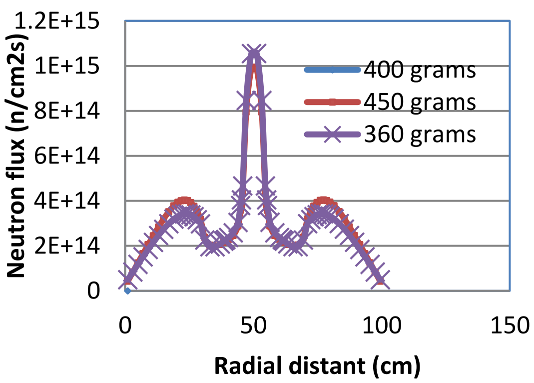

FIGURE 6. Thermal neutron flux in the MTR core

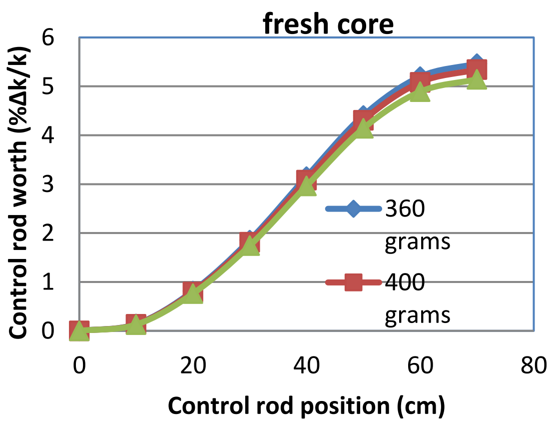

FIGURE 7. Control rod reactivity

CONCLUSION

The results of this study indicate the loading of fresh MTR reactor fuel for a new core configuration with thermal power of 50 MW. The optimal core that meets the safety limits uses U7Mo-Al fuel with a uranium mass of 450 grams and a cycle length of 39 days (1950 MWD). Based on these results for maximum capacity utilization, the fresh core configuration with a 5 × 5 grid with uranium mass 450 grams uses single discharged fuel management using two safety control rods. However, if it does not use a safety control rod, the MTR core can only be used with a load of 400 grams. The maximum thermal neutron flux in the center of the core is 1.0×10^{15} n /cm^{2}s, a minimum shutdown margin of -5.52% Δk/k and a cycle length of 30 days, and the core has an irradiation facility with a maximum thermal neutron flux of 4.0×10^{14} n/cm^{2}s.

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