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Neutronic Parameter Analysis of Plate-Type Fueled TRIGA 2000 Reactor by MCNPX

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ABSTRACT

A novel simulation to calculate the neutronic parameters of the TRIGA 2000 reactor using plate-type fuel has been performed. The plate fuel used was produced by the Indonesian Nuclear Industry (PT INUKI) with U_3Si_2-Al material. Neutronic parameters based on INUKI's plate-type fuel dimension and the current TRIGA's configuration were simulated using MCNPX. The simulation was performed by modeling the complete reactor's configuration on a fresh fuel core state. We obtained the kinetic parameter values from the simulation, i.e., delayed neutron fraction of 8.11×10^{-3}, a prompt neutron lifetime of 2.0551×10^{-4} s, and an average neutron generation time of 1.87×10^{-4} s. The excess reactivity of the reactor was 9.02 %Δk/k, while reactivity in the one-stuck-rod state was below -0.5 $ with an average value of -3.40 %Δk/k (-4.19 $). The average thermal neutron flux peak occurred at the central irradiation position with the value of 3.0×10^{13} to 3.1×10^{13} n/(cm^2 s). The reactor has a power peaking factor of 1.379 in the control rod position of 0 % on D3 fuel. The reactor had a negative feedback reactivity coefficient, except for the moderator coefficient. These results suggest that the current configuration of plate-type fuel met the nuclear reactor neutronic safety standards.

INTRODUCTION

The TRIGA (Training, Research, Isotopes General Atomic) 2000 reactor is one of Indonesia's prides, being operated and maintained by National Atomic Energy Agency (BATAN). This reactor was built on January 1, 1964, in the Bandung Nuclear Zone and inaugurated on February 20, 1965. As the name implies, this reactor plays a role in training, research, and isotope production and is manufactured by General Atomics. The benefits of this research reactor are felt by BATAN researchers and the Indonesian public [1].

This reactor has undergone a name change and has increased its power several times since it was first built. The TRIGA 2000 reactor was originally named TRIGA Mark II with a capacity of 250 kW. In 1971, the reactor power was increased to 1 MW. Subsequently, on 24 June 2000, the reactor power was again increased to 2 MW. The name was changed to TRIGA 2000, which was inaugurated by the vice president of Indonesia at the time, Megawati Soekarno Putri.

The G.A. Siwabessy reactor is the main reactor for supplying radioisotopes to meet the ASEAN regional needs [2]. As this reactor is planned to undergo thorough maintenance, a buffer reactor is needed for radioisotope production. The TRIGA 2000 reactor is predicted to become the buffer reactor [2,3]. For this reason, preparations must be made to ensure that the reactor can meet the radioisotope needs in the ASEAN region, especially in Indonesia.

Currently, the TRIGA 2000 reactor uses cylinder-type fuel with the composition of U-ZrH produced by General Atomic. However, at the TRIGA reactor user meeting on 24 March 2010 in Marrakesh, General Atomics announced that they would no longer produce the aforementioned cylinder-type fuel elements [4]. As a result, the reactor's operation was hampered by the exhaustion of currently used fuel. Due to the extensive benefits of this reactor, it is necessary to find a solution to allow the reactor operations to resume.

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The fuel that can substitute for the TRIGA 2000 reactor fuel is plate-type fuel with U_3Si_2-Al material. This fuel can be produced by the Indonesian Nuclear Industry, Ltd. (PT INUKI) and is already being used by the G.A. Siwabessy reactor with satisfactory results [3]. The use of fuel produced by PT INUKI will also reduce dependence on imported fuel. It is strongly believed that the reactor's modification due to the fuel change can be carried out because changes in the reactor core support had been made to increase reactor power several years ago [5].

Changes in the reactor core can lead to changes in the reactor's characteristics. One of the characteristics that need to be studied is the neutronic parameter. After this change, the features of the reactor must be ensured to meet the conditions of safety and security standards [6]. Neutronic parameters also describe the behavior of neutrons in the reactor, which plays a vital role in reactor operation. The neutronic parameters discussed in this study are core excess reactivity, shutdown reactivity, control rod worth, one stuck rod reactivity (shutdown margin), flux distribution, power peaking factor (PPF), negative feedback reactivity coefficient, and kinetic parameters.

Several studies have researched the modification of the TRIGA 2000 reactor fuel using plate fuel. In 2014, P. Basuki [7] conducted a study on modifying the TRIGA 2000 reactor to use plate-type fuel with two 5 × 5 configurations. The study was conducted for two scenarios, namely using Be reflectors and not using reflectors. In the study, it was found that reactors without reflectors have very low reactivity, and reactors with Be reflectors have a positive shutdown margin. In addition, the use of Be reflectors requires replacing existing reflectors.

In 2019, the I. S. Hardiyanti study [4] resulted in a higher reactivity than [7] using the same core configuration as in Fig. 1. Data were obtained using the MCNP6 code. However, this study used a simplified design. The design consisted of a reflector, water pool, fuel, and control rods, without a beam port, core grid, thermalizing column, or thermal column. In the same year, I. S. Hardiyanti [8] also conducted research to determine the minimum amount of fuel so that the reactor could be critical. From this research, it was found that the reactor can be critical with 15 fuel elements. However, the reactor will have less reactivity.

Other research with the same core configuration was performed by Surian Pinem [5] by using several codes, *i.e.*, WIMSD/5B, BATAN-FUEL, and BATAN-3DIFF. The research was carried out when the core was in an equilibrium state.

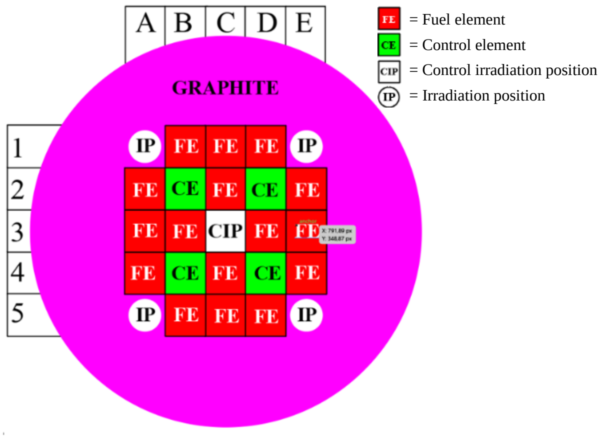

Fig. 1. Reactor core configuration using plate fuel.

This study examines the neutronic parameters after the reactor was modified using plate-type fuel. The core configuration was 5 × 5, with each end used as Irradiation Position (IP), and the middle part used as Central Irradiation Position (CIP), as shown in Fig. 1. Unlike the previous study by Pinem [5], the calculation of the neutronic parameters was carried out on a fresh fuel core state. The core configuration of fresh fuel elements will provide the most reactive core condition, providing the least core shutdown margin. The configuration gives the highest excess reactivity.

Moreover, this study made minimal modifications by maintaining the existing core to fill the gap of previous studies [4,7,8]. This study also models the TRIGA 2000 reactor as realistically as possible so that the results can be close to the actual conditions.

Neutronic parameter calculations were carried out by computer simulation using Monte Carlo N-Particle eXtended (MCNPX) software, a particle transport code software developed by Los Alamos National Laboratory. MCNPX was chosen because of its ability to model three-dimensional geometry in general, precise representation of transport effects, and continuous energy cross-sectional data [9]. It also can use neutron sources and perform flexible calculations. The use of MCNPX in criticality modeling of the TRIGA 2000 reactor can provide a detailed description of its neutronic characteristics because the geometry can be made as detailed as possible [10]. Moreover, MCNPX has been benchmarked with satisfactory results [11,12]. In this study, we used MCNPX v.2.7.0 and ENDF/B-VII as the nuclear library.

METHODOLOGY

Geometry specification of TRIGA 2000

The plate fuel produced by PT. INUKI is the Material Testing Reactor (MTR) type. Currently, this fuel is produced for utilization in the RSG-GAS Serpong reactor. The fuel contains U_3Si_2-Al material with a uranium density of 2.96 g/cc and 19.75 % enrichment. Each fuel element contains 21 fuel meats, and each control rod has 15 fuel meats. The fuel meats are then clad with AlMg2.

Plate-type fuels have several advantages. According to Wardhani's research [13], TRIGA 2000 reactor using plate fuel must use forced cooling. Subekti [14] conducted a study on plate-type fuel in the RSG-GAS reactor, showing that each sub-channel on the fuel-with a 2.55 mm width-has an even distribution of water flow velocity. This means that each fuel meat is being cooled down evenly. The fuel meat is also moderated uniformly, thereby increasing the value of the thermal neutron flux.

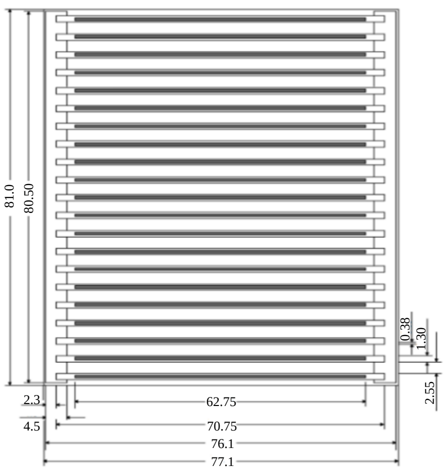

Fig. 2. Dimension of PT INUKI's plate fuel element, in mm [5].

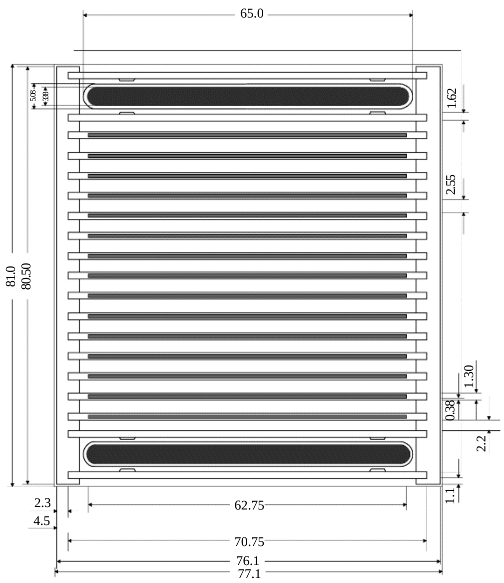

Fig. 3. Dimension of PT INUKI's plate control rod, in mm [5].

Figures 2 and 3 show the diagrams of fuel and control rods, respectively. The parameters of those fuel and control rods are shown in Table 1, while Table 2 shows the additional parameters of the TRIGA 2000 reactor.

Table 1. Parameters of PT INUKI's plate fuel and control rod [5].

Parameter	Value
Fuel and control rod dimension (mm)	77.1 × 81 × 600
Fuel plate thickness (mm)	1.3
Sub-channel coolant thickness (mm)	2.55
Number of plates in fuel	21
Number of plates in the control rod	15
Plate cladding material	$AlMg_2$
Plate cladding thickness (mm)	0.38
Fuel meat dimension (mm)	0.54 × 62.75 × 600
Fuel meat material	U_3Si_2-Al
U-235 enrichment (%)	19.75
Uranium density in meat (g/cm3)	2.96
U-235 content per fuel (g)	250
U-235 content per control rod (g)	178.6
Meat absorption material	Ag-In-Cd
The thickness of absorption material (mm)	3.38
Cladding material	SUS-321
Cladding thickness (mm)	0.85

Table 2. Additional dimensions of the TRIGA 2000 reactor [6].

Parameter	Value
Outer reflector radius	54.29 cm
Beam port radius	3 cm
IP height	55.4 cm
CIP dimension	8.05 cm × 7.61 cm × 55.4 cm

Neutronic parameters

Flux distribution

The neutron flux is defined by Eq. (1) as the number of neutrons in one cm^3 of volume multiplied by the average velocity of these neutrons [15].

$$\phi\left[\frac{\mathrm{n}}{\mathrm{cm}^2.\mathrm{s}}\right] \equiv n\left[\frac{\mathrm{n}}{\mathrm{cm}^3}\right]\bar{\mathrm{v}}\left[\frac{\mathrm{cm}}{\mathrm{s}}\right] \quad (1)$$

In MCNPX, flux neutron (tally F4) is calculated using Eq. (2).

$$F4 = \frac{1}{V}\int_{E_t} dE \int_{t_j} dt \int dV\, \phi(\vec{r}, E, t) \quad (2)$$

The normalization factor is used to adjust the simulation condition with the actual state of the reactor. This is calculated with Eq. (3).

$$\left(\frac{1\frac{\mathrm{J}}{\mathrm{s}}}{\mathrm{W}}\right)\left(\frac{1\ \mathrm{MeV}}{1.602\times10^{-13}\ \mathrm{J}}\right)\left(\frac{\mathrm{fission}}{200\ \mathrm{MeV}}\right) = 3.121\times10^{10}\,\frac{\mathrm{fission}}{\mathrm{W.s}} \quad (3)$$

For a 2 MW power source, the neutron flux is calculated in Eq. (4).

$$(2.0 \times 10^{6}\ \mathrm{W})\left(3.121 \times 10^{10}\frac{\mathrm{fission}}{\mathrm{W.s}}\right)\left(2.43\frac{\mathrm{n}}{\mathrm{fission}}\right) = 1.52 \times 10^{17}\frac{\mathrm{n}}{\mathrm{s}} \quad (4)$$

The actual flux is obtained by multiplying the amount of the output flux of the MCNPX with the source strength, as in Eqs. (5) and (6).

$$\Phi\left(\frac{\mathrm{n}}{\mathrm{cm^2.s}}\right) = \frac{\text{tally F4}\frac{1}{\mathrm{cm^2}} \times 1.52\times10^{17}\frac{\mathrm{n}}{\mathrm{s}}\times\frac{1}{k_{eff}}}{Volume} \quad (5)$$

$$\Phi\left(\frac{\mathrm{n}}{\mathrm{cm^2.s}}\right) = \mathrm{TMESH}\frac{1}{\mathrm{cm^2}} \times 1.52 \times 10^{17}\frac{\mathrm{n}}{\mathrm{s}} \times \frac{1}{k_{eff}} \quad (6)$$

There are two neutron flux equations as given in Eqs. (5) and (6). Equation (5) is used to normalize the flux released by F4, while Eq. (6) is used to obtain the flux using TMESH. Flux calculation for the F4 output uses Eq. (5) because the flux in F4 is volume-dependent, as in Eq. (2). To simplify the coding process, the value of volume in the MCNPX calculation is set to one so that the actual volume value is only considered in the normalization process. As for the flux normalization factor using TMESH, Eq. (6) is used because TMESH obtains the flux without considering the geometry but by tracking the particles one by one.

Power peaking factor (PPF)

Local Power Density (LPD) is the ratio between the maximum power density of fuel and the average power density of the entire reactor. PPF is the maximum value of LPD [16]. This parameter is used as one of the determinants of the reactor safety system. The reactor is expected to have an even distribution of power. The LPD can also determine whether fuel from a reactor has the potential to melt.

PPF can be generated using the F7 tally. This tally produces an output in the form of fission energy deposits in units of MeV/g. The power density may be obtained by multiplying the output of F7 by the density of the material calculated in F7, which results in the final unit of MeV/cm^3. Equation (7) examines the power density of each fuel meat (U_3Si_2-Al) on each fuel and control rod.

$$power\ density = tally\ F7\ \times\ meat\ density \quad (7)$$

Since PPF is the highest value of LPD, it is necessary to find the value of LPD for all fuels in the reactor. LPD can be obtained using Eq. (8).

$$LPD = \frac{\bar{p}_c}{\bar{p}_r} \quad (8)$$

with $\bar{p}_c$ is the average power density in a fuel element/control rod, and $\bar{p}_r$ is the average power density in the reactor.

Negative feedback reactivity coefficient

There are three negative feedback reactivity coefficients: moderator temperature, fuel temperature, and void. MCNPX cannot change the void percentage of moderated water [17]. The void coefficient calculation is done indirectly. The change in the percentage of voids in moderator water is replaced by changing the water density. This is due to the change of water to voids affecting the average density of all water in the moderator. A similar study was conducted by [18]. The relationship between the percentage of voids and water density is shown in Table 3. Table 3 assumes that the water vapor density is very small compared to the water density; hence, the total density of the moderator in x% void percentage is a result of $n(100\,\% - \%void) \times water\ density$ at 0 % $void\ percentage$.

Table 3. The relation between density and %void [17].

Density (g/cm^3)	Void (%)
0.9982	0 %
0.8984	10 %
0.7986	20 %
0.6987	30 %
0.5989	40 %
0.4991	50 %
0.3993	60 %
0.2995	70 %
0.1996	80 %

The moderator temperature and fuel temperature coefficients can be found by changing the characteristics of the material at a certain temperature. The material characteristics can be changed in the material data on MCNPX. The material characteristics of the MCNPX are contained in the ZAID code. The ZAID code for the characteristic data of uranium and water is shown in Table 4.

Table 4. ZAID Data of water and uranium for feedback reactivity coefficient [19].

Material	ZAID	Temperature (°C)
Water	lwtr.01t	26.85
Water	lwtr.03t	226.85
U-235	92235	20.45
U-235	92235.13c	226.85
U-238	92238	20.45
U-238	92238.13c	226.85

Kinetic parameters

The kinetic parameters of a reactor determine the state of the reactor over time. Kinetic parameters that are typically analyzed include prompt neutron lifetime (τ_r), delayed neutron fraction (β_{eff}), and average neutron generation time (Λ). τ_r is the average lifespan of fast neutrons produced by fission. Neutron lifespan is the time it takes for a neutron from birth to death (absorbed or underwent fission with the atomic nucleus). β_{eff} is the ratio of the delayed neutrons to all the neutrons born in the reactor, while Λ is the average lifetime of each neutron generation in the reactor. β_{eff} can be calculated using Eq. (9).

$$\beta_{eff} = \frac{k_t - k_p}{k_t} \tag{9}$$

with k_t is the total multiplication factor (delayed neutron and prompt neutron), and k_p is the prompt neutron multiplication factor.

Equation (9) is a method for finding β_{eff} known as the prompt method, which has been utilized by Mweetwa [17], Hassanzadeh [20], Henry [21], Mghar [22], and Michalek [23], with excellent results. Thus, Λ can be obtained using Eq. (10).

$$\Lambda = \frac{\tau_r}{k_{eff}} \tag{10}$$

with k_{eff} is the effective multiplication factor.

Kinetic parameters can determine the reactor's behavior during operation. The delayed neutron can disturb the stability of the reactor for a certain period.

RESULTS AND DISCUSSION

The TRIGA 2000 reactor geometric modeling

The reactor's parts being modeled in this study are the pool, four beam ports, fuel, control rod, reflector, thermal column, thermalizing column, core grid, and rotary specimen rack (Lazy Susan). The reactor has a pool with a depth of 6.26 m and a diameter of 99.06 cm [6]. The reactor is modeled to have a pool with 100 % pure water with an even water density of 0.9982 g/cm^3 throughout the reactor. The reactor core is at a depth of 5.0498 m, calculated from the water level to the center of the reflector [6].

Figures 4 and 5 show a side view of the TRIGA 2000 reactor core. Fuel elements made by PT INUKI have a length that matches the existing core; therefore, almost all parts of the fuel (especially the fuel which contains uranium) are in the reflector. In the middle of the core, there is an irradiation position. Fuel and control rods are arranged on top of the core grid. The existing core grid in the reactor must be replaced to match the fuel used.

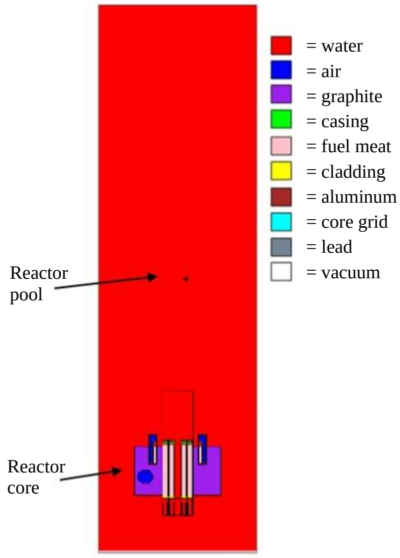

Fig. 4. Side view of the TRIGA reactor.

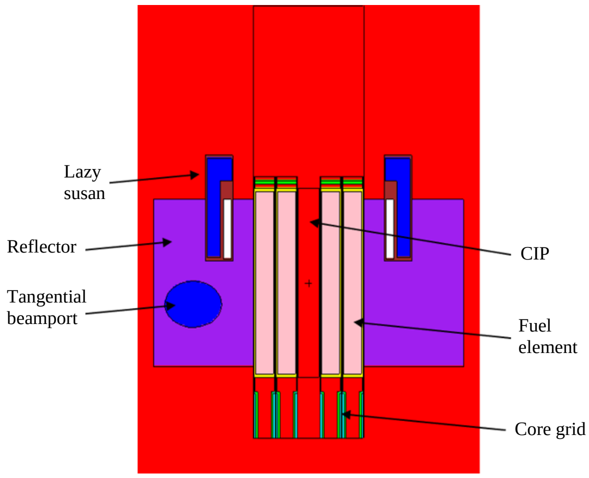

Fig. 5. Side view of the reactor core.

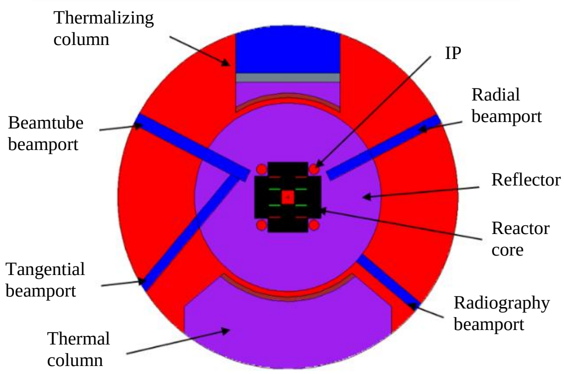

Fig. 6. The top view of all the modeled reactor parts.

Figure 6 shows the top view of the TRIGA 2000 reactor after being modified using plate-type fuel. One of the changes made was in the reflector.

The outer diameter of the reflector stays unchanged but the inner diameter of the reflector changes. The inner cross-section of the reflector was originally circular but has now ben changed according to the fuel arrangement. Changes were also made to the beam port. Each beam port has a tapering reflector whose thickness increases from the end of the beam port to the reactor core. In addition, the previous radial beam port (top right) and beam tube (top left) directly penetrate the reactor. After modification, the beam port is blocked by the graphite.

The fuel and control rods are assembled in a 5 × 5 position. Based on research by Pinem [5], the 5 × 5 arrangement in Fig. 1 is the best one. They are assembled with one Central Irradiation Position (CIP) in the middle and four Irradiation Positions (IP) in each corner. The fuel structure cannot be larger than this as it will hit the rotary specimen rack. The fuel is arranged symmetrically to facilitate the fuel reshuffling process. The reactor has a reflector outer diameter of 54.29 cm.

This study calculates the neutronic parameters in a fresh fuel state. Fresh fuel is a condition where all fuel has a burnup value of 0 %. Newly operated reactors, or in this case reactors changing their type of fuel, will use new fuel (fresh fuel). Afterward, the reactor can use the reshuffling and refueling methods with certain strategies. The average state of the reactor in each reshuffling and refueling period is called the state of equilibrium.

In this study, the particles that were considered in the simulation were neutrons. The calculation was carried out with 300 cycles with each cycle simulating 200 000 neutron particles. One neutron sample was inserted into each center of the fuel meat for a total of 396 neutron samples. The other 199 604 neutron samples were arranged automatically by MCNPX, which were scattered throughout the reactor. The tally calculation starts at the 76th cycle. This is because the first cycle to the 75th cycle had a large uncertainty value. The configuration above produces k_{eff} and tally calculations with an uncertainty below 0.1 %.

Kinetic parameter

The kinetic parameters of the TRIGA 2000 reactor are shown in Table 5, which are calculated at the 100 % lifted control rod position. The kinetic parameters obtained have a greater value than the ones obtained from research conducted by Pinem [5] at equilibrium conditions and also have a more excellent value than the TRIGA Mark II reactor. The delayed neutron fraction in the Pinem's [5] study was 7.153×10^{-3}, in the previous reactor was 7.2×10^{-3} [24], while in this study it was 8.11×10^{-3}. Higher delayed neutron fraction means more neutrons resulting after fission occurs - mainly from the decay of fission products - and will affect the reactor's dynamic during operation.

Table 5. Kinetic parameters of the TRIGA 2000 reactor.

Parameter	Value
Delayed neutron fraction (β_{eff})	8.11×10^{-3}
Prompt neutron lifetime (τ_r)	2.0551×10^{-4} s
Average neutron generation time (Λ)	1.87×10^{-4} s

The prompt neutron lifetime of the TRIGA 2000 after modification was 2.0551×10^{-4} s. This means that the lifespan of the prompt neutrons in the reactor was 2.0551×10^{-4} s. The TRIGA Mark II reactor has a prompt neutron lifetime of 6.0×10^{-5} s [25].

As a comparison, the average neutron generation time in the study by Pinem [5] in the equilibrium state was 5.229×10^{-5} s, whereas in this study was 1.87×10^{-4} s.

Several things cause this difference. The difference with the TRIGA 2000 reactor is due to different fuels and core configurations. The difference with Pinem's study [5] is due to differences in fresh fuel and equilibrium conditions. The difference is also caused by the calculation method used. The program code used by Pinem [5] was the deterministic calculation method, while MCNPX used the probabilistic approach.

The lifetime of prompt neutrons in the reactor after modification is longer than that before modification. One of the reasons for the shorter lifetime of the prompt neutrons in the reactor before modification is that the U-ZrH fuel itself contains a moderator. The possibility of neutrons immediately turning into slow neutrons and undergoing other fission reactions is more significant. The neutrons in the core, after modification, must pass through the water first to experience a moderation process and again undergo a fission reaction. Nonetheless, the reactor system is complex. There is a plausible cause of this other than the moderator-contained fuel. Further research should investigate the longer lifetime of prompt neutrons in the reactor after modification.

Several kinetic parameters are also calculated indirectly. The delayed neutron fraction is calculated by comparing the difference in the reactor's effective multiplication factor, including prompt and delayed neutrons, with the multiplication factor of the reactor regardless of the delayed neutrons. However, this method has been tested by [22] on the Moroccan TRIGA Mark II reactor with a difference of 1.65 % from the results of the direct study. The results of the prompt neutron lifetime are calculated directly by MCNPX by tracking the neutron movement one by one. The average neutron generation time is determined using Eq. (10).

Reactivity

The reactivity of the TRIGA 2000 reactor after being modified using plate-type fuel is shown in Table 6. The reactor has an excess reactivity of 9.02 %Δk/k, which is sufficient to keep the reactor critical. The excess reactivity in the study by Pinem [5] in the state of fresh fuel has a value that is not too dissimilar from the value of 10.40 %Δk/k obtained in this study. The excess reactivity in the equilibrium state is 6.61 %Δk/k. The reactor has a shutdown reactivity of -10.11 %Δk/k, and all control rods in the reactor can reduce reactivity by 19.13 %Δk/k.

Table 6. Reactivity of the TRIGA 2000 reactor.

Parameter	k_{eff}	Reactivity (%Δk/k)
Excess reactivity	1.09908	9.02
Shutdown reactivity	0.90815	-10.11
Control Rod Worth	-	19.13

Based on the difference in reactivity value, the calculation of the neutronic parameters in the fresh fuel and equilibrium states has quite different results. This is because the fresh fuel core still contains a lot of uranium. At the equilibrium core, as the uranium has undergone fission reactions, the rate of neutron multiplication is lower than that in the fresh fuel core.

The reactor's reactivity in a one-stuck rod state is shown in Table 7. The condition of one stuck rod on the four control rods has met the requirements, namely having reactivity of less than -0.5 $. The data also shows that each control rod has almost the same ability to absorb neutrons (worth) due to the reactor's symmetrical arrangement. In comparison, the reactor before modification has three types of control rod: shim, regulating, and safety. Those three control rods have different abilities to absorb a neutron.

Table 7. Reactivity of the TRIGA reactor at the state of one stuck rod.

Control rod	k_{eff}	Reactivity (%Δk/k)	Reactivity ($)	Safety Limit ($)
B-2	0.96555	-3.57	-4.40	< -0.5 $
B-4	0.96685	-3.43	-4.23	< -0.5 $
D-2	0.96764	-3.34	-4.13	< -0.5 $
D-4	0.96854	-3.25	-4.01	< -0.5 $

Reactivity of the reactor in a one-stuck rod state at the equilibrium core also has an even value. In the fresh fuel state, the mean value of reactivity is -3.4 %Δk/k, while other studies resulted in -2.96 %Δk/k [5] that is slightly different. Reactivity at the equilibrium core has a mean value of -6.60 %Δk/k. Similar to the excess reactivity, the reactivity values in the one stuck rod state between fresh fuel and equilibrium are quite different.

Based on the current safety standards, where the reactivity value of one stuck rod has to be less than -0.5 $ [7], it is determined that the current modified configuration of the TRIGA 2000 reactor using plate-type fuel has met the safety standard.

The graph in Fig. 7 presents the relation between the withdrawn control rod position and the reactivity. The reactor will be critical in the ideal condition when the entire control rod is removed at ±24 cm. However, the reactivity value shown in this graph does not consider other parameters. Reactivity may decrease due to changes in other parameters or by neutron poison. Therefore, the critical condition in the reactor will occur at the control rod position of higher than 24 cm. The graph in Fig. 7 can also analyze reactivity insertion accidents.

Figure 8 shows the relation between the change of the withdrawn control rod position and the change in reactivity. The difference in reactivity due to the control rod position change turns out to be non-linear. Figure 8 shows that the most significant shift in reactivity occurs during the evolution of the control rod position from 12 cm to 18 cm and from 18 cm to 24 cm.

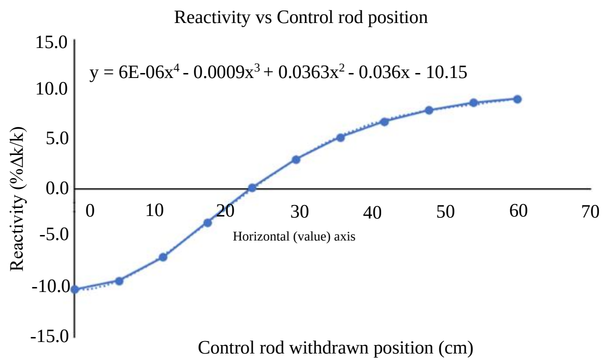

Fig. 7. Relation between reactivity and withdrawn control rod position.

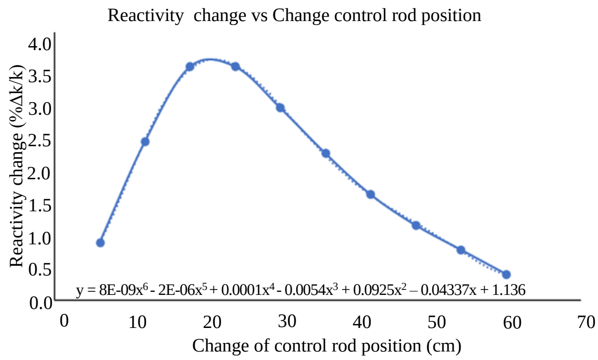

Fig. 8. Relation between reactivity change and withdrawn control rod position.

Thermal neutron flux

Thermal neutrons are an essential component in research reactors. This is because the research reactor uses these neutrons to irradiate certain materials. The research reactor is expected to have a high thermal neutron flux value. As stated before, the thermal neutron flux can be calculated by MCNPX using several methods. The first method is the F4 tally, and the second is TMESH. In this study, the F4 tally was used to find the average neutron flux in beam port, IP, and CIP cells. On the other hand, TMESH was used to find the distribution of thermal neutrons in the core in more detail. TMESH was examined in a cube geometry with a length of 110 cm (-55 cm to 55 cm on the x-axis), 110 cm wide (-55 cm to 55 cm on the y-axis), and 54 cm high (-27 to 27 on the z-axis).

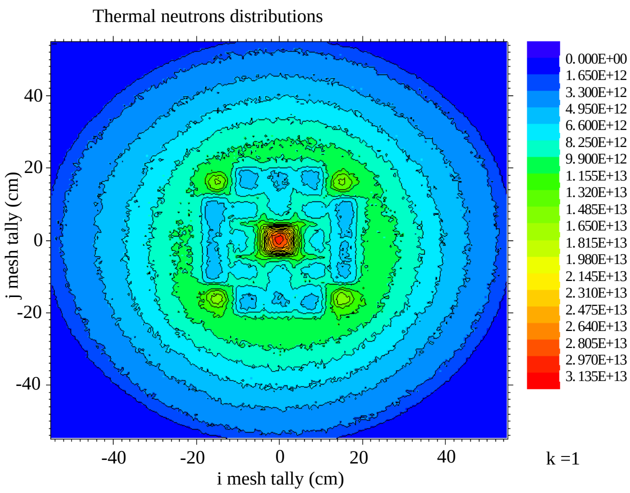

Fig. 9. Thermal neutron distribution of the TRIGA 2000 reactor on the XY plane.

Figure 9 shows an overview of the thermal neutron distribution of the TRIGA 2000 reactor using plate-type fuel. This distribution value is the average of the neutron flux values on the z-axis of the calculation domain. The peak of the neutron flux is obtained at CIP as 3.0×10^{13} - 3.1×10^{13} n/(cm^2 s). The thermal neutron distribution across all IPs has the same value of 4.7×10^{12} - 4.9×10^{12} n/(cm^2 s), while the thermal neutron fluxes at IP and CIP in the equilibrium state are 6.0×10^{13} n/(cm^2 s) and 2.0×10^{13} - 3.0×10^{13} n/(cm^2 s), respectively [5]. The thermal neutron flux at CIP has a smaller value than that at the central thimble in the reactor before modification, namely 5.18×10^{13} n/(cm^2 s) [1]. Before modification, the central part of the reactor was named central thimble, and afterward, it was called CIP. Nevertheless, the modified reactor has more irradiation positions than before [5], thus serving as an advantage for the TRIGA 2000 reactor.

Figures 10 and 11 show the distribution of thermal neutrons in the XZ and YZ planes. These two figures show that the value of the distribution of thermal neutrons tends to be more significant in the center of the reactor. These results cannot be compared to the previous study of Pinem because the z-axis range of the thermal neutron flux [5] and the central thimble was not known. The maximum thermal neutron flux value (not the maximum of the average) at the CIP on the fresh fuel core will be greater than the current value.

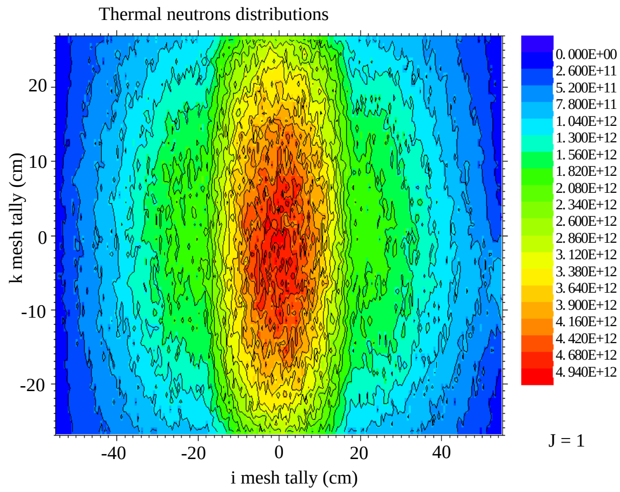

Fig. 10. Thermal neutron distribution of the TRIGA 2000 reactor on the XZ plane.

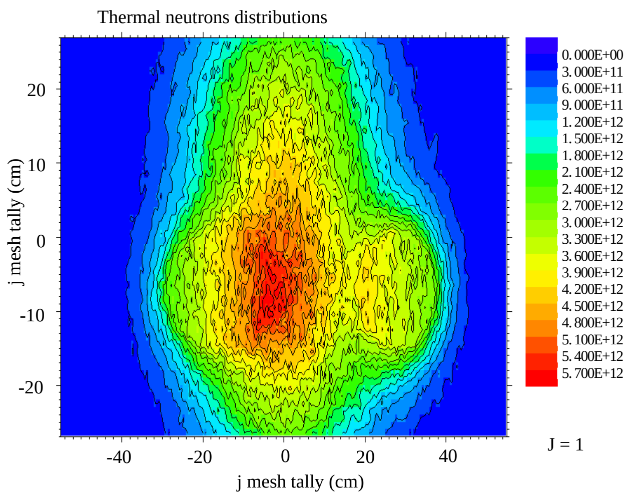

Fig. 11. Thermal neutron distribution of the TRIGA 2000 reactor on the YZ plane.

Changes in the reactor core, especially in the reflector, cause changes to the beam port. An additional reflector must cover the beam port that previously penetrates directly into the fuel. The beam port that does not directly penetrate the fuel must experience an increase in the thickness of the reflector. Table 8 shows the average flux value for the beam port using the F4 method. The thermal neutron flux value is still considerably high.

However, it is necessary to do further study, especially at the end of the beam port outside the reactor concrete wall, as this part was not examined in this study.

Table 8. The average flux at several positions of the reactor using F4.

Position	Average flux (n/(cm^2 s))
Beam port Radial	5.47×10^{12}
Beam port Radiography	1.14×10^{12}
Beam port Beamtube	1.09×10^{12}
Beam port Tangential	2.57×10^{12}
IP A5	1.76×10^{13}
IP A1	1.69×10^{13}
IP E5	1.86×10^{13}
IP E1	2.84×10^{13}
CIP	3.55×10^{13}

Power peaking factor (PPF)

PPF is required to indicate which part of the control rod consumes the most energy. PPF is the highest value of LPD, which is obtained by comparing the energy density of a fuel with the average energy density of the reactor. LPD in MCNPX can be obtained through F7, which is the energy deposit resulting from the fission process [19]. In other words, LPD describes the ratio of the fission energy density produced in fuel with the average fission energy density in the reactor. This study examines LPD in every fuel meat, followed by the average on each fuel or control rod. The graphs in Figs. 12 and 13 illustrate the average LPD in each fuel and control rod, as well as the change in the percentage of the withdrawn control rod.

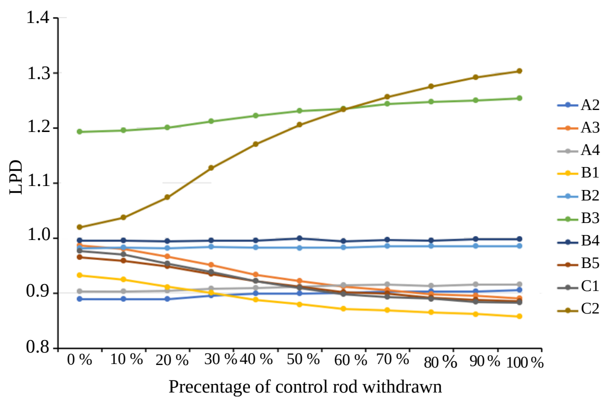

Fig. 12. Relation between LPD and the percentage of withdrawn control rod (Part 1).

The graphs in Figs. 12 and 13 show that D3 and C4 fuel elements dominate the PPF in the reactor. When the control rod is withdrawn from 0 % to 60 %, PPF occurs at D3. When the control rod is withdrawn from 70 % to 100 %, PPF occurs at C4. This is because the control element is in the middle of the reactor core. Based on the graph in Fig. 9, the distribution of thermal neutrons is concentrated in the middle. This neutron causes the fuel in the center of the reactor to undergo more fission than other parts. This reason is also in line with the higher LPD in B3 and C2 fuel elements than in other fuel elements.

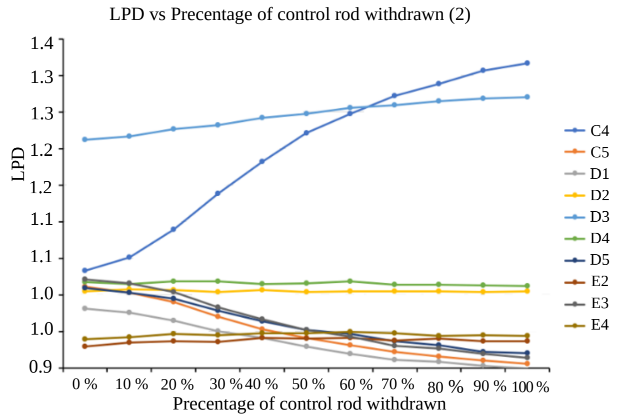

Fig. 13. Relation between LPD and the percentage of withdrawn control rod (Part 2).

The higher LPD value in the fuel in the middle of the reactor results in the fuel being used up more quickly. The data on the LPD distribution will help in the fuel reshuffling strategy to ensure that fuel utilization can be more efficient. In addition, the LPD data can be used to determine the hottest part of the reactor, namely the D3 and C4 fuel elements.

Based on these data, the PPF value of each fuel element does not exceed 1.4 (see Table 9). This shows that the fuel elements has met the safety standards. Another advantage is the even distribution of LPD for all fuels, except for fuel in the center position. However, if all the LPD values are averaged, the result will be close to one, as can be seen in Table 10.

Table 9. PPF value at each withdrawn control rod position.

Withdrawn control rod position (%)	PPF Position	PPF Value
0 %	D3	1.379
10 %	D3	1.364
20 %	D3	1.336
30 %	D3	1.296
40 %	D3	1.267
50 %	D3	1.243
60 %	D3	1.228
70 %	C4	1.227
80 %	C4	1.228
90 %	C4	1.233
100 %	C4	1.234

Table 10. Summary of PPF and LPD.

Parameter	Control rod position	PPF position	Value
Highest PPF Value	100 %	D3	1.379
Average PPF	0 %-100 %	D3 and C4	1.276
Average LPD	0 %-100 %	All control rods and fuels	1.010

Negative feedback reactivity coefficient

The negative feedback reactivity coefficient is essential for maintaining the reactor's stability when there is an error in the function or during the normal operational condition. Doppler coefficient can be found with the change in fuel temperature from 20.45 °C to 226.85 °C, while the moderator coefficient is found with the change in water temperature from 26.85 °C to 226.85 °C. Moreover, the void coefficient can be found by the change in void percentage from 0 % to 80 % indirectly using the water density data in Table 3. All the changes are done in a holistic manner, which means that the change in water density due to a change in void percentage occurs in the entire body of water.

Table 11 shows the Doppler coefficient value of the TRIGA 2000 reactor after modification, which has a negative value. This indicates that increased fuel temperature will decrease the reactor's reactivity, suggesting a good result. During operation, the change in fuel temperature will not make the reactor out of control.

Table 11. Doppler coefficient.

Temperature (°C)	k_{eff}	Reactivity (%Δk/k)	Doppler Coefficient ((%Δk/k)/°C)
20.45	1.09908	9.01481	-0.00132
226.85	1.09581	8.74330	

Table 12 shows the moderator coefficient value of the TRIGA 2000 reactor after modification, which has a positive value. This suggests a poor result as a positive coefficient value means that the reactor may be uncontrollable. However, Table 11 shows that the reactor has a smaller Doppler coefficient value than the moderator coefficient value. If both are added, the change in both fuel and water temperatures will generate a negative value. Doppler coefficient may help control the reactor affected by the difference in the moderator temperature.

Table 12. Moderator coefficient.

Temperature (°C)	k_{eff}	Reactivity (%Δk/k)	Moderator Coefficient ((%Δk/k)/°C)
26.85	1.09908	9.01481	0.00082
226.85	1.10107	9.17925	

Table 13. Void Coefficient.

Water Density (g/cm^3)	Void (%)	k_{eff}	Reactivity (%Δk/k)	Void Coefficient ((%Δk/k)/%void)
0.9982	0 %	1.09908	9.01481	-
0.8984	10 %	1.08045	7.44597	-0.1569
0.7986	20 %	1.04857	4.63202	-0.2814
0.6987	30 %	1.01188	1.17405	-0.3458
0.5989	40 %	0.96237	-3.91014	-0.5084
0.4991	50 %	0.90311	-10.72848	-0.6818
0.3993	60 %	0.81473	-22.74005	-1.2012
0.2995	70 %	0.70472	-41.90033	-1.9160
0.1996	80 %	0.55062	-81.61345	-3.9713

Table 13 shows the void coefficient values. The void coefficient data in Table 13 shows the coefficient value of the previous void percentage with the data on that row. For example, a change in the void coefficient at line 10 % represents a change from 0 % to 10 %. The minimal void coefficient value indicates that the reactor is very sensitive to changes in the void percentage in the reactor.

The graph in Fig. 14 shows the void coefficient value for the change in %void. The void coefficient tends to decrease at a higher void change and has a negative value. This shows that the reactor will not lose its stability when the void percentage in the reactor increases. A minimal void coefficient value can also help stabilize the reactor caused by the moderator coefficient.

The feedback reactivity coefficient values of the TRIGA 2000 reactor after the modification have been obtained. Most of the TRIGA 2000 reactor coefficients after modification show negative values. This is preferable because the increase in these parameters will not increase the reactor reactivity. In addition, although the moderator coefficient has a positive value, the Doppler and void coefficients can help stabilize the reactor. From these data, the reactor has a built-in safety system. The reactor can be said to have met the safety standards.

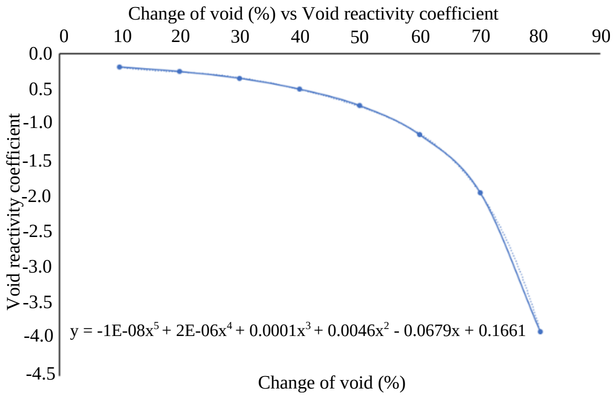

Fig. 14. The %void and void reactivity coefficient.

CONCLUSION

The TRIGA 2000 reactor neutronic parameters using plate fuel have been examined. Neutronic parameters were discussed in the state of the fresh fuel core. The kinetic parameter values of the TRIGA 2000 reactor are delayed neutron fraction of 8.11×10^{-3}, prompt neutron lifetime of 2.0551×10^{-4} s, and average neutron lifetime of 1.87×10^{-4} s. The excess reactivity of the reactor is 9.02 %Δk/k. The shutdown reactivity value for the reactor is -10.11 %Δk/k, and the control rod's worth value is 19.13 %Δk/k. Reactivity in the one stuck rod state of the reactor is below -0.5 $ with an average value of 3.40 %Δk/k (-4.19 $). The mean thermal neutron flux peak occurs at the CIP at 3.0×10^{13} - 3.1×10^{13} n/(cm^2 s). The PPF of the reactor is 1.379 at the control rod position of 0 % on D3 fuel. Except for the moderator coefficient, the reactor has a negative feedback reactivity coefficient. However, the reactor has a very small Doppler and void coefficient to help stabilize the reactor. It can be concluded that the current modification and fuel configuration have met the nuclear reactor neutronic safety standards.

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AUTHOR CONTRIBUTION

Arby Nuryana, Rida S. N. Mahmudah, and Azizul Khakim have participated sufficiently in this paper, including participation in the concept, design, analysis, writing, and manuscript revision. All authors have read and approved the final version of the article.

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