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Power and neutron flux distribution analysis in the RSG-GAS reactor: preliminary study to identify the reactor readiness as Power Ramp Test Facility (PRTF)

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Abstract. PRTF RSG-GAS is an irradiation facility that used especially for testing fuel pin element of nuclear power plant reactor. Before, during and after the testing, reactor shall remain in critical and steady state condition. In this research, we determine the PRTF readiness before testing. Pre-experimental analysis was performed using Monte Carlo method by MCNP-6 software. K$_{eff}$ value of RSG-GAS obtained in this study is $1.10234\pm0.00011$ at power operation (15 MW). The total power was obtained 13.6 MW with difference of 9.3% to the actual operating power at RSG GAS. The maximum power released by one fuel and the average rod power at operating power reach $2.4 \times 10^{-2}$ MW and $9.5 \times 10^{-3}$ MW. The total neutron flux of PRTF at J7 and K7 grid position respectively are $0.64 \times 10^{14}$ n.cm$^{-2}$s$^{-1}$(70.80% thermal neutron flux and 29.20% fast neutron flux); and $0.26 \times 10^{14}$ n.cm$^{-2}$s$^{-1}$(84.89% thermal neutron flux and 15.11% fast neutron flux). Both testing location show that neutron profile dominated by thermal neutron. This condition enables to use irradiated fuel elements. However, if the neutron flux present in PRTF is generate to excessive fission reactions, it will adversely affect to the fuel element which testing and the reactor. Therefore, the existence of thermal neutrons and their changes when the irradiation process take place becomes an interesting topic to do.

1. Introduction

In 2030, Indonesia requires the addition of 4.1 GW electricity supply every year. Renewable energy are needed to cover the energy shortage [1]. Nuclear power plants is one of the alternative energy that has been planned. The Government commissioned Badan Tenaga NuklirNasional (BATAN) to conduct an in-depth study on the preparation of development nuclear power plant in Indonesia. One of the indicator is the availability of fuel element testing facilities. The RSG-GAS has been equipped Power Ramp Test Facility (PRTF) for irradiation testing of Pressurized Water Reactor (PWR) fuel. These facilities have been prepared but has not been installed [2]

The Reaktor Serba Guna G.A Siwabessy (RSG-GAS: previous name MPR 30) is multipurpose research reactor, which was commissioned in 1987. The reactor maximum power is 30 MW with 15 MW operation power. The core uses the Silicide Fuel (U$_3$Si$_2$Al) with plate type fuel elements [3]. On the 10×10 core grid position, there were 40 standards Fuel Elements (FEs one fuel consists 21 fuel
plates), 8 Control Elements (CEs each consist 15 fuel plates), Be reflector elements, and other irradiation facilities as depicted in Figure 1 [4].

The criticality of RSG GAS at full configuration core is 1.10132 by Liem with JEND-4.0 data library. In the same paper reactivity value obtained 9.24%Δk/k [5]. These results are used to validate the data library that we use in this work (ENDF/B-VII.1).

Fuel elements testing aim to determine the standard of quality and safety. In the process of the testing, the fuel elements are moved up and down. Fission reactions occur during the process. Excessive fission reactions will increase the value of reactivity, a quantity value is related to $k_{eff}$ [6]. Increasing of the reactivity value is proportional to $k_{eff}$ value. The reactor with $k_{eff} > 1$ indicates supercritical condition. If the neutron flux changes in the core is more than 16%, the reactor will scram. The objective of the activity is to investigate the power and neutron flux distributions before the fuel elements insert to the PRTF. A recent data of neutron flux calculated by Taryo[7] is shown in Table 1.

Table 1. Thermal and Fast neutron in PRTF RSG-GAS [7]

| Thermal neutron flux (×10^{14} n.cm^{-2}.s^{-1}) | Fast neutron flux |
|-----------------------------------------------|-------------------|
| PRTF K7                                       | 1.2296            |
| PRTF J7                                       | 1.2296            |
|                                               | 0.1321            |
|                                               | 0.1321            |

As a research reactor, RSG GAS can be used for reactor fuel elements testing. Consequently, the power on the core must be kept constant during the testing process (as a safety function) [8]. Sutrisno, suggests that the generated power increase close to linearity, proportional to the position of the fuel elements against the reactor core [9].

This paper is organized as followed: experiment method including how to get power and neutron flux distributions, discussion of the measurement result with other papers, and the conclusion of the present work.

2. Numerical Methods

Monte Carlo is the main method in this work. The neutronic aspect were simulated with Monte Carlo N Particles (MCNP-6) with ENDF/B.VII.1 library. CEUP, the input name of RSG GAS model in our research, is shown in Figure 1. This geometry at the beginning of cycle (BOC) and other specification can be obtained elsewhere [10]. In this report, we assumed that CEUP was installing at operation power (15MW) and maximum power (30MW). The result reported in this work have been carried out with adequate number of histories so that the relative errors to be less than 1%.

To calculate the reactor criticality with MCNP, KCODE card has to be defined. The criticality value are obtained at full core configuration and the reactivity value measured can be written as followed [6]

$$\rho = \frac{k_{eff} + 1}{k_{eff}}$$

(1)
Figure 1. Core-reflector configuration of (Fig 1.a) RSG-GAS (Fig 1.b) CEUP, geometry model of RSG-GAS as input for MCNP code.

The flux neutron were tallied F4:N and the distribution of neutron flux were tallied FMESH4:N. Therefore to normalized the F4 Tally in order to reach the steady state thermal power of a critical system, the following scaling per units time should be used [11]:

\[
S = \frac{P[\text{Watt}] \times \frac{\text{Neutron}}{\text{fission}}}{1.6022 \times 10^{-13} \times \frac{1}{\text{MeV}} \times \text{fission} \times \text{keff}}
\]

The last, to get power distribution at the core we used F7:N Tally. Using Eq (2) as normalization factor of the power at each plate fuel element, the power can be determined [11]:

\[
P_i[\text{Watt}] = PF7_i \times \frac{\text{watt}}{\text{gr}} \times S \times 1.6 \times 10^{-13} \times \frac{\text{J}}{\text{MeV}} \times m[\text{gr}]
\]

The total power at core can be determined with Eq (4) and Eq (5) to obtain the average power at each fuel assembly [11]:

\[
P_{tot} = \sum P_i
\]

\[
P_{fuel \ av} = \frac{P_{tot}}{N}
\]

3. Results and discussion
In order to give confidence on the accuracy of MCNP code, it is recommended to validate the code with another data. Table 2. shows the comparison of Monte Carlo calculation results for this study with another data base (JEND-4.0).
Table 2. Comparison of Monte Carlo calculation results of RSG GAS reactor with another data base.

|                  | JEND-4.0 [5] | ENDF/B.VII.1 |
|------------------|--------------|--------------|
| $k_{\text{eff}}$ (full configuration core) | 1.10132      | 1.10234±0.00011 |
| Reactivity value ($\rho \Delta k/k$)     | 9.24%        | 9.3%         |

The MCNP6 codes agrees very well with difference of 0.0092% compare with the JEND-4.0 data library. According to the results presented in Table 2, the $k_{\text{eff}}$ value of CEUP is 1.10234±0.00011. This value is ideal for the supercritical condition, because the value is still close to the critical state of the reactor. In the same table, the reactivity value is performed 9.3% $\Delta k/k$.

The profile of thermal neutron flux at RSG GAS core is reported in Fig.2 (a). In this case, the thermal neutron fluxes are concentrated at the irradiation facilities. There are 5 peaks, 4 of which are located on the Irradiation Position (IP) and the other one is in the Central Irradiation Position (CIP). This condition caused by water moderator generating high density of neutron. Population of fast neutron in the core is caused by fuel fission reaction. Therefore the fast neutron fluxes are concentrated at the element fuel as performed at Fig. 2 (b).

![Figure 2](image_url)

**Figure 2** Distribution of (a) Thermal neutron flux (b) Fast neutron flux at RSG-GAS core (operation power = 15 MW)

![Figure 3](image_url)

**Figure 3**. Distribution of neutron energy at PRTF grid J7 & K7
To determine the condition of PRTF more specifically before testing, we calculated the distribution fluxes of neutron at PRTF. Fig. 3 shows the characteristics of neutron fluxes. The neutron fluxes have a maximum peak in the thermal energy range, of (0.01 - 1)eV and the peak of the fast neutron (1 - 10) MeV. This conditions shows that PRTF like the same as irradiation facilities in general, which have a higher thermal neutron population than fast neutrons. The presence of thermal neutrons in PRTF allows this facility to be used for fuel irradiation tests. The average flux neutron at PRTF is performed in Table 3.

Table 3. Average neutron flux at PRTF

| Grid    | Thermal neutron flux\(^{(a)}\) \(\times 10^{14}\) n.cm\(^{-2}\).s\(^{-1}\) | Fast neutron flux\(^{(a)}\) \(\times 10^{14}\) n.cm\(^{-2}\).s\(^{-1}\) | Thermal neutron flux\(^{(b)}\) \(\times 10^{14}\) n.cm\(^{-2}\).s\(^{-1}\) | Fast neutron flux\(^{(b)}\) \(\times 10^{14}\) n.cm\(^{-2}\).s\(^{-1}\) |
|---------|-------------------------------------------------|-------------------------------------------------|-------------------------------------------------|-------------------------------------------------|
| PRTF J7 | 1.229                                           | 0.132                                           | 0.907                                            | 0.455                                            |
| PRTF K7 | 1.229                                           | 0.132                                           | 0.442                                            | 0.222                                            |

(a) Result by Taryo, 2014 with maximum power 30 MW  
(b) This work

Figure. 4 Distribution of (a) Thermal neutron flux (b) Fast neutron flux at PRTF as function of position from nearest fuel (operation power = 15 MW).

In this case, neutron pollution decrease as a function of distance from the active core of reactor. The most neutrons in the reactor are generated from the fission reaction of fuel. Therefore an area close to fuel will have more neutrons. This is shown in Fig. 3 or Table 3, where the total neutron flux in the K7 grid is smaller than the J7 grid. This condition is clarified by Fig. 4, which shows the contours of the thermal neutron population (Fig. 4 (a)) and the fast neutron population (Fig. 4 (b)).

In this part, the average neutron flux will be compared to the results obtained by Taryo. If we compare it for each grid, this study has a maximum difference of 64.03%. This value is very high, but if we look at Table 3 shows that the neutron flux on the PRTF grid J7 and K7 obtained by Taryo has the same value. We assume that the grid is merged (J7 + K7) so that the results obtained are a combination of two grids. Besides that, both the results have the same trend, where the average thermal neutron flux has a higher value than the average fast neutron flux. This condition enables the fission reactions occur during the testing process. If the neutron flux in PRTF is generate to excessive fission reactions, it will adversely affect the fuel element testing and the reactor. Therefore, the existence of thermal neutrons and their changes during the irradiation process becomes an interesting topic to investigate.
The last aspect needs to be assessed to ensure the readiness of RSG GAS as the fuel elements testing facility in the reactor core power distribution. The results presented in Fig. 4 would be a benchmark during the fuel irradiation process. In this work, the operating power and the maximum power are sequence reach 13.6 MW (difference 9.3%) and 34.5 MW (difference 15%) with actual power at RSG GAS. The average value of each plate is 0.014 MW and the average power of one assembly is 0.30 MW. The power peaking factor are the link between the nuclear and the thermal hydraulic analysis of the reactor core as they define maximum power released locally in the core. The peaking factor in this case is 1.69.

4. Conclusion
The $k_{eff}$ value in this work is given $1.10234 \pm 0.00011$ with reactivity value is 9.3% $\Delta k / k$. The total power was obtained 13.6 MW with difference of 9.3% to the actual operating power at RSG GAS. The maximum power released by one fuel and the average rod power at operating power reach $2.4 \times 10^{-2}$ MW and $9.5 \times 10^{-3}$ MW. The total neutron flux of PRTF at J7 and K7 grid position respectively are $0.64 \times 10^{13}$ n.cm$^{-2}$ s$^{-1}$ (70.80% thermal neutron flux and 29.20% fast neutron flux); and $0.26 \times 10^{14}$ n.cm$^{-2}$ s$^{-1}$ (84.89% thermal neutron flux and 15.11% fast neutron flux). Both testing location show that neutron profile dominated by thermal neutron. This condition enables to use irradiated fuel elements. However, if the neutron flux present in PRTF is generate to excessive fission reactions, it will adversely affect to the fuel element which testing and the reactor. Therefore, the existence of thermal neutrons and their changes when the irradiation process take place becomes an interesting topic to do.

5. References
[1] BBPT. (2016). OUTLOOK Energi Indonesia 2016 Pengembangan Energi Untuk Mendukung Industri Hijau.
[2] Suwardi. (2014, Juni). Irradiation Test of Surrogated PWR Fuel Pin Design Manufactured With nUO2. Urania, 20 No.2, 56-108.
[3] BATAN. (1992). Safety Analysis Report, Multipurpose Reactor G.A Siwabessy. Jakarta
[4] Hong, L. P., & Sembiring, T. M. (2010). Design of Transition Cores of RSG-GAS (MPR-30) with higher loading silicide fuel. Nuclear Engineering and Design, 240, 1433-1442.
[5] Liem, P. H., & Sembiring, T. M. (2012). Benchmarking the New JENDL-4.0 Library on Criticality Experiment of a Research Reactor with Oxide LEU (20w/o) Fuel, Light Water Moderator and Beryllium Reflectors. *Annals of Nuclear Energy, 44*, 58-64.

[6] Kimov, A. (1975). *Nuclear Physics and Nuclear Reactor*. Moscow: MIR.

[7] Taryo, T., & Sembiring, T. M. (2014). Analysis of Neutron Flux Distribution in RSG-GAS Reactor with U-Mo Fuels. *Atom Indonesia, 39*, 21-34.

[8] Yidoung, Z. P. (2003). Power Ramp Testing Methode for PWR Fuel Rod at Research Reactor. *JAERI*, 51-52.

[9] Sutrisno, H. S. (2016, April). Pengujian Irradiasi Kelongsong Pin PRTF dengan laju Alir Sekunder 750 l/jam. *Buletin Pengelolaan Reaktor Nuklir*, pp. 19, 71-82.

[10] Pinem, S., Liem, P. H., Sembiring, T. M., & Surbakti, T. (2016). Fuel element burn up measurements for the equilibrium LEU silicide RSG GAS (MPR-30) core under a new fuel management strategy. *Annals of Nuclear Energy, 98*, 211-217.

[11] Snoj, L., & Ravnik, M. (2006). Calculation of Power Density with MCNP in TRIGA reactor. *International Conference, Nuclear Energy for New Europe 2006.109*, pp. 1-6. Slovenia: nss.