Analysis of oxide fuel element temperature of RSG-GAS by using experiment and calculation

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Abstract. Multipurpose Reactor GA Siwabessy is a research reactor designed to have nominal power of 30 MWth, located in PUSPIPTEK area Serpong and operated by BATAN. RSG GAS utilizes LEU-MTR with enrichment 19.75% and uranium density of 2.96 g U/cm³, 40 fuel rods and 8 control rods, 4 irradiation position (IP) and 1 central irradiation position (CIP) in 2 × 2 lattice, and reactor core surrounded by beryllium reflector. Forced convection cooling mode is applied to remove heat of fission product at normal operation, while natural convection cooling mode is used for reactor physics experiment, or cooling after transient condition. Forced convection cooling mode does not employ primary cooling system, where power is limited to 1% of the nominal power, while natural convection cooling occurs after transient process due to loss of coolant. Heat transfer occurs because of difference in coolant flow density where coolant flows upward since natural circulation valve is opened. This research is aimed to analyze heat removal capability in natural convection cooling mode at the most severe condition following reactor scram. The experiment of cladding and coolant temperature measurement is carried out in fuel coolant channel using instrumented oxide fuel element. Capability of generating reactor power in natural convection cooling is calculated using NATCON code. The measurement result of the highest temperature of plate fuel element is 121 °C. Meanwhile, the calculation results of NATCON show that the initial temperature of nucleate boiling $T_{ONB}$ occurs at power 685.86 kW and cladding temperature is 126.35 °C. These experiment results indicate that nucleate boiling does not occur in natural convection cooling due to scram or in other words safety margin has been met.

Key Words: temperature measurement, instrumented fuel element, residual heat, NATCON

1. Introduction

The Multipurpose reactor GA Siwabessy (RSG GAS) is a research reactor which is designed at nominal power of 30 MW, located in Serpong PUSPIPTEK area and operated by BATAN. RSG GAS uses LEU-MTR type fuel with enrichment of 19.75% and uranium density 2.96 g U/cm³. The reactor core consist of 40 fuel elements and 8 control rods, 4 irradiation positions (IP) and 1 position of central irradiation position (CIP) in the grid 2x2, the reactor core is surrounded by a beryllium reflector, as shown in Fig. 1. Light water is used as a coolant and moderator.

In RSG-GAS normal operation, forced cooling mode run by a primary pump is employed RSG-GAS. The pump is equipped with fly wheel to avoid the drop in the primary coolant flow rate in case it is failed when the reactor is in high power. It is possible the reactor operates at 1% of nominal power without the primary pump availability for some application such as foil irradiation at fuel element position [1]. When the pump is failed, the reactor core is cooled by natural convection. This condition occurs if the primary coolant flow rates drops. Low reactor power may also cause the primary and
secondary cooling system to fail and then trigger the natural convection cooling mode to run. The reactor safety is provided in order to prevent any release of radioactivity from releasing from the reactor core to environment. It is important to have enough margin of reactor safety in the reactor normal operation and any postulated accident conditions, for example force convection, natural convection, and loss-of-flow accidents. RSG-GAS uses both of forced and natural convection cooling mode, in the process of heat transfer from the reactor core. When the reactor operates with natural convection mode, reactor power is limited to 1% of nominal power, where the reactor cooling system does not work. Natural convection cooling mode is used for reactor physics experiments. In addition, natural convection cooling modes can also occur after the reactor scram. After scram occurs, the natural circulation flaps located under the plenum will open, caused by the cooling pressure difference between the reactor and the plenum. In this condition heat transfer only occurs because of differences in water density in the coolant channel. The natural convection in the gap of plate type fuels is important to take into account because it uses natural phenomena for residual heat dissipation on a relatively narrow gap, for that we need a limit where the safety margin for the initial occurrence temperature of onset of nucleate boiling (ONB) must be fulfilled.

![Diagram of RSG-GAS Core Configuration](image)

Note:
BE = Beryllium element, BS+ = Beryllium reflector with stopper, FE = Fuel element, CR = Control rod, IP = irradiation position, CIP = central irradiation position, PNRS = pneumatic system, HYRA = hydraulic system, PRTF = power ramp test facility.

Fig. 1. RSG-GAS Core Configuration [1]

Many studies have been performed regarding natural convection reactor analysis by researchers. Mochizuki (2013) performs natural convection test modeling and analysis from Phenix fastbreeder
(FBR) reactors. The purpose of this study was to assess the application of the verified NETFLOW ++ code, using various water facilities and validated using a plant data from the loop-type FBR "Monju" and the "Joyo" loop type fast reactor[2]. J Zou (2014) Conducting the residual heat removal, using a simple passive PWR model. The simulation is an observation of effective residual heat removal (PRHR), in normal feed water and split feed water channels. The study has been conducted to learn the effectiveness of the system against the PRHR capacity [3]. Ambrosini (2008) conduct a study on the phenomenon of flow stability in single channel and heated channel with boiling and supercritical fluids. This study also observed transpiration effects in condensation and evaporation and deterioration of heat transfer to supercritical fluids [4]. Hastuti EP et al (2006) conduct an experimental and analysis results on loss of flow transient using instrumented fuel element, under IAEA coordinated research projects on safety significance of postulated initiating events for different research reactor types and assessment of analytical tools [5]. Hastuti EP (2013) after the Fukushima accident, also conducted an analysis of natural convection heat transfer on RSG GAS spent fuel storage using Grace and NATCON calculation programs [6]. Furthermore, Hastuti EP (2018) also conducted an analysis of abnormal control rod withdrawal for innovative research reactors using PARET-ANL codes [7]. Pavel Zitek (2014) summarizes the basis for the solution heat removal by natural convection of conventional nuclear reactors and reactors with coolant flow through the fuel [8].

The researchers also conducted analysis and benchmarking of various thermohydraulic phenomena in research reactors using IAEA 10 MW reactor data. Tewfik (2011) conducted a residual heat transfer study with natural convection mode in the MTR research reactor, using the RELAP5/Mod3.2 calculation program. The results of calculation of fuel cladding temperature are compared with IAEA 10 MW reactor data [9]. Adorni (2005) conduct a safety analysis using IAEA 10 MW benchmark reactor, to study fuel conversion from the use of high enriched uranium to the use of low enriched uranium fuels, under flow blockage of a single Fuel Assembly (FA) conditions, using thermal-hydraulic best estimate RELAP5 code [10]. Tewfik (2011) was study the application of techniques for combining computer codes for reactor safety analysis. This study uses three-dimensional code, neutronic-kinetic-thermal hydraulic. Model (3D-NKTH) based on the combined code of PARCS and RELAP5 / Mod3.3, this computer code has been developed for the IAEA Patent on fuel with high enrichment uranium [11]. Meanwhile Omar (2013) perform an analysis of upward flow when LOFA occurs, investigated with kinetic and thermal hydraulic neutron coupling models for a slow and fast LOFA, the benchmark done with IAEA 10MW MTR reactor [12].

The objective of this paper is to discuss the comparison between calculation and measurement results of instrumented fuel element for natural convection. NATCON code is employed to assess the thermal hydraulic of the natural convection based on the reactor core RSG-GAS model. The output of the code for natural convection condition is compared with the measurement results for temperature distribution in fuel, cladding, and coolant core thermo-hydraulics. The main focus of this study is at temperature distribution of the fuel element, cladding material, as well as the analysis of on-set of critical heat flux phenomenon in research reactor. Investigation on some major parameters, i.e. fuel cladding temperature and bulk coolant temperature, is performed to find out whether or not the exceed their safety limit in case any nucleate boiling occurs in the coolant when the reactor experiences scram.

2. Methodology
The methodology used in this analysis is performed by calculation using NATCON computer code. The calculation results obtained include prediction of thermohydraulic and safety parameters as a function of reactor power. Meanwhile, the experiment that will be done include plate and coolant temperature measurement in fuel channel, using two instrumented oxide fuel element.

2.1. NATCON Computer Code
Heat transfer calculation in natural convection cooling mode for RSG GAS is carried out using a natural convection calculation code for research reactor with plate-type fuel, NATCON. NATCON code was developed for thermohydraulic analysis in steady state of plate fuel of the research reactor cooled by
natural convection [13]. The maximum limit of the reactor operating power in the natural Convection cooling is the occurrence of onset of nucleate boiling (ONB). This code will calculate buoyant force, friction force, coolant flow rate, heat transfer coefficient, cladding temperature, fuel temperature, and wall temperature where initial nucleate boiling occurs based on Bergles-Rohsenow correlation. Cooling water flows to remove the heat generated by fuel with upward flow, because of mass density difference in cooling water in the fuel gap. The buoyant force produced is compensated by friction force in the opposite direction caused by cooling water with particular density. The buoyancy and friction forces are shown in Equation (1) and (2) below [13].

\[
F_n = g(\bar{\rho}_c - \rho_{AMB}) \cdot A_c \cdot L_c
\]

(1)

where:

- \(\rho_c\) = average coolant density of heated column, can be expressed as:

\[
\bar{\rho}_c = \frac{1}{L_c} \int_0^{L_c} \rho_c(x) \, dx
\]

(2)

- \(\rho_{AMB}\) = ambient water density in the reactor tank, [g/cm³];
- \(A_c\) = flow area of coolant channel, [cm²];
- \(L_c\) = length of heated water column, [cm].
- \(g\) = gravitation acceleration, [cm/det²]; \(g = 9.80665 \text{ m/det}^2\);

Buoyant force makes flow inhibited by friction force produce pressure drop. Flow velocity will reach certain value, which is called terminal velocity, where buoyant force is in equilibrium with friction force. This friction force can be expressed as [13]:

\[
F_f = \frac{\left(\rho \nu\right)^2_{in}}{2g} \cdot A_c \cdot \left[ \frac{1}{2} \rho_{in} + \sum_{i=1}^{n} \frac{f \Delta z_i}{\rho_i D_H} + \frac{1}{\rho_{out}} \right]
\]

(3)

where:

- \(\rho\) = water density at any point 'z', [g/cm³];
- \(f\) = friction factor;
- \(\nu\) = water velocity in the inlet channel, [cm/detik];
- \(\rho_{in}\) = water density at inlet channel, [g/cm³];
- \(\rho_{out}\) = water density at outlet channel, [g/cm³];
- \(\Delta z_i\) = axial location along the coolant channel, [cm];
- \(D_H\) = hydraulic diameter of coolant channel, [cm].

The code employs some particular equations to calculation the temperature of fuel meat, fuel plate, and coolant, as well as pressure drop, coolant flow rate, heat flux, and safety margin towards ONB. Bergles-Rohsenow correlation is used to compute the heat flux at ONB.

\[
q_{ONB} = \frac{P_z^{1.156}}{9.23} \left[ 1.8(T_s - T_{sat}) \right]^{0.463} P_z^{0.0234}
\]

(4)

where:

- \(q_{ONB}\) = heat flux at onset of nucleate boiling, [W/cm²];
- \(P_z\) = pressure at coolant at any point 'z', [bar abs.];
- \(T_s\) = clad surface temperature, [°C];
- \(T_{sat}\) = saturation temperature of water, [°C];
- \(P\) = pressure at channel exit, [bar abs.];
Point boiling occurs when ONB occurs. If at this time flux is raised, then fraction of void will occur in the cooling channel causing flow instability.

2.2. Experimental

Two thermocouple-instrumented oxide fuel elements (IFE) are used to measure fuel element temperature [5]. Three thermocouples are embedded to IFE RI-10 for fuel element surface temperature at various height. On the other hand, three thermocouples are installed at IFE RI-11. One of them is used for fuel plate temperature measurement and the other two are used for the measurement of coolant inlet and outlet temperature. The IFEs are positioned in the core position for the highest radial power density factor.

The IFE is placed at the hottest position for fuel plate temperature measurement. The reactor operates at a specific power level using forced convection cooling provided by two primary pumps. After the reactor achieved steady power, the measurement of reactor power and fuel plate temperature is carried out every two hours to achieve steady recorded data. Meanwhile, a continuous data recorder is used to measure the temperature of IFE at natural convection mode. Data recorded are obtained before the application of natural convection mode. When the two primary pumps are shut down and the primary isolation valve is closed, the difference of the coolant pressure in the plenum and reactor core causes the natural circulation valve to open. The position of natural circulation flaps are shown in Figure 3.
Data are acquired during the reactor operates in steady state to scrams and the pumps are on. The data obtained are assessed and analysed to study the temperature characteristics as a function of time and any possible phenomenon that might occur. Moreover, the measurement results are studied and compared to the calculation analysis, which is done for the reactor nominal power level. For this analysis, the inputs are inlet coolant temperature, coolant flow rate, and measured power. Eventually, NATCON code is employed to analyse the fuel element temperature and safety margin as a function of power in natural convection cooling mode.

3. Results and Discussion
The calculation results of thermohydraulic parameters in the natural convection cooling at several power level with oxide fuel are summarized in Table 1. As explained earlier, the operation safety in natural convection mode is limited by the maximum power permitted, i.e. the temperature initiating the occurrence of onset of nucleate boiling ($T_{ONB}$). The calculation is started by power generation of 10 kW without operating the primary and secondary coolant, then raised by 100 kW step to reach the maximum power. The coolant upward flow occurs naturally due to density difference caused by coolant temperature, which receives the heat produced by fuel element. Coolant flow velocity occurs if buoyant force ($F_B$) is equal or greater than friction force ($F_F$). The force causing the coolant flow will increase linearly as the generated power increases, as shown in Table 1.
The maximum power is achieved when $\Delta T_{ONB} = 0^\circ$C. The maximum power permitted in the natural convection cooling is 685.56 kW, with coolant flow rate 7,697 cm/second because of coolant density difference from the lower axial nodes to the upper axial nodes, where the Re number is between 741.6 and 1312.0. Therefore, it indicates that, in natural convection cooling system, the heat generated in RSG-GAS is removed by laminar flow. Since there is no forced heat removal, then the cladding and fuel temperatures are nearly as high as the fuel temperature. As the temperatures are almost the same, accumulation of heat in the cladding occurs and initiates nucleate boiling temperature. The maximum cladding and fuel temperatures are 126.35°C and 127.33°C, respectively.

The power level at natural convection cooling mode reaches 685.56 kW. The reactor power margin does not change in natural convection cooling mode because of power limitation 1% of the nominal power or about 300 kW. The results of safety analysis indicate that RSG-GAS is able to operate above the operating condition specified. The thermal hydraulic profile along the fuel element at 685.56 kW at natural convection is shown in Figure 4. The axial highest temperature is resulted from heat transfer of fission product heat of the fuel element and coolant flow rate through channel. The hottest fuel plate indicates the highest fuel element temperature.

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**Table 1. Calculation results of RSG-GAS with natural convection cooling mode using NATCON**

| Power, kW | 10    | 100   | 299   | 300   | 400   | 500   | 600   | 685.6 |
|-----------|-------|-------|-------|-------|-------|-------|-------|-------|
| $Q^*$ per plate, kW | 0.0325 | 0.325 | 0.65  | 0.975 | 1.300 | 1.625 | 19.500 | 22.281 |
| $T_{ONB}$, °C | 72.38  | 56.95 | 47.9  | 37.71 | 27.56 | 17.72 | 8.13  | 0     |
| $FB$, Pa | 7.41E+04 | 2.35E+05 | 3.32E+05 | 4.10E+05 | 4.74E+05 | 5.34E+04 | 5.99E+05 | 6.32E+02 |
| $V$, cm/s | 0.806 | 2.687 | 3.697 | 4.886 | 5.709 | 6.470 | 7.162 | 7.697 |
| Flow rate, kg/s | 0.0014 | 0.0046 | 0.0066 | 0.0083 | 0.00968 | 0.011 | 0.0121 | 0.013 |
| $T_{in}$, °C | 44.5  | 44.5  | 44.5  | 44.5  | 44.5  | 44.5  | 44.5  | 44.5  |
| $T_{out}$, °C | 51.6  | 65.6  | 73.6  | 79.3  | 84.3  | 88.3  | 92    | 97.87 |
| $T_{clad}$, °C | 52.04 | 67.65 | 76.81 | 83.64 | 98.34 | 108.35 | 118.1 | 126.35 |
| $T_{fuel}$, °C | 52.05 | 67.67 | 76.84 | 83.73 | 98.91 | 109.07 | 118.96 | 127.33 |
| Mass flowrate, kg/m²·s | 7.99 | 26.63 | 38.71 | 48.41 | 56.57 | 64.11 | 70.96 | 76.26 |

**Table 2. Maximum power and operating limit values in the RSG GAS natural convection mode**

| Reactor Power, kW | 300   | 685.56 |
|-------------------|-------|--------|
| Heat flux per plate, kW | 0.9750 | 2.2281 |
| $\Delta T_{ONB}$, °C | 37.71  | 0.00   |
| $FB$, Pa | 41,0317 | 63.1775 |
| $V$, cm/s | 4,886 | 7.697 |
| Flow rate, kg/s | 0.0083 | 0.0130 |
| $T_{in}$, °C | 44.5  | 44.5  |
| $T_{out}$, °C | 79.3  | 97.87 |
| $T_{clad}$, °C | 83.64 | 126.35 |
| $T_{fuel}$, °C | 83.73 | 127.33 |
| Mass flowrate, kg/m²·s | 48.41 | 76.26 |

The reactor power margin does not change in natural convection cooling mode because of power limitation 1% of the nominal power or about 300 kW. The results of safety analysis indicate that RSG-GAS is able to operate above the operating condition specified. The thermal hydraulic profile along the fuel element at 685.56 kW at natural convection is shown in Figure 4. The axial highest temperature is resulted from heat transfer of fission product heat of the fuel element and coolant flow rate through channel. The hottest fuel plate indicates the highest fuel element temperature.
GAS is removed by laminar flow. Since there is no forced heat removal, then the cladding temperature is nearly as high as the fuel temperature. As the temperatures are almost the same, accumulation of heat in the cladding occurs and initiates nucleate boiling temperature. The maximum cladding and fuel temperatures are 126.35 °C and 127.33 °C, respectively.

The power level at natural convection cooling mode reaches 685.56 KW and the reactor power margin does not change in natural convection cooling mode because of power limitation 1% of the nominal power or about 300 KW. The results of natural convection cooling experiment after the reactor scram. After the reactor has been operated at 30 MW, the highest cladding temperature is 121°C, while the maximum cladding temperature according to the Safety Analysis Report of RSG-GAS is 145°C. This temperature 121°C as measured indicates condition when stagnant return flow occurs, in which buoyant force exceeds friction force and then initiates the occurrence of coolant flow in fuel gap due to density difference. The natural convection coolant flow rate at this condition is 7.697 cm/s. Based on analysis of NATCON calculation results, the residual power is 641.82 kW generating heat that is still possible to be transferred by natural convection. The results of safety analysis indicate that RSG-GAS is able to operate above the operating condition specified. The temperature profile along the fuel element at 685.56 kW at natural convection is shown in Figure 4.

For the reactor protection system, the reactor is not allowed to operate when the primary coolant is not available at 300 kW, which is 1% on nominal power. This condition is set to provide sufficient safety margin. By comparing the measurement and calculation results to the design data, the safety margin of the fuel element temperature (<145°C) is achieved.

CONCLUSION
After the maximum fuel plate temperature characteristics in the core with natural convection cooling mode is evaluated, natural convection cooling mode can be turned on during the reactor normal operation for reactor experiments and residual heat dissipates following a reactor scram. The IFE reaches its maximum cladding temperature, i.e. 121°C. It concludes that the maximum safety margin is achieved since the temperature limit of the fuel plate is 145°C. NATCON code gives results that the residual power is 641.82 kW at the maximum cladding temperature.

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