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Simulation of Modified TRIGA-2000 with Plate-Type Fuel under LOFA Using EUREKA2/RR-Code

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ABSTRACT

The TRIGA-2000 research reactor in Bandung, Indonesia, has operated for over 50 years. Recently, the problem of fuel availability arises, since its fuel is no longer produced. A modification of reactor core with new plate-type fuel has been suggested. The study of the neutronic assessment of plate-type fuel elements reactor core had been done. The next assessment that needed to be done was thermal-hydraulic analysis. The purpose of this study is to simulate the thermal-hydraulic characteristics of major parameters, such as reactor power, fuel cladding temperature, and departure from nucleate boiling ratio (DNBR) due to LOFA transient, using EUREKA2/RR code. During steady-state condition, downward flow forced convection mode for core coolant system is operated. The upward flow occurs when the natural circulation mode takes place. Hottest core channels temperature during LOFA conditions was considered. The reactor core was modeled as three channels, *i.e.*, the hottest channel, the average channel, and the channel for control assemblies, respectively. The simulation was based on the steady-state condition of 2 MWt reactor power, cooling mass flow rate of 63.5 kg/s, and inlet coolant temperature to the core of 35.5 °C. The result shows that the hottest fuel cladding temperature does not cause a nucleate boiling. During LOFA, the residual heat was removed by natural circulation flow that occurred slowly. In order to have larger inertia force, provision of the flywheel in the shaft of primary coolant pump is suggested.

INTRODUCTION

The TRIGA-2000 is one of the research reactors in Indonesia that has operated for over 50 years. It is a pool-type reactor, using zirconium hydride as moderator and light water as coolant. Its power output has been increased from 1000 kW to 2000 kW. The heat that is generated from the TRIGA-2000 reactor core is currently transferred through natural convection mode and transmitted by the coolant pump to the plate-type heat exchanger.

After a long operation, the problem of fuel availability arises, since the producer, General Atomics, could not produce new fuel further. The supplier of TRIGA fuel element, CERCA/TRIGA International, is considering discontinuation of the manufacture of TRIGA fuel [1].

A new modification of reactor core with plate-type fuel has been suggested to solve that problem, since PT INUKI (*Industri Nuklir Indonesia*) Persero has produced plate-type fuel elements which have been used successfully for many years in RSG-GAS, as well as the replacement component of control rods [2].

Assessment of neutronic aspects of a conceptual design of a 2-MWt research reactor core has been done by P. Basuki *et al*. (2014) using certain boundaries. A reactor using rod-type

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fuel elements can be converted to allow it to use plate-type fuel elements [3,4]. The study was continued with the neutronic assessment of reactor core configuration for the plate-type fuel elements; further, the next step, thermal-hydraulic analysis, is required [5]**.** In order to meet the safety requirement of reactor operation, each modification needs safety assessment prior to its utilization. Hence, the use of plate-type fuel elements in the TRIGA-2000 requires some thermal-hydraulic simulation related to safety aspects. In this case, the core cooling system should be changed and modified from natural circulation mode to forced circulation. Sudjatmi *et al*. (2015) recommend that the core cooling system should be done by forced convection, since it is indicated that if natural circulation is used for 2 MW a boiling of reactor coolant can occur [6]. Forced circulation mode allows a high heat transfer rate, and therefore can avoid coolant boiling that causes damage to reactor fuel. Furthermore, the cladding temperature should be kept lower than the saturation temperature at the power of 2 MW utilizing forced convection**.** The thermal-hydraulic transient analysis of modified TRIGA-2000 plate-type fuel is an extension of several other studies reported previously. Therefore, the transient simulation using EUREKA2/RR code can be carried out. This code is widely used to analyze the thermal-hydraulic transient of plate-type fueled research reactor cores. Heat transfer and critical heat flux (CHF) correlations used in EUREKA2/RR have been specially developed for research reactors in which plate-type fuel is adopted. Besides, the code provides a coupled thermal-hydraulic and point kinetics capability [7]. EUREKA2/RR code has been verified with a satisfactory result and successfully used for the analyses of thermal-hydraulic characteristics of reactivity insertion accident (RIA) as well as loss of flow accident (LOFA) of the JRR-4 reactor in Japan, the RSG-GAS reactor in Indonesia, and the TRIGA Mark-II research reactor in Bangladesh [8,9].

The scenario of LOFA impacts reactor safety parameters due to loss of flow in the primary coolant. The main objective for reactor safety is to keep the fuel in a thermally safe condition with adequate safety margins during all operational modes (normal and abnormal states). To achieve this purpose, a preliminary analysis LOFA is useful for assessing reactor safety. The core thermal-hydraulic parameters such as temperature, pressure, coolant flow rate and critical heat are important to be studied. Generally, the cladding temperature before reactor scram and reverse flow change from downward to upward are crucial conditions that should be considered. Therefore, the objective of the present work is to simulate the transient characteristics of major parameters such as reactor power, fuel cladding surface temperature, and departure from nucleate boiling ratio (DNBR) due to LOFA using EUREKA2/RR code. Then, it will be determined if the parameters exceed the safety limit or if any nucleate boiling occurs in the hottest fuel coolant channel.

THEORY

Reactor description

The TRIGA-2000 is a pool-type reactor surrounded with a concrete shield. It has two primary cooling pumps, and natural circulation flaps will be installed in the bottom plenum. The new modified configuration of the core consists of 16 standard fuel elements and four control fuel elements. An irradiation channel is located in the middle of the core. Figure 1 shows the new modified core configuration. The fuel material is U_3Si_2-Al enriched to 19.7 %, while the absorber material is Ag-In-Cd.

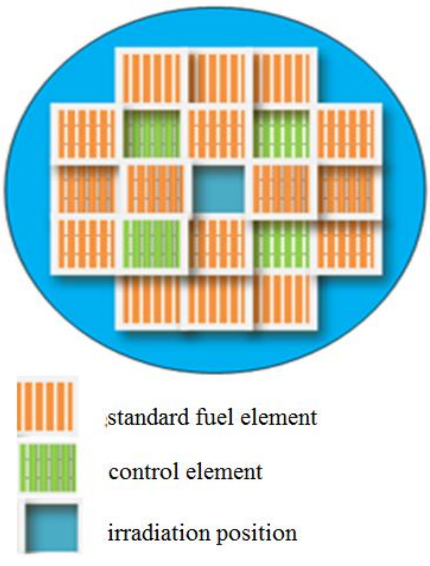

Fig. 1. The reactor core configuration [5].

Currently, the flow of primary coolant system is maintained by the centrifugal pumps that provide a total flow rate of 215.8 m^3/h. Heat generated in the reactor core is transferred to a water-to-water heat exchanger of primary coolant loop, while the secondary water loop is cooled by an external cooling tower. In the normal steady state power of 2 MWt, the modified TRIGA-2000 reactor uses the downward flow of forced-convection cooling mode. At present, the temperature of primary cooling water flowing into the tank is 32.2 °C [10]. Therefore, to remove the heat from the reactor core, a primary coolant system and a secondary coolant system are provided. In the natural convection mode of LOFA, the fuel elements are cooled by natural convection in which the flow direction is upward.

Fuel elements

The data on standard plate-type fuel elements and control fuel elements, taken from the fuel element specification of RSG-GAS, is presented on Fig. 2 and Table 1. The standard fuel element consists of 21 plates, while the control fuel element consists of 15 fuel plates and enclosed by absorber material. The cladding material of standard fuel elements and control fuel elements is AlMg.

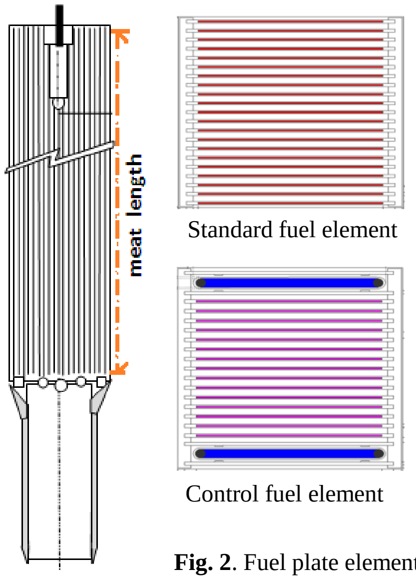

Fig. 2. Fuel plate element.

Table 1. Fuel element specifications

No.	Fuel element parameter	Design value
1	Fuel meat	
	width (mm)	62.75
	length (mm)	600.00
2	Distance between top of plate and top of meat (mm)	12.50
3	Fuel plate length (mm)	625.00
4	Cooling channel	
	width (mm)	67.10
	gap (mm)	2.557
5	Flow area of 1 standard fuel element (m^2)	36.03×10^{-4}
6	Flow area of 1 control fuel element (m^2)	2.57×10^{-3}
7	Number of fuel element	18.857

METHODOLOGY

In principle, there are two prevalent steps in the simulation process, namely modeling and running. The first step, namely modeling, involves obtaining and defining the data of the modified TRIGA-2000 reactor core including dimension/size and operating parameters. The second step, running the simulation, can be divided further into were running EUREKA2/RR at steady state condition followed by running LOFA transient simulation. Therefore, determination of steady-state condition and careful examination of the input data to perform the initial conditions prior the running computer of transient modeling are important steps. In the steady state thermal-hydraulic calculation, the COOLODN2 code was used. Figure 3 charts the methodology used in the LOFA transient simulation.

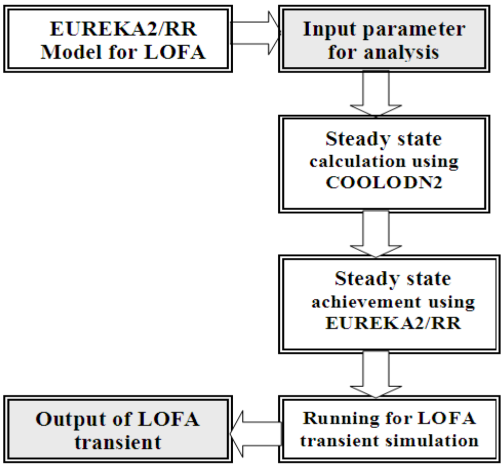

Fig. 3. Schematic of simulation methodology.

Simulation scenario

First, the reactor core was cooled by forced downward flow under normal condition. A LOFA was stipulated to occur due to loss of electrical power supply for the circulation pumps. Since the LOFA transient was protected against, a trip signal will be initiated by the reactor protection system (RPS) when the flow rate of the primary coolant reaches 85 % of its initial value. Second, the scram took place with 0.5 second delay time. This means that the control rods would start to drop into the core to shut the reactor down. Residual heat will be reduced after the flow inversion by the natural circulation through the flap valves in the bottom of core. Flow coast down will occur with loss of the flow rate through the core. The simulation scenario of coast down flow (CDF) of modified TRIGA-2000 using the measurement data taken from existing operation of pump in which mass flow rate becomes 0 kg/s (no flow) after nine seconds from loss of energy supply. In this simulation, it was assumed that once an accident occurred, it would not be followed by another type of accident; thus, it is a single failure. As a note, this CDF time is faster than the CDF of the TRIGA-IEA-R1 in Bangladesh which is approximately 27 s [11].

Assumptions

Simulation of the LOFA has been conducted based on several assumptions as follows: (i) An equilibrium core condition had been achieved before LOFA occurred, (ii) The steady state thermal-hydraulic performance of modified TRIGA-2000 core is designed for operation of the reactor at 2 MWt and the inlet coolant temperature to the core is a realistic value of 35.5 °C, (iii) The radial peaking factor to average radial power ratio is 1.24 [5], (iv) The reactor core is cooled using a pump which provides an effective mass flow rate of 63.0 kg/s, (v) The kinetics parameter are as follows: $\beta_{effective}$ = 0.0072, reactivity coefficient of fuel temperature = -2.0×10^{-5} Δk.k^{-1}/°C, and reactivity coefficient of coolant temperature = -6.0×10^{-5} Δk. k^{-1}/°C [12].

EUREKA2/RR core model

In the modified TRIGA-2000 core model in EUREKA2/RR, only the fuel region with upper and lower plenum was considered. The whole core was divided into several regions called channels and these channels differ from each other by power generation, coolant mass flow rate, and hydraulic diameters. Each fuel element, surrounded by water moderator, was considered in defining the channel. As shown in Fig. 4, channel-I represented the hottest sub-channel position, while channel-II and channel-III represented the average channels for standard fuel elements and the control fuel channel, respectively [9]. The axial power distribution factors were divided into 10 node segments. A by-pass node was used to simulate natural circulation after flow inversion occurs.

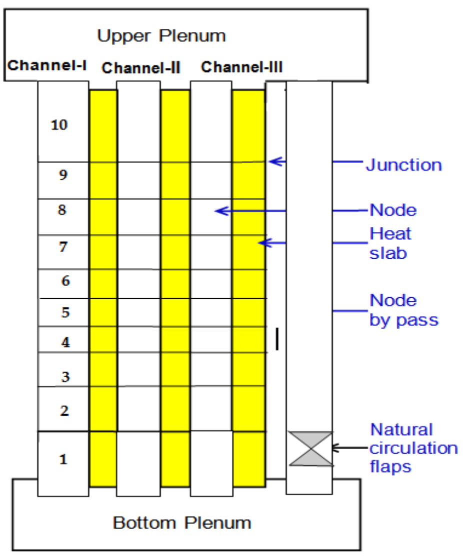

Fig. 4. Analysis model of EUREKA2/RR [9].

RESULTS AND DISCUSSION

Initially, steady state calculations were performed and the results of thermal-hydraulic parameters in the hottest channel of fuel element were obtained. The steady-state thermal-hydraulic performance of the modified TRIGA-2000 core was determined with good agreement with COOLODN2 code. Table 2 shows that the calculation results of EUREKA2/RR agree very well with the results of COOLODN2. Table 2 shows that cladding surface temperature, DNBR (departure from nucleate boiling ratio) and OFIR (onset of flow instability ratio) remain within safety limits. Furthermore, in addition to steady-state calculations, the LOFA transient have been simulated.

Table 2. Steady state results of the hottest channel

Hottest channel	EUREKA2/RR	COOLODN2
Coolant velocity, m/s	0.947	0.982
Meat temperature,°C	86.67	84.20
Sat. temperature,°C	115.37	113.22
Cladding temp,°C	85.27	83.50
Outlet coolant temp,°C	50.83	51.78
DNBR (-)	3.74	3.17
MOFIR	-	2.58
Heat flux, KC/m^2.hr	2.160×10^5	2.240×10^5

In the simulation of the LOFA transient, various parameters such as reactor power, cooling flow rate, cladding surface temperature, saturation temperature, and DNBR were considered as outputs. Figure 5 shows the transients of reactor power, hottest cladding surface temperature, sub-channel cooling flow rate, and saturation temperature. In this simulation, the steady state had been determined for the first 10.0 s; simultaneously, transient thermal-hydraulic parameter are presented. Since the coast down flow of TRIGA-2000 only takes 9.0 s, the residual coolant flow rate decreases rapidly as a consequence of the lack of flywheel in the shaft of the pump.

There is one case of LOFA transient simulation that requires special attention, *i.e.*, the crucial peak temperatures occuring just after coolant pump loss of power and removal of residual heat by coolant flow that is decreasing rapidly. Meanwhile, the expected reversal flow of natural circulation of about 0.015 kg/s does not indicate sufficient coolant flow to remove the residual heat present. As shown in Fig. 5, the cladding surface temperature increased due to loss of flow (before trip of control rod); furthermore, for a brief time after reactor trip, cladding surface temperature decreased rapidly, and then it increased to 107.7 °C. However, it was lower than saturation temperature (temperature difference was 14.5 °C). It means that nucleate boiling did not occur in the hottest position of reactor core with margin of 14.5 °C.

Figure 6 shows the simulation result for DNBR, outlet bulk water temperature from the coolant channel, and heat flux in the hottest sub-channel. The DNBR attained the minimum value of 3.55 after 0.75 s from the start of transient in which the minimum allowable limit was 1.50 [13]. It means that the conditions of LOFA transient stayed within safety margin even as the heat contained in the reactor core could not be transferred to the secondary system. The forced conversion coolant mode is safer than the natural cooling mode as expressed by previous studies [6,14]. Further, as shown in Fig. 6, the outlet bulk water temperature from the hottest channel gradually decreased and did not rise further, although residual heat removal by natural circulation flow proceeded slowly. The upward reverse flow of natural circulation is caused by the difference in density between the hot water and the cold water [15,16]. At the initial stage of the LOFA condition, it is very important to maintain a downward core flow.

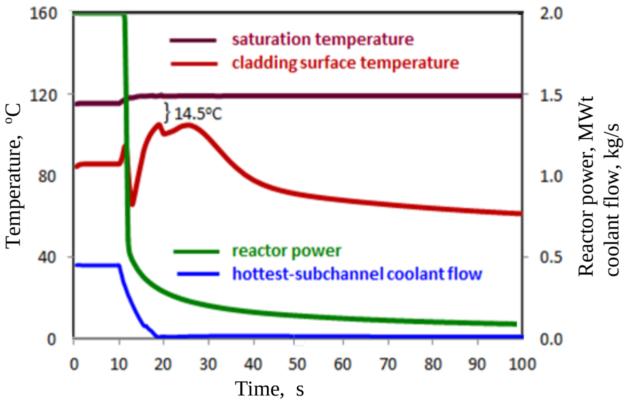

Fig. 5. Hottest channel transient of reactor power, cladding and saturation temperature at CDF 9 s.

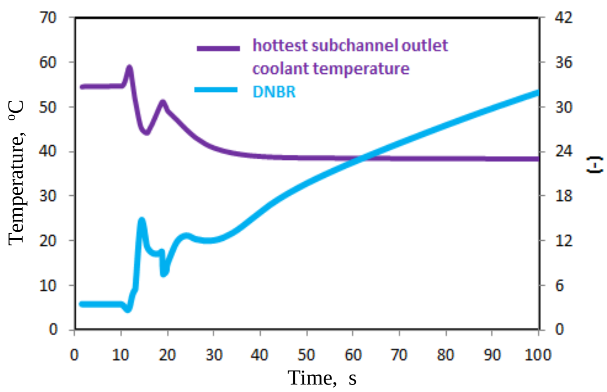

Fig. 6. Hottest channel transient of DNBR and outlet water temperature at CDF 9 s.

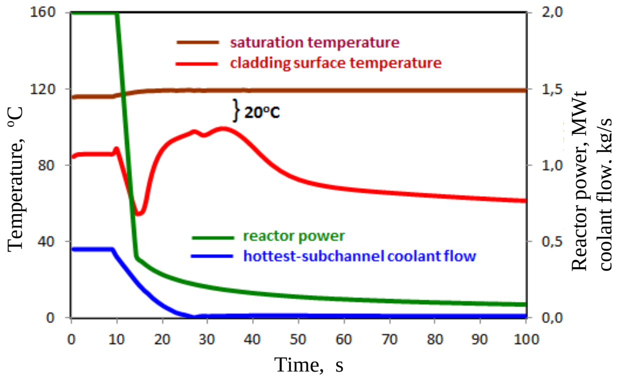

Fig. 7. Hottest channel transient of reactor power, cladding temperature, and saturation temperature at 18 s of CDF.

Figure 7 shows the results of CDF which, at 18 seconds, lasted twice as long as the previous one. In this case, the temperature difference between saturation temperature and the hottest cladding surface temperature increased to 20.0 °C. In order to have a larger inertial force, provision of flywheel in the shaft of primary coolant pump is suggested. Using flywheel, the removal of residual heat by CDF will be improved. Therefore, this simulation of LOFA transient is expected to support the previous study of transition of rod-type fuel elements into the plate-type fuel elements.

CONCLUSION

Simulation results show that the LOFA at reactor power of 2 MW, effective mass flow rate of 63.0 kg/s, and inlet coolant of 35.5 °C does not cause nucleate boiling. The LOFA simulation shows that the reactor is sufficiently safe. The crucial peak temperatures occurs just after coolant pump loss of power and heat removal toward residual heat by coolant flow that decreases rapidly. EUREKA2/RR code has performed the simulation effectively; hence, a general picture of the core thermal-hydraulic during LOFA can be considered. Maintaining a downward core flow at the initial stage of the LOFA condition is very important in a reactor modified to use downward flow. Although the installation of a flywheel in the shaft of the primary coolant pump to increase the inertia force is not performed, the decay heat still can be transferred to the coolant water present in the modified TRIGA-2000 reactor core through natural convection. However, it is suggested to install the flywheel.

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