Steady state and LOCA analysis of Kartini reactor using RELAP5/SCDAP code: The role of passive system

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Abstract. Safety is the priority for nuclear installations, including research reactors. On the other hand, many studies have been done to validate the applicability of nuclear power plant based best estimate computer codes to the research reactor. This study aims to assess the applicability of the RELAP5/SCDAP code to Kartini research reactor. The model development, steady state and transient due to LOCA calculations have been conducted by using RELAP5/SCDAP. The calculation results are compared with available measurements data from Kartini research reactor. The results show that the RELAP5/SCDAP model steady state calculation agrees quite well with the available measurement data. While, in the case of LOCA transient simulations, the model could result in reasonable physical phenomena during the transient showing the characteristics and performances of the reactor against the LOCA transient. The role of siphon breaker hole and natural circulation in the reactor tank as passive system was important to keep reactor in safe condition. It concludes that the RELAP/SCDAP could be use as one of the tool to analyse the thermal-hydraulic safety of Kartini reactor. However, further assessment to improve the model is still needed.

Keywords: research reactor, safety, thermal-hydraulic, LOCA, RELAP5

1. Introduction

A research reactor is an important facility to support the development of nuclear science and technology in a country. That research reactor is in general used for education and training in terms of basic nuclear reactor. A research reactor is also very useful to help the research and development activities. As a neutron source, the research and development in developing new materials could be conducted in a research reactor. Other benefit is that research reactor could be used to produce radio isotopes, which are widely used in industries and medicine. There are 228 type of research reactors in operation in the world today [1]. In Indonesia, there are three research reactors, i.e. two of TRIGA type and one of MTR type. These reactors are all operated by National Nuclear Energy Agency of Indonesia (BATAN).

As other nuclear and radiation facilities, the safety must be assured by the operator. In order to assure the excellent level of safety, the International Atomic Energy Agency (IAEA) has established a fundamental safety principles and specific set of requirements for research reactor [2]. One of the requirements is to conduct safety analysis using mainly deterministic method. In order to comply with this requirement, several thermal-hydraulic analytical codes have been developed. In the past, conservative approach codes were used, but as the need to have more realistic results emerged, best estimate approach codes have been developed. Moreover, most of these codes are developed initially...
for nuclear power plant, but then they are also validated to be used in research reactor, such as RELAP5 code.

Study to validate RELAP5 code for different type of research reactor has been done since couple decades. Hari et al. used the RELAP5/Mod3.2 code to simulate two transient events without scram on a MTR type reactor named HIFAR [3]. RELAP5/MOD3 has also been qualified against the steady state and transient data from MARIA reactor, a pool type reactor with pressurized fuel channels containing concentric multi-tube assemblies of highly enriched uranium [4]. Other studies on the applicability of RELAP5 computer code for MTR type reactor (RSG-GAS and IEA-R1) have been done by Abdelrazek et al [5], Chatzidakis et al [6], Chatzidakis et al [7] and Hainoun et al [8].

Moreover, some works have been performed to study the applicability of the RELAP5 code to TRIGA type research reactors. The study against IPR-R1, a TRIGA Mark-I type has been done by Costa et al [9] and Reis et al [10,11]. While Marcum et al [12] assessed the validity of RELAP5 for OREGON State TRIGA reactor, and Antariksawan et al studied the applicability of RELAP5 for TRIGA-2000, a TRIGA Mark-II type reactor [13,14].

Since each research reactor has unique features and none of the previous research have been performed for all possible events happen in a research reactor, the present study is aimed to study the applicability of the RELAP5 for Kartini research reactor. The reactor is a pool, water cooled reactor of 100 kW thermal power. The present study differs from previous study especially in terms of the model and the experimental data used, which are specific to the Kartini reactor. On the other hand, the current model is the improvement of the model used in previous study [15], especially in core inlet model, the transient model and the comparison with the latest experimental data. The study presents the results of steady state calculation and also the analysis result of a hypothetical accident known as loss-of-coolant accident (LOCA). This kind of accident is considered as the one of design basis accident. It is important to predict the response of the reactor against such accident in order to know the performance of its safety provisions. In addition, this study is also important in the context of the further use of RELAP5 code as an analytical tool to support the operational safety of Kartini reactor.

2. Methodology

2.1. Reactor description

Kartini reactor is similar to TRIGA Mark-II type reactor, a pool type with U-ZrH cylindrical fuel of about 752 mm length. The core is constituted of 68 fuels and 3 control rods placed in annular configuration. Figure 1 shows the core configuration. The vertical cut view of reactor and core configuration is delineated in figure 2. The core is seated in bottom of reactor tank with about 2 m of diameter and about 5 m beneath water surface. The light water serves as moderator and coolant. The core cooling is done by natural circulation.

The hot water flowing upward from the core is pumped through primary piping system to a shell and tube type heat exchanger in about 6.5 kg/s to transfer the heat to the secondary cooling system. The colder water is circulating back to the reactor tank.

2.2. Code used: RELAP5/SCDAP/MOD3.2 code

The RELAP5/SCDAP/MOD3.2 used in this study is developed by Innovative System Software (ISS) based on publically available RELAP5/MOD3.3 code. The code is designed to describe the overall reactor coolant system thermal-hydraulic and core behavior. The code could be used to simulate the normal operating conditions or under design basis or severe accident events. The RELAP5 models calculate the overall thermal-hydraulic response and SCDAP models calculate the core and vessel behavior during normal and accident conditions. It has been validated for a wide range of accident conditions using a variety of experiments and plant data [16].
2.3. Calculation

In this study, the model of Kartini reactor was developed for simulating steady state condition and LOCA transient due to pipe break and beam tube rupture. Nodalization scheme of the core and primary cooling system is described in figure 3.

![Figure 1. Core configuration of Kartini reactor [15].](image1)

![Figure 2. Vertical cut view of Kartini reactor.](image2)

Reactor tank is divided into core and coolant region. The last is further differentiated in several regions: lower plenum (B100), core by-pas (P110), mixing region (P145, P150, P160 and P165) and upper plenum (P170). The core is divided in three region: hot channel, average channel and cold channel. The hot channel represents 6 fuels in the most inner ring (P120 and HS120) while the remaining fuel elements is represented by the average channel. The primary piping system and the heat exchanger are included. The secondary cooling side model is simplified by using time dependent volume as a heat sink. In this model, the reactivity feedback effects are not yet accounted for.

![Figure 3. Nodalization scheme of Kartini reactor.](image3)
As first step, the steady state calculation is done to validate the model. The calculation results are compared with the available measurement data, especially at the nominal power of 100 kW. Once the model is considered providing good steady state results, the transient calculation is conducted. A short transient due to pump trip without reactor scram is simulated. The calculation result is compared with the measurement data obtained during reactor power calibration. To simulate the LOCA, models for pipe break and beam tube rupture are developed. The pipe break is assumed occur in the pipe circulating cold water to the reactor in the lowest part. A guillotine pipe break is considered to simulate the worst case. In the case of a beam tube rupture, the leak from the beam tube with the size equal to the beam tube diameter is assumed.

3. Results and discussion

3.1. Steady state condition

Figure 4 and 5 show the coolant mass flow rate and fuel and coolant temperature at steady state at 100 kW, respectively. The steady state condition could be achieved after about 3000 s. With regard to the fuel temperature and coolant temperature at the upper tank, calculation obtains 126.67 °C and 35.56 °C, respectively. While, the measurement data indicates the fuel temperature 128.3 °C and the upper pool coolant temperature of 35.4 °C [17]. So, the difference for both is only less than 2%.

![Figure 4. Coolant mass flow rate at 100 kW.](image1)

![Figure 5. Fuel and coolant temperature at 100kW.](image2)

On the other hand, no data of mass flow measurement in the pool is available. However, we could see from the calculation results that when the coolant is flowing out from the inlet pipe at the rate of about 6.5 kg/s, the major part of the flow is going directly to the pool (mixing region), only less than one-third of mass flow coolant rate goes to the core. This is because the reactor core is cooled by natural circulation such that the flow to the core is determined by the reactor power.

Figure 6 shows the fuel temperature increases with the power. Figure 7 provides the calculation results on the increase of the coolant mass flow rate to the core with the reactor power from 10 to 100 kW. The instability at the beginning of the mass flow curve in figure 7 is due to numerical instability (steady state is not yet achieved).

The comparison of calculated and measured fuel temperature is delineated in table 1 and figure 8. The differences between the measured and calculated fuel temperatures at various powers are between -1.67% and +10.5%. The difference is higher in the lower temperature. The difference between measurement and calculation, could be caused by error in power measurement which could attained about 9.7%. Another source of error is from temperature measurement which is approximately 1%. Considering that, the calculation results are acceptable.
Figure 6. Fuel temperature variation with power

Figure 7. Coolant flow variation with power

Table 1. Fuel temperature variation with power: measured vs calculation,

| Power (kW) | Temperature (°C) | Difference (%) |
|------------|------------------|----------------|
|            | Calculated       | Measurement    |               |
| 10         | 43.22            | 39.5           | 9.39          |
| 20         | 54.28            | 49.7           | 9.10          |
| 30         | 64.41            | 58.3           | 10.50         |
| 40         | 74.09            | 68.5           | 8.10          |
| 50         | 83.38            | 78.5           | 6.17          |
| 60         | 92.36            | 89.3           | 3.47          |
| 70         | 101.11           | 97.3           | 3.90          |
| 80         | 109.64           | 108.3          | 1.25          |
| 90         | 117.97           | 118.8          | -0.68         |
| 100        | 126.15           | 128.3          | -1.67         |

Figure 8. Measured vs calculated fuel temperature.

3.2. Transient during pump trip steady state

In addition, the comparison is also done during transient following primary pump trip. The scenario is as follows: at the steady state operation at 100 kW, primary pump is stopped, but the reactor is not scammed. During about 60 minutes the coolant temperature at the upper pool is measured once a minute. Figure 9 describes the primary inlet flow to the tank and coolant flows to the core. As in figure 9 it could be shown that after pump trip, the flow to the core is practically unchanged. It is because the flow is dominated by the natural circulation, which in this case the heat generated in the core plays the role pumping the coolant upward. Figure 10 shows the comparison of measured and calculated of the upper pool coolant temperature evolution. The temperature difference of both is quite obvious, about 5 °C. This is probably due to the early measurement of the coolant temperature, i.e. just at a short time after the power reaches 100 kW when the coolant temperature is not yet homogenized. The data from the reactor operation shows that it takes about 7 hours for the coolant temperature, which initially at about 30 °C, to reach steady state at 35 °C. Despite of that difference, the rate of increase of the coolant temperature is identical in both cases, which is about 0.0014 °C/s. It proves that the model of natural convection in the reactor pool is satisfactory.
3.3. LOCA due to a pipe break
It is assumed the lowest part of cold leg experiences double ended break (guillotine break). Following
that, the coolant leaks with the rate of about 12 kg/s at the beginning and consequently, the coolant
level in tank decreases. As the level reduces up to 20 cm below the normal level, the reactor scram and
pump trip are assumed. In the reality, there is a siphon breaker hole in the inlet pipe, but it is not
considered in this simulation to obtain the worst case. Figure 11 depicts the flow of primary coolant,
leaks and coolant level in tank. The break starts at 500 s of calculation time. At 30 s later, the coolant
level is down by 20 cm and it triggers reactor scram and pump trip. The leak continues until 1450 s
before stopping when the level reaches about 1.1 m from the bottom of the core or at about the level of
the tip of inlet pipe. The leak flow stops because the siphon affect is interrupted. Following to that,
the decay heat is removed by natural convection in the remaining coolant. The fuel temperature in hot
channel and coolant temperature in various regions in the tank are shown in Figure 12. It could be seen
that up to the end of calculation time, the temperatures are stabilize at about 34 °C.

In the field, a siphon breaker hole is added at the inlet pipe. Its location is about 40 cm below the
tank's tip. This is a safety facility that is used to prevent water efflux in case of pipe break [18,19]. In
such case the coolant loss is reduced and also avoid the core from being uncovered.
3.4. LOCA due to a beam tube

A hypothetical one beam tube rupture accident is simulated to see the characteristic of Kartini reactor following that accident. Figure 13 shows that following the beam tube rupture, coolant leaks from the ruptured beam tube. It causes the coolant level decreases fast and 750 s after the initiation of the leak, the coolant reaches the final level at 0.35 m or at the level of the beam tube. At that time, the leak flowing through the beam tube practically stops. In this scenario of the accident, the reactor scram and pump trip are determined to occur when the coolant level decreases by 20 cm from the normal level. The decay heat remains after reactor scram.

As in figure 14, even the coolant level reaches at mid of the core and upper half of fuel element is uncovered, the remaining coolant, vapor and air could cool the fuel element. In this figure, the fuel temperature, which represents the mid-height of fuel, decreases following the decay heat, but then increases before decreasing again and attains about 50 °C at about 20,000 s after the initiation of the accident. The fuel temperature increases because the coolant cooling capacity decreases, but when the decay heat continues to decrease and the vapor and air could remove the decay heat, the fuel temperature decreases again. The remaining coolant at the bottom part of the core increases slightly and stabilizes at about 40 °C. Other study investigated extreme loss of coolant due to bottom of tank rupture causing complete core uncovered. It showed the fuel temperature increased quickly to more than 1000 °C in about 50 s [20].

![Figure 13. Break flow and coolant level during LOCA due to beam tube rupture.](image1)

![Figure 14. Fuel and coolant temperature during LOCA due to beam tube rupture.](image2)

4. Conclusion

A RELAP5/SCDAP model for Kartini research reactor has been developed and used to simulate the steady state operational and transient events. The steady state simulation showed that the RELAP5/SCDAP provided reasonable results when compared with available measurement data. The model could also predict the reactor characteristics following a hypothetical LOCA events. The reactor is predicted that it could keep its good safety performance against LOCA due to worse pipe break and beam tube rupture. The simulation is also shown that the passive system and phenomenon, i.e. siphon breaker hole and natural circulation inside the tank, plays an important role to keep the safety of the reactor during loss of coolant events. The improvement is still needed, especially by extending the comparison with the measurement data. The integration of the reactivity feedback effect would also improve the model. It is more needed to simulate the reactivity insertion accident type which are commonly hypothesized in a research reactor.

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