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TRIGA-2000 RESEARCH REACTOR THERMAL-HYDRAULIC ANALYSIS USING RELAP/SCDAPSIM/MOD3.4

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ABSTRACT

Any events presumed to risk the safety of a nuclear reactor should be analyzed. In a research reactor, the applicability of best estimate thermal-hydraulic codes has been assessed for safety analysis purposes. In this paper, the applicability of the RELAP/SCDAPSIM/MOD3.4 thermal-hydraulic code to one Indonesian research reactor, which is named TRIGA-2000, is performed. The aim is to validate the model and use the model to analyze the thermal-hydraulic characteristics of TRIGA-2000 for main transient events considered in the Safety Analysis Report. The validation was done by comparing the calculation results with experimental data mainly in steady state conditions. The comparison of calculation results with the measurement data showed good agreement with little discrepancies. Based on these results, simulations for thermal-hydraulic analyses were performed for loss of coolant transients. The calculation results also properly depicted the physic of the thermal-hydraulic phenomena following the loss of coolant transients. These results showed the adequacy of the model. It could be shown that the engineered safety features of TRIGA-2000 play an important role in keeping the reactor safe from the risk of postulated loss of a coolant accident.

Keywords: Loss of coolant; Loss of flow; RELAP5; Research reactor; TRIGA

1. INTRODUCTION

Indonesia electricity demand tends to increase about 8% to 9% annually (Berawi et al., 2016). To fulfil that demand, Indonesia government has decided to constructs new power plants with the total capacity of 35,000 MW. The construction of power plants should consider the location and type of power plant. Regarding the type of power plant, considering significant increase of carbon dioxide concentration in the atmosphere, the use of renewable energy is one way to limit the concentration of carbon dioxide (Setiawan & Asvial, 2016). In the Indonesia's national energy plan, until 2050, the use of new and renewable energy in energy mixed is planned to increase considerably, i.e. up to 31% (Republic of Indonesia, 2014). One of the new energy sources option is nuclear energy. It is generally agreed that nuclear power plants can serve to avoid greenhouse gas (GHG) emissions, such as carbon dioxide (Besmann, 2010). As the world practice, the first step to master the nuclear energy is to build a nuclear research reactor.

A research reactor is used primarily as a neutron source for research and other purposes, but not

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for power generation. The International Atomic Energy Agency (IAEA) database shows that 244 research reactors are in operation in the world today (IAEA, 2016). Among these, the TRIGA (Training, Research, Isotopes, General Atomics) type is widely used. General Atomic has installed 66 TRIGA reactors in 24 countries (General Atomic, 2017).

The National Nuclear Energy Agency of Indonesia (BATAN) operates three research reactors: two TRIGA Mark-II type reactors (named TRIGA-2000 and Kartini) as well as one MTR (Material Testing Reactor) type reactor (named RSG-GAS). Initially, TRIGA-2000 was operated at 250 kW, but in the year 2000 the reactor was upgraded to 2000 kW. TRIGA is designed with a high degree of inherent safety features. However, any possibility of occurrences that risk the safety of the reactor should be considered during the design and operation. Moreover, after the Fukushima Dai-ichi Nuclear Power Plant accident, an appropriate amount of attention needs to be given to the operator of the nuclear power plant in order to perform a safety reassessment of the reactor (IAEA, 2015).

Safety analysis of a nuclear reactor was previously performed based on conservative modeling assumptions. To provide more realistic analysis results, that approach has been replaced by the so-called best-estimate methodology. Over the last four decades, advanced computational codes that are based on the best-estimate approach have been developed, such as RELAP5. Many of these computational tools have been developed for power reactors but then adopted for research reactors. For that purpose, a benchmark against experimental data and thermal-hydraulic computational methods has been conducted (Hainoun et al., 2014).

RELAP5 computer code was originally a light water reactor thermal-hydraulic analysis code developed by the IDAHO National Engineering Laboratory (INEL) for the United States Nuclear Regulatory Commission (USNRC) (Fletcher & Schultz, 1995). The code has been used to model and simulate transient analysis for different scenarios of accidents in a Pressurized Water Reactor (PWR) or a Boiling Water Reactor (BWR). Meanwhile, Innovative System Software (ISS) developed RELAP/SCDAPSIM. The code utilizes publicly available RELAP/MOD3.3 and SCDAP/RELAP5/MOD3.2 models (Antariksawan et al., 2005). In this code, RELAP5 models calculate the overall thermal-hydraulic response, and SCDAP models calculate the core and vessel behavior during normal and accident conditions. The RELAP5 has been validated for a wide range of reactor types and accident conditions using a variety of experiments and plant data, including TMI-2 (Allison & Hohorst, 2010). RELAP5 has also been used for experimental analyses other than in nuclear reactor facilities (Kusuma et al., 2017). As in the other field of technology, the use of a computer code is increasing in the safety analysis. The reason for increasing use of computer codes is their ability to simulate complex phenomena into efficient, effective and coherent synthetic environment (Berawi, 2013).

Furthermore, some works have been performed to study the applicability of the RELAP5 code for different types of research reactors. Simulation of two transient events without scram based on the HIFAR reactor have been done by using RELAP5/MOD3.2 (Hari et al., 2000). A modification was proposed to account for the critical heat flux (CHF) in annular fuel geometry. The RELAP/SCDAPSIM/MOD3.4 model has been developed for several research reactor types (TRIGA, LVR-15 and CARR), and the preliminary nodalization for the qualification of the input model has been described in (Antariksawan et al., 2005). A study on steady state and flow transient in TRIGA-2000 had been performed using RELAP/SCDAPSIM/MOD3.4 (Antariksawan, 2006). Some discrepancies during the comparison of calculation results and measurements have been indicated, which were caused by the limitations of the model. Another work was carried out to assess the RELAP5 model for the University of Massachusetts Lowel Research Reactor, an open-pool reactor using light water as moderator and coolant, and with plate-type fuel (Bousbia-Salah et al., 2006). It concluded that the RELAP5 model provided good agreement for the steady state operational condition, pool heat up, and reactivity insertion transient considered in the study. An assessment of the RELAP5 model has also been performed for the Oregon State TRIGA Reactor (Marcum et al., 2010). The result showed that an increase of the channel number model gave the closest results to the measurements. Another study was conducted in order to use the RELAP5 model to analyze the IPR-R1 TRIGA research reactor's thermal-hydraulic characteristics during a steady state and loss of flow event (Costa et al., 2010). The model showed good agreement with the experimental data for the 50 kW steady state, but it could not match the data of pool heat up during the loss of flow transient condition. A better result for the transient condition was obtained when the same model was modified by dividing the reactor pool into two regions and adding a model of cross flow between both regions (Reis et al., 2010). The number of channels in the core was also found to be important to refine the calculation results (Reis et al., 2012). Other studies assessed several thermal-hydraulic codes, including RELAP5, against an MTR-type reactor for a steady state condition, loss of flow, and loss of coolant transient (Abdelrazek et al., 2014; Chatzidakis et al., 2014; Chatzidakis et al., 2013; Hedayat et al., 2007; Karimpour & Esteki, 2015). In those studies, the RELAP5 code was generally able to simulate the steady state condition with good agreement but with a higher discrepancy during the transient considered in the study.

Considering those previous research studies, it is obvious that validation of the RELAP5 code against the research reactor, especially for a TRIGA-type reactor, is still rare. The TRIGA-type reactor operates at various thermal power levels (from less than 0.1 to 16 MW), and each one could have specific features depending on its objectives. As a result, the validation of the RELAP5 model for the TRIGA reactors is still necessary prior to being used as a safety analysis tool. The present work is on the development of preceding works; it consists of improving the model (especially the inlet core and cross flow model), gap conductance model, and the addition of the experimental data for validation. In addition, the current research will focus on loss of coolant transients. The objective is to validate the improved model and to obtain the thermal-hydraulic characteristics of the reactor following the loss of coolant transients. The model will first be validated for a steady state condition and specific transients by comparing the calculation results with measurement data and/or other calculations from validated codes. Once the model is qualified, it will then be used to predict the responses of the reactor during the loss of coolant transients. The loss of coolant transients are important for the safety consideration as described in TRIGA-2000's Safety Analysis Reports (SAR) and are not well analyzed using RELAP5 in the previous studies. The results from this study are also expected to contribute to the validation of thermal-hydraulic codes for research reactors especially for TRIGA-type reactors, and also to support the TRIGA-2000 reactor re-operation after long periods of being shut down for maintenance.

2. METHODOLOGY

2.1. Description of TRIGA-2000

TRIGA-2000 is a pool-type reactor with 2000 kW of thermal power. The core is placed at the bottom of the reactor tank at a depth of about 6 m. A 28 cm-thick graphite ring surrounds the core. Figure 1 and Figure 2 show the vertical cut-view of the reactor and core configuration, respectively (Antariksawan, 2006; Antariksawan et al., 2005). The reactor is fueled with low enriched uranium in the form of U-ZrHx. The fuel is in cylindrical geometry with an outer diameter of 3.75 cm and a total length of 72 cm.

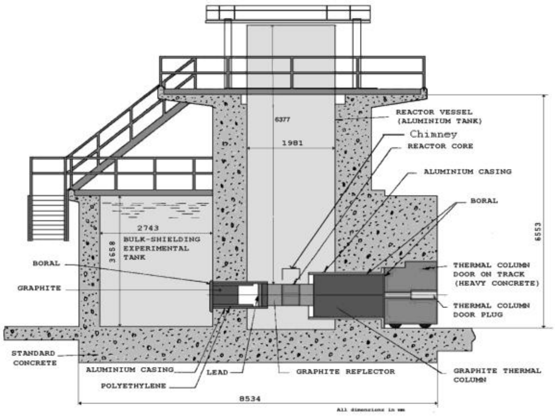

Figure 1 TRIGA-2000 reactor vertical cut view

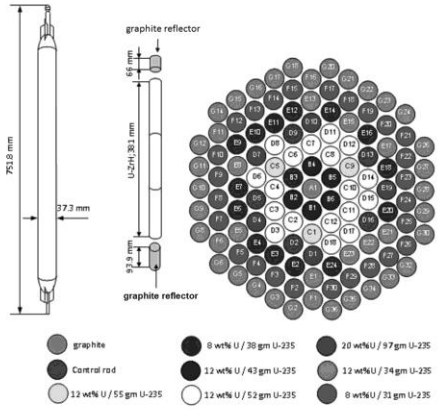

Figure 2 Core and fuel element

The primary cooling system is diagrammed in Figure 3, and the main characteristics are given in Table 1 (BATAN, 2001). The reactor core is cooled by light water. During normal operation, the reactor coolant is circulated by one centrifugal pump. Though the main cooling of the reactor core is assured by the natural circulation, the forced convection coming from the primary cooling system affects the flow pattern along the reactor core (Umar et al., 2009).

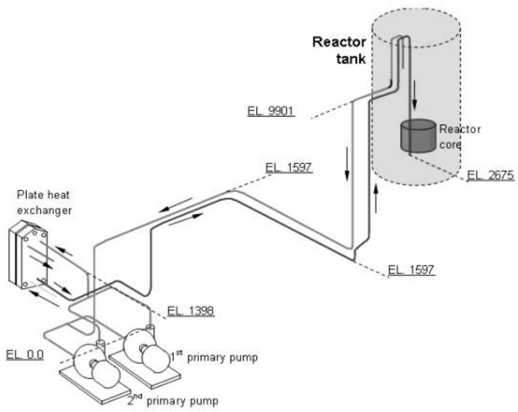

Figure 3 Primary cooling diagram

Table 1 Main characteristics

Component	Characteristic
Primary cooling	$\dot{m}_{max}$: 950 gpm (50.5kg/s) T_{in} : 42.2^{o}C T_{out} : 32.2^{o}C
Secondary side cooling	$\dot{m}$: 1200 gpm (86.2 kg/s) T_{out} : 37^{o}C T_{in} : 29^{o}C
Pipes	Material : Aluminum Inner diameter : 14.24 cm Thickness : 0.5 cm
Fuel element	Composition : U-ZrHx Enrichment : 19.75%± 0.2% Total length : 72.0cm Diameter : 37.5cm

TRIGA-2000's reactor protection system consists of some limiting operation conditions and engineered safety features. The reactor scram occurs with 0.5 s of delay from the initiation of the scram signal. To mitigate the postulated loss of coolant, an Emergency Core Cooling System (ECCS) was installed. It is a 15.5 m^3 open water tank located at about a 5.3 m height above the reactor. The water is poured gravitationally to the reactor tank when the ECCS trip valve is opened. The ECCS is automatically actuated when the coolant level is 5 m below normal with a delay of 6 s. TRIGA-2000 also has a simple passive safety system that is called siphon breaker holes, which are located at both the coolant inlet and outlet pipes. These holes are about 12.7 mm in diameter, and they are about 80 cm below the normal coolant level in the reactor tank.

2.2. Nodalization

This study comprises steady state and transient calculations. The steady state calculation is performed for validating the RELAP5 model of TRIGA-2000. The model of TRIGA-2000 is composed of a reactor tank and a primary coolant piping system, including a plate-type heat exchanger. The secondary system is not modelled as a full circuit but only to provide a boundary conditions using time-dependent volume. Figure 4a shows the nodalization of the system (Antariksawan, 2006). The model of the fuel element consists of a fuel where the heat is generated, gap, and cladding—as shown in Figure 4b.

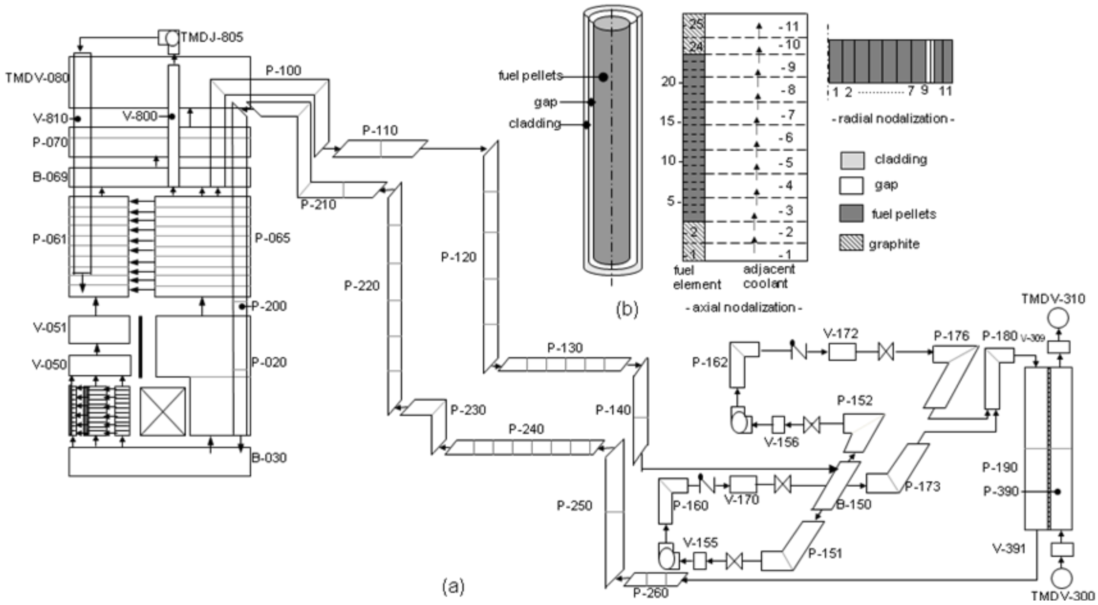

Figure 4 Model and nodalization of: (a) the cooling system; and (b) the fuel element of TRIGA-2000

Several steady state and transient data measurements from TRIGA-2000's operations using different thermal power were used; relevant data generated by other analytical model are also applied in the validation process. After the validation process (in which the model is confirmed to be feasible and applicable), the results from the steady state simulation are used to perform the transient analysis. Those transient calculations are aimed to assess the thermal-hydraulic characteristics of the reactor following several transient events that are considered in the safety analysis.

3. RESULTS AND DISCUSSION

3.1. Steady State

To validate the RELAP/SCDAPSIM steady state model for TRIGA-2000, the operation data of TRIGA-2000 at a nominal power of 2000 kW were used. Figure 5 shows the variation in reactor power and fuel center line temperature during the reactor's start-up. The reactor's power was increased gradually (as determined by the operator) to reach the nominal steady state condition at 2000 kW. It could be seen that the trend of calculated power and fuel temperature are similar with those of the reactor operation measurement. The steady state values of the fuel temperature at 2000 kW are also in good agreement.

However, Figure 6 shows the variation of coolant mass flow rate in the hot channel, the core by-pass predicted by RELAP/SCDAPSIM during start-up, and the total coolant mass flow rate. When the power and temperature of the fuel increased during start-up, the flow into the hot channel also increased, while the core by-pass flow rate decreased. This indicates that the core cooling is dominated by natural circulation. Therefore, the increase of the temperature in the fuel increases the temperature of the coolant in the core, and it augments the buoyancy force, causing the flow to the core to increase. As shown in Figure 6, the model predicted a coolant mass flow rate of about 0.12 kg/s in the hot channel. This result is similar to the calculation using CFD in a previous study (Umar et al., 2003).

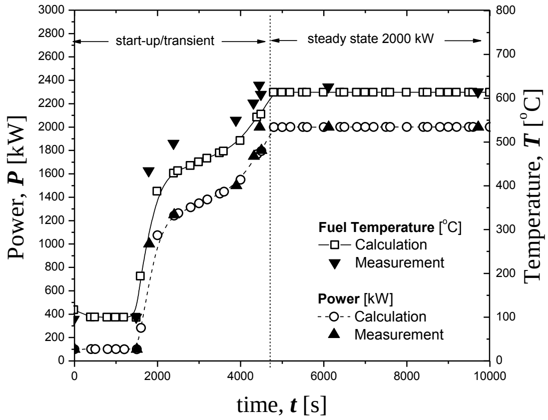

Figure 5 Power and fuel temperature during start-up

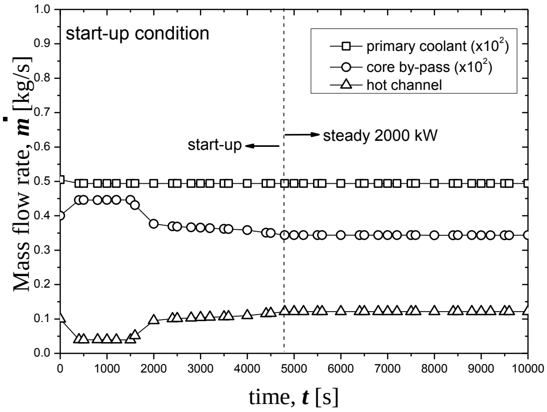

Figure 6 Coolant mass flow rate during start-up

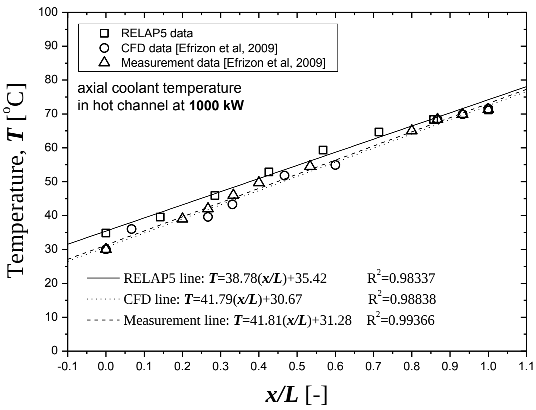

Figure 7 Measured and calculated axial coolant temperature in the hot channel at 1000 kW

Table 2 Main results at steady state 2000 kW

Parameters	Measurement	SCDAP/ RELAPSIM	Diff (%)*
T_{out} (°C)	46.90	47.37	0.8
T_{in}(°C)	37.30	37.73	1.1
ΔT (°C)	9.64	9.64	0.0
2nd Tout (°C)	37.60	37.44	0.4
T_{fuel} (°C)	615.00	613.78	0.2

*Diff (%) = $\frac{|calculation - measurement|}{measurement} \times 100$

In addition, the model was compared with the transient data from measurement during a calorimetric power calibration technique at 300 kW. The reactor was operated at a steady state of 300 kW, and the primary circulation pump was then shut down. The temperature of the coolant in the tank was measured in some locations. Figure 8 shows the variation of the coolant temperature in the reactor tank at one point of measurement, which was about a 1 m depth for 1 hour following the pump trip—both from the measurement and calculation. It could be seen that the model could predict the variation of the temperature (as in the measurement), which increased at the rate of about 3.15°C/hour with a difference between -4.43% and +3.72%, and with an average deviation of 0.089%. This result and the above steady state comparison confirmed that the model is feasible and applicable to be used for transient analysis.

3.2. Loss of Coolant

3.2.1. Cold leg rupture

Figure 9 shows the evolution of the mass flow rate of coolant leakage as well as the coolant tank level. The steady calculation is done up to 500 s, at which the transient started. The primary coolant began leaking from the pipe, both from the upstream and downstream sides of the guillotine break point. The leak caused the water level in the tank to decrease. After only about 12 s from the beginning of the leak, the water level reached the operational limit at about 0.5 m below the normal level. At that time, the reactor scram occurred, and the pump was tripped. The leak flow terminated about 50 s after the break, and the water level in the tank decreased approximately 1 m from the initial level. Following that, the core cooling was done by convective natural circulation. As shown in Figure 10, the reactor core could be cooled by that natural circulation, and the fuel temperature could be maintained as low without significantly increasing the coolant temperature. The occurrence of natural circulation in the reactor pool was also found in another study (Chatzidakis et al., 2013).

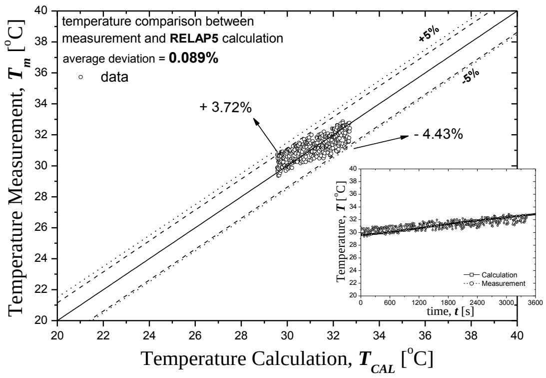

Figure 8 Coolant temperature variation with time after the pump trip at 300 kW

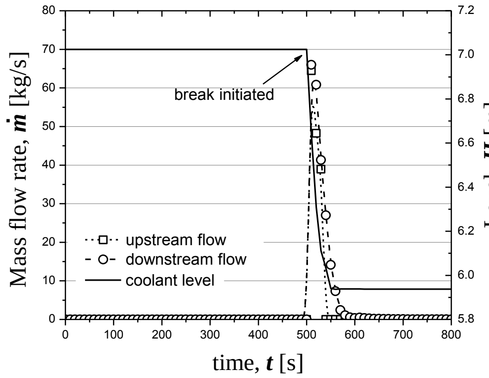

Figure 9 Mass flow rate of coolant leakage and water level in the tank during pipe break

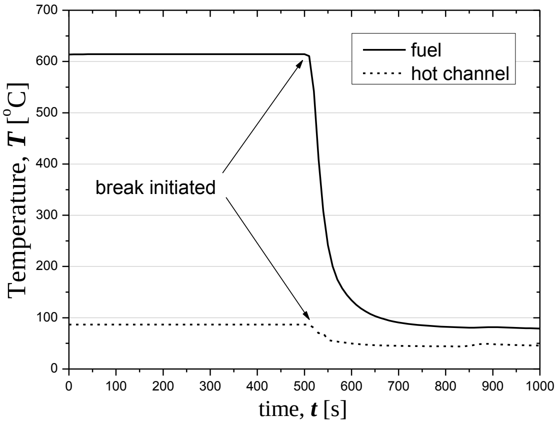

Figure 10 Fuel and coolant temperature evolution during the pipe break transient

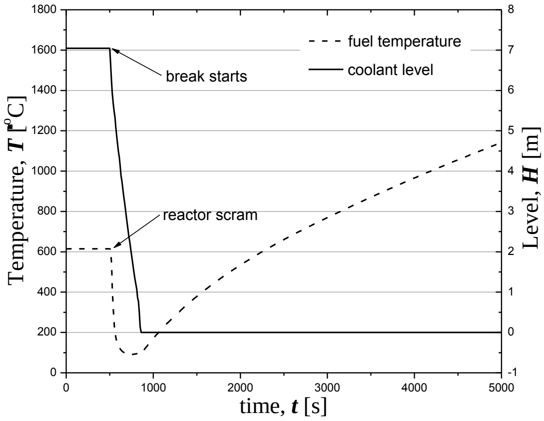

Figure 11 Water level and fuel temperature transient during the pipe break transient

From this result, it could be seen that the siphon breaker holes were very important because they stopped the leakage. The leakage termination occurred when the water in the tank reached the holes and the penetrated air blocked the water flow. However, the above result predicted that there was an undershooting level of water. The level of the water was about 20 cm below the siphon breaker holes. This was because the inertia of the leak flow was relatively high, while the size of the holes was relatively small, causing the small penetrated air flow rate (Kang et al., 2013). In contrast, if there are no siphon breaker holes, the consequences to the pipe break would be the worst. Figure 11 shows the simulation results of such a case. In the absence of the siphon breaker holes, the water leak could occur until the water level reaches the tip-end of the cold leg, which is at the level of the core's lower grid. That occurred in only about 360 s. It would make the core totally uncovered, and, if the ECCS is unavailable, the fuel temperature would increase above the safety limit because the core's heat removal is degraded.

3.2.2. Beam tube rupture

Figure 12 shows mass flow leakage, ECCS flow, and coolant level transient in the reactor tank during the beam tube rupture. In that figure, t = 0 s was the start of the calculation, and at t = 500 s the beam tube rupture occurred. About 10 s after the rupture, the reactor scram and pump trip occurred because of the low level trip signal. The reactor power decreased to the decay heat. The coolant level continued to descend; it reached the level of ECCS initiation at about 185 s and the beam tube level at about 300 s. At that time, half the top of the fuel was uncovered. When ECCS initiated, the coolant level continued to decrease because the ECCS flow rate was smaller than the drained out water flow. However, the poured water from ECCS could help to cool the core, so the temperature of fuel was kept below 100°C. In contrast, if ECCS fails to function, the uncovered part of the fuel would only be cooled by air and steam. This results in the decrease of the heat removal capacity, and it causes a continuous increase of the fuel temperature as shown in Figure 13. The maximum fuel temperature limit of 950 °C was exceeded about 5000 s after rupture. These results show the important role of ECCS, and the existing ECCS is predicted to be adequate.

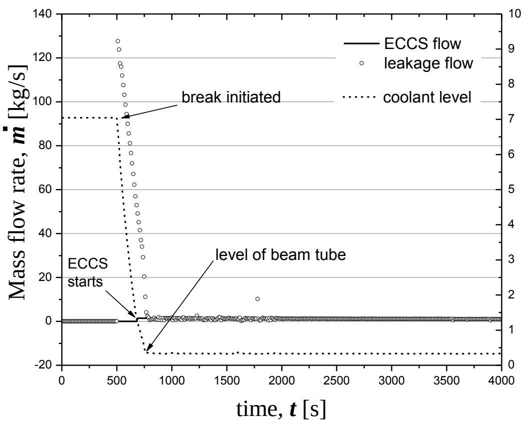

Figure 12 Leakage and ECCS mass flow rate and water level in the tank during beam tube break

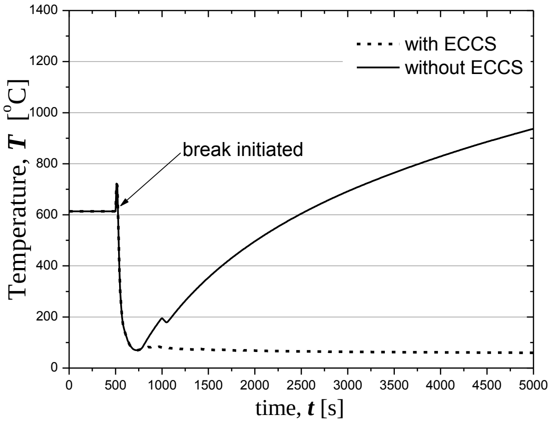

Figure 13 The hottest fuel temperature evolution during beam tube break with and without ECCS

4. CONCLUSION

The RELAP/SCDAP/MOD3.4 has been used to calculate the steady state condition and loss of coolant transient of the TRIGA-2000 reactor. The comparison of the calculation results of several operational parameters with existing measurement data for the steady state condition showed good agreement. The difference was less than 1.2% for some of the values being compared. Then, the model was qualified to simulate the reactor system in a steady state condition, and it could be used to simulate the transient events.

Based on the transient simulation results, it could be shown that the model provides fairly good results for describing reasonable characteristics of thermal-hydraulic phenomena during the transient. The simulation has shown the significant role of natural circulation inside the tank, the siphon breaker holes, and ECCS for keeping the reactor safe in case of the loss of coolant following a coolant pipe break or a beam tube rupture.

Further validation with experimental data and sensitivity analysis of several important parameters, especially for transient events, should be considered to improve the validity of the model.

5. NOMENCLATURE

h	height (m)		*Subscript*	
$\dot{m}$	mass flow rate		max	maximum
L	length (m)		in	inlet
P	power (kW)		out	outlet
t	time (s)		*Acronyms*	
T	temperature (°C)		ECCS	Emergency Core Cooling System
x	distance (m)		OLC	Operating and Limit Condition
ΔT	temperature difference (°C)		SAR	Safety Analysis Report

6. ACKNOWLEDGEMENT

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