KnE Energy

ICoNETS Conference Proceedings
International Conference on Nuclear Energy Technologies and Sciences (2015), Volume 2016

Conference Paper

Loss of Flow Event Analysis of the RRI-50 Conceptual Design

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Abstract

The conceptual design of Indonesia's Research Reactor (RRI-50) core has reached the design optimization phase. RRI-50 is designed to produce the maximum neutron flux of 1.0×10^{15} n/cm²s and thermal neutron flux of $5,0\times10^{14}$ n/cm²s. The analysis of design basis accident should have anticipated any possible accidents, one of which is loss of flow event that should be met according to the IAEA safety standard. There are many conditions that cause loss of flow accident, such as pump failure. This series of event may trigger an accident due to loss of coolant. Therefore, the event should be deeply analyzed. Loss of flow event analysis has been carried out using PARET-ANL code. The margin toward flow instability and two-phase flow becomes a design limit. The analysis was performed during initial period of event. The reactor scram occurred at 2.1 second because flow rate trip reaches 85% from its nominal value. Based on loss of flow event of the design basis accident of RRI-50 core, it can be concluded that the margin toward flow instability at the early transient, which is the worst condition, has not generated two-phase flow at a constant pressure.

Keywords: Conceptual design, RRI-50, loss of flow event, simulation, code

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Received: 29 July 2016
Accepted: 21 August 2016
Published: 21 September 2016

Publishing services provided by Knowledge E

1. Introduction

The conceptual design of Indonesia's Research Reactor (RRI-50) has been at the design optimization stage. The RRI-50 is designed to achieve the average neutron flux as high as 1.0×10^{15} n/cm²s and thermal neutron flux of 5.0×10^{14} n/cm²s, while it is expected to operate at the lowest possible power. The reactor core components are built in 5×5 array configuration of 16 fuel elements and 4 fuel rods and 5 irradiation positions. This reactor is water (H_2O) cooled with heavy water (D_2O) moderator. The fuel used is uranium molybdenum with high uranium content (7 – 9 g U/cm³). Its core and power is in accordance with the User Criteria Document requirements and will generate high heat generation [1]. To support the design, the reactor center should be designed as compact as possible. Consequently, heat flux density at each fuel element becomes high and creates a challenge to be overcome in term of its heat removal.

In the previous research, core cooling system has been designed to indicate that heat flux generated is 590 W/cm³ [1]. To prevent boiling, reactor core should be located in a pressured vessel and coolant flow rate in the parallel fuel channels is estimated high. The calculation results obtained from the previous study indicated that design criteria of DNBR≥2, ΔT_{ONB}, and V ≤2/3 of critical flow rate not exceeded for P_{inlet} = 8 kgf/cm² with flow rate of 900 kg/s [2]. Sufficient coolant design should be completed with Design Basic Accident (DBA) of

How to cite this article: Endiah Puji Hastuti, Surip Widodo, "Loss of Flow Event Analysis of the RRI-50 Conceptual Design," *KnE Energy*, vol. 2016, 10 pages. DOI 10.18502/ken.v1i1.461.

the primary coolant ensuring fuel safety in case of loss of coolant flow. To facilitate neutronic design and thermal hydraulic design, calculation of thermal hydraulic design has been carried out. As thermal hydraulic safety margin defines reactor power operation, reactor power will be in proportion with thermal neutron flux obtained.

Conceptual design of RRI-50 is considered as a research reactor with high power. This reactor concept has some similarities with China Advanced Research Reactor (CARR). The characteristics of CARR reactor have been analyzed by researchers through computer modeling. Tian et al. (2005) calculated theoretically reactor core thermal hydraulic characteristics of CARR at steady and transient state [3], while Qing Lu et al. (2009) performed a similar analysis using THAC-PRR code developed in Visual Fortran 6.5 [4]. Meanwhile, Ronghua Chen et al. (2011) carried out accidental analysis on CARR due to reactivity insertion, caused by unexpected control rod withdrawal in full power condition, using TSAC1.0 code [5]. To study the CARR thermal hydraulic safety characteristics at accident condition due to station blackout, Wenxi Tian et al. (2007) conducted an accidental analysis using TSACC code developed [6]. The use of computer codes for calculation such as PARET-ANL, RELAP-5 et cetera has proved that these codes can be used to analyze thermal hydraulic safety at steady state and transient state as carried out by Hamidouche (2009), Omar S. AL-Yahia et al. (2013), and Chatzidakis (2014) [7-9].

One of the engineered safety features (ESF) in the research reactor is residual emergency core cooling system (RECCS). The main function of RECCS is to remove heat from the core after reactor scrams and the primary cooling system does not operate, while core cooling by natural circulation is not possible. For an open pool reactor, there is an alternative design to avoid the use of such system. Passive system, such as coolant flow, fly-wheel inertia, flap valve and core chimney, is sufficient to overcome black out event. This system is not available for a research reactor with a pressure vessel in the pool, in which RCCS is provided and supplied by an onsite emergency power supply to overcome black out event [10].

The RRI-50 fuel is plate type, which is cooled by primary coolant flowing through parallel gaps and supplied by primary coolant pump. When the primary pumps inadvertently shutdowns, the core should shutdown. When the core shutdown, there is residual heat that will decreases with time. To remove this residual heat, the primary coolant pump should be designed to be able to flow the coolant for residual heat removal. The fly wheel in the coolant pump should have enough time to continuously flow the coolant. In addition, there are many cases, including pump failure, that cause loss of coolant flow. These events can initiate an accident due to loss of coolant. Therefore, they should be evaluated in depth. Loss of coolant flow event is one DBA that should be analyzed to meet IAEA safety standards. The RRI-50 primary coolant safety design should be able to anticipate black out event, in which no power is available to run the primary cooling pump, causing loss of coolant flow. The pump design should anticipate any pump failures, in which reactor core cooling safety by the primary coolant system should be ensured. Based on the previous analysis on high power research reactor such as CARR, and analysis results obtained from computation code used in the accident analysis at high power research reactor, this paper is aimed to describe an analysis on thermal hydraulic characteristics of RRI-50 core due to loss of coolant flow event during early accident, using PARET-ANL code.

2. Theory

The thermal hydraulic condition in the reactor coolant channel during transient can be outlined using mass, momentum, and energy conservation laws. The fundamental laws of mass, momentum, and energy conservation for irregular shape with constant control volume V, and surface area S, are as follows (Liepmann, 1957; Delhaye, 1981) [8]:

$$\frac{\partial}{\partial t}\left[\int_{v}^{\square} \rho dv\right]+\oint_{s} r\vec{u}.\,d\vec{s}=0 \tag{1}$$

$$\frac{\partial}{\partial t}\left[\int_{v} (\rho\vec{u})dv\right]+\oint_{s} (\rho\vec{u})\vec{u}.\,d\vec{S}=\oint_{s} (\vec{p})ds+\oint_{v} \rho\vec{\psi}\,dv \tag{2}$$

$$\frac{\partial}{\partial t}\left[\int_{v} (\rho e)dV\right]+\oint_{s} (\rho e)\vec{u}.\,d\vec{S}=-\oint_{s} \vec{\varphi}.d\vec{S}+\oint_{v} q'' dV+\oint_{s} (\vec{u}.\vec{p})d\vec{S}+\int_{v}^{\square} \rho\vec{u}.\vec{\varphi}V \tag{3}$$

where:

ρ	:	Volume weight density for two phase coolant, (kg/m^3)
$\vec{u}$	:	Fluid flow vector, (kg/s.m^2)
p	:	Pressure, (Pa)
$\vec{\varphi}$	:	Heat flux vector, (MW/m^2)
$\vec{\psi}$	:	Body force vector per mass unit, (Pa)
e	:	Internal energy per mass unit, including intrinsic heat (H-P/ρ); and kinetic energy½ u^2, MW.s.
q″	:	Internal volumetric heat generation rate, (MW/m^3)

Simulation of loss of coolant flow is carried out using PARET-ANL code. Before transient calculation is performed, steady state condition should be identified first where it is assumed that the reactor has been in operation for 24 hours so that equilibrium condition is achieved. Distribution of axial and radial power peak factor employs the calculation results of neutronic group, UMo-Al fuel with fuel loading of 360 gr U/cm^3. The axial power peak is divided into 21 nodes, where the distribution of axial and radial power peak has considered nuclear and engineering uncertainty.

Coolant System Design

Based on the generated power, generally, research reactor can be classified into three groups: low power, medium power, and high power research reactor. Low power research reactor has power density of less than 100 kW/liter. Medium research reactor operates above 10 MW, typically about 30 – 40 MW until 70 MW with power density over 100 kW/liter. High power research reactor group is based on its core power and compactness in term of high power density and uses vessel designed in a pool. For its safety, a high power research reactor requires forced coolant flow for several hours after reactor shutdown. To ensure coolant flow availability, battery or diesel generator is needed to provide power supply to operate the emergency coolant pump. After such period, the reactor is in safe condition as long as all fuels are under the water.

RRI-50 operates with forced convection cooling mode. In shutdown operation mode where there is no more fission reaction, the fuel should always be cooled using forced convection. In a loss of coolant flow event, the residual heat is cooled using an emergency core cooling system (ECCS). The thermal hydraulic parameters in this mode are defined through steady state and transient analysis. The objectives of the analysis are to obtain reactor core responses during the postulated severe operational event or accident and to determine safety condition that can be selected at steady state and transient state. The conceptual design of RRI-50 cooling system is to remove heat generated by the fuel with nominal power of 50 MW. To prevent boiling, RRI-50 core is designed at relatively high pressure with operational pressure at 8 kgf/cm^2. This system should be maintained to make the core operate at any condition. To ensure the operation of forced cooling system during normal operation, the primary cooling is equipped with 3 centrifugal pumps (two pumps operate and one pump is in standby mode). These pumps flow the primary coolant downward into reactor core through hot leg to the delay chamber. The design of the delay chamber is to decay N-16 radionuclide to safe level. The primary coolant from the delay chamber then transfers its heat to a heat exchanger and returns to the reactor pool through a distributor ring. Coolant distributor ring is installed at reactor vessel wall, 1.8 m above the core. From the distributor ring, water is sprayed to the reactor core. To ensure sufficient coolant during heat removal due to loss of flow transient, the pump is equipped with flywheel. In addition, ECCS should be able to operate automatically as decay heat is in large amount.

Postulated Initiating Events

The postulated event that might occur in case of loss of pump power supply, which is the main component in the primary cooling system, is categorized as a decrease in heat removal accident. The potential accidents that might occur are tabulated in Table 1.

Table 1: Potential accidents on primary coolant system [8].

No.	Potential Accident
1.	Coolant reduction due to valve failure.
2.	Leakage of the primary coolant boundary beyond the isolation valves.
3.	Leakage of the primary coolant pipe between reactor pool and primary system isolation.
4.	Leakage of the primary coolant pipe inside the reactor pool.
5.	Reactor pool leakage.
6.	Leakage of the warm water layer system.
7.	Leakage of the fuel element store purification system.
8.	Heat exchanger leakage.
9.	Rupture of a beam tube.
10.	Loss of primary pumps and flow coast down.
11.	Flow blockage to single cooling channels.

Among those potential accidents, loss of primary pumps and flow coast down is chosen as the most severe accident. If the reactor core is securely safe during this accident, then the primary cooling system design meets the required safety requirements.

3. Methodology

Loss of Flow Event Simulation

The event initiating an accident due to loss of flow is that the primary coolant pump is suddenly not functioning. This event is simulated by assuming that the reactor is in operation at nominal power of 50 MW, primary coolant flow rate of 900 kg/s, and coolant pressure of 8 kgf/cm^2 and power supply is suddenly lost. This event is simulated using a transient computation code, PARET-ANL. In the simulation, the reactor is protected to scram if the coolant flow rate reaches 85% of its nominal capacity (765 kg/s) or if the power rises 114% of its nominal power (57 MW), any of which reached first will trigger reactor protecion system.

Input Data

The input data use in this analysis are shown in Table 2.

TABLE 2: Input data [1].

No.	PARAMETERs	Value
1.	Reactor power, MW Power generated in fuel element,% Core flow rate, kg/s Pressure, kgf/cm²	50 100 900 8,0
2.	Fuel element: Type Enrichment, % Geometry: Fuel element dimension,mm Fuel plate dimension, mm Fuel plate thick, mm Gap width, mm Number of plate per fuel element Number of plate per control element	 U9Mo-Al 19, 75 77, 1x81 0, 54x62, 75x700 1, 3 2, 55 21 15
3.	Cladding: Type Cladding thickness, mm	 $AlMg_2$ 0,38
4.	Power peaking factor: Radial power peaking factor, FR $F_{cool;}$ $F_{film;}$ $F_{hflx;}$ $F_{cladding}$, F_{bond}, F_{meat} Axial power peaking factor, F_A	 3,000 1,167;1,200;1,200 1,00; 1,00; 1,00 1,2
5.	Flow area, m²	0,07578078
	Mass flow rate, kg/s.m² t=0	11876,36
	Mass flow rate, kg/s.m², t=150	819,39
6.	Power trip, Mw	57
	Flow trip,%	85

4. Results and Discussions

Steady State Condition

RRI-50 that is operating at steady state condition, before LOFT occurs, is shown in Figure 1 and 2 and Table 3. Figure 1 depicts thermal hydraulic parameters at steady state condition, in which observation is carried out at the hottest channel. As shown in Figure 1, point 0 is the fuel outlet/bottom side, while point 70 is the fuel inlet/top side. It is shown in the figure that heat flux generated is 500 W/cm² and the coolant outlet temperature at the hot channel reaches 91.38°C, making 21.24°C difference from the average channel temperature. This difference is caused by the use of uncertainty factor at the hot channel. Meanwhile, the cladding temperature distribution has the same profile with axial heat flux distribution, in which the maximum cladding temperature reaches 164.54°C. Bulk temperature and cladding temperature have not reached two-phase temperature. Compared to DNBR temperature, the margin is 47.95°C. This margin toward two-phase temperature indicates that boiling has not occurred at the steady state condition.

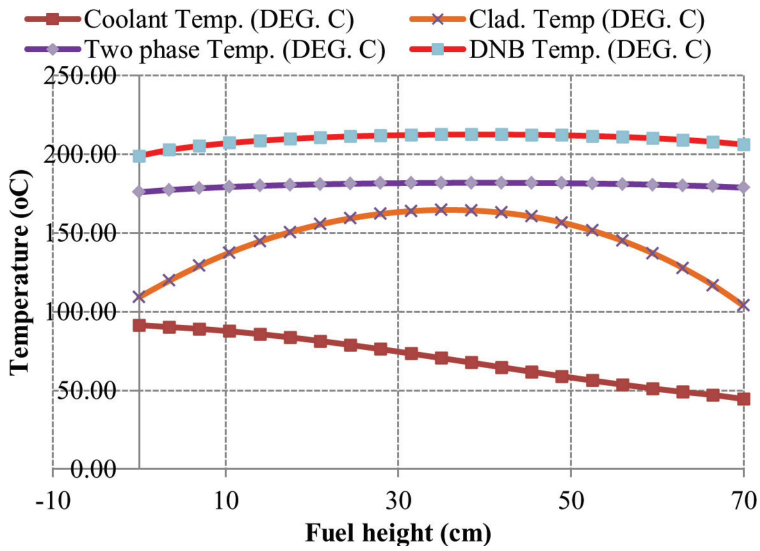

Figure 1: Thermal hydraulic parameters at steady state condition.

Figure 2 shows characteristics of heat transfer temperature at radial direction in one fuel plate, starting from the fuel central meat to the cladding and coolant. Temperature gradation patterns as a function of heat transfer process in the average channel and heat channel are indicated. There is a significant temperature difference at fuel region, between average channel and heat channel. The temperature of two-phase flow and DNBR temperature are the same for these two channels. The figure shows that there is sufficient temperature margin on two-phase temperature and DNBR temperature. Table 3 provides explanation for Figure 2 in term of figures at each region. This table outlines thermal hydraulic characteristics for RRI-50 at steady

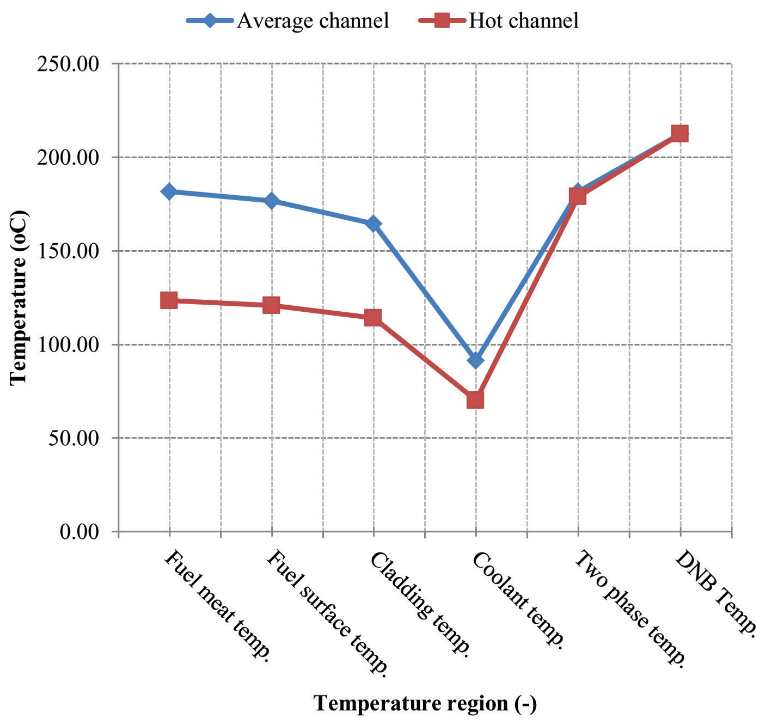

Figure 2: Thermal hydraulic parameters at fuel element regions.

Table 3: Comparison between hot channel and average channel at steadystate condition.

	Fuel meat temp.	Fuel surface temp.	Cladding temp.	Coolant temp.	Two phase temp.	DNB temp.
	(°C)	(°C)	(°C)	(°C)	(°C)	(°C)
Hot Channel	181,61	176,89	164,54	91,38	181,81	212,49
Ave. Channel	123,52	120,94	114,20	70,14	178,94	212,49

state in the average channel and hottest channel. The temperature difference is caused by the use of nuclear and engineering uncertainty factor in the hottest channel. On the other hand, at the average channel, both factors are assumed to 1.

Loss of Flow Event Simulation

As explained previously, LOFT event, which is anticipated at RRI-50, is one of DBA that should be considered as loss of flow event might occur any time during reactor operation. Figure 3 depicts the event at early period of LOFT. It is apparent that the decrease in flow rate occurs at 2.1th second when the flow rate reaches 85% of its normal value causing a signal due to minimum flow. The 0.5 second delay time makes the reactor scram at 4.62th second and causes the coolant and cladding temperature to decrease. Meanwhile, the flywheel of the pump provides residual flow, helping to take residual heat. At 10th second when the remaining reactor power is about 17% of the nominal power, the fuel and coolant temperatures get lower to 107°C and 73°C.

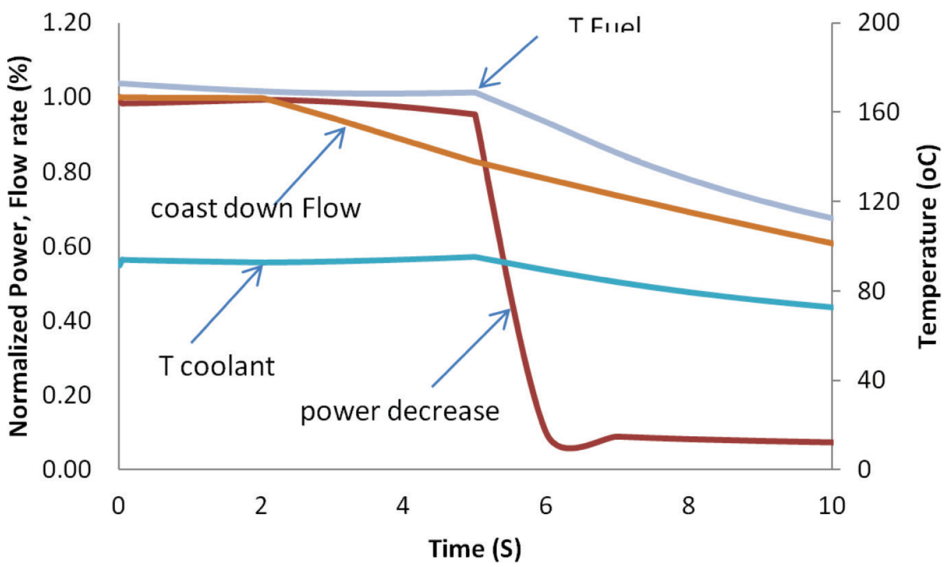

Figure 3: Normalized power and flow rate and temperature at the beginning of LOFT.

Figure 4 indicates the coolant and cladding temperature profiles at the beginning of LOFT. Significant temperature reduction is due to reactor scram. The fuel temperature decreases more rapidly than that of coolant because of the presence of cooling mode change from forced convection to natural convection. This change makes the two temperatures reach equilibrium soon after the transient.

DOI 10.18502/ken.v1i1.461

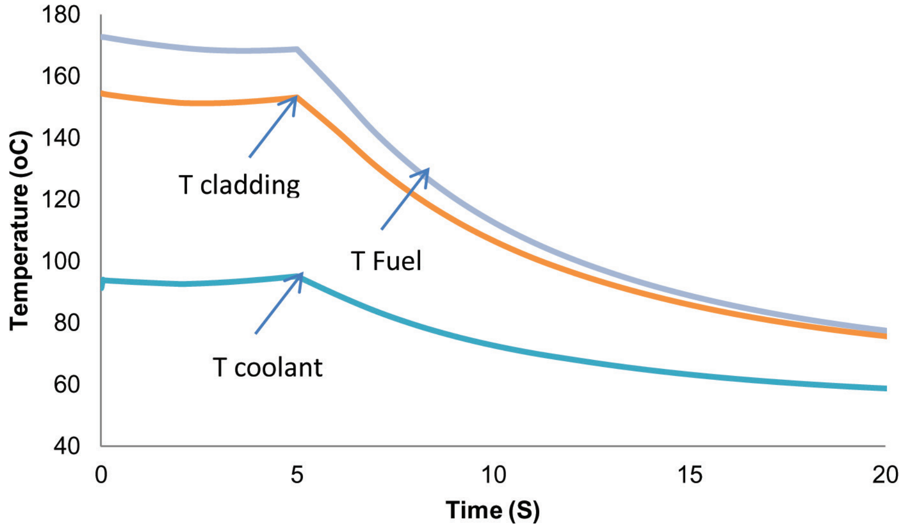

Figure 4: Coolant and cladding temperature profile at the beginning of LOFT.

Figure 5 shows the correlation among normalization of the primary coolant system, normalized power, and core reactivity during transient at early period of LOFT. It is shown that trip occurs at 4.6th second due to the flow rate of 85% of its normal value. Based on the assumption that there is a delay between trip signal and a control rod drop, scram time is 5.1 seconds. Since the reactor scrams, the reactor power drops significantly to 7.8% or 3.9 MW in 10 seconds. At the same time, core reactivity swings from -0.007$ to -17.40$. The presence of Doppler effect indicates that reactivity feedback occurs and causes reactor scram.

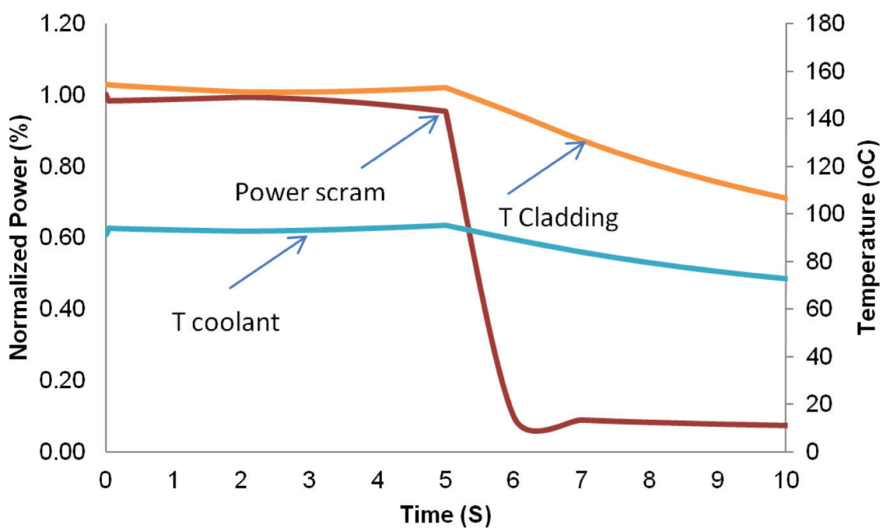

Figure 5: Fluence of reactivity and flow to power decrease.

The correlation of coolant temperature, cladding temperature and safety margin towards flow instability (S) is shown in Figure 6. The observation carried out at the first 40 seconds during loss of flow event indicates that S reaches the lowest value of 7.7 when the reactor scrams. Core reactivity change causes a decrease in cladding and coolant temperature to its respective equilibrium, i.e. 55°C and 67°C. At this equilibrium temperature, the margin towards S reaches its equilibrium value in 25 seconds, which is a safe value.

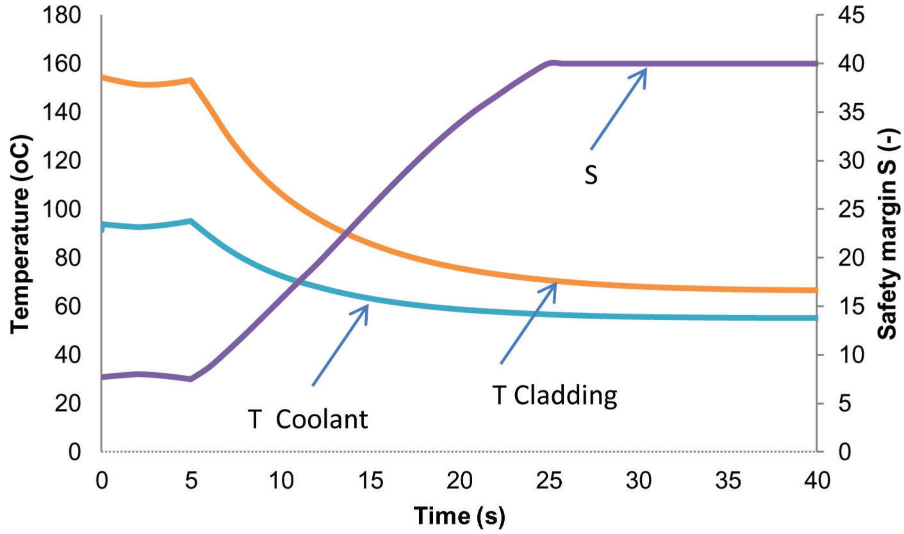

Figure 6: Corelation between coolant and cladding temperature and safety margin.

5. Conclusion

Based on the analysis results on thermal hydraulic characteristics of RRI-50 core towards loss of coolant flow transient at the beginning of accident using PARET-ANL code, it is concluded that the safety margin towards flow instability is still met at early period of LOFT and two-phase flow has not occurred at constant pressure.

6. Acknowledgment

This work has been carried out under the collaboration of RRI-50 Conceptual Design Project by PTKRN BATAN. I would like to thank you to the Neutronic Group for preparing some important parameters for this work.

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