Preliminary Safety Analysis of Anticipated Transient without Scram (ATWS) Events for the Prototype GEN-IV SFR (PGSFR) using MARS-LMR

Chiwoong Choi [a*] and Kwiseok Ha [a]
[a]*Korea Atomic Energy Research Institute, 989-111, Daedeok-daero, Yuseong-gu, Daejeon, 305-353, Korea*
[*]*Corresponding author: cwchoi@kaeri.re.kr*

1. Introduction

A safety analysis of ATWS for the recently designed Prototype GEN-IV Sodium Cooled Fast Reactor (PGSFR) was conducted. The MARS-LMR code has been used as a safety analysis tool, which was developed with new coolant properties and heat transfer and pressure drop correlations for liquid metals [1]. Unprotected Transient Over-Power (UTOP), Unprotected Loss OF Flow (ULOF), and Unprotected Loss Of Heat Sink (ULOHS) were selected as representative events for the ATWS. In an unprotected condition, the power in the core is only controlled by reactivity feedbacks, which are interacted with the thermal-hydraulic characteristics of the components in the plant. Heat is removed by the steam generator (SG) and decay heat removal system (DHRS). Therefore, the major objectives of the safety analysis of the ATWS events are to investigate the thermal hydraulic characteristics of the DHRS and the SG, the neutron kinetic characteristics of the reactivity feedback, and the interaction between the neutron kinetics, and the thermal-hydraulics during the events.

2. Safety Analysis of ATWS

2.1 Basic analysis parameters

The reactor is modeled with five channels, i.e., inner, outer, hot subassembly, hot pin, and non-fueled subassembly. In this analysis, the hot pin is newly modeled to check the hottest pin behavior. In addition, in the near future, this hot pin will be used to evaluate a cumulative damage function (CDF) as a safety limit. A coolant flow is driven by two primary mechanical pumps, and passed by the core region though the inlet plenum. The primary heat transport system (PHTS) is divided into cool and hot pools based on the core. Heat is removed by two steam generators, which are connected by two intermediate heat exchangers (IHXs) with one pump for each loop, which is called an intermediate heat transport system (IHTS). The DHRS consists of passive DRC and active DRC, which have two loops with DHX and air-sodium heat exchangers by each. In addition, one of the major differences from the previous design is the location of the DHX, which is moved from the hot pool to the cold pool. The nodalizations of the PGSFR are described in Fig. 1.

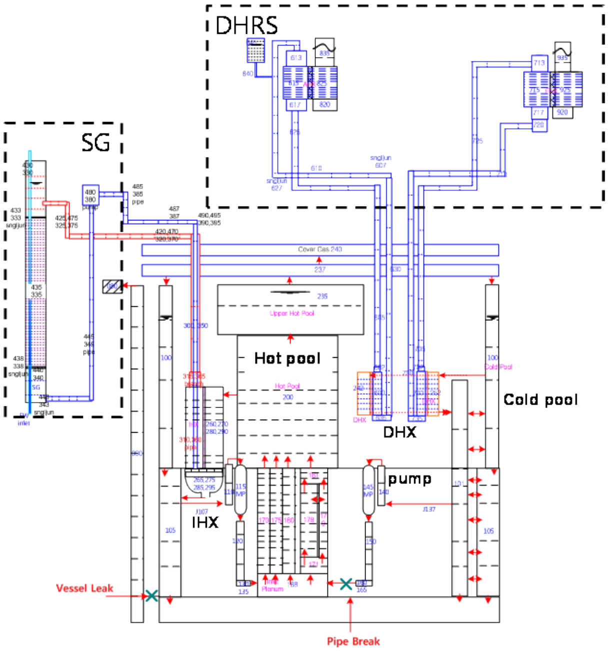

Fig. 1 Nodalization of the PGSFR

The total power in the core is 392.6 MWt, and the total dissipation heats in the pumps of PHTS and IHTS are 1.3 MWt and 0.87 MWt, respectively. The heat removal rate in the SG is 393.6 MWt. The heat removal rate in the single DHRS loop is 1 MWt, and thus the total heat removal rate of DHRS is 4 MWt for all four loops. The capacity of the DHRS was reduced from the previous 20MWt. Based on the design values, a steady state result was obtained, which was used as the initial condition for the ATWS event analysis.

2.2 Reactivity feedback models

In the PGSFR, there are additional reactivity feedbacks other than the Doppler and coolant density, which are related to the structural thermal expansions. The MARS-LMR was implemented with three kinds of reactivity feedback models including the fuel axial, core radial, and control rod/reactor vessel expansions [2]. The definitions of these reactivity feedbacks are as follows:

<u>Fuel Axial Expansion</u>

$$\delta R^A = \sum_i C_i^A \left(1 - \frac{\rho_{Fi}^{t=0}}{\rho_{Fi}^{t=0}} \right) \quad (1)$$

<u>Core Radial Expansion</u>

$$\delta R^R = C^R \ln \left(\begin{array}{l} \frac{N_{LP}}{N_t} \sum_i W_i^{LP} \varepsilon_i^{LP} + \\ \frac{N_{GP}}{N_t} \sum_i W_i^{GP} \varepsilon_i^{GP} \end{array} \right) \quad (2)$$

<u>CRDM Expansion</u>

$$\delta R^{CR} = C^{CR} \left(\Delta Z_{CR} - \Delta Z_{RV} \right) \quad (3)$$

where R is the reactivity [$], C is the reactivity coefficients, N is the ring number covered by the load pad (LP) or the grid plate (GP), W is the weighting factor, and Z is the displacement for the control rod (CR) or the reactor vessel (RV). The superscripts A, R, and CR represent axial, radial, and CRDM expansions.

Based on the reactor design in the beginning of cycle (BOC), the coefficients were evaluated. The U-Zr metallic fuel density property was used for the fuel axial expansion. The representative structures for the LP and GP were assigned to the subassembly and the inlet plenum respectively. The control rod and the reactor vessel were modeled individually. At the BOC condition, the primary control rods were insulated 0.35 m from the top of the active core region. The details for the input calculation are similar to a previous KAERI report [3].

3. Results

3.1 UTOP

The UTOP event is initiated with 30 cents of reactivity insertion for 15 seconds at 10 seconds, in which the initiating condition is taken from a previous analysis. The power is saturated at about 1.25-times the nominal power at around 3×10^4 seconds. The highest power is about 1.65-times the nominal power, as shown in Fig. 2. Fig. 3 shows reactivity feedback during the UTOP event. The major negative feedback component is the radial expansion. The new reactivity equilibrium reached around 600 seconds, in which the heat rejection capacity is exceeded over the reactor power, as shown in Fig. 4. The major heat rejection system is the steam generator. In addition, the peak cladding temperature is 1017 K at 90 seconds. In addition, the equilibrium cladding temperature is 962 K. Based on the initial position of the primary control rods worth, the maximum worth by the withdrawal of the single control rod is approximately 50 cents.

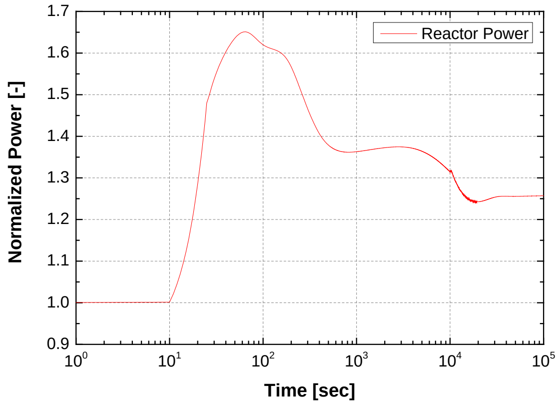

Fig. 2 Normalized power during the UTOP event

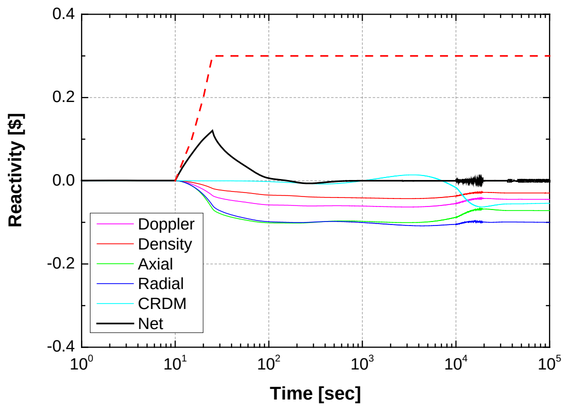

Fig. 3 Reactivity feedbacks during the UTOP event

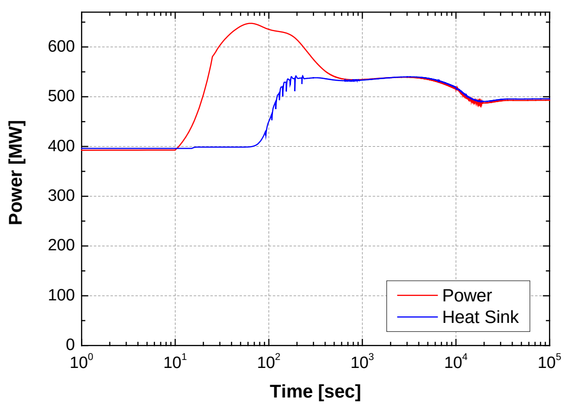

Fig. 4 Heat balance during the UTOP event

Accurate initiating conditions need to be considered for a future analysis.

3.2 ULOF

The ULOF assumes that the primary and secondary pumps are tripped at 10 seconds. The coastdown halving times of the pumps in the PHTS and the IHTS are 8 and 4 seconds, respectively. The transient is calculated for 1.5×10^5 seconds. The power is reduced due to negative reactivity feedbacks corresponding to rising temperatures. Fig. 5 shows the power during the transient, which indicates the inherent reactor shutdown is achieved by the reactivity feedbacks. The dominant reactivity feedback component is the radial expansion as shown in Fig. 6. Initially, the temperatures of the coolant, cladding, and fuel are increased due to a mismatch between the power and flow. Thus, the peak

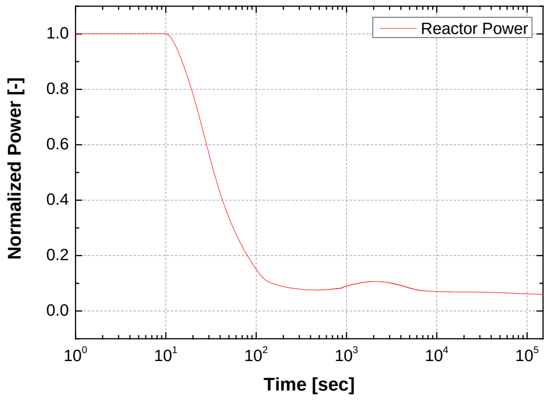

Fig. 5 Normalized power during the ULOF event

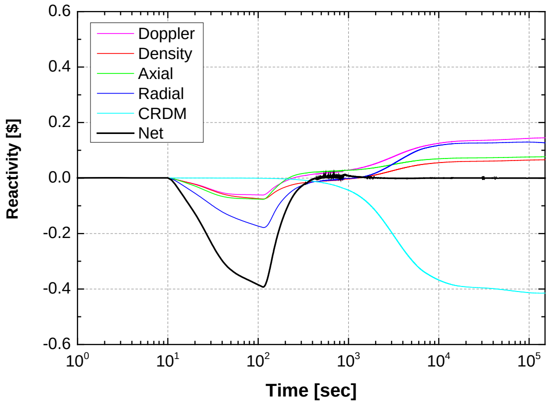

Fig. 6 Reactivity feedbacks during the ULOF event

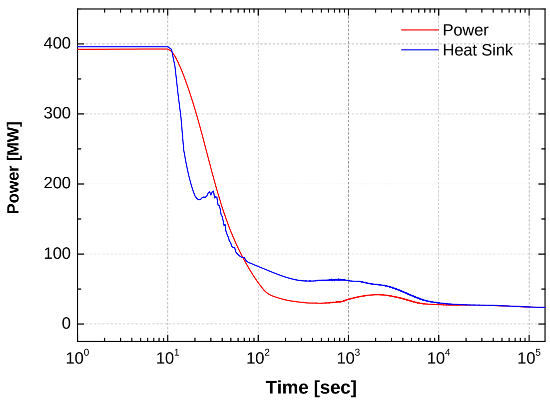

Fig. 7 Heat balance during the ULOF event

cladding temperature reached 1046 K at 113 seconds. In addition, the temperatures of the system are continuously decreased by reaching a new equilibrium.

In addition, because of the reduction of temperatures of the coolant and the structures except the temperature difference between the inlets/outlets of the core, all reactivity feedbacks are changed to positive except the CRDM expansion. Fig. 7 shows heat balance during the transient. At around 90 seconds, the capacity of the heat sinks including SG and DHRS are excessed over the reactor power. Then, the inherent shut-down is achieved at about 10^4 seconds.

3.2 ULOHS

The ULOHS assumes that SGs are failed at 10 seconds. Therefore, the heat rejection has to be accomplished by only the DHRS, which is activated at 5 seconds after the initiating of the event. One of the major goals for the ULOHS analysis is to evaluate the performance of the DHRS. Fig. 8 shows the normalized power. The negative reactivity feedback inherently makes the power decrease. However, at around 2500 seconds, the power has a small peak due to CRDM reactivity feedback, as shown in Fig. 9. The power is being decreased and the primary pumps are still working, and thus, the

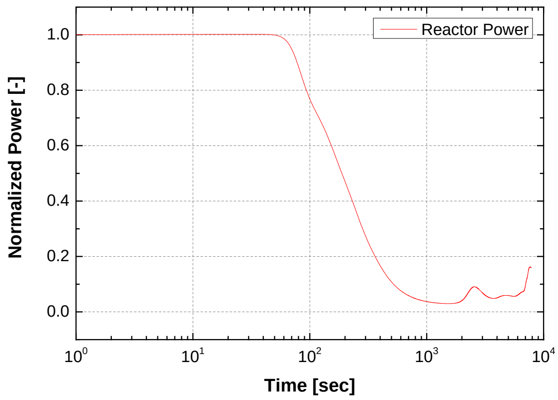

Fig. 8 Normalized power during the ULOHS event

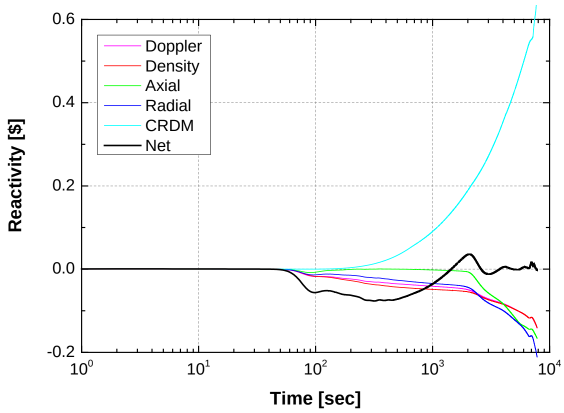

Fig. 9 Reactivity feedbacks during the ULOHS event

temperature difference between the inlet and outlet of the core is decreased. In addition, the DHRS heat removal capacity is still not enough to cool the core down, which means the pool temperature rise. Owing to the drastic rise of both cold and hot pool temperatures, the reactor vessel expansion is more predominant than the control rod expansion. Thus, the CRDM reactivity changed to positive and continuously increased as the pool temperature increased. Thus, the net reactivity is changed to positive, and a small peak of the power occurred.

The ULOHS calculation was failed at about 8000 seconds due to a higher pressure over 10 MPa in the SG. Currently, the design of the expansion vessel in the SG is being developed. Thus, the cover gas in the SG is modeled with an engineering sense. Thus, this calculation failure indicates that the capacity of this cover gas volume may not be enough during the ULOHS transient. Considering the rupture disk setting pressure of 1MPa, the higher pressure than the set pressure in the SG, will make the liquid sodium be drained to a dump tank. Therefore, recalculation of the ULOHS with the designed expansion vessel will be carried out.

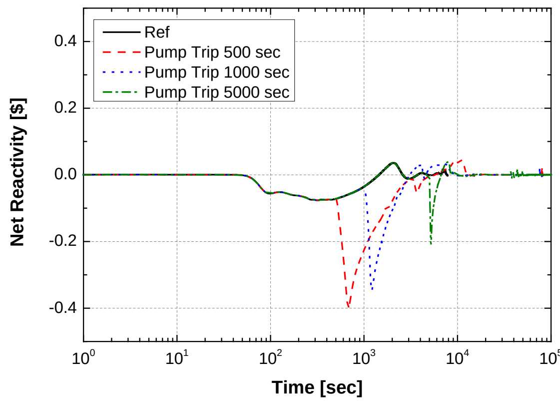

Fig. 10 Net reactivity during the ULOHS event for different trip times of the primary pumps

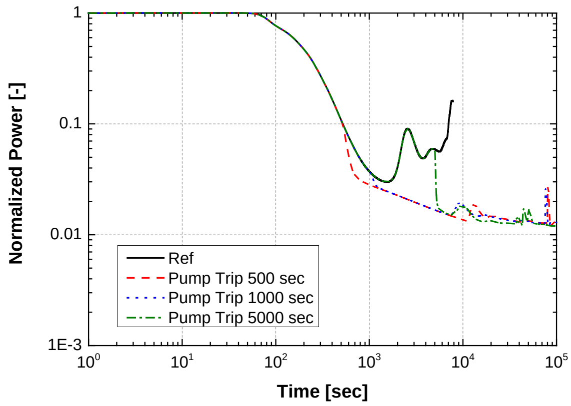

Fig. 11 Normalized power during the ULOHS event for different trip times of the primary pumps

The dissipation heat from the primary pump is 1.3 MW. In addition, the capacity of a single DHRS loop is 1 MW. Therefore, at least two DHRS loops are necessary to remove heat from the primary pumps. For long-term cooling, the primary pump should be tripped. The PRISM safety analysis for ULOHS reported that the primary pump will be a major contributor as a heat source in the long-term cooling [4]. In this study, the primary pumps were tripped at 500, 1500, and 5000 seconds. Fig. 10 and 11 shows the net reactivity and normalized power, respectively. When the primary pumps are tripped, additional negative reactivities are inserted due to a much higher temperature rise, which directly enhances the power reduction. Therefore, the DHRS can cover the decay heat sooner. The timing of the trip is not an influential parameter for long-term cooling.

3. Conclusions and Further Works

The safety analysis for ATWS events: UTOP, ULOF, and ULOHS are carried out using the MARS-LMR. Based on the interaction between the neutron kinetics in the reactivity feedback components and thermal-hydraulics in the DHRS, the inherent equilibrium of the UTOP and the ULOF are successfully achieved. However, the ULOHS has an issue related to overpressure in the steam generator. It will be considered with the updated designs of the expansion vessel. In addition, during the ULOHS transient, the trip of the primary pumps is tested with different trip times. The trip of the primary pumps is helpful in reducing the power with additional negative feedback. However, the timing itself is not an effective parameter for long-term cooling. In the near future, a sensitivity test for unprotected events with the developed designs of the PGSFR will be conducted.

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