A Sensitivity Analysis of a Pipe Break Accident in a Preliminary Specific Design of the PGSFR

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1. Introduction

The PGSFR (Prototype Gen-IV Sodium-cooled Fast Reactor) is a pool type sodium cooled fast reactor with a thermal power of 392.1 MW which has been developed in accord with an enhanced safety, an efficient utilization of uranium resources and a reduction of a high level waste volume in the Korea Atomic Energy Research Institute (KAERI) since 2012 under a National Nuclear R&D Program. The PGSFR has an inherent safety characteristic owing to the design to have a negative power reactivity coefficient during all operation modes and it has a passive safety characteristic due to the design of a passive decay heat removal circuit. For an evaluation of the safety features of the PGSFR, a sensitivity analysis has been performed for a pipe break accident which is one of the most important DBEs in the PGSFR. For a sensitivity analysis, some design variables are applied to be conservative. An effect of uncertainties is evaluated on a Doppler reactivity and a sodium density. Sensitivity studies have been also performed to find the most conservative condition for an air flow rate and an air temperature. The results of the sensitivity analysis provide a cumulative damage function (CDF) which is related to a fuel damage or threat to its structural integrity during the transients for the considered DBEs.

2. Analysis Methods

Figure 1 describes a nodalization of the PSGFR applied to the MARS-LMR code. The PGSFR is composed of a Primary Heat Transport System (PHTS), an Intermediate Heat Transport System (IHTS), a Steam Generating system (SG) and a safety-grade decay heat removal system (DHRS). The DHRS is composed of two units of Passive Decay-heat Removal Circuits (PDRCs) and two units of Active Decay-heat Removal Circuits (ADRCs).

As shown in the nodalization, two main pumps take sodium from a pool and discharge it into inlet pipes in the primary system. Then the sodium flows into an inlet plenum where one of the inlet pipes is assumed to be broken as an initial event.

The pipe break event was analyzed using MARS-LMR code. The event was assumed to start at 102% power. The ANS-79 model was used for a core decay power after a reactor scram. At 5 seconds after the reactor trip, SG feed-water lines were isolated and the primary and secondary pumps were tripped corresponding to a loss of off-site power (LOOP). One independent PDRC and one ADRC were assumed to be available in accordance with a single failure criterion and maintenance. AHX and FHX dampers were assumed to be open at 5 seconds after the reactor trip.

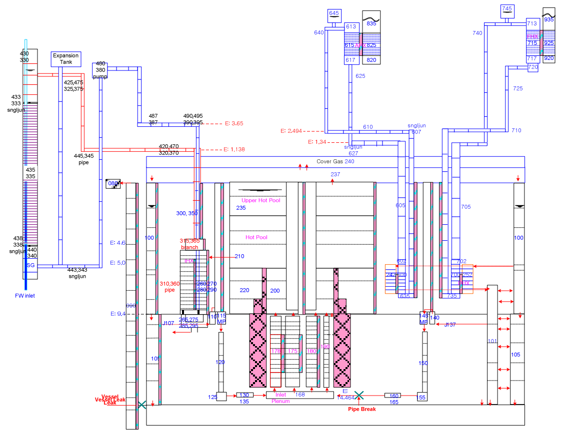

Figure 1 Nodalization of the PGSFR

In this simulation, each unit of the DHRS can remove 0.625% of a nominal power. The ADRC can also be operated in a passive mode, which corresponds to 0.3125% heat removals.

Table 1 depicts a reactor protection signals and their set-points. The reactor is scrammed by a high power to PHTS flow ratio of 121.4 %, high core outlet temperature of 570.9 ℃, high core inlet temperature of 415.9 ℃, high SG shell outlet temperature of 364.9 ℃, or low hot pool level of 30 cm below from a normal level.

Table 1. Reactor protection signal and set-point

Parameter	Set-point (Uncertainty)
High core outlet temperature	565℃ (±5.9 ℃)
High core inlet temperature	410℃ (±5.9 ℃)
High power to PHTS flow ratio	119 % (±2.4 %)
SG shell outlet temperature	359 ℃ (±5.9 ℃)
Low hot pool level	20 cm below 100% operating level (±10 cm)

A safety acceptance criterion for a safety analysis of the PGSFR is evaluated on the CDF which is dependent on some parameters such as cladding thickness, fuel gas plenum pressure, burn-up rate, pin power, etc. This CDF model is applied to the MARS-LMR code. 0.05 is used as a limit of the pipe break event.

3. Transient Results of Pipe Break Accident

The primary coolant flows into the inlet plenum from four pipes connected with two PHTS pumps. The primary pipe break accident was assumed to be initiated from a pipe break of one of the four pipes. The flow through the broken pipe is discharged into a cold pool.

An imbalance between the reactor power and primary flow is a main safety concern of the pipe break event. To prevent an occurrence of a severe imbalance between the power and flow, the reactor was designed to be tripped by a high power/flow trip.

In this simulation, the accident occurs at 10 seconds. A reactor is scrammed at 17.5 seconds by a high power trip and the reactor power and primary flow decreased. The power decreased drastically owing to the reactor trip, and the cladding temperature showed the highest value. Figs. 2 and 3 show a cladding mid-wall temperatures and coolant temperatures through a core during the pipe break accident, respectively. The initial cladding temperature increases due to a decrease of sodium flux into the reactor core as shown in Fig. 4.

Both PHTS pumps and IHTS pumps are stopped with an assumption of the LOOP at the same time of the reactor trip. Therefore, a residual heat removal is achieved only by an evaporation of water in SG tubes and by the DHRS. Fig. 5 shows heat removals by SGs, which decrease rapidly due to a feedwater isolation. Water inside of SG tubes is left until about 1000 seconds.

Fig. 6 compares a decay heat removal rate of the DHRS with the reactor power. After about 6500 seconds, the amount of heat removals by the DHRS is higher than a core residual heat production, and a core outlet temperature decreased continuously.

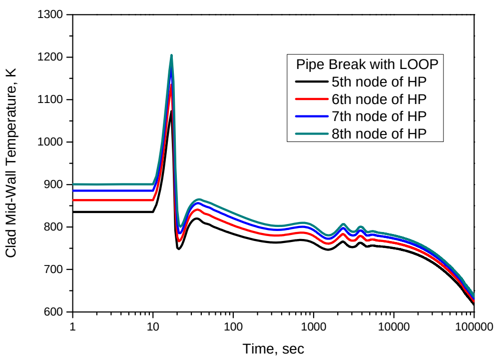

Figure 2 Clad mid-wall temperature change during PB accident

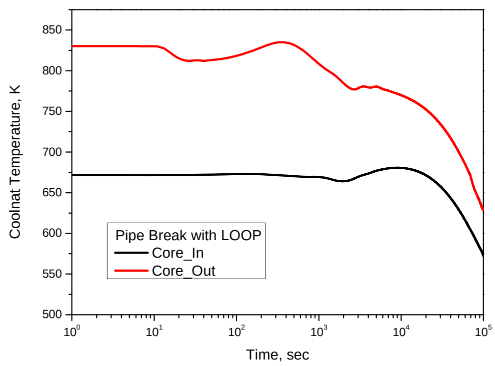

Figure 3 Coolant temperature change during PB accident

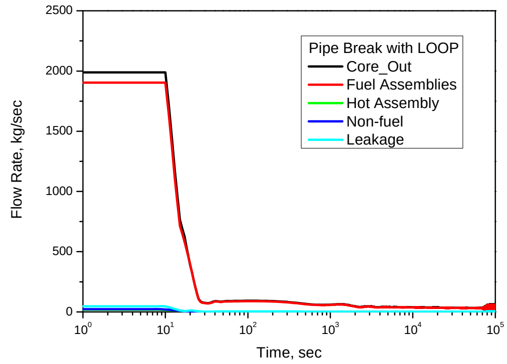

Figure 4 Flow rates through core channels during PB accident

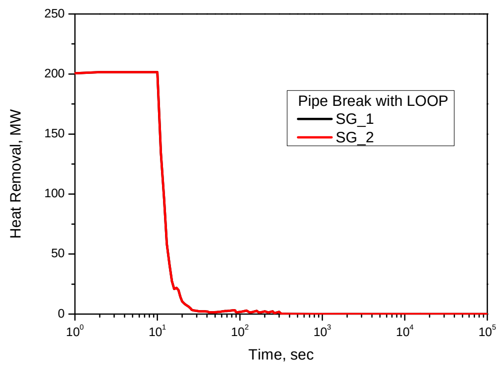

Figure 5 Heat removals by SGs during PB accident

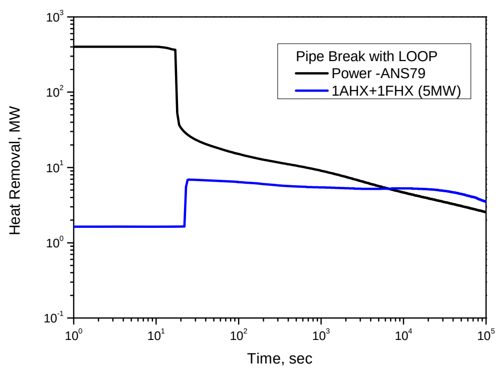

Figure 6 Heat removals by DHRS compared with decay power during PB accident

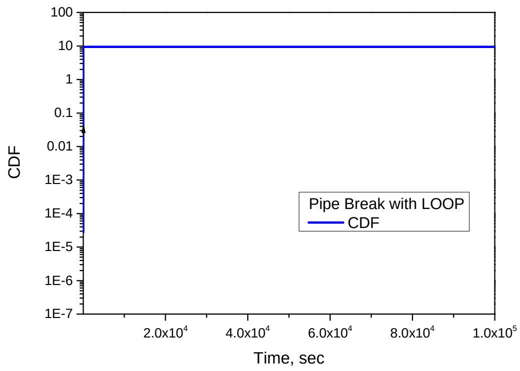

Figure 7 Calculated CDF during Pipe Break accident

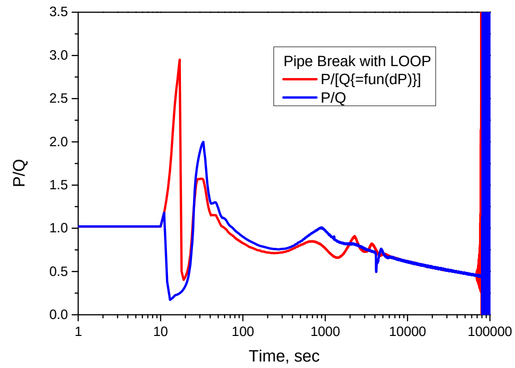

Figure 8 Power to flow ratio in comparison P/Q with P/Q{=f(d P)}

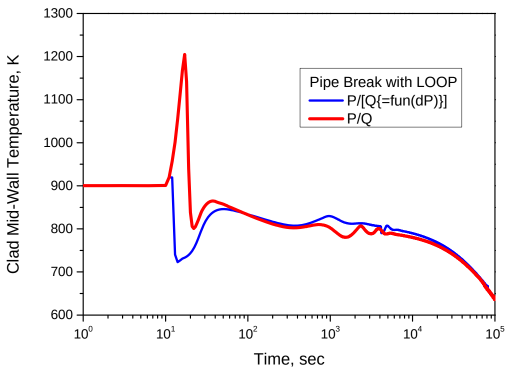

Figure 9 Clad mid-wall temperature during PB accident

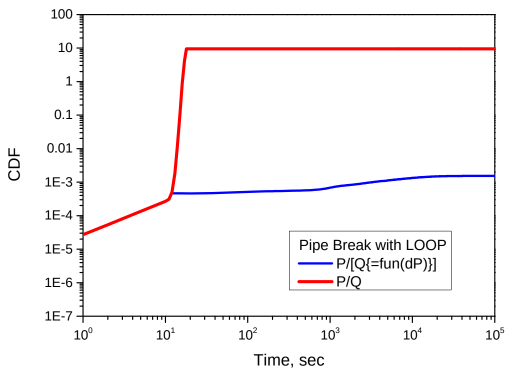

Figure 10 Calculated CDF in comparison P/Q with P/Q{=f(d P)}

Fig. 7 shows the calculated CDF during the PB accident. Immediately after a pipe was broken, the CDF is drastically increased and then it goes over 0.05. Therefore, a faster detection is needed to prevent the fuel damage. A method of extracting the PHTS flow rate from a pressure drop was used instead of a measurement of the PHTS flow rate. A correlation between a pressure drop and a PHTS flow rate is indicated in a following equation (1). The pressure drop is estimated as the difference from a core exit to a PHTS pump discharge chamber.

$$Q_f=(dP/dP_{nominal})^{0.56} \qquad (1)$$

Where, $dP=P_{pump\_discharge\_chamber} - P_{core\_exit}$

The analysis results using the method of extracting the PHTS flow rate from the pressure drop, is shown in Figs. 8 to 10. Immediately after a pipe was broken, the reactor is tripped faster than the previous result, and then the CDF is calculated under 0.05. Therefore, it is concluded that a faster detection is important in a pipe break accident. Based on this result, a sensitivity analysis was also performed to find the most conservative condition.

4. Sensitivity Analysis

For the sensitivity analysis, some design variables are applied to be conservative. Table 3 shows a range of design parameters and reactivity parameters with their uncertainties. An air flow rate, air temperature, Doppler reactivity, and density reactivity are selected for the sensitivity variables.

Table 3. The range of sensitivity parameters

Parameter	Range	Uncertainty
Air flow rate	-50%~100%	50%
Air temperature	10℃~ 50℃	
Doppler reactivity	-15% ~ 15%	15%
Density reactivity	-20% ~ 20%	20%

Figs. 11 to 13 show the results for the sensitivity calculations. As shown in the Figs., the CDF is not sensitive for the variation of the air temperature, Doppler reactivity, and density reactivity but the variation of the air flow rate affects more severely on the CDF. The less air flows into the AHX, the less heat removal is achieved, and then the larger CDF is calculated. Therefore, it is concluded that the air flow rate is the most sensitive and conservative variable in the pipe break accident, and then it is important to know an accurate uncertainty of the variable.

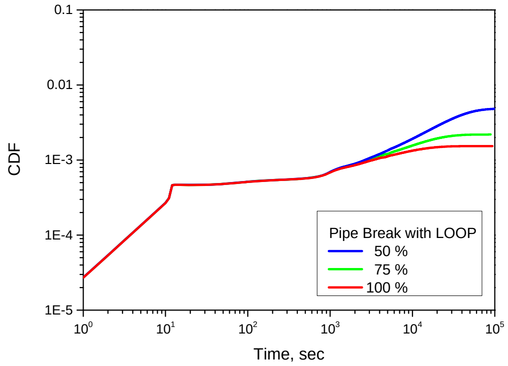

Figure 11 Calculated CDF versus air flow rate

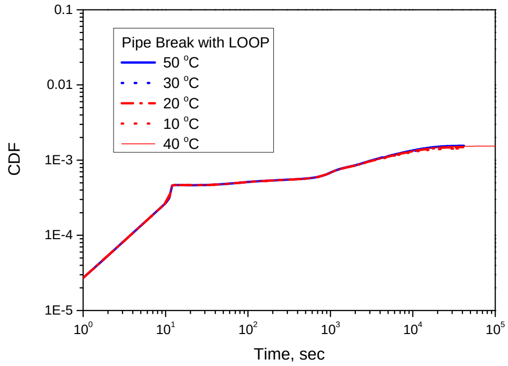

Figure 12 Calculated CDF versus air temperature

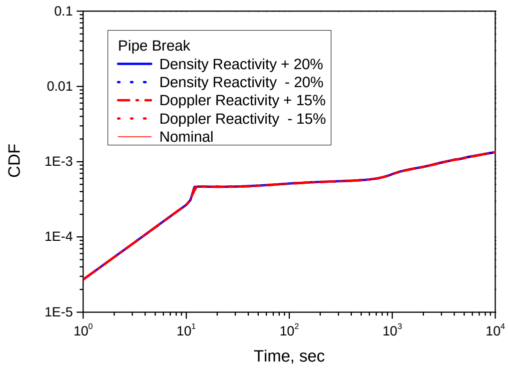

Figure 13 Calculated CDF versus reactivity feedback

5. Conclusions

In order to assess the inherent safety features of the PGSFR, a safety analysis was performed for a pipe break accident with MARS-LMR. And, the sensitivity studies were also performed to find the most conservative condition. As a result, the PGSFR was appropriately tripped by a high power to PHTS flow ratio using the method of extracting the PHTS flow rate from the pressure drop. The air flow rate was the most sensitive variable in the sensitivity analysis. Therefore, it is important to know the accurate uncertainty of the air flow rate in the AHX.

Acknowledgements

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REFERENCES

[1] K. S. Ha, Safety Evaluation for Transients of Demonstration SFR, Korea Atomic Energy Research and Institute, 2011.