An Assessment of Transient Over-Power Accident in the PGSFR

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Abstract. KAERI (Korea Atomic Energy Research Institute) has been developing a preliminary specific design of the PGSFR (Prototype Gen-IV Sodium-cooled Fast Reactor), which is a pool-type sodium cooled fast reactor with a thermal power of 392.2 MW. 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 TOP (Transient Over-Power) which is one of most important DBEs (Design Basis Events) in the PGSFR using MARS-LMR code. MARS-LMR contains the sodium property table including dynamic properties, heat transfer correlations for the liquid metal, and the models describing the flow resistance by wire-wrap spacer in the core, which shows a good agreement with the experimental data conducted in the EBR-II plant and the appropriateness of the models related to liquid metal reactor. 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. Conservative assumptions are applied to the analysis of the plant responses during the postulated DBAs, which are 102 % of power condition with ANS-79 decay power model, 5.0 seconds delay in opening of AHX and FHX dampers, and loss of off-site power (LOOP) is taken into account. Additionally, one PDHRS (Passive Decay Heat Removal System) and one ADHRS (Active Decay Heat Removal System) are available in accordance with a single failure criterion and maintenance. As a result, the preliminary specific design of PGSFR, meets safety acceptance criteria with a sufficient margin during the TOP event and keep accidents from deteriorating into more severe accidents.

Key Words: PGSFR, TOP, DBE, MARS-LMR

1. Introduction

SFR (sodium fast reactor) design technologies have been developed in South Korea since 1997 under a National Nuclear R&D Program intended to achieve enhanced safety, efficient utilization of uranium resources, and reduction in the volume of high level waste. In 2015, the preliminary specific design of the PGSFR was completed. It is a pool-type sodium-cooled fast reactor with the thermal power of 392.2 MWt, and uses metallic fuel of U-Zr(10%) in a core having inherent reactivity feedback mechanisms and high thermal conductivity.

The PGSFR consists of the PHTS (Primary Heat Transport System), the IHTS (Intermediate Heat Transport System), the SGs (Steam Generators) including BOP (Balance of Plant), and the DHRS (Decay Heat Removal System) shown in Fig.1. The PHTS is placed in a large pool to make the system transients slower, which provides greater probability that abnormal events will terminate before they propagate to become accidents. The IHTS loop is thermally coupled to the PHTS and to the SGs (steam generators). The IHTS transfers the reactor-generated heat from the IHX (Intermediate Heat eXchanger) of the PHTS to the SG. The IHTS consists of two loops, and each loop has two IHXs, one EM (electro-magnetic) pump, one expansion tank, and one steam generator. The SGs consist of two independent steam generation loops and where sub-cooled water is converted to super-heated steam. The DHRS has a heat transfer capability of 10 MWt, and is composed of two units of PDHRS and two units of ADHRS. In addition, a damper driven by the emergency diesel generator is attached to the AHX (Natural-draft sodium-to-air Heat Exchanger) and to the FHX (Forced-draft sodium-to-air Heat Exchanger). The damper design concept is a passive fail-open type. The ADHRS has been designed to operate at half capacity by natural circulation, even if the EM-pump of the ADHRS stops [1].

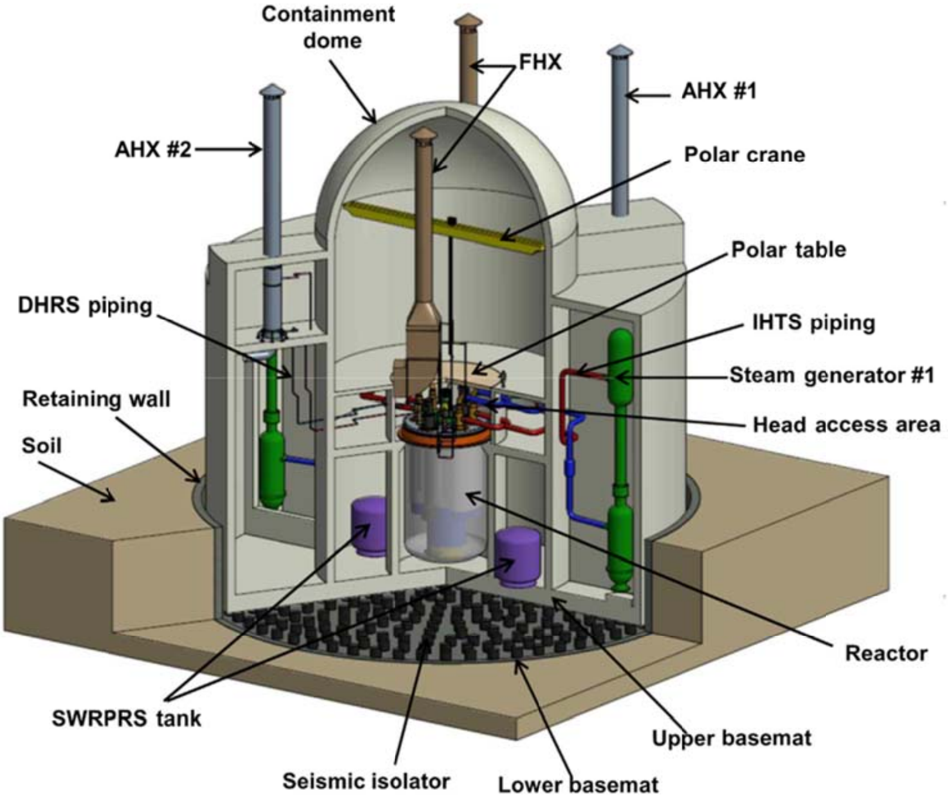

Figure 1. Overall Configuration of the PGSFR

The fundamental approach to design of a safe nuclear reactor is defense-in-depth. The performance of safety functions is assured, in normal operation and under accident conditions, by including multiple, independent, redundant means in the design. The cladding and end seals of each fuel pin are the first barrier to protect against the escape of radiological material to the environment. Table I shows the safety acceptance criteria of the fuel and cladding for each event category. An acceptance criterion for AOO (Anticipated Operational Occurrences) and DBA (Design Basis Accident) Class 1 is established on the basis of CDF (Cumulative Damage Function). CDF is introduced as a measure to protect against rupture due to thermal creep. The acceptance criteria for DBA class 2 and DEC (Design Extension Condition) are based on temperatures of pin melting and coolant boiling. Two temperatures are important to assure pin coolable geometry and no propagation. First, the peak fuel temperature should remain below the solidus temperature and second, the cladding temperature should remain below 1,075℃, a threshold temperature for rapid eutectic penetration [2]. At this temperature the eutectic penetration rate jumps by two orders of magnitude associated with melting of a protective solid iron-uranium compound. In DEC events, massive fuel melting is allowed, as long as the molten core is contained in a vessel with a coolable geometry. In such a scenario, the coolant temperature is a more important factor than the fuel or cladding temperatures, and it should be maintained below the sodium boiling temperature.

Based on the safety acceptance criteria described in Table 1, studies on system transients are carried out to assess the inherent safety features of the PGSFR.

TABLE I: SAFETY ACCEPTANCE CRITERIA FOR EVENT CATEGORY

Event Category	**AOO**	**DBA Class 1**	**DBA Class 2**	**DEC**
Fuel/ Cladding	CDF*∑AOO< 0.05 Strain<1%	CDFevent< 0.05 Strain<1%	Fuel T<Solidus T Clad T<1075 °C Coolant T<Boiling T	Coolant T <Boiling T

2. Analysis Method

Figure 2 shows the MARS-LMR nodalization for the PGSFR preliminary specific design. The core is modeled as four parallel flow channels (i.e., hottest sub-assembly, fuel assemblies, non-fuel assemblies, and leakage flow). Active fuel regions are axially divided into eight nodes. The PHTS is placed in a large pool with two temperature zones. Four sodium-to-sodium decay heat exchangers (DHX), and two pumps, are located in the cold pool; while four IHXs are located in the hot pool to transfer the reactor-generated heat from the PHTS to the SG. The IHTS consists of the two IHXs tube side, piping, one EM-pump, and one SG shell side. The steam generator tubes are divided into a total of 30 nodes. The SG inlet feed-water boundary region is described with a constant mass flow-rate condition, and the SG outlet boundary region near the high-pressure turbine is described with a constant pressure condition. Each DHRS is modeled in passive and active modes (i.e., using PDHRS and ADHRS, respectively). The DHX is immersed in the cold pool region and the sodium-to-air heat exchanger is located in the upper region of the reactor building. Air boundary regions in the mode are imposed at the entrance and exit of this part.

The reactor shutdown system requires inclusion of a mandatory protection system to prevent deterioration of the plant during all conceivable accidents. Table 2 lists the trip parameters and the set points (with uncertainties) of the reactor protection system.

Conservative assumptions are applied to the analysis of plant responses during the postulated DBAs. These include 102 % of power condition with ANS-79 decay power model [3], 5.0 seconds (s) delay in opening of AHX and FHX dampers, and loss of off-site power (LOOP). Additionally, one PDHRS and one ADHRS are available in accordance with a single failure criterion and maintenance.

TABLE II: TRIP PARAMETERS AND SET POINTS

Parameter	**Set-point (Uncertainty)**
High core inlet temperature	410 (±6) ℃
High power to PHTS flow ratio	119 (±2.4) %
SG shell outlet temperature	359 (±6) ℃
Low hot pool level	0.2 m below 100% operating level (±10 cm)

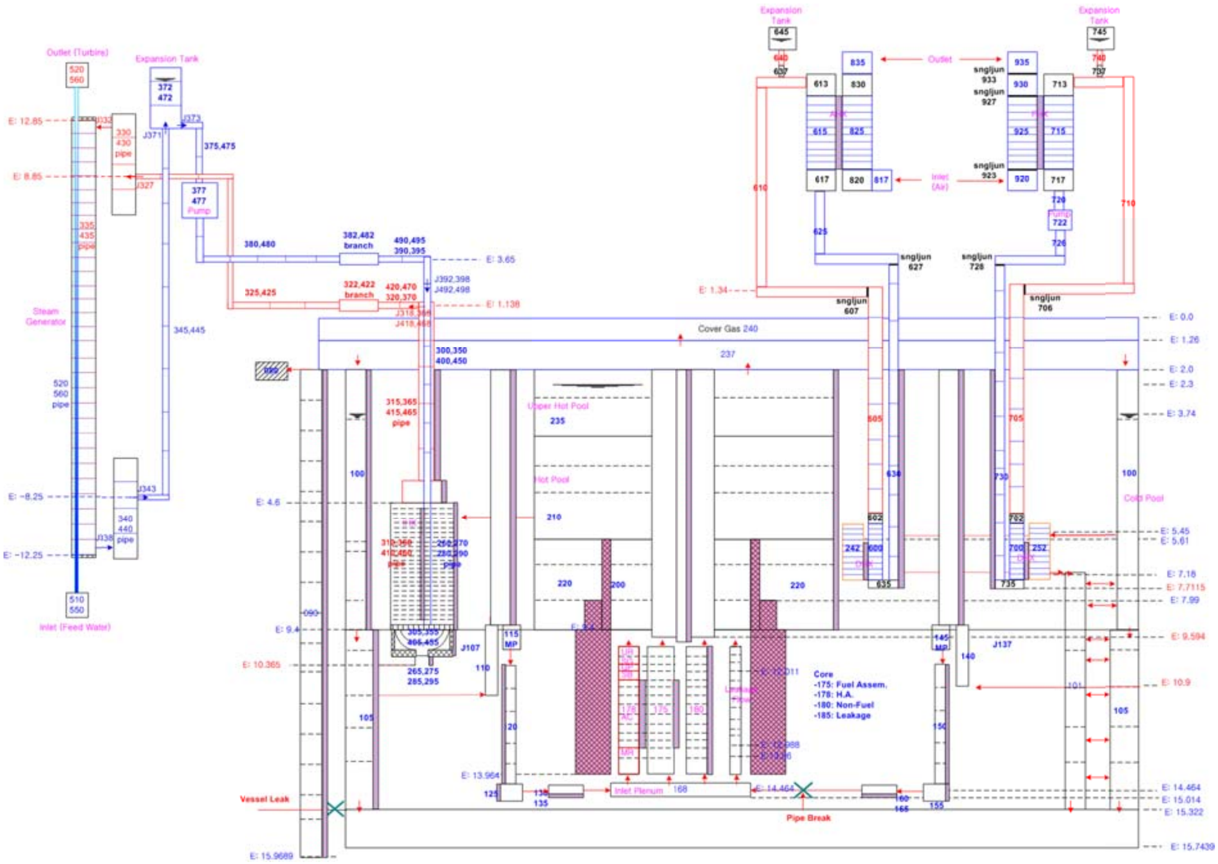

FIG. 2. MARS-LMR Nodalization for PGSFR

3. Analysis Results

A TOP accident is assumed to occur due to a single rod withdrawal. The event is initiated at 10.0 s, and a positive reactivity of 30 ¢ is inserted for 15 s. The core power rapidly increases due to the positive reactivity insertion after the initiation of rod withdrawal and the cladding temperature shows the highest value as shown in Fig.3.

The reactor power drastically decreases due to the reactor trip by a high power/flow signal as shown in Fig.4. To prevent an occurrence of a severe imbalance between the power and flow, PGSFR was designed to be tripped by a high power/flow trip.

In 5 seconds after the reactor trip, the stop signal occurs at the steam generator feed water and PHTS pump follows the pump coastdown. The core inlet and outlet temperature start to rise due to decreased flow rate by the PHTS pump coastdown together with reduced heat transfer to IHTS by feedwater isolation as shown in Fig.5.

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. At about 1000 seconds, the natural circulation flow rate increases in PHTS due to the increase of the natural circulation flow in IHTS as shown in Fig.6 to Fig.7.

Fig. 8 compares a decay heat removal rate of the DHRS with the reactor power. After about 5200 seconds, the amount of heat removals by the DHRS is higher than a core residual heat production, and a core outlet temperature decreased continuously.

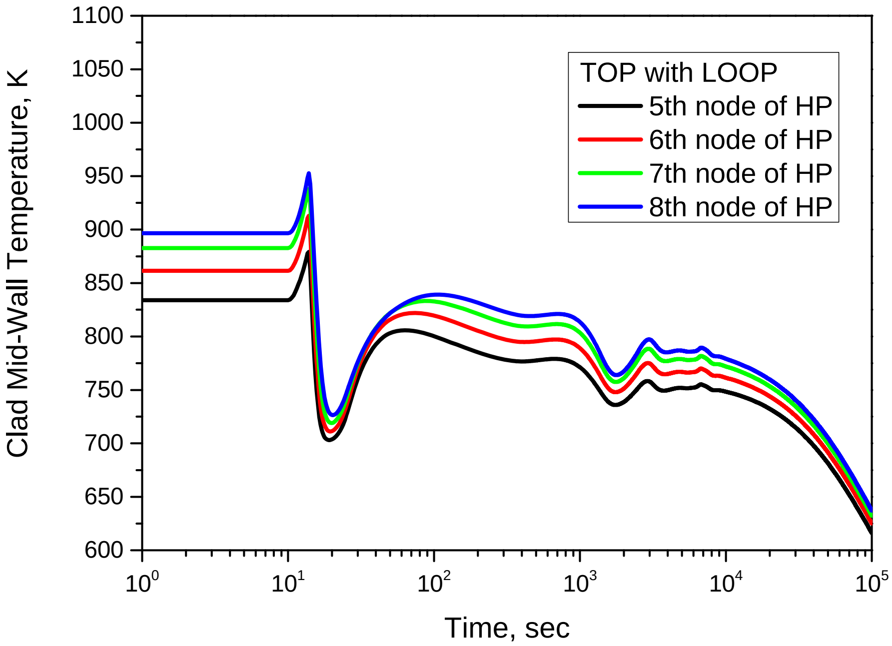

FIG. 3. Clad mid-wall temperature change

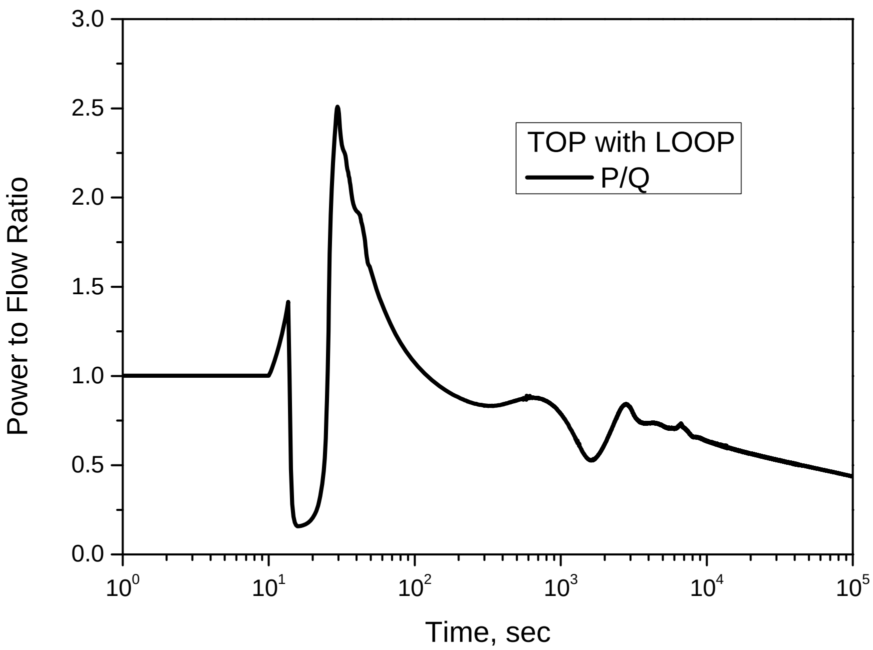

FIG. 4. Power to flow ratio

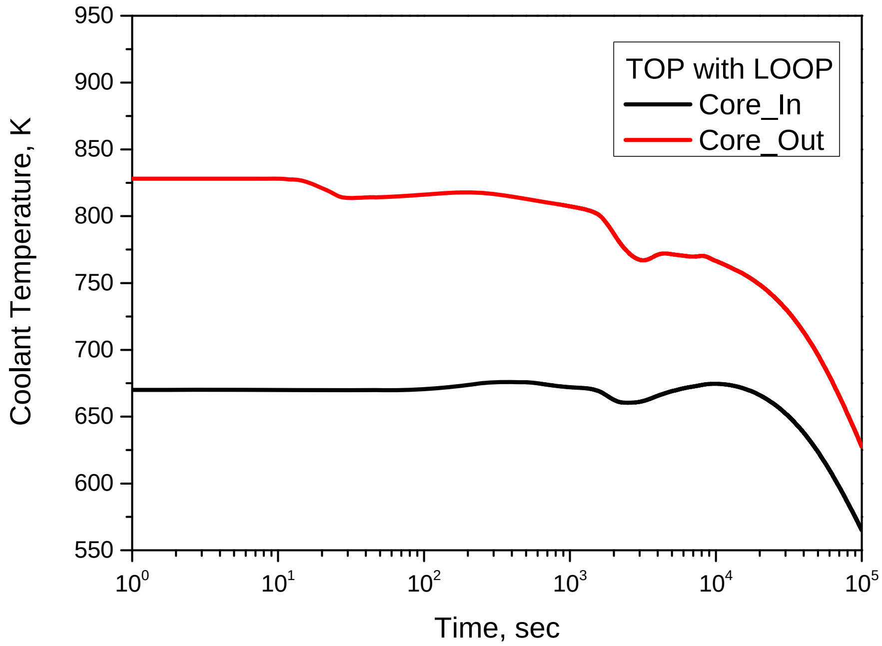

FIG. 5. Coolant temperature change

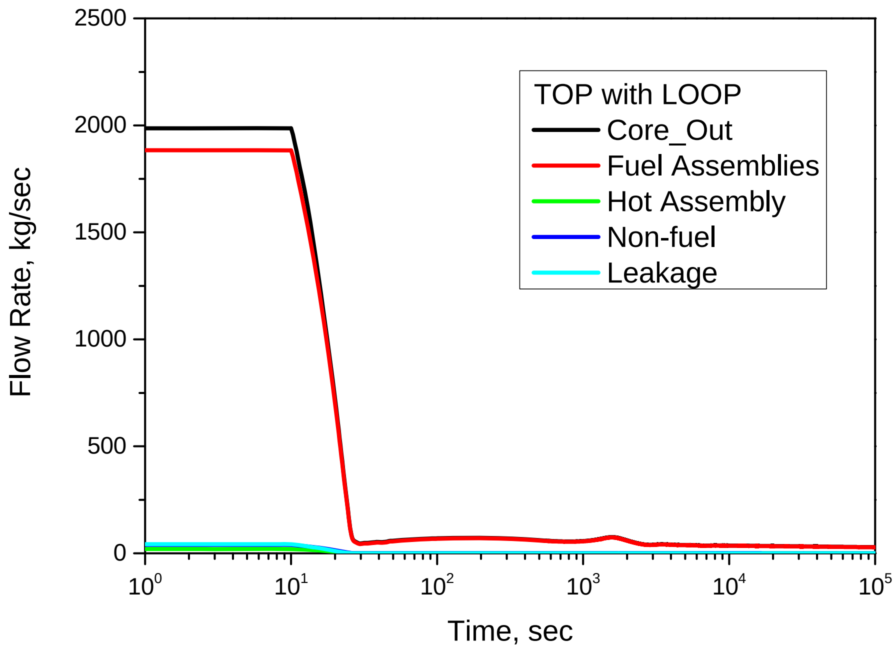

FIG. 6. Flow rates through core channels

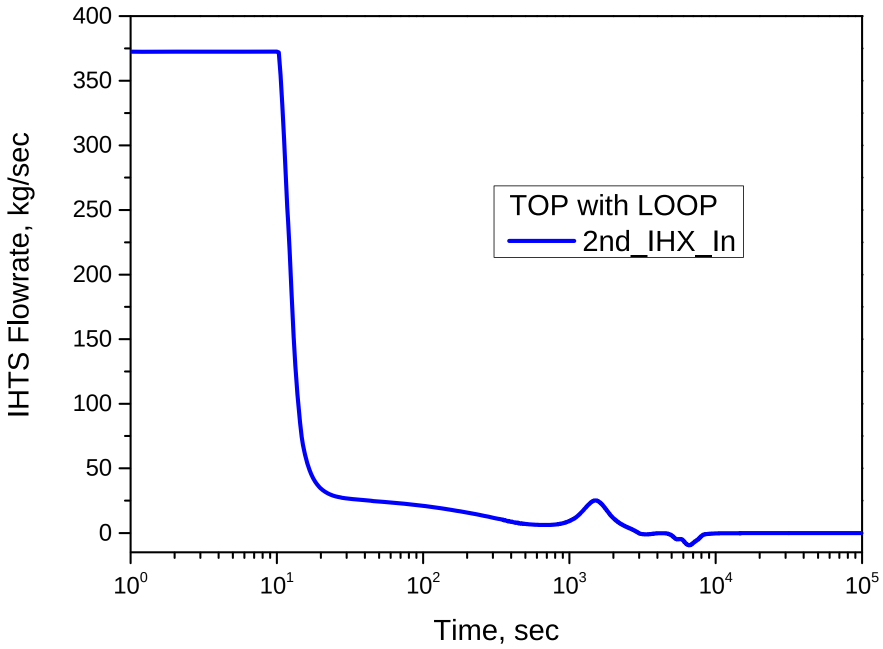

FIG. 7. Flow rates through IHTS

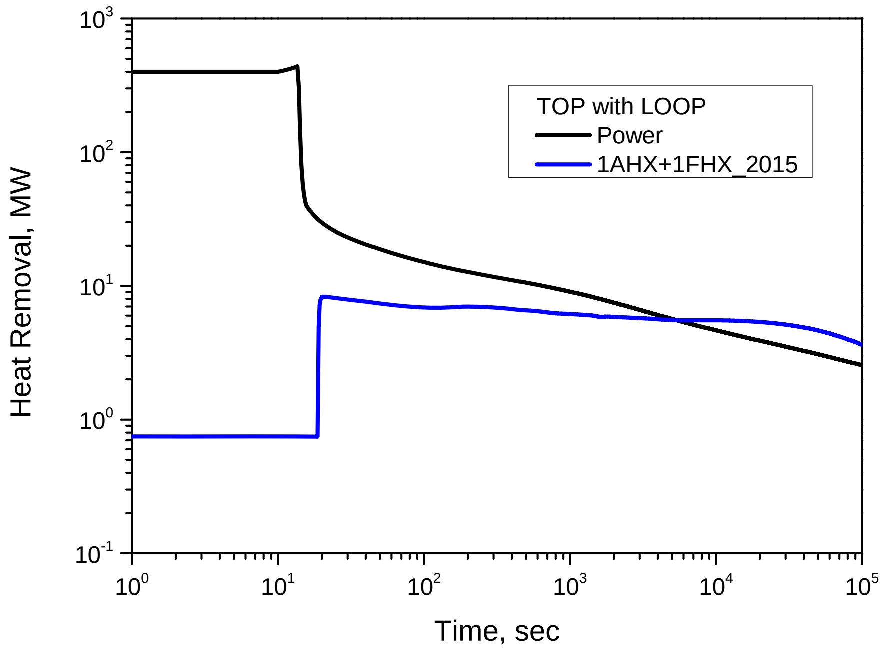

FIG. 8. Heat removals by DHRS compared with decay power

3.1. Sensitivity Results

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 III: THE RANGE OF SENSITIVITY PARAMETERS

Parameter	**Range**
Air flow rate	97%~100%
Air temperature	10℃~ 40℃
Doppler reactivity	-30% ~ 30%
Density reactivity	-35.4% ~ 35.4%

Figs. 9 to 11 show the results for the sensitivity calculations. As shown in the Figs, the clad temperature is not sensitive for the variation of the air temperature, the air flow rate, Doppler reactivity, and density reactivity in their uncertainty ranges. The variation of the air flow rate can affect more severely on the capacity of the decay heat removal systems. The less air flows into the AHX, the less heat removal is achieved, and then the larger clad temperature is calculated. But the uncertainty of the air flow rate in the PGSFR is just 3 %, which is proved by the AHX test facility.

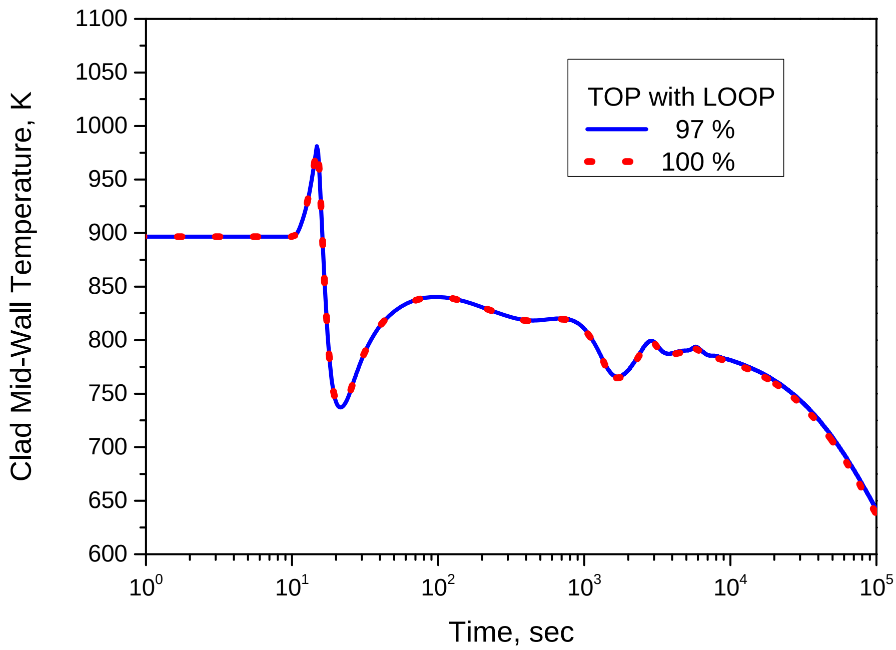

FIG. 9. Clad mid-wall temperature change versus air flow rate

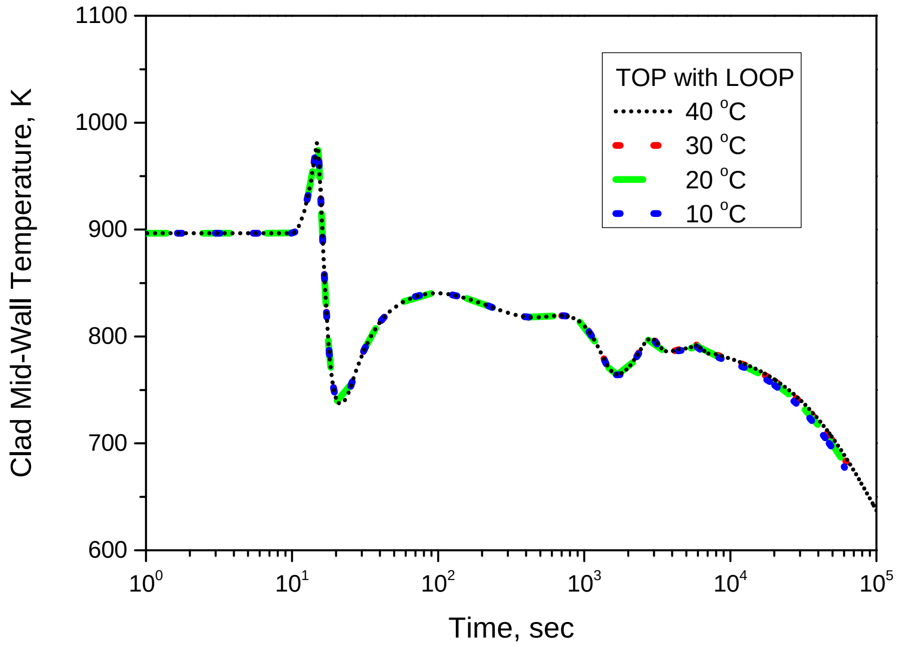

FIG. 10. Clad mid-wall temperature change versus air temperature

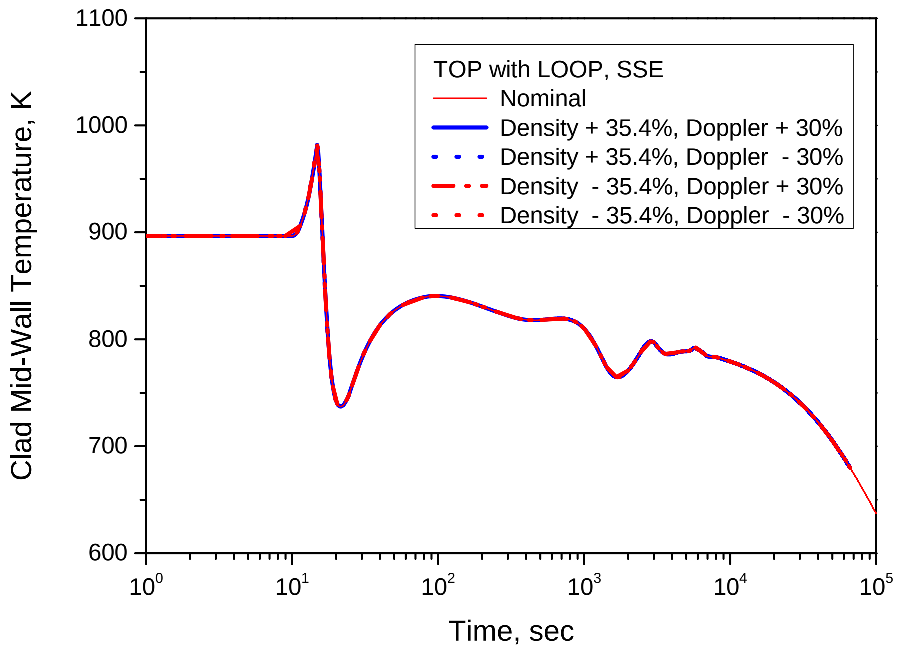

FIG. 11. Clad mid-wall temperature change versus reactivity feedback

4. Conclusion

In order to assess the inherent safety features of the PGSFR, a safety analysis was performed for the TOP accident with MARS-LMR and the sensitivity analysis was also performed to find the most conservative condition. As a result, the PGSFR was appropriately tripped by the RPS (Reactor Protection System) and cooled by the DHRS during the TOP. Besides, the clad temperature is not sensitive for the variation of the air temperature, the air flow rate, Doppler reactivity, and density reactivity in their uncertainty ranges. In conclusion, the preliminary specific design of PGSFR meets the safety acceptance criteria with a sufficient margin during the TOP event.

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