2014 PGSFR Safety Analysis for Loss of Flow

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1. Introduction

KAERI has been developing a conceptual design of the PGSFR (Prototype Gen-Ⅳ Sodium-cooled Fast Reactor) with the thermal power of 392.1 MWt, which is the pool type SFR (Sodium-cooled Fast Reactor) with metal fuel. The PGSFR consists of the PHTS (Primary Heat Transport System), the IHTS (Intermediate Heat Transport System), and the DHRS (Decay Heat Removal System).

A LOF (Loss Of Flow) accident has been investigated for a safety evaluation of the PGSFR using the MARS-LMR code. The safety analysis is evaluated by a CDF (Cumulative Damage Fraction). In case of the LOF accident, the tentative safety criterion is the CDF of under 0.05 [1].

2. Modeling and Results

2.1 PGSFR Input Modeling

Figure 1 demonstrates a nodalization for the MARS-LMR input with the PGSFR. The PHTS is placed in a large pool similar to the demonstration fast reactor. The IHTS transfers the reactor-generated heat from the IHX (Intermediate Heat eXchanger) of the PHTS to the SG (Steam Generator).

The IHTS consists of two loops, and each loop has two IHXs, one EM (Electro-Magnatic) pump, one expansion tank, and one steam generator. The SGs consists of two independent steam generation loops and converts the sub-cooled water to a super-heated steam by transferring the heat from the intermediate sodium to the water and steam.

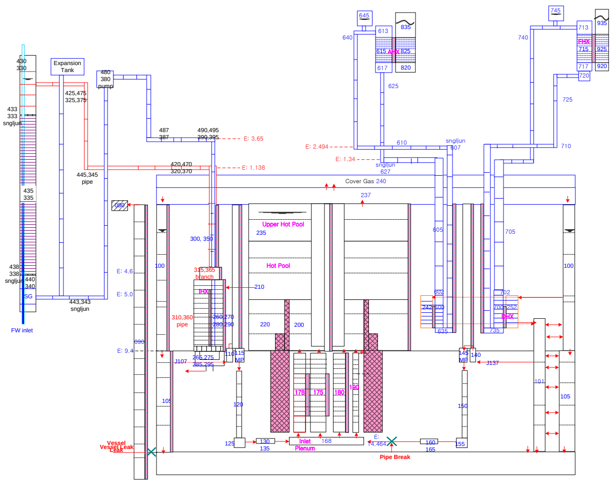

Fig. 1. Nodalization of PGSFR for MARS-LMR

The DHRS with the heat transfer capability of 10 MWt is composed of two units of PDHRS (Passive Decay Heat Removal System) and ADHRS (Active Decay Heat Removal System) and each loop is equipped with DHX (sodium-to-sodium Decay Heat eXchanger). In addition, a damper driven by the emergency generator (Diesel Generator) is attached to the AHX (Natural-draft sodium-to-air Heat Exchanger) and the FHX (Forced-draft sodium-to-air Heat Exchanger), which are even opened at the LOOP (Loss Of Off-site Power).

The event is assumed to start at of 102 % power condition of normal plant operating with HCF (Hot Channel Factor). The ANS-79 model is used for the core decay power after a reactor shut-down. It has been assumed that one PDHRS and one ADHRS are available by applying a single failure and a single maintenance criterion.

2.2 LOF Accident Scenario

The accident was initiated by both of Primary Heat Transport System (PHTS) pumps trip at 10 seconds in this present study. In addition, the LOOP was also assumed for a conservative point of view, and thus both of IHTS pumps trip and both of SG feed-water isolation are tripped at the same moment of PHTS pump trip.

2.3 LOF Accident Results

Figure 2 shows the coolant temperature behaviors

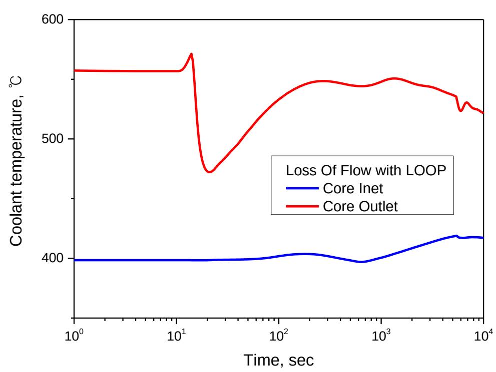

Fig. 2. Coolant temperature behavior for LOF

during the LOF accident. The core outlet temperature rapidly increases by both of PHTS pump trip at 10 seconds, and then, decreases nearly vertically after the reactor shut-down by trip signal of a power to flow-rate ratio at 13.5 seconds, and then the core inlet and outlet temperature rise due to both decreased mass flow-rate by the PHTS pump trip with coast-down during 16 seconds and the diminution of the heat transfer to the IHTS by the isolation of the feed water.

Figure 3 shows the decay heat removal rate of DHRS compared with the reactor power. The AHX dampers are assumed to open at 5 seconds after the reactor shut-down. The DHX heat removal of 5 MWt exceeds the core decay heat power of 5 MWt at about 5000 seconds, and the core outlet temperature decreases as shown in Fig. 2.

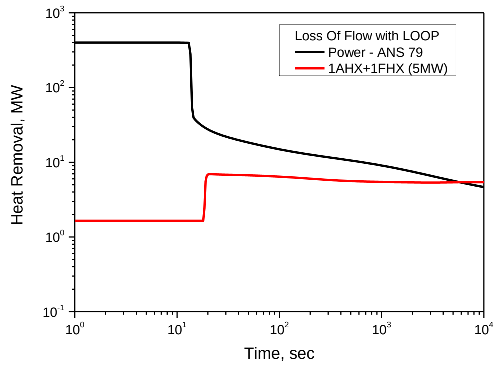

Fig. 3. Reactor power compared with DHRS heat removal for LOF

Figures 4 and 5 show the CDF and the peak cladding temperature behaviors with time, respectively. The increase of the peak cladding mid-wall temperature leads to the increase of the CDF. After the peak cladding temperature in Fig. 4 is decreasing by the reactor shut-down and DHX heat removal, the CDF is not increasing continuously as shown in Fig. 5.

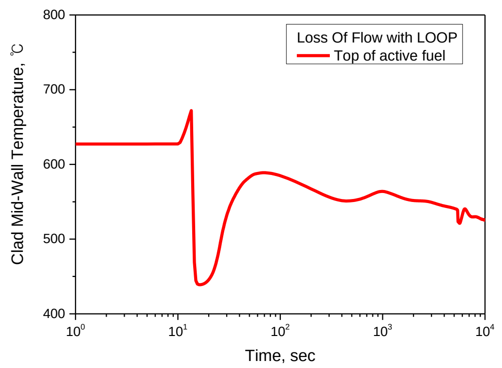

Fig. 4. Cladding peak temperature behavior for LOF

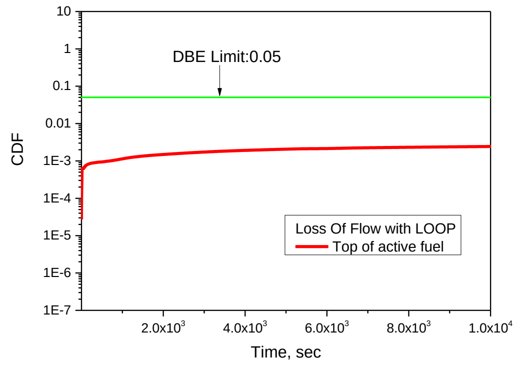

Fig. 5. CDF behavior for LOF

3. Conclusions

The LOF accident has been evaluated in the PGSFR using MARS-LMR. The accident was initiated by both of PHTS pump trip.

In the results, the CDF was predicted below a tentative safety criterion of 0.05 with a sufficient margin. The DHRS acceptably functioned for removing the core decay heat during long-term cooling period.

REFERENCES

[1] K. L. Lee et al., Safety evaluation for a preliminary specific design of the PGSFR in 2014, KAERI/TR-5905/2015, Korea Atomic Energy Research and Institute, 2015.