Safety Analysis for Loss of Flow related to Loss of Off-site Power in 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 392.2 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 consists of the PHTS (Primary Heat Transport System), the IHTS (Intermediate Heat Transport System), the SGs (two Steam Generators), and the DHRS (Decay Heat Removal System). The PGSFR has an inherent safety features to have a negative power reactivity coefficient during all operation modes. It has a passive safety characteristic due to the design of the DHRS.

In this study, a LOF (Loss Of Flow), one of the most important accidents in DBEs (Design Basis Events), has been investigated for the PGSFR using the MARS-LMR code. Furthermore, the effect of LOOP (Loop Of Off-site Power) has been investigated, and CDF (Core Damage Fraction) of safety criteria is applied to confirm the safety margin.

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. 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 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.

The DHRS with the heat transfer capability of 10 MWt is composed of two units of PDHRS (Passive Decay Heat Removal System) and two units of 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.

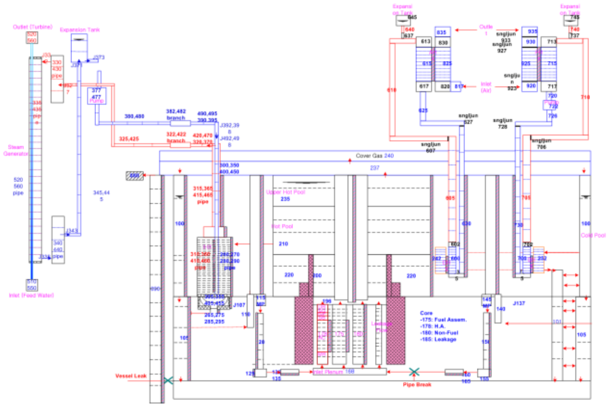

Fig. 1. Nodalization of PGSFR for MARS-LMR

In safety analysis of LOF, conservative approach, which is 102% of power condition with HCF (Hot Channel Factor), ANS-79 decay power model [1], 5.0 seconds delay in opening of AHX and FHX dampers, and LOOP is taken into account. Additionally, one PDHRS and one ADHRS are available in accordance with a single failure and a single maintenance criterion.

2.2 LOF Accident Scenario

The accident was initiated by stop of both PHTS pumps 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 and both of SG feed-water isolations are tripped at the same moment of PHTS pump stop.

2.3 LOF Accident Results

Figure 2 shows the coolant temperature behaviors during the LOF accident. Reactor is tripped at 14.36 seconds right after PHTS pump stop of both PHTS pumps at 10 seconds. The outlet coolant temperature

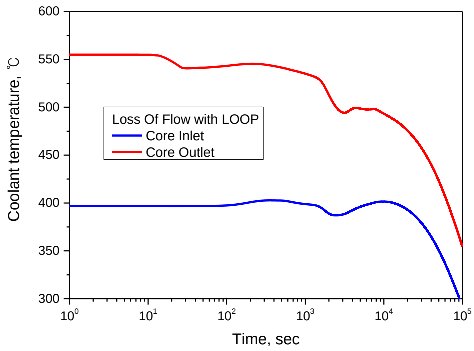

Fig. 2. Coolant temperature behavior for LOF

decreases nearly vertically after the reactor shut-down by trip signal of a power to flow-rate ratio at 13.36 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. 3.

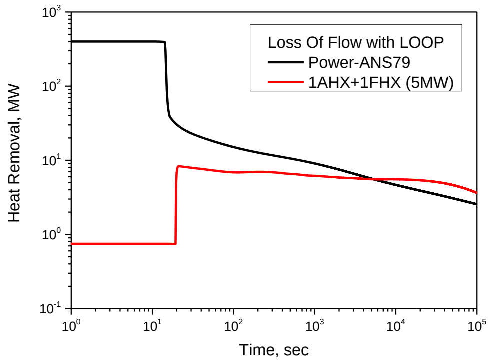

Fig. 3. Reactor power compared with DHRS heat removal for LOF

Figures 4 and 5 show the peak clad temperature and the CDF behaviors with time, respectively. CDF [2] in MARS-LMR can be defined by Eqs. (1)-(3):

$$CDF = \int_{t=0}^{t=t} \frac{1}{t_r} dt \qquad (1)$$

$$t_r = \theta \exp \frac{Q}{\mathrm{R}} \cdot \frac{1}{T} \qquad (2)$$

$$\begin{aligned} \mathrm{In}\theta &= -34.8 + \tanh \frac{\sigma - 200}{50} \\ &+ \frac{12}{1.5 + 0.5\tanh \dfrac{\sigma - 200}{50}} \mathrm{In}\left[\mathrm{In} \frac{730}{\sigma}\right] \\ &- 0.5\left[1 + \tanh \frac{\sigma - 200}{50}\right] \\ &\cdot \left\{0.75\left[1 + \tanh\left(\frac{\dot{T} - 58}{50}\right)\right]\right\} \end{aligned} \qquad (3)$$

where, t_r is a rupture time in second, σ is a hoop stress (MPa), T is transient temperature (K), $\dot{T}$ is heating rate (K/sec), activation energy Q is 70170 (cak/mole), gas constant R is 1.986 (cal/mole/K).

The increase of the peak clad mid-wall temperature leads to the increase of the CDF. After the peak clad 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.

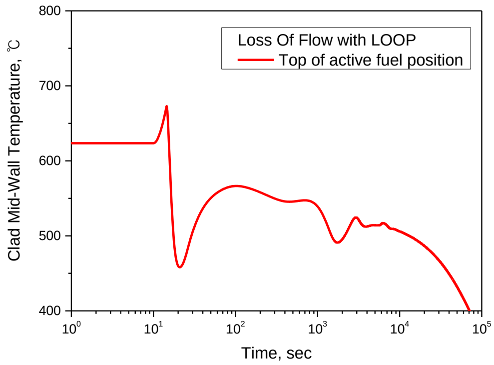

Fig. 4. Clad peak temperature behavior for LOF

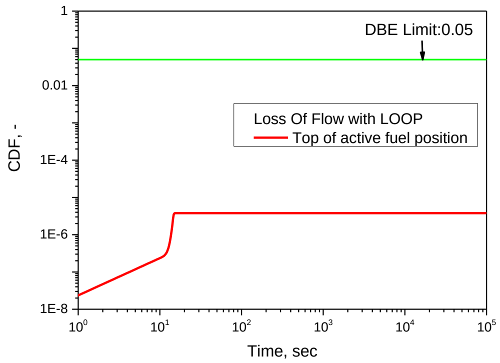

Fig. 5. CDF behavior for LOF

2.4 Effect of LOOP at LOF Accident

In order to be conservative, LOOP has been adopted to the safety analysis. Effect of LOOP at LOF accident has been investigated in this section. In case of LOF accident with non-LOOP, the accident was initiated by PHTS pump stop at 10 seconds in this present study.

Figure 6 shows the peak clad temperature behaviors with time for LOF with LOOP and non-LOOP, respectively. The fist peak clad temperature in LOF with LOOP and non-LOOP rapidly increases by stop of both PHTS pumps at 10 seconds. The first peak clad temperature in LOF with LOOP and non-LOOP decreases nearly vertically after the reactor shut-down by the trip signal of a power to flow-rate ratio. The second peak clad temperature in LOF with non-LOOP does not increase as higher as that in LOF with LOOP, because the SG feed-water is not isolated in LOF without LOOP.

Figure 7 shows the CDF behaviors with time for LOF with LOOP and non-LOOP, respectively. The CDF value in LOF with LOOP is slightly higher than that in LOF with non-LOOP.

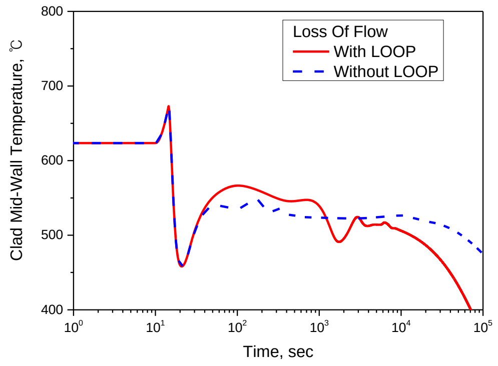

Fig. 6. Clad peak temperature behaviors for LOF with LOOP and non-LOOP

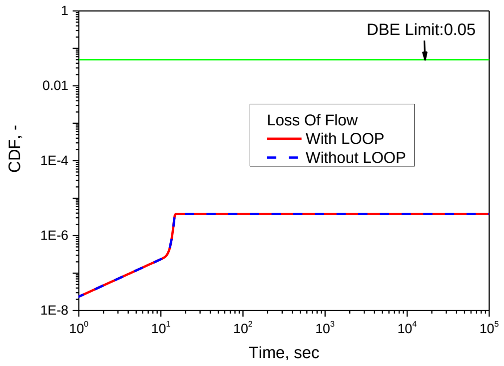

Fig. 7. CDF behaviors for LOF with LOOP and non-LOOP

3. Conclusions

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

In the results, the CDF was calculated below a safety criterion of 0.05 with a sufficient margin. The DHRS acceptably functioned for removing the core decay heat during long-term cooling period. Furthermore, it has been elucidated that LOF with LOOP is more conservative than LOF with non-LOOP.

ACKNOWLEDGEMENTS

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[1] American Nuclear Society, Decay Heat Power in Light Water Reactors, an Americal National Standard, ANSI/ANS-5.1, 1979.
[2] C. W. Choi et al., Supplement of Cumulative Damage Function in MARS-LMR, SFR-960-DS-486-001, 2014.