Sensitivity Study on LOHS Analysis for a demonstration SFR

Won-Pyo Chang, Kwi-Seok Ha, Kwi-Lim Lee, Seok-Hun Kang, Chiwoong Choi, Hae-Yong Jeong
Korea Atomic Energy Research Institute
P.O.Box 105, Yuseong, Daejeon, 305-600, Korea
**Corresponding author: wpchang@kaeri.re.kr*

1. Introduction

A LOHS(Loss Of Heat Sink) accident was analyzed for an safety evaluation of the conceptual design of a demonstration SFR (**S**odium cooled **F**ast **R**eactor) with 600 MWe. The accident initiator in the present analysis was an inadvertent isolation of the feed-water valves among various causes. In addition, the loss of off-site power was also postulated in the analysis for a conservative point of view, and thus each two pumps in both the primary and intermediate systems would be tripped in the transient. The DRC(Decay heat Removal Circuit) is the unique safety grade system for removing decay heat. Therefore, the study was aimed to evaluate whether the core damage in the accident could be prevented within the designed DRC capacity.

2. Analysis

2.1 Input model and accident simulation

Figure 1 demonstrates a nodalization for the MARS-LMR input with the demonstration SFR [1]. The safety grade decay heat removal system, DRC consists of two loops of a passive system and two loops of an active system. Since one emergency diesel generator out of two was assumed to malfunction in the analysis, only an active loop with the two passive loops would be ready to provide the service in the accident scenario. Besides, the pumps in the primary and secondary systems were tripped and the air control valves for the AHX(Air Heat Exchanger) and FAHX(Forced Air Heat Exchanger) were opened at the loss of off-site power.

2.2 LOHS analysis

2.2.1 Accident scenario

The accident was initiated by the feed-water isolation signal at 10.0 s. The IHX(Intermediate Heat Exchanger) inlet and outlet temperatures went up due to a sudden absence of heat removal through the SGs. The reactor trip signal was caused by a high IHX inlet temperature with a delay of 8.5 s, and then the reactor was tripped 0.01 s later. The reference case for the analysis was based on the nominal power and primary flow without a reactivity coefficient change when the loss of off-site power took place with a time delay of 2.5 s after the reactor trip.

2.2.2 Results

The analysis results exhibited that the IHX set-point was reached at 50.80 s, and then the reactor trip signal occurred at 59.30 s. Subsequently, the reactor was tripped at 59.31s. The pump coast-down began at 61.83 s resulting from the loss of off-site power.

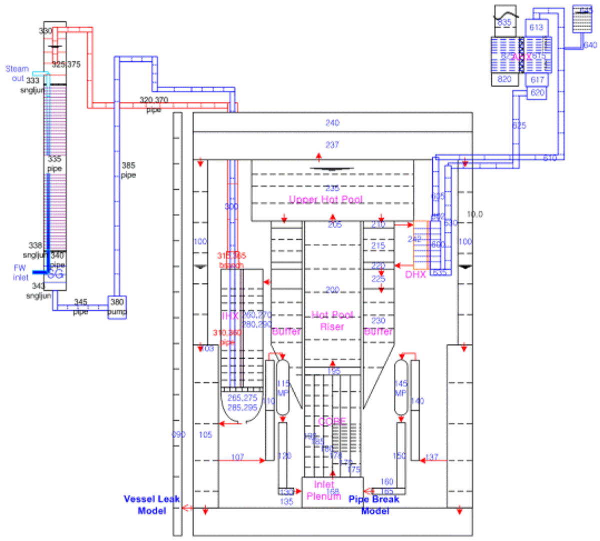

Figure 1 Nodalization for MARS-LMR to the demonstration SFR

Figure 2 displays results for the core inlet and outlet temperatures. The core outlet temperature which showed a peak just before the reactor trip plummeted as the power fell down. After the core outlet temperature showed the second peak about 220 s, it turned into a descending trend with increase of the natural circulation flow in the primary system as long as coolant in the SGs was available. Meanwhile, the core inlet temperature after the second peak continuously increased until actual DHX heat removal would be available as a result of the reduced heat transfer through the IHX.

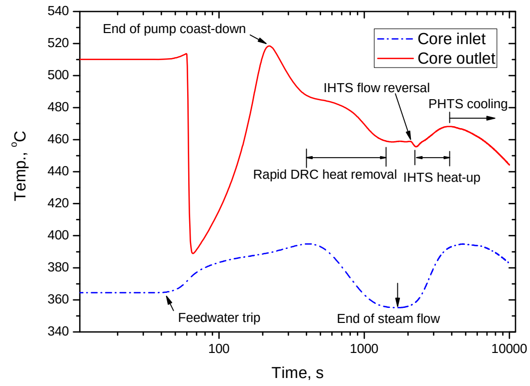

Figure 2 Behaviors of core inlet and outlet temperatures

Figure 3 represents the cladding temperatures for 4 axial nodes. The maximum temperatures in the 8th axial node at 59 s, sharply fell due to the reactor trip. While the maximum temperatures largely depended on the core power before the reactor trip, sodium coolant temperature played a dominant role later on.

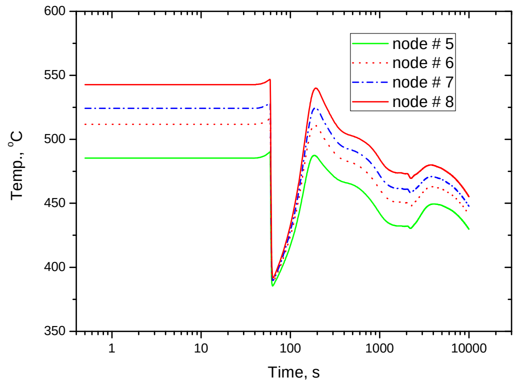

Figure 3 Axial cladding temperatures

2.3. Sensitivity analysis results

Sensitivity analysis was performed such parameters as the core power, the core flow, the reactivity coefficients, and the timing of the loss of off-site power after the reactor trip.

2.3.1 Sensitivity study on the core power

Sensitivity study was made on 98, 100, and 102 % of the nominal core power to take into account the uncertainty. Figure 4 compares those 3 cases for the maximum cladding temperature which was calculated in the 8th axial node. The highest temperature was found for the case of 102 % of the nominal power as expected, and the temperature was estimated at 551.4 ^{o}C. The maximum temperatures for 100 % and 98 % were 546.9 ^{o}C and 542.4 ^{o}C, respectively. The temperature difference was less than 5 ^{o}C. The difference was not serious considering the margin to the safety limit.

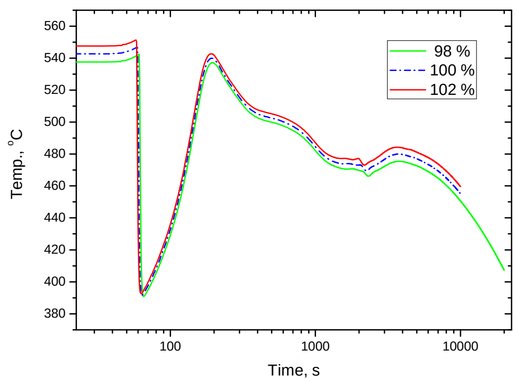

Figure 4 Result of sensitivity study on core power

2.3.2 Sensitivity study on the core flow

The evaluation was made on 95 and 100 % of the nominal core flow. Figure 5 compares the results for the two cases, but a noticeable difference was not seen in the results.

2.3.3 Sensitivity study on the reactivity coefficients

The evaluation was made on $\pm$ 20 % of the nominal value. Almost no difference was found. The uncertainty on the reactivity coefficients is no more concern in the analysis.

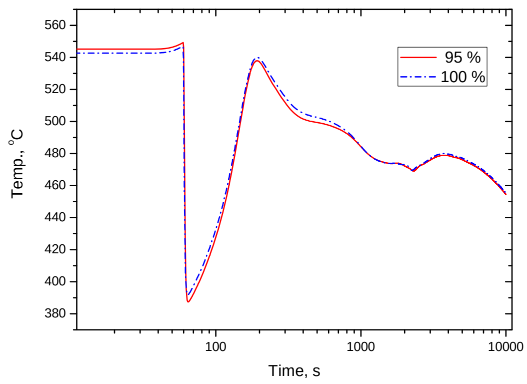

Figure 5 Result of sensitivity study on core flow

2.3.4 Sensitivity study on the timing of a loss of off-site power

The scenario with the loss of off-site power could be essential to the accident result because it leads to pump trip. The analyses, thus, were performed with the occurrence time delay of 2.5 s and 5.0 s including 0.0 s. The results for the cladding temperature, however, were almost same, because coolant temperature turned to be affected by the P/Q(Power to Flow ratio) more dominantly than the core flow in the early transient. It could be said that the loss of flow timing did not affect the results.

3. Conclusion

Analyses were performed for a loss of heat sink accident in the demonstration SFR using the MARS-LMR. Sensitivity study was also carried out on the parameters such as the core power, the core flow, the reactivity coefficients, and the timing of the loss of off-site power.

In result, the most sensitive parameter was the core power, and the core flow followed the next. The timing of the loss of off-site power negligibly affected the result. The highest cladding temperature was calculated at 553.6 ^{o}C with the core power of 102 %, the core flow of 95 %, and the reactivity coefficients of +20 %.

The safety limits for the fuel centerline and cladding temperatures are 950 ^{o}C and 650 ^{o}C, respectively. Therefore, the estimated maximum fuel centerline and cladding temperatures were sufficiently lower than the respective safety limit.

REFERENCES

[1] K.S. Ha, “Input description of the Design Basis Events for the demonstration SFR,” SFR-SA310-WR-01-2010Rev.00 (2010).