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Effects of primary trip parameter's failure during the feedwater line break in SALUS

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1. Introduction

In order to ensure the safety of the nuclear power plant, critical safety principles such as defense-in-depth and diversity are taking into accounts from the initial design stage of the plant. In the design of reactor protection system, diversity can be implemented in various ways, such as sensing different parameters, using different technologies, using different logic or algorithms, or using different actuation means to provide several ways of detecting and responding to a significant event [1]. It is also described that diversity is complementary to the principle of defense-in-depth and increases the chances that defenses at a particular level or depth will be actuated when needed.

As one of the procedure to verify the reliability of the protection system in the SALUS(Small, Advanced, Long-cycled and Ultimate Safe sodium-cooled fast reactor), which is a 100MWe long fuel cycle sodium-cooled fast reactor system under development in KAERI, a series of calculations for representative Anticipated Operational Occurrences (AOOs) have been done to identify the diversity of trip parameters using the MARS-LMR code[2]. As part of subsequent research, an evaluation of the diversity of trip variables for design basis accidents is underway. In this research, the effects of a failure in the primary trip parameter on the thermal-hydraulic behavior and reactor safety were analyzed for a Loss of Heat Sink (LOHS) accident resulting from feedwater line break in feedwater system.

2. Analysis Methods

2.1 Trip parameters and safety acceptance criteria

The following sentence related to signal diversity can be found in NUREG-series publication[1].

Signal diversity is the use of different sensed parameters to initiate protective action, in which any of the parameters may independently indicate an abnormal condition, even if the other parameters fail to be sensed correctly.

The list of trip parameters and related nominal set point can be found in ref [2]. The safety acceptance criteria for Design Basis Accidents category I(DBA-I) is shown in Table 1.

Table 1 Safety acceptance criteria for DBA-I

Frequency/r·y	Offsite Radiological Consequence	Fuel, Cladding, Structure, Containment Damage Limit
1E-4 ≤ F < 1E-2	10% of 10CFR100.11	- No fuel melting - A small fraction of fuel pin failure, CDF_{each} < 0.05 - Core coolability - ASME Service Level C - Maintain design leakage rate

2.2 Analysis assumptions

A pipe rupture in the feedwater system was assumed. The MARS-LMR code was used for the evaluation. Single failure of safety components designed by single failure criteria was considered in the event sequences. It was also assumed that a loss of offsite power occurred immediately upon a reactor trip. The PHTS, IHTS, and the feedwater pumps were assumed to stop at the time of the reactor trip.

To perform a conservative analysis, the most conservative conditions among the Limiting Conditions for Operation (LCO) were selected for core inlet temperature, flow rate, and power output for a conservative evaluation. For core inlet temperature and power output, the upper limit of the LCO (core inlet temperature: 372°C, core power output: 272.34 MWt (102%)) was selected, and for flow rate, the lower limit (1256.4 kg/s) was chosen.

To select the conservative reactivity feedback combination, a sensitivity analysis was conducted on the reactivity combination based on the individual reactivity information for each fuel cycle.

To identify the primary and secondary reactor trip variables, the MARS-LMR code calculations were performed up to 10,000 seconds and repeated twice[2,3]. The primary reactor trip variable was identified in the 1st calculation. In the 2nd calculation, the failure of the primary trip variable, which had been identified in the 1st calculation was assumed.

3. Results

The considered reactivity feedback mechanism in the core were Doppler effect, sodium density, fuel axial expansion, core radial expansion, and control rod drive line/reactor vessel (CRDL/RV) expansion. To consider the various reactor condition, different reactivity worth was derived for both the beginning of cycle (BOC) and end of cycle (EOC) conditions. A sensitivity analysis was conducted to identify the respective contribution to fuel peak temperature. The small and the large absolute value was designated as LST and MST, respectively, among BOC and EOC. Based on these individual reactivity data, a total of 32 conditions for the core were constructed for sensitivity analysis.

Figure 1 shows the averaged fuel peak temperature with respect to individual reactivity feedback on fuel peak temperature. As shown in the figure, the large absolute value (MST) of sodium density and CRDL/RV expansion contributed more to the increase in fuel peak temperature compared to the small absolute value (LST). The LST values of the others were found to contribute more to the increase in fuel peak temperature. Therefore, the aforementioned reactivity feedback combination was selected as a conservative set.

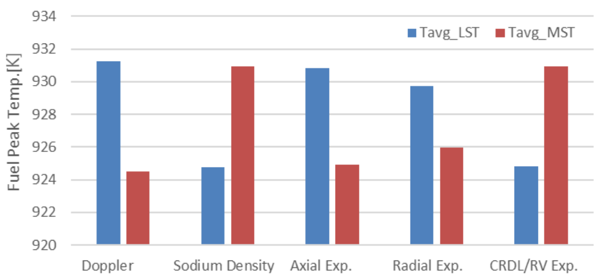

Figure 1 Dependency of fuel peak temperature on individual reactivity feedback

Figure 2 shows the variation of core inlet temperature (CIT) and central subassembly outlet temperature (CSOT), which were found to be the primary and secondary trip parameters from 1st and 2nd calculation, respectively.

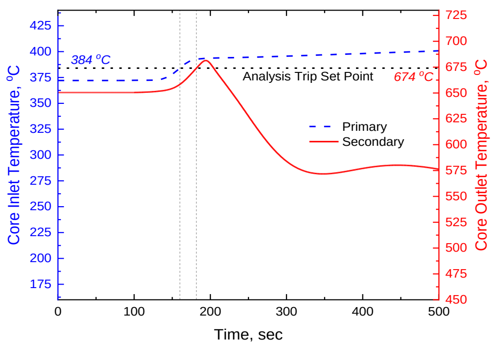

Figure 2 Primary and secondary trip parameter

The Primary and secondary parameters for ESFAS were found to be the same as those of the trip parameters in this scenario. Secondary trip is available about 21.7 seconds after primary trip failure. The time difference is evaluated by comparing the time of the reactor trip actuation between 1st and 2nd calculation.

Figure 3 and 4 shows the variation of averaged core inlet and outlet temperature and cumulative damage fraction (CDF) induced by primary trip failure under feedwater line break in feedwater system. As shown in figure, the effect of primary trip parameter's failure on reactor behavior was very limited in this scenario. Therefore, the increase of CDF was also confined and the CDF by secondary reactor trip variable has enough margin against safety acceptance criteria.

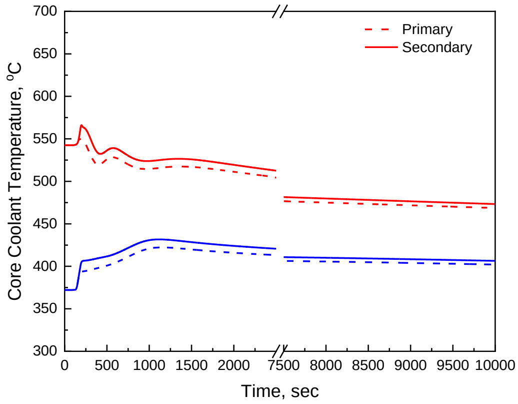

Figure 3 Effects on averaged core temperature

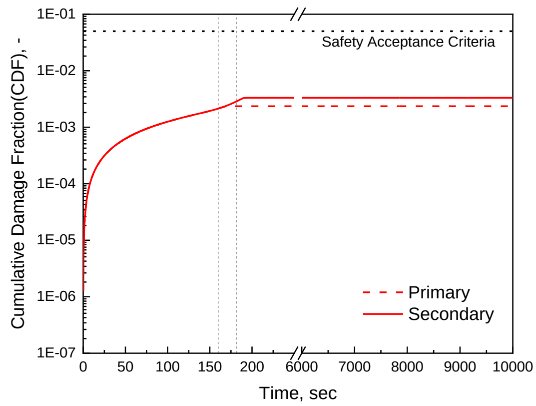

Figure 4 Effects on CDF

3. Conclusions

The effects of primary trip failures were evaluated during feedwater line break accident using MARS-LMR code. The conservative reactivity feedback combination for the accident was derived based on sensitivity analysis. The effects of time delay in trip and ESFAS actuation due to the failure of primary trip parameter were found to be limited. The HCSOT was found to be a valid secondary trip parameter for feedwater line break accident.

4. Acknowledgement

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References

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