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Evaluation of Multiple SGTR at the innovative SMR using SPACE Code

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1. Introduction

A steam generator tube rupture (SGTR) accident in a nuclear power plant can occur due to stress corrosion cracking (SCC) caused by tube corrosion during operation or by foreign material ingress from external sources. A multiple steam generator tube rupture (MSGTR) accident is defined as the simultaneous rupture of multiple tubes within a single steamgenerator which results in a greater release of reactor coolant compared to a single-tube rupture. According to SECY 93-087 [1], the U.S. Nuclear Regulatory Commission(NRC) considers the rupture of 2–5 tubes in passive pressurized water reactors (PWRs) and recommends demonstrating the integrity of accident mitigation measures, given that accident progression in such cases occurs more rapidly than in a single-tube rupture scenario.

Furthermore, the Korea Institute of Nuclear Safety (KINS) regulatory guide 16.1 [2] stipulates that evaluations should be performed assuming the simultaneous rupture of five tubes in a single steam generator. SMART-100, the first pressurized water–type small modular reactor (SMR) in Korea to obtain Standard Design Approval (SDA), assessed MSGTR events in its Standard Safety Analysis Report (SSAR) [3 using a best-estimate methodology. Considering the helical steam generator design characteristics of the innovative small modular reactor (i-SMR), the thermal and dynamic loads are expected to be lower than those of large PWR steam generator tubes, the analysis in this study adopts the rupture of five tubes in accordance with regulatory guide 16.1

2. Description of the i-SMR

As illustrated in Figure 1, the i-SMR adopts an integral reactor configuration to eliminate the possibility of a large-break loss-of-coolant accident (LBLOCA) [4]. The reactor coolant system (RCS) flow path proceeds upward through the core and the upper riser region within the core support barrel (CSB), and then downward through the reactor coolant pumps (RCPs) located on the side of the steam generator (SG) shell. The flow subsequently returns to the core via the lower plenum. The coolant is forcibly circulated by four RCPs positioned at the upper section of the reactor vessel. Heat generated in the core is transferred to the secondary system through monobloc, once-through, helical coil steam generators.

The control rod drive mechanism (CRDM) is centrally located on the reactor vessel closure head (RVCH), and the major components of the nuclear steam supply system (NSSS) including the RCPs, SGs, and pressurizer (PZR) are all housed within a single reactor vessel. In accordance with the design characteristics of the i-SMR, the four feedwater and steam pipes penetrating the containment vessel are each divided into two lines, resulting in a total of eight feedwater nozzles and eight steam nozzles. These connect to a single steam generator located within the reactor vessel. This steam generator, which produces superheated steam, is equipped with eight feedwater headers and eight steam headers. The feedwater and steam nozzles are connected to the corresponding headers via eight internal feedwater lines and eight steam lines, respectively. In the steam generator, the reactor coolant flows on the shell side, i.e., outside the heat transfer tubes, while the secondary feedwater flows inside the tubes.

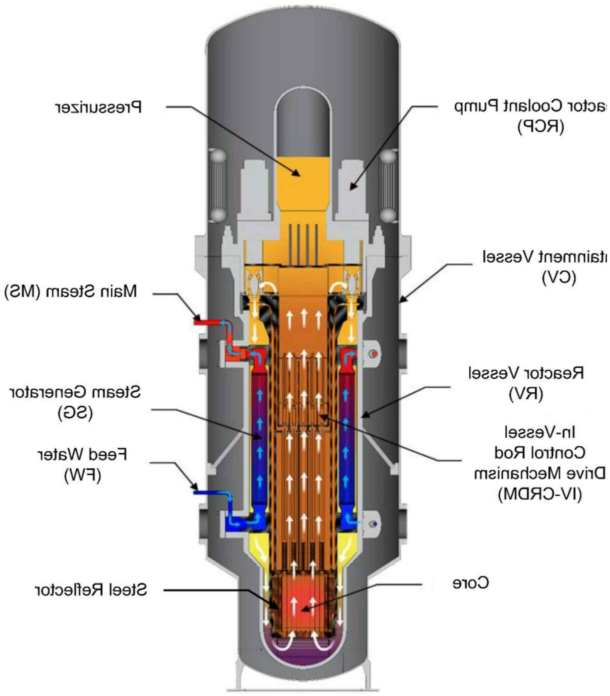

Fig. 1. Configuration of i-SMR

3. Modeling of Multiple SGTR

To simulate the multiple SGTR accident, the break area was modeled assuming the rupture of five tubes. As shown in Figure 2, the steam generator tubes were reconfigured into separate nodes representing the intact side and the ruptured side. Unlike in conventional large PWRs, the SGTR flow path in the i-SMR is directed from the downcomer toward the steam generator tubes. To ensure conservative conditions, the rupture location was modeled at the region with the highest temperature.

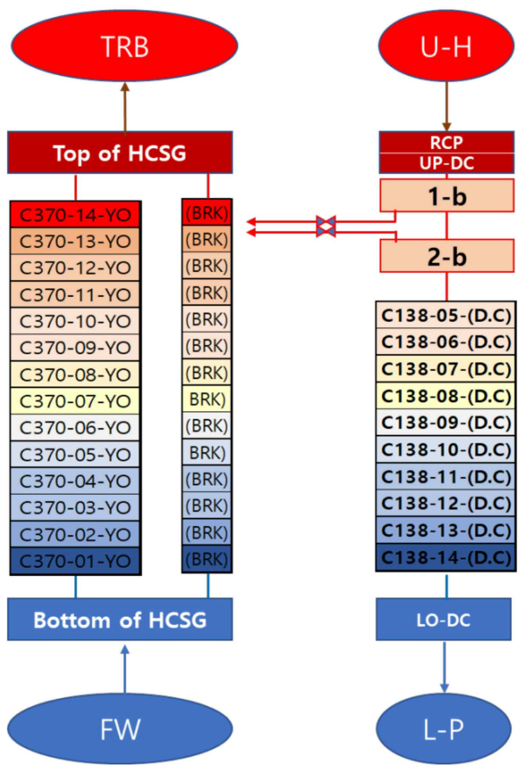

Fig. 2. Nodalization of break model

4. Calculaation

This section discusses the calculation results for the multiple SGTR accident in the i-SMR. The reactor was assumed to operate at 100% full power at the onset of the event, with a simultaneous loss of offsite power (LOOP) occurring at reactor trip. All control systems were assumed to operate in automatic mode, and no operator actions were considered. In addition, due to the design characteristic of the i-SMR in which the secondary system design pressure is identical to that of the primary system, MSSVs are not installed. Consequently, the influence of MSSVs was not evaluated. The analysis instead focused on maintaining reactor coolant inventory and ensuring long-term decay heat removal capability. The physical behavior and event sequence of accident are discussed in Section 4.2.

4.1 Steady-state

The calculations were performed using the SPACE code [5] based on a best-estimate methodology and assumptions. The reactor coolant system included the core, pressurizer, reactor coolant pumps, helical steam generators, as well as the major safety and control systems of the NSSS. The initial conditions and other reactor design parameters for this model were selected to be appropriate for transient analysis. The steadystateresults, which served as the initial conditions for the transient simulations, are summarized in Table 1.

Table I. Initial condition for transient calculation

Parameter	Values
Core power [MW]	520.0
Primary system pressure [MPa]	15.5
Primary system inlet temp. [K]	543.65
Primary system outlet temp. [K]	573.85
Secondary system pressure [MPa]	4.34

4.2 Transient state

The calculation results for the MSGTR accident in the i-SMR are presented in Figures 3–6, with the event sequence summarized in Table 2. (Figure 3: RCS and SG pressures, Figure 4: integrated break flow, Figure 5: pressurizer water level, Figure 6: RCS temperature)

At the early stage of the SGTR accident, the rapid decrease in RCS pressure triggers a reactor trip signal due to low pressurizer water level. In this analysis, the reactor trip occurred at 69 sec. Since a LOOP was assumed to coincide with reactor trip, both main feedwater and main steam lines were isolated by the closure of the isolation valves, while the PAFS valves were opened. Continuous leakage of coolant through the ruptured tubes into the isolated secondary side caused the secondary system pressure to increase until it reached equilibrium with the RCS pressure, leading to a sharp reduction in break flow. In this analysis, it was confirmed that the SG peak pressure time was almost similar to the time when it was synchronized with the RCS pressure.

Table II. Chronology of the transient main event

Event	Time [s]
Break	0.0
Low Pressurizer Level	68
Reactor & Turbine Trip	68
LOOP, RCP coast down	68
MFIV,MSIV isolation signal	68
PAFS Actuation	69
Peak pressure of SG	197
Safety Shutdown	11,201
Calculation end	36,000

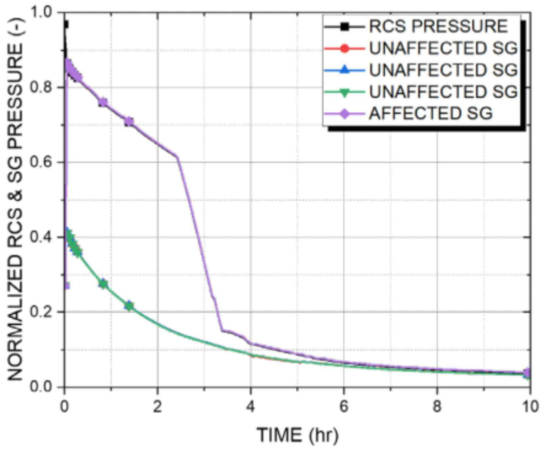

Fig. 3. RCS and SG Pressure

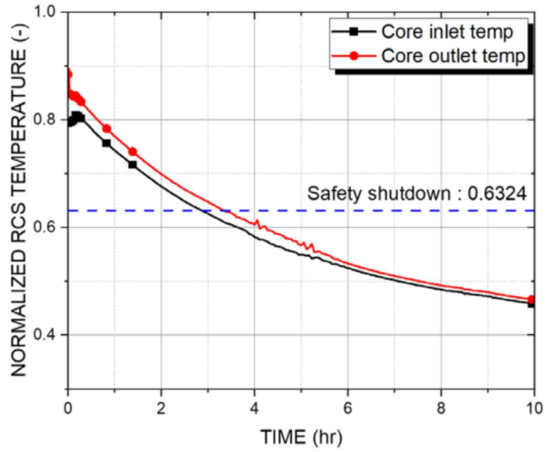

Fig. 6. RCS temperature

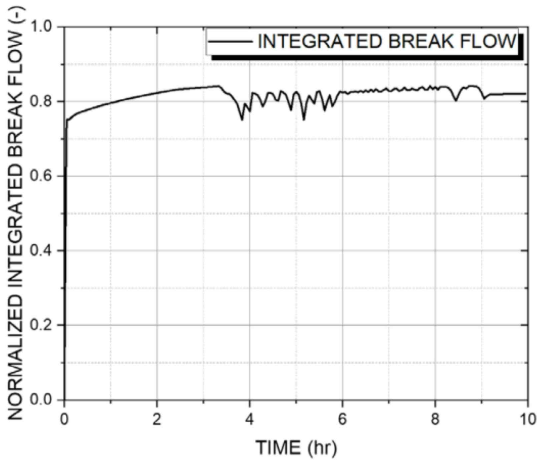

Fig. 4. Integrated break flow

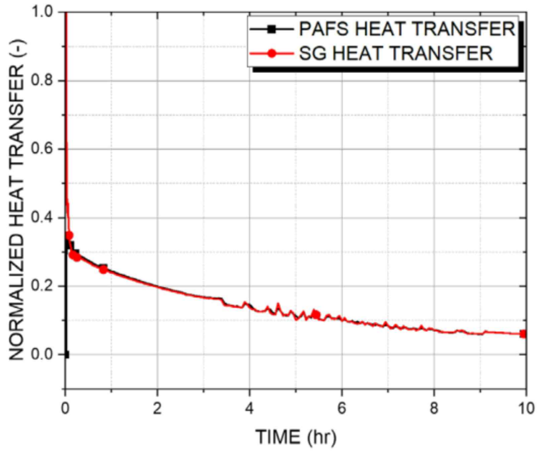

Fig. 7. Heat flux of PAFS

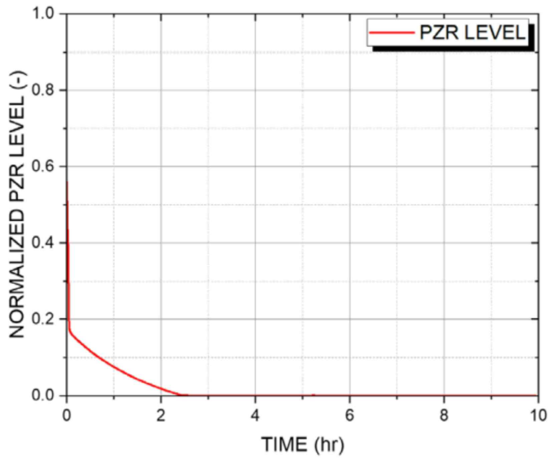

Fig. 5. PZR collapsed water level

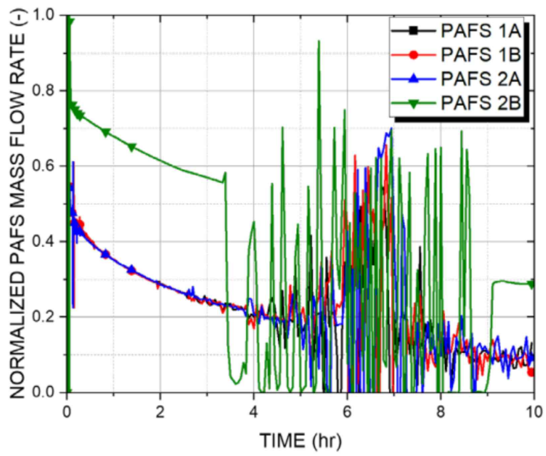

Fig. 8. Mass flow rate of PAFS

After reactor trip, with the secondary side isolated,the secondary pressure increased rapidly to a peak and subsequently equalized with the RCS pressure. In the MSGTR scenario, pressure equalization occurred at approximately 175 sec, accompanied by a significant reduction in break flow. The pressurizer water level dropped rapidly following reactor trip and eventually depleted. Once the pressure difference between the primary and secondary systems reached equilibrium,the RCS inventory was maintained, enabling continuous cooling by the PAFS and preventing core damage. The closed-loop flow path established by the PAFS sustained long-term heat removal from the RCS via the passive cooling system, as shown in Figures 7 and 8.

Approximately 2.5 hours, the observed sharp decrease in RCS and broken-SG pressures coincided with the complete depletion of the pressurizer collapsed water level. The depletion of the pressurizer inventory indicates that no saturated liquid remained, and only subcooled liquid was present in reactor vessel. For this reason, both the RCS pressure and the broken-side SG pressure dropped sharply.

After the rapid depressurization of the RCS ended, the pressure difference between the broken and intact SGs was not significant. Similarly, the pressure difference between the RCS and the broken SG was small, leading to reverse flow through the break. This reverse flow caused a fluid void difference in the PAFS line, which in turn resulted in flow fluctuations in the PAFS connected to the broken SG. The PAFS flow rate of the intact SG also showed slight fluctuations owing to the oscillatory behavior of the broken-side PAFS flow. However, these PAFS flow fluctuations did not significantly affect the heat transfer performance presented in Figure 7. This is because the RCS had already been sufficiently cooled by that stage of the transient.

5. Conclusions

The reactor trip results in turbine shutdown, which generates a main steam isolation signal. Closure of the main steam isolation valves forms a closed-loop circulation path between the steam generator and the PAFS. This closed-loop circulation prevents any direct leakage of reactor coolant to the external environment, thereby inherently limiting the release of radioactive materials.

Owing to this design feature, the present analysis was performed to demonstrate that the reactor coolant temperature can be reduced to below the safe shutdown criterion. The evaluation was conducted using the SPACE code, based on best-estimate methodology and assumptions. In this study, a detailed analysis was performed for a scenario involving the rupture of five SG tubes, assuming only automatic system actuation without any operator actions for 72 hours. The results indicate that the pressure difference between the primary and secondary systems reaches equilibrium, and the RCS inventory is maintained. Continuous cooling by the PAFS enables the reactor to reach the safe shutdown temperature without difficulty.

In conclusion, for 72 hours following the MSGTR accident, no operator actions were required, and the RCS water inventory was maintained above the top of the active fuel. Therefore, no fuel damage was observed under the assumed conditions. A sensitivity analysis of the heat removal performance of the PAFS will be performed as furtherwork.

ACKNOWLEDGMENTS

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