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Preliminary Analysis of Total Loss of Feedwater Accident for i-SMR using MELCOR Code

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1. Introduction

Over 70 small modular reactor (SMR) designs are under development worldwide, driven by advantages such as shorter construction periods, location flexibility near demand areas, and potential for integration with various industries [1]. i-SMR, one of the most expected SMR designs, is being developed through a collaborative effort by organizations including Korea Hydro & Nuclear Power (KHNP), Korea Atomic Energy Research Institute (KAERI), and KEPCO E&C in the Republic of Korea. The i-SMR is an integrated pressurized water reactor (IPWR) type SMR with electric power of 170 MWe and thermal power of 520 MWth [2]. Its design incorporates several key features, including boron-free operation, in-vessel control rod drive mechanism (IV-CRDM), helical coil steam generators, metal containment vessel (CV), and passive safety systems (PSSs).

The PSSs play a critical role in ensuring passive safety, which is one of the top tier requirements of the i-SMR development [2]. The PSSs of i-SMR incorporate passive emergency core cooling system (PECCS), passive containment cooling system (PCCS), and passive auxiliary feedwater system (PAFS). The PECCS is primarily designed to mitigate loss of coolant accident (LOCA). It depressurizes the reactor vessel by discharging steam into the CV through emergency depressurization valves (EDVs). The PCCS reduces the temperature and pressure of the CV by condensing the steam released into the CV. The condensate in the CV is then recirculated into the reactor vessel through emergency recirculation valves (ERVs). The heat exchanger of the PAFS is isolated from the secondary system during normal operation. In the event of an accident, the PAFS removes decay heat transferred to the steam generator through heat exchange between the PAFS heat exchanger and emergency cooling tanks (ECTs).

Previous studies have shown that given the PSSs operate as designed, they can effectively manage various accident scenarios including modular makeup and purification system (MMPS) charging/letdown line LOCA, station blackout (SBO), and total loss of feedwater (TLOFW) [2], [3]. However, from a regulatory and conservative perspective, it is necessary to evaluate the potential of severe accident progression under conservative assumptions, some or all of the safety systems fail to operate. For instance, preliminary severe accident analyses for representative scenarios such as TLOFW and the MMPS charging line LOCA, have been conducted using the CINEMA code [4], [5]. To further enhance the reliability of the safety assessment for the i-SMR, it is important to build a database of wide range of accident scenarios and assumptions using different numerical analysis codes. To expand the severe accident analysis database, this study performed an accident analysis using the MELCOR code for the TLOFW scenario.

2. Methodology

2.1 MELCOR input model

In this study, we developed a MELCOR input model of the 520 MWth i-SMR. The overall nodalization of the input model is shown in **Figure 1**. The model comprehensively includes the reactor coolant system (RCS), the secondary system, the CV, the reactor building (RB), and PSSs. The RCS consists of major components such as reactor core, pressurizer, and reactor coolant pumps (RCPs). The reactor core was designed into 5 axial and 5 radial rings to model the detailed power and temperature distribution. For the PSSs, the PECCS consists of two EDVs at the top of the pressurizer and two ERVs on the side of the RV, which were modeled as flow paths between the RV and CV. Two pressurizer safety valves (PSVs) were also modeled to open and close based on pressure setpoints, thereby preventing over-pressurization of the RCS. The PCCS was modeled using heat structures that simulate heat transfer between the CV and the PCCS heat exchangers.

A steady-state calculation was conducted to ensure the reliability of the input model. The normalized results, summarized in **Table I**, demonstrated that the main operation parameters have a relative error of less than 2 % from the standard design values. Moreover, heat balance between the primary and secondary sides of the steam generators was examined as a verification work of the input model. The steady-state calculation results were set as the initial condition for the subsequent accident analysis.

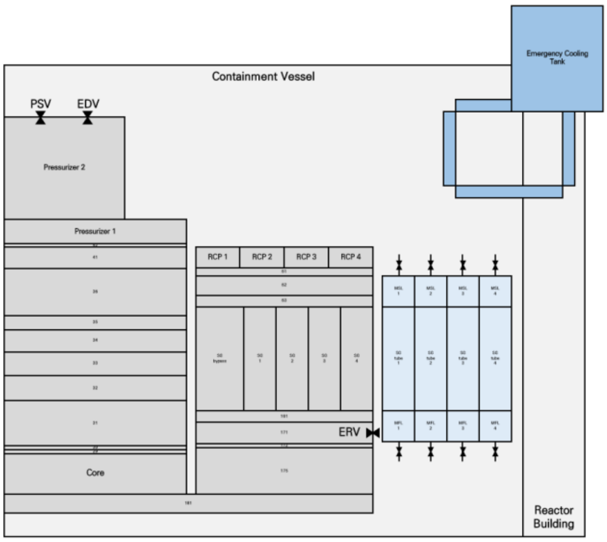

Fig. 1. MELCOR nodalization for i-SMR

Table I: Normalized steady state calculation result

Parameter	Design value	MELCOR simulation
Core thermal power	1	1
RCS pressure	1	1.0001
Core inlet temperature	1	1.0022
Core outlet temperature	1	1.0017
RCS mass flow rate	1	1.0004
SG 2nd inlet temperature	1	1.0003
SG 2nd outlet temperature	1	1.0179
Feedwater mass flow rate	1	1

2.2 Accident Scenario

The accident scenario, considered in this study, is the TLOFW, initiated by a sudden termination of feedwater supply to the secondary system. The MELCOR simulation was performed for 72 hours following the initiating event. To conduct a conservative accident analysis, it was assumed that both the PECCS and PAFS would remain inoperative, even if their actuation signals were generated. Accordingly, the valves associated with the systems were not allowed to open. Both PSVs were assumed to operate normally according to their pressure setpoints, and all trains of the PCCS were assumed to remain fully functional throughout the accident progression.

3. Results and Discussion

The sequence of major accidental event was summarized in **Table II**. The accident was initiated by the trip of the main feedwater pump in the secondary system flow. The loss of heat removal capability of steam generator caused the RCS pressure to rise. Once the RCS pressure reached the reactor trip set point, the reactor shutdown signal was generated, followed by the opening of the PSVs when the pressure exceeded their set point.

Table II: Major accident events

Event	Time (sec)
Accident initiation	0
MFWPs shutdown	0
Reactor trip; RCPs shutdown	15.1
Initial PSVs opening	25.4
PECCS, PAFS operation	N/A
Core uncover	N/A

Due to the steam release through the PSV and decay heat from the core, the PSV continued to cycle its opening and closing. The RV pressure showed repeated fluctuations, while the CV pressure peaked as the PSV opened. However, the CV pressure decreased after each peak due to the heat removal through the PCCS and CV wall, as shown in **Figure 2, 3**. The PSV last opened at about 166,500 seconds, but the flow rate did not appear in **Figure 3** because the plot time step was set too large.

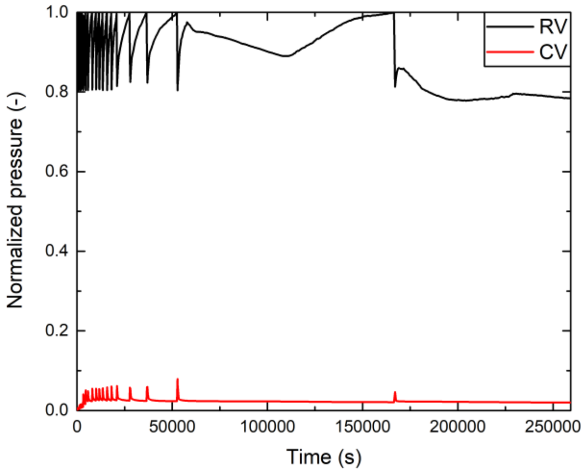

Fig. 2. Normalized pressure of RV and CV

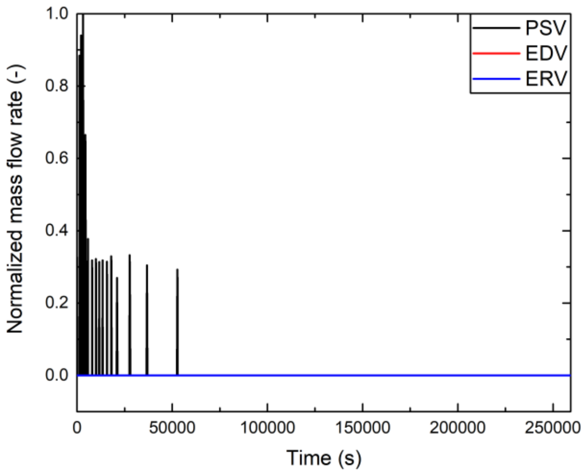

Fig. 3. Normalized mass flow rate of valves (PSV, EDV, ERV)

As a result of coolant discharge, the water level in RV gradually decreased, while the released coolant condensed inside the CV, causing the CV water level to rise as shown in **Figure 4**. **Figure 6** shows the significant heat transfer through RV structure, particularly at the lower head, which resulted in a substantial radial temperature gradient as illustrated in **Figure 5**. The heat transferred from the RV was delivered to the CV and subsequently removed through the PCCS and CV wall. As shown in **Figure 7** the heat removal rate on the PCCS heat exchanger and CV wall sharply increased at each PSV opening, producing distinct peaks. As the accident proceed and the CV wall temperature increased, the PCCS became the dominant heat sink. Also natural circulation flow was established within the CV. As shown in **Figure 8**, this flow consisted of an upward stream of steam and a downward stream of water, which effectively removed the heat released from the RV wall.

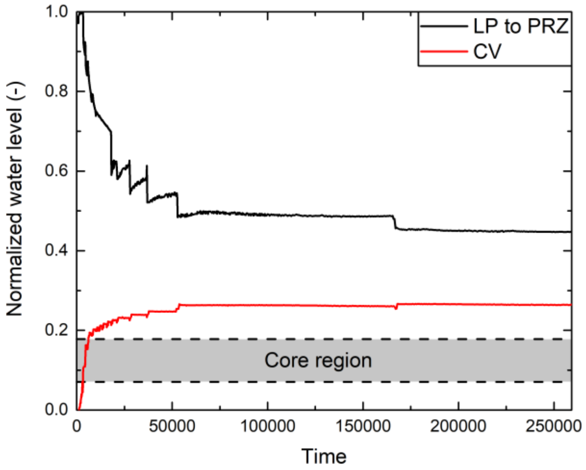

Fig. 4 Normalized water level in RV and CV

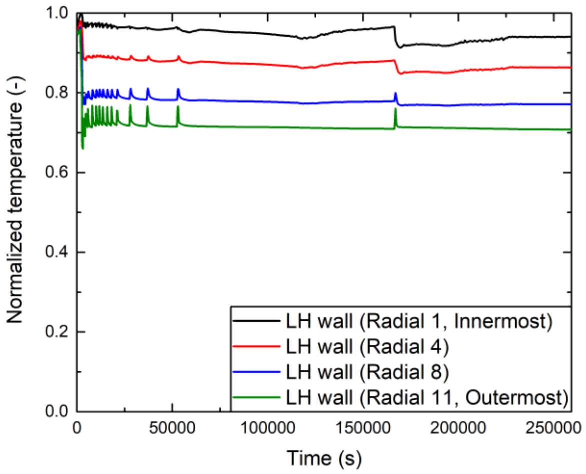

Fig. 5. Radial distribution of normalized lower head wall temperature

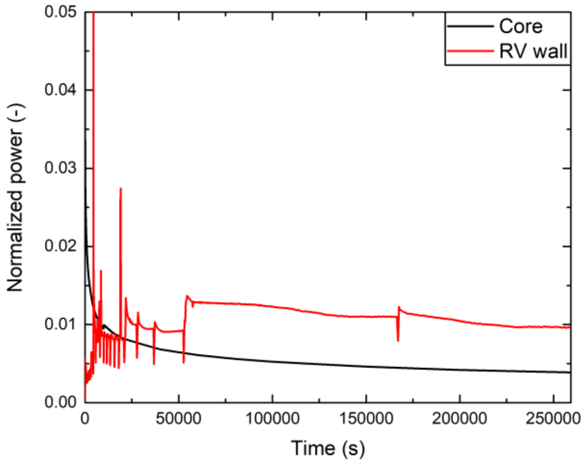

Fig. 6. Normalized core power and heat transfer rate of RV wall

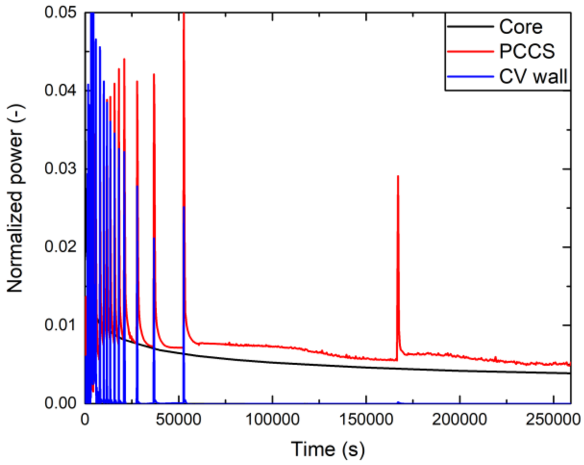

Fig. 7. Normalized heat transfer rate of PCCS and CV wall

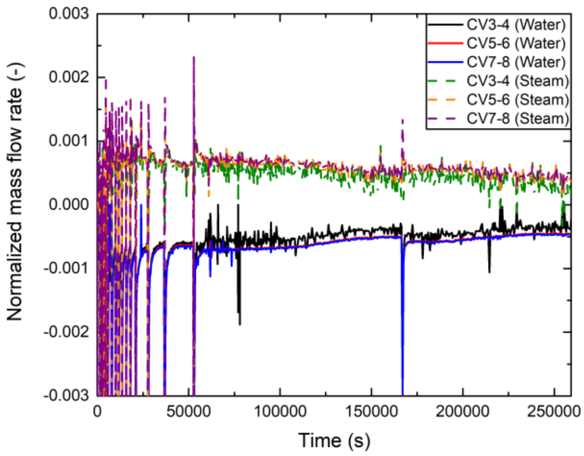

Fig. 8. Normalized mass flow rate of CV flow paths

From this analysis, it was confirmed that the heat generated in the core was transferred to the CV through the PSVs and RV wall, and then effectively removed by the PCCS and CV wall. As shown in **Figure 9**, the fuel, cladding, and core exit temperature stayed within appropriate range during the 72 hours. Consequently, the cladding temperature did not reach the oxidation reaction temperature, preventing both oxidation reactions and hydrogen generation. Furthermore, as shown in **Figure 10**, the fuel and cladding temperatures did not reach their melting point, ensuring that the initial inventory remained stable during 72 hours. Therefore, although the PECCS and PAFS did not operate in this scenario, the core heat was adequately removed, preventing core melting and progression into a severe accident.

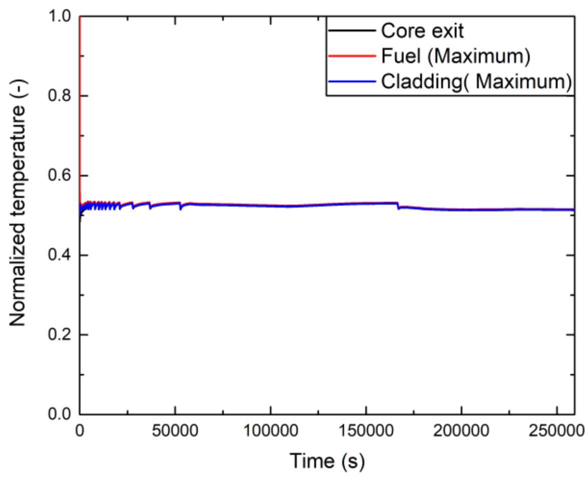

Fig. 9. Normalized temperature of core exit, fuel and cladding

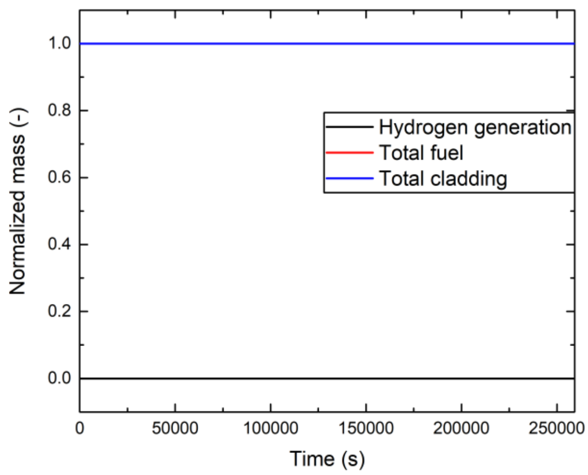

Fig. 10. Normalized total mass of hydrogen generation, fuel and cladding

4. Conclusions

This study presented a preliminary analysis of the TLOFW accident in the i-SMR using the MELCOR code under conservative assumptions that both PECCS and PAFS were unavailable. The major findings are as follows:

1) The repeated opening and closing of the PSVs released steam, which was effectively condensed by heat transfer through the PCCS and CV wall.
2) The discharged steam condensed, forming a stable water level in the CV and enabling RV wall cooling, thereby preventing core uncover inside the RV.
3) The fuel and cladding temperature remained below the melting temperatures, avoiding oxidation, hydrogen generation, and core melting throughout 72 hours after the accident.

Based on these results, it appeared that, even under conservative assumptions, the i-SMR effectively removes decay heat, thereby preventing severe accident progression during TLOFW accident. However, it should be noted that the results of this study are preliminary, and further modifications of the system design and advancement on the input model may change the numerical prediction of the accident progression.

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