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Preliminary analysis of high pressure severe accident in i-SMR using CINEMA code

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1. Introduction

Small Modular Reactors (SMRs) have attracted significant interest due to their modular design, which enables shorter construction times, enhanced safety, and flexible applications compared to traditional large-scale nuclear power plants. Among these, the PWR type SMR is particularly noteworthy because known PWR nuclear technology can be used. Consequently, there is a focus on research in South Korea on this PWR type SMR, so called i-SMR.

Figure 1 shows the preliminary design of the i-SMR [1]. The i-SMR is characterized by having a metal Containment Vessel (CV) outside the Reactor Vessel (RV) and utilizing passive safety systems such as the Passive Auxiliary Feedwater System (PAFS) and the Passive Containment Cooling System (PCCS). In this study, as part of the ongoing research into severe accident analysis for i-SMR[2,3], we present the preliminary analysis results of accident scenarios that could lead to high-pressure severe accidents.

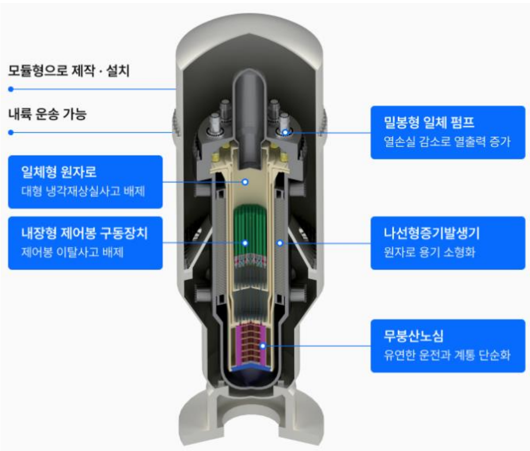

Fig. 1. Concept design of i-SMR [1]

2. TLOFW accident scenario

In this study, the TLOFW (Total Loss of Feed Water) was selected as the possible high-pressure accident scenario. In this scenario, the feedwater supply to the secondary side is interrupted, leading to a decrease in the water level within the RV pressurizer, causing an RCP (Reactor Coolant Pump) trip. As a result, the coolant inside the RV evaporates, maintaining a high-pressure state within the RV. During this time, only the Pressure Safety Valve (PSV), which maintains the RV pressure at a constant level, is assumed to function correctly, while all other safety valves are assumed to be inoperable due to malfunctions. As the coolant inside the RV continues to evaporate, RV pressure increases. Once RV pressure exceeds the allowable PSV pressure, steam is released from the RV into the CV. The steam released into the CV undergoes heat exchange with the outside environment via the PCCS and condenses. This condensed water can be used to cool the RV outer walls at the lower part of the CV.

The accident analysis was performed using the CINEMA(Code for INtegrated severe accident Evaluation and MAnagement) code, which is a domestic severe accident analysis code developed by a consortium of KHNP, KAERI, KEPCO E&C and FNC[4]. CINEMA has been previously applied to PWRs such as the OPR-1000[5]. Therefore, the code is suitable to analysis of the i-SMR, which is also a PWR type Reactor. The analysis covered a 72-hour period following the onset of the TLOFW.

3. Results and discussions

Figure 2 shows the pressure profiles inside the RV and CV during the TLOFW accident scenario. Figure 3 shows the collapsed water levels in the RV and CV. At the early stage of the accident, the coolant inside the RV evaporates due to decay heat from the reactor core, leading to an increase in RV pressure. Then, the RV pressure stabilizes within a certain high-pressure range through the operation of the PSV. The water level inside the RV continuously decreases, and approximately 10 hours after the accident, the reactor core becomes completely dry. It was observed that the CV pressure slightly increased due to the steam released from the RV through the PSV, while the water level inside the CV rose as the steam condensed through the PCCS, eventually submerging the outer wall of the RV.

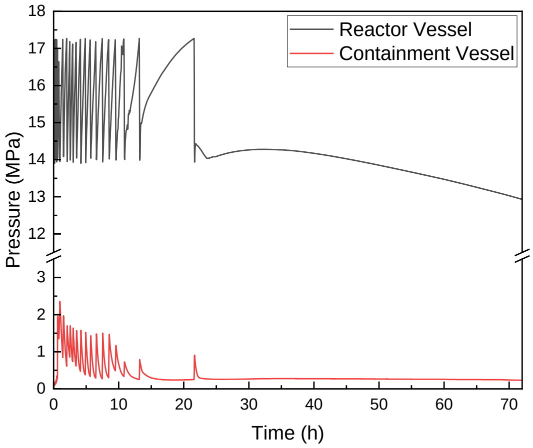

Fig. 2. Pressure profile inside the RV and CV

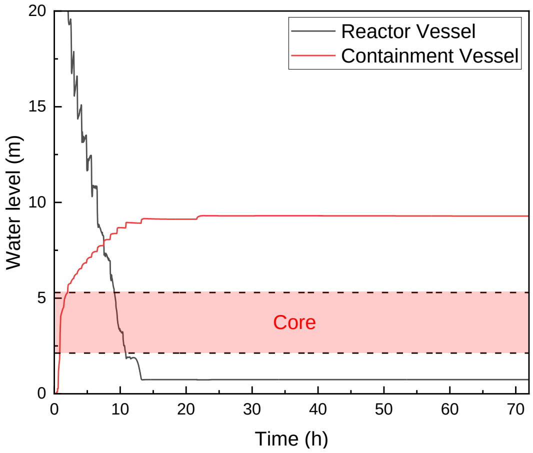

Fig. 3. Collapsed water levels in the RV and CV

Figure 4 shows temperatures of the reactor core at the center with different heights. The lowest point is denoted as Node[0], with higher nodes (up to Node[4]) representing positions further up of the core. In the initial conditions, the thermal power distribution of the fuel rods was set as cosine curve, resulting in higher temperatures at the central nodes (1, 2, 3) and lower temperatures at the ends (Nodes 0 and 4). Around 10 hours, when the core dried out, fuel temperature increased gradually due to decay heat. However, the temperature did not reach high enough to oxidize the cladding. Interestingly, around 25 hours after the accident, the fuel temperature began to decrease. From the detailed analysis, it was found that this phenomenon was due to the circulation of high pressure steam within the reactor building, which cooled heated cores.

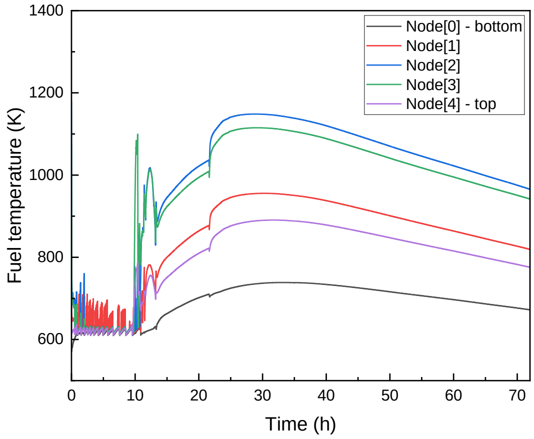

Fig. 4. Temperature of the reactor core at the center

Figure 5 presents the steam flow rates at upstream(downcomer) and downstream of the core. As the RV maintained a high-pressure state, the presence of high density steam inside the RV, coupled with the filling of water in the CV, resulted in the cooling of the RV wall. This initiated a circulation flow of high pressure steam between the hot core region and the cold RV wall, transferring heat from the core to the CV, and eventually out of the CV through the PCCS. It was also confirmed that the PCCS removed heat larger than the decay heat generated by the reactor core at 72 hours after TLOFW accident. Consequently, this CINEMA analysis demonstrated that even during a high pressure accident, the i-SMR system remained stable without progressing into a severe accident involving core meltdown for up to three days.

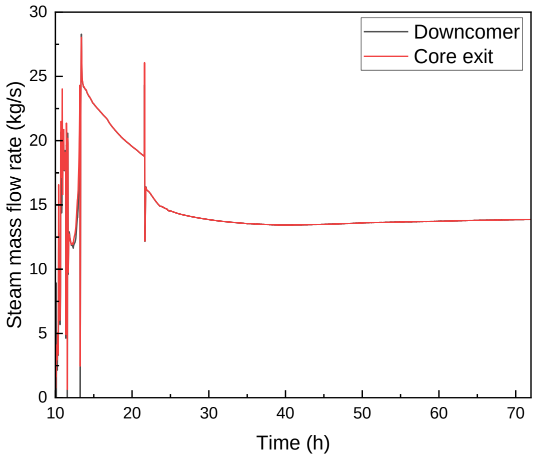

Fig. 5. Steam flow rates at upstream and downstream of the core

4. Conclusion

In this study, a preliminary analysis of a high pressure accident for i-SMR was performed using CINEMA code. Results showed that when high pressure accident conditions are established due to the evaporation of coolant within the RV in TLOFW accident, steam released from the RV through the PSV condenses via the PCCS, filling the CV with water. This water cools the RV from the exterior, while the high pressure steam inside the RV circulates naturally, transferring heat from the core to the outside, thereby preventing the progression to a severe accident in i-SMR system. However, it should be noted that these results are preliminary, and they may change based on future design modifications and the operational status of safety systems.

ACKNOWLEDGMENTS

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