Safety Analysis of i-SMR under LOCA Condition using SPACE Code

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ABSTRACT

i-SMR (innovative Small Modular Reactor), currently being developed in Korea, is a new type of reactor that integrates the core, pressurizer, steam generator, and reactor coolant pumps within a single vessel. Due to this integral design, large-diameter piping is eliminated, thereby fundamentally excluding the possibility of a LBLOCA (Large Break Loss of Coolant Accident). To satisfy the top-tier requirements of i-SMR—specifically, preventing core uncovery during design basis accidents—passive safety systems, including PAFS (Passive Auxiliary Feedwater System), PCCS (Passive Containment Cooling System), and PECCS (Passive Emergency Core Cooling System), have been adopted, significantly enhancing safety.

Safety analyses of i-SMR under SBLOCA (Small Break Loss of Coolant Accident) conditions were performed using the SPACE code, a thermal-hydraulic system code. In the event of an accident, reactor coolant is discharged into the containment vessel. Subsequently, PAFS and PCCS actuate to cool and condense the hot steam. This condensed coolant collects at the bottom of the containment vessel and is naturally injected back into the reactor vessel through the PECCS, driven by the pressure difference (hydraulic head) between the reactor vessel and the containment vessel.

The results confirmed that, even under conservative assumptions, the core remains covered, and stable conditions are maintained for 72 hours without operator actions. Consequently, i-SMR meets not only the regulatory acceptance criteria for LOCA but also the top-tier requirements. This demonstrates that i-SMR ensures improved safety compared to conventional large-scale PWRs.

Keywords: *i-SMR, LOCA, Deterministic Safety Analysis, SPACE code*

1 INTRODUCTION

Recently, carbon neutrality has been emphasized to mitigate climate change, and research on nuclear energy as a CO2-free energy is being actively conducted. In particular, safe and flexible SMRs are garnering significant attention worldwide. Accordingly, Korea has completed the standard design of i-SMR. As shown in Figure 1[1], i-SMR is an integral reactor where components such as the core, pressurizer, steam generator, and reactor coolant pumps of conventional commercial plants are integrated within a single reactor vessel. Due to this integral design, large-diameter piping is eliminated, thereby fundamentally excluding the possibility of a LBLOCA.

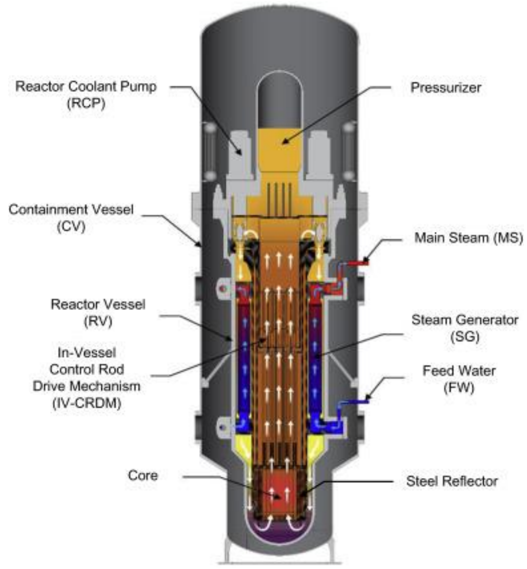

Figure 1: Schematic diagram of the overall i-SMR system

Furthermore, passive safety systems (Figure 2[1]) significantly enhance safety, allowing the reactor to maintain a safe shutdown state for 72 hours without external power in the event of SBLOCA. PAFS (Passive Auxiliary Feedwater System) is designed with four independent trains, each equipped with a passive condensation heat exchanger. Two trains share a single ECT (Emergency Cooling Tank). During DBAs (Design Basis Accident), PAFS forms a closed loop with the secondary steam line to remove primary heat transferred through the steam generator. PCCS (Passive Containment Cooling System) is composed of two independent trains, each equipped with a passive condensation heat exchanger, and shares the ECT with PAFS as its heat sink. Located at the top of the containment vessel, PCCS removes heat from containment atmosphere, thereby reducing its pressure and temperature. PECCS (Passive Emergency Core Cooling System) consists of three EDV (Emergency Depressurize Valve) and two ERV (Emergency Recirculation Valve). When EDV mounted on the pressurizer is opened, high-temperature, high-pressure steam is discharged from the reactor vessel to depressurize. ERV on the side of the reactor vessel, once opened, allow the condensed coolant accumulated at the bottom of the containment vessel to flow back into the reactor vessel, driven by the difference in hydrostatic head. To evaluate the performance of passive safety systems, LOCA analyses were performed using the SPACE code, a thermal-hydraulic system analysis code. LOCA was selected as the representative DBA because mitigating its consequences requires the actuation of all three aforementioned passive safety systems.

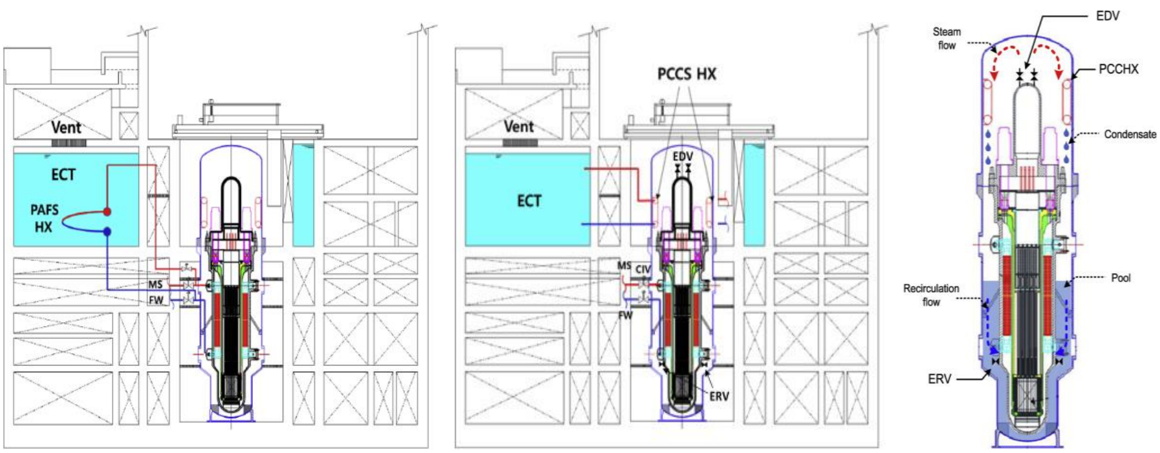

Figure 2: Schematic diagram of passive safety systems

2 ANAYSIS METHODOLOGY

2.1 Scenario

LOCA is defined as a potential accident in which a breach in a reactor's pressure boundary causes the coolant water to rush out of the reactor faster than makeup water can be added back in[2]. In i-SMR, the elimination of large pipes means that only SBLOCAs can occur, with potential break locations being the MMPS (Modular Makeup and Purification System) charging, letdown, and pressurizer spray lines. Generally, the cladding temperature serves as the FOM (Figure of Merit) for LOCA analyses. However, since i-SMR design precludes core uncovery and subsequent cladding heat-up, the collapsed water level of the reactor vessel was adopted as the FOM instead. From the perspective of the water level, a lower break location results in a lower collapsed water level; thus, the MMPS charging line near the main feedwater line was determined to be the limiting break location.

The LOCA scenario was divided into three phases based on the actuation of key safety systems.

- Phase A

Phase A is defined as the period from the initiation of the accident to the reactor trip. During this phase, the reactor coolant is discharged into the containment vessel through the break, causing the core pressure and pressurizer water level to decrease. Throughout Phase A, the RCS (Reactor Coolant System) remains filled with liquid coolant, and subcooled or saturated coolant is discharged through the break into the containment vessel.

- Phase B

Phase B is defined as the period from the reactor trip to the opening of the PECCS. Following the reactor trip, the rapid depressurization of the primary system lowers the saturation temperature. Consequently, flashing occurs in the core even under the low thermal power conditions of the shutdown process. After the reactor trip, the secondary piping and containment penetration lines are isolated, and the PAFS is operated by the opening of its actuation valves. The continuous discharge of coolant through the break leads to a steady decline in the primary system inventory and a gradual rise in containment pressure. When the dropping riser water level reaches the PECCS actuation setpoint, an actuation signal is generated. Subsequently, EDV and ERV open when the pressure difference between the reactor and containment vessels reaches the SOPM (Spurious Operation Protection Module) release criterion.

- Phase C

Phase C is defined as the period following the actuation of the PECCS. Since the pressurizer steam is released via the EDVs, the reactor and containment vessels achieve pressure equilibrium; at this moment, the pressure and temperature of the containment vessel may hit their maximum levels. Upon the opening of the ERVs, a coolant backflow is established from the reactor vessel to the containment vessel via the ERVs. Consequently, the collapsed water level of the reactor vessel steadily decreases until recirculation starts, potentially reaching its minimum. Subsequently, when the hydrostatic head of the condensed coolant at the bottom of the containment vessel builds up enough to overcome the pressure difference between the reactor and containment vessels, recirculation to the reactor vessel starts through the ERVs. The coolant injected into the reactor downcomer restores the core water level and prevents core uncover by maintaining the reactor inventory throughout the accident. Since the decay heat from the core is continuously removed by the PAFS and PCSS, the RCS reaches a safe shutdown state and sustains it for over 72 hours.

2.2 Nodalization

SPACE nodalization for i-SMR includes two hydraulic channels for the core and the downcomer, a single riser channel, one pressurizer, four RCP (Reactor Coolant Pump), four helical steam generator heat exchanger channels located at 90-degree azimuthal intervals, and a single channel containment vessel enveloping the reactor vessel. In the secondary system, the helical steam generators are modelled with four hydraulic channels. To account for their independent system configurations, the PAFS and PCCS are explicitly modelled as individual trains. Additionally, the model includes valve components for the three EDVs and two ERVs. For the core, downcomer, and steam generators, the multiple channels within each region are connected using cross-flow to simulate multidimensional behaviours. Excluding the core and PAFS, default modelling option were applied to all components in accordance with the user guidelines. For the PAFS, a dedicated heat transfer models developed for i-SMR analysis were implemented.

2.3 Initial conditions and assumptions

The major core and system parameters used in the LOCA analysis are summarized in Table 1. To account for measurement uncertainties, the initial core power is assumed to be 102% of the nominal power, and a conservative top-skewed axial power shape is applied. For the decay heat, 1.2 times the ANS-71 curve is used. Additionally, the maximum design data is applied to the tube plugging rate of the helical steam generators. During a LOCA, when the containment pressure reaches the reactor trip setpoint, a reactor trip signal is generated, and the reactor trips. Following the reactor trip signal delay time, assuming a turbine trip and a LOOP (Loss of Offsite Power), the RCPs begin to coast down, and the containment penetration lines and secondary piping are isolated. For a conservative evaluation of the RCS inventory during a LOCA, the lower bound design values are applied to the containment pressure and temperature, as well as the ambient temperature outside the containment. These assumptions yield a conservative prediction of the break flow, effectively delaying both the time to reach pressure equilibrium and the initiation of recirculation.

Table 1: Major Initial and Boundary Conditions for i-SMR

Variables	Values
Core power	102% of nominal power
RCS flow	TDF (Thermal Design Flow)
Inlet temperature of the Core	Nominal
Outlet temperature of the Core	Nominal
System pressure	Nominal
Steam generator plugging rate	Maximum

3 CALCULATION RESULTS

The break spectrum of the MMPS charging line considered in the LOCA analysis ranges from 1% to 100% of the pipe cross-sectional area. The calculation results for the 100% break size are summarized in Table 2.

Based on the results of the sensitivity study on possible single failures within the passive safety systems, the most limiting single failure during a LOCA is the failure of a PAFS actuation valve to open, which results in the unavailability of one PAFS train. In this case, the core heat removal capability is minimized due to the loss of that PAFS train. The degraded PAFS heat removal capability leads to a lower depressurization rate of the reactor vessel, thereby delaying the time to reach the SOPM release pressure compared to other single failure scenarios. This delayed valve actuation means that the valves open at a lower RCS coolant level, resulting in the minimum margin to core uncovery.

Following the reactor trip triggered by the high containment pressure setpoint, the reactor power rapidly decreases, as shown in Figure 3. Concurrently, the system pressure also drops due to the break flow (Figure 4). Subsequently, the PAFS actuation valves open to initiate core decay heat removal via the PAFS, and the high-temperature steam discharged into the containment from the onset of the accident is condensed and cooled by the PCCS (Figure 5). The continuous coolant discharge depletes the reactor vessel inventory, generating a riser low-water-level signal; however, the PECCS is not actuated due to SOPM. As the sustained cooling by the PAFS and PCCS brings the differential pressure between the reactor and containment vessels below the SOPM setpoint, the EDVs and ERVs open simultaneously, as depicted in Figure 6 and Figure 7. Until the reactor vessel and the containment reach pressure equilibrium (Figure 4), coolant is discharged through the ERVs from the relatively higher-pressure reactor vessel into the containment. At this point, the minimum collapsed water level above the core is observed, as shown in Figure 8. Thereafter, as the system continues to depressurize via the EDVs, natural recirculation through the ERVs begins once the hydrostatic head of the containment condensate exceeds the differential pressure between the two vessels (Figure 7). During the recirculation cooling period, the containment water level remains higher than the reactor vessel water level (Figure 8), and the collapsed liquid level above the core recovers driven by the recirculation flow after hitting its minimum. Throughout the transient, both the cladding and coolant temperatures decrease steadily (Figure 9 and Figure 10). The core decay heat is continuously removed by the PAFS, PCCS, and ambient heat losses from the containment (Figure 5). Ultimately, the primary coolant temperature reaches entry condition for a safe shutdown state, as illustrated in Figure 9.

Table 2: Sequence of Events

Events	Time, sec
Break Occur	0
Reactor Trip	5
PAFS Actuation	5
PECCS Actuation	1,358
PCT	N/A
Minimum Collapsed Water Level above Core	1,650
Safe Shutdown State	3,622

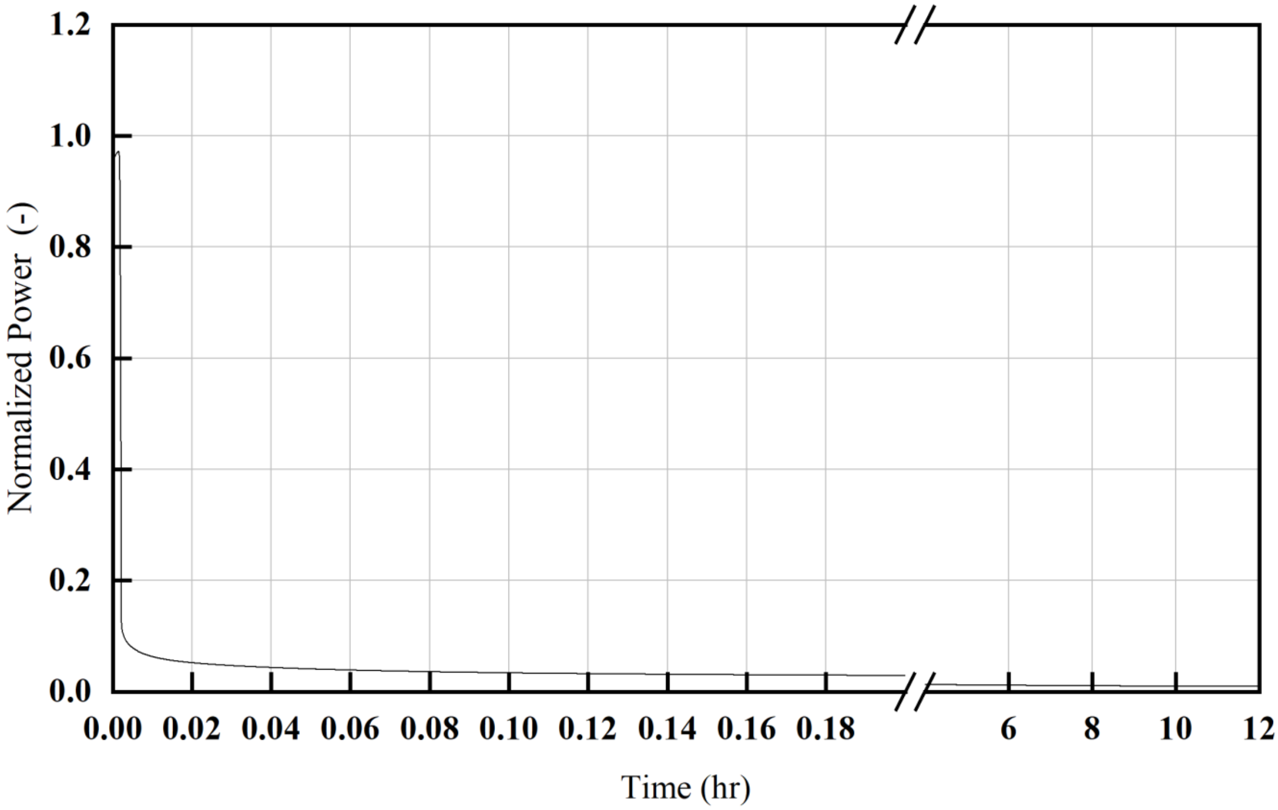

Figure 3: Core Power

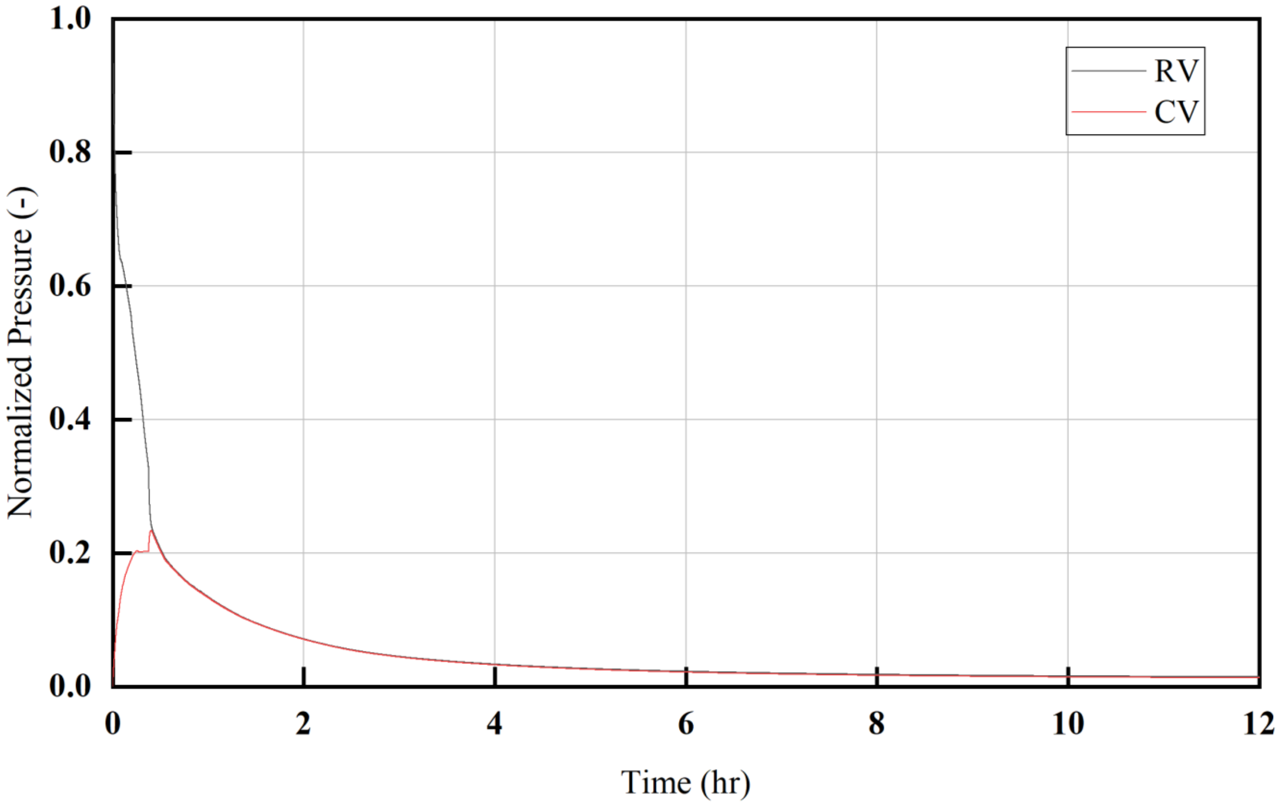

Figure 4: Reactor and Containment Vessels Pressures

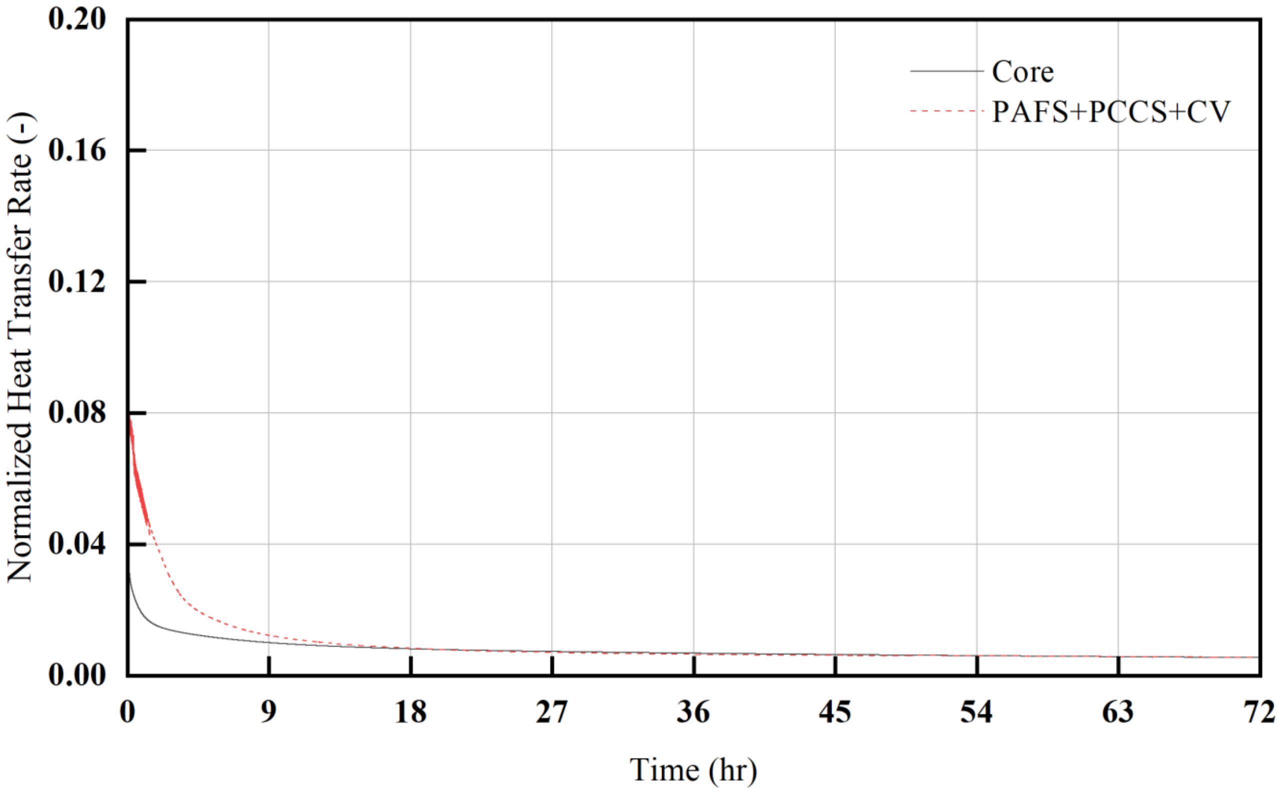

Figure 5: Comparison of Core Decay Heat and Passive Safety System Heat Removal

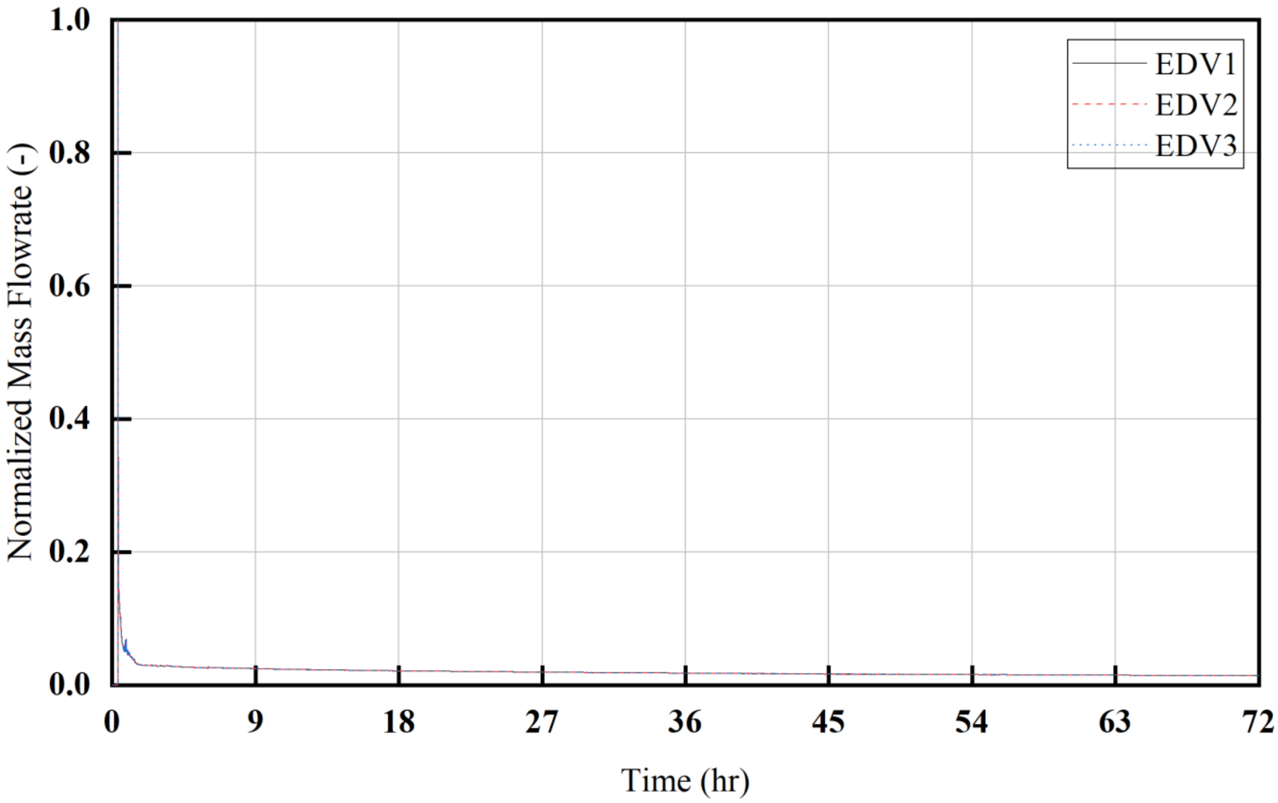

Figure 6: Mass Flowrate through EDVs

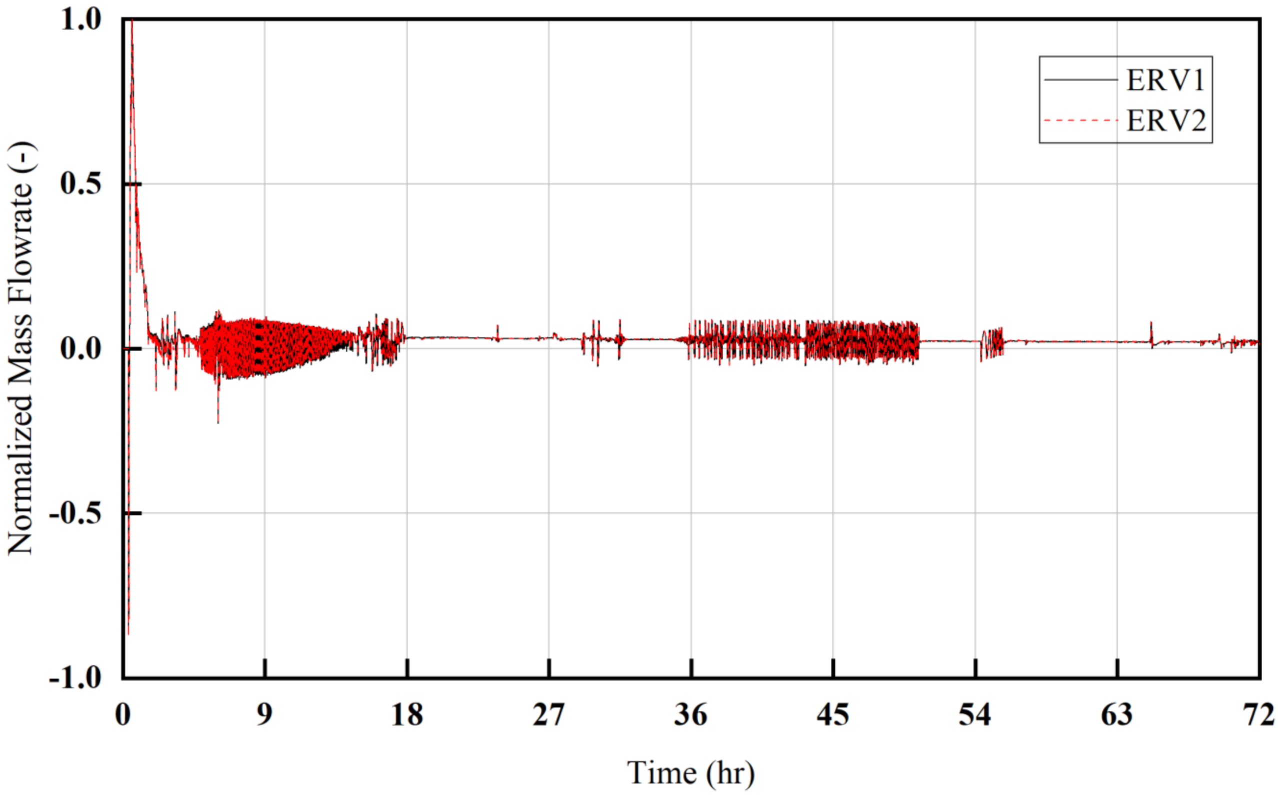

Figure 7: Mass Flowrate through ERVs

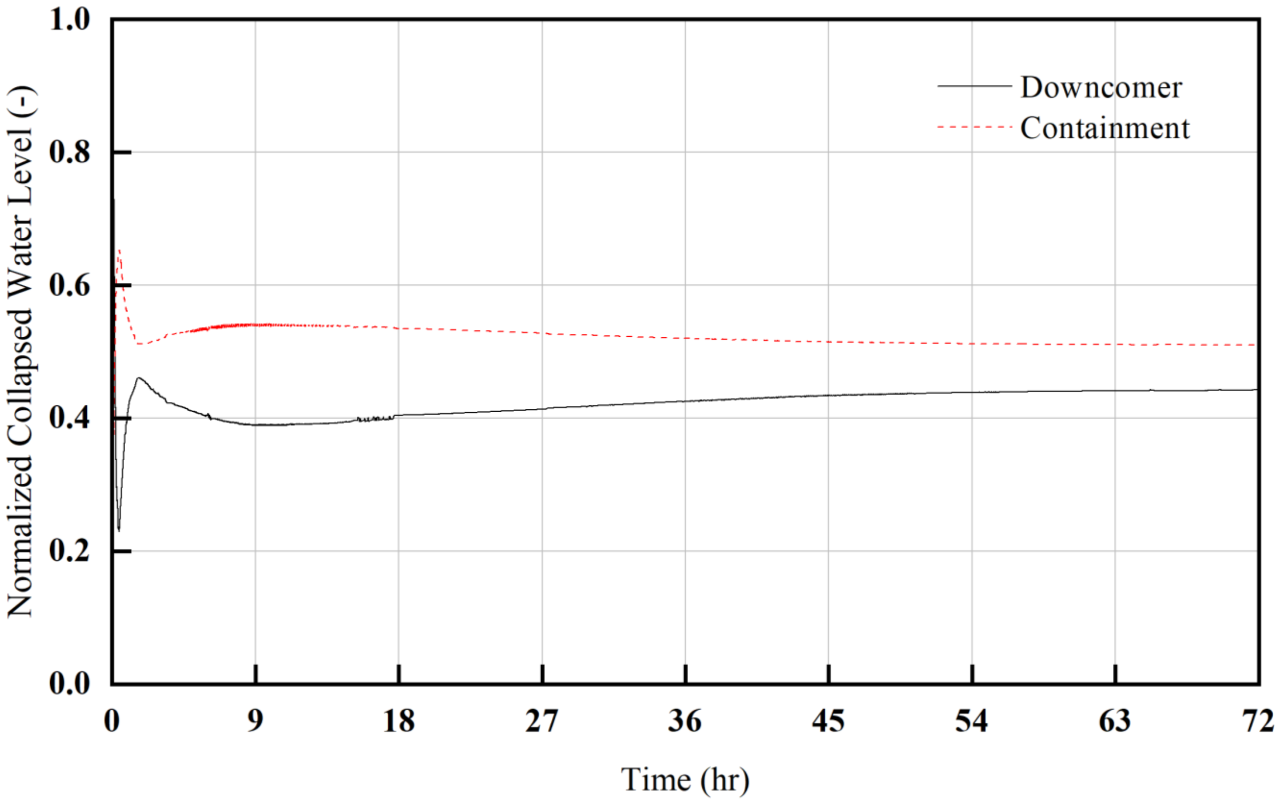

Figure 8: Downcomer vs. Containment Vessel Collapsed Water Level

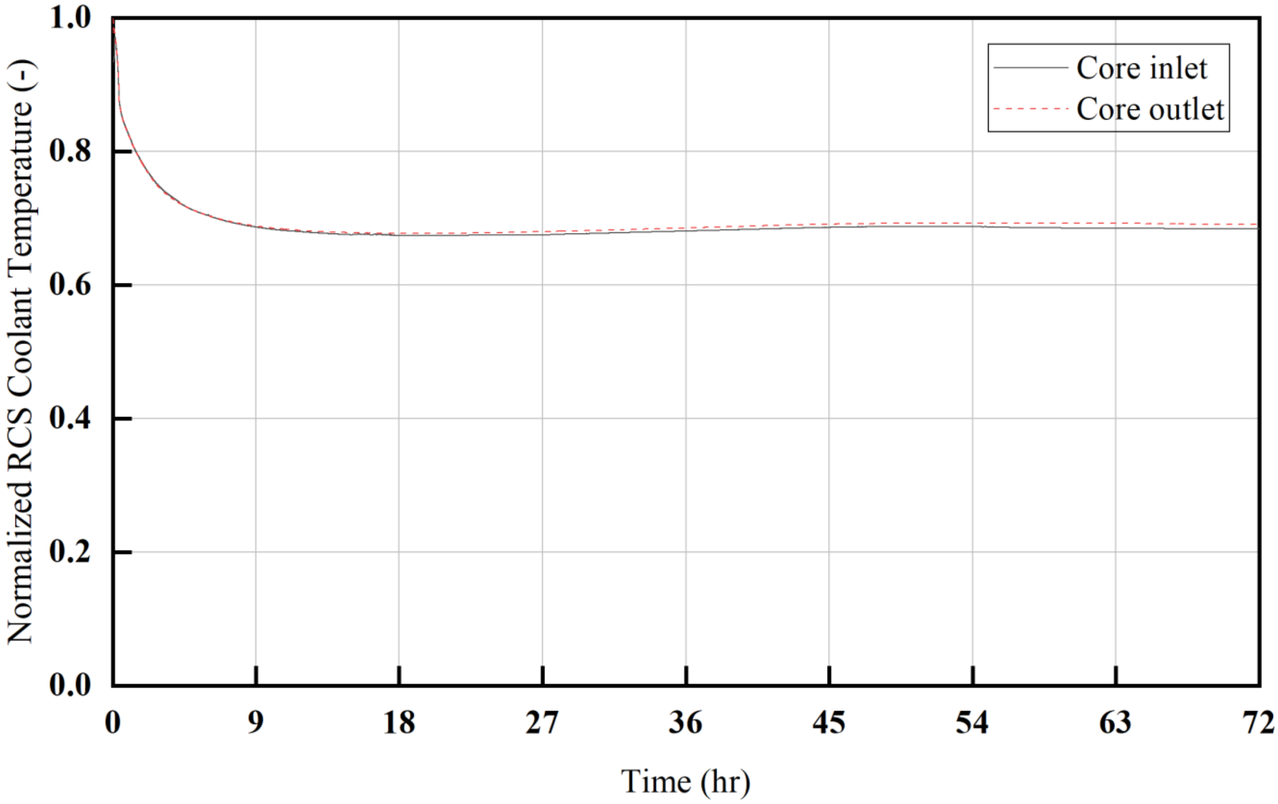

Figure 9: RCS Coolant Temperature

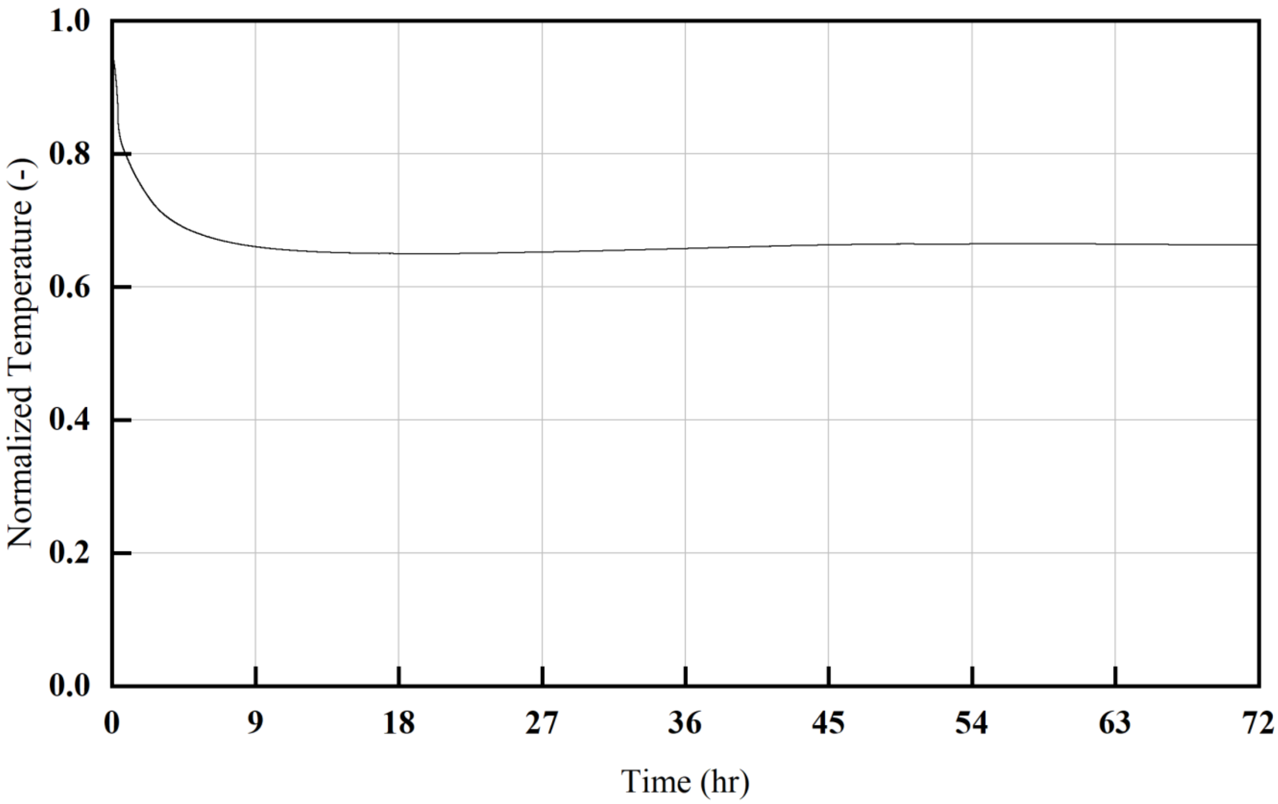

Figure 10: Hot Spot Cladding Temperature

4 CONCLUSION

This paper presented safety analyses of i-SMR under SBLOCA conditions using the SPACE code. Throughout the post-break period, core uncovery did not occur, and it was confirmed that the core decay heat was stably removed by the passive safety systems. Throughout the transient, the cladding temperature continuously decreased, and both the cladding oxidation and core-wide hydrogen generation remained below acceptance criteria. In conclusion, the PECCS of the i-SMR satisfies the safety acceptance criteria across the entire LOCA break spectrum.

REFERENCES

[1] S. G. Lim, H. S. Nam, D. H. Lee, S. W. Lee, Design characteristics of nuclear steam supply system and passive safety system for innovative Small Modular Reactor (i-SMR), *Nucl. Eng. Technol. 2025*, 57 (10), 103697, https://doi.org/10.1016/j.net.2025.103697

[2] U.S. Nuclear Regulatory Commission, Loss of coolant accident (LOCA), NRC Glossary. Retrieved [March 3, 2026], from https://www.nrc.gov/reading-rm/basic-ref/glossary/loss-of-coolant-accident-loca.html

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