

Evaluation of External Wall Heat Removal Capability Based on Containment Vessel Modelling in i-SMR

Yerim Park*, Dong-Hyuk Lee

Korea Hydro & Nuclear Power Co., LTD. Central Research Institute, 70, Yuseong-daero, 1312 beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*Corresponding author: yerim.park@khnp.co.kr

***Keywords :** i-SMR, Wall Heat Removal, Safety Analysis, SPACE

1. Introduction

Under the leadership of Korea Hydro & Nuclear Power Co. (KHNP), the Korean nuclear industry is advancing the standard design of an innovative small modular reactor (i-SMR). The goal is to obtain the standard design approval by 2028. To verify the safety of the plant design, it is necessary to analyze the phenomenon that can occur in the nuclear power plant under various transient events.

The previous studies have confirmed that the i-SMR gradually cools down even when the secondary heat removal capability is lost [1,2]. In these cases, the primary cooling mechanisms are discharge through pressurizer safety relief valves (PSRV) and external cooling through the reactor vessel (RV) wall.

In this paper, the effect of external cooling of RV wall was evaluated under various containment vessel (CV) model. To focus on the heat transfer performance of the RV wall, heat removal through the secondary system is assumed to be unavailable.

2. Methods

2.1 i-SMR Characteristics

The SPACE 3.3.2 code is used to simulate a transient states. The SPACE is a safety system code for nuclear power plant used for plant design and safety analysis [3].

In the i-SMR, the passive safety systems are applied to enhance the safety of the nuclear power plant and ensure maximum safety margins. The i-SMR has passive auxiliary feedwater system (PAFS), passive emergency core cooling system (PECCS) and passive containment cooling system (PCCS). The PAFS can decrease the reactor coolant temperature to the safety shutdown criteria within 36 hours following a reactor trip without alternating current power supply or operator actions, and perform its safety function for 72 hours. The PCCS heat exchangers are located at the upper region of the CV and condense steam discharged from PSRV and emergency depressurization valve (EDV). The condensed water accumulated at the bottom of CV and then recirculated into the RV through emergency recirculation valve (ERV) [4].

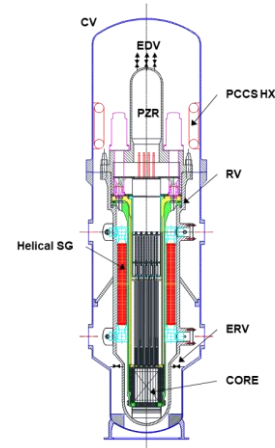


Fig. 1. Characteristic of the i-SMR

2.2 Containment Vessel Nodalization

The CV is modeled with PIPE component in SPACE code. The annular region of the CV is divided into 11 axial volume cells and, the upper and lower regions are modeled as single volume. Each volume is connected through flow paths. The geometric parameters, such as cell volume and flow area, are based on the design values of the i-SMR. The top view of node for analyzed cases are shown in Figure 2. The CV in the Base case is modeled as a single annular channel. In case Annular-2, the CV is modeled as two annular regions divided along the radial direction from the RV. For case Channel-2, the CV node is divided into two annular channels. In case Annular-2 and Channel-2, each volume cell is connected through cross flow junctions.

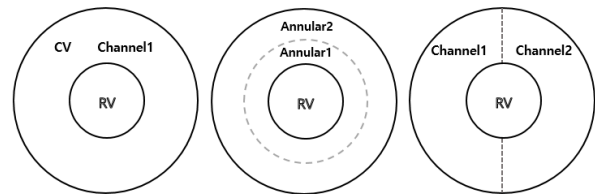


Fig. 2. Top view of CV nodalization (a) Base (b) Annular-2 (c) Channel-2

The steady state result is shown in Table I. The difference between the steady-state simulation result of i-SMR SPACE model and the design value is extremely

small, as shown in the Table I. It is confirmed that the SPACE model properly reflects the design specification.

Table I: Comparison for design value and SPACE simulation value

Parameter	Difference (%)
Core Power	-0.10
Pressurizer Pressure	0.01
Pressurizer Level	0.00
Core Average Temperature	-0.01
Reactor Coolant Flow Rate	0.00
Steam Generator Pressure	-1.64
Steam Generator Temperature	-0.27
Feedwater Temperature	0.00

3. Results

3.1 Transient Scenario

The malfunction of main feedwater system is assumed to simulate loss of normal feedwater accident. The heat removal through secondary side decreases leading to an increase of RV pressure. The RV pressure increases to setpoint of high pressurizer pressure signal. Then reactor trip and PAFS actuation signal are generated. In this paper, it is assumed that the PAFS isolation valves fail to open.

The PSRVs are opened and closed to maintain the RV pressure. The steam discharged through the PSRVs is condensed by the PCCS heat exchanger and falls to the bottom of the CV. The accumulated coolant at the lower CV cools the external wall of the RV. Due to heat release through the valve and RV external wall cooling, the core temperature decreases very gradually.

3.2 Sensitivity Analysis on Containment Vessel Node

The Figure 3 shows the normalized heat transfer rate of the PCCS. The heat transfer rate of the PCCS depends on the discharge rate of the steam through the PSRVs. As the PSRVs repeatedly open and close to release steam, the heat transfer rate shows periodic peak. In every cases, heat from the RV external wall is removed through natural circulation of single phase liquid. The heat transfer rate of case Channel-2 shows instability about at 9 hours, but it does not significantly affect the accident progression.

The heat transfer rate from the RV external wall to accumulated water in the CV is shown in the Figure 4. Figure 5 shows the RV wall temperature and Figure 6 shows the maximum cladding temperature.

Compared to the other cases, case Annular-2 achieves the fastest cooldown in both the temperature of RV wall and core. The subdivision of nodes in the radial direction provides a more finely resolved liquid temperature gradient near the RV external wall. . In case Annular-2, active cross flow is induced and facilitates effective natural circulation.

In case Channel-2, the division of the volume into two annular channels promotes active cross flow between the channels, leading to effective heat removal from the RV external wall.

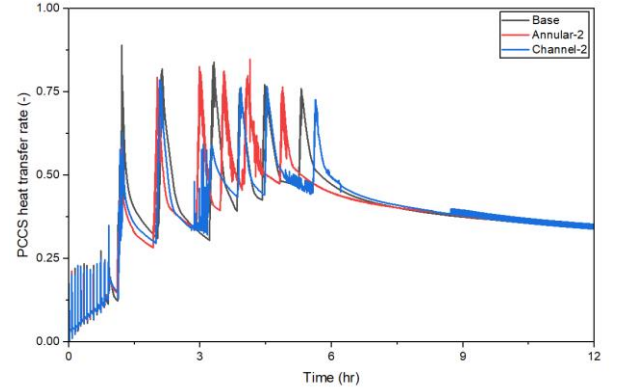


Fig. 3. Heat transfer rate of the PCCS

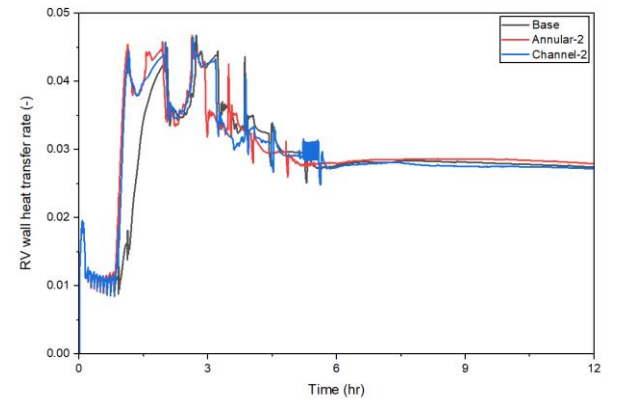


Fig. 4. Heat transfer rate of the RV external wall

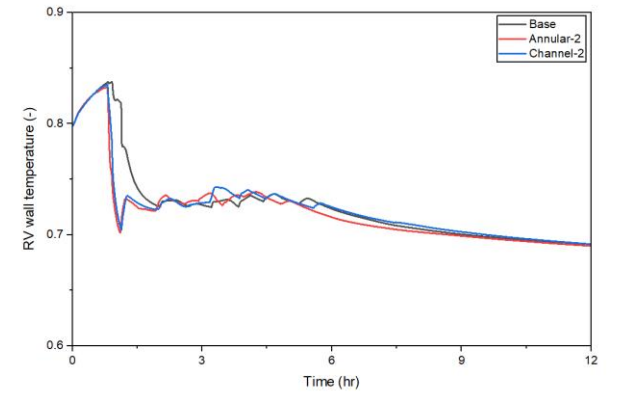


Fig. 5. RV external wall temperature

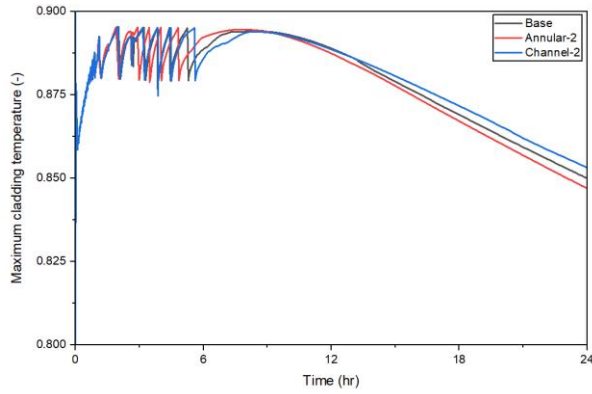


Fig. 6. Maximum cladding temperature (24 hr)

4. Conclusions

In this study, the transient event was simulated on various CV node model. It was confirmed that subdividing the CV volume node enhances natural circulation and RV wall cooling capability. It was found that the CV modeling affects the cooling capability of the RV wall during early phase of the accident. While the CV node influences the reduction behavior of cladding temperature in the long-term, but the temperature difference is not remarkably large. In the i-SMR SPACE simulation, the core temperature gradually decreases in the long-term with little sensitivity to the CV modeling, when the secondary side heat removal capability is lost.

ACKNOWLEDGMENTS

This work was supported by the Innovative Small Modular Reactor Development Agency grant funded by the Korea Government (MSIT) (No. RS-2024-00403548).

REFERENCES

- [1] Y. Park et al., Analysis of Loss of Feedwater System Accident in Small Modular Reactor using SPACE Code, KOSEE, 2025.
- [2] Y. Park et al., Sensitivity Analysis of Operator Actions during Loss of Feedwater System Accident in Small Modular Reactor, KOSEE, 2026.
- [3] Korea Hydro & Nuclear Power (KHNP), SPACE 3.3.2 manual, Volume 2 User manual, 2025.
- [4] Korea Hydro & Nuclear Power (KHNP), LOCA safety analysis report for iSMR, iSMR-K-X-PDR(R)-24014-P (Rev.0), (2024).