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Steady-State Modeling and Verification of the i-SMR Using MELCOR

Dosu Park [a], Sihyeong Yu [a], Wonung Jeong [a], Joongoo Jeon [b]*
[a] *Department of Quantum System Engineering, Jeonbuk National University, Jeonju, Republic of Korea*
[b] Division of Advanced Nuclear Engineering, Pohang University of Science and Technology, Pohang, Republic of Korea
***Corresponding author: jgjeon41@postech.ac.kr**

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1. Introduction

Small Modular Reactors (SMRs) are regarded as next-generation nuclear power systems that offer the advantages of reducing construction time and cost through downsizing and modularization, while simultaneously ensuring a high level of safety based on passive safety systems. Owing to these characteristics, SMRs have been actively researched and developed worldwide as promising alternatives to complement the limitations of conventional large-scale light water reactors.

However, in the event of an accident, it is essential to employ sophisticated and reliable analysis codes to comprehensively evaluate the thermal-hydraulic behavior of the reactor, the accumulation and transport of released gases, and the performance of passive safety systems. Against this background, this study performed system modeling of the Innovative Small Modular Reactor (i-SMR) and verified the steady-state calculation results using MELCOR, a representative severe accident analysis code. The objective of this study is to confirm that the primary, secondary, and passive safety systems can maintain a stable steady state without applying the COR_Package, and to verify that the same steady-state are preserved after incorporating the COR_Package, thereby ensuring the validity of the modeling.

MELCOR provides a wide range of safety-related information, including thermal-hydraulic behavior of the reactor, generation and transport of fission product gases, heat transfer phenomena in the containment, and the operation of passive safety systems under various accident scenarios. Through this capability, MELCOR enables systematic safety evaluation of the i-SMR and quantitative verification of reactor responses under different accident conditions. Instead of employing the COR Package, which is commonly used for core modeling, this study adopted the CV_SOU function within the CVH Package to directly represent the core heat source. Given that the i-SMR is still under development, specific details of the core design are not yet publicly accessible. Nevertheless, this approach allows the simplified reproduction of core thermal behavior in the initial stage of model development and provides a foundation for subsequent application to various accident scenarios.

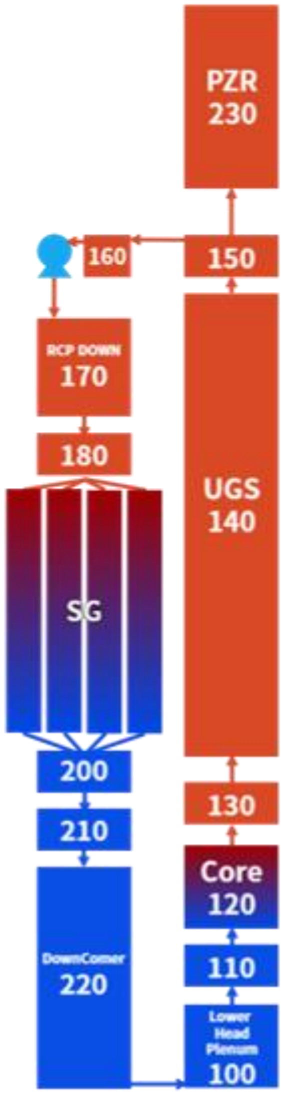

Figure 1. Configuration of the primary system

2. MELCOR Modeling

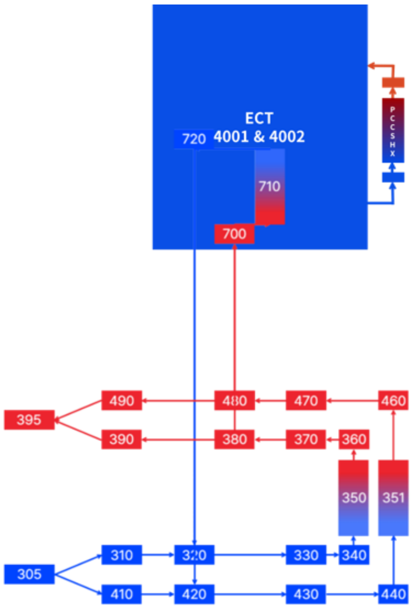

Figure 2. Configuration of the secondary system and Passive Safety System

In general, the i-SMR can be divided into three major subsystems. First, as shown in **Figure 1**, the primary system forms a circulation structure in which heat generated in the core (node 120) is transported by the coolant, transferred to the secondary side through the steam generator (SG), and then recirculated. Second, as illustrated in **Figure 2**, the secondary system receives feedwater at node 305, where it is heated in the steam generator (node 350) by the primary side, converted into steam, and discharged through node 395. Finally, as also shown in **Figure 2**, the passive safety systems consist of the Passive Auxiliary Feedwater System (PAFS), which condenses steam from the secondary side through heat exchange with the water in the Emergency Cooling Tank (ECT) and returns it as feedwater to the steam generator, and the Passive Containment Cooling System (PCCS), which condenses steam inside the containment through heat exchange with the ECT water.

2.1 Primary System Modeling

The primary system transfers the heat generated in the core to the secondary system via the coolant. The thermal output of the i-SMR is 520 MWt. Normally, the core would be modeled using the COR Package; however, in this study, the core was simulated by assigning a continuous heat generation of 520 MWt to the control volume representing the core. When the reactor pressure is low, the pressurizer increases it by activating the heaters; conversely, when the pressure is high, it decreases it through the spray system. The heater and spray functions were modeled using CV_SOU [1]. In this approach, a positive heat source was applied when the pressurizer pressure decreased, whereas a negative heat source was applied when the pressure increased, thus reproducing the behavior of the heaters and spray system. In the actual design, the secondary-side steam generator consists of eight headers. In this study, however, it was simplified by grouping two headers into one steam generator, resulting in a total of four steam generators. Furthermore, to enable a more detailed analysis of the thermal-hydraulic behavior, each steam generator was subdivided into ten sections [1].

2.2 Secondary System Modeling

The secondary system functions by converting the feedwater into steam as it passes through the helical coils of the steam generator, where it absorbs heat from the primary system, and by discharging the generated steam through the steam line. In accordance with the subdivision of the primary piping into ten nodes, the secondary-side steam generator was also divided into ten nodes, allowing for a more detailed analysis of the thermal-hydraulic behavior [1]. When setting the boundary conditions of the steam generator, the convective boundary condition was applied instead of the Helical Steam Generator option. In this process, the HS input was implemented by modeling a single heat transfer tube and adjusting the multiplicity factor.

2.3 Passive Safety System Modeling

Passive safety systems are designed to operate automatically without external power or operator intervention during accident conditions, thereby maintaining reactor safety. These systems respond to abnormal situations such as overpressure, depressurization, or the accumulation of residual heat in the core, stabilize pressure and temperature, and ultimately restore the reactor to a safe state. The Emergency depressurization valve (EDV) and Emergency Recirculation Valve (ERV) were modeled to open when the pressure difference between the Containment Vessel (CNV) and the pressurizer falls below a specified threshold. PCCS was modeled to allow continuous circulation of ECT coolant even under normal operating conditions. PAFS condenses the steam generated in the secondary side through ECT and supplies the condensed water back to the steam generator, thereby sustaining the removal of decay heat from the primary side through a natural circulation cooling process. In the event of a Loss of Coolant Accident (LOCA), the pressure difference between the CNV and the primary system gradually decreases, and once it drops below the threshold, the EDV and ERV are activated. At this point, the pressurizer gases are discharged into the CNV through the EDV, while the coolant from the primary system is released into the CNV through the ERV. The discharged steam undergoes heat exchange via the PCCS, condenses into water, and accumulates at the bottom of the CNV. As the water level rises and reaches the height of the ERV, it flows back into the primary system, establishing a circulation loop that enables effective heat removal under accident conditions.

3. Results

The steady-state conditions of the i-SMR model were evaluated by calculating error against the reference steady-state values specified in the i-SMR design. As shown in **figure 4**, the primary-side temperature, considered a key indicator for assessing steady-state conditions, was found to closely match the design steady-state value.

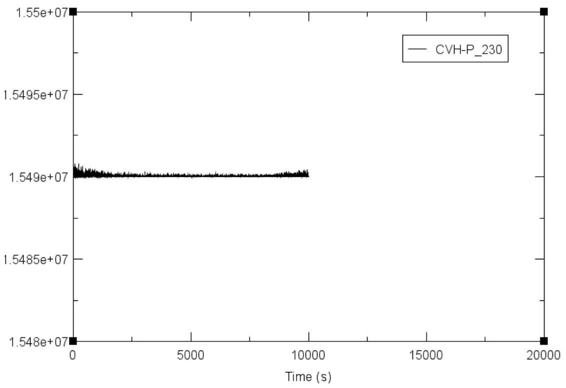

Figure 3. Primary System Pressure

Figure 3 shows the pressurizer pressure of the primary system. The calculated result exhibited an error rate of 0.064% compared to the reference value, confirming that the model is in very close agreement with the target condition.

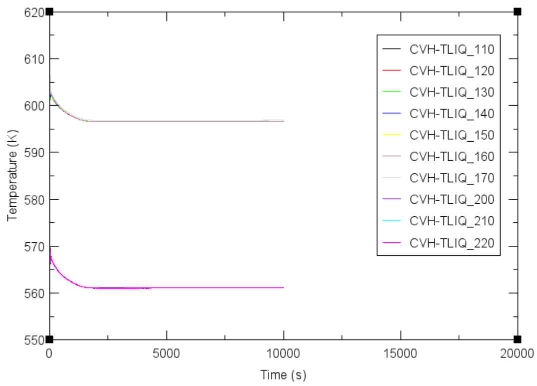

Figure 4. Primary System Temperature

Figure 4 presents the temperature results of the primary system, including T_cold and T_hot. The error rates were 0.376% and 0.443%, respectively, indicating that the steady-state thermal conditions were well reproduced by the model. In addition, the temperature difference (ΔT) between the hot and cold legs showed an error rate of 1.51%, which is slightly larger than that of the individual temperatures but still within an acceptable range for steady-state validation.

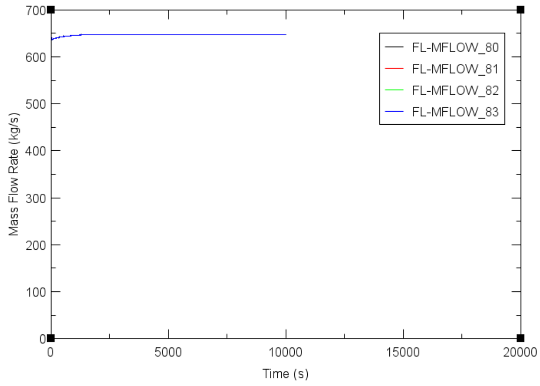

Figure 5. Reactor Coolant Pump (RCP) Mass Flow Rate

Figure 5 illustrates the RCP flow rate in the primary system. The result showed an error rate of 2.60% compared to the reference value, indicating that the steady state operation of the pump was appropriately reproduced.

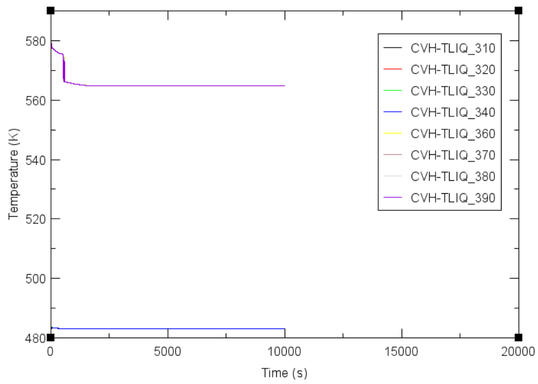

Figure 6. Secondary System Temperature

Figure 6 shows the feedwater and steam temperatures in the secondary system. The error rates were 0.021% for the feedwater side and 0.157% for the steam side, both of which indicate very good agreement with the reference values.

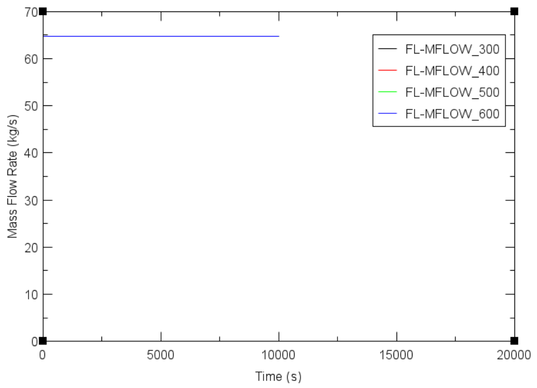

Figure 7. Secondary System Mass Flow Rate

Figure 7 presents the flow rate of the secondary system. The calculated result showed a very low error rate of 0.026%, demonstrating excellent agreement with the reference value. Overall, all major thermal hydraulic variables demonstrated close agreement with the reference data, and the developed model was confirmed to reliably reproduce the steady-state conditions of the i-SMR.

4. Conclusions

In this study, MELCOR was employed to develop a modeling framework for evaluating the safety of the i-SMR. The model was constructed by dividing the system into three major parts the primary system, the secondary system, and the passive safety systems using system information obtained from materials provided by the Nuclear Regulatory Research Division. The developed model achieved a stable steady state, and its validity was confirmed through comparison with reference results, demonstrating that the current modeling approach is reasonable. The developed i-SMR model can simulate various accident scenarios and calculate the source term of combustible gases even without employing the COR_Package. However, the absence of the COR_Package limits the ability to accurately analyze fuel behavior and the detailed characteristics of combustible gas phenomena. Therefore, future research will incorporate the COR_Package to achieve more precise analyses of fuel behavior and gas release, enabling more reliable evaluations across a range of accident scenarios. Ultimately, the goal of this research is to establish a comprehensive and sophisticated MELCOR model that can provide an integrated assessment of i-SMR safety.

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[1] C. H. Song, J. H. Song, and S. J. Kim, Evaluation of accident mitigation capability of passive core and containment cooling systems of i-SMR under representative LOCA and non-LOCA conditions, Nuclear Engineering and Technology, vol. 57, 103556, 2025.