

Assessment of Inherent Safety Features and Pressure Integrity of i-SMR during Anticipated Transient Without SCRAM

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1. Introduction

As the global community strives for carbon neutrality, nuclear power is gaining renewed attention as a carbon-free energy source. In this context, competition for the development of Small Modular Reactors (SMRs) is intensifying due to their potential to ensure both safety and economic efficiency. The innovative Small Modular Reactor (i-SMR), currently under development in Republic of Korea, is a model designed to drastically enhance safety during emergency situations by introducing innovative passive safety features, such as Passive Auxiliary Feedwater (PAFS), Passive Containment Cooling System (PCCS), and Passive Emergency Core Cooling System (PECCS).

According to IAEA-TECDOC-1791 [1], an Anticipated Transient Without SCRAM (ATWS) is defined as a representative Design Extension Condition (DEC) involving a common-cause failure of the reactor protection system. Specifically, it is categorized under DEC-A, which requires verifying the shutdown and cooling capabilities under multiple failure conditions before core melt occurs. In traditional large-scale PWRs such as the APR-1400, the occurrence of a turbine trip significantly influences reactivity feedback and primary system pressure behavior. Thus, trip and non-trip scenarios have been strictly analyzed to ensure.

However, the i-SMR possesses fundamental design characteristics that distinguish it from large-scale PWRs in terms of system configuration and actuation logic. In the APR-1400, the Auxiliary Feedwater System (AFW) operates based on a low steam generator level signal, and the Main Steam Isolation Valves (MSIV) respond to signals such as low steam generator pressure. In contrast, the i-SMR is designed such that a reactor trip signal directly triggers the immediate closure of the MSIVs and the actuation of PAFS. Due to these differences in signal architecture, it is not appropriate to directly apply the analysis assumptions of large-scale plants to the i-SMR, necessitating a safety evaluation tailored to the i-SMR's unique operating conditions and scenarios.

Accordingly, this study performs a thermal-hydraulic analysis of the i-SMR under ATWS conditions for the Loss of Main Feedwater (LOMF), General Transients (GTRN) by Reactor Coolant Pump (RCP) trip, and Loss of Coolant Accident (LOCA) while reflecting its specific design features. Specifically, we focused on the

contribution of the Pressurizer Safety Relief Valves (PSRV) to pressure control and verified whether the pressure limit (3200 psi) is maintained depending on the availability of secondary cooling sources. Through this investigation, we aim to demonstrate that the i-SMR secures sufficient safety margins even under severe ATWS scenarios, leveraging its lower power density and superior PSRV steam discharge capacity compared to large-scale nuclear power plants.

2. Methodology

The methodology for evaluating the safety of the i-SMR during ATWS events involves a comprehensive thermal-hydraulic analysis using a validated system code and a set of scenarios that reflect the unique design characteristics of the reactor. This section describes the modeling approach, the integrated signal logic, the specific analysis scenarios including reactivity feedback effects, and the acceptance criteria used to verify the safety margins.

2.1 Analysis Code and Plant Modeling

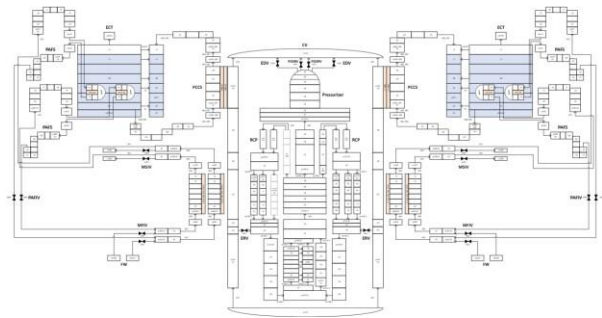


Fig. 1. i-SMR MARS-KS input model nodalization

The thermal-hydraulic system analysis in this study was performed using MARS-KS 2.0, and advanced best-estimate system code maintained and validated by the Korea Institute of Nuclear Safety (KINS) [2]. Unlike conventional large-scale PWRs, the i-SMR adopts an integrated reactor design, where the RCPs, Steam Generators (SGs), and the Pressurizer (PZR) are all housed within a single Reactor Vessel (RV). The integrated configuration of the reactor was modeled with detailed nodalization as shown in Fig. 1. This integrated configuration fundamentally eliminates the possibility of

Large-Break Loss of Coolant Accidents (LBLOCA) by removing large-loop piping. The modeling in this study precisely nodalized the complex internal flow paths and component interactions within the RV to capture the unique natural circulation and pressure transient characteristics of the i-SMR.

2.2 Signal Integration

A defining safety feature of the i-SMR is its signal integration logic, which distinguishes it from the APR-1400[3]. In the APR-1400, the AFW operates based on independent signals such as low SG water level. In contrast, the i-SMR is designed with a synchronized logic where a reactor trip signal immediately triggers the closure of the MSIVs and the simultaneous actuation of the PAFS. The differentiation in the accident sequences between the APR-1400 and the i-SMR is illustrated in Fig. 2. This study incorporates this integrated logic to simulate how the early isolation of the secondary side and the immediate initiation of passive cooling effectively suppress primary system pressure during an ATWS event. The specific trip signal logics for the major systems in the APR-1400 and the i-SMR are detailed in Table I and Table II.

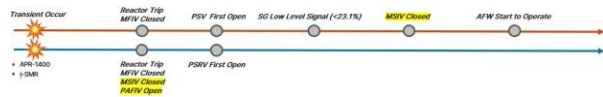


Fig. 2. Difference of ATWS sequence between APR-1400 and i-SMR

Table I: APR-1400 Main System's Trip Signals

Main System	Trip Signal	Remark
PSV (Pressurizer Safety Valve)	PZR High Pressure	PZR Pressure > 16.4MPa
MSIV	SG Low Pressure	SG Pressure < 6.05MPa
MFIV	Reactor Trip Signal	-
AFW	SG Low Level	SG WR Level < 23.5%

Table II: i-SMR Main System's Trip Signals

Main System	Trip Signal	Remark
PSRV	PZR High Pressure	PZR Pressure > 17.0MPa
MSIV	Reactor Trip Signal	-
MFIV	Reactor Trip Signal	-
AFW	Reactor Trip Signal	-

2.3 Analysis Scenarios and Assumptions

To evaluate the safety margins of the i-SMR, ATWS scenarios were developed for LOMF, GTRN, and LOCA. The analysis matrix was constructed by combining the following variables:

- Turbine Trip vs Non-trip: Following the conservative methodology used for large-scale plants, cases were divided based on whether a turbine trip occurs in synchronization with the reactor trip signal or the turbine remains operational, maintaining secondary steam discharge. The specific sequence of events for the ATWS

non-trip scenario, including the thermal-hydraulic and neutronic coupling, is depicted in Fig. 3.

- PSRV Success vs Failure: The success or failure of the PSRV was set as a critical analysis variable. This sensitivity analysis aims to verify if the i-SMR's inherent design characteristics can maintain pressure integrity even in the event of active valve failure.

- PAFS Availability: The contribution of Passive Safety Systems (PSSs) was evaluated by varying the number of operational PAFS trains (0/4, 2/4, and 4/4). The 0/4 configuration represents a limiting conservative scenario where the i-SMR must rely on thermal inertia, PCCS, and PSRVs without any assistance.

- LOCA-ATWES Conditions: For scenarios combining a pipe break with a failure to SCRAM, the most limiting break conditions allowed by the integrated structure were selected to verify core inventory and cooling performance. The output of the detector model is input to the cable model. The cable is modeled using the code PSpice and includes the cable characteristics of capacitance, resistance, characteristic impedance, and length. Although the effect of cable on the shape of the output signal of a detector is usually negligible, in our application to high-rate counting, its effect is significant. For our application, we found a distributed parameter model was necessary for accurate modeling of the cable.

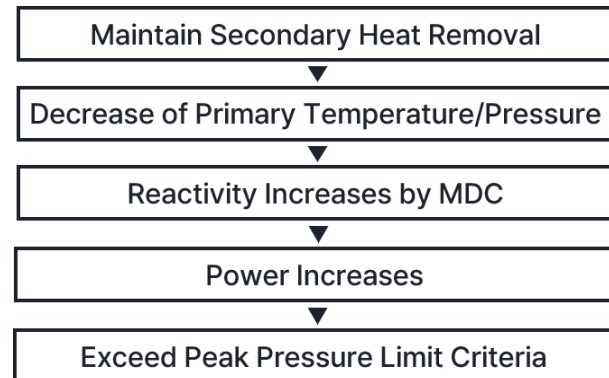


Fig. 3. Accident sequence for ATWS – non trip case

2.4 Acceptance Criteria

The safety of the i-SMR under ATWS conditions was evaluated against the following three criteria, based on regulatory guidelines and design requirements:

1. Peak Pressure Limit: The peak primary system pressure must not exceed 3,200psi. This criterion is particularly critical for verifying that the reactivity-induced power increases in turbine non-trip scenarios do not compromise the Reactor Coolant Pressure Boundary (RCPB).
2. Peak Cladding Temperature (PCT): In accordance with 10 CFR 50.46 [4], the PCT must remain below 1,477K (2,200°F) to ensure that the core geometry remains intact and cooling is maintainable despite any power transient.

3. Long-term Cooling (ECT Inventory): The inventory of the Emergency Cooling Tank (ECT) must be sufficient to support the sustained decay heat removal required to stabilize the plant following the initial power and pressure peaks.

3. Results and Discussion

3.1 Initial Thermal-Hydraulic Response and Pressure Boundary Integrity

This section evaluates the rapid transients occurring within the first few minutes (2,000sec) of the accident and the resulting integrity of the RCPB. Specifically, the impact of turbine trip status and the operational success of the PSRV in the internal pressure of the RV was analyzed.

In most ATWS scenarios, the i-SMR effectively mitigated the initial transient through its inherent reactivity feedback characteristics. As the primary coolant temperature rose immediately following the initiation of the event, negative reactivity was inserted into the core due to the strongly negative Moderator Density Coefficient (MDC) and Doppler Temperature Coefficient (DTC). This resulted in an immediate reduction in reactor power. This power reduction, combined with the discharge capacity of the PSRVs, allowed the primary system pressure to be stably controlled at approximately 17MPa, near the design operating pressure. This demonstrates a safety margin relative to peak pressure limit 22MPa, confirming that the inherent safety features of the i-SMR effectively suppress early ATWS behavior. For the LOMF turbine trip scenario with successful PSRV actuation, the reactor power and RV pressure trends are shown in Fig. 4 and Fig. 5, respectively.

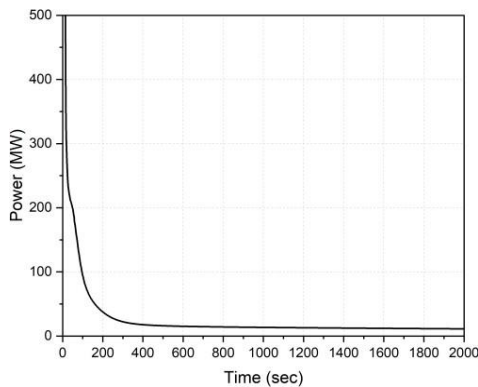


Fig. 4. Power for LOMF turbine trip case (PSRV Success)

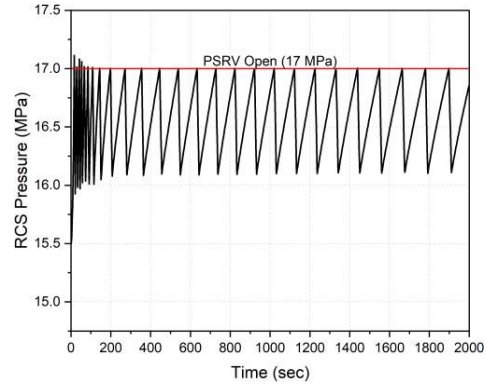


Fig. 5. RV Pressure for LOMF turbine trip case (PSRV Success)

In the GTRN non-trip scenario, the unique safety robustness of the i-SMR design was observed, even under the assumption of a PSRV failure. This case simulated a simultaneous failure of both the turbine trip and Main Feedwater (MFW) isolation, resulting in continuous feedwater supply. The analysis showed that while the core power did not decrease rapidly and remained relatively stable, the continuous MFW flow effectively removed a thermal load equivalent to the total reactor power output to the secondary side. Consequently, even without PSRV actuation, the cooling performance of the MFW alone was sufficient to decrease the primary pressure at a stable level. This proves that the i-SMR system configuration provides complementary cooling paths through the secondary side, ensuring pressure integrity even during multiple system failures. In the GTRN non-trip case under PSRV failure, the power and pressure transients are presented in Fig. 6 and Fig. 7.

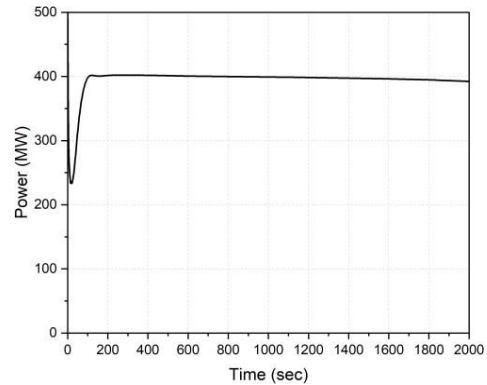


Fig. 6. Power for GTRN turbine non-trip case (PSRV Failure)

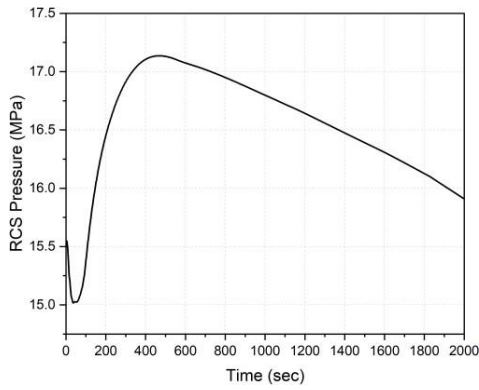


Fig. 7. RV Pressure for GTRN turbine non-trip case (PSRV Failure)

Conversely, in scenarios such as the LOMF where secondary cooling is unavailable, a simultaneous failure of the PSRVs leads to a complete loss of energy discharge capability. In this case, the primary pressure rose sharply within 50-100 seconds of the event initiation, exceeding the 22MPa (3,200 psi) limit. This contrast highlights that while the early safety of the i-SMR primarily relies in PSRV reliability. The limiting case of LOMF non-trip with PSRV failure resulted in a rapid pressure excursion, as illustrated in the power and pressure trends in Fig. 8 and Fig. 9.

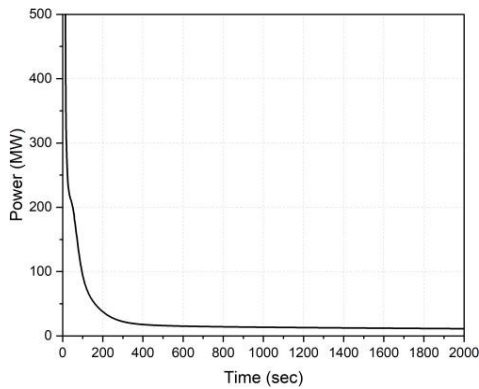


Fig. 8. Power for LOMF turbine non-trip case (PSRV Failure)

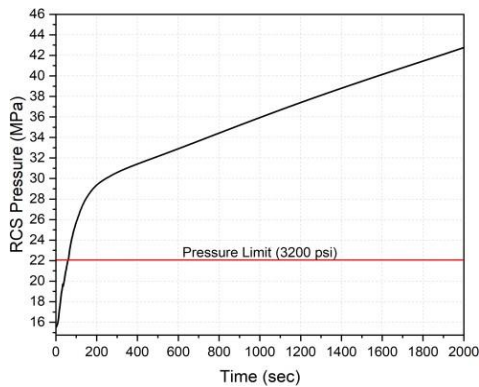


Fig. 9. RV Pressure for LOMF turbine non-trip case (PSRV Failure)

3.2 Long-term Cooling Performance and Heat Sink Sustainability

Following the initial pressure peak, the reactor transitions into a long-term cooling mode, where the PAFS becomes the primary mechanism for decay heat removal. The analysis demonstrates that while increasing the number of operational PAFS trains significantly enhances the initial cooling gradient, it also accelerates the evaporation of coolant within the ECT. This higher heat removal rate leads to a more rapid depletion of the ECT inventory compared to the 2/4 configuration. This physical trade-off indicates that while more passive trains provide faster stabilization, they simultaneously shorten the available mission time of the ultimate heat sink. The impact of the number of operational PAFS trains on the heat sink sustainability is evaluated in Fig. 10 and Fig. 11, which compare the ECT water level depletion rates between the 2-train and 4-train configurations.

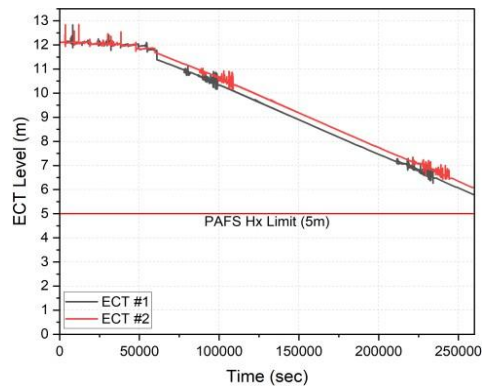


Fig. 10. Two train of ECT Level for LOMF ATWS in success for PAFS 2 trains

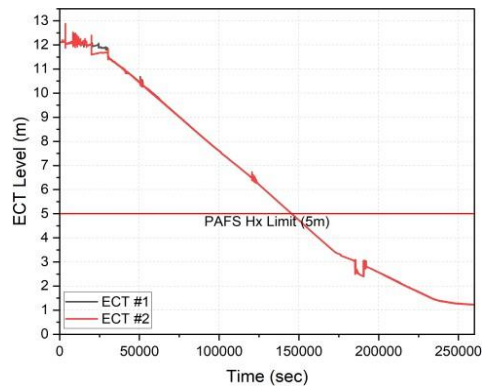


Fig. 11. Two train of ECT Level for LOMF ATWS in success for PAFS 4 trains

3.3 Safety Evaluation of LOCA-ATWS

The safety of the i-SMR was further evaluated under the extreme conditions of a LOCA-ATWS. In all analyzed scenarios, the PCT remained well below the regulatory limit of 1,477K, ensuring the preservation of core geometry. Furthermore, the density-driven natural

circulation between the core and the SGs remained stable even after the cessation of RCPs, maintaining sufficient core collapsed water level to prevent core uncover. The preservation of core geometry and fuel integrity during the LOCA-ATWS event is verified through the RV water level and PCT trends shown in Fig. 12 and Fig. 13.

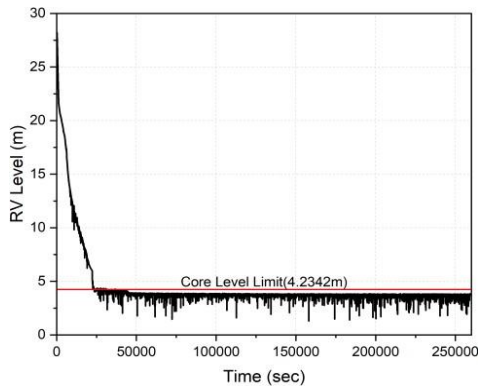


Fig. 12. RV Level for SLOCA-ATWS in success for PAFS 2 trains

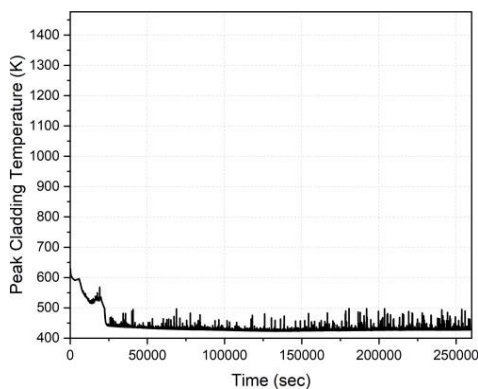


Fig. 13. PCT for SLOCA-ATWS in success for PAFS 2 trains

3.3 Synthesis: Inherent Safety and Design Advantages of the i-SMR

The comprehensive analysis of various ATWS scenarios confirms that the i-SMR maintains superior safety margins compared to traditional large-scale reactors. These enhanced margins are primarily attributed to the reactor's inherent design characteristics. First, the low power density and the large primary coolant volume within the integrated vessel provide substantial thermal inertia, which effectively slows down the progression of transients. Second, the synchronized signal integration logic ensures the immediate commencement of passive cooling, thereby suppressing pressure spikes more effectively than independent safety systems. These integrated features allow the i-SMR to autonomously maintain fundamental safety functions even under multiple system failures.

4. Conclusion

In this study, a thermal-hydraulic analysis of i-SMR ATWS scenarios was performed to evaluate safety margins under design extension conditions. The results indicated that primary pressure integrity during transient events is heavily determined by the operational success of the PSRVs, signifying a high level of dependence on PSRV reliability for early-stage mitigation.

Furthermore, it was observed that when reactor power is maintained at a certain level, the thermal burden in the ECT inventory increases, highlighting the importance of inventory management strategies for long-term cooling. Conversely, in the case of LOCA-ATWS, core power was effectively reduced through negative reactivity feedback, and the analysis confirmed that safety limits are maintained without core damage as long as the relevant PSSs operate as designed.

In conclusion, SMRs like the i-SMR adopt fundamental design concepts that differ significantly from those of conventional large-scale pressurized water reactors. Therefore, rather than simply following the analytical assumptions and evaluation methods used for large reactors, a differentiated approach that reflects the unique physical characteristics and safety philosophy of SMRs is required. The technical insights derived from this study are expected to serve as essential foundational data for future regulatory licensing and safety optimization of the i-SMR.

Acknowledgements

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