# A Safety Analysis Methodology for Non-Loss of Coolant Accidents (Non-LOCA) in an Innovative Small Modular Reactor (i-SMR)

**Bong Oh Choi, Min Shin Jung, Ung Soo Kim**
KEPCO E&C
269 Hyeoksin-ro, Gimcheon-si, Republic of Korea
bochoi@kepco-enc.com, msjung@kepco-enc.com, uskim@kepco-enc.com

## ABSTRACT

In this work, a safety analysis methodology for Non-Loss of Coolant Accidents (Non-LOCA) in the innovative Small Modular Reactor (i-SMR) is presented. The i-SMR features an integral design that houses major components within a single Reactor Pressure Vessel (RPV), thereby eliminating large-break LOCA risks. A conservative deterministic approach is employed using the Safety and Performance Analysis CodE for nuclear power plants (SPACE) for thermal-hydraulic analysis and the Thermal Hydraulic AnaLyzer for Enhanced Simulation of cores (THALES) for fuel safety assessment via Departure from Nucleate Boiling Ratio (DNBR) calculations.

The analysis covers Anticipated Operational Occurrences (AOOs) and Postulated Accidents (PAs) based on Title 10 of the Code of Federal Regulations (10CFR) criteria, applying conservative assumptions for initial conditions and single failures. The design utilizes fully passive safety systems that require no operator action. Results confirm that the i-SMR maintains system stability and fuel integrity under all Non-LOCA scenarios, satisfying all safety criteria, including radiological limits. These findings validate the design's adequate safety margins and the proposed methodology as a reliable basis for design verification and future licensing.

***Keywords:*** *Innovative Small Modular Reactor (i-SMR), Non-LOCA, Safety analysis, Integral design, Passive safety*

## 1 INTRODUCTION

The global nuclear industry has increasingly focused on Small Modular Reactors (SMRs) as a promising solution to enhance safety, economics, and flexibility in power generation. Among various SMR concepts, the integral-type Pressurized Water Reactor (PWR) has garnered significant attention due to its inherent safety features. The i-SMR adopts an integral design that houses major Reactor Coolant System (RCS) components—including the steam generators (SGs), reactor coolant pumps (RCPs), and pressurizer (PZR)—within a single RPV. This configuration eliminates large external piping, thereby inherently precluding Large-Break Loss of Coolant Accidents (LBLOCA) [1], which has been a dominant design basis accident in traditional loop-type PWRs. Consequently, the design paradigm for the i-SMR shifts towards enhancing safety margins for Non-LOCA events.

While the elimination of LBLOCA significantly reduces the risk of rapid depressurization, ensuring safety under Non-LOCA conditions remains critical for design verification and licensing. Therefore, a comprehensive safety analysis methodology tailored to the unique characteristics of the i-SMR is essential to demonstrate compliance with regulatory requirements such as 10 CFR Part 50.

This study aims to establish a robust safety analysis methodology for Non-LOCA events during the standard design phase of the i-SMR. To achieve this, a conservative deterministic approach is employed, utilizing SPACE for thermal-hydraulic analysis and THALES for fuel safety assessment via DNBR calculations. The analysis covers a spectrum of AOOs and PAs based on 10

CFR criteria, incorporating conservative assumptions for initial conditions and single failures. Through this study, the applicability of the proposed methodology is verified by confirming that the i-SMR maintains system stability and fuel integrity within safety limits under all analyzed scenarios.

## 2 I-SMR DESIGN CHARACTERISTIC

As shown in Figure 1, the i-SMR employs an integral configuration to eliminate the risk of LBLOCA [1]. The primary components, including the RCP, SG, and PZR, are installed within a single reactor vessel. The RCS transfers core heat to the secondary system via a once-through helical coil SG. Coolant flows upward through the core, turns downward through the RCPs, and returns to the lower plenum. Four RCPs force this circulation, while the in-vessel control rod drive mechanism (CRDM) and PZR are located in the upper riser and central reactor vessel closure head (RVCH), respectively. Table 1 summarizes the major design parameters for the Nuclear Steam Supply System (NSSS) of the i-SMR [2].

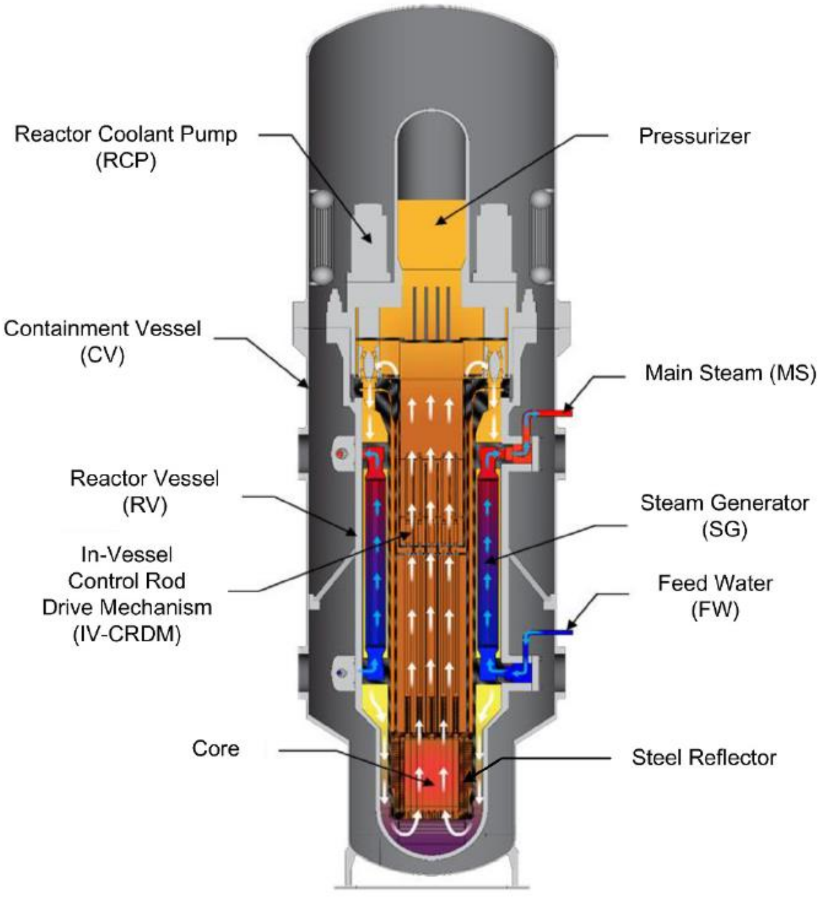

Figure 1: Configuration of nuclear steam supply system and containment vessel

Table 1: Major design parameters of nuclear steam supply system

| Parameter | Values |
|---|---|
| Thermal/electrical Power, MW(t)/MW(e) | 520/170 |
| Primary circulation | Forced circulation (Four canned-motor reactor coolant pump) |
| Primary system pressure, MPa | 15.5 |
| Primary inlet/outlet Temperature, °C | 295.5 / 320.0 |
| Reactor coolant pump type | Seal-less canned-motor |
| Fuel type/assembly array | $UO_2$ pellet / 17 x 17 square |
| Number of fuel assemblies | 69 |

| | |
|---|---|
| Fuel enrichment, %/Refueling cycle, months | <5.0 / >24 |
| Discharge burnup, GWd/ton | <62.0 |
| Reactivity control mechanism | Control rods (In-Vessel type), Burnable absorber, Moderator temperature coefficient |
| Secondary system pressure, MPa | 5.25 |
| Steam generator type | Helical coil |
| Superheating degree, °C | >30 |

### 2.1 Reactor Core

The core comprises 69 fuel assemblies (17×17, enriched to 5%) with an active height of 2.4 m. Refueling occurs in two batches over a 24-month cycle, supported by a stainless-steel reflector. Reactivity is controlled by control rods and burnable absorbers, utilizing a negative moderator temperature coefficient in a boron-free core. This boron-free design eliminates boron dilution risks and Primary Water Stress Corrosion Cracking (PWSCC), enhances safety by preventing re-criticality during Small-Break Loss of Coolant Accidents (SBLOCA), and flattens power profiles to improve thermal margins [3].

### 2.2 Reactor Coolant Pump

Four vertical canned-motor RCPs are installed at the upper portion of the RVCH. These hermetically sealed units eliminate the need for external seals, inherently precluding seal LOCAs and enhancing safety and economics. The forced circulation provides greater thermal power than natural circulation designs and ensures flow stability during load-following operations by mitigating secondary system instabilities.

### 2.3 Steam Generator

The SG uses a monobloc helical coil type located in the annular region between the Core Support Barrel (CSB) and the reactor vessel. Primary coolant transfers heat to secondary feedwater, which is superheated by approximately 30°C, eliminating the need for a turbine moisture reheater. The tubes are spirally wound in multiple layers to maximize heat transfer area and are divided into eight groups connected to feedwater and steam headers. Tube orifices are installed to prevent flow instability.

### 2.4 Pressurizer

The PZR is designed with sufficient capacity to prevent RCS overpressurization during transients. The i-SMR maintains a large initial water inventory (about 20% of RCS volume) to manage accidents without external injection sources. It includes two safety valves for overpressure protection and heaters at the bottom.

### 2.5 Description of the Passive Safety System

The i-SMR safety system is fully passive [4], comprising the Passive Emergency Core Cooling System (PECCS) for LOCA, the Passive Auxiliary Feedwater System (PAFS) for Non-LOCA, and the Passive Containment Cooling System (PCCS) for steel containment vessel (CV) cooling. Designed without Class 1E power or operator intervention, it employs fail-safe principles and physical separation. Key components like the reactor vessel (RV) and CV are located underground for protection against external hazards. This section details their design characteristics, operating mechanisms, and validation experiments. Figure 2 shows the schematic of PECCS, PCCS, and PAFS of the i-SMR.

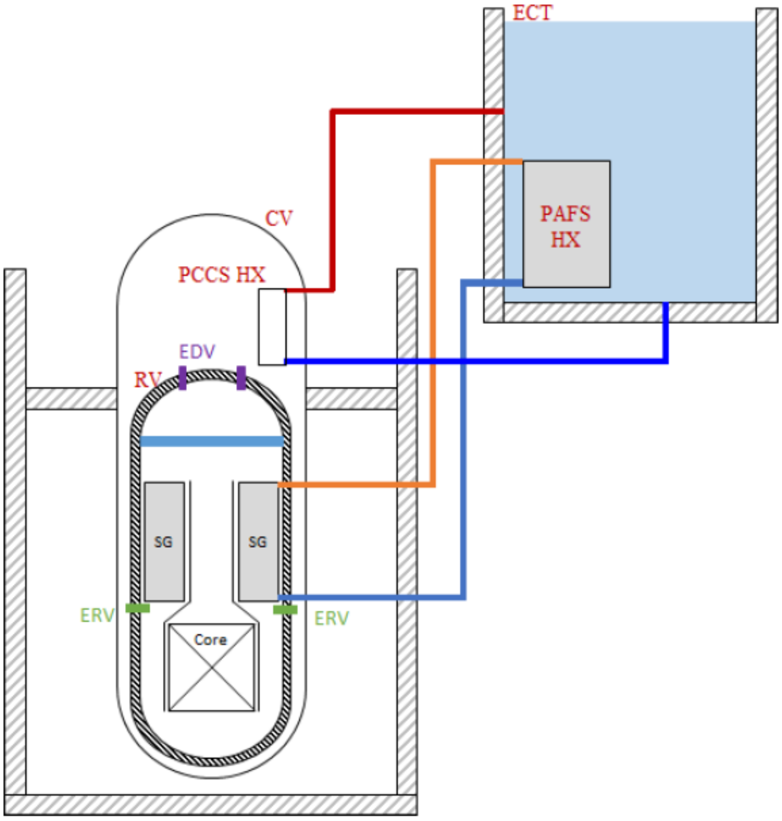

Figure 2: Schematic of PECCS, PCCS, and PAFS of i-SMR

### 2.5.1 Passive Containment Cooling System

The PCCS removes heat from the steel CV following Design Basis Accidents (DBAs) to reduce pressure and temperature, ensuring sufficient condensate for the PECCS. It employs vertical tube heat exchangers (PCCHX) inside the CV connected to the Emergency Cooling Tank (ECT), operating without actuation valves to keep the CV dry. Unlike previous designs for concrete containments, the i-SMR PCCS operates under vacuum with pure steam condensation. New experiments will be conducted under higher pressure (up to 5 MPa) to validate this optimized design.

### 2.5.2 Passive Auxiliary Feedwater System

The PAFS removes core decay heat for all DBAs, including station blackout, maintaining safe shutdown for 36 hours via passive cooling. It utilizes a U-shaped heat exchanger (PAFHX) in the ECT, establishing a natural circulation loop with the SG to transfer residual heat. While the tube geometry matches the Advanced Power Reactor 1400 (APR1400), the i-SMR PAFS uses superheated steam during early operation. Previous APR+ experiments validated the design, but new tests will investigate superheated steam condensation effects specific to i-SMR conditions.

### 2.5.3 Passive Emergency Core Cooling System

The PECCS consists of Emergency Depressurization Valves (EDVs) and Emergency Recirculation Valves (ERVs) to manage LOCA scenarios. The EDVs rapidly depressurize the RCS to equalize pressure with the CV, after which the ERVs enable gravitational recirculation of condensed water back into the core. Valves are fail-safe and designed to prevent core uncovery. A separate effect test is planned to investigate the recirculation flow characteristics coupled with the CV.

# 3 NON-LOCA SAFETY ANALYSIS METHODOLOGY

Section 3.1 describes methods for the systematic identification, categorization, and grouping of postulated initiating events and accident scenarios addressed in the Non-LOCA analysis. Section 3.2 presents the acceptance criteria applied to the Non-LOCA safety analysis for the i-SMR. Section 3.3 discusses the computer codes and plant models used. Section 3.4 describes the deterministic safety analysis applied to the Non-LOCA safety analysis.

## 3.1 Design Basis Accident Categorization

Table 2 presents the initiating events and event classifications for the i-SMR, derived by considering the major design characteristics of the i-SMR based on the initiating events and event classifications for the PWR presented in References [5] through [8].

Non-LOCA generally refers to all events and accidents among the DBAs described in Chapter 15 of the Safety Analysis Report (SAR), excluding the LOCA where the leakage amount of the RCS exceeds the coolant replenishment by charging flow. For i-SMR, based on the classification system presented in 10 CFR of the U.S. Nuclear Regulatory Commission (NRC), they are classified into Normal, AOOs, and PAs, and each event falls into one of the following categories.

Table 2: Initiating Event and Event Classification

| Chapter | Initial Event | Category |
|---|---|---|
| 15.1 | Increase in Heat Removal by the Secondary System | |
| 15.1.1 | Decrease in Feedwater Temperature | AOO |
| 15.1.2 | Increase in Feedwater Flow | AOO |
| 15.1.3 | Increased Main Steam Flow | AOO |
| 15.1.4 | Inadvertent Opening of a Turbine Bypass Valve | AOO |
| 15.1.5 | Steam Line Break Inside and Outside Containment Vessel | PA |
| 15.1.6 | Loss of Containment Vacuum/Containment Flooding | AOO |
| 15.1.7 | Inadvertent Operation of Passive Auxiliary Feedwater System | AOO |
| 15.2 | Decrease in Heat Removal by the Secondary System | |
| 15.2.1 | Loss of External Load | AOO |
| 15.2.2 | Turbine Trip | AOO |
| 15.2.3 | Loss of Condenser Vacuum | AOO |
| 15.2.4 | Main Steam Isolation Valve Closure | AOO |
| 15.2.5 | Steam Pressure Regulator Failure | N/A |
| 15.2.6 | Loss of Non-Emergency AC Power to the Station Auxiliaries | AOO |
| 15.2.7 | Loss of Normal Feedwater Flow | AOO |
| 15.2.8 | Feedwater System Pipe Breaks Inside and Outside Containment Vessel | PA |
| 15.3 | Decrease in Reactor Coolant Flow Rate | |
| 15.3.1 | Total Loss of Reactor Coolant Flow | AOO |
| 15.3.2 | Flow Controller Malfunction Causing Flow Coastdown | N/A |

|  |  |  |
|---|---|---|
| 15.3.3 | Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power | PA |
| 15.3.4 | Reactor Coolant Pump Shaft Break | PA |
| 15.4 | Reactivity and Power Distribution Anomalies |  |
| 15.4.1 | Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low Power Conditions | AOO |
| 15.4.2 | Uncontrolled Control Element Assembly Withdrawal at Power | AOO |
| 15.4.3 | Single Control Element Assembly Drop | AOO |
| 15.4.3A | Single Control Element Assembly Withdrawal | AOO |
| 15.4.4 | Startup of an Inactive Reactor Coolant Pump | AOO |
| 15.4.5 | Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate | N/A |
| 15.4.6 | Inadvertent Deboration | N/A |
| 15.4.7 | Inadvertent Loading of a Fuel Assembly into the Improper Position | PA |
| 15.4.8 | Control Element Assembly Ejection | N/A |
| 15.5 | Increase in RCS Inventory |  |
| 15.5.1 | Inadvertent Operation of the MMPS | AOO |
| 15.6 | Decrease in Reactor Coolant System |  |
| 15.6.1 | Inadvertent Opening of a Pressurizer Safety/Relief Valve | AOO |
| 15.6.2 | MMPS Line Failure Outside Containment Vessel | AOO |
| 15.6.3 | Steam Generator Tube Rupture | PA |
| 15.6.4 | Radiological Consequences of Main Steam Line Failure Outside Containment (BWR) | N/A |
| 15.6.5 | Loss-of-Coolant Accident | PA |
| 15.6.6 | Inadvertent Operation of Passive Emergency Core Cooling System | AOO |
| 15.7 | Radioactive Material Release from a Subsystem or Component |  |
| 15.7.1 | Radioactive Gas Waste System Failure | PA |
| 15.7.2 | Radioactive  Liquid Waste System Leak or Failure | PA |
| 15.7.3 | Postulated Radioactive Releases due to Liquid-Containing Tank Failures | PA |
| 15.7.4 | Fuel Handling Accident | PA |
| 15.7.5 | Spent Fuel Cask Drop Accidents | PA |
| 15.8 | Anticipated Transient without Scram | N/A |

### 3.1.1 Loss of Containment Vacuum/Containment Flooding (15.1.6)

The i-SMR maintains a vacuum in the containment during normal operation without insulation on the RCS. Loss of vacuum or flooding (e.g., PCCS tube rupture) increases convective heat transfer, cooling the system. Based on guidelines for single faults and small pipe ruptures, this is classified as an AOO, subject to Probabilistic Safety Assessment (PSA) verification.

### 3.1.2 Inadvertent Actuation of PAFS (15.1.7)

PAFS cools the SG using external condensers. Inadvertent actuation does not cause backflow due to higher feedwater pressure and check valves, leaving thermal hydraulic parameters stable. Following guidelines for valve malfunctions, this is classified as an AOO, subject to Probabilistic Safety Assessment (PSA) verification.

### 3.1.3 Inadvertent Reactor Coolant Boron Dilution (15.4.6)

Boron-free operation is applied, so this event does not occur.

### 3.1.4 Control Rod Cluster Ejection (15.4.8)

The application of In-Vessel CRDM prevents control rod ejection by system pressure, unlike external mechanisms. Thus, this accident does not occur.

### 3.1.5 Malfunction of MMPS (15.5.1)

The Modular Make up System (MMPS) functions similarly to the Chemical and Volume Control System (CVCS) but with a modular, boron free design. It controls inventory via charging and letdown flows. Like the CVCS malfunction, this is classified as an AOO, subject to Probabilistic Safety Assessment (PSA) verification.

### 3.1.6 Inadvertent Actuation of PECCS (15.6.6)

The PECCS uses EDVs and ERVs. Inadvertent actuation causes coolant discharge and inventory loss, moving the event category from "Inventory Increase" to "Inventory Decrease." Based on guidelines for valve malfunctions, this is classified as an AOO, subject to PSA verification following the completion of the Integrated Assembly Block (IAB) design.

## 3.2 Acceptance Criteria

If the risk of an event is defined as the product of the frequency of the event and its consequences, the power plant shall be designed such that all AOOs and PAs have the same level of risk. This is generally reflected in regulatory requirements that for events with relatively high frequency (AOOs), serious consequences must be prevented, while for relatively rare events (PAs), more serious consequences are permitted [8]. The analysis and evaluation results for individual initiating events are examined to verify compliance with the applicable analysis acceptance criteria for each event. Alternative acceptance criteria suitable for the design and operation of a specific power plant may also be applied, as in a new nuclear reactor design license application. In such cases, the alternative acceptance criteria are reviewed to determine whether they are equivalent to the current acceptance criteria described below in terms of function and consequences.

### 3.2.1 Analysis Acceptance Criteria for AOOs

The following are specific criteria to satisfy the requirements required by Reference [9] for AOOs.

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit.

3. An AOO should not generate a PA without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

### 3.2.2 Analysis Acceptance Criteria for PA

Unlike an AOO, a PA could result in sufficient damage to preclude resumption of plant operation. The basic criteria to satisfy the requirements required by Reference [9] for PAs are as follows.

1. Pressure in the RCS and main steam system should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.
2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit. If the minimum DNBR does not meet these limits, then the fuel is assumed to have failed.
3. The release of radioactive material shall not result in offsite doses in excess of the guidelines of Reference [9] Section 15.0.1.

## 3.3 Computer Codes and Plant Model

This section mainly covers an overview of SPACE, which is used for the thermal-hydraulic analysis of the i-SMR. Additionally, an overview of THALES, a core thermal-hydraulic analysis code used for Critical Heat Flux (CHF) calculations, is described.

### 3.3.1 SPACE

SPACE is a system safety analysis code for nuclear power plant design. This code was developed by Korea Hydro & Nuclear Power (KHNP) with the participation of KEPCO Engineering & Construction Company (KEPCO E&C), KEPCO Nuclear Fuel (KNF), and the Korea Atomic Energy Research Institute (KAERI), under the support of the Ministry of Trade, Industry and Energy (MOTIE), to localize nuclear power plant design technologies and secure international competitiveness for the domestic nuclear power plant industry. In March 2017, licensing approval was obtained for SPACE and the Non-LOCA safety analysis methodology for the APR1400 using this code. Unlike conventional codes that analyze two-phase flow using liquid and gas phases, SPACE employs a two-phase, three-field model (liquid, vapor, and droplets) to simulate two-phase flow phenomena more accurately. The code is object-oriented, programmed in C++ to facilitate modularity and interface with other computational codes. SPACE consists of input/output (I/O) modules, a hydraulic solver, auxiliary equations, component and special phenomenon models, reactivity models, trip/control system models, and thermal structure/core dynamics models. The I/O module manages file operations, error checking, and data exchange between modules. The hydraulic solver provides the numerical framework using ten governing equations—mass, momentum, and energy conservation for three fields plus non-condensable gas—discretized via the Finite Volume Method and solved using a Semi-Implicit Scheme. Auxiliary equations determine constitutive relations based on flow regime maps. The code also includes models for critical flow, counter-current flow limitation, and rapid area changes. The SPACE nodalization for the i-SMR was constructed as shown in Figure 3.

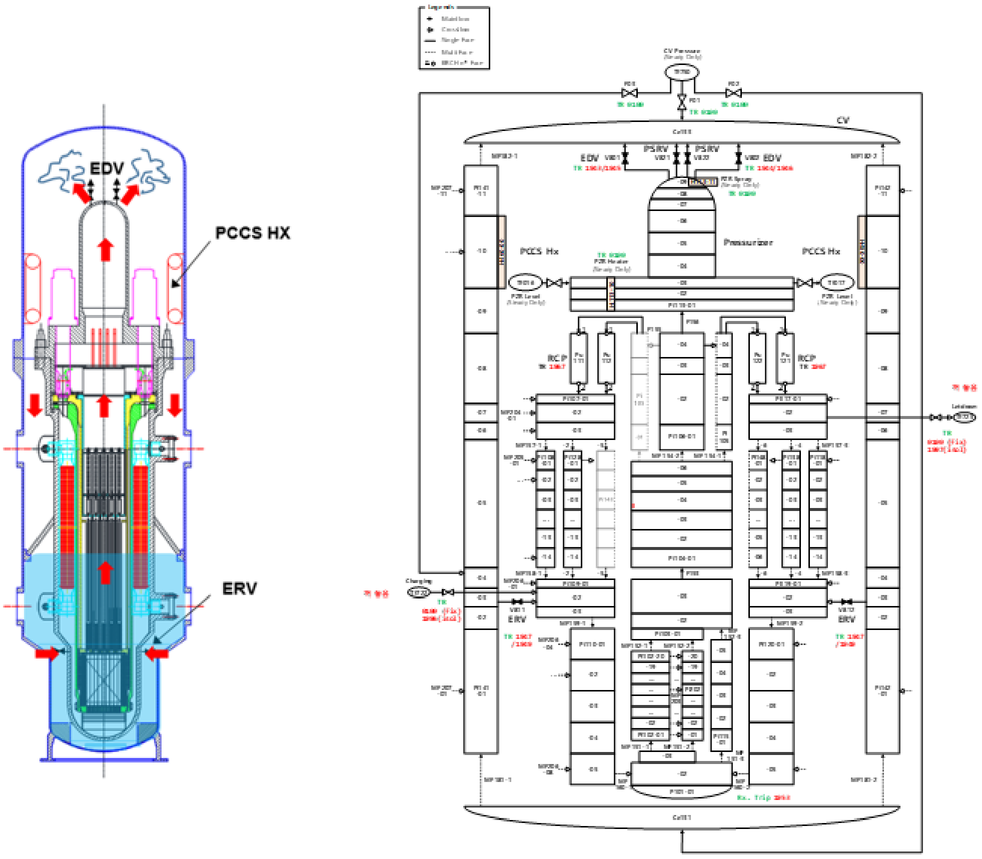

Figure 3: SPACE Nodalization for i-SMR

### 3.3.2 THALES

THALES is a computer code developed for the core thermal-hydraulic design of PWRs. Approved in January 2013, it is currently used for Optimized Power Reactor 1000 (OPR1000) and APR1400 core design. THALES employs a subchannel analysis methodology based on mass, energy, and momentum conservation equations for a two-phase mixture. It utilizes a sparse matrix solver and variable relaxation factor to enhance computational speed and convergence stability. The code accounts for pressure drops across support grids and fuel rods and applies a turbulence model to simulate cross-flow between subchannels. While users can select models for two-phase friction, quality, void fraction, and heat transfer, only licensed combinations are permitted for PWR design. In Non-LOCA safety analysis, THALES is used for DNBR analysis based on the Reference Thermal Margin (RTM) model. For i-SMR, the RTM model incorporates the KNF-I CHF correlation and ensures conservative DNBR evaluation, with conservatism verified periodically. In Non-LOCA safety analysis, transient core heat flux and thermal-hydraulic conditions from SPACE are input into the RTM model to perform DNBR analysis.

## 3.4 Deterministic Safety Analysis

Table 3 presents the various options available for performing deterministic safety analyses, which differ in the level of conservatism associated with the computer code used, assumptions regarding system availability, and the initial and boundary conditions applied [10]. The SPACE code applied to the i-SMR Non-LOCA safety analysis is a best-estimate code. The initial conditions were derived through sensitivity analyses within the allowable range and applied to the design basis event (DBE) safety analysis. In this study, the analysis setpoints and delay times of the assumed Reactor Safety System (RSS) and Core Protection System (CPRS) were assumed to be conservative values for each accident. Therefore, this approach corresponds to Option 2 in Table 3.

Table 3: Options for Performing deterministic Safety Analysis

| Option | Computer code type | Assumptions about system availability | Type of initial and boundary condition |
|---|---|---|---|
| 1. Conservative | Conservative | Conservative | Conservative |
| 2. Combined | Best estimate | Conservative | Conservative |
| 3. Best estimate plus uncertainty | Best estimate | Conservative | Best estimate<br>Partly most unfavourable conditions |
| 4. Realistic | Best estimate | Best estimate | Best estimate |

## 4 SAFETY ANALYSIS RESULTS OF NON-LOCA EVENTS

As an example of the Non-LOCA events of the i-SMR, deterministic safety analysis for an inadvertent opening of the turbine bypass valve (IOTBV) event, categorized under Section 15.1.4, using the SPACE and THALES codes is representatively reviewed. The results are presented in Figures 4 through 7. Since the IOTBV event is a cooling event caused by the secondary system, the analysis focused on thermal-hydraulic behavior rather than primary and secondary peak pressures.

This event is classified as an AOO, as summarized in Table 2. The turbine bypass valve may be opened inadvertently by an operator or due to a failure in the control system actuating the valve. The inadvertent opening increases the main steam flow rate. The analysis considers the steam release flow that induces the most severe nuclear boiling departure rate within the possible range of transient steam releases.

For the event described in this section, the primary variable of interest is the minimum DNBR in the core hot channel. This variable indicates whether the fuel allowable damage limit has been exceeded or whether the nuclear fuel cladding has been damaged.

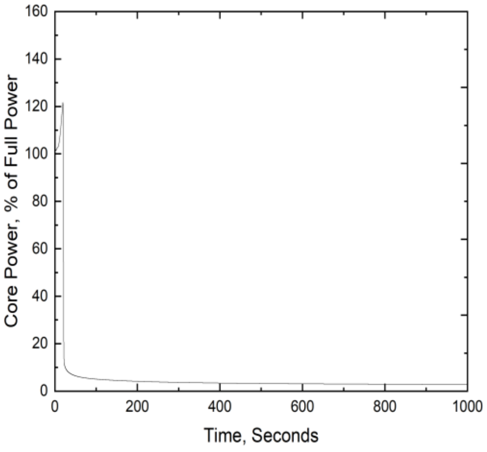

Figure 4 : Core Power vs. Time (IOTBV)

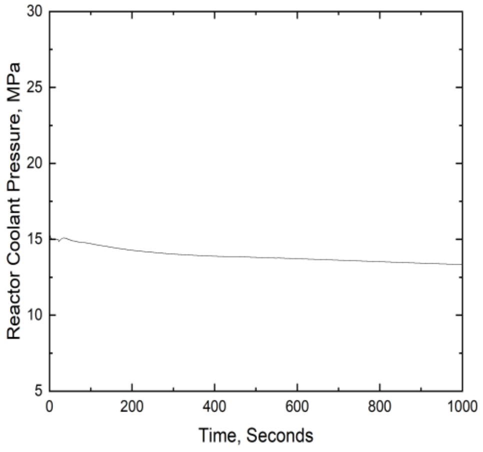

Figure 5 : RCS Pressure vs. Time (IOTBV)

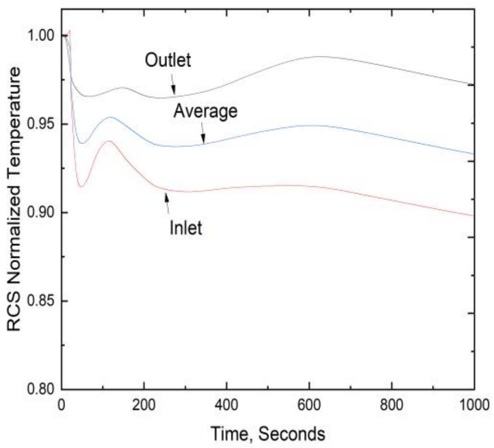

Figure 6 : RCS Normalized Temperature vs. Time
(IOTBV)

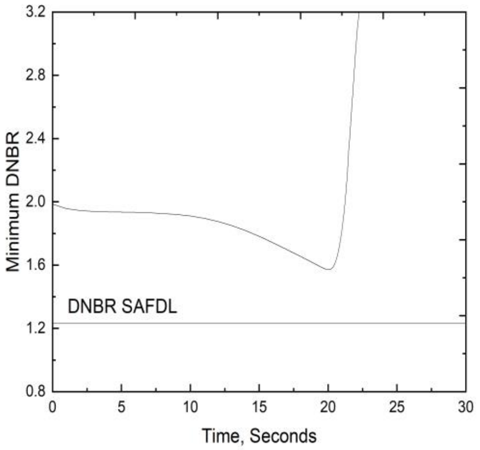

Figure 7 : Minimum DNBR vs. Time
(IOTBV)

## 5 CONCLUSION

A comprehensive review on safety-analysis methodology for Non-LOCA events in the i-SMR, addressing its integral architecture and fully passive safety systems, has been performed. By systematically categorizing initiating events per 10 CFR and applying conservative deterministic assumptions, the methodology reliably evaluates plant response to AOOs and PAs. The combined use of SPACE (thermal-hydraulic) and THALES (DNBR) provides a rigorous assessment of system stability and fuel integrity.

Analysis results confirm that the i-SMR design maintains sufficient safety margins under all examined scenarios:

- Peak RCS pressures ≤ 110 % of design (AOOs) and ≤ design pressure (PAs).
- Minimum DNBR ≥ 95/95, ensuring cladding integrity.
- Passive safety systems (PAFS, PCCS) effectively mitigate accident consequences without operator action or AC power.

These findings validate the robustness of the i-SMR's inherent safety design and demonstrate that the proposed methodology offers a reliable technical foundation for standard-design approval and future licensing applications.

## 6 ACKNOWLEDGEMENT

This work was supported by the Ministry of Science and ICT (MSIT), through the National Research Foundation of Korea (NRF), under Grant No. RS-2024-00403548. The authors also wish to express their sincere gratitude to the researchers at KEPCO E&C and Korea Hydro & Nuclear Power (KHNP) for their valuable technical advice and support throughout the development of the safety evaluation methodology.

## REFERENCES

[1] Sang Won Lee, Heo Sun, Do-Hwan Lee, Conceptual design of i-SMR, in: International Congress on Advanced Nuclear Power Plants, Gyeongju, Republic of Korea, April 23-27, 2023.

[2] 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), Central Research Institute, Korea Hydro & Nuclear Power Co., Ltd., Daejeon, Korea, 2016. (for a technical report)

[3] Bum Hee Jo, Chang Joo Hah, Investigation on long-term daily load follow operation capability of soluble boron-free SMR, Ann. Nucl. Energy 149 (2020) 107764.

[4] Sang-Gyu Lim, Hyun-Jin Lee, Seok-Jong Yoon, Uiju Jeong, Sang-Won Lee, Design characteristics of passive safety system for i-SMR, in: International Congress on Advanced Nuclear Power Plants, Gyeongju, Republic of Korea, April 23-27, 2023.

[5] Regulatory Guide 1.70, Rev.3, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,” US NRC, 1978.11.

[6] NUREG-0800, Rev.3, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” US NRC, 2007.03.

[7] KINS/GE-N001 , Rev.8, “Safety Review Guidelines for PWR,” KINS, 2024.11.

[8] KINS/GE-N020 , Rev.0, “Safety Review Guidelines for i-SMR,” KINS, 2025.12.

[9] Nuclear Safety and Security Commission, Regulation on Technical Standards for Reactor Facilities, etc., NSSS Rule No. 36, Jan. 20, 2026 (as amended).

[10] International Atomic Energy Agency, Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2, IAEA, Vienna (2010).