Overview of i-SMR Non-LOCA Safety Analysis

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ABSTRACT

The innovative Small Modular Reactor (i-SMR) is a next-generation reactor model developed in the Republic of Korea. It features an integral reactor design with a double steel vessel structure, incorporating advanced characteristics such as Boron-Free (BF) operation and a fully passive safety system to enhance safety. This paper presents the transient safety analysis of the i-SMR, performed using the SPACE thermal-hydraulic code, ASTRA nuclear design code and the THALES subchannel code. The study specifically focuses on two key accident scenarios: the Complete Loss of Reactor Coolant Flow (CLOF) and the Single Control Element Assembly Withdrawal (SCEAW). These two scenarios represent the most limiting cases in terms of fuel integrity evaluation for SAR sections 15.3 and 15.4, respectively.

The integral design of SMRs inherently limits the available internal space. This constraint requires the elimination of flywheels on the Reactor Coolant Pumps (RCPs), resulting in a rapid flow coastdown during a loss of AC power event. Such characteristics present a significant challenge to ensuring fuel integrity. However, the application of a BF core design maintains a sufficiently negative Moderator Temperature Coefficient (MTC) throughout the entire cycle, thereby effectively maintaining fuel integrity.

The increased control rod worth, resulting from the adoption of a BF core, poses a potential threat to fuel integrity during a SCEAW event. However, the core is effectively protected by the core quadrant power deviation reactor trip, which is triggered by in-core detectors that immediately sense the resulting power deviation.

The results confirm that relevant safety acceptance criteria are satisfied with sufficient margins, providing insights into the safety behaviour of the i-SMR design under representative non-LOCA conditions.

Keywords: *i-SMR, transient analysis, SPACE, complete loss of flow, single CEA withdrawal*

1 INTRODUCTION

The i-SMR adopts an integral reactor configuration in which major primary components, including the reactor coolant pumps and steam generators, are installed within a reactor vessel [1]. The elimination of large primary piping effectively removes the possibility of large-break loss-of-coolant accidents. In addition, the i-SMR employs fully passive safety systems designed to operate without the need for operator action during the transient. Figure 1 illustrates the conceptual configuration of the i-SMR integral reactor design.

Figure 1: Conceptual Configuration of the i-SMR

However, the compact configuration of the integrated double steel vessel—comprising the reactor and containment vessels—imposes spatial constraints that prevent the installation of flywheels on the RCPs. This absence of flywheels leads to a rapid flow coastdown during loss of AC power events, such as a CLOF, which must be evaluated to ensure fuel integrity. Additionally, the adoption of a BF core results in increased control rod worth. This characteristic poses a potential threat to fuel integrity during reactivity-initiated transients, such as a SCEAW, and therefore must be managed through effective reactor protection systems.

To address these safety concerns, this paper presents a detailed transient analysis using a SPACE [2] and THALES [3] code system. The SPACE code is employed for system-level thermal-hydraulic evaluation, while the THALES subchannel analysis code is utilized to perform Departure from Nucleate Boiling Ratio (DNBR) calculations. The study focuses on evaluating the safety margins for the CLOF and SCEAW events under the most limiting conditions.

The remainder of this paper is organized as follows: Section 2 describes the analysis methodology and code modeling; Section 3 presents the simulation results and discussions for each transient; and Section 4 summarizes the conclusions and the safety implications for the i-SMR design.

2 ANALYSIS METHODOLOGY

2.1 Code System and Coupling Procedure

The safety analysis for the i-SMR is conducted using the ASTRA, SPACE, and THALES codes. The overall frameworks adopted for the CLOF and SCEAW events are illustrated in Figures 2 and 3, respectively. Generally, the ASTRA code, a nuclear design code, generates general nuclear design parameters and event-specific data—rod worth, radial power peaking factor (Fr), axial power distribution—for the SPACE code. During a SCEAW event, ASTRA is specifically utilized to calculate the core quadrant power tilt, which is critical for determining the reactor trip time. In addition, the Fr used in the subchannel analysis is directly obtained from the ASTRA calculation to reflect the power redistribution caused by control rod movement.

The SPACE code, a thermal-hydraulic system code, evaluates the overall plant response by modelling both the primary and secondary systems, utilizing general nuclear design data and event-specific data provided by ASTRA. The key transient parameters from SPACE—such as core heat flux, RCS temperature, pressure, and core inlet mass flow rate—are extracted as boundary conditions for the subchannel analysis. For CLOF event, the Fr value is calculated in the detailed core model of the SPACE calculation. In this case, both the average fuel rod and the hot fuel rod are explicitly modeled, and the Fr is calculated as the ratio of their respective power levels.

Finally, the THALES code calculates the Minimum DNBR (MDNBR) by incorporating the thermal-hydraulic inputs from SPACE and the core design-specific data—such as power distribution—generated by the ASTRA code. This integrated approach ensures a conservative assessment of the most limiting subchannel during the transient events.

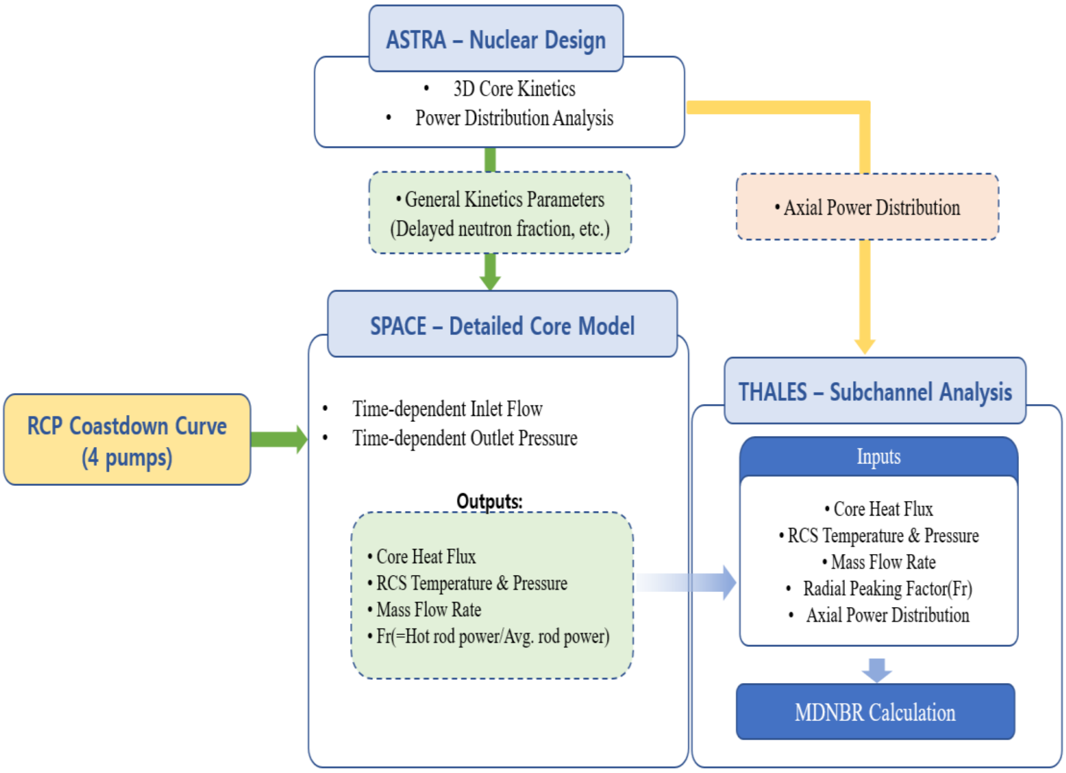

Figure 2: Analysis Procedure for the CLOF Event

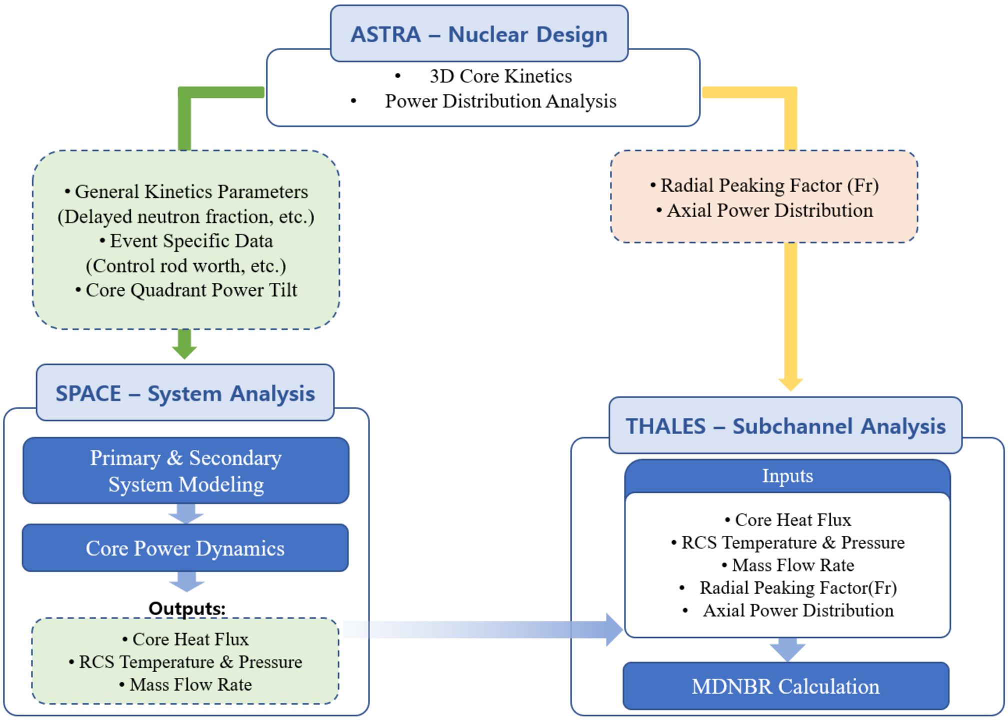

Figure 3: Analysis Procedure for the SCEAW Event

2.2 Description of Modelling

The CLOF event was evaluated using a detailed core model developed with the SPACE code. To capture the precise thermal-hydraulic behaviour during the flow coastdown, the active core region is axially discretized into multiple cells. The boundary conditions for this model are defined such that the core inlet is modelled with a time-dependent flow boundary, while the core outlet is governed by a time-dependent pressure boundary. Additionally, axial heat structures are coupled to each cell to accurately simulate the heat transfer from the fuel rods to the coolant during the transient. The rationale for using a detailed core model without a full plant system simulation is based on the transient characteristics of the CLOF event. In a CLOF scenario, the reactor trips and power decreases occurred within a few seconds after the initiation of the flow coastdown. Consequently, the MDNBR also occurs during the initial period. During this early coastdown phase, system pressure and coolant temperature exhibit negligible deviation from initial conditions. Therefore, the transient behaviour relevant to DNBR is primarily governed by the rapid reduction in core flow and the corresponding power response. This justifies the use of a core-focused model without performing a full-scale plant system simulation for the CLOF event.

Unlike the CLOF analysis which utilizes a detailed core model, the modeling for the SCEAW event incorporates a plant system model. This model includes a single pressurizer, four RCPs, helical-coil steam generators, and the passive safety systems. A distinctive design feature of the i-SMR is that the Control Element Drive Mechanisms (CEDMs) are installed internally within the Reactor Vessel (RV). Due to the harsh environment inside the RV—characterized by high temperature, high pressure, and high radiation levels, the conventional Reed Switch Position Transmitters (RSPT) type Control Element Assembly (CEA) position indicator can't be reliably applied. Instead, a single-channel solenoid-type CEA position indicator is adopted for the i-SMR design. However, implementing a dual-channel configuration for the solenoid-type indicator may cause signal interference between channels. Therefore, the system is designed with a single-channel configuration. To supplement the limited monitoring capability of this single-channel system, a reactor trip signal based on the core quadrant power tilt from the In-Core Instrumentation (ICI) is explicitly considered in the safety analysis to ensure timely mitigation of the SCEAW event.

2.3 Major Assumptions and Initial Conditions

The safety analyses for both the CLOF and SCEAW events were performed under conservative initial and boundary conditions selected within the Limiting Conditions for Operation (LCO). The objective of these assumptions is to ensure that the evaluated thermal margins represent the most adverse plant states with respect to fuel integrity. The key assumptions and initial conditions applied in this study are summarized in Table 1. For both transients, the initial conditions were conservatively selected to minimize initial DNBR. The reactor was assumed to operate at the maximum rated power, maximum RCS inlet temperature, minimum pressurizer pressure, minimum core flow rate within the allowable operating range. In addition, the maximum radial power peaking factor and the limiting axial power distribution were applied to ensure bounding local heat flux conditions.

For the CLOF analysis, a four-pump flow coastdown curve corresponding the inherent design characteristics of the i-SMR RCP configuration was applied. As illustrated in Figure 4, the core flow rate decreases more rapidly following a pump trip than conventional PWR plant. This relatively steep coastdown behavior is associated with the integral reactor configuration, which employs long-shaft RCPs installed within the compact vessel structure. In this design, flywheels are not incorporated in order to maintain mechanical integrity and structural stability within the limited internal space. As a result of the reduced rotational inertia, the core flow decreases more rapidly during the early phase of the transient compared to typical large-scale loop-type PWR designs. This characteristic leads to a more pronounced reduction in thermal margin during the early phase of the transient and therefore necessitates a rigorous evaluation of early-time DNBR behavior. The reactor trip was initiated by the undervoltage signal, and a conservative trip delay corresponding to the maximum allowable protection system response time was assumed.

For the SCEAW analysis, the control rod withdrawal was modeled with the maximum bank worth and withdrawal speed. The quadrant power tilt trip function was credited based on the ICI configuration. The i-SMR core is equipped with 20 ICI, which generate local power signals. The tilt signal A-E are obtained by calculating the power deviation between symmetrically located ICI detectors in each core quadrant. A reactor trip is initiated when the calculated quadrant power tilt exceeds the trip setpoint. To ensure conservatism in the safety evaluation, the analysis conservatively assumed that reactor trip is initiated upon the third tilt signal exceeding the setpoint. This assumption reflects the consideration of single failure and operability concern of the ICI system. In addition, a conservative reactor protection system response time was applied to account for signal processing and actuation delays.

Table 1: Key Assumptions and Initial Conditions

Parameters		Value
Core Power, % of full power (520 MWt)		HFP
Pressurizer Pressure		Min.
Core Inlet Temperature		Max.
Core Inlet Mass Flow Rate		Min.
Single Failure		N/A
Loss of Offsite Power, sec.		0
Turbine Trip Delay, sec.		3
Operator Action		N/A
ASI		Top peaked
Trip	CLOF	Undervoltage
	SCEAW	Core quadrant power tilt

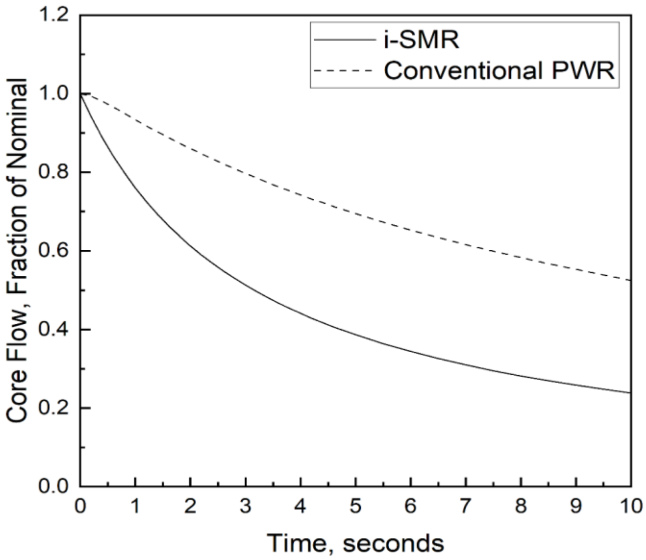

Figure 4: Normalized Four Pump Flow Coastdown

3 RESULTS

3.1 Analysis Results of CLOF

The safety of the i-SMR during a CLOF event is primarily determined by the dynamic balance between the decreasing core flow rate and the subsequent reduction in reactor power. Figure 5 shows the transient response of the core flow rate and the relative reactor power. As discussed previously, the flow rate decreases rapidly due to the low rotational inertia of the reactor coolant pumps. However, the reactor trip is promptly initiated by the undervoltage signal, and the BF core ensures a rapid power decrease through a strongly negative MTC. Although the flow coastdown is steeper than that of conventional reactors, the synchronized reduction in power effectively compensates for the loss of flow. Figure 6 presents the transient behavior of the DNBR. The acceptance criterion for fuel integrity is defined based on the minimum DNBR limit corresponding to the applied critical heat flux (CHF) correlation. The calculated minimum DNBR during the CLOF transient occurs shortly after pump trip initiation and subsequently rebounds following reactor trip. As a result, the MDNBR remains above the acceptance limit, providing a sufficient safety margin under the analysed conditions. These results demonstrate that the inherent reactivity feedback characteristics of the BF core and the prompt trip action maintain sufficient thermal margin even under the rapid flow reduction scenario characteristic of the i-SMR design.

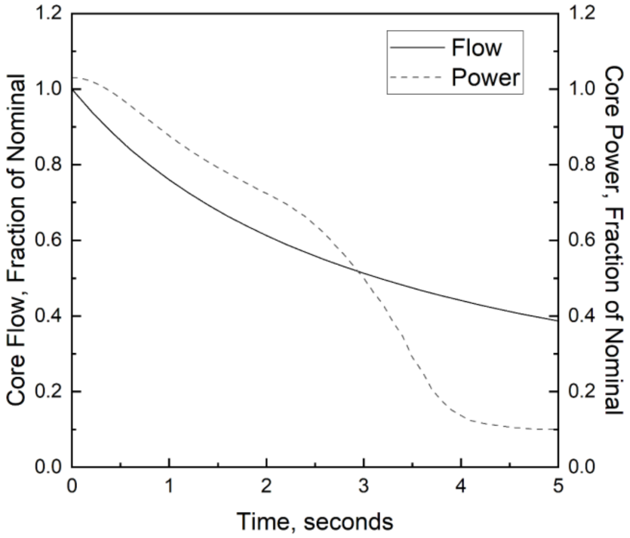

Figure 5: Transient Responses of Core Flow and Core Power (CLOF)

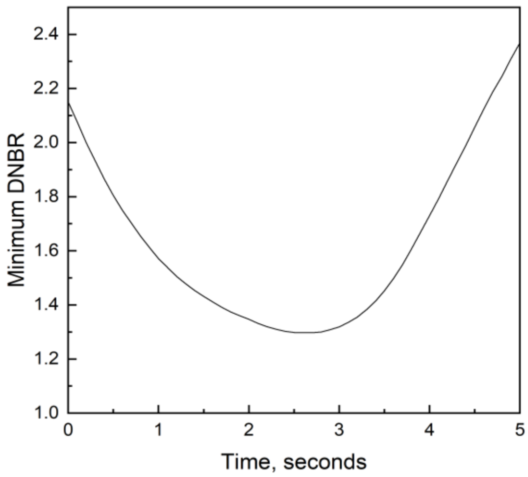

Figure 6: Transient Responses of DNBR (CLOF)

3.2 Analysis Results of SCEAW

The SCEAW transient is primarily governed by the reactivity insertion resulting from control rod withdrawal and the subsequent response of the reactor protection system. Unlike the flow-driven CLOF event, the key safety concern during SCEAW is the increase in core power, the associated spatial power redistribution, and the resulting local heat flux escalation.

Figure 7 presents the transient behavior of the core power and Figure 8 shows the quadrant power tilt following the initiation of control rod withdrawal. The quadrant power tilt signal reaches the trip setpoint within approximately 20 seconds, initiating reactor shutdown. As shown in Figure 7, the maximum power occurs shortly before reactor trip, after which prompt insertion of the CEAs results in a decrease in core power.

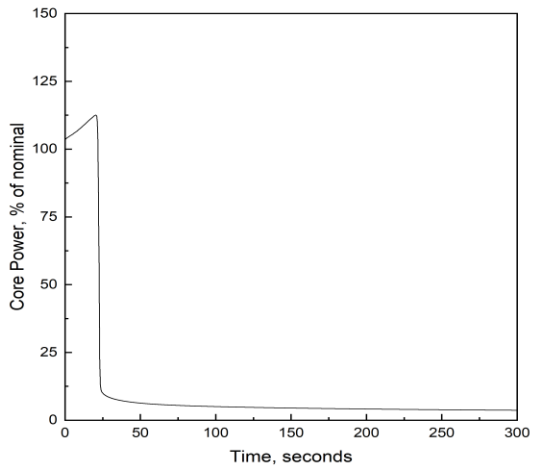

Figure 7: Transient Responses of Core Power (SCEAW)

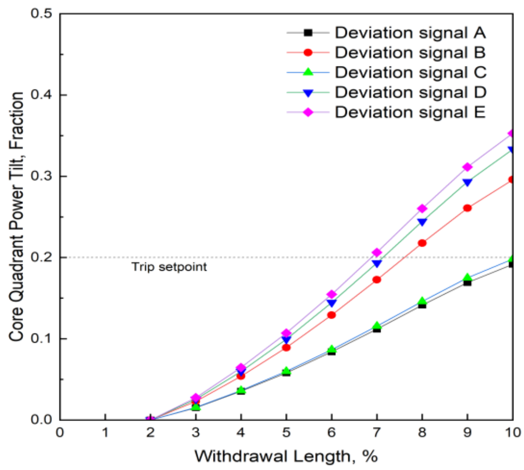

Figure 8: Core Quadrant Power Tilt (SCEAW)

Figure 9 presents the DNBR response during the SCEAW transient. The minimum DNBR occurs during the transient following control rod withdrawal. However, the DNBR remains above the acceptance criterion based on the applied CHF correlation, indicating that sufficient thermal margin is maintained.

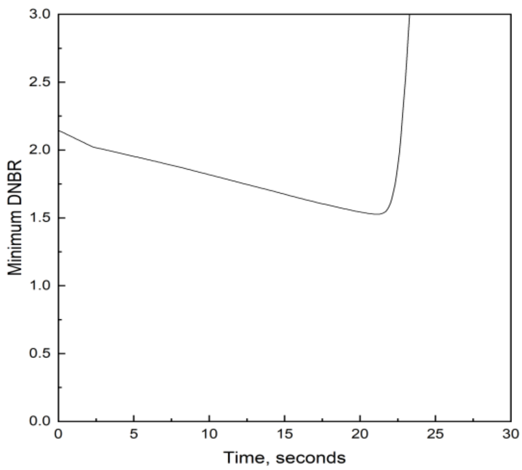

Figure 9: Transient Responses of DNBR (SCEAW)

Although the SCEAW event induces a significant localized power increase due to the high rod worth, the credited quadrant power tilt trip function effectively mitigates the reactivity insertion. These results demonstrate that the i-SMR design maintains sufficient thermal margin under the most limiting reactivity-initiated transient conditions.

4 CONCLUSION

This paper presented a deterministic safety evaluation of the i-SMR under representative non-LOCA transients corresponding to SAR Sections 15.3 and 15.4, focusing on the CLOF and SCEAW events.

The results indicate that the integral reactor configuration of the i-SMR leads to a relatively steep flow coastdown during CLOF events due to the reduced effective rotational inertia of the reactor coolant pump system, associated with the compact integral vessel design and the absence of flywheels. This rapid reduction in core flow reduces thermal margin during the transient. However, the BF core design, characterized by a strongly negative moderator temperature coefficient and rapid power reduction, effectively compensates for the loss of flow. As a result, the BF core and integral configuration interact in a complementary manner, enabling the i-SMR to maintain sufficient DNBR margin under CLOF conditions.

During a SCEAW event, the BF core design results in relatively high control rod worth, which may introduce a significant positive reactivity into the core. However, the resulting core power asymmetry activates the quadrant power tilt trip, effectively preventing fuel damage.

The calculated results indicate that, despite the rapid flow reduction in CLOF and the significant reactivity insertion in SCEAW, the i-SMR maintains sufficient DNBR margin relative to the acceptance criterion under the most limiting initial conditions and protection system assumptions. The credited reactor protection functions, including undervoltage trip and quadrant power tilt trip, effectively mitigate the transient excursions and preserve fuel integrity.

Overall, the present study confirms that the i-SMR design, when evaluated using conservative analysis methodology, satisfies fuel thermal margin requirements for representative non-LOCA events.

REFERENCES

[1] S. G. Lim, 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] C. K. Yang, SPACE Code Development and Assessment Plans, Transactions of the Korean Nuclear Society Autumn Meeting, Paper 484, Jeju, Korea, Oct. 21-22, 2010

[3] B. J. Jang, Implementation of Boundary Condition to THALES Code, Transactions of the Korean Nuclear Society Autumn Meeting, Paper 16A-319, Gyeongju, Korea, Oct. 27-28, 2016

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