Analysis of Formation Mechanisms to Identify the Causes of Thermal Neutron Flux Degradation in Ex-core Detector of i-SMR

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1. Introduction

For the safe operation of a nuclear reactor, continuously monitoring the core power distribution and its changes is essential, and ex-core detectors generate signals from neutrons leaking through the Reactor Vessel (RV). The Core Protection Calculator System (CPCS) uses safety channel signals from the ex-core detectors to calculate the Departure from Nucleate Boiling Ratio (DNBR) and Local Power Density (LPD). When these parameters exceed the Limiting Safety System Settings (LSSS), a reactor trip signal is activated to safely protect the reactor core and systems.

Innovative Small Modular Reactor (i-SMR) adopts a radial solid reflector to enhance neutron economy. However, this may lead to a reduction in neutron leakage, which can negatively affect signal formation for core protection. Furthermore, to satisfy environmental qualification requirements, the ex-core detectors are planned to be located behind the Containment Vessel (CV). This change is expected to result in a significant difference in the thermal neutron flux in the safety channels compared with a conventional power plant.

This study was conducted to analyze the causes of thermal neutron flux degradation in the safety channels of ex-core detectors in i-SMR from a neutron physics perspective. Accordingly, the calculated thermal neutron flux in the safety channels of i-SMR was compared with that of a conventional power plant. To analyze the causes of the differences from a design perspective, the contribution of each material to thermal neutron flux formation in the safety channels was quantitatively decomposed, enabling an analysis of primary mechanisms of thermal neutron flux formation. Furthermore, a sensitivity analysis was conducted on key design parameters, including the coolant volume fraction of the solid neutron reflector and the radial distance of detector, to assess the influence of these design parameters.

2. Computational Results and Analysis

2.1 Computational Methods for Neutron Transport

The thermal neutron flux in the ex-core detector safety channels of i-SMR was calculated using the DORT [1] transport code, which is based on the two-dimensional discrete ordinates method. The macroscopic cross sections were generated using the GIP code based on the BUGLE-96 library (47 neutron groups and 20 photon groups) [2]. The scattering cross-sections were treated using a P_3 Legendre expansion, and the angular discretization was conducted using the S_8 quadrature [3]. The convergence criteria (ε) was set to less than 10^{-3}.

2.2 Computational Transport Model

As shown in Fig. 1, a quarter-core model in the R–θ coordinate system was established to calculate the thermal neutron flux in the ex-core detector safety channels of i-SMR. Since the ex-core detector design for i-SMR has not yet been finalized, the conventional ex-core detector design was adopted to examine neutron flux distribution characteristics. Also, while the installation location of the ex-core detectors has not yet been finalized, the analysis was conducted using a representative location just behind the CV and as close as practicable to minimize distance-related effects.

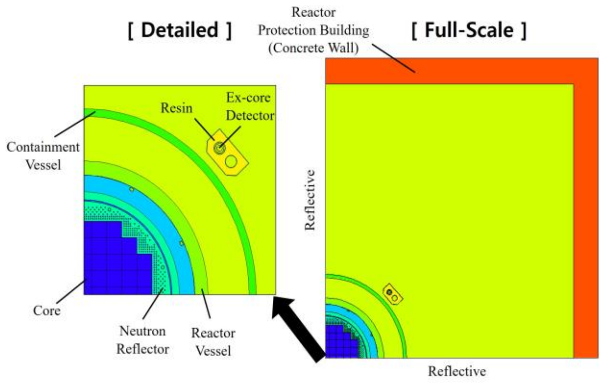

Fig. 1. R - θ Transport Calculation Model for Thermal Neutron Flux in i-SMR Ex-core Detector

2.3 Comparsion of Thermal Neutron Flux of Safety Channel

Table 1 shows the differences in the design parameters that may affect radiation transport characteristics between i-SMR and the conventional power plant (SKN 5&6).

Table 1. Differences in Design Parameters between i-SMR and Conventional Power Plant (SKN 5&6).

	SKN 5&6	i-SMR
Volumetric Fission Density (fission/cm^3 ·sec)	3.487×10^{12}	2.407×10^{12}
Location of Ex-core Detector	Behind RV	Behind CV
Azimuthal Angle of Safety Channel	35 °	45 ° (TBD)
Effective Distance of Safety Channel *	125.36 cm	177.22 cm (TBD)
Material of Neutron Reflector	H_2O	SS304

※ Effective Distance of Safety Channel
= (Distance of Safety Channel from Core Center) – (Core Equivalent Radius)

Based on the differences in reactor design as shown in Table 1, the calculated thermal neutron flux in the safety channels of i-SMR was compared with that of the conventional power plant using the DORT computational code, as summarized in Table 2. The result shows that the thermal neutron flux in i-SMR is about 6.54% of that in the conventional power plant (SKN 5&6).

Table 2. Calculated Thermal Neutron Flux in i-SMR and SKN 5&6.

	Calculated Value of Thermal Neutron Flux at Safety Channel (#/cm^2·sec)	Ratio (i-SMR / SKN 5&6)
SKN 5&6	2.03×10^9	6.54 %
i-SMR	1.33×10^8	

2.4 Quantitative Contribution Analysis of the Thermal Neutron Flux Formation Mechanism in Safety Channel

2.4.1 Computational Models for Contribution Analysis

To analyze the causes of the differences shown in Table 2 from a design perspective, a computational model was established to quantitatively analyze the contribution of each design material to the thermal neutron flux in the safety channels. Four computational models from Case A to Case D, considering whether resin and concrete wall are present, were established based on the Case A model as shown in Fig. 1 and are summarized in Table 3. For each model, the neutron sources were categorized into “Core-Originated” and “Concrete-Reflected” sources. The "Core-Originated" source means the neutron flux in the safety channel formed by neutrons directly leaking from the Containment Vessel. The "Concrete-Reflected" source means the neutron flux in the safety channel formed by neutrons scattered and reflected from the concrete wall.

Table 3. Computational Models for Analyzing Contributions to Thermal Neutron Flux Formation.

Origin of Source	Case	Materials		Calculated Thermal Neutron Flux
		Resin	Concrete Wall	
Core-Originated + Concrete-Reflected	Case A	O	O	Φ_a
	Case B	X	O	Φ_b
Core-Originated	Case C	O	X	Φ_c
	Case D	X	X	Φ_d

2.4.2 Decomposition of Thermal Neutron Flux Formation Mechanism

Neutrons leaking from vessel undergo moderation and absorption reactions in the resin surrounding the ex-core detector. In addition, reflected neutrons are produced after moderation in the Reactor Protection Building (Concrete Wall). To quantify the contributions of each design material (resin and concrete wall) to the thermal neutron flux in the safety channel, the contributions were defined by distinguishing between the “Core-Originated” forward neutron source and “Concrete-Reflected” albedo neutron source. Each contribution is defined to express the net change, considering both the gain and loss of thermal neutrons.

(1) Background Flux [Φ_d]

: Thermal neutron flux from the “Core-Originated” source with both resin and concrete wall removed (Case D).

(2) Resin net Contribution Flux [$\Phi_c - \Phi_d$]

: In Case C, the net contribution of resin to the thermal neutron flux in the safety channel from the “Core-Originated” neutron source. (with the concrete reflection effect excluded)

(3) Concrete net Contribution Flux [$\Phi_b - \Phi_d$]

: In Case B, the net contribution of concrete wall to the thermal neutron flux in the safety channel from the “Concrete-Reflected” neutron source. (with the resin effect excluded)

(4) Rear Resin net Contribution Flux

$$[\ (\Phi_a - \Phi_c) - (\Phi_b - \Phi_d) = \Phi_a - \Phi_b - \Phi_c + \Phi_d\ ]$$

: In Case A, the net contribution of the rear resin to the thermal neutron flux in the safety channel from the “Concrete-Reflected” neutron source.

A positive “Rear Resin net Contribution Flux” indicates that the interaction of “Concrete-Reflected” neutrons with rear resin is dominated by gain reactions due to neutron moderation. A negative “Rear Resin net Contribution Flux” indicates that the interaction of

"Concrete-Reflected" neutrons with rear resin is dominated by loss reactions due to neutron absorption.

In Case A, the thermal neutron flux (Φ_a) in the ex-core detector safety channel can be expressed as gain and loss terms resulting from primary neutron interactions with materials (resin and concrete wall). Using the terms defined above, the following Eq. (1) represents the formation mechanism of the thermal neutron flux in the safety channel.

$$\Phi_a = \sum \Phi_{gain} + \sum \Phi_{loss}$$

$$= \sum (\Phi_{moderation} + \Phi_{background\ flux}) + \sum \Phi_{absorption}$$

$$= \underbrace{\overbrace{\Phi_d}^{\text{Background flux}} + \overbrace{(\Phi_c - \Phi_d)}^{\text{Resin net Contribution Flux}}}_{\text{Core-Originated}} + \underbrace{\overbrace{(\Phi_b - \Phi_d)}^{\text{Concrete net Contribution Flux}} + \overbrace{(\Phi_a - \Phi_b - \Phi_c + \Phi_d)}^{\text{Rear Resin net Contribution Flux}}}_{\text{Concrete-Reflected}} \qquad (1)$$

2.4.3 Contribution Analysis Results for Thermal Neutron Flux Formation

Based on the methodology in Section 2.4.2, the material-wise contributions of thermal neutron formation were analyzed. The results were compared between the conventional power plant (SKN 5&6) and i-SMR, as summarized in Table 4.

Table 4. Comparison of Material Contributions to the Formation of Thermal Neutron Flux in the Safety Channel.

Origin of Source	Description	i-SMR		SKN 5&6	
		Thermal Neutron Flux (#/cm²·sec)	Contribution (%)	Thermal Neutron Flux (#/cm²·sec)	Contribution (%)
Core-Originated	Background flux (Φ_d)	7.83×10^4	0.06 %	1.61×10^6	0.08 %
	Resin net Contribution Flux ($\Phi_c - \Phi_d$)	9.73×10^7	73.38 %	1.21×10^9	59.60 %
	Total Core-Originated Contribution Flux (Φ_c)	9.74×10^7	73.43 %	1.21×10^9	59.68 %
Concrete-Reflected	Concrete net Contribution Flux ($\Phi_b - \Phi_d$)	7.49×10^7	56.48 %	1.22×10^9	60.07 %
	Rear Resin net Contribution Flux ($\Phi_a - \Phi_b - \Phi_c + \Phi_d$)	-3.97×10^7	-29.91 %	-4.00×10^8	-19.75 %
	Total Concrete-Reflected Contribution Flux ($\Phi_a - \Phi_c$)	3.52×10^7	26.57 %	8.17×10^8	40.32 %

The results show that the direct thermal neutron leakage (Φ_d , background flux) from the "Core-Originated" neutron source, without absorption in the resin, is less than 1 % in both i-SMR and the conventional power plant. Therefore, resin moderator should be installed to utilize the more abundant fast neutrons from the "Core-Originated" source rather than thermal neutrons to enhance the detector signal efficiently. According to the results in Table 4, it can be inferred that most of the thermal neutron flux in the ex-core detector safety channel is formed through moderation of fast neutrons from the "Core-Originated" neutron source. Since they are directly coming from the core, the resulting signal reflects core power reliably and is suitable for generating reactor protection signals. Thus, the dominant cause of thermal neutron flux degradation in the safety channel of i-SMR ex-core detector is the reduction of fast neutrons leaking from the CV.

In i-SMR, the "Total Concrete-Reflected Contribution Flux" (26.57 %) is lower than that of the conventional power plant (40.32 %). This is because, in conventional power plant, the concrete wall is installed close to the RV to serve as a Primary Shield, resulting in a stronger neutron reflection effect than in i-SMR.

The negative value of the "Rear Resin net contribution Flux" indicates that, for neutrons reflected from the concrete wall, absorption losses in the resin dominate over moderation gains within the resin.

Thus, resin plays a dual role. It moderates "Core-Originated" fast neutrons to form the majority of the ex-core detector signal. At the same time, it reduces the contribution of "Concrete-Reflected" thermal neutrons, which are expected to interfere with core power, by absorbing them. The thermal neutron contribution of each material in the safety channel of i-SMR is summarized in Fig. 2.

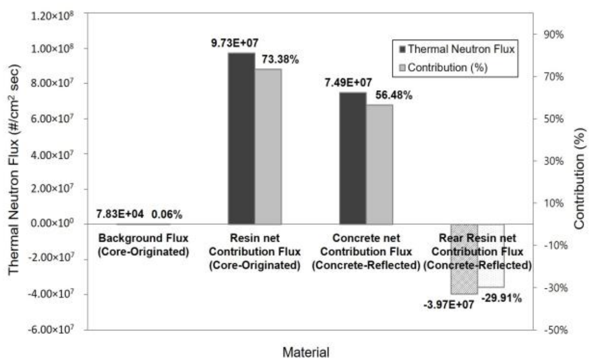

Fig. 2. Material-wise Thermal Neutron Flux in the Safety Channel of i-SMR and the Corresponding Quantitative Contributions.

2.5 Sensitivity Analysis of Design Parameters

2.5.1 Sensitivity Analysis of Coolant Volume Fraction in the Neutron Reflector

Since the neutron reflector is one of the major differences in the reactor internal design between the conventional power plant and i-SMR, its shielding effect was analyzed. In solid reflectors, the formation of coolant flow paths for heat removal is essential, and their extent of these paths can influence neutron leakage. Therefore, a homogenized model of the neutron reflector and its flow paths was established, and the coolant volume fraction was selected as the variable to conduct a sensitivity analysis of the thermal neutron flux in the ex-core detector safety channels, as shown in Fig. 3.

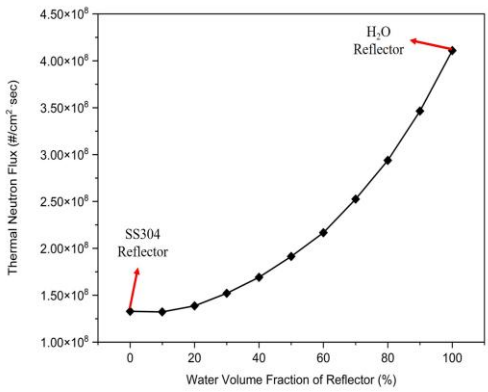

Fig. 3. Sensitivity Results for Thermal Neutron Flux to Coolant Volume Fraction in the Neutron Reflector.

The increase in coolant volume within the neutron reflector reduces neutron reflection. However, the leakage of fast neutrons increases. As a result, the thermal neutron flux in the safety channel differs by up to 3 times. As the leaked fast neutrons increase, moderation in the resin is enhanced, leading to a higher thermal neutron flux level at the detector.

2.5.2 Sensitivity Analysis of Ex-core Detector Radial Distance

Considering variables such as mechanical interferences, the actual installation location of the ex-core detector may be farther than that shown in Fig. 1. Because this may lead to additional reduction in the thermal neutron flux, a sensitivity analysis was conducted on the installation location of ex-core detector. Models with radial distances of 10 cm increments up to 100 cm from the safety channel location in Fig. 1 were established, and the thermal neutron flux was calculated for each position. Additional study shows that changes in the neutron flux due to the azimuthal angle were found to be negligible, and therefore only the effects of radial distance were analyzed in this study.

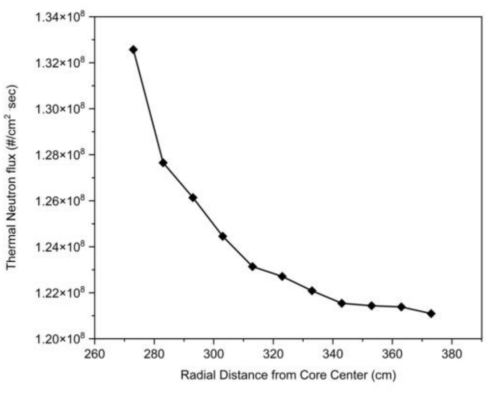

Fig. 4. Sensitivity Results for Safety Channel Thermal Neutron Flux to the Radial Distance from Core.

Fig. 4 shows the sensitivity of the thermal neutron flux in the safety channels to the radial distance from the core center. As the radial distance increased, the thermal neutron flux decreased; however, beyond a certain distance, the rate of decrease per 10 cm fell below 1 %, indicating a relatively small impact. This is because the ex-core detector is located sufficiently far from the neutron source (core), and the fast neutrons that primarily contribute to detector signal are negligibly attenuated in air. As a result, the reduction of thermal neutron flux with increasing radial distance is relatively small.

3. Conclusions

Based on calculations in this study, the primary cause of the degradation of thermal neutron flux in the ex-core detector safety channels of i-SMR was found to be the decreased leakage of fast neutrons originating from the core. The main causes of this phenomenon were identified as the addition of solid neutron reflectors and installation location of ex-core detector behind CV structure, and the reduction of neutron source strength per unit volume. Consequently, this may cause problems in signal formation for the ex-core detector.

The results of this study suggest that, to maintain sufficient signal intensity in the ex-core detectors of i-SMR, design modifications should be necessary rather than adopting the same detector design as in conventional power plants. However, design modifications must be carefully considered, taking into account not only neutron physics but also engineering factors such as material availability and manufacturability.

It should be noted that the results of this study are preliminary and intended only to identify characteristics of current design, as the design and installation location of the ex-core detector have not yet been finalized. Furthermore, to compensate for methodological limitations of the code calculations, bias corrections should be applied when using these values for design purposes. Therefore, rather than focusing on absolute values, it is desirable to use the results for comparison with conventional power plants and for identifying general trends.

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