A Cylindrical Shielding Design Concept for the Prototype Gen-IV Sodium-cooled Fast Reactor

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1. Introduction

A Sodium-cooled Fast Reactor (SFR) has the potential of radioactive waste reduction and resolving the spent fuel storage problems by utilizing fast spectrum neutrons. The Korea Atomic Energy Research Institute (KAERI) has been developing an SFR to aim at specific design approval of a Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR) up to 2020. In the PGSFR, a metal fueled, blanket-free, pool type SFR concept is adopted to acquire the inherent safety characteristics and high proliferation-resistance.

In the pool type fast reactor, the intermediate heat exchangers (IHXs), which transfer heat from the primary sodium pool to a secondary sodium loop, are placed inside of the reactor vessel. Hence, secondary sodium passing the IHXs can be radioactivated by a $^{23}Na(n,\gamma)^{24}Na$ reaction, and radioactivated secondary sodium causes a significant dose in the Steam Generator Building (SGB) [2, 3]. Therefore, a typical core of a pool type fast reactor is usually surrounded by a massive quantity of shields [2]. In addition, the blanket composed of depleted uranium plays a role as superior shielding material; a significant increase in shields is required in the blanket-free pool type SFR.

In this paper, a new cylindrical shielding design concept is proposed for a blanket-free pool type SFR. In a conventional shielding design, massive axial shields are required to prevent irradiation of secondary sodium passing IHXs and they should be replaced according to the subassembly replacement in spite of negligible depletion of the shielding material. The proposed shielding design concept minimizes the quantity of shields without their replacement.

To prevent an angle quadrature and number of group problems in the fast reactor shielding analysis reported in reference [4], the MCNP code with a continuous energy library is used for the PGSFR shielding analysis.

2. PGSFR Shielding Analysis Model

The basic design data of the candidate core of thr PGSFR is shown in Table I. The core part of the MCNP model is established based on reference [5] with conservative approximation by adopting a single enrichment core. For the fuel subassembly, 19.53 w/o U-10-Zr fuel is loaded while 15 w/o U-10-Zr fuel is assumed as spent fuel in In Vessel Storage (IVS). For the cladding, lower shield, and radial reflector materials, HT9m is used [5] while B_4C is used for the shields. The in-vessel components such as IHX and DHX in the MCNP model are established based on reference [6]. The detailed configurations are shown in Figs. 1 through 3.

Table I: Basic Design Data for PGSFR Candidate Core

	Number	Number of pins
Fuel Subassembly	112	217
IVS	50	217
Radial reflector	78	61
Replaceable Radial shield	114	7
Control Rod	9	61

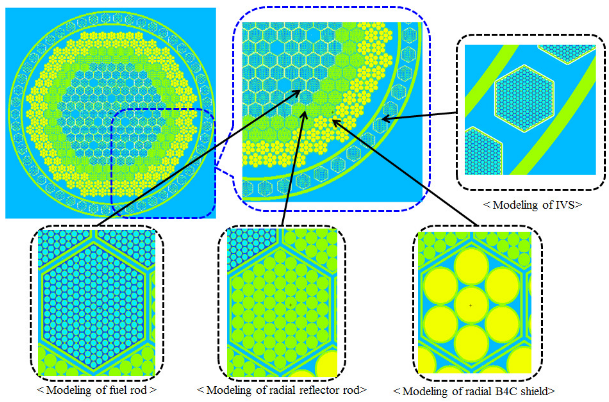

Fig. 1. Radial view of MCNP model for PGSFR Candidate Core

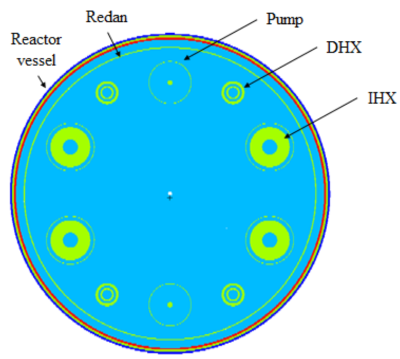

Fig. 2. Radial view of MCNP model for PGSFR in-vessel components at AA' plane

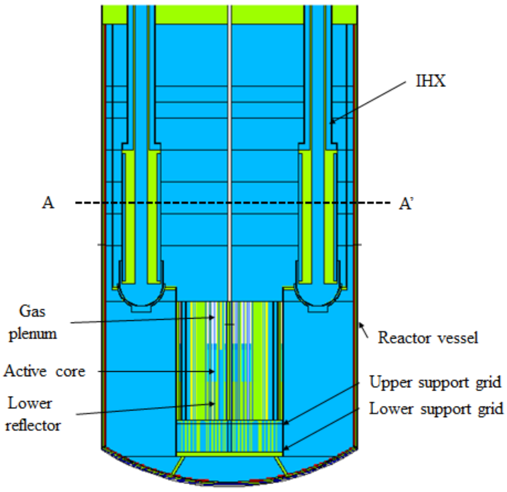

Fig. 3. Axial view of MCNP model for PGSFR in-vessel components

2. Calculation Method for PGSFR Shielding Analysis

2.1 Shielding Analysis Strategy

The established model is analyzed by the MCNP6.1 [7] code with continuous energy ENDF/B-VII.1 libraries for the following temperatures: 663.15 K, 740.65 K, 818.15 K, and 890.65 K. Since the shielding calculation is a kind of deep penetrating problem, the weight window method based on the WWG card is used to reduce the computation time. To prevent fission source bias due to a weight window method, a strategy shown in Fig. 4 is utilized for the shielding analysis. By employing the strategy in Fig. 4, one inactive generation calculation is sufficient for a parametric study because different configurations of the shields do not influence the fission source distribution.

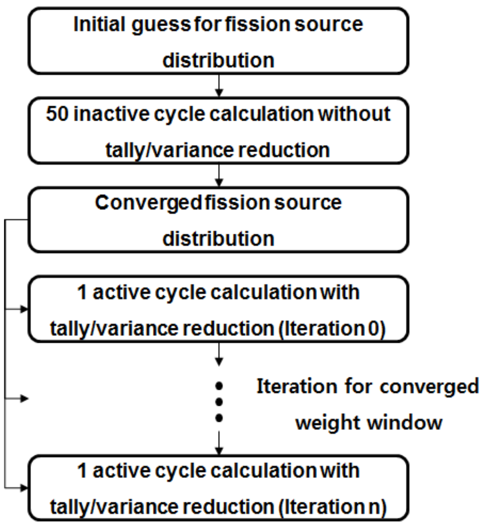

Fig. 4. A strategy for PGSFR shielding analysis

2.2 Conversion Factors for Shielding Analysis

Since the purpose of the MCNP code is the analysis of multiple particle transport, at least neutron, gamma, and beta particle transport calculations should be performed by the MCNP code to obtain the total q-values in the core. However this multi-particle calculation requires a tremendous calculation time, and the following power conversion factor is considered;

$$F = P_{total} \times \frac{P_{fission}}{P_{total}} \times \frac{1}{\sum_i \frac{\sigma_f^i \phi}{\sum_j \sigma_f^j \phi} Q_{fiss}^i} \times \frac{1}{1.60219 \times 10^{-13}} \times \nu, \quad (1)$$

where $P_{fission}$ is the fission power calculated by the deterministic code with neutron and gamma transport, P_{total} is the total thermal power of reactor, $\sigma_f^i \phi$ is total fission reaction of isotope i in reactor, Q_{fiss}^i is fission q-value isotope i, and ν is number of neutrons born at each fission.

Although the established model is based on a uranium fueled core, the PGSFR will be operated with TRU fuel after its validation. Hence, the following fuel type effect is considered;

$$F' = \frac{(\text{core average flux at TRU equilibrium core})}{(\text{core average flux at U equilibrium core})}. \quad (2)$$

In addition, the established model is for a Begin of Cycle (BOC) core, and the following cycle effect is also considered:

$$f = \frac{(\text{maximum core average flux during operation})}{(\text{core average flux at BOC of uranium core})} = \frac{(\text{core average flux at EOC of 6th cycle})}{(\text{core average flux at BOC of uranium core})}. \quad (3)$$

Finally, the activity of secondary sodium passing IHXs is calculated with simple algebra as

$$\lambda_{24} N^{24} = \frac{\sigma_c^{23} \phi N^{23} \left(1 - e^{-\lambda_{24} \Delta t_1}\right)}{1 - e^{-\lambda_{24}(\Delta t_2 + \Delta t_1)}} \times f \times F' \times F, \quad (4)$$

where λ_{24} is the decay constant of ^{24}Na, $\sigma_c^{23} \phi$ is the capture reaction rate of ^{23}Na in the secondary sodium passing IHXs, Δt_1 is the irradiation time of the secondary sodium, and Δt_2 is the cooling time of the secondary sodium.

2.3 Design Limit of Secondary Sodium Activation

The design limit of secondary sodium activation is obtained from reference [8]. In reference [8], 0.69 $\mu Ci / Kg$ was reported for a 0.25 *mRem/hr* worker dose limit in an SGB.

3. Analysis Results

3.1 Original PGSFR without shields

Sodium activation distributions and secondary sodium activation in IHXs for various IHX positions are shown in Fig. 5 and Table II, respectively.

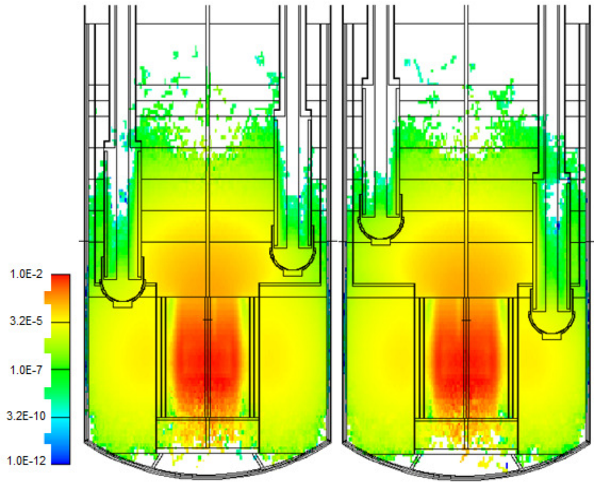

Fig. 5. Axial view of sodium activation distribution in original model for various IHX axial positions

Table II: Secondary sodium activations in IHXs and DHX for various IHX axial positions

IHX position	Activation [μCi / Kg]	Associated dose level in SGB [$mRem/hr$]
Active core top + 0.0 m	4.06E+04	1.47E+04
Active core top + 1.0 m	2.48E+04	8.99E+03
Active core top + 2.0 m	1.49E+04	5.40E+03
Active core top + 3.0 m	6.21E+03	2.25E+03
DHX	8.14E+02	2.95E+02

As mentioned in section 1, a significant dose rate is achieved without shields for every axial position of the IHXs in the reactor.

3.2 The First Sketch of Cylindrical Shielding Concept

Fig. 6 shows the first sketch of the cylindrical shielding concept for various axial positions of IHXs. Since the most significant irradiation was observed at the bottom part of the IHX, as shown in Fig. 5, a thick radial shield is placed below the IHX, while a relatively thin upper shield is placed inside the IHX.

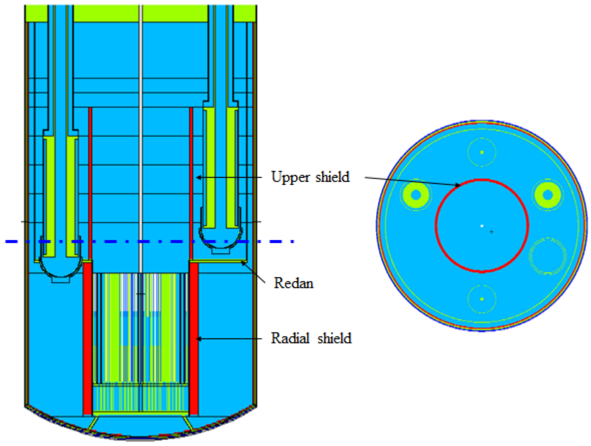

Fig. 6. The first sketch of cylindrical shielding concept

Sodium activation distributions of the first sketch of the cylindrical shielding concept are shown in Figs. 7 and 8. The radial view is the blue dotted line plane in Fig 6. As shown in Figs. 7 and 8, the cylindrical shield concept showed a good shielding effect for neutron streaming originated from the top part of the core. However, as shown in Table III, thicker shields are required for the first sketch of the cylindrical shielding concept.

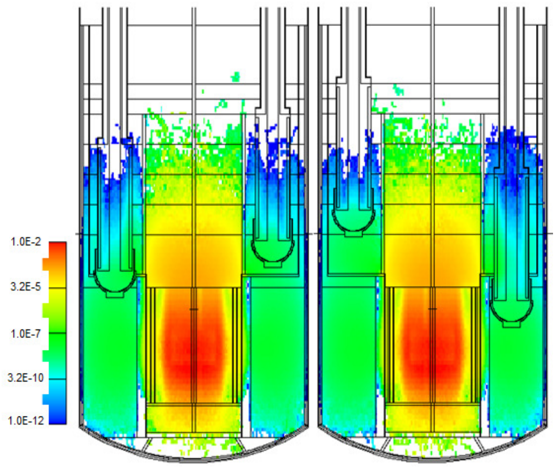

Fig. 7. Axial view of sodium activation distribution for the first sketch of cylindrical shielding concept

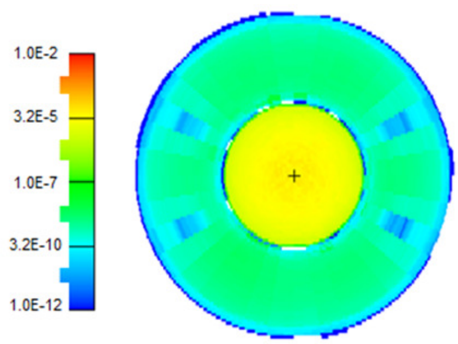

Fig. 8. Radial view of sodium activation distribution for the first sketch of cylindrical shielding concept

Table III: Secondary sodium activations in IHXs and DHX for various IHX axial positions

IHX position	Activation [μCi / Kg]	Associated dose level in SGB [*mRem/hr*]
Active core top + 0.0 m	2.68E+01	9.71E+00
Active core top + 1.0 m	2.58E+01	9.35E+00
Active core top + 2.0 m	1.76E+01	6.38E+00
Active core top + 3.0 m	5.76E+00	2.09E+00

3.3 The Final Design of Cylindrical Shielding Concept

The final design of the cylindrical shielding concept is shown in Fig. 9. For the final design of the cylindrical shielding design concept, the following are considered.

a) The second upper shield is added.
b) To preserve a stable primary sodium flow to the IHX, the height of the upper shield is limited up to the IHX inlet. Hence, the axial position of the IHX is chosen to be 2.2 m above the active core top.

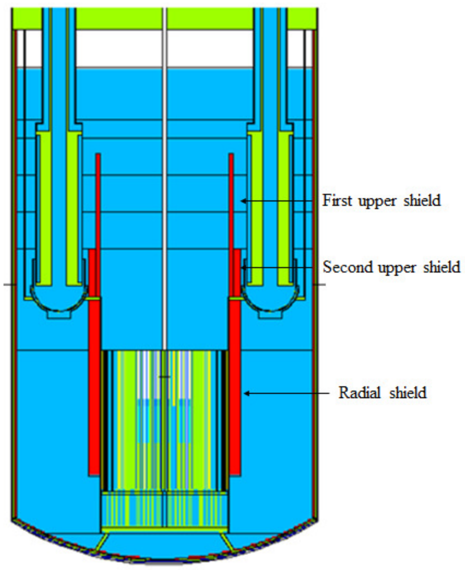

Fig. 9. The final design of cylindrical shielding concept

Sodium activation distributions and secondary sodium activation in the IHXs and DHXs for the final design of thr cylindrical design concept are shown in Fig. 10 and Table IV, respectively.

Table IV: Secondary sodium activations in IHXs and DHX for the final design of cylindrical shielding concept

	Activation [μCi / Kg]	Associated dose level in SGB [*mRem/hr*]
IHX	0.69E+00	2.50E-01
DHX	0.13E+00	4.71E-02

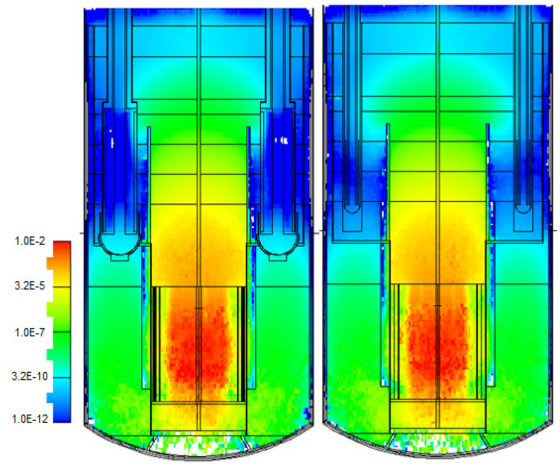

Fig. 10. Axial view of sodium activation distribution for the final design of cylindrical shielding concept

The final cylindrical shielding design concept satisfied the secondary sodium activation limit in both IHXs and DHXs.

3. Conclusions

In this paper, a new cylindrical shielding design concept is proposed for a blanket-free pool type SFR such as a PGSFR. The proposed design concept satisfied the dose limit in the steam generator building successfully without introducing a large quantity of B_4C shielding inside the subassembly.

Although the proposed cylindrical shielding design minimizes the in-vessel shields and omits a B_4C shield within the fuel subassembly, the reactor vessel height will be increased by 1.2 m (~8.5 %) due to an increase of the IHX position. Hence, alternatively, another cylindrical shielding concept, which maintains the reactor vessel height, by employing an upper reflector within the fuel subassembly and adopting a cap type shield below the IHXs, is also considered as a candidate PGSFR shielding design option [9].

ACKNOWLEDGMENTS

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