Preliminary ShutDown Dose Rate Evaluation for the Marine-Based Molten Salt Reactor

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1. Introduction

The Molten Salt Reactor (MSR), classified as a Generation IV nuclear reactor by the Generation IV International Forum (GIF), utilizes liquid nuclear fuel, offering several advantages, including enhanced accident tolerance, the ability to operate at low pressure, and improved fuel management. These characteristics suggest that MSRs are promising candidates for next-generation nuclear energy systems [1].

Furthermore, the liquid nature of the nuclear fuel in MSRs eliminates many of the constraints associated with conventional solid-fuel reactors. Consequently, a variety of MSR designs are currently under development, varying in application, geometry, and power output to accommodate diverse operational needs. In this study, a preliminary SDDR (ShutDown Dose Rate) analysis is performed for a 100 MWth MSR being developed for ship propulsion applications. Through this analysis, the fundamental shielding requirements necessary for its deployment are derived.

2. Models and Approximations

The calculation model is established as outlined in Reference [2] and is illustrated in Fig. 1. The core is characterized by a cylindrical structure with a radius of 91 cm, which is capped by hemispherical sections of the same radius at the top and bottom. An 8.8 cm-thick structural material surrounds the core, followed by a 50 cm-thick BeO reflector. Beyond the reflector, a 2.5-centimeter-thick reactor vessel encloses the entire system. The core is filled with a NaCl-KCl-UCl_3 molten salt fuel, utilizing HALEU (High-Assay Low-Enriched Uranium) [3].

Outside the reactor vessel, approximately 275 cm from the reactor center, there is a 25 cm-thick layer of SPROULE WR-1200 insulator [4]. Beyond this layer, a 25 cm-thick shield is enclosed by a 2.5 cm-thick structure.

The calculations were performed using the Direct One Step (D1S) method, which is based on the MCNP 6.1 code [5]. For the ENDF/B-VII.1 library, a temperature of 900 K was approximated, while a separately prepared delayed gamma library was utilized. The results from the MCNP simulations were normalized according to Eq. (1), which is based on a 100 MWth power. In this process, the reaction rate terms in Eq. (1) were assumed based on the values used in Reference [6].

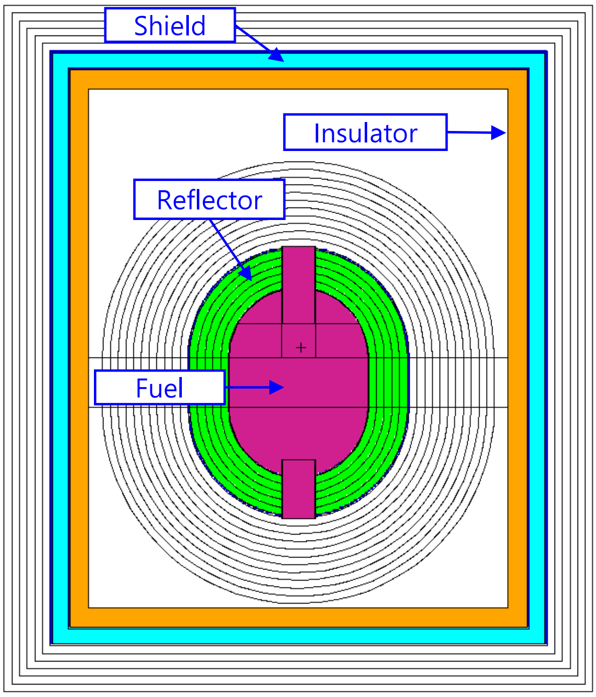

Fig. 1 Configuration of calculation model

$$F = P_{total} \times \frac{P_{fission}}{P_{total}} \times \frac{1}{1.60219 \times 10^{-13}} \times \nu \times \frac{1}{\left(\frac{\sigma_f^{u^{235}}\phi}{\sigma_f^{u^{235}}\phi + \sigma_f^{u^{238}}\phi}\right) Q_{fiss}^{u^{235}} + \left(\frac{\sigma_f^{u^{235}}\phi}{\sigma_f^{u^{235}}\phi + \sigma_f^{u^{238}}\phi}\right) Q_{fiss}^{u^{238}}} \quad (1)$$

The SDDR calculation assumed that the reactor operated for 30 years at 90% of its full power capacity, followed by a 12-hour cooling period.

3. Effect of Shield Material

Two shielding materials were considered in this study: case 1) a composite material consisting of 50% water and 50% structural material, and case 2) a material composed entirely of 100% water.

As illustrated in Figs. 3 and 4 the delayed gamma dose rate distribution is examined for the two considered cases, while Figs. 4 and 5 depict the delayed gamma source distribution for each case.

In Case 1, where structural material is mixed within the shield, the activation of the shield itself is relatively higher. In contrast, Case 2, where the shield is composed entirely of water, shows negligible activation within the shield.

However, despite the increased activation in Case 1, the structural material within the shield effectively attenuates gamma radiation originating both from the reactor vessel and other gamma sources in the core. Conversely, in Case 2, due to the absence of structural material within the shield, the gamma radiation from the reactor vessel and other gamma sources in the core is not adequately attenuated, resulting in a gamma dose rate that is approximately 15 times higher than in Case 1.

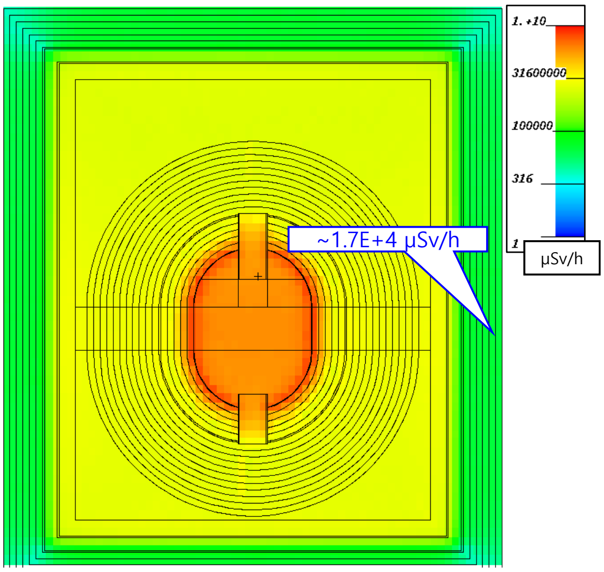

Fig. 2 Delayed gamma dose rate distribution for case 1

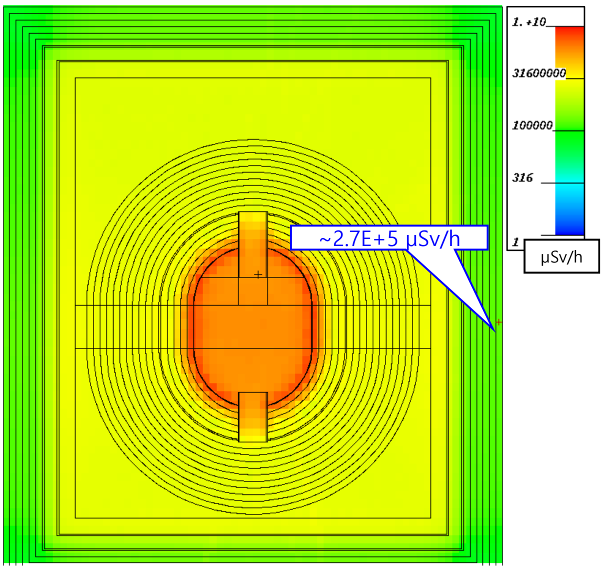

Fig. 3 Delayed gamma dose rate distribution for case 2

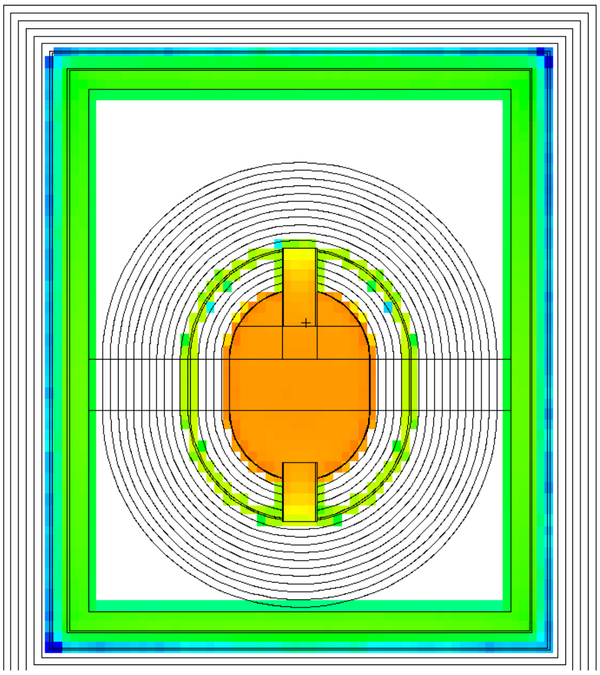
Fig. 4 Relative delayed gamma source distribution for case 1

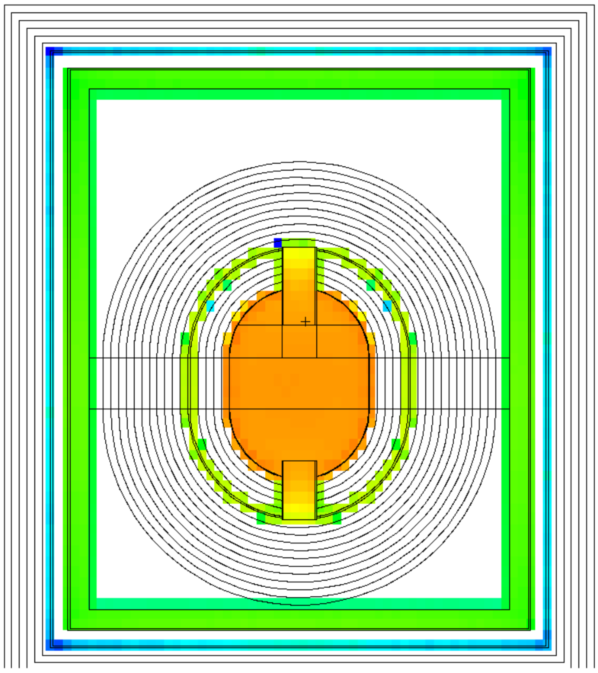
Fig. 5 Relative delayed gamma source distribution for case 2

4. Effect of RAFM Steel as an Outside Shield

In the cases investigated previously, the SDDR was found to be considerably high. Therefore, the potential impact of installing an additional shielding layer for delayed gamma radiation outside the existing shielding structure in the current model was evaluated. In this case, the additional shielding must be designed to minimize the activation effects caused by neutron leakage penetrating the insulator and shield during reactor operation. To address this, ARAA (Advanced

Reduced Activation Alloy), which developed for a structural material of fusion reactor to reduce the activation by removing long-lived radionuclide, was considered as the delayed gamma shielding material [7]. As shown in Fig. 6, the additional shield was placed 10 cm away from the outer surface of the existing shield.

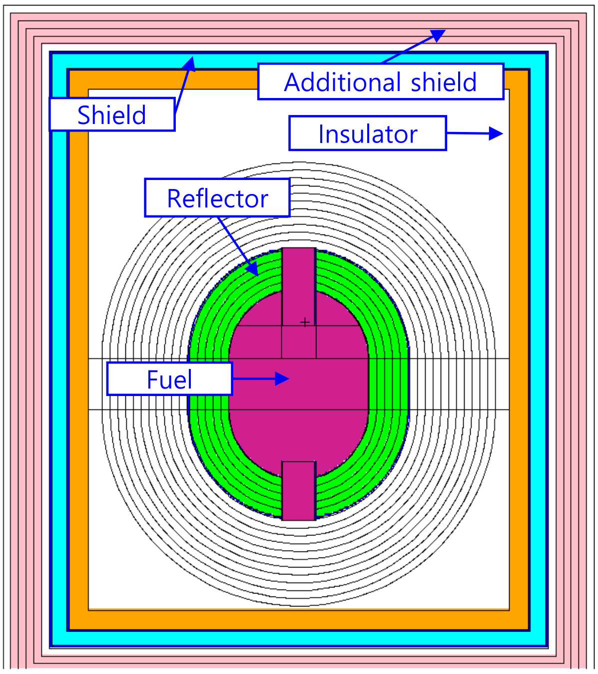

Fig. 6 Configuration of calculation model with additional shield

Figs. 7 and 8 illustrate the delayed gamma dose rate distribution when applying an additional shield with a thickness of 20 cm and 40 cm, respectively. Meanwhile, Figs. 9 and 10 depict the distribution of delayed gamma sources for each case.

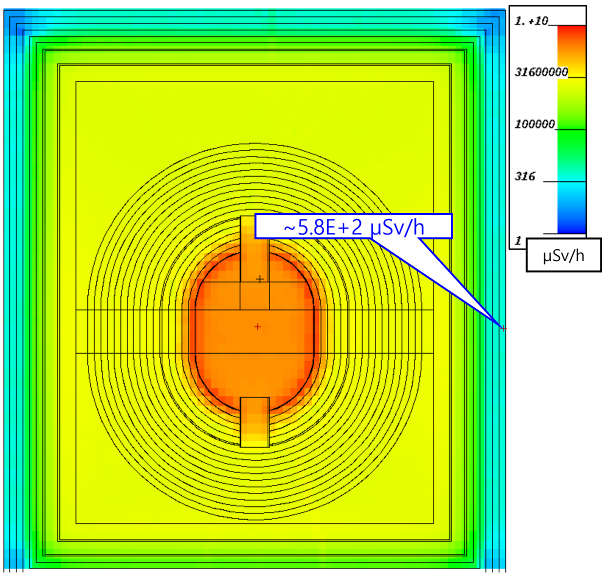

Fig. 7 Delayed gamma dose rate distribution for 20 cm thickness additional shield case

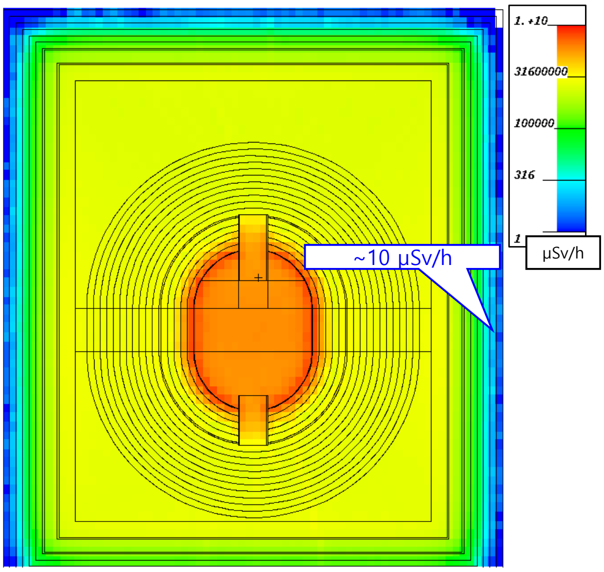

Fig. 8 Delayed gamma dose rate distribution for 20 cm thickness additional shield case

With the application of a 20 cm-thick ARAA additional shield, the dose rate was reduced by approximately 460 times. When a 40 cm-thick ARAA additional shield was applied, the dose rate was further reduced, reaching 10 μSv/h, which corresponds to the occupational exposure limit recommended by ICRP-60 [8]. Furthermore, consistent with the relatively low activation characteristics of ARAA, the activation of the additional shield was evaluated to be relatively lower compared to the structural material used in the existing shield.

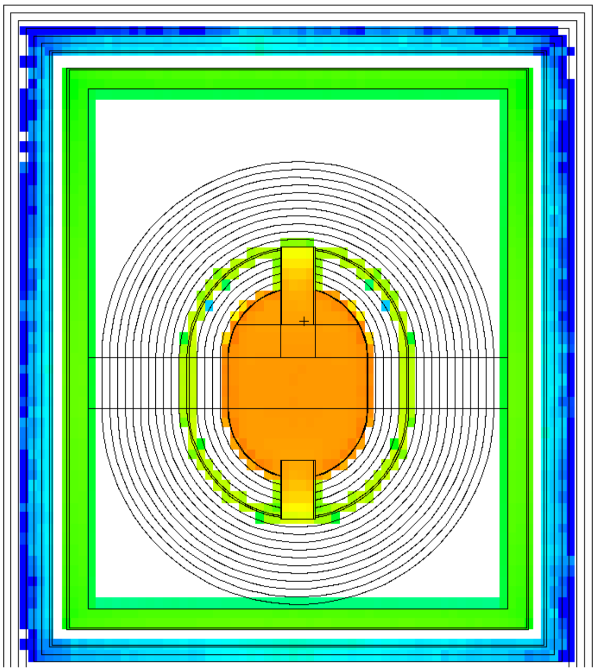
Fig. 9 Relative delayed gamma source distribution for 20 cm thickness additional shield case

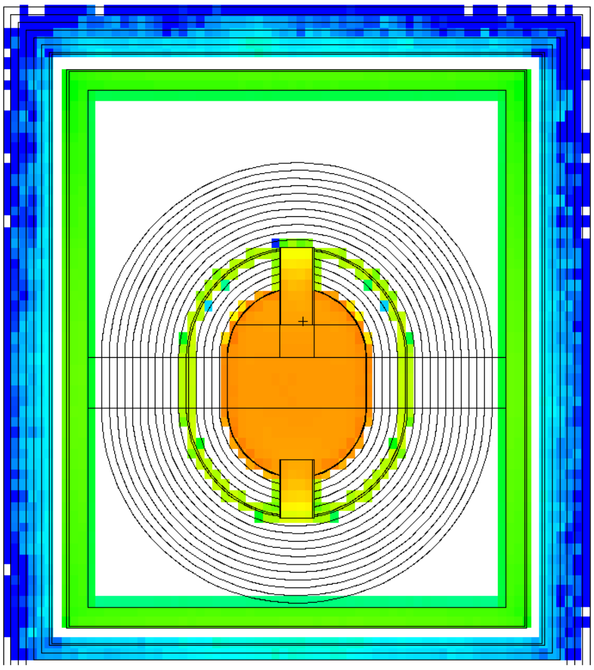

Fig. 10 Relative delayed gamma source distribution for 40 cm thickness additional shield case

4. Conclusions and Future Study

In this study, preliminary SDDR for a 100 MWth marine-based MSR was evaluated using the D1S method. The following key conclusions were drawn:

1) A shielding material composed of 50% water and 50% structural material was found to be more effective than a shield consisting of 100% water.
2) By incorporating an additional shield with a thickness of approximately 40 cm into the existing design, the delayed gamma dose rate at the outer boundary was confirmed to meet the occupational exposure limit recommended by ICRP-60.
3) When using ARAA as the additional shielding material for SDDR mitigation, it was observed that concerns for activation of additional shield were considerably lower compared to conventional structural materials.

However, incorporating a 40 cm-thick additional shield still presents a design challenge. Therefore, further study is required to optimize the shielding design by reducing the thickness of the additional shield. For instance, an alternative approach could involve integrating neutron shield within the insulator, which could mitigate the activation of both the insulator and shield, finally leading to a reduction in the SDDR.

ACKNOWLEDEMENT

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