

Advanced radiation shielding methodology for small modular reactors using a multithreaded two-step source approach

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ABSTRACT

The increasing demand for compact and efficient radiation shielding in advanced nuclear systems, such as small modular reactors in nuclear-powered ships, necessitates the development of improved radiation shielding techniques. This study presents an advanced methodology using a multithreaded surface source approach for radiation shielding analysis and applies it to the URANUS lead-bismuth cooled fast reactor. Unlike traditional surface source write/read (SSW/SSR) methods, which do not support multithreading and often suffer from prolonged computation times to reduce relative error, the proposed method ensures consistent statistical accuracy and computational efficiency. By using MCNP6's advanced capabilities, this approach overcomes prior limitations and enables more reliable and effective shielding analysis. Comparative analyses with conventional shielding methods, energy and directional sensitivity studies, and preliminary optimization results demonstrate the effectiveness of the newly developed methodology in enhancing radiation shielding performance for space-constrained nuclear systems.

1. Introduction

The global maritime industry is undergoing a paradigm shift toward sustainability, driven by increasingly stringent environmental regulations. The International Maritime Organization (IMO) has set ambitious targets to cut carbon emissions by 70% and total greenhouse gas emissions by 50% by 2050, relative to 2008 levels (IMO adopts historical agreement on, 2023). To meet these requirements, various alternative ship propulsion technologies are being explored, among which nuclear propulsion has reemerged as a viable solution due to its ability to provide high energy output, long operational lifetimes, and zero direct carbon emissions (Kim et al., 2025).

While nuclear propulsion is not a new concept—previous implementations such as the NS *Savannah* and Soviet nuclear icebreakers have demonstrated its feasibility—modern applications demand more compact, efficient, and safer reactor designs that align with the modular and scalable nature of small modular reactors (SMRs) (Bagheri and Khalafi, 2023). One of the major challenges in integrating nuclear propulsion into commercial and military vessels is radiation shielding. Unlike conventional land-based reactors that can accommodate large and heavy shielding, marine reactors require highly optimized shielding that minimizes both weight and volume while ensuring adequate

protection for personnel and sensitive equipment (Junchong Yu, 2020). This highlights the importance of developing a conservative yet efficient source definition in radiation shielding calculations to achieve optimal shielding design. To address these challenges, advanced computational tools such as Monte Carlo (MC) simulations and optimization algorithms have been employed to design lightweight and efficient radiation shielding systems.

The potentially hazardous radiation sources produced in a nuclear reactor include high-energy neutrons, gamma rays, and fission fragments that are highly radioactive, emitting alpha, beta, and gamma radiation (WOOD, 1982). While alpha particles and beta rays have limited penetration and can be effectively blocked by thin materials, fast neutrons and high-energy gamma rays require much thicker shielding to ensure adequate radiation protection (Chen et al., 2018). The harmful biological effects of such radiation are well documented, underscoring the need for proper shielding. To determine the appropriate shield thickness, the radiation dose behind the shielding must be maintained below 10 μ Sv/hr at full power, considering an exposure duration of 40 working hours per week (Bagheri and Khalafi, 2023).

To solve the reactor shielding problem, numerical solutions of the Boltzmann transport equation have historically followed three tracks: point-kernel method, discrete ordinates method, and Monte Carlo (MC)

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method. The point-kernel method treats sources as differential isotropic point sources and evaluates attenuation along straight paths while accounting for multiply scattered particles through an empirical buildup factor. While this technique is fast and convenient for preliminary gamma ray and fast neutron shielding calculations (Bagheri and Khalafi, 2023; Chen et al., 2018), significant errors can arise when dealing with complex geometries or complicated source descriptions when compared with the MC method (Matijević et al., 2025). A recent application example is provided by Houtkoop (2022). In this work, the shielding is idealized as a spherical inner neutron shield (water or concrete) plus an outer lead for gamma shielding. Prompt gamma emission is grouped into energy intervals; for each group, the flux at radius R is computed with a Taylor-form buildup factor, then converted to exposure using group-dependent mass absorption coefficients for air. Lead thickness is increased iteratively using an average halving distance until the combined neutron and gamma dose rates falls below a prescribed design limit. This workflow is documented together with the lookup tables for attenuation, buildup, and mass-absorption coefficients, and is used to plot shield weights against historical and conceptual marine reactors to sanity-check the analytical estimates (Houtkoop, 2022).

The discrete ordinates methods solve the transport equation deterministically by discretizing energy and angle and expanding scattering in Legendre polynomials. Widely used implementations offer high efficiency. But such methods introduce errors induced by multi-group approximation and can struggle with complex geometries or strong streaming features (Wang et al., 2021). A representative application is the MRX compact marine reactor study (YAMA JI and SAKO, 1994), where radial and axial shields were optimized using 1-D cylindrical ANISN calculations; DOT3.5 and QAD were also employed. The authors reported specific primary and secondary gamma dose ratios at the pressure-vessel and containment outer surfaces and designed to conservative criteria after benchmarking ANISN against OTTO-HAHN shield-tank experiments using a factor-of-three margin on calculated dose rate at key locations (YAMA JI and SAKO, 1994).

Finally, MC Method directly simulates particle histories with continuous-energy cross sections with few approximations, offering the ability to handle arbitrary source and geometry descriptions more naturally than point-kernel or discrete ordinate methods. But in this case the tradeoff is computational cost, especially for deep-penetration problem (Chen et al., 2018).

In typical radiation shielding analysis using the MC method, the Monte Carlo N-Particle (MCNP) transport code is used, which employs KCODE criticality simulations. KCODE not only calculates the effective multiplication factor but also provides essential data such as neutron flux, gamma flux, dose rates, reaction rates, the average number of neutrons emitted per fission, and particle energy spectra, among other key parameters (Kulesza et al., 2022). Neutron and gamma fluxes and their corresponding dose rates are obtained by tracking individual particle histories and simulating their interactions to represent the reactor's average behavior. To improve the accuracy of the analysis, a three-dimensional heterogeneous reactor model can be employed that enables more detailed assessments under various operational conditions. Although KCODE criticality calculations are crucial for primary radiation shielding design, excessively long computational times are required to achieve low relative errors. As a consequence, using KCODE alone for radiation distribution assessments is often impractical. To efficiently analyze extensive radiation distributions, variance reduction techniques (VRTs) are indispensable. Traditional approaches, such as surface source write/read (SSW/SSR), involve recording radiation sources's energy, direction and weight at a designated vessel outer surface and using them as secondary sources instead of KCODE criticality sources for deep penetration problems, particularly when dealing with thick biological shielding (Alizadeh et al., 2013). This methodology has been practically applied by Alizadeh et al. in the shielding analysis of a VVER-1000 reactor, where surface source modeling and iterative VRTs were employed to obtain accurate dose rates through multi-layer

concrete shields (Alizadeh et al., 2013). However, SSW/SSR has inherent limitations, including the inability to leverage multithreading during source recording and the potential risk of insufficient particle sampling. Moreover, users must ensure that a sufficient number of tracks are recorded on the surface source to enhance computational efficiency (Kulesza et al., 2022). Similarly, the weight window generation (WWG) technique is also computationally demanding, requiring multiple iterations to achieve an optimized variance reduction parameter set (Sweezy et al., 2005).

To overcome these challenges and improve computational performance for radiation shielding analysis using MCNP6, this study proposes a methodology applying a newly developed VRT that supports multithreading and is compatible with ADVANTG, a deterministic code used to generate weight windows (Van Riper and Metzger, 2017). By integrating these aspects, the proposed approach aims to significantly reduce computational time while enhancing the accuracy and practicality of deep penetration shielding calculations for optimized shielding design, which is required for marine applications in particular.

The reactor model considered in this study is URANUS (Yoo et al., 2014). Developed by Seoul National University, URANUS is a lead-bismuth eutectic (LBE)-cooled fast reactor designed for a thermal output of 100 MWth and a 20-year core life without refueling. Lead-cooled fast reactors offer several advantages for marine propulsion, including high thermal conductivity and excellent heat capacity. Lead also has a high boiling point, reducing the risk of coolant boiling under operational conditions, and is chemically inert, thereby avoiding the risk of sodium–water reactions seen in sodium-cooled fast reactors (OECD/NEA, 2015). In addition, the URANUS design employs fully passive cooling via natural circulation, eliminating the need for active pumping systems and enhancing safety through simplified system architecture. This passive cooling system also ensures heat removal in accident scenarios, which improves overall reliability (Yoo et al., 2014).

The rest of this paper is organized as follows. Section 2 provides the background on MCNP, VRTs, and the URANUS reactor design. Section 3 introduces the proposed methodology using a high efficiency, multi-threaded cylindrical surface source for three-dimensional heterogeneous shielding analysis. This section also discusses the differences between the conventional SSW/SSR-based VRT and the newly developed VRT, and then for optimization of radiation shielding design, a genetic algorithm is employed to minimize both shield volume and weight while maintaining dose rates below the regulatory target. Next, Section 4 presents the sensitivity analysis and optimization results of the newly developed VRT including the source term across various energy and directional bins, assessed in terms of both radiation dose rate and computational efficiency. Comparison results are also given between the shielding performance of the proposed design and the initial configuration with respect to biological shield volume, weight, and dose rate reduction. Section 5 summarizes and concludes the study.

2. Monte Carlo method and reactor geometry description

This section first introduces the MCNP code used for radiation shielding analysis. Then the VRT applied to reduce the computational time required to achieve results with sufficient statistical precision is discussed, followed by a description of the URANUS reactor geometry.

2.1. MCNP code

The MCNP code is a widely used tool for simulating neutron and gamma ray transport, performing criticality calculations, and analyzing radiation shielding problems. It allows users to perform various tallies related to particle current, particle flux, and energy deposition. These tallies are normalized per source particle, providing standardized output for comparative analysis. The relative error of the tally results is obtained by dividing the standard deviation ($\sigma_{\bar{x}}$) by the average tally value

($\bar{X}$), which serves as a measure of the simulation's statistical precision (Kulesza et al., 2022), as in Eq. (1):

$$\text{Relative error} = \frac{\sigma_{\bar{X}}}{\bar{X}}. \quad (1)$$

Radiation sources in MCNP can be defined using one of three main methods: a general fixed source, a surface source utilizing the SSW/SSR feature, or a criticality source defined by KCODE. In the KCODE calculation, MCNP first allows the fission neutron distribution to converge during the inactive cycles. Then during the active cycles, the code simulates N fission neutrons per cycle and calculates the effective multiplication factor (k_{eff}) in each generation. This process is repeated over several cycles to ensure an accurate estimate of k_{eff} (Goorley, 1994).

2.2. Variance reduction technique (VRT)

For deep penetration problems in MCNP, VRTs are essential when calculating the dose rate at a tally located behind thick shielding, due to the large attenuation in particle number that would otherwise lead to long computation times and high relative errors (Cramer, 1984). The efficiency of MCNP simulations is commonly evaluated using a figure of merit (FOM) defined as the inverse of the product of the squared relative error (R) and the total computation time (T) in minutes (Kulesza et al., 2022) as in Eq. (2),

$$\text{FOM} = \frac{1}{R^2 \cdot T}. \quad (2)$$

where a higher FOM indicates better efficiency (Martínez-Fernandez, 2024).

Among various VRTs, weight windows play a central role by controlling the statistical weight of particles, thus ensuring adequate sampling in regions of interest while avoiding wasted computational effort in unimportant areas. One widely adopted VRT in MCNP is the mesh-based WWG technique. This method utilizes a user-defined mesh to assign importance values or weight ranges across the problem geometry. During particle transport, MCNP adjusts the weight of each particle according to the mesh region it enters: particles with weights outside the assigned window are either split or subjected to Russian roulette. This adaptive control of particle weights significantly improves the tally statistics, particularly in shielding problems. However, generating an optimal weight window often requires repetitive iteration, which can lead to excessive computation. The process typically involves an initial forward simulation to collect flux data, followed by the generation of a mesh-based weight window file. The updated weight window is then applied in a subsequent simulation for improved accuracy and reduced relative error (Culbertson and Hendricks, 2000).

To address the challenges associated with the repetitive iteration required for WWG, a hybrid strategy has been proposed that takes advantage of the strengths of both MC and deterministic methods. For example, ADVANTG, which works in conjunction with MCNP, deterministically computes source biasing and transport biasing parameters, particularly weight window and biased source distributions, especially in shielding analyses involving deep penetration problems (Van Riper and Metzger, 2017).

2.3. URANUS geometry

The URANUS reactor is a 100 MWth LBE-cooled fast-spectrum SMR capable of operating for 20 years without refueling. This long operational lifespan supports sustained energy supply with minimal maintenance. The reactor employs a passive cooling system based on natural circulation, eliminating the need for active coolant pumps and thereby enhancing system simplicity and reliability. The main design parameters of URANUS are summarized in Table 1. As seen in Fig. 1, depicting its hexagonal fuel assembly configuration and heterogeneous core structure, the core of URANUS is designed to flatten the power distribution by

Table 1

Description of the major design parameters of URANUS (Yoo et al., 2014).

Parameter	Specification
Thermal power	100 MWth
Fuel	UO ₂
Enrichment (Inner core/Outer core)	9.35 wt%/17.75 wt%
Cladding	T91
Absorber	B ₄ C
Reflector	LBE
Primary coolant	LBE
Gap	He
Core lifetime	20 years
Assembly geometry	Hexagonal
Reactivity swing	109.8 pcm
Secondary coolant	Water
Core inlet temperature	304.95 °C
Core outlet temperature	457.4 °C
Primary cooling method	Natural circulation
Primary heat transfer system	Compact pool type

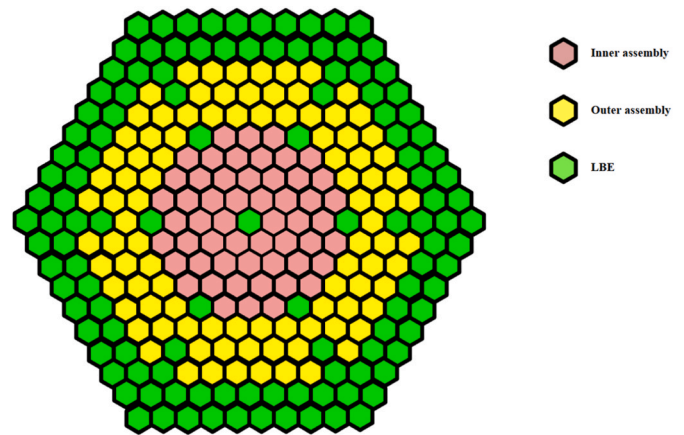


Fig. 1. Radial layout of the URANUS active core (Yoo et al., 2014).

including 54 inner fuel assemblies with an enrichment of 9.35% and 96 outer fuel assemblies with an enrichment of 17.75%. The reactor incorporates T-91 ferritic steel cladding, known for its high resistance to corrosion.

For the purpose of conservative shielding evaluation, six control assemblies, six shutdown assemblies, and one ultimate shutdown assembly—each containing neutron-absorbing materials—are excluded from the model (Yoo et al., 2014). Instead, the reactor is modeled in an all-rods-out configuration under steady-state conditions (Alizadeh et al., 2013). The total height of the primary side is 986 cm, while the active core height is 180 cm. To conservatively evaluate bulk radiation shielding, the tally for dose rate measurement is positioned at the shielding region behind the reactor, specifically within the range $z = -90$ cm to 90 cm, corresponding to the active core height.

To enhance computational efficiency, a commonly used geometry truncation method is applied (Kulesza et al., 2022), as seen in Fig. 2. This method simplifies the modeled geometry by excluding regions that

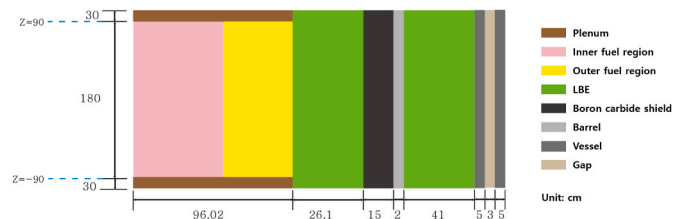


Fig. 2. Diagram of the multiple shielding layers surrounding the URANUS reactor core.

contribute negligibly to the tally results, which reduces the computational time per particle history. To maintain sufficient accuracy while simplifying the model, ± 30 cm axial extensions above and below the active core are included (Alizadeh et al., 2013), as shown in Fig. 2. In this computational model, the active core is surrounded by a 26.1 cm thick LBE reflector. To minimize radiation-induced damage to reactor materials and to attenuate neutron and gamma flux outside the reactor, a boron carbide shield is incorporated (Yu et al., 2023). Additionally, the reactor comprises a barrel next to a downcomer situated in the annular space between the outer wall of the barrel and the inner wall of the inner vessel, through which LBE coolant flows via natural circulation to absorb heat from the core. Beyond this, there is an insulating gap and the outer vessel.

3. Methodology

This section introduces the proposed methodology for improved computation in radiation shield analysis. The radiation source in the reactor is described, then the traditional SSW/SSR approach is explained for comparison. The newly developed VRT of the proposed methodology using a cylindrical surface source model is then discussed, followed by the optimization algorithm applied to minimize shielding volume and weight under dose constraints.

3.1. Source definition

Among the various sources generated during reactor operation, delayed fission neutrons, photoneutrons, and activation gamma rays are not considered due to their negligible contribution to shielding calculations. In contrast, the radiation sources that must be considered in shielding calculations include prompt fission neutrons, which occur simultaneously with or shortly after fission events; prompt fission gamma rays, emitted concurrently with the formation of fission products; and capture gamma rays, produced when a nucleus absorbs a neutron through radiative capture, significantly contributing to shielding calculations especially around the biological shield region (Schaeffer, 1973). These sources are defined using the KCODE card in MCNP. The KCODE calculation parameters are as follows: 50,000 source histories per cycle, 100 inactive cycles, and 8900 active cycles. The standard deviation of the calculated effective multiplication factor is 0.000023, which is below 100pcm (Rabir and Usang, 2012). The results of MCNP tallies are normalized to the number of source particles. The number of source particles, which are fission neutrons, is determined using Eq. (3). The thermal power of the reactor, P , is set to 100 MWth, and the recoverable energy per fission, Q , is assumed to be 200 MeV. The average number of neutrons emitted per fission, ν , is taken as 2.496 based on MCNP calculations. Since the reactivity swing of URANUS is 109.8 pcm, which is very close to criticality, the effective multiplication factor, k_{eff} , can be approximated as 1 (Žerovnik et al., 2014). Using this equation, the calculated average number of fission neutrons is 7.79×10^{18} per second:

$$S = \frac{P \cdot \nu}{Q \cdot k_{eff}} \quad (3)$$

3.2. Traditional surface source

To improve computational efficiency in radiation shielding optimization, particularly when the material composition or thickness of the biological shield is varied, the SSW and SSR features in MCNP are typically employed (Hendricks et al., 2022). In this study, the traditional SSW/SSR approach and the newly developed VRT were compared. Fig. 3 presents the top view of the reactor model, where points (a) and (b) mark the surfaces on which the neutron spectrum was measured. As neutrons pass through the multiple shielding layers surrounding the reactor core, the fluence of fast neutrons with energy above 0.1 MeV,

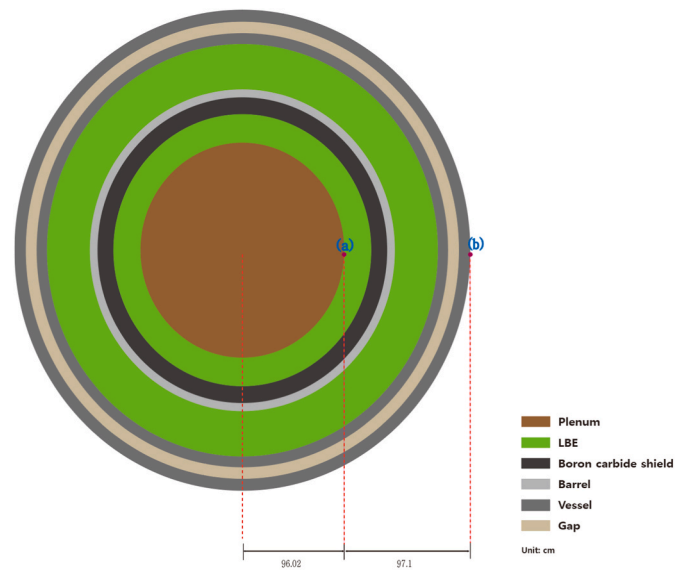


Fig. 3. Locations of neutron spectrum measurements (reactor top view).

which has a significant impact on shielding calculations, is reduced to 0.000179 times its initial value from the active core outer surface to the outer surface of the outer vessel (Hamzah et al., 2018), with the relative error in the MCNP calculation remaining below 0.5%.

In this study, the SSW card was used to record the source particle information (such as particle weight, direction, and position) on the outer surface of the reactor's outer vessel. After recording the source, the active core region was removed from the geometry, and the simulation was re-initiated using the SSR card from the surface source, bypassing the need for a full criticality calculation in each iteration (Cuneo and Conn, 2013). This method is particularly advantageous in deep penetration problems where the computational load increases significantly due to the low probability of particles reaching distant tally regions (Yang et al., 2022). By starting the particle histories directly from the outer vessel surface, the simulation concentrates solely on the shielding region of interest, which reduces unnecessary computational overhead.

However, there are limitations to this approach. The SSW process does not support multithreading, meaning that source recording can be time-consuming for large numbers of particles. Moreover, when the number of particles in the new SSR calculation exceeds the number of particles stored in the surface source file, the relative error tends not to decrease effectively, which compromises simulation accuracy. As a result, a sufficiently large number of particle histories must be recorded during the SSW step to ensure statistical reliability in subsequent SSR-based simulations (Kulesza et al., 2022). In addition, as the mesh-based WWG function built into MCNP is time-consuming and the SSW/SSR approach is incompatible with ADVANTG (Van Riper and Metzger, 2017), repetitive iterations remain required to achieve convergence (Sweezy et al., 2005). To overcome these challenges, a new variance reduction method is introduced in this study.

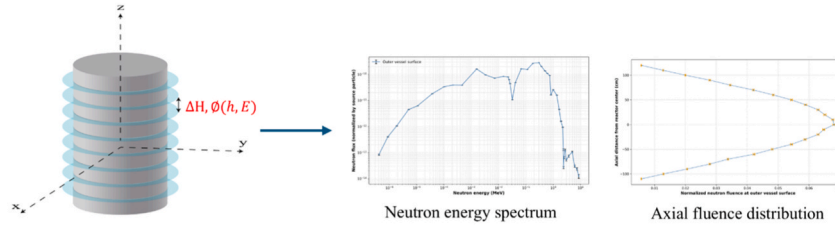
3.3. Surface source in the newly developed VRT

To more efficiently define the source, we propose a two-step VRT without using the SSW card, as illustrated in Fig. 4.

In Step 1, the reactor is modeled within the axial range $z = -120$ cm to 120 cm, and the MCNP KCODE criticality calculation is performed. Using the F2 tally along the height of the outer surface of the outer vessel, the height and energy dependent neutron fluence, $\phi(h, E)$, is recorded; i.e., the neutron fluence is binned by height and energy. Given the reactor's symmetric geometry, the neutron fluence is assumed to be symmetric with respect to the polar angle in the cylindrical coordinate

Step 1. Source recording

- Reactor modeling
- KCODE criticality calculation
- The energy and height dependent neutron fluence is recorded on the exterior surface of the outer vessel.



Step 2. Flux/Dose calculation with cylindrical surface source

- Definition of a cylindrical surface source using the recorded height and energy dependent fluence distribution at the outer vessel surface
- Simplification of reactor geometry by excluding unnecessary regions
- Definition of the external biological shield
- Radiation shielding analysis

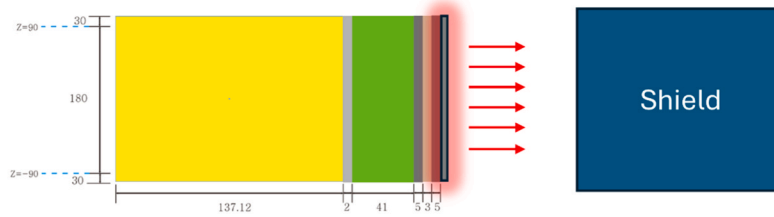


Fig. 4. Flow chart of source generation in the newly developed VRT.

system (Lamarsh et al., 2001). To avoid overestimating the recorded fluence due to back-scattered neutrons from the biological shield, external shielding layers are excluded during this tallying process (Hendricks et al., 2022).

In Step 2, $\phi(h, E)$ is used to define a cylindrical surface source in MCNP using the SDEF card. This card specifies the source particle's position, axis, and energy as a function of height, particle type, and source height. In earlier versions of MCNP and MCNPX, cylindrical surface sources on a problem surface could not be defined using only the SDEF card, and source particles were emitted isotropically from the

source point. However, by using the SUR card along with the SDEF card in MCNP, it is possible to constrain the source direction outward to the cylindrical surface, enabling accurate modeling of a cylindrical surface source (Hendricks et al., 2005). Subsequently, the biological shielding is modeled to attenuate the dose caused by this source. To account for the effect of neutrons that are scattered back from the vessel wall and contribute to the dose inside the shield, the internal geometry of the vessel (excluding the active core) is included in the model. In this study, not only the active core but also the strong neutron absorber (the boron carbide shield) was removed (Rizwan et al., 2020). Figs. 5 and 6

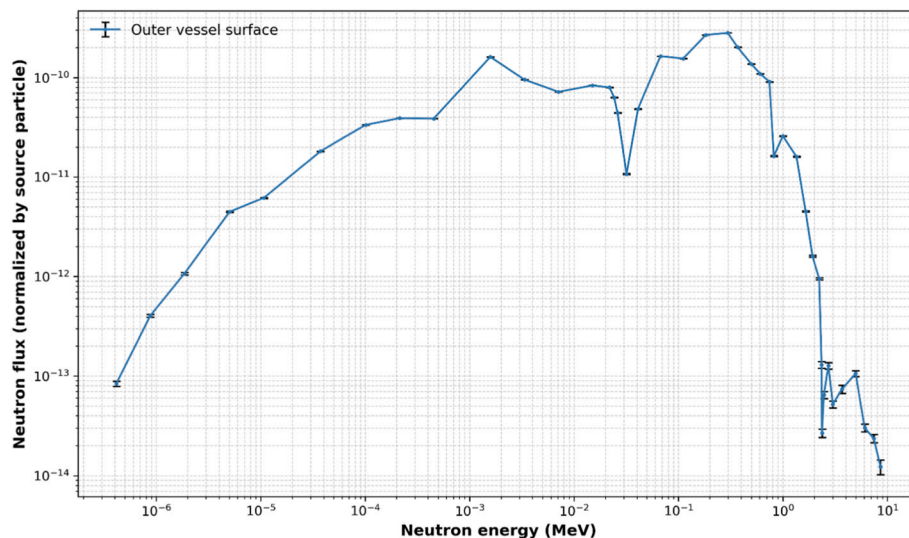


Fig. 5. Neutron energy spectrum on the outer surface of the outer vessel.

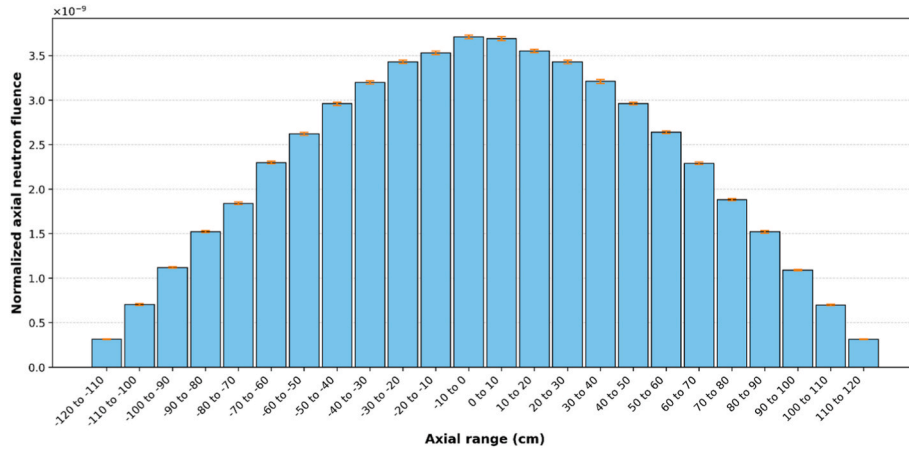


Fig. 6. Axial neutron fluence on the outer surface of the outer vessel.

respectively present the neutron energy spectrum observed on the outer surface of the outer vessel and the axial distribution of neutron fluence on the cylindrical surface.

To evaluate not only the neutron source but also the gamma rays emitted from the reactor vessel, two separate simulations are conducted to compare the dose contribution from primary gamma rays emerging from the vessel and secondary gamma rays produced through interactions with surrounding shielding materials. First, to estimate the dose rate from primary gamma rays, the SSW method is used to record the source at the vessel surface, and the dose rate is then calculated using a reference material, 150 cm thick concrete, which is effective at attenuating both neutrons and gamma rays (El-Khayatt, 2011). The second simulation is performed using only the neutron source emitted from the vessel. This allows for an evaluation of the secondary gamma dose rate resulting from radiative capture reactions in the shielding material. Fig. 7 illustrates the model used in this analysis, in which a 150 cm thick concrete layer is placed 10 cm away from the vessel.

Fig. 8 illustrates the variation in gamma dose rate as the thickness of the concrete increases, up to 150 cm. In deep penetration scenarios involving thick shielding, the contribution of primary gamma rays to the total dose rate becomes negligible compared to that of secondary gamma rays. Therefore, when defining the reactor as a source, only neutrons are used to define the cylindrical surface source. To account for secondary gamma rays generated through radiative capture, the MCNP simulation is performed with the “mode n p” option enabled, which allows for coupled neutron and photon transport calculations. All MCNP simulations reported in Fig. 8 were converged with relative errors below 2%.

For conservative estimation, the axial range of the tallies is limited to the active core height ($z = -90$ cm to 90 cm). F2 tallies are used to calculate particle flux, and the dose rates behind the biological shield for both neutrons and gamma rays are obtained using ICRP-116 conversion factors (Petoussi-Henss et al., 2010). In this approach, the radiation dose rate behind the biological shield is calculated using Eq. (4):

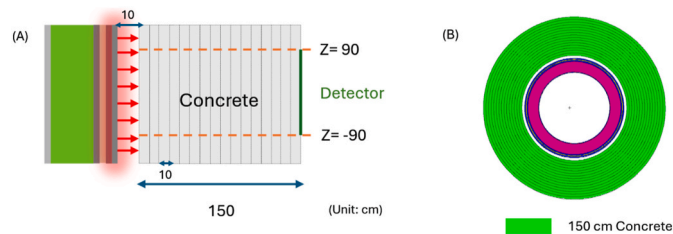


Fig. 7. Geometry of the shielding model with a 150 cm concrete layer positioned 10 cm from the vessel: (A) side view, (B) top view.

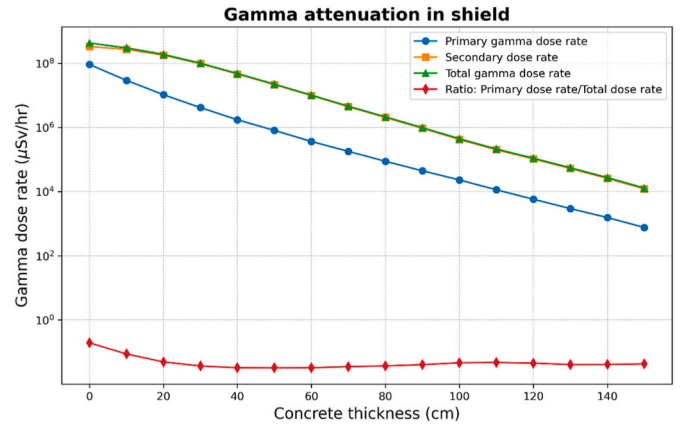


Fig. 8. Dose rate contributions of primary and secondary gamma rays by concrete shield thickness.

$$\text{Dose rate} = \text{Dose rate}_{\text{fixed}} \times \frac{\phi_{\text{KCODE},0}}{\phi_{\text{fixed},0}} \times S. \quad (4)$$

To account for the difference in neutron weight normalization between the KCODE and fixed-source calculations on the outer surface of the outer vessel, the dose rate obtained from the newly developed method, $\text{Dose rate}_{\text{fixed}}$, is multiplied by the ratio of the neutron flux at the outer vessel surface calculated from KCODE ($\phi_{\text{KCODE},0}$) to the neutron flux at the outer vessel surface calculated from the newly developed method ($\phi_{\text{fixed},0}$). Finally, the result is scaled by the number of fission neutrons derived from Eq. (3) (Žerovnik et al., 2014).

3.4. Optimization using genetic algorithm

In this study, a genetic algorithm (GA) was employed to optimize the multilayer radiation shielding design of the URANUS fast reactor. The objective of the design is to minimize the volume and weight of the shielding while satisfying the dose rate constraint of less than 10 μSv/hr behind the biological shield. Unlike conventional trial-and-error methods, the GA provides an automated and efficient means of exploring the design space, yielding lightweight and compact shielding configurations under multiple constraints (Bagheri et al., 2024).

The shielding structure consists of five layers as shown in Fig. 9, where the material and thickness of the first four layers are randomly selected between water and stainless steel, depending on the optimization sampling. Water is selected for its high neutron removal cross section relative to density, making it effective for neutron attenuation,

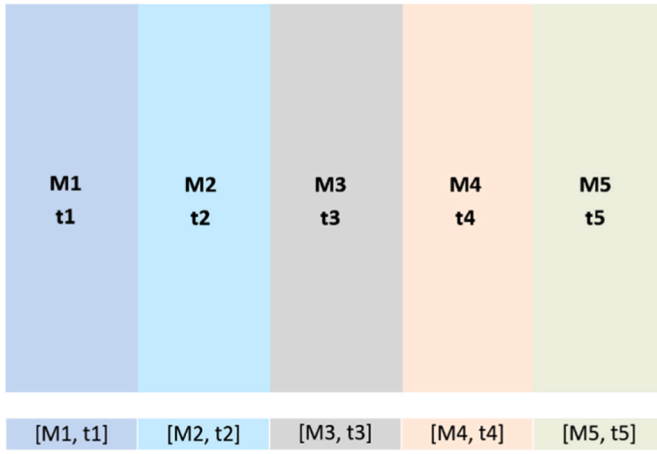


Fig. 9. Representation of the chromosome structure in the genetic algorithm for multilayer shielding design (M: material; t: thickness).

while steel provides superior shielding for both neutrons and gamma rays (Chen et al., 2018; El-Khayatt, 2010). The material of the final layer, M5, is fixed as lead, which effectively attenuates capture gamma rays produced in the preceding layers, while its thickness is randomly selected within a specified range during the optimization process.

Each chromosome represents a candidate shielding configuration in the form [Material1, Thickness1], ..., [Material5, Thickness5], where the thickness values are sampled within predefined bounds specific to each material. The initial population consists of 200 chromosomes randomly generated within the constraints.

To evaluate each candidate design, a Python-based intermediate script was used to automatically generate MCNP and ADVANTG input files, enabling radiation transport simulation with weight windows. The resulting neutron and gamma dose rates were combined to calculate the total dose equivalent.

The following objective function is designed to minimize the volume and weight of the shielding system under the condition that the dose rate limit is not exceeded:

$$F(X) = C_R f_R(X) + C_V f_V(X) + C_W f_W(X), \quad (5)$$

where

$$f_R(X) = \frac{R(X)}{R_0} < 1 \quad (6)$$

$$f_V(X) = \frac{V(X)}{V_0} < 1 \quad (7)$$

$$V(X) = \sum_{i=1}^M V_i \quad (8)$$

$$f_W(X) = \frac{W(X)}{W_0} < 1 \quad (9)$$

$$W(X) = \sum_{i=1}^M \rho_i V_i \quad (10)$$

$$X = [x_1, x_2, \dots, x_n] \quad (11)$$

$$L_j < x_j < U_j \quad (12)$$

$$E(X) = 1 - \frac{F(X) - F_{min}}{F_{max} - F_{min}} \quad (13)$$

The functions $f_R(X)$, $f_V(X)$, and $f_W(X)$ represent the dimensionless objective components corresponding to the total dose rate including neutron dose rate and gamma dose rate, total volume, and total weight of the biological shield, respectively. The normalized weighting

coefficients C_R , C_V , and C_W are assigned to each objective component. In this study, the coefficients were set to 0.1, 0.45, and 0.45, respectively, for cases in which the total dose remained below the regulatory limit $R_0 = 10 \mu\text{Sv/hr}$, assuming an occupational exposure scenario of 40 h per week and 50 working weeks per year. The total shield volume $V(X)$ is expressed in cubic meters (m^3), and the total shield weight $W(X)$ is given in metric tons. The reference values V_0 and W_0 correspond to the volume and weight of the baseline shielding design, respectively (Cai et al., 2018). $E(X)$ represents the fitness function corresponding to the objective function. The maximum and minimum values of the objective function, F_{max} and F_{min} , are defined as the highest and lowest values of $F(X)$ among all chromosomes within a single generation. The initial five-layer shielding model used for reference was relatively thick and conservatively designed, consisting of 50 cm of water, 40 cm of stainless steel, 70 cm of water, 10 cm of stainless steel, and 10 cm of lead, totaling 140 cm in thickness. This configuration yielded a total volume V_0 of 55.549 m^3 , a total weight W_0 of 231.8 metric tons, and a dose rate of $7.08 \mu\text{Sv/hr}$.

Fig. 10 presents the flowchart of the GA applied to the shielding design in conjunction with MCNP. Initially, the material types and thicknesses for each shielding layer are randomly selected, and the corresponding volume and weight of the shielding configuration are calculated. These candidate chromosomes are evaluated against predefined constraints, including the reference volume V_0 , reference weight W_0 , and other design limits. If a candidate satisfies the constraints, it is accepted into the initial parent population P_1 ; otherwise, it is either discarded or regenerated.

For each valid chromosome, an MCNP input file is generated using an intermediate script. After the MCNP simulation is completed, the total

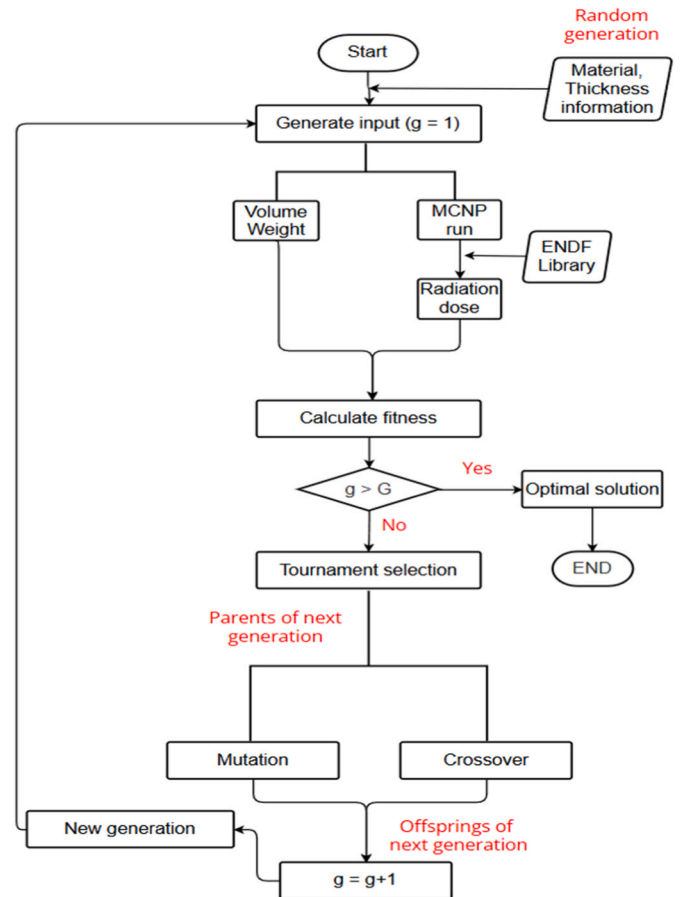


Fig. 10. Flowchart of reactor shielding design using MCNP with the genetic algorithm.

dose rate for each chromosome is obtained. If the dose rate exceeds 10 $\mu\text{Sv/hr}$, the chromosome is eliminated from the population. Next, the objective function $F(X)$ is calculated, and the fitness function $E(X)$ is used as the selection basis. A standard binary tournament selection is employed to form the next parent population P_2 . The offspring population O_2 is generated from P_2 through crossover, i.e., combining the material and thickness information from two parent chromosomes, or through mutation, which randomly modifies a gene in a single parent chromosome. The parent and offspring populations are merged to create the combined population C_2 , and the process is repeated until the maximum number of generations G is reached (Chen et al., 2020). In the preliminary optimization, the GA was executed with an initial population of 200 chromosomes over 50 generations, with each generation consisting of 100 parent chromosomes and 100 offspring chromosomes.

4. Results

4.1. Sensitivity analysis on the newly developed source

To develop a source model that is consistently conservative, less biased, and more efficient than the KCODE source, a sensitivity analysis was conducted. In Step 1 (see Section 3.3), the recording region at the outer surface of the outer vessel, spanning from $z = -120$ cm to 120 cm, was divided into 0, 6, 12, 24, 48, 96, and 192 axial segments, and the energy spectra of neutrons reaching each segment were recorded in 47 energy bins. In Step 2, the height-dependent energy spectrum of the source was defined in two ways: using the maximum energy value within each energy bin, and by sampling uniformly across each energy interval. The performance of ADVANTG's energy biasing function was also evaluated for comparison. Fig. 7 was used for the analysis model, where the neutron dose rates calculated at every 10 cm depth within a 150 cm thick concrete shield (Dose rate) were compared against the dose rate obtained using the KCODE source ($Dose_{rate_0}$). To ensure conservative calculation, the dose ratio (Eq. (14)) should be greater than 1,

$$Dose\ ratio = \frac{Dose\ rate}{Dose\ rate_0} \quad (14)$$

In Figs. 11 and 12, the notation H(a)erg47(b) represents the number of height divisions (a) and the sampling method for each energy bin (b), where 'h' indicates uniform sampling across the energy bin and 'd' indicates sampling only at the maximum energy. The x-axis represents the concrete thickness, and the y-axis indicates the dose ratio at that thickness. Lastly, 'ssw' refers to the dose ratio of the cylindrical surface source using the MCNP WWG and SSW functions, as described in

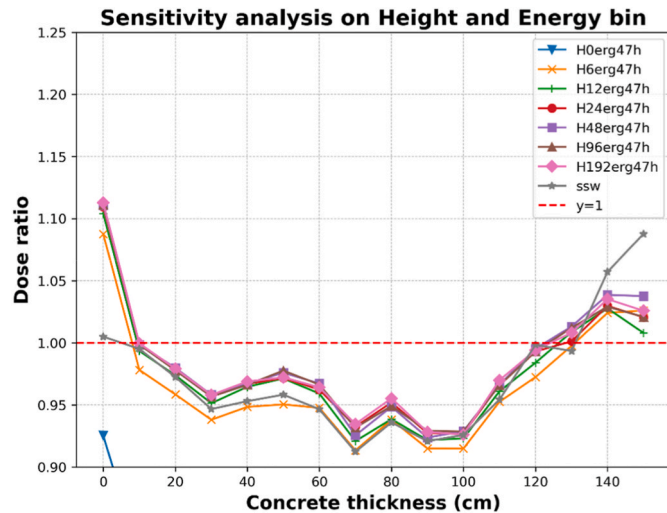


Fig. 11. Uniform neutron energy distribution across each energy interval.

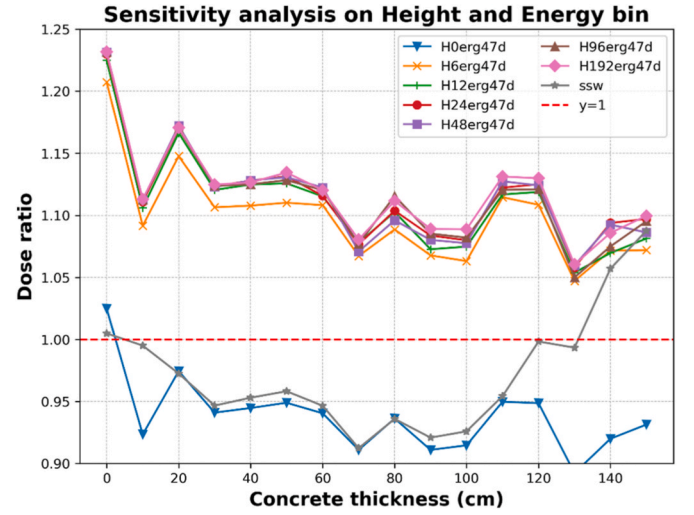


Fig. 12. Maximum neutron energy distribution across each energy interval.

Sections 2.2 and 3.2.

Fig. 11 plots the dose ratios for sources defined with 47 neutron energy bins, each sampled uniformly across its interval, for various height division methods. The case with no height division and only energy distribution defined (H0erg47h) resulted in significant underestimation, with dose ratios falling below 0.9 for most concrete thicknesses. Additionally, all other methods including SSW showed underestimation. In Fig. 12, each energy bin was sampled at its maximum energy. Except for the case without height division, all methods showed conservative results and tended to converge toward the dose rate calculated by KCODE as the shielding thickness increased.

Fig. 13 presents the results when sampling only at the maximum energy in each energy bin while additionally applying ADVANTG's energy biasing technique, which enhances the sampling of high-energy particles compared to analog calculations (Mosher et al., 2015). The direction biasing option was not used. The results show little difference from those in Fig. 12, but the FOM, an indicator of calculation efficiency, increased and the relative error decreased, as discussed below. In all MCNP simulations presented in Figs. 11–13, the relative errors were maintained within 5%, indicating the statistical reliability of the calculated results.

Fig. 14 illustrates the FOM obtained from Eq. (2), which quantifies

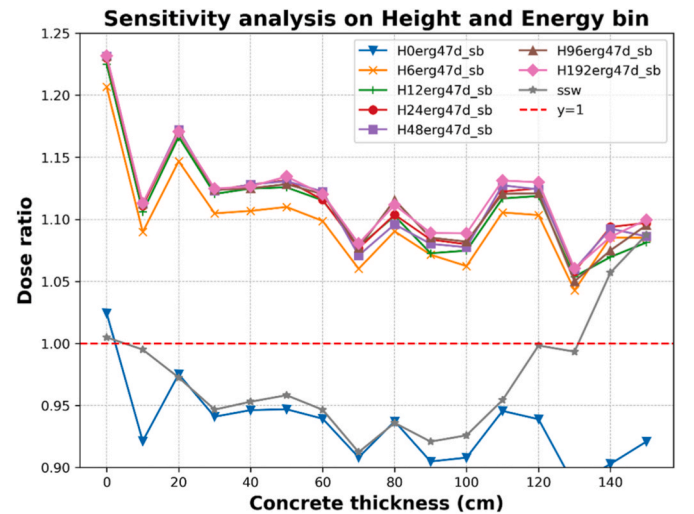


Fig. 13. Maximum neutron energy distribution across each energy interval with energy biasing.

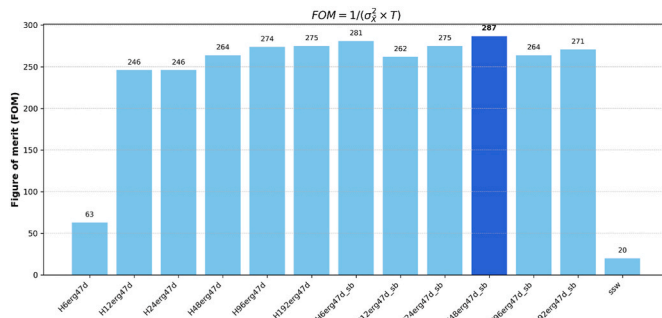


Fig. 14. Calculation efficiency (FOM) of each source definition method.

the computational efficiency, for each source definition method under an equal computing time of 120 min. All simulations were performed on an AMD Ryzen 9 9950X. The method that achieved the highest computational efficiency involved defining the energy in each bin by its maximum value, dividing the outer surface of the vessel into 48 axial segments for source recording, and applying energy biasing through ADVANTG. This method, namely H48erg47d_sb, highlighted in blue in the figure, consistently yielded conservative results while achieving the highest efficiency. Furthermore, its computational efficiency was 14.35 times higher than that of the SSW-defined source. Based on these findings, this source definition was selected for the subsequent shielding analysis. It is also noted that the conventional approach based on the SSW-defined source required 38.17 min to generate the weight window. In contrast, when ADVANTG was used, only 3.73 min were needed.

4.2. Shielding optimization results

To determine the source term for a shielding analysis of the URANUS reactor, source particles were recorded using surface tallies on the outer surface of the outer vessel and reformulated into a cylindrical source with height- and energy-dependent distribution. Subsequently, preliminary shielding design optimization was performed using the GA (see Section 3.4) to determine the optimal thicknesses of the shielding layers. In all MCNP simulations performed in the optimization process, the relative error of the dose rate calculations was maintained below 5%, ensuring high reliability of the results. Fig. 15 plots the minimum value of the objective function in the combined population C_g at the g th generation. While keeping the dose rate below $10 \mu\text{Sv/hr}$, the objective function was designed to minimize the dose rate while reducing the shield volume and weight, with the combined weights for the dose rate, volume, and weight components set as $C_R = 0.1$, $C_V = 0.45$, and $C_W = 0.45$, respectively. The plot exhibits a consistent decrease in the objective function over generations, indicating convergence toward an optimal shielding configuration.

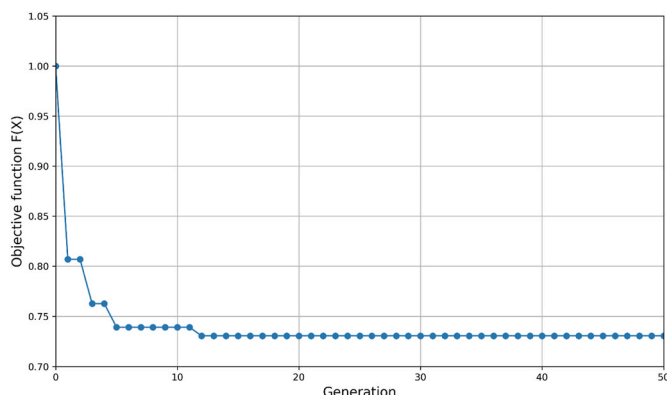


Fig. 15. Reduction of the objective function $F(X)$ over generations.

Fig. 16 presents a cross-sectional comparison between the initial and optimized shielding designs. Detailed information on the material, density, and thickness of each layer in the optimized shielding configuration is provided in Table 2. To compare the attenuation performance between the initial shielding design and the proposed design, the dose rates were calculated as a function of the biological shield thickness outside the outer vessel. Fig. 17 illustrates the attenuation of neutron, gamma, and total dose rates with respect to shield thickness for both the initial and proposed designs, while also presenting the corresponding relative errors of the neutron and gamma dose rates. Although the total dose rate in the initial shielding design reaches 10 $\mu\text{Sv/hr}$ at a thickness of 139 cm, the configuration is over-shielded. In the proposed design, the neutron dose rate decreases more steeply than that of the initial design between approximately 13 cm and 103 cm, demonstrating an effective suppression of neutron penetration behind the main shield region. For gamma rays, the proposed design maintains a lower dose rate than the initial design beyond approximately 40 cm, effectively suppressing the dominant gamma contribution to the total dose rate at the rear side of the shield. This shows that the optimized configuration achieves more efficient overall shielding by focusing on the attenuation of the dominant gamma component in the deeper regions. Overall, compared to the initial design, which is relatively thick and heavy, the proposed design achieved a 33.3% reduction in shielding volume and a 15.6% reduction in shielding weight while successfully maintaining the dose rate below the target limit of 10 $\mu\text{Sv/hr}$.

5. Conclusion

In this study, an advanced methodology was developed to enable efficient and accurate three-dimensional shielding analysis for heterogeneous reactors. The main objective was to address several practical challenges in shielding simulations, particularly those associated with conventional SSW/SSR-based VRTs.

By directly recording the energy and axial distributions of the neutron source on the outer surface of the outer vessel using KCODE, a cylindrical surface source was formulated. In this source model, the outer vessel height was divided into 48 axial segments, and the energy spectrum was conservatively discretized into 47 energy bins conservatively, which enabled effective variance reduction without reliance on conventional SSW or WWG methods.

Based on this source, a shielding design for the URANUS fast reactor was optimized using a GA, resulting in a 33.3% reduction in shielding volume and a 15.6% reduction in weight compared to the initial shielding design, while maintaining dose rates below 10 $\mu\text{Sv/hr}$.

Additionally, the newly developed source definition technique demonstrated high fidelity in reproducing the spatial and energy distribution of the original neutron flux, while also offering significant improvements in computational efficiency. Based on these advantages,

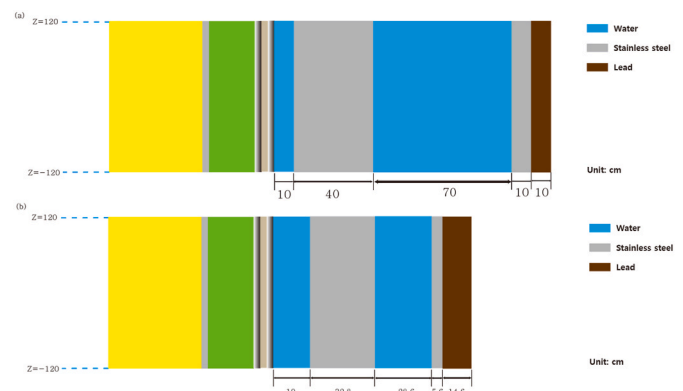


Fig. 16. Cross-sectional view of the (a) initial shield design and (b) proposed design.

Table 2

Shielding material properties and geometrical parameters of the proposed biological shield.

Material	Density (g/cm ³)	Thickness (cm)	Outer radius (cm)
Water	0.98	19	212.12
Stainless steel	7.85	32.8	244.92
Water	0.997	28.6	273.52
Stainless steel	7.85	5.6	279.12
Lead	11.34	14.6	293.72

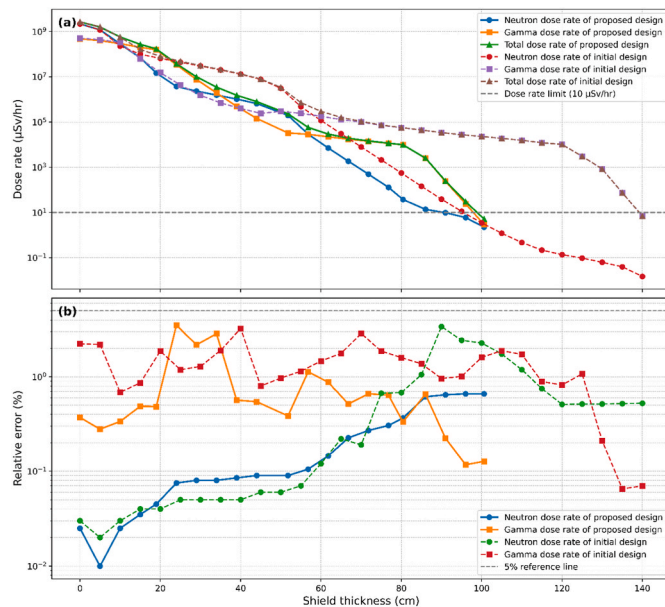


Fig. 17. (a) Comparison of dose rates between initial and proposed shielding designs. (b) Corresponding relative errors of neutron and gamma dose rates.

the methodology presented in this study contributes to improving the efficiency of shielding calculations for advanced nuclear systems, particularly for those in compact areas. The proposed cylindrical surface source formulation allows for deep-penetration problems to be tackled more efficiently with conservative dose rates.

Despite these achievements, the current optimization used only a limited set of materials such as water, stainless steel, and lead. Future work should explore the application of advanced shielding materials with superior attenuation performance and lower mass, as well as consider material cost and thermal properties for more practical reactor shielding design optimization.

In conclusion, this work provides a new method for efficient and conservative source generation for reactor shielding calculations. By introducing a cylindrical surface source compatible with high-performance computation, the proposed method not only addresses current challenges in computational efficiency but also lays the foundation for future applications in transportable, space-constrained, and optimized nuclear reactor shielding systems.

CRediT authorship contribution statement

Seungjae Jo: Writing – review & editing, Writing – original draft, Visualization, Methodology, Investigation, Formal analysis, Data curation, Conceptualization. **Jaehyun Cho:** Supervision, Software, Resources, Project administration, Funding acquisition.

Declaration of competing interest

The authors declare that they have no known competing financial

interests or personal relationships that could have appeared to influence the work reported in this paper.

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Appendix A. Supplementary data

Supplementary data to this article can be found online at <https://doi.org/10.1016/j.pnucene.2026.106422>.

Data availability

Data will be made available on request.

References

- Alizadeh, A., Shirani, A.S., Kashi, S., 2013. Neutron and gamma-ray Deep Penetration Calculation Through Biological Concrete Shield of VVER-1000 Reactor by a New Technique Based on Variance Reduction.
- Bagheri, S., Khalafi, H., 2023. SMR, 3D Source Term Simulation for Exact Shielding Design Based on Genetic Algorithm.
- Bagheri, S., et al., 2024. An optimal radiation shield design of a small modular nuclear power reactor by genetic algorithm. *Nucl. Technol.*
- Cai, Y., et al., 2018. Metaheuristic optimization in shielding design for neutrons and gamma rays. *Ann. Nucl. Energy*.
- Chen, Y.Q., Yan, B., 2018. The Technology of Shielding Design for Nuclear Reactor: a Review.
- Chen, Z., Zhang, Z., Xie, J., Guo, Q., Yu, T., Zhao, P., Liu, Z., Xie, C., 2020. Multi-objective optimization strategies for radiation shielding design with genetic algorithm. *Comput. Phys. Commun.* 252, 107267.
- Cramer, S.N., 1984. VARIANCE REDUCTION METHODS APPLIED TO DEEP-PENETRATION PROBLEMS.
- Culbertson, Christopher N., Hendricks, John S., 2000. An Assessment of the MCNP4C Weight Window. Los Alamos National Laboratory Report. LA-UR-00-1656.
- Cuneo, M.E., Conn, R.W., 2013. Critical Source Options for MCNP6 Surface Source Files. Los Alamos National Laboratory. LA-UR-13-21223.
- El-Khayatt, A.M., 2010. Calculation of fast neutron removal cross-sections for some compounds and materials. *Ann. Nucl. Energy* 37 (2), 218–222.
- El-Khayatt, A.M., 2011. Calculation of radiation attenuation coefficients for shielding concretes. *Ann. Nucl. Energy* 38 (6), 1280–1285.
- Goorley, Tim, 1994. Criticality Calculations with MCNP5, second ed. A Primer.
- Hamzah, A., Suwoto, S., Rohanda, A., Adrial, H., Bakhri, S., Sunaryo, G.R., 2018. Neutron fluence and DPA rate analysis in pebble-bed HTR reactor vessel using MCNP. *J. Phys., Conf. Ser.* 962 (1), 012044.
- Hendricks, J.S., McKinney, G.W., Waters, L.S., Prael, R.E., Haight, R.C., 2005. MCNPX EXTENSIONS VERSION 2.5.0. Los Alamos National Laboratory, Tech. Rep. LA-UR-05-2675.
- Hendricks, J.S., Swinhoe, M.T., Favalli, A., 2022. In: Monte Carlo N-Particle Simulations for Nuclear Detection and Safeguards: an Examples-based Guide for Students and Practitioners. Cham, Switzerland. Springer Nature.
- Houtkoop, K.C.F., 2022. Nuclear Reactors for Marine Propulsion and Power Generation Systems. M.Sc. thesis. Dept. of Marine Technology, Delft University of Technology, Delft, The Netherlands.
- IMO adopts historical agreement on net zero emissions target for 2050. *Rederiforbundet*, Jul. 7, 2023 [Online]. Available: <https://www.rederi.no/articles/imo-adopts-historical-agreement-on-net-zero-emissions-target-for-2050/>.
- Junhong Yu, 2020. Marine Nuclear Power Technology.
- Kim, K., Kwon, C.S., Kim, S., 2025. A conceptual study of 15,000 TEU SMR-powered container ships. *Int. J. Nav. Archit. Ocean Eng.* 17, 100662. <https://doi.org/10.1016/j.jnaoe.2025.100662>, 1–15.
- Kulesza, J.A., et al., 2022. In: MCNP® Code Version 6.3.0 Theory & User Manual, Rev. 1. Los Alamos National Laboratory. LA-UR-22-30006.
- Lamarsh, J.R., Baratta, A.J., 2001. Introduction to Nuclear Engineering.
- Martínez-Fernández, E., 2024. Neural network-based source biasing to speed-up challenging MCNP simulations. *Fusion Eng. Des.* 202, 114406.

- Matijević, M., Trontl, K., Šadek, S., Družijanić, P., 2025. Point-Kernel code development for gamma-ray shielding applications. *Appl. Sci.* 15 (14), 7795. <https://doi.org/10.3390/app15147795>.
- Mosher, S.W., Johnson, S.R., Beville, A.M., Ibrahim, A.M., Daily, C.R., Evans, T.M., Wagner, J.C., Johnson, J.O., Grove, R.E., 2015. ADVANTG—An automated variance reduction parameter generator. ORNL/TM-2013/416 Rev. 1. Oak Ridge National Laboratory, Oak Ridge, TN, USA, Aug.
- OECD/NEA, 2015. Handbook on lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermalhydraulics and Technologies.
- Petoussi-Henss, N., Bolch, W.E., Eckerman, K.F., Endo, A., Hertel, N., Hunt, J., Pelliccioni, M., Schlattl, H., Zankl, M., International Commission on Radiological Protection, 2010. ICRP publication 116 conversion coefficients for radiological protection quantities for external radiation exposures. et al. *Ann. ICRP* 40, 1–257.
- Rabir, M.H., Usang, M.D., 2012. Modeling the Puspiti Triga reactor using mcnp code. In: *Proceedings of the R and D Seminar*.
- Rizwan, S., Hamid, M.A.A., Jusoh, N., Rahman, R.A., Arshad, M.K.A., 2020. Characterization of boron carbide particles and its shielding behavior against neutron radiation. *Results Phys.* 16, 102939.
- Schaeffer, N.M., 1973. *Reactor Shielding for Nuclear Engineers*.
- Sweezy, J., Brown, F., Booth, T., Chiaramonte, J., Preeg, B., 2005. Automated variance reduction for MCNP using deterministic methods. *Radiat. Protect. Dosim.* 116 (1–4), 508–512. <https://doi.org/10.1093/rpd/nci257>.
- Van Riper, Kenneth A., Metzger, Robert L., 2017. Automated ADVANTG variance reduction in a proton driven system. *M&C 2017 - International Conference on Mathematics & Computational Methods Applied to Nuclear Science & Engineering*. Jeju, Korea.
- Wang, X., Zhang, B., Dai, N., Chen, Y., 2021. Multicollision source method for discrete ordinates neutron transport calculation. *Ann. Nucl. Energy* 156, 108216. <https://doi.org/10.1016/j.anucene.2021.108216>, 1–12.
- Wood, J.A.M.E.S., 1982. *Computational Method in Reactor Shielding*.
- Yama Ji, Akio, Sako, Kiyoshi, 1994. *Shielding design to obtain compact marine reactor*. J. Nucl. Sci. Technol.
- Yang, H., Liu, J., Li, J., Wu, H., 2022. An auto-importance sampling method for deep penetration problems. *Ann. Nucl. Energy* 178, 109398.
- Yoo, J.-W., Kim, Y.J., Sungyeol, C., Jaehyun, C., Soon, H.I., 2014. Advanced Passive Design of Small Modular Reactor Cooled by Heavy Liquid Metal Natural Circulation.
- Yu, H., Zhu, G., Zou, Y., Yan, R., Liu, Y., Kang, X., Dai, Y., 2023. Neutron/Gamma Radial Shielding Design of Main Vessel in a Small Modular Molten Salt Reactor.
- Žerovnik, G., Podvratnik, M., Snoj, L., 2014. On normalization of fluxes and reaction rates in MCNP criticality calculations. *Ann. Nucl. Energy* 63.