Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 22-23, 2025

Monte Carlo Shielding Simulation for a Cylindrical Well Storage Containing Irradiated Stainless Steel Capsules

Jati Eka Putri and Juyoul Kim*
Department of Nuclear Power Plant Engineering, KEPCO International Nuclear Graduate School, 658-91 Haemaji-ro, Seosaeng-myeon, Ulju-gun, Ulsan 45014
**Corresponding author: jykim@kings.ac.kr*

Keywords: cylindrical well storage, stainless steel radioactive waste, effective dose, PHITS

1. Introduction

The assessment of dose rates around radioactive sources is critical for ensuring safety in environments where radioactive materials are handled. This paper presents a simulation study focused on a cylindrical well storage containing radioactive wastes, specifically designed with concrete and lead shielding. Such configurations are prevalent in the safe storage of radioactive materials. The study employs Particle and Heavy Ion code Systems (PHITS) [1] to simulate and analyze dose rates at various height positions around the well. This research is conducted at Indonesia's Radioactive Waste Management Installation (RWMI) [2]. The facility aims to develop safe and efficient storage solutions for high-activity radioactive waste generated from research and industrial applications as shown in Fig. 1.

Fig. 1 Interim storage with underground cylindrical well

The primary objective of this study is to simulate and analyze dose rates around a cylindrical well containing radioactive waste from stainless steel (SS) capsules, ensuring that radiation exposure remains within safe limits as mandated by Indonesian regulations. This research aims to assess the effectiveness of concrete and lead shielding of well structure design in reducing radiation exposure as shown in Fig. 2. The results will support the development of safe storage solutions at the Interim Storage Facility and contribute valuable insights to enhance radiation protection policies and practices in Indonesia's radioactive waste management framework.

Fig. 2 Cylindrical well storage with rack hole

2. Method and Results

The one cylindrical well is designed to house 21 stainless steel (SS) holes, each containing seven SS capsules stacked vertically without gaps. The SS capsules measure 50 cm in length and 5 cm in diameter, contributing to an overall source activity of 58,800 mCi for Co-60, equivalent to 2.18×10^{12} Bq for PHITS input. The shielding configuration includes a 3 cm thick concrete wall surrounding the well, topped with a 33 cm concrete cover and a 5 cm thick lead cover acting as a circular cap. An air gap of 50 cm exists above the sources, extending to the surface cover. The well dimensions and materials were modeled according to these specifications. This study aligns with Indonesian regulations on radiation protection. BAPETEN Regulation No. 4 of 2013 [3] sets dose limits for radiation workers (20 mSv/year averaged over five years) and the public (1 mSv/year), ensuring exposures are kept as low as reasonably achievable (ALARA). From that regulation, the management of the facility conducts dose constraints for the workers by 15 mSv/year [4]. The Occupational Health and Safety division manages the radiation exposure limit for the interim storage area by 15 µSv/h [5]. This limit was monitored at a distance of 100 cm above the surface.

2.1 Simulation Setup

The simulation setup for this study utilized the PHITS code to model the dose rates around a cylindrical well containing radioactive sources. The simulation focused on photons, specifically simulating the decay of Co-60. The geometry included a cylindrical well with 21 stainless steel holes, each housing seven stainless steel capsules, topped with 33 cm concrete and a 5 cm lead cover, with a 50 cm air gap above the sources. Dose rates were calculated at various vertical positions (2 cm, 10 cm, 50 cm, and 100 cm above the well). To ensure statistical reliability, a total of 10 million particle histories were simulated, allowing for a comprehensive analysis of radiation exposure in the defined geometry.

Table 1. Material Composition

Material	Density (g/cm^3) [6]	Thickness (cm)
Source: Stainless steel (SS 304)	8.03	0.3
Air (inside source)	0.001205	5

Concrete-portland (Well structure)	2.3	3
Concrete-Portland (top cover)	2.3	33
Pb (top cover)	11.35	5
Soil	1.52	surrounding the well

Based on the material composition in Table 1, the design was made. The cylindrical well was simulated within the layer of shielding from inside to outside are concrete and lead to cover the well. Fig. 3 shows an illustrated design for one hole (assuming one source of SS capsules) inside the well. The results of the geometry input parameter are shown below in Fig. 4. The geometry shows air inside the source and a gap between the source and the well. Additionally, air is used outside the well as a medium for dose-rate transfer to the environment.

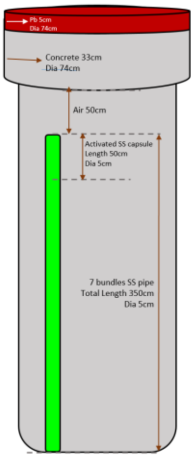

Fig. 3 Illustrated design for one hole

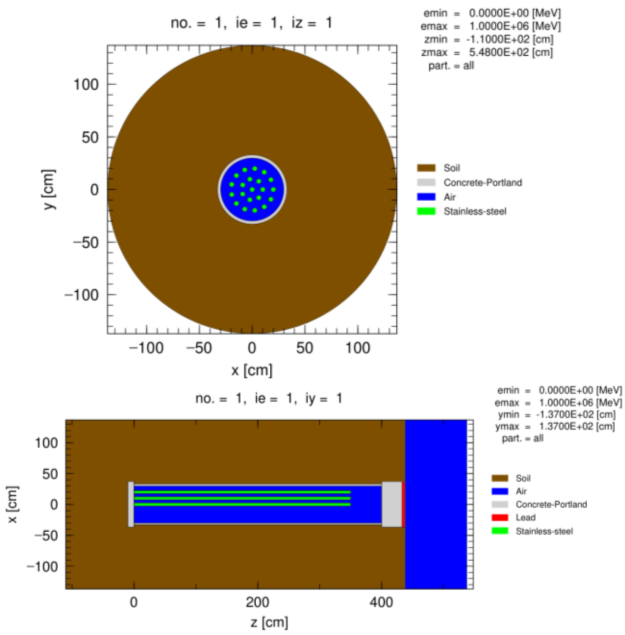

Fig. 4 Geometry in XY and XZ axis

2.2 Simulation Results

Co-60 is a radioactive isotope that decays by beta emission, producing energetic beta particles (electrons) and gamma radiation (photons) [7]. Co-60 undergoes beta decay, emitting beta particles (electrons). These beta particles are significant because they contribute to the radiation dose close to the source but are typically absorbed quickly in materials and are less of a concern for shielding design compared to gamma rays. The more critical aspect of shielding is the gamma radiation. Co-60 emits two primary gamma photons with energies of approximately 1.17 MeV and 1.33 MeV [8]. These high-energy photons can penetrate materials more deeply, requiring robust shielding to protect against them. Concrete is widely used as a shielding material in radiation protection due to its advantageous properties. Its high density, around 2.3 g/cm³, allows it to attenuate radiation effectively by absorbing and scattering gamma rays, significantly reducing their intensity. The substantial mass of concrete structures further enhances their shielding capabilities, as thick layers can create effective barriers against radiation [8]. The particles fluence distribution results for all, photons, and electrons were as shown in Fig. 5.

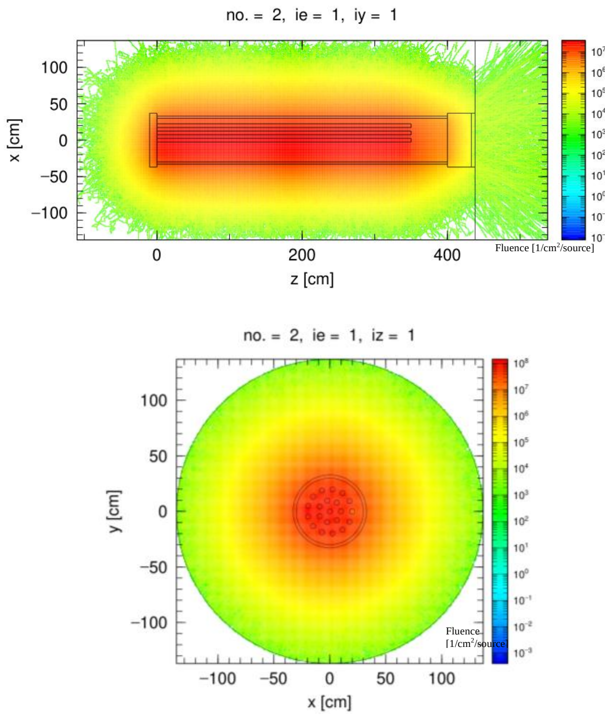

Fig. 5 Particles fluence distribution for photons

The dose rate distribution along the height direction shows the effective dose that will be received by the environment and worker. The axial dose-rate distribution refers to the variation of the radiation dose rate at different distances from the surface of well cover, moving outward vertically. In the context of the well, this distribution helps in assessing the effectiveness of the shielding materials in reducing radiation levels as one moves away from the well. The use of lead and concrete layers in the top cover plays a significant role in reducing the dose rate in this direction. The axial dose-rate distribution chart showed that the dose rates above the surface of the well were well exceeding acceptable limits. Fig. 6 shows the dose rate along the height of the well.

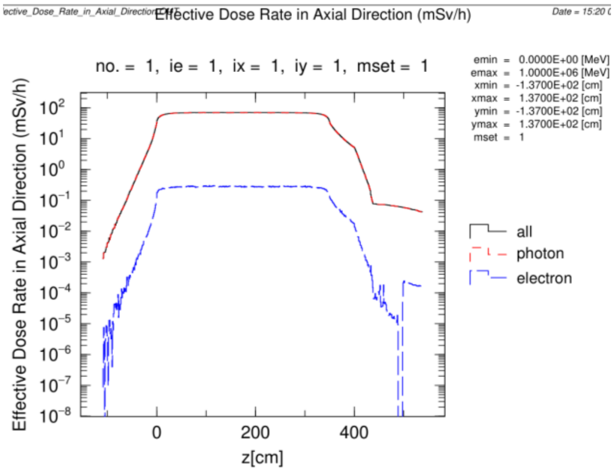

Fig. 6 Axial effective dose rate distribution

The PHITS simulation provided a comprehensive analysis of the dose rate distribution around the cylindrical well. The effective dose rate evaluation is crucial for ensuring compliance with regulatory limits. Table 2 presents the simulated dose rates (in µSv/h) at various distances from the surface of the well.

Table 2. Effective dose rate results

Distance from top/surface (cm)	*Dose rate (µSv/h)*
2	77.24
10	74.08
50	63.83
100	42.62

The results indicate a clear trend that dose rates increase as the measurement point approaches the storage well, consistent with exponential attenuation of gamma radiation. The shielding configuration, consisting of high-density concrete and lead, plays a crucial role in reducing radiation exposure. Additionally, the 50 cm air gap above the source contributes to dose attenuation by allowing radiation to disperse before encountering to the cover lid layers. However, the dose rate at 100 cm from the surface (42.62 µSv/h) exceeds the facility's operational limit of 15 µSv/h, highlighting the need for further shielding optimization. This suggests that while the current lead-concrete combination significantly reduces radiation exposure, additional shielding enhancements are necessary for practical implementation. One possible improvement is replacing the air gap with a solid concrete plug, which could further reduce radiation towards the surface. These findings underscore the importance of continuous evaluation and adaptive shielding modifications to ensure compliance with regulatory limits. Moreover, the results provide valuable insights for designing future storage improvements, reinforcing the necessity of regular monitoring and potential shielding upgrades.

3. Conclusions

The PHITS simulation results demonstrate that while the current shielding design of the cylindrical well storage significantly reduces radiation exposure, it does not yet ensure full compliance with the operational dose rate limit. The combination of 33 cm concrete and 5 cm lead provides substantial attenuation, however, the dose rate at 100 cm from the surface (42.62 µSv/h) exceeds the permissible limit of 15 µSv/h, indicating the need for additional shielding modifications. Further optimization is necessary to achieve regulatory compliance while maintaining a practical and cost-effective shielding design. Replacing the air gap with a solid concrete plug or incorporating additional shielding layers could significantly reduce dose rates. Additionally, continuous radiation monitoring, periodic reassessment of shielding performance, and potential material enhancements are essential to sustaining long-term radiation protection and ensuring the facility operates within safe exposure limits.

Acknowledgment

This research was supported by the 2025 Research Fund of KEPCO International Nuclear Graduate School (KINGS), the Republic of Korea.

REFERENCES

[1] J. A. E. Agency, "PHITS," [Online]. Available: Japan Atomic Energy Agency.

[2] T. G. o. t. R. o. Indonesia, "NATIONAL REPORT NATIONAL REPORT THE JOINT CONVENTION ON THE SAFETY OF SPENT FUEL MANAGEMENT AND ON THE SAFETY OF RADIOACTIVE WASTE MANAGEMENT," Draft, Jakarta, 2024.

[3] BAPETEN, BAPETEN Regulation No. 4 of 2013 Concerning Radiation Safety in the Utilization of Nuclear Energy, Jakarta: BAPETEN, 2013.

[4] RWMI, Directorate of Nuclear Facility and Management. BRIN. Target and Program
Occupational Safety and Health Management System (SMK3). Jakarta, 2022.

[5] Safety Analysis Report of Radioactive Waste Management Installation, Directorate of Nuclear Facility and Management. BRIN. Jakarta, 2022.

[6] Pacific Northwest National Laboratory (PNNL), the Compendium of Material Composition Data for Radiation Modeling, 200-DMAMC-128170, PNNL-15870, Rev. 2, April 2021.

[7] https://ionactive.co.uk/resource-hub/guidance/co-60-cobolt-60-radiation-safety-data

[8] https://www.ptb.de/cms/en/ptb/fachabteilungen/abt6/fb-62/621-dosimetry-for-radiotherapy/gamma-irradiation-facility/properties-of-co-60-radiation.html