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Monte Carlo Simulation of Interim Storage Containers for Dismantled Disused Sealed Radioactive Sources

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1. Introduction

Radioactive wastes consist of radioactive substances, materials, and equipment that have been exposed to or have become radioactive due to the operation of nuclear installations and can no longer be used [1]. One type of radioactive waste is Disused Sealed Radioactive Sources (DSRS) which are generated after radionuclide isotopes have been used for industrial purposes. Many countries maintain centralized radioactive waste management facilities for storing DSRS. The conditioning of DSRS is challenging due to the need to reduce volume and the limitations of interim storage facilities. Some countries such as the Philippines [2], Ghana [3] and Turkey [4], conducted DSRS conditioning in concrete drums. In Indonesia, concrete drum conditioning for DSRS is not yet employed; instead, an alternative method is utilized. This involves dismantling DSRS, placing the dismantled sources into capsules and stainless-steel drums, and storing them in well-type interim storage as shown in Fig. 1.

Fig. 1. Interim storage for high exposure radiation waste consisting of well-type and pool type storage [5].

In 2017, center of radioactive waste technology (CRWT) which is the facility responsible for operating the interim storage for radioactive waste in Indonesia conditioned 282 DSRS. These DSRS were dismantled from their original containers and placed in eight stainless steel capsules. This included 171 sources of ^{60}Co with a total activity of 0.32 Ci (placed in three capsules), 101 sources of ^{137}Cs with a total activity of 14.02 Ci (placed in four capsules), and 10 sources of ^{241}Am/Be (placed in one capsule) [6]. The distribution of activity in each capsule is detailed as shown in Table I.

Table I: Data on stainless-steel capsules from conditioning DSRS [6]

No.	Radionuclide	Number of sources	Activity (Ci)
1	^{60}Co	61	0.03
2	^{137}Cs	36	7.44
3	^{137}Cs	9	0.05
4	^{241}Am/Be	10	0.36
5	^{60}Co	80	0.21
6	^{60}Co	30	0.08
7	^{137}Cs	36	4.55
8	^{137}Cs	20	1.98

All capsules are stored in 60-liter containers stainless steel drums within interim storage for high exposure radiation waste. These facilities have well-type dimensions of 4 meters in depth, with a radius of 30 cm, concrete walls 10 cm thick, and a cover consisting of 33 cm of concrete and 5 cm of lead as shown in Fig. 2. In a previous study, this configuration was simulated using MicroShield 7.02 with specific data on the capsules, types of radionuclides and total activity, assuming a point source geometry [6]. The calculation showed a dose rate of 0.284 mR/h (≈2.84 µSv/h) on the surface cover of the well, which is below the interim storage dose limit of 25 µSv/h.

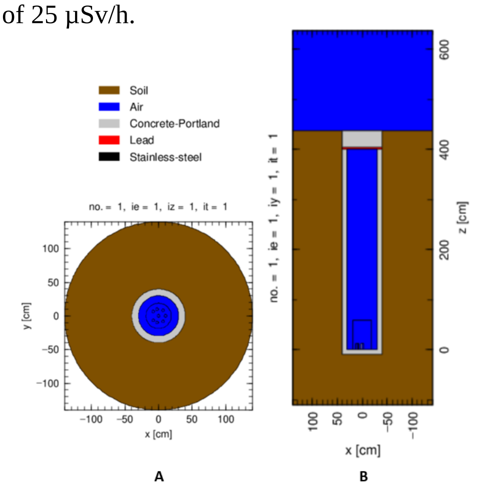

Fig. 2. Simulation of dimensions in well-type interim storage: (A) Cross-section along the XY-axis, (B) Cross-section along the XZ-axis.

Our main goal is to re-assess the calculations using Monte Carlo radiation simulation using the Particle and Heavy Ion Transport System (PHITS) code and enhance the safety in order to ensure that the total dose rate remains below the dose limit for interim storage. We used data from a previous study and assumptions such as the dimensions of the 60-liter stainless steel drum: height of 58.5 cm, radius of 19 cm, body thickness of 1 mm, lid thickness of 2 mm, and bottom thickness of 5 mm [7].

2. Methods and Results

We performed multiple simulations using the PHITS code, which is a versatile Monte Carlo radiation transport code that can simulate the behavior of a wide range of particle species, with energies reaching up to 1 TeV per nucleon for ions [8]. PHITS has been developed collaboratively by JAEA, RIST, KEK, and several other institutes. This code is essential for research in various fields, such as accelerator technology, radiotherapy, space radiation, and many other areas related to particle and heavy ion transport phenomena, including radiation shielding calculations [9]. The PHITS input code used for radiation shielding calculations in this study includes the following sections: title, parameters, source, material, material name color, surface, cell, volume, importance, t-track, and t-dchain, with a total of 1,000,000 particles for each run. Each specific data point for shielding material in this calculation is based on a reference [10].

2.1 Benchmarking with the previous study

Using the data above, we performed a simulation using the PHITS code. The distribution of the effective dose rate is shown in Fig. 3.

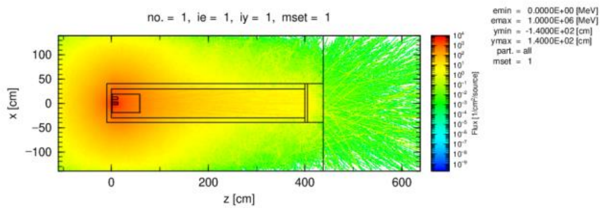

Fig. 3. Distribution of effective dose rates (mSv/h) for simulating all dismantled DSRS in eight capsules stored in well-type interim storage.

Our results show that the surface dose rate at 448 cm on the Z-axis is 0.0488 mSv/h (48.8 μSv/h), which exceeds the dose limit of 25 μSv/h, and at 1 meter from the surface (538 cm on the Z-axis), it is 0.022 mSv/h (22 μSv/h). This is higher than in the previous study, which reported a dose rate of 2.84 μSv/h. We investigated the contributing factors and found that the radiation dose from photons at the surface is 3.7 x 10^{-3} mSv/h (3.7 μSv/h) and at 1 meter from the surface, it is 1.77 x 10^{-3} mSv/h (1.77 μSv/h). This photon dose is of the same magnitude as the result from the previous study, which reported 2.84 μSv/h. However, the neutron dose at the surface is 0.045 mSv/h (45 μSv/h), and at 1 meter from the surface, it is 0.0202 mSv/h (20.2 μSv/h). Based on this fact, we found that the neutron dose significantly impacts the overall dose rate as shown Fig. 4.

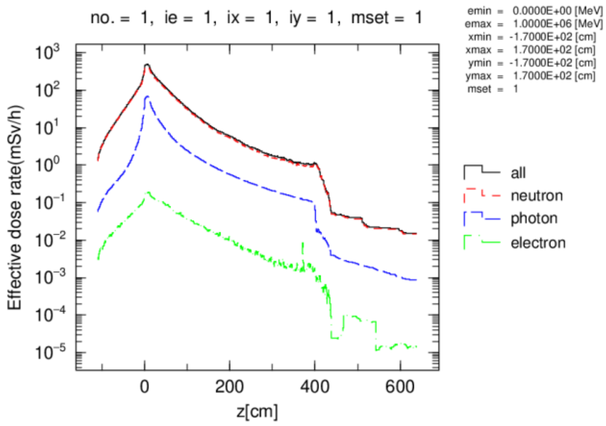

Fig. 4. Effective dose rate along the height from all radionuclides (^{60}Co, ^{137}Cs, and ^{241}Am/Be).

We found that the previous calculation did not include neutron radiation in the dose calculation [6]. The significant contributor to this was the neutron radiation from the ^{241}Am/Be sources. Based on this, we conducted two scenarios. First, we removed the ^{241}Am/Be source from interim storage and calculate the activation from the drum, and calculate the dose rate without ^{241}Am/Be. In the second scenario, we added additional shielding materials to reduce the contribution of neutron dose to the total dose.

2.2 Scenario calculation of dose rate without ^{241}Am/Be and activated container after removing ^{241}Am/Be

To ensure the contribution of ^{241}Am/Be to the total dose rate, we conducted a calculation excluding the ^{241}Am/Be sources to compare the differences with and without it. The results showed that, without the ^{241}Am/Be sources, the surface dose rate was 2.6 x 10^{-5} mSv/h (or 0.026 μSv/h), and at 1 meter from the surface, there was no detectable radiation. This level is significantly below the dose limit for the control area in this interim storage, which is 25 μSv/h. However, the activation of the drum container after removing the ^{241}Am/Be sources following 2 years of irradiation in interim storage must be considered.

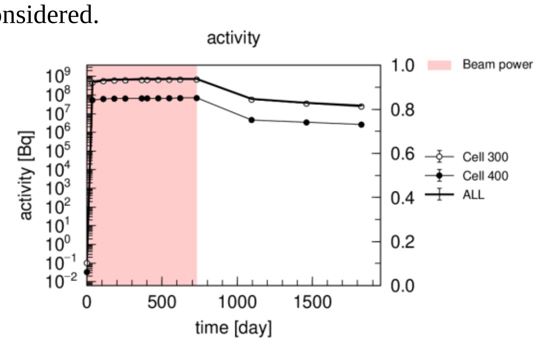

Fig. 5. Graph of activity (Bq) vs. time (days) for an activated container drum

Fig. 5 shows that the activated container drum (cell 300) and lid drum (cell 400) contains around 10^8 Bq after 2 years. Even after removing ^{241}Am/Be, the activated container drum (including the lid drum) contains between 10^7 and 10^8 Bq in the next 3 years.

2.3 Scenario with additional shielding materials

In the final scenario, we suggested an additional shielding with non-borated polyethylene placed inside the drum to cover all capsules. We conducted a sensitivity analysis with various thicknesses of non-borated polyethylene on the top side and a thickness of 3 cm on the sides to determine the optimal thickness that would keep the dose rate on the surface cover below the dose limit. As shown in Fig. 6, a thickness of non-borated polyethylene greater than 5 cm on the top capsule can reduce the total dose rate below the 25 µSv/h dose limit for the control area in interim storage.

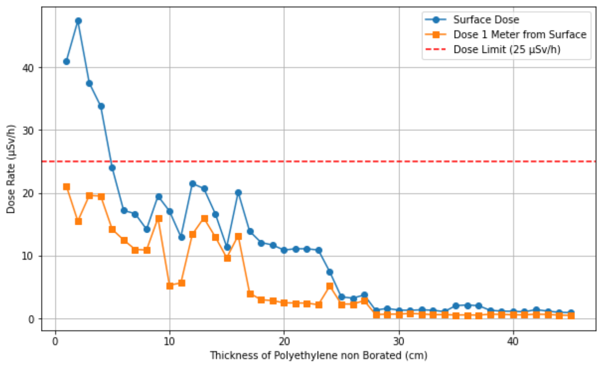

Fig. 6. Graph of dose rate versus thickness of non-borated polyethylene

3. Conclusions

Monte Carlo simulation to calculate the effectiveness of radiation shielding can be performed using the PHITS code. In this study, dismantled DSRS containing ^{60}Co, ^{137}Cs, and ^{241}Am/Be in well-type interim storage facility needs to consider the contribution of neutron dose from ^{241}Am/Be. Two scenarios can reduce the dose rate to comply with the dose limit in the control area of the interim storage. In the first scenario, the removal of the ^{241}Am/Be source must consider activated shielding materials during the irradiation time. In the other scenario, adding additional shielding materials such as non-borated polyethylene with a minimum thickness of 3 cm on the sides and 5 cm on the top capsule. Both scenarios can effectively reduce the total dose rate below the dose limit. In a real situation of 2019, facilities removed ^{241}Am/Be from well-type interim storage. The reuse process has been carried out for ^{241}Am/Be for industrial purposes [5].

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