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Original Article

Radiological safety assessment of lead shielded spent resin treatment facility with the treatment capacity of 1 ton/day

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ABSTRACT

The radiological safety of the spent resin treatment facility with a ^{14}C treatment capacity of 1 ton/day was evaluated in terms of the external and internal exposure of worker according to operation scenario. In terms of external dose, the annual dose for close work for 1 h/day at a distance of more than 1 m (19.8 mSv) satisfied the annual dose limit. For 8 h of close work per day, the annual dose exceeded the dose limit. For remote work of 2000 h/year, the annual dose was 14.4 mSv. Lead shielding was considered to reduce exposure dose, and the highest annual dose during close work for 1 h/day corresponded to 6.75 mSv. For close work of 2000 h/year and lead thickness exceeding 1.5 cm, the highest value of annual dose was derived as 13.2 mSv. In terms of internal exposure, the initial year dose was estimated to be 1.14E+03 mSv when conservatively 100% of the nuclides were assumed to leak. The allowable outflow rate was derived as 7.77E-02% and 2.00E-01% for the average limit of 20 mSv and the maximum limit of 50 mSv, respectively, where the annual replacement of the worker was required for 50 mSv.

1. Introduction

In heavy-water reactors, ion-exchange resins are used to purify liquid radioactive waste that is generated during operation. The resins are used in moderator deuteration and de-deuteration systems, liquid waste handling systems, and shield cooling systems, among others [1]. Subsequently, the spent resin is stored in a resin tank, as shown in Fig. 1 [2]. The spent resin with volume fraction of 80%, the zeolite with volume fraction of 10%, and activated carbon are stored in the tank. The stored spent resin should be handled based on future decommissioning plans [3–6].

According to the Nuclear Safety Act in Korea, intermediate-level radioactive waste (ILW) cannot be disposed of in caves. This is because it is not approved for cave disposal, where the concentration of ^{14}C in the resin exceeds its tolerable limit (2.22E+05 Bq/g) for near-surface disposal of low-level radioactive waste (LLW). The spent resin from the heavy-water reactors contains various radionuclides (including ^{3}H, ^{14}C, ^{60}Co, and ^{137}Cs) that are classified as ILW with a high concentration of ^{14}C (8.06E+06 Bq/g), which has a half-life of 5730 years [7–14]. Therefore, spent resin treatment is necessary to reduce the radioactivity concentration of ^{14}C below the LLW disposal criteria. Treatment of spent resin typically results in secondary waste and can be inefficient in terms of the disposal costs. Currently, in order to eliminate these shortcomings, a ^{14}C treatment facility with a high removal efficiency is being developed based on the previous spent resin treatment device with a lab scale of 1 kg/day [2]. This would convert ^{14}C from spent resin into gaseous form at a treatment capacity of 1 ton/day and is expected to lead to commercial recycling of ^{14}C from the spent resin as a labeled compound of radioisotopes [15].

In order to determine the radiological safety of radiation workers for the use of 1 ton/day commercial spent resin at the treatment facility, VISIPLAN, which is the exposure dose assessment code, was used as an as low as reasonably achievable (ALARA) planning tool for nuclear facilities. It had been developed in 1999 at the SCK-CEN laboratory in Belgium [16]. The external dose assessment of workers was conducted using the VISIPLAN code based on the 3D radiation environment modeling according to the workers' location and situation. In addition, an internal dose assessment was conducted using the inhalation dose conversion factor in international commission on radiological protection (ICRP) 119 [17]. Only the inhalation dose conversion factor was considered in this study because it is expected that workers will not ingest radioactive material even if an accident occurs in the treatment facility.

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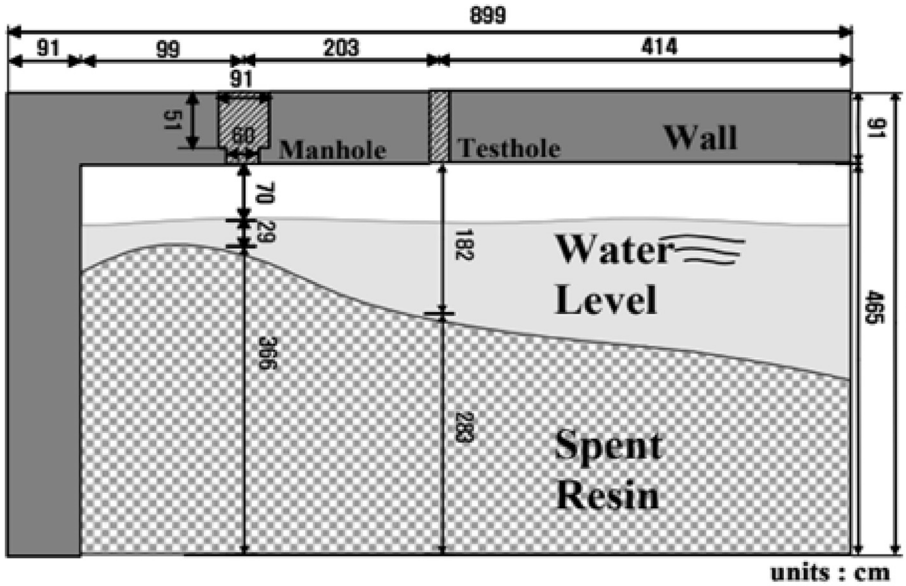

Fig. 1. Structure of resin storage tank and composition from heavy water reactor.

2. Modeling and source term of 1 ton/day spent resin treatment facility

2.1. Modeling of 1 ton/day spent resin treatment facility

As shown in Fig. 2, the 1 ton/day spent resin treatment facility has been modeled using the VISIPLAN code. The spent resin mixtures generated from the heavy water reactors are separated into spent resin, activated carbon, and zeolite through the spent resin mixture separator to ensure disposal safety. The spent resin is then subjected to a desorption process using microwaves, which are a form of dry decontamination. The separated zeolite and activated carbon enter the storage tank, and the separated spent resin containing ^{14}C flows into the microwave reactor through the spent resin feed hopper. In the microwave reactor, the desorption of ^{14}C occurs, and the generated CO_2 gas is then adsorbed to the adsorption tower in the form of gas through the condensate water tank. The condensate water tank collects moisture and ions by condensing the steam generated from the spent resin in a microwave reactor. The dried spent resin from which ^{14}C is desorbed, enters the spent resin storage tank located under the microwave reactor.

Lead shielding was assumed in the facility to reduce the exposure dose of workers. Lead shielding was applied to the parts of the

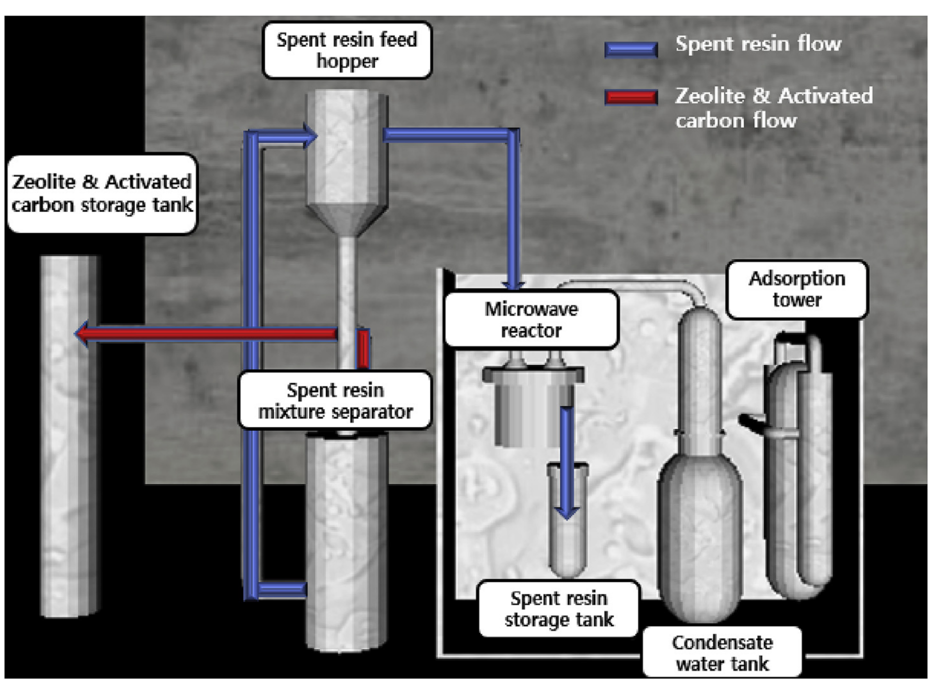

Fig. 2. 3D modeling of a 1 ton/day spent resin treatment facility using VISIPLAN.

Table 1
Radioactivity concentration of the spent resin mixture.

Nuclides	Zeolite (Bq/g)	Activated carbon (Bq/g)	Spent resin (Bq/g)
^{3}H	8.55E+03	1.56E+04	3.30E+04
^{14}C	1.98E+02	2.22E+03	1.54E+05
^{57}Co	0	0	2.05E+01
^{60}Co	4.98E+01	1.52E+02	3.82E+02
^{51}Cr	0	0	2.05E+02
^{134}Cs	2.39E+01	1.80E+00	1.33E+01
^{137}Cs	3.22E+04	1.63E+03	1.16E+04
^{54}Mn	0	0	1.60E+01
^{95}Nb	2.89E-01	5.92E+00	3.67E+01
^{125}Sb	0	9.90E+00	2.80E+02
^{95}Zr	0	0	2.68E+01
^{152}Eu	0	0	4.44E+02
^{154}Eu	0	0	3.48E+01

Table 2
Radioactivity of the spent resin mixture (100 kg of zeolite and activated carbon and 400 kg of spent resin).

Nuclides	Zeolite (Bq)	Activated carbon (Bq)	Spent resin (Bq)
^{3}H	8.55E+08	1.56E+09	1.32E+10
^{14}C	1.98E+07	2.22E+08	6.16E+10
^{57}Co	0	0	8.20E+06
^{60}Co	4.98E+06	1.52E+07	1.53E+08
^{51}Cr	0	0	8.19E+07
^{134}Cs	2.39E+06	1.80E+05	5.31E+06
^{137}Cs	3.22E+09	1.63E+08	4.66E+09
^{54}Mn	0	0	6.41E+06
^{95}Nb	2.89E+04	5.92E+05	1.47E+07
^{125}Sb	0	9.90E+05	1.12E+08
^{95}Zr	0	0	1.07E+07
^{152}Eu	0	0	1.78E+08
^{154}Eu	0	0	1.39E+07

zeolite and activated carbon storage tank, zeolite and activated carbon separator, spent resin feed hopper, microwave reactor, and spent resin storage tank where the source term was located. The adsorption tower was not considered. This is because it is shielded by the device itself because of the presence of a beta-emitting nuclide, ^{14}C. The thickness of the lead shield increased from 0.5 cm to 2 cm in increments of 0.5 cm. For close and remote work, the change in the exposure dose of workers and spatial dose distribution was derived.

2.2. Source term of 1 ton/day spent resin treatment facility

The radioactivity of radionuclides in the spent resin mixture was determined to evaluate the exposure dose of the spent resin treatment facility. As shown in Table 1, the spent resin mixture was sampled from storage tank #2 of Wolseong Unit 1 to determine the radioactivity concentration value of each nuclide. The radioactivity values corresponding to the most existing nuclides in the spent resin mixture (100 kg of zeolite and activated carbon and 400 kg of spent resin) in the facility during operation are listed in Table 2. The VISIPLAN code was used to set the source term values inside the spent resin treatment facility. The location of the source term was assumed to be within the spent resin mixture separator (100 kg of spent resin, 12.5 kg of zeolite, and activated carbon), separated zeolite and activated carbon storage tank (87.5 kg of zeolite and activated carbon), spent resin feed hopper (100 kg of spent resin), microwave reactor (100 kg of spent resin), and the spent resin storage tank (100 kg of spent resin). The workers' external dose assessment was conducted based on the location of the source term.

3. Method and tools

3.1. Assessment of the workers' external dose due to the spent resin treatment facility

The VISIPLAN code calculated the external dose based on the point kernel method as shown in equations (1) and (2) [16].

$$\varphi = \int \frac{S \cdot B \cdot e^{-\mu}}{4\pi r^2} dV \quad (1)$$

φ: Photon flux [$\# \cdot m^{-2} \cdot s^{-1}$]
S: Source strength per unit volume [$\# \cdot s^{-1} \cdot m^{-3}$]
B: Build-up factor
μ: Attenuation effectiveness coefficient [= linear attenuation coefficient (1/cm) multiplied by thickness of absorber (cm)]
r : Distance from point source [m]
V : Volume [m^3]

Each small point source was termed as a kernel and an integration process in which contributions to the dose of each point were combined, was termed as each small "point kernel" integration. Based on the photon flux at each point, the effective dose rate was calculated based on the dose conversion factor selected in the calculation.

$$\dot{H} = \sum_i C_i \cdot \varphi_i \quad (2)$$

$\dot{H}$: Effective dose rate (Sv/s)
C_i: Dose conversion factor for the photon energy of radionuclide, i [$Sv/(\# \cdot m^{-2})$]
φ_i: Photon flux for photon energy of radionuclide, i [$\# \cdot m^{-2} \cdot s^{-1}$].

3.2. Assessment of the workers' internal dose due to the spent resin treatment facility

In the case of internal exposure due to the spent resin treatment facility, it is deemed that it does not occur during normal operation. However, while the probability is very low even during normal operation, internal exposure can occur due to the release of the nuclides. The internal dose assessments were conservatively conducted when 100% of the source term was leaked from the facility. The allowable outflow rate of the facility was determined when compared to the annual dose limit (100 mSv for 5 years without exceeding 50 mSv/year) [18]. The allowable outflow rate is the rate at which the worker satisfies the radiological safety criteria by assuming that radionuclides are leaked from the facility, as shown in equation (3). The annual dose limit for workers corresponds to 20 mSv or less on average, although it can range up to 50 mSv. Therefore, an acceptable outflow rate was derived for both values.

$$L = \frac{(ALD - ED)}{\sum_i A_i \times C_i} \quad (3)$$

L: Outflow rate of the spent resin treatment facility [%]

ALD: Annual dose limit of worker [mSv]
ED: External dose [mSv]
A_i: Radioactivity of radionuclide i [Bq]
C_i: Dose conversion factor of radionuclide i [mSv/Bq]

The inhalation dose conversion factor of ICRP publication 119 (Compendium of Dose Coefficients based on ICRP Publication 60) and the average breathing rate of 1.2 m^3/h for adult workers were considered [19]. In addition, 100% of the radioactivity was assumed to enter the body through breathing. Further, based on the APF value of 10 for the basic Half mask/Dust mask of OSHA 3352–02 (Assigned Protection Factors for the Revised Respiratory Protection Standard, 2009), the effective dose due to breathing was derived. Based on these factors, the workers' internal dose was calculated, as shown in Equation (4) [20,21].

$$\mathrm{H(inh)} = \sum_i \{A_i \times C_i\} \times BR \times HW \times APF \tag{4}$$

H(inh): Committed effective dose for inhalation [mSv]
A_i: Air concentration of nuclide i [Bq/m^3]
C_i: Dose conversion factor of nuclide i [mSv/Bq]
BR: Breathing rate [1.20 m^3/h]
HW: Exposure time in workspace [h]
APF: Assigned Protection Factors [0.1]

4. Worker scenario for the spent resin treatment facility

4.1. Close work for the spent resin treatment facility

Dose assessment was conducted according to the distance from the center of the facility based on the source term and modeling technique, as presented in Fig. 3. The working time for the spent resin treatment facility was assessed on an hourly basis, and the error of the working time was set to 10 min.

4.2. Remote work for the spent resin treatment facility

Remote work was assumed to reduce the exposure dose. The remote work room was assumed to be based on a domestic container (iron of 1.2 mm thickness) and was at a distance of 5 m from the facility. It was assumed that the worker was remotely operating and monitoring the facility inside the remote work room. The annual dose was derived for 2000 h/year by assuming 8 h of work per day. The worker was assumed to work for 5 days in a week and 50 weeks in a year. The scenario for remote work was assumed to be monitored by the worker in the remote work room for 8 h/day.

5. Results and discussion

5.1. Assessment of the distribution of exposure dose in space due to the operation of the spent resin treatment facility

As shown in Fig. 4, the dose distribution in space due to the operation of the spent resin treatment facility was derived. The dose level of the storage tank, where the separated zeolite and

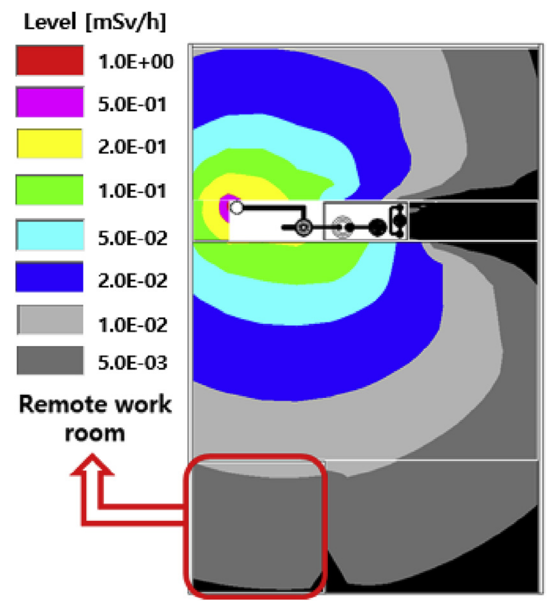

Fig. 4. Distribution of the exposure dose due to the operation of the spent resin treatment facility (Z-view).

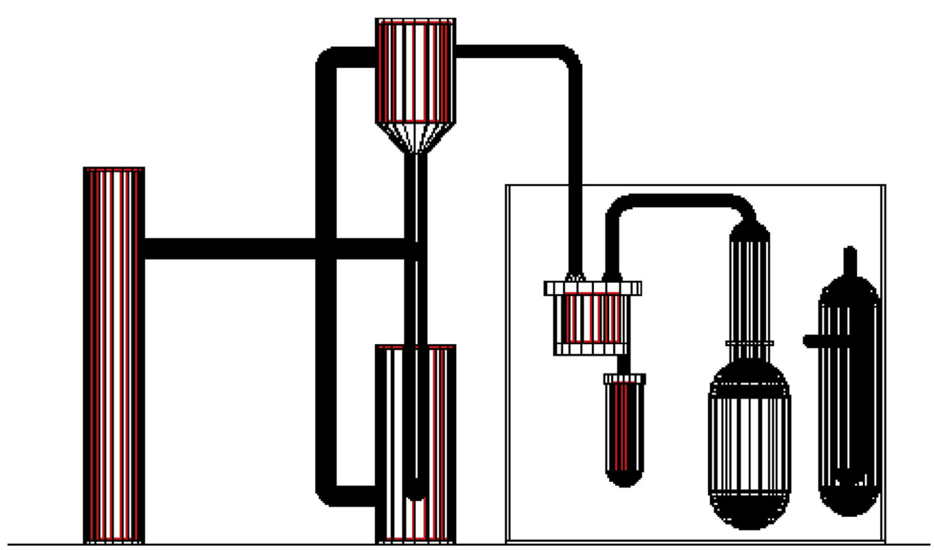

Fig. 3. Location of the source term inside the 1 ton/day spent resin treatment facility.

activated carbon were stored, was determined to be 170–220% higher than in other locations of the source terms. The dose rate within approximately 30 cm from the storage tank was derived to vary from 0.11 mSv/h to 1.2 mSv/h. In addition, it was confirmed that the dose level decreased when the distance from the spent resin treatment facility increased.

5.2. External dose assessment based on work scenario

5.2.1. External dose assessment of workers based on the location in close range

We assumed that the workers were working at a distance of approximately (20 cm–2 m) from the spent resin treatment facility, and location-specific dose assessments were conducted based on the distance from the facility. The results are summarized in Table 3. The assessment was performed on a unit at a distance of 20 cm from the surface of the center of the facility. The dose rate at the nearest 20 cm was derived to be 1.60E-01 mSv/h. The conversion of this dose rate to the annual dose for 2000 h/year indicated that the annual dose limit for workers was not satisfactory. In the case of 1 h of work per day at a distance of more than 100 cm from the facility, the results indicated that the workers' annual dose limit was satisfied. The annual dose at a distance of 100 cm from the facility, which is the starting point for satisfying the annual dose limit, was derived as 19.8 mSv.

5.2.2. External dose assessment of workers in the case of remote work

As shown in Fig. 4, the remote work room on the left side was 5 m away from the facility. The range of the dose level in the remote work room was 4.4E-03 mSv/h ~1.1E-02 mSv/h. As shown in Table 4, the external dose rate in the remote work room corresponds to 7.2E-03 mSv/h. The evaluation of the annual dose based on the corresponding dose rate resulted in 1.8E+00 mSv for 250 h/year and 1.44E+01 mSv for 2000 h/year. Hence, work conducted in a remote work room was confirmed to satisfy the annual dose limit of workers.

5.2.3. External dose assessment of workers in the case of lead shielding on the facility

To satisfy the annual dose limit of the workers conservatively by decreasing their exposure dose, dose assessment was conducted by additionally considering lead shielding in the spent resin treatment facility. The distribution of exposure dose from the spent resin treatment facility when the lead thickness increases to 0.5, 1.0, 1.5, and 2.0 cm, additionally including the case of no lead shielding is shown in Fig. 5. As the thickness of lead increased, the results indicated that the range of dose distributions decreased and the value of the external dose also gradually decreased. The results of dose assessment in close work for the facility with lead shielding are summarized in Tables 5 and 6, and the results of dose assessment in remote work are summarized in Table 7.

Table 4
Result of workers' external dose assessment during remote work.

Dose rate (mSv/h)	7.2E-03
Annual dose - 250 h (mSv)	1.8E+00
Annual dose - 2000 h (mSv)	1.44E+01

As shown in Tables 5 and 6, for close work for 250 h/year, the highest dose value was 6.75 mSv and the lowest value was 0.45 mSv, which satisfied the annual dose limit. For 2000 h/year, it was confirmed that the annual dose limit of the workers was satisfied irrespective of the distance from the lead of thickness at least 1.5 cm. As shown in Table 7, for remote work, a dose range of 5.25E-01–1.10E-01 mSv was derived for 250 h/year and a dose range of 4.20–8.80E-01 mSv for 2000 h/year. The results of the external dose assessment based on the thickness of the lead shield are graphically shown in Figs. 6 and 7. Based on these results, it is necessary to determine the optimal thickness of the lead in the future for the design and manufacturing of the actual facility.

5.3. Assessment of the ratio of nuclides affecting workers with change in lead thickness

With respect to the nuclides listed in Table 2, the ratio of nuclides affecting the workers was evaluated. As summarized in Table 8, the percentage of the effect of each nuclide on the worker inside the remote room was determined when the lead thickness gradually increased, including that when lead shielding was not present. Among the 11 nuclides that affected the external exposure, it was found that the ^{60}Co, ^{137}Cs, and ^{152}Eu nuclides affected the dose rate of workers because ^{60}Co, ^{137}Cs, and ^{152}Eu were the most abundant in the facility and had the highest radioactivity concentration. The contribution rates for the three nuclides are shown in Fig. 8. In the absence of lead shielding, ^{137}Cs exhibited the greatest influence on workers at 86.2%, ^{60}Co exhibited the second largest effect at 7.44%, and ^{152}Eu exhibited an influence of 4.97%. When the thickness of the lead shield increased, the effect ratio of ^{137}Cs gradually decreased, whereas that of ^{60}Co increased. This can be attributed to the difference in the attenuation coefficient of gamma rays due to the considered lead shielding wherein ^{137}Cs (0.662 MeV) with a relatively lesser emission energy than ^{60}Co (1.17, 1.33 MeV) was observed to decrease the relative effect when the thickness of lead shield increased.

Table 3
Results of the external dose assessment during close work for the spent resin treatment facility.

Distance (cm)	Dose rate (mSv/h)	Annual dose - 250 h (mSv)	Annual dose - 2000 h (mSv)
20	1.60E-01	4.00E+01	3.20E+02
40	1.20E-01	3.00E+01	2.40E+02
60	1.00E-01	2.50E+01	2.00E+02
80	9.00E-02	2.25E+01	1.80E+02
100	7.90E-02	1.98E+01	1.58E+02
120	6.70E-02	1.68E+01	1.34E+02
140	5.80E-02	1.45E+01	1.16E+02
160	5.10E-02	1.28E+01	1.02E+02
180	4.40E-02	1.10E+01	8.80E+01
200	3.90E-02	9.75E+00	7.80E+01

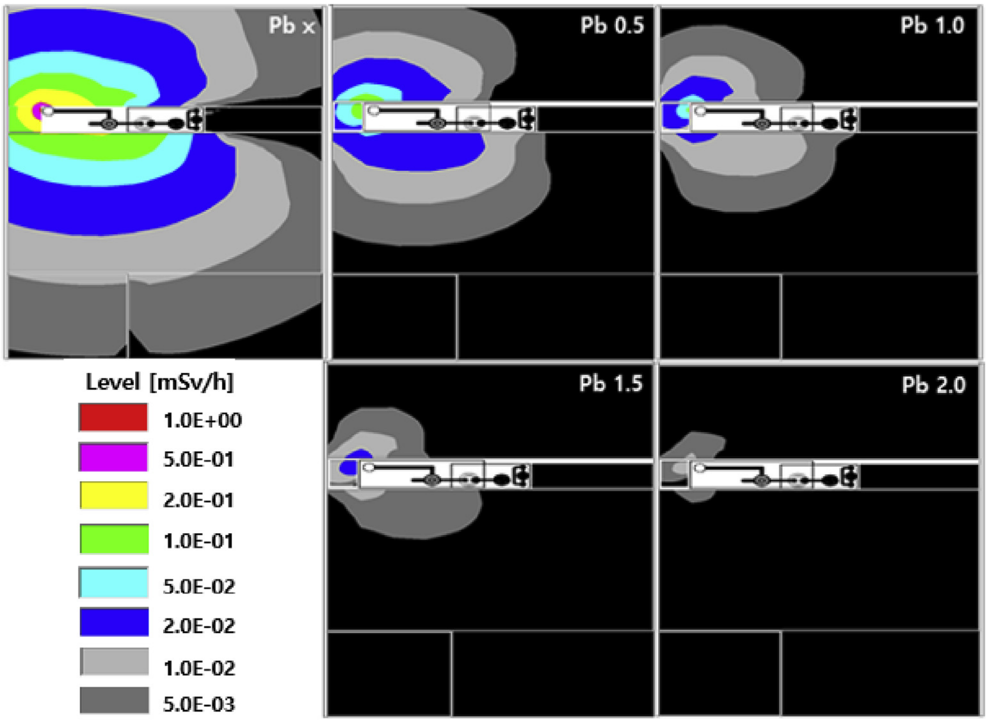

Fig. 5. Distribution change of the exposure dose from the spent resin treatment facility based on the thickness of lead shield (Z-view).

Table 5
Annual dose (mSv) assessment based on the thickness of the lead shield during close work (250 h).

Distance (cm)	Pb 0.5 cm	Pb 1.0 cm	Pb 1.5 cm	Pb 2.0 cm
20	6.75E+00	3.25E+00	1.65E+00	9.75E-01
40	5.75E+00	3.25E+00	1.85E+00	1.13E+00
60	5.00E+00	2.75E+00	1.55E+00	9.50E-01
80	4.75E+00	2.45E+00	1.40E+00	8.25E-01
100	4.25E+00	2.25E+00	1.23E+00	7.25E-01
120	3.75E+00	2.00E+00	1.10E+00	6.50E-01
140	3.50E+00	1.80E+00	9.75E-01	5.75E-01
160	3.00E+00	1.63E+00	9.00E-01	5.25E-01
180	2.75E+00	1.48E+00	8.25E-01	4.75E-01
200	2.45E+00	1.35E+00	7.50E-01	4.50E-01

Table 6
Annual dose (mSv) assessment based on the thickness of the lead shield during close work (2000 h).

Distance (cm)	Pb 0.5 cm	Pb 1.0 cm	Pb 1.5 cm	Pb 2.0 cm
20	5.40E+01	2.60E+01	1.32E+01	7.80E+00
40	4.60E+01	2.60E+01	1.48E+01	9.00E+00
60	4.00E+01	2.20E+01	1.24E+01	7.60E+00
80	3.80E+01	1.96E+01	1.12E+01	6.60E+00
100	3.40E+01	1.80E+01	9.80E+00	5.80E+00
120	3.00E+01	1.60E+01	8.80E+00	5.20E+00
140	2.80E+01	1.44E+01	7.80E+00	4.60E+00
160	2.40E+01	1.30E+01	7.20E+00	4.20E+00
180	2.20E+01	1.18E+01	6.60E+00	3.80E+00
200	1.96E+01	1.08E+01	6.00E+00	3.60E+00

5.4. Assessment of the allowable outflow rate of the facility and internal dose of workers

As shown in Table 9, the committed effective dose equivalent for 50 years was derived when 100% of the radionuclides were conservatively leaked from the facility. The committed effective dose of the worker corresponds to 2.46E+04 mSv, as shown in equation (4). ^{137}Cs exhibited a value of 1.83E+04 mSv, which is the most influential value for the worker, and ^{152}Eu and ^{60}Co showed 4.07E+03 mSv and 1.52E+03 mSv, respectively as the second and third contributors. As shown in Table 10, the values of the internal dose of workers over the initial year were derived. The value of the internal dose of workers for the first year corresponded to 1.14E+03 mSv. In addition, the allowable outflow rates of the facilities were derived using equation (3). The workers' dose limit was 100 mSv for five years, the maximum value of one year corresponded to 50 mSv, and the outflow rate in the case with and without the change of workers for a year was derived for the criterion of 50 mSv/y and 20 mSv/y, respectively. Using the numerical value of 0.45 mSv for the external dose in equation (3) based on the lead shield with a thickness of 2.0 cm, the allowable internal doses were 49.55 mSv and 19.55 mSv, respectively, for close work of 1 h/day. The allowable outflow rates corresponding to 2.00E-01% and 7.77E-02% were derived by dividing by the effective dose value of 2.46E+04 mSv. This indicates that the allowable outflow rate of the facility should be within 2.00E-01% and 7.77E-02% for the cases with and without the replacement of workers each year, respectively, during normal operation. With respect to remote work, remote operation and monitoring will be performed in a space separate from the treatment facility; hence, internal exposure due to inhalation will not occur even if radionuclides are leaked from

Table 7
Annual dose assessment based on the thickness of the lead shield during remote work.

Distance (cm)	Pb 0.5 cm	Pb 1.0 cm	Pb 1.5 cm	Pb 2.0 cm
Annual dose - 250 h (mSv)	5.25E-01	3.00E-01	1.80E-01	1.10E-01
Annual dose - 2000 h (mSv)	4.2E+00	2.4E+00	1.44E+00	8.80E-01

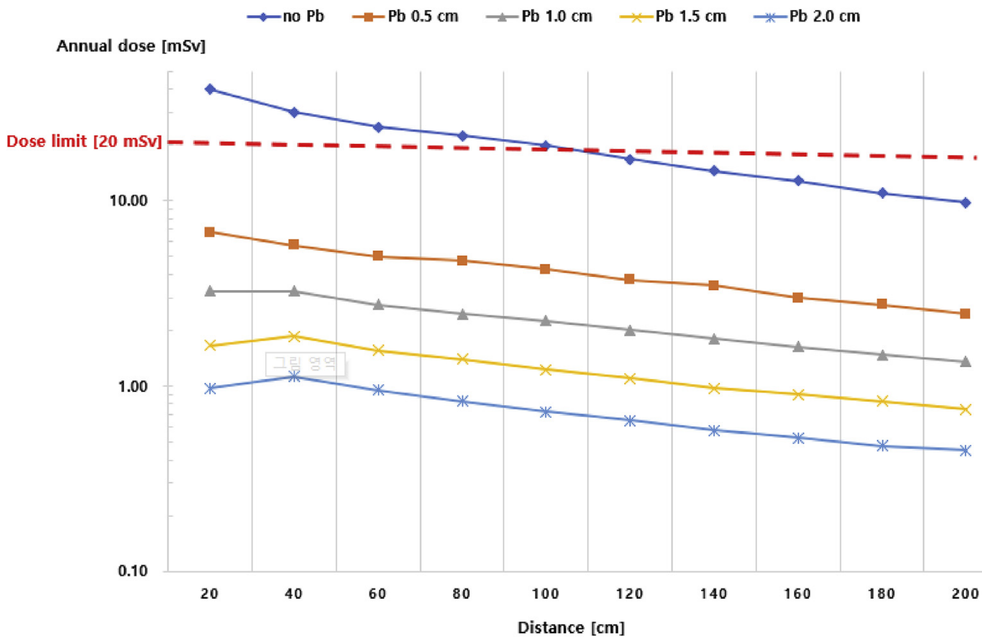

Fig. 6. Changes in the worker exposure dose for close work (250 h) based on the thickness of the lead shield.

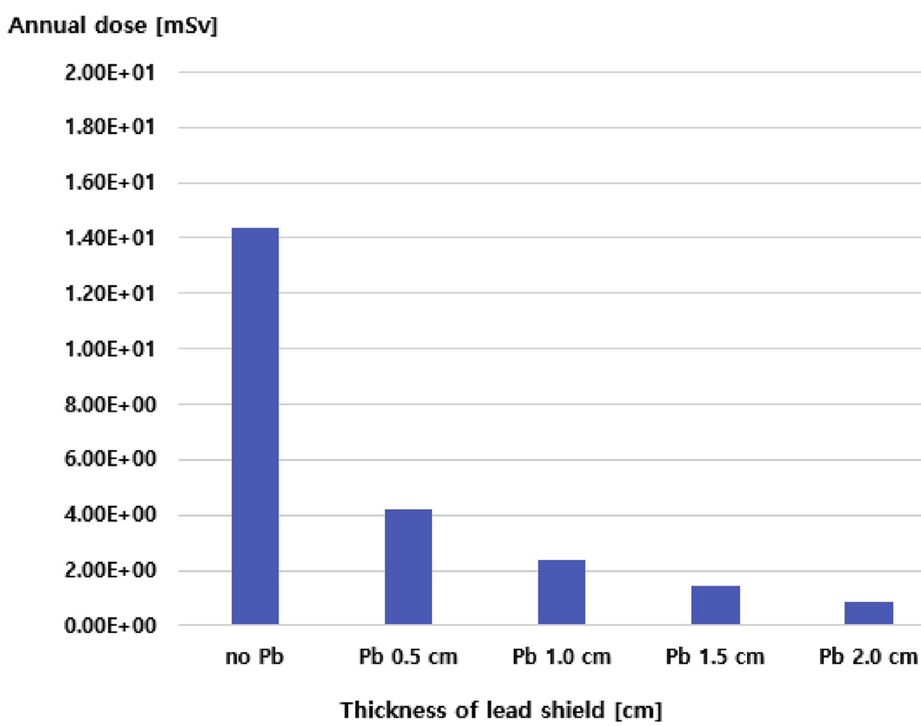

Fig. 7. Changes in the worker exposure dose for remote work (2000 h) based on the thickness of the lead shield.

the treatment facility.

6. Conclusion

The radiological safety assessment of workers during the operation of a 1 ton/day spent resin treatment facility in several scenarios was conducted. It was confirmed that the annual dose limit of workers was not satisfied for 8 h of close work per day. Close work for 1 h per day was confirmed to satisfy the annual dose limit at a distance of 1 m or more. Based on the results, it is expected that if close work is conducted, it is necessary to work 1 h/day at a distance of more than 1 m. With respect to remote work,

Table 8
Percentage of the effect of nuclides on the worker inside the remote room based on the thickness of the lead shield (%).

Radionuclide	no Pb	Pb 0.5 cm	Pb 1.0 cm	Pb 1.5 cm	Pb 2.0 cm
^{57}Co	8.17E-04	9.29E-05	8.59E-05	7.54E-05	6.56E-05
^{60}Co	7.44E+00	1.26E+01	1.67E+01	2.17E+01	2.79E+01
^{51}Cr	3.42E-01	2.37E-03	4.92E-04	1.00E-04	1.78E-05
^{134}Cs	2.35E-01	2.63E-01	2.82E-01	3.24E-01	3.21E-01
^{137}Cs	8.62E+01	8.22E+01	7.71E+01	7.00E+01	6.29E+01
^{54}Mn	6.10E-02	8.71E-02	9.78E-02	1.33E-01	1.08E-01
^{95}Nb	1.77E-01	2.21E-01	2.50E-01	2.92E-01	2.64E-01
^{125}Sb	6.32E-01	3.75E-01	3.11E-01	2.94E-01	2.08E-01
^{95}Zr	1.37E-01	1.44E-01	1.63E-01	1.82E-01	1.75E-01
^{152}Eu	4.97E+00	3.97E+00	4.98E+00	6.37E+00	7.81E+00
^{154}Eu	2.87E-01	4.18E-01	5.34E-01	6.83E-01	8.12E-01

the annual dose limit would be satisfied irrespective of the working time. In addition, as the workers work in a separate space, the inhalation of radionuclides will not occur. Therefore, in terms of the radiological safety of the worker, it is deemed that remote work is more appropriate. In addition, an external dose assessment was conducted considering lead shielding for the facility to reduce the exposure dose of workers. The annual dose limit was satisfied irrespective of the distance in the lead shield with a thickness of more than 1.5 cm for 8 h of close work and more than 0.5 cm for 1 h close work. Even for remote work, increased thickness of lead shield led to a reduction in the exposure dose. Therefore, it is expected that it is necessary to derive the optimal lead thickness in the actual manufacturing of the facility. In addition, the radionuclides that maximally affected the dose of the workers corresponded to ^{137}Cs, ^{60}Co, and ^{152}Eu. When the thickness of lead increased, the ratio of the influence of ^{137}Cs decreased gradually and ^{60}Co tended to increase. The allowable outflow rate of the spent resin treatment facility was evaluated for a value of 50 mSv, where the worker was changed every year, and for a value of 20 mSv, where the worker was not changed. Hence, the facility will be maintained below the allowable outflow rate to ensure the radiological safety of the worker.

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

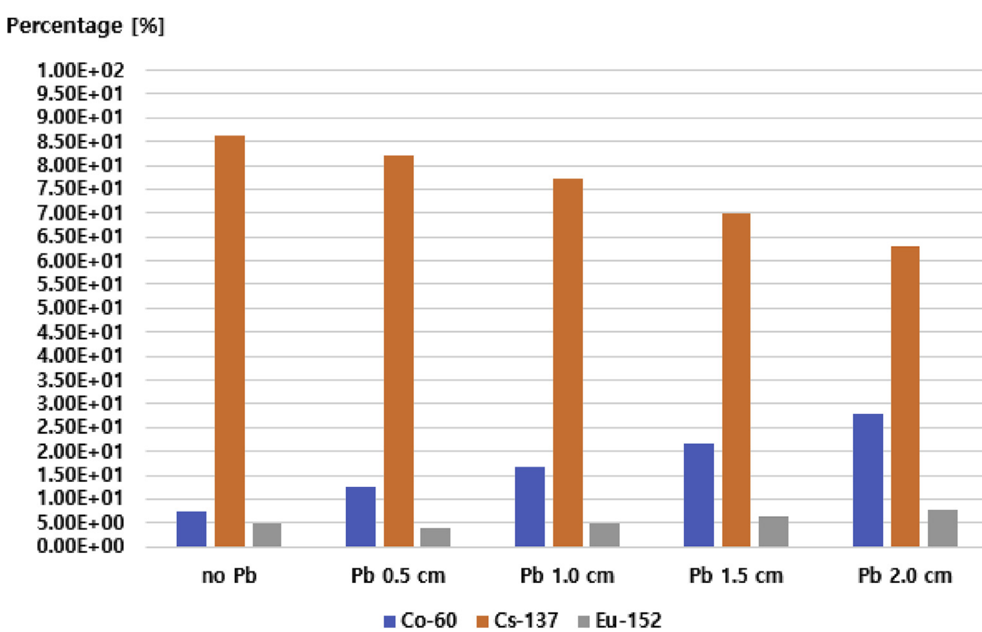

Fig. 8. Change in ^{60}Co, ^{137}Cs, and ^{152}Eu influence ratios with thickness of lead shield.

Table 9
Assessment of workers' internal dose for the spent resin treatment facility.

Radionuclide	Radioactivity (Bq)	Inhalation dose conversion factor (mSv/Bq)	Inhalation internal exposure dose (mSv)
^{3}H	1.56E+09	4.10E-08	3.75E+01
^{14}C	6.18E+09	6.50E-09	2.35E+01
^{57}Co	8.20E+06	9.40E-07	4.52E+00
^{60}Co	1.53E+08	1.70E-05	1.52E+03
^{51}Cr	8.19E+07	3.60E-08	1.73E+00
^{134}Cs	5.31E+06	9.60E-06	2.99E+01
^{137}Cs	4.66E+09	6.70E-06	1.83E+04
^{54}Mn	6.41E+06	1.20E-06	4.51E+00
^{95}Nb	1.47E+07	1.60E-06	1.38E+01
^{125}Sb	1.12E+08	3.30E-06	2.17E+02
^{95}Zr	1.07E+07	2.50E-06	1.57E+01
^{152}Eu	1.78E+08	3.90E-05	4.07E+03
^{154}Eu	1.39E+07	5.00E-05	4.07E+02
Total exposure dose	–	–	2.46E+04

* Inhalation internal exposure dose is calculated by considering the space volume, APF value, and breathing rate.

Table 10
Workers internal dose for the operation of the spent resin treatment facility in the first year.

Radionuclide	Half life (year)	Inhalation internal exposure dose for first year (mSv)
^{3}H	1.23E+01	2.18E+00
^{14}C	5.73E+03	4.72E-01
^{57}Co	7.45E-01	2.74E+00
^{60}Co	5.26E+00	1.88E+02
^{51}Cr	7.59E-02	1.73E+00
^{134}Cs	2.06E+00	8.53E+00
^{137}Cs	3.02E+01	6.08E+02
^{54}Mn	8.55E-01	2.50E+00
^{95}Nb	9.59E-02	1.38E+01
^{125}Sb	2.76E+00	4.81E+01
^{95}Zr	1.75E-01	1.54E+01
^{152}Eu	1.35E+01	2.20E+02
^{154}Eu	8.59E+00	3.21E+01
Total exposure dose	–	1.14E+03

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