Original Article

# Radiological safety evaluation of dismantled radioactive concrete from Kori Unit 1 in the disposal and recycling process

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A B S T R A C T

For evaluating the radiological safety of dismantled concrete, the process of disposal and recycling of the radioactive concrete generated during the dismantling of Kori Unit 1 is analyzed. Four scenarios are derived based on the analysis of the concrete recycling and disposal process, and the potential exposure to the workers and public during this process are calculated. VISIPLAN and RESRAD code are used for evaluating the dosages received by the workers and public in the following four scenarios: concrete inspection, transport of concrete by the truck driver, driving on a recycled concrete road, and public living near the landfilled concrete waste. Two worker exposure scenarios in the processing of concrete and two public exposure scenarios in recycling and disposal are considered; in all the scenarios, the exposure dose does not exceed the annual dose limit for each representative.

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## 1. Introduction

The Kori Unit 1 in South Korea was operational from 1978 to 2017. After 40 years of operation, Korea hydro and nuclear power (KHNP) decided not to apply for the life extension of this unit. Although Korean research reactors, such as Korea research reactors-1and 2 (KRR-1 and 2), have been decommissioned [1—3], Kori Unit 1 is the first case of decommissioning a commercial Korean NPP. Kori Unit 1 was commissioned in 1978 with an operating license terminating in June 2017. The experience gained from decommissioning Korean research reactors cannot be directly applied to Kori Unit 1 because research and commercial reactors require different approaches for decommissioning due to differences in the reactor power, design, and activation level. Therefore, considerable preparation is necessary for decommissioning Kori Unit 1.

Some of the concrete, which is the main building material for nuclear facilities, is contaminated or activated during the operation and maintenance of a nuclear power plant [4]. Concrete is an important radioactive waste generated along with metal waste, when dismantling nuclear facilities. A large quantity of radioactive concrete is generated during decommissioning, accounting for 15%

of the weight of the total radioactive waste for 900—1300 MWe PWR [5]. Radioactive concrete is dismantled in a strictly controlled environment and processed in various ways. It undergoes different processes including waste processing to recycling the less contaminated parts or disposal to the waste facility as low level contaminated waste. According to the NUREG-1640 report by NRC, there are eight scenarios, where workers or the public can be exposed, from dismantling to disposal [6]. In the concrete disposal process, radiological safety depends on the applied regulations and reduction techniques. Approximately 45,000 tons of radioactive concrete waste were expected to be shipped during the dismantling of the Maine Yankee nuclear power plant (860 MWe, PWR); however, 284 tons, other than the primary shield, were generated during the dismantling of the Trojan nuclear power plant (1130 MWe, PWR) [7,8]. This is due to the regulations and reduction methods adopted for the radioactive concrete waste.

Existing facilities can be used for processing such concrete, but it is expensive to transport the radioactive and nonradiative concrete generated in the process of dismantling a nuclear power plant. Therefore, it is advantageous to build a simple processing facility onsite and reduce the transportation cost of considerable concrete waste. Radiological safety evaluation is necessary in the disposal and recycling process, using these facilities. It has been demonstrated that appropriate quantities of waste and suitable scenarios are necessary for evaluating the safety of the disposal and recycling process of the radioactive concrete generated during the

dismantling of nuclear power plants.

In this study, for evaluating the radiological safety of concrete dismantling, the process of radioactive concrete disposal and recycling during the dismantling of Kori Unit 1 is analyzed, and the potential exposure to the workers and public during waste disposal are calculated. The ALARA evaluation code VISIPLAN and RESRAD-ONSITE are used for dose assessment. Based on these results, the radiological safety is evaluated with reference to the annual dose limits.

## 2. Scenarios and assumptions

Scenario analysis of the work process is necessary for evaluating the received dose during the recycling and disposal of concrete. In general, the dismantling process includes five steps: processing, transportation, disposal, landfill, and recycling [9]. Dose assessment requires detailed analysis of the disposal process.

Initial processing of concrete involves the crushing and sorting of concrete into a form suitable for disposal or recycling. Concrete is processed along the conveyor belt, and crushed and sorted through a hopper, feeder, jaw crusher, and cone crusher. Concrete processing removes unnecessary impurities, and separates the crushed concrete into gravel (>5 mm), sand (1—5 mm), and cement paste (<1 mm) [10]. The concrete generated while dismantling the bio-shield is separated in the processing facility into gravel, sand, and cement paste through physical and thermal treatment [11]. In automated processing facilities, all the operations such as crushing, conveying, and sorting are performed automatically. As the entire process is executed automatically without worker intervention, workers are not required for concrete processing. An inspector is considered necessary to visually inspect for impurities that may be difficult for the machine to detect; hence, dose assessment for these inspectors is required.

Contaminated concrete above the clearance-level concentration is transported to a waste disposal site, whereas that below the clearance level can be landfilled as self-disposal of general industrial waste or sent to a concrete recycling process facility for recycling. It is assumed that landfill disposal is performed onsite. Furthermore, it is assumed that the recycling facility is installed near the decommissioning site, like the waste processing facility. On the other hand, it is necessary to locate the disposal facility in a region different from the waste processing facility. Dose assessment must be performed for the truck driver because waste movement to the disposal facility involves radioactivity above the clearance level.

After transportation, the concrete waste is stored using the rock-cavern disposal method. An annual dose limit of 0.1 mSv/y is required for medium and low-level radioactive waste disposal facilities as per the nuclear safety act. The dose generally satisfies the annual dose limit for the workers or public because of the disposal facility's own protective management and system. However, public who live near the landfill-site would continue to be exposed after landfill disposal; therefore, the dose calculation for the public should be considered. In the recycling process, it is necessary to consider the dosage according to the recycling scenario. Concrete can be recycled from the site to a shield or used for road construction; the road construction described in NUREG-1640 can be used as a representative case for the recycling of concrete [6]. Recycled concrete is not utilized for general roads, but only for limited applications, such as in highways, to minimize possible exposure. Therefore, pedestrians were not separately considered in this scenario. The flow chart of the concrete disposal process is shown in Fig. 1.

Four scenarios were derived, based on the analysis of the concrete recycling and disposal process, involving potential exposure to the workers and public by the radioactive concrete generated during the dismantling of the bio-shield.

- Scenario 1: The concrete waste inspector in the processing facility (occupational exposure)
- Scenario 2: The truck driver transporting concrete waste to the disposal facility (occupational exposure)
- Scenario 3: A driver on a recycled concrete road (public exposure)
- Scenario 4: Public living near a general landfill facility (public exposure)

## 3. Radioactive inventory analysis

The radioactivity needs to be identified for dose evaluation during the disposal and recycling of the dismantled concrete from the bio-shield. The radioactive inventory was evaluated in a previous study [12], using Monte Carlo code MCNP6. The response of the neutron flux to each part of the simulated bio-shield was evaluated Due to lack of actual data on the activation of the Kori Unit 1 bio-shield, the radioactivity must be determined through simulation based on the surrounding geometry, i.e., the reactor flux onto the bio-shield. MCNP6 is suitable for probabilistically evaluating the reactions between the bio-shield and the neutrons generated by the reactor over a long period [7,13]. In a previous research, MCNP6 was used to analyze the distribution of the axial and the radial directions. Based on these results, it was decided that the disposal process should commence 13.5 years after suspension, according to the decommissioning plan, and the radioactivity was calculated accordingly. The evaluated activation was calculated for the activity and volume of each area, and the volume and average radioactivity for concrete above the clearance level were calculated. The radiation attenuation rate due to intermediate shielding is constant irrespective of the radioactivity, and the dose is calculated by multiplying this by the dose factor and time. As the dose attenuation rate is less related to the radioactivity, it is expected that the same value will be obtained, if the average value is used. On excluding concrete 812 $m^3$ below the clearance level from the existing evaluation results, the volume of radioactive concrete was 327 $m^3$. Radioactive nuclides are distributed in the paste matrix throughout the paste-volume portion, which accounts for approximately 23% of the total volume; when separated, the radioactive quantity can be considerably reduced. On processing the concrete, the radioactivity concentration decreases to 0.17% for gravel and 0.67% for sand, whereas that of the paste increases, in contrast [14]. Therefore, an increased nuclide concentration was used for transportation evaluation, and the nuclide concentration of a mixture of sand and gravel was considered for recycling evaluation. Table 1 shows the volume of the concrete composition and radioactivity of each part derived through evaluation.

## 4. Dose assessment methods

Scenario 1 involves workers, who visually inspect concrete at the processing facility. Automated concrete processing eliminates the need for workers during processing and is designed such that the operator need not approach the concrete stream unless there is an accident, such as sudden machine stoppage. As remote monitoring has limitations, visual inspection may be required. Here, maximum exposure is expected because the concrete stream must be inspected visually at close range. A worker monitors the flow near the jaw crusher, checks for abnormalities in the stream, and removes the impurities. When the derived radioactive concrete is processed, the processing operation ends. The processing time is
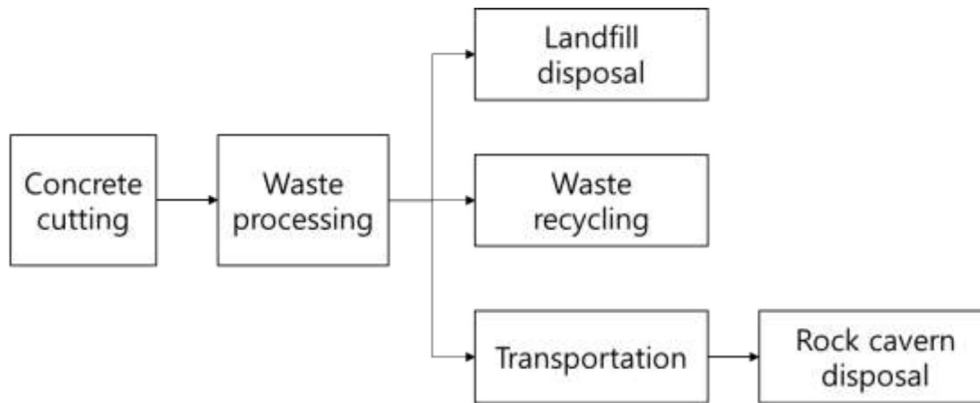
**Fig. 1.** Schematic of the concrete recycling and disposal process.

**Table 1**
Volume and radioactivity derived from the concrete waste.

| Parameter | Cement paste | Sand | Gravel |
|---|---|---|---|
| Volume (m$^3$) | 80 | 131 | 116 |
| $^3$H concentration (Bq/m$^3$) | 1.77E+11 | 7.35E+07 | 2.90E+08 |
| $^{55}$Fe concentration (Bq/m$^3$) | 3.05E+08 | 1.27E+05 | 5.00E+05 |
| $^{60}$Co concentration (Bq/m$^3$) | 2.91E+09 | 1.21E+06 | 4.77E+06 |
| $^{152}$Eu concentration (Bq/m$^3$) | 2.69E+07 | 1.12E+04 | 4.42E+04 |
| $^{154}$Eu concentration (Bq/m$^3$) | 6.20E+05 | 2.58E+02 | 1.02E+03 |

the same as the working time of the worker, who is exposed during this period. The quantity of derived radioactive concrete was 327 m$^3$, and it was assumed that approximately 10 h may be required, conservatively, considering the throughput of the jaw crusher, for processing. Depending on the task, if a worker works 8 h per day, two working days may be required. Even for these one or two days of work, the dose should be compared with the annual dose limit for safety.

Scenario 2 is intended for truck drivers involved in the transportation process. The truck driver remains in the truck and receives exposure from the loaded concrete. During loading or unloading, another assistant is involved in loading the waste concrete using the loader, and is also exposed to the concrete; however, as the driving time is longer than the loading time, if the representative truck driver does not exceed the dose limit in the dose evaluation results, the assistant can be considered to be within the radiological safety limit. It was assumed that the distance from the temporary waste processing facility at the Kori Unit 1 site to the low-level waste disposal facility in South Korea was approximately 71 km. Assuming that the vehicle operates at low speed (40 km/h), the required transport time is 1.77 h, and the driving time is calculated as 2 h, considering the loading time. As in the processing, the exposure time is also limited because the quantity of radioactive waste is limited. In the case of paste, a highly compound material was used, which was disposed during the disposal process. It was mixed with water to form a water-binder ratio of 0.6. Thereby, it can be expected to increase 1.6 times, and only 288 tons of paste are transferred to the disposal facility, necessitating 12 roundtrips, assuming a 25.5-ton truck for transport. Therefore, if one driver performs all the work, it is expected to be completed in 24 h. As the concrete waste is blocked within the container, the driver does not receive internal exposure, and only external exposure can be considered.

Disposal and recycling are both possible options for the generated waste. To consider both scenarios conservatively, it is assumed that all the waste is either recycled or disposed to landfill disposal.

Scenario 3 involves drivers driving on a recycled concrete road. In Scenario 3, public exposure is considered because the subject is a regular driver on a recycled road. Under Korean law, recycled concrete should be less than 30% of the total aggregate volume when mixed; the recycled aggregate volume was assumed to be 24% as per these regulations. For road construction, 3 m (W) × 100 m (L) × 0.2 m (h) dimensions was applied for the domestic regulations. It was assumed that recycled concrete was used on a road near the site, and that workers entering and exiting the area were exposed on the recycled concrete road while driving. Therefore, the exposure time can be calculated according to Equation (1).

$$T = \frac{2L}{V} \tag{1}$$

where

$T =$ Dwelling time (h)

$L =$ Length of the road (m)

$V =$ Vehicle velocity (m / s)

The driver was exposed twice, while entering and leaving the site, and the length of the road was calculated by dividing the derived volume of the concrete by the width and height of the road. The road was assumed to be 3 m (W) × 429 m (L) × 0.2 m (h), and the velocity of the vehicle was considered to be 30 km/h. Assuming that the number of entries per year was 250 days, the dwelling time was 14.29 h, conservatively considered as 15 h.

Scenario 4 involves residents living on landfilled concrete waste after self-disposal. Residents living on landfilled concrete waste are continuously exposed; some of the concrete nuclide flows into the groundwater, and the residents may directly ingest the groundwater or indirectly ingest the agricultural products grown using the groundwater. It was assumed that residents continue to live 24 h per a year on landfilled concrete waste; the thickness of the landfilled layer was assumed to be 45 cm, as per Korean waste management regulations. The effluent nuclide is affected by the surrounding climate. KMA (Korea Meteorological Administration) data was used to account for the wind speed (2.6 m/s) and precipitation (1302.5 mm) near the power plant for five years [15]. Data from the integrated dose assessment code (INDAC) was used for the agricultural and marine product intake fraction [16].

As the interaction between the radioactive concrete and surroundings in the disposal and recycling process is complex, it is

necessary to perform dose assessment for the workers and public using appropriate tools. VISIPLAN and RESRAD code were used to evaluate the dosage to the workers and public. In the case of external exposure, VISIPLAN, a 3D dose assessment tool, was used to evaluate the dosage to the processing workers, truck driver, and the users of recycled items; RESRAD, which is a computer code that can be used to estimate the radiation dose and risks, was used for evaluating the dosage to the public after disposal. It evaluates many exposure pathways such as an external gamma dose, a dust inhalation dose, soil ingestion dose, and food (plant, meat, and milk) ingestion dose or drinking water, and is suitable for evaluating the dosage to the public.

VISIPLAN and RESRAD were used to calculate the dose rate using point-kernel calculations, and the dosage for each scenario was evaluated according to Equation (2) [17,18].

$$D_t = \sum_{x=i}^{n} D_x T_x \qquad (2)$$

where

$D_t$ = External dose in the scenario (mSv). $D_x$ = External dose rate of a specific sub-scenario x (mSv/h).

$T_x$ = Working time of sub-scenario x (h).

For evaluating the recycling process of scenario 1, it was assumed that the processing facility comprised a hopper, jaw crusher, feeder, and cone crusher, and that the concrete moves along the conveyor belt between each equipment, as depicted in Fig. 2. During this process, the worker may additionally be exposed to concrete dust, which may result in internal exposure. This was evaluated using Equation (3) [19–21].

$$H_i = D_i B C_i E R t \qquad (3)$$

where

$H_i$ = Inhalation dose of radionuclide $i$ (mSv)

$D_i$ = Dose coefficient for the inhalation of radionuclide $i$ (mSv/Bq)

$B$ = Breathing rate $\left( m^3 \middle/ h \right) = 1.2 \, m^3 \middle/ h$      [20]

$C_i$ = Concentration of radionuclide $i$ in concrete $\left( \text{Bq} \middle/ \text{m}^3 \right)$

$E$ = Air release fraction

$R$ = Volume ratio of concrete to air in the working space

$t$ = Working time (h)

The dose coefficients for radionuclide inhalation are described in ICRP 119, where there is a difference in the dose coefficients between dust sizes of 1 μm and 5 μm [20]. The particle size was referenced to the dust distribution of the general aggregate using the EPA report [22]. A breathing rate of 1.2 m$^3$/h, as per ICRP 119, was used for the calculations. For the air release fraction, air emission factors of 1.2 g/ton and 0.023 g/ton due to the crusher and conveyer belt, respectively, were applied based on the US EPA data [23]. R is the volume ratio of concrete to air in the working space and indicates the degree of scattering of the radionuclides in concrete into air. The size of the working space was assumed to be 100 m$^3$, and the volume of concrete was assumed to be that of the jaw crusher and conveyor belt [24].

To evaluate the dosage to the driver in scenario 2, it was assumed that the concrete waste was transported in a 25.5-ton truck, simulated as shown in Fig. 3. For recycling in scenario 3, it was assumed that the mixture was recycled after mixing with ordinary paste [6]. The geometries for scenarios 1, 2, and 3, respectively, are listed in Table 2.

## 5. Results and discussions

The doses received during the recycling and disposal process of the radioactive from the bio-shield were calculated, and the results are depicted in Table 3. The annual dose was calculated by adding the external and internal dose rates and multiplying by the annual dwelling time.

The dose assessment results are as follows: 9.03E-01 mSv/y in scenario 1, 1.04E-03 mSv/y in scenario 2, 1.40E-03 mSv/y in scenario 3, and 4.47E-08 mSv/y in scenario 4, which do not exceed the annual dose limit of the representative in each scenario.

For scenario 1, the external and internal dose rates were calculated to be 7.5E-02 mSv/h and 1.53E-02 mSv/h, respectively. The effect of external exposure was expected to be five times greater than that of the internal. In the internal exposure evaluation process, the dust production rate was very low (<0.0012 kg/t),
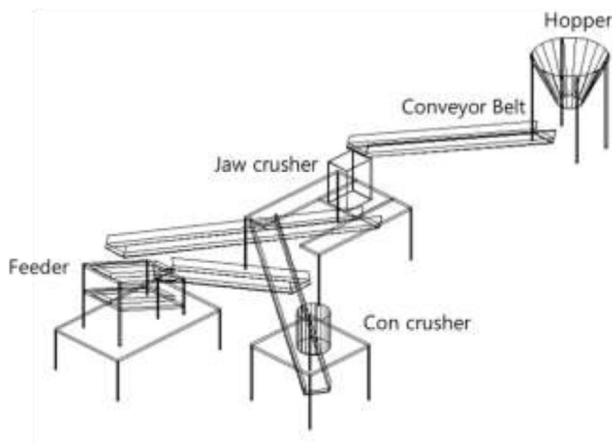


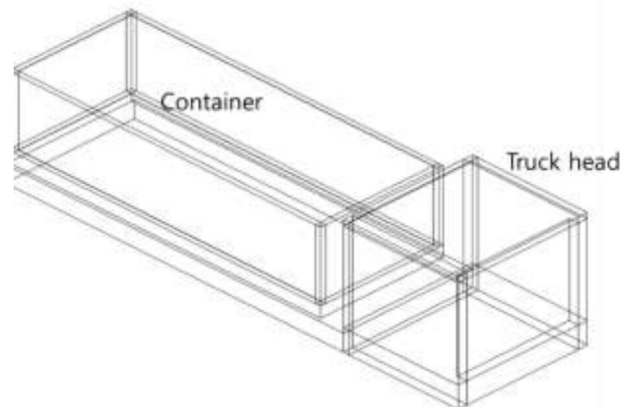Fig. 2. Layout of the concrete waste processing facility for simulation.



Fig. 3. Layout of the 25.5-ton truck for simulation.

**Table 2**
Geometric factors for simulation.

| Processing | Volume of Hopper | 6.68 m³ |
|---|---|---|
| | Volume of Jaw crusher | 2.25 m³ |
| | Volume of Feeder | 1.23 m³ |
| | Volume of Con crusher | 1.93 m³ |
| | Geometry of Conveyor Belt (W · H) | 1 · 1.7 m² |
| Processing | Geometry of Truck (W · D · H) | 2.38 · 8.5 · 2.1 m³ |
| | Volume of container | 15.9 m³ |
| Recycling | Geometry of Road (W · D · H) | 100 · 3 · 0.2 m³ |

therefore, it was considered that the dose was lower than that of the external exposure.

In the case of scenario 2, the dose rate was evaluated to be as low as 1.04E-03 mSv/y because there was no internal exposure for the driver as the concrete was blocked and the driving time was not considerable.

In scenarios 3 and 4, which are the recycling and disposal scenarios, respectively, the dose rates were significantly lower than the individual annual dose limit of 10 μSv. In the recycling case, the dose rate was evaluated to be 9.36E-05 mSv/h, which was below the annual dose limit for the public because a driver on the road consumed less time to pass through the recycled concrete, and the exposure time was less. Concrete can be recycled in various ways, and there may exist an environment where residents are subjected to high exposure, such as buildings. In this case, the derived dose rate cannot be directly substituted; however, if the dwelling time is large in the recycling process, detailed analysis of the dose would be required for safe usage of the recycled concrete.
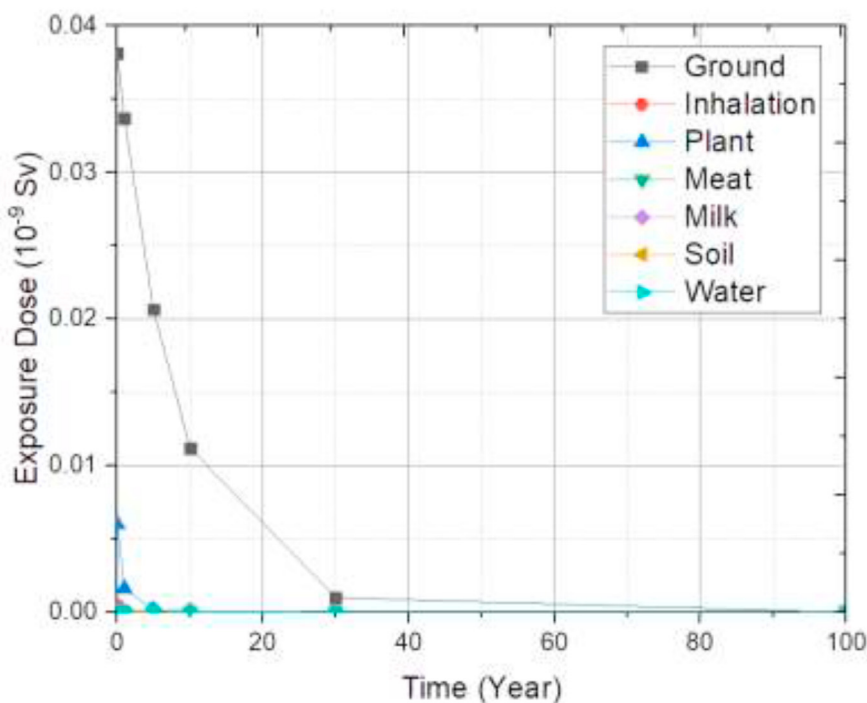
In the case of scenario 4, the exposure dose over time, after disposal, can be expressed as shown in Figs. 4 and 5.

In Fig. 4, there is an effect due to plant contamination or inhalation at approximately 15%, but with time, the ground dose accounts for the majority with more than 99%. At ground dose, the effect due to $^{60}$Co initially accounts for 99%; this decreases with time because of decay, with a half-life of 5.56 years. After 100 years, $^{152}$Eu with a relatively long half-life accounts for more than 90% of the dose. For plant contamination and inhalation, the effect of $^{3}$H is almost 100% immediately after disposal; however, it decreases with time, and the effect of $^{3}$H at 5 years is 0%; the effect of $^{60}$Co is primary.

After disposal, the annual dose is 4.47E-08 mSv. This is below the annual dose limit of 10 μSv, which is the clearance criteria specified by the nuclear safety act in South Korea. Therefore, the generated concrete waste from the bio-shield after processing is considered radiologically safe.

## 6. Conclusions

In this study, the disposal and recycling process of the concrete waste generated by the dismantling of the bio-shield was analyzed, and the potential doses received by the workers and public during this process were evaluated. In the disposal and recycling of the radioactive concrete from the bio-shield, four scenarios involving

**Table 3**
Dose assessment results of the disposal and recycling process.

| Scenario | External dose rate (mSv/h) | Internal dose rate (mSv/h) | Dwelling time (hour/year) | Total dose (mSv/year) |
|---|---|---|---|---|
| Scenario 1 | 7.50E-02 | 1.53E-02 | 10 | 9.03E-01 |
| Scenario 2 | 6.50E-05 | – | 16 | 1.04E-03 |
| Scenario 3* | 9.36E-05 | – | 15 | 1.40E-03 |
| Scenario 4* | 4.35E-12 | 7.5E-13 | 8760 | 4.47E-08 |

* Point in time immediately after disposal or recycling.



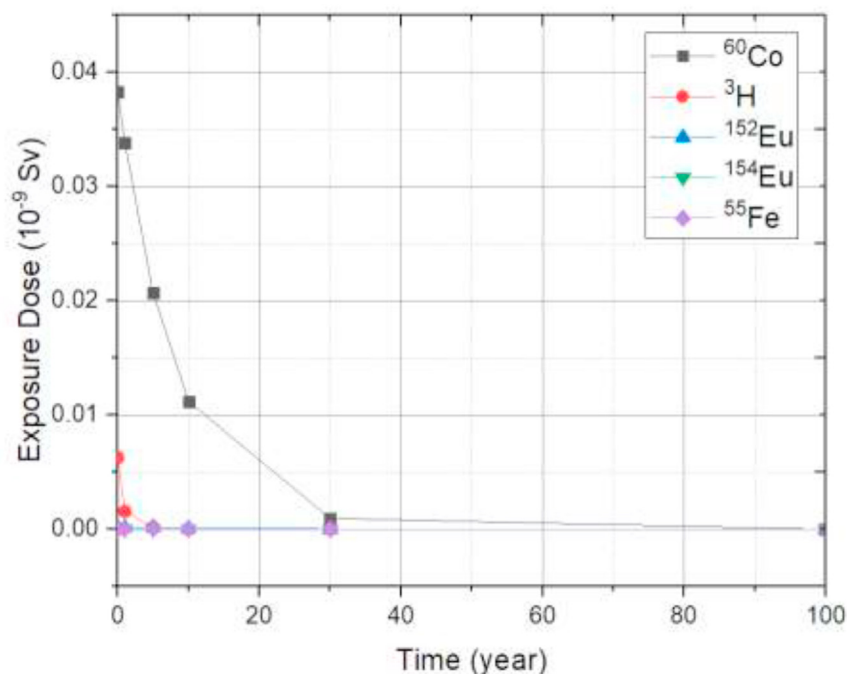**Fig. 4.** Exposure dose from the disposed concrete with time, as exposure paths.

**Fig. 5.** Exposure dose from the disposed concrete with time, as radionuclides.

worker or public exposure were analyzed, and the exposure doses were calculated. The evaluation results indicated that the exposure doses for all the scenarios did not to exceed the annual dose limits for the workers and public. Based on these results, it is possible to create waste disposal and recycling procedures, criteria for securing worker and public safety, and identify factors for safety assessment.

### Declaration of competing interest

The authors declare the following financial interests/personal relationships which may be considered as potential competing interests

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