



## Original Article

# Radiological safety of workers for the beta-nuclide removal treatment of the mixed waste from Wolsong nuclear power plant

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## ARTICLE INFO

## Keywords:

Decommissioning  
Dose assessment  
VISIPLAN 4.0  
Radioactive waste

## ABSTRACT

This study evaluates the radiological safety of the beta-nuclide removal process for mixed radioactive waste (waste resin, activated carbon, and zeolite) generated during the operation of Wolsong Unit 1. In addition, the potential for reducing worker radiation exposure in accident scenarios was investigated to optimize waste processing capacity. Using the VISIPLAN dose assessment code, external exposure was analyzed in scenarios involving the leakage of airborne radioactive particulates during the beta nuclide removal process. Internal exposure was assessed by taking into account the concentration of airborne radioactive particulates and effectiveness of the protective equipment. The 1 kW beta-nuclide removal reactor processes activated carbon and zeolite, while the 9 kW beta-nuclide removal reactor processes waste resin. Under the conditions of airborne radioactive particulates leakage, the maximum total dose rates for the 1 and 9 kW beta nuclide removal reactors reached  $1.51 \times 10^{-2}$  and  $7.22 \times 10^{-2}$  mSv/h, respectively. Based on the annual dose limit of 20 mSv over a five-year period, the permissible operating hours for the 1 kW and 9 kW reactors were determined to be 1325 h and 277 h, respectively. It is expected that these results will contribute to advances in radiation safety assessment techniques for reactor decommissioning and radioactive waste management.

## 1. Introduction

Wolsong Unit 1, Korea's first commercial heavy-water reactor, has been permanently shut down. During operation, various purification systems, including those for reactors and related equipment, are used to remove radioactive nuclides. These processes generate ion-exchange resins, zeolites, and activated carbon [1]. These materials are classified as mixed waste and stored in a mixed state within a Spent Resin Tank (SRT).

The average activity concentration of  $^{14}\text{C}$  in the spent resin mixture generated from the Wolsong Nuclear Power Plant is  $8.06 \times 10^6$  Bq/g. This value exceeds the regulatory limit of  $2.22 \times 10^5$  Bq/g for low-level radioactive waste specified in the "Regulations on the Classification and Self-Disposal Criteria for Radioactive Waste" [2]. The estimated total volume of this mixed waste is approximately 9000 drums (based on 200-L drums) [3]. Within the storage tank, the mixture consisted of an ion exchange resin, spent activated carbon, and zeolite in a weight ratio of 8:1:1. These components are considered critical for radioactive waste management and require proper treatment and handling.

It is estimated that domestic heavy-water reactors collectively store

approximately  $1.19 \times 10^8$  kg of mixed waste, comprising  $9.56 \times 10^5$  kg spent resin,  $1.19 \times 10^5$  kg zeolite, and  $1.19 \times 10^5$  kg activated carbon [1].

Radioactive nuclides present in the mixed waste include gamma-emitting nuclides, such as  $^{60}\text{Co}$  and  $^{137}\text{Cs}$ , and beta-emitting nuclides, such as  $^3\text{H}$  and  $^{14}\text{C}$ . These mixed wastes must be treated due to the different properties and hazards associated with each nuclide.  $^{60}\text{Co}$  and  $^{137}\text{Cs}$  pose significant external exposure risks due to their high-energy gamma emissions, while  $^3\text{H}$  and  $^{14}\text{C}$ , though less penetrating, are major contributors to internal exposure with long-term biological effects [4–11]. The concentrations of radioactive nuclides analyzed in the collected samples are listed in Table 1 [12].

As of January 2022, the data on the capacity of the disposal silo show that the storage volume of mixed waste generated during the operation of Wolsong Unit 1 was approximately 384 m<sup>3</sup>. Based on previous reports, the spent resin was found to contain radioactive nuclides at an average concentration of  $6.66 \times 10^5$  Bq/g [10], and the average density of the mixed waste was determined to be  $1.18 \times 10^6$  g/m<sup>3</sup>, which corresponds to a total estimated radioactivity of around  $3.02 \times 10^{14}$  Bq.

This volume represents approximately 82.5 % of the available

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<https://doi.org/10.1016/j.net.2025.103802>

Received 17 April 2025; Received in revised form 12 June 2025; Accepted 14 July 2025

Available online 15 July 2025

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**Table 1**  
Radioactivity concentration by nuclides of the samples of zeolite, activated carbon and spent resin.

[Bq/g]			
Nuclide	Zeolite	Activated carbon	Spent resin
<sup>57</sup> Co	$0.00 \times 10^0$	$0.00 \times 10^0$	$2.91 \times 10^1$
<sup>60</sup> Co	$9.37 \times 10^1$	$1.85 \times 10^2$	$4.94 \times 10^2$
<sup>51</sup> Cr	$0.00 \times 10^0$	$0.00 \times 10^0$	$2.58 \times 10^2$
<sup>134</sup> Cs	$6.60 \times 10^1$	$2.47 \times 10^0$	$1.57 \times 10^1$
<sup>137</sup> Cs	$9.11 \times 10^4$	$2.45 \times 10^3$	$1.72 \times 10^4$
<sup>54</sup> Mn	$0.00 \times 10^0$	$0.00 \times 10^0$	$2.67 \times 10^1$
<sup>95</sup> Nb	$8.68 \times 10^{-1}$	$7.31 \times 10^0$	$4.39 \times 10^1$
<sup>125</sup> Sn	$0.00 \times 10^0$	$1.55 \times 10^1$	$4.25 \times 10^2$
<sup>95</sup> Zr	$0.00 \times 10^0$	$0.00 \times 10^0$	$2.75 \times 10^1$
<sup>152</sup> Eu	$0.00 \times 10^0$	$0.00 \times 10^0$	$5.12 \times 10^2$
<sup>154</sup> Eu	$0.00 \times 10^0$	$0.00 \times 10^0$	$4.33 \times 10^1$
<sup>3</sup> H	$8.55 \times 10^3$	$1.56 \times 10^4$	$3.30 \times 10^4$
<sup>14</sup> C	$1.98 \times 10^2$	$2.22 \times 10^3$	$1.54 \times 10^5$

capacity at designated low- and intermediate-level radioactive waste disposal facilities. This corresponds to 82.5 % of the capacity of low- and intermediate-level radioactive waste disposal facilities.

Direct disposal of this mixed waste would significantly burden current disposal limits, emphasizing the necessity of nuclide concentration reduction for sufficient disposal space usage.

Since <sup>14</sup>C is a beta-emitting radionuclide present at intermediate-level concentrations, its removal is essential from spent resin, zeolite, and activated carbon. Due to their differing radioactivity levels and physicochemical properties, these materials require separate treatment processes. As <sup>14</sup>C is a beta nuclide present in intermediate-level concentrations, a beta nuclide removal process must be implemented for spent resin, zeolite, and activated carbon. Given that spent resin, zeolite, and activated carbon differ in terms of radioactivity levels and physicochemical properties, they must be treated separately and independently of each other.

In facilities developed for the treatment of mixed waste, the separation process generates airborne radioactive particulates on the perforated plates installed for the discharge of liquid waste. Accordingly, potential accident scenarios involving the release of airborne radioactive particulates must be considered, considering both external and internal exposure risks [13].

In this study, a dose assessment was performed for workers under a potential accident scenario involving the leakage of airborne radioactive particulates during the operation of a beta nuclide removal reactor to reduce the radioactive concentration.

## 2. Models and methodology

### 2.1. Modeling of the integrated treatment facility

The mixed waste generated during Wolsong Unit 1's operation was categorized into spent resin and activated carbon/zeolite to facilitate effective disposal. The mixed waste generated during the operation of Wolsong Unit 1 was divided into two categories, spent resin and activated carbon/zeolite, to ensure safe disposal. The VISIPLAN dose assessment code, developed in 1999 by the Belgian research center SCK-CEN and widely used as a planning tool for As Low As Reasonably Achievable (ALARA) in nuclear facilities [14], was used to check the radiation safety of workers in an integrated treatment facility for mixed waste from CANDU reactors.

Using the VISIPLAN code, a model of the radiation work environment was created and accident scenarios were defined to derive the external dose rate results based on worker locations and arm lengths. For the modeling process, a 3D model of the demonstration space was created with dimensions of 6 m (width), 8 m (length), and 13 m (height), and a total volume of 624 m<sup>3</sup>. The reactors for the removal of beta nuclides were modeled within this space. The demonstration space was

constructed with 600 mm thick concrete walls, and the beta nuclide removal reactors were made of Stainless Steel 304. The spent resin was processed using a 9 kW beta-nuclide removal reactor, whereas activated carbon and zeolite were processed using a 1 kW beta-nuclide removal reactor. Fig. 1 presents the 3D models of the 1 kW and 9 kW beta-nuclide removal reactors within the demonstration space.

These reactors aim to lower beta nuclide levels in materials—zeolite, activated carbon, and spent resin—which are known to contain both beta and gamma emitters from Wolsong Unit 1's spent resin tank.

The waste contains a high concentration of the beta-emitting radionuclide <sup>14</sup>C, requiring the application of an appropriate treatment method to reduce radioactive contamination. The model applied in this study employs microwave irradiation to lower contamination levels, reduce waste volume before final disposal, and improve disposal safety. Since the waste contains moisture, a drying process is necessary. During this process, microwave energy generated by a magnetron is used [15]. Microwave heating enables uniform and rapid heating and is particularly effective for removing <sup>3</sup>H from water.

The direct heating method offers the advantage of removing beta nuclides and residual free water in a short time. Simultaneously, agitation causes physical abrasion and dissociation of the mixed waste resin. This detachment process may lead to carbonization and cracking of the resin, generating airborne radioactive particulates. Secondary waste generated during treatment includes <sup>14</sup>C and <sup>3</sup>H. These secondary products can be further concentrated and potentially reused as valuable resources, for instance, industry application to radioisotope tracer, contributing not only to the reduction of radioactive waste but also to the promotion of resource recycling [16].

The 1 kW and 9 kW beta-nuclide removal reactors consist of an air in panel, an air out panel, a motor, and magnetrons, as shown in Fig. 2. The dimensions of the 1 kW beta-nuclide removal reactor are 107 cm in width, 60 cm in length, and 131.2 cm in height, and those of 9 kW reactor are 150 cm in width, 83.95 cm in length, and 175 cm in height.

The air in panel draws in air from outside to create constant air flow inside the reactor. The air out panel discharges the air generated inside, primarily releasing heat and other by-products generated during the treatment process, so that a stable environment is maintained inside the device. The motor supports the operation of the reactor as a driving mechanism. Both the 1 kW and 9 kW beta-nuclide removal reactors feature a 1 kW magnetron and three 3 kW magnetrons that generate high-frequency electromagnetic waves that promote ionization and facilitate beta nuclide removal.

### 2.2. Scenario of leakage of airborne radioactive particulates from a beta-nuclide removal reactor

Beta-nuclide removal technology reduces the concentration of radioactive nuclides by removing beta nuclides from mixed waste through microwave-induced self-heating.

During normal system operation, workers are only required to perform monitoring tasks, such as checking the process status via a control panel to ensure smooth operation. However, in the scenario assumed in this study where airborne radioactive particulates become trapped in the perforated mesh the affected area must be opened, and the worker is required to manually remove the accumulated particles through a cleaning operation. Initially, the mixed waste is introduced into the system using a transport container, and subsequent transfer within the system is conducted via internal piping.

To determine the generation rate of airborne radioactive particulates, calculations were made considering nuclide concentration and the space volume where leakage may occur.

In the beta-nuclide removal reactor addressed in this study, when airborne radioactive particulates accumulate on the perforated mesh, workers perform a cleaning task by opening the corresponding section and manually removing the particulates. For this task, it was assumed that the worker wears a mask with an Assigned Protection Factor (APF)

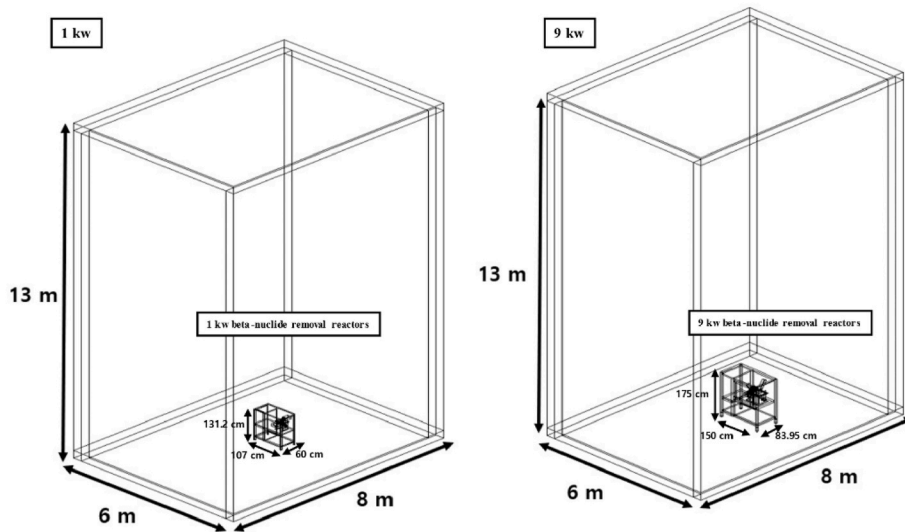


Fig. 1. 3D modeling of the 1 kW and 9 kW beta-nuclide removal reactors within the demonstration space.

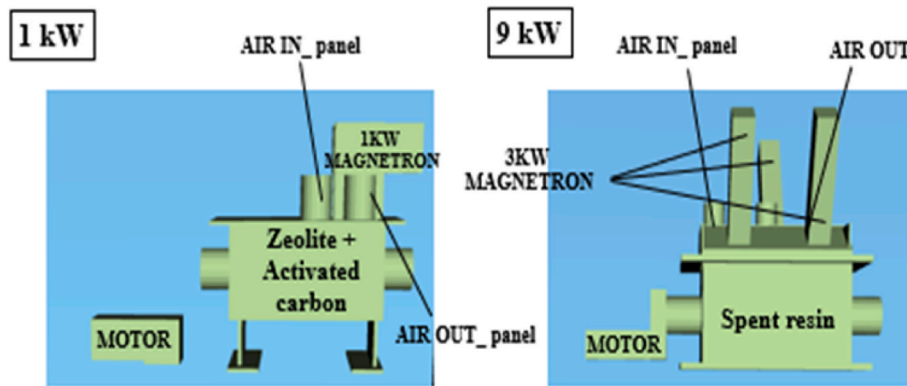


Fig. 2. 3D modeling of the 1 kW and 9 kW beta-nuclide removal reactor.

of 10.

The worker enters the demonstration area where the beta-nuclide removal reactor is installed through an access door, performs the airborne radioactive particulates removal task at the contaminated section, and exits through the same access route. This series of movements defines the operational pathway considered for dose assessment.

The processing capacities of the 1 kW and 9 kW beta-nuclide removal reactors were 1 and 9 kg per batch, respectively.

After the demonstration experiment, the conditions of the separated waste resin in the storage tank were checked to verify the separation process performance of the treatment system. It was confirmed that approximately 20 % of the airborne radioactive particulates was stuck between the perforated mesh designed for liquid waste discharge.

The estimated concentration of airborne radioactive particulates generated by the 1 kW and 9 kW reactors are 0.32 g/m<sup>3</sup> and 2.88 g/m<sup>3</sup>, respectively. However, to ensure that the system operates stably and effectively removes the waste even in the event of unexpected changes in operating conditions or increases in concentration, evaluations were conducted at intervals of 0.05 g/m<sup>3</sup> from 0.15 g/m<sup>3</sup> to 0.40 g/m<sup>3</sup> for the 1 kW reactor and 0.50 g/m<sup>3</sup> from 1.50 g/m<sup>3</sup> to 4.00 g/m<sup>3</sup> for the 9 kW reactor.

### 2.3. Radioactivity inventory of beta-nuclide removal reactors and input parameters for the dose calculation

The mixed waste considered in the airborne radioactive particulates

leakage scenario includes zeolite, activated carbon, and spent resin. In the 1 kW beta-nuclide removal reactor, zeolite and activated carbon were present in a 1:1 ratio.

Table 2 lists the radioactive inventory of the 1 kW beta-nuclide removal reactor by nuclide and concentration.

In the 9 kW beta-nuclide removal reactor, only the spent resin was considered. Table 3 lists the radioactivity of the 9 kW beta-nuclide removal reactor, broken down by radioactive nuclide concentration.

Both external and internal doses during radiation work in a mixed-waste treatment facility were assessed considering airborne radioactive particulates leakage scenario.

An external dose assessment was performed for a scenario in which airborne radioactive particulates leaked while the beta nuclide removal reactor was in operation. The external dose was assessed at a horizontal distance of 30 cm from the reactor based on the average arm length of an adult male [17].

The external dose was estimated using the point-kernel method with photon buildup corrections. The photon fluence rate for the external dose assessment was calculated using Equation (1) at a dose measurement point located at a certain distance from the point source [14].

$$\phi = \int \frac{S \cdot B \cdot e^{-x}}{4 \cdot \pi \cdot r^2} dV \tag{1}$$

where  $\phi$  is the photon fluence rate to be calculated (m<sup>-2</sup>·s<sup>-1</sup>), S is the strength of the source nuclide per unit time and volume (m<sup>-3</sup>·s<sup>-1</sup>), B is the buildup factor, x is the linear attenuation coefficient (cm<sup>-1</sup>)

**Table 2**  
Radioactivity of the 1 kW beta-nuclide removal reactor.

[Bq]						
Nuclide	Airborne radioactive particulates concentration [g/m <sup>3</sup> ]					
	0.15	0.20	0.25	0.30	0.35	0.40
<sup>60</sup> Co	1.30 × 10 <sup>4</sup>	1.74 × 10 <sup>4</sup>	2.17 × 10 <sup>4</sup>	2.61 × 10 <sup>4</sup>	3.04 × 10 <sup>4</sup>	6.48 × 10 <sup>4</sup>
<sup>134</sup> C	3.20 × 10 <sup>3</sup>	4.27 × 10 <sup>3</sup>	5.34 × 10 <sup>3</sup>	6.41 × 10 <sup>3</sup>	7.48 × 10 <sup>3</sup>	8.55 × 10 <sup>3</sup>
<sup>137</sup> Cs	4.38 × 10 <sup>6</sup>	5.84 × 10 <sup>6</sup>	7.30 × 10 <sup>6</sup>	8.76 × 10 <sup>6</sup>	1.02 × 10 <sup>7</sup>	1.17 × 10 <sup>7</sup>
<sup>95</sup> Nb	3.83 × 10 <sup>2</sup>	5.10 × 10 <sup>1</sup>	6.38 × 10 <sup>2</sup>	7.65 × 10 <sup>2</sup>	8.93 × 10 <sup>2</sup>	1.02 × 10 <sup>3</sup>
<sup>125</sup> Sn	7.25 × 10 <sup>2</sup>	9.67 × 10 <sup>2</sup>	1.21 × 10 <sup>3</sup>	1.45 × 10 <sup>3</sup>	1.69 × 10 <sup>3</sup>	1.93 × 10 <sup>3</sup>
<sup>3</sup> H	1.13 × 10 <sup>6</sup>	1.51 × 10 <sup>6</sup>	1.88 × 10 <sup>6</sup>	2.26 × 10 <sup>6</sup>	2.64 × 10 <sup>6</sup>	3.01 × 10 <sup>6</sup>
<sup>14</sup> C	1.13 × 10 <sup>5</sup>	1.51 × 10 <sup>5</sup>	1.89 × 10 <sup>5</sup>	2.26 × 10 <sup>5</sup>	2.64 × 10 <sup>5</sup>	3.02 × 10 <sup>5</sup>

**Table 3**  
Radioactivity of the 9 kW beta-nuclide removal reactor.

[Bq]						
Nuclide	Airborne radioactive particulates concentration [g/m <sup>3</sup> ]					
	1.50	2.00	2.50	3.00	3.50	4.00
<sup>57</sup> Co	2.72 × 10 <sup>4</sup>	3.63 × 10 <sup>4</sup>	4.54 × 10 <sup>4</sup>	5.45 × 10 <sup>4</sup>	6.36 × 10 <sup>4</sup>	7.26 × 10 <sup>4</sup>
<sup>60</sup> Co	4.62 × 10 <sup>5</sup>	6.17 × 10 <sup>5</sup>	7.71 × 10 <sup>5</sup>	9.25 × 10 <sup>5</sup>	1.08 × 10 <sup>6</sup>	1.23 × 10 <sup>6</sup>
<sup>51</sup> Cr	2.41 × 10 <sup>5</sup>	3.22 × 10 <sup>5</sup>	4.02 × 10 <sup>5</sup>	4.83 × 10 <sup>5</sup>	5.63 × 10 <sup>5</sup>	6.44 × 10 <sup>5</sup>
<sup>134</sup> Cs	1.47 × 10 <sup>4</sup>	1.96 × 10 <sup>4</sup>	2.45 × 10 <sup>4</sup>	2.94 × 10 <sup>4</sup>	3.43 × 10 <sup>4</sup>	3.92 × 10 <sup>4</sup>
<sup>137</sup> Cs	1.61 × 10 <sup>7</sup>	2.15 × 10 <sup>7</sup>	2.68 × 10 <sup>7</sup>	3.22 × 10 <sup>7</sup>	3.76 × 10 <sup>7</sup>	4.29 × 10 <sup>7</sup>
<sup>54</sup> Mn	2.50 × 10 <sup>4</sup>	3.33 × 10 <sup>4</sup>	4.17 × 10 <sup>4</sup>	5.00 × 10 <sup>4</sup>	5.83 × 10 <sup>4</sup>	6.66 × 10 <sup>4</sup>
<sup>95</sup> Nb	4.11 × 10 <sup>4</sup>	5.48 × 10 <sup>4</sup>	6.85 × 10 <sup>4</sup>	8.22 × 10 <sup>4</sup>	9.59 × 10 <sup>4</sup>	1.10 × 10 <sup>5</sup>
<sup>125</sup> Sn	3.98 × 10 <sup>5</sup>	5.30 × 10 <sup>5</sup>	6.63 × 10 <sup>5</sup>	7.96 × 10 <sup>5</sup>	9.28 × 10 <sup>5</sup>	1.06 × 10 <sup>6</sup>
<sup>95</sup> Zr	2.57 × 10 <sup>4</sup>	3.43 × 10 <sup>4</sup>	4.29 × 10 <sup>4</sup>	5.15 × 10 <sup>4</sup>	6.01 × 10 <sup>4</sup>	6.86 × 10 <sup>4</sup>
<sup>152</sup> Eu	4.79 × 10 <sup>5</sup>	6.39 × 10 <sup>5</sup>	7.99 × 10 <sup>5</sup>	9.58 × 10 <sup>5</sup>	1.12E × 10 <sup>6</sup>	1.28 × 10 <sup>6</sup>
<sup>154</sup> Eu	4.05 × 10 <sup>4</sup>	5.40 × 10 <sup>4</sup>	6.75 × 10 <sup>4</sup>	8.11 × 10 <sup>4</sup>	9.46 × 10 <sup>4</sup>	1.08 × 10 <sup>5</sup>
<sup>3</sup> H	3.09 × 10 <sup>7</sup>	4.12 × 10 <sup>7</sup>	5.15 × 10 <sup>7</sup>	6.18 × 10 <sup>7</sup>	7.21 × 10 <sup>7</sup>	8.24 × 10 <sup>7</sup>
<sup>14</sup> C	1.44 × 10 <sup>8</sup>	1.92 × 10 <sup>8</sup>	2.40 × 10 <sup>8</sup>	2.88 × 10 <sup>8</sup>	3.36 × 10 <sup>8</sup>	3.84 × 10 <sup>8</sup>

multiplied by the distance traveled through the material, which follows the source-dose point line of sight through the material  $i$ , ( $x = \sum \mu_i \tau_i$ )  $\mu_i$  is the linear attenuation coefficient of material  $i$  (cm<sup>-1</sup>), which indicates how much the radiation is attenuated in the respective material.  $\tau_i$  is the distance that radiation travels within material  $i$  (cm),  $r$  distance from the source (m),  $V$  volume (m<sup>3</sup>).

The assessment of the internal dose assumes that radioactive materials are inhaled by the human body. This assessment calculates the internal dose resulting from inhaled radioactive substances, taking into account the worker's breathing rate and the use of an APF10 mask [18].

The dose conversion factors (DCFs) used in this assessment are based on (International Commission on Radiological Protection (ICRP 119). The internal dose was calculated using Equation (2) [19,20].

$$D_{inh} \text{ (mSv/h)} = \sum \{DCF_i \times C_i\} \times BR \times APF \tag{2}$$

where  $D_{inh}$  is the committed effective dose for inhalation (mSv/h),  $DCF_i$  is the dose conversion factor of nuclide (mSv/Bq),  $C_i$  is the air concentration of the nuclide (Bq/m<sup>3</sup>),  $BR$  is the breathing rate (1.20 m<sup>3</sup>/h) [21], and  $APF$  is the assigned protection factor (0.1) [22]. This internal dose is determined based on the airborne radioactive particulates concentration of radioactive nuclides, the DCF values for each nuclide, and the worker's breathing rate. The exposure reduction effect was also taken into account by considering the filtration efficiency of protective equipment such as masks in filtering airborne radioactive particulates. This approach ensures that all relevant factors, including the airborne radioactive particulates concentration, DCF values, breathing rates, and mask efficiency, are fully incorporated into the internal dose assessment.

### 3. Results and discussion

#### 3.1. External dose assessment

The values of the external dose rate for the scenario of leakage of airborne radioactive particulates from the 1 kW beta nuclide removal reactor are listed in Table 4. When the estimated airborne radioactive particulates generation rate of 20 %, derived from actual experiments, is assumed for the demonstration space volume of the 1 kW beta-nuclide removal reactor, the external dose rate, based on airborne radioactive particulates concentration of 0.32 g/m<sup>3</sup>, is 5.3 × 10<sup>-5</sup> mSv/h. The external dose rate values for the airborne radioactive particulates leakage scenario in the 9 kW beta-nuclide removal reactor are listed in Table 5.

Likewise, applying the 20 % generation rate found in experiments to

**Table 4**  
External dose rate for the scenario of airborne radioactive particulates leakage from the 1 kW beta-nuclide removal reactor.

[mSv/h]						
Nuclide	Airborne radioactive particulates concentration [g/m <sup>3</sup> ]					
	0.15	0.20	0.25	0.30	0.35	0.40
<sup>60</sup> Co	1.70 × 10 <sup>-7</sup>	2.80 × 10 <sup>-7</sup>	4.10 × 10 <sup>-7</sup>	4.80 × 10 <sup>-7</sup>	6.00 × 10 <sup>-7</sup>	6.10 × 10 <sup>-7</sup>
<sup>134</sup> Cs	4.10 × 10 <sup>-8</sup>	5.70 × 10 <sup>-8</sup>	6.60 × 10 <sup>-8</sup>	8.30 × 10 <sup>-8</sup>	9.90 × 10 <sup>-8</sup>	1.10 × 10 <sup>-7</sup>
<sup>137</sup> Cs	2.50 × 10 <sup>-5</sup>	3.20 × 10 <sup>-5</sup>	3.50 × 10 <sup>-5</sup>	5.00 × 10 <sup>-5</sup>	5.40 × 10 <sup>-5</sup>	5.60 × 10 <sup>-5</sup>
<sup>95</sup> Nb	3.70 × 10 <sup>-9</sup>	3.40 × 10 <sup>-9</sup>	4.40 × 10 <sup>-9</sup>	5.10 × 10 <sup>-9</sup>	5.40 × 10 <sup>-9</sup>	6.70 × 10 <sup>-9</sup>
<sup>125</sup> Sn	1.70 × 10 <sup>-9</sup>	2.20 × 10 <sup>-9</sup>	3.50 × 10 <sup>-9</sup>	3.30 × 10 <sup>-9</sup>	3.90 × 10 <sup>-9</sup>	4.20 × 10 <sup>-9</sup>
SUM	2.52 × 10 <sup>-5</sup>	3.23 × 10 <sup>-5</sup>	3.55 × 10 <sup>-5</sup>	5.06 × 10 <sup>-5</sup>	5.47 × 10 <sup>-5</sup>	5.67 × 10 <sup>-5</sup>

**Table 5**  
External dose rate for the scenario of airborne radioactive particulates leakage from the 9 kW beta-nuclide removal reactor.

[mSv/h]						
Airborne radioactive particulates concentration [g/m <sup>3</sup> ]						
Nuclide	1.50	2.00	2.50	3.00	3.50	4.00
<sup>57</sup> Co	1.60 × 10 <sup>-8</sup>	2.00 × 10 <sup>-8</sup>	4.00 × 10 <sup>-8</sup>	3.10 × 10 <sup>-8</sup>	3.50 × 10 <sup>-8</sup>	4.20 × 10 <sup>-8</sup>
<sup>60</sup> Co	6.25 × 10 <sup>-6</sup>	8.00 × 10 <sup>-6</sup>	1.10 × 10 <sup>-5</sup>	1.40 × 10 <sup>-5</sup>	1.60 × 10 <sup>-5</sup>	1.60 × 10 <sup>-5</sup>
<sup>51</sup> Cr	4.00 × 10 <sup>-8</sup>	6.00 × 10 <sup>-8</sup>	6.80 × 10 <sup>-8</sup>	7.90 × 10 <sup>-8</sup>	9.60 × 10 <sup>-8</sup>	1.20 × 10 <sup>-7</sup>
<sup>134</sup> Cs	1.30 × 10 <sup>-7</sup>	1.90 × 10 <sup>-7</sup>	2.10 × 10 <sup>-7</sup>	2.60 × 10 <sup>-7</sup>	3.40 × 10 <sup>-7</sup>	3.50 × 10 <sup>-7</sup>
<sup>137</sup> Cs	5.00 × 10 <sup>-5</sup>	7.80 × 10 <sup>-5</sup>	1.10 × 10 <sup>-4</sup>	1.20 × 10 <sup>-4</sup>	1.20 × 10 <sup>-4</sup>	1.40 × 10 <sup>-4</sup>
<sup>54</sup> Mn	1.20 × 10 <sup>-7</sup>	1.50 × 10 <sup>-7</sup>	2.00 × 10 <sup>-7</sup>	2.30 × 10 <sup>-7</sup>	3.00 × 10 <sup>-7</sup>	3.00 × 10 <sup>-7</sup>
<sup>95</sup> Nb	1.90 × 10 <sup>-7</sup>	2.50 × 10 <sup>-7</sup>	3.10 × 10 <sup>-7</sup>	3.80 × 10 <sup>-7</sup>	4.90 × 10 <sup>-7</sup>	5.90 × 10 <sup>-7</sup>
<sup>125</sup> Sn	6.60 × 10 <sup>-7</sup>	8.90 × 10 <sup>-7</sup>	1.10 × 10 <sup>-6</sup>	1.30 × 10 <sup>-6</sup>	1.80 × 10 <sup>-6</sup>	1.70 × 10 <sup>-6</sup>
<sup>95</sup> Zr	1.60 × 10 <sup>-7</sup>	1.60 × 10 <sup>-7</sup>	2.00 × 10 <sup>-7</sup>	2.30 × 10 <sup>-7</sup>	3.20 × 10 <sup>-7</sup>	3.10 × 10 <sup>-7</sup>
<sup>152</sup> Eu	3.20 × 10 <sup>-6</sup>	4.20 × 10 <sup>-6</sup>	5.20 × 10 <sup>-6</sup>	6.00 × 10 <sup>-6</sup>	8.50 × 10 <sup>-6</sup>	7.60 × 10 <sup>-6</sup>
<sup>154</sup> Eu	3.00 × 10 <sup>-7</sup>	3.70 × 10 <sup>-7</sup>	4.80 × 10 <sup>-7</sup>	5.90 × 10 <sup>-7</sup>	7.00 × 10 <sup>-7</sup>	7.90 × 10 <sup>-7</sup>
SUM	6.11 × 10 <sup>-5</sup>	9.23 × 10 <sup>-5</sup>	1.29 × 10 <sup>-4</sup>	1.43 × 10 <sup>-4</sup>	1.49 × 10 <sup>-4</sup>	1.68 × 10 <sup>-4</sup>

the 9 kW reactor volume resulted in an external dose estimate at a concentration of 2.88 g/m<sup>3</sup> is 1.38 × 10<sup>-4</sup> mSv/h. The grid-based dose estimation with VISIPLAN, may lead to discrepancies between the actual and estimated doses for volume sources [14].

No probabilistic approach was used and linear interpolation was applied. Although linear interpolation is a useful and simple estimation method, it may not fully capture nonlinear physical relationships, which may lead to nonlinear results in the dose calculation.

### 3.2. Assessment of the internal dose

The internal dose rate values for the scenario of airborne radioactive particulates leakage from the 1 kW beta-nuclide removal reactor are listed in Table 6. In this scenario, the internal dose rate at airborne radioactive particulates of 0.32 g/m<sup>3</sup>, corresponding to 1 kW, is 1.18 × 10<sup>-2</sup> mSv/h when internal exposure occurs.

**Table 6**  
Internal dose rate for the 1 kW beta-nuclide removal reactor.

[mSv/h]						
Airborne radioactive particulates concentration [g/m <sup>3</sup> ]						
Nuclide	0.15	0.20	0.25	0.30	0.35	0.40
<sup>60</sup> Co	4.26 × 10 <sup>-5</sup>	5.69 × 10 <sup>-5</sup>	7.11 × 10 <sup>-5</sup>	8.53 × 10 <sup>-5</sup>	9.95 × 10 <sup>-5</sup>	1.14 × 10 <sup>-4</sup>
<sup>134</sup> Cs	5.92 × 10 <sup>-5</sup>	7.89 × 10 <sup>-6</sup>	9.86 × 10 <sup>-6</sup>	1.18 × 10 <sup>-5</sup>	1.38 × 10 <sup>-5</sup>	1.58 × 10 <sup>-5</sup>
<sup>137</sup> Cs	5.64 × 10 <sup>-3</sup>	7.52 × 10 <sup>-3</sup>	9.40 × 10 <sup>-3</sup>	1.13 × 10 <sup>-2</sup>	1.32 × 10 <sup>-2</sup>	1.50 × 10 <sup>-2</sup>
<sup>95</sup> Nb	1.18 × 10 <sup>-7</sup>	1.57 × 10 <sup>-7</sup>	1.96 × 10 <sup>-7</sup>	2.36 × 10 <sup>-7</sup>	2.75 × 10 <sup>-7</sup>	3.14 × 10 <sup>-7</sup>
<sup>125</sup> Sn	4.60 × 10 <sup>-7</sup>	6.14 × 10 <sup>-7</sup>	7.67 × 10 <sup>-7</sup>	9.21 × 10 <sup>-7</sup>	1.07 × 10 <sup>-6</sup>	1.23 × 10 <sup>-6</sup>
<sup>3</sup> H	8.91 × 10 <sup>-6</sup>	1.19 × 10 <sup>-5</sup>	1.49 × 10 <sup>-5</sup>	1.78 × 10 <sup>-5</sup>	2.08 × 10 <sup>-5</sup>	2.38 × 10 <sup>-5</sup>
<sup>14</sup> C	1.41 × 10 <sup>-7</sup>	1.89 × 10 <sup>-7</sup>	2.36 × 10 <sup>-7</sup>	2.83 × 10 <sup>-7</sup>	3.30 × 10 <sup>-7</sup>	3.77 × 10 <sup>-7</sup>
SUM	5.70 × 10 <sup>-3</sup>	7.60 × 10 <sup>-3</sup>	9.50 × 10 <sup>-3</sup>	1.14 × 10 <sup>-2</sup>	1.33 × 10 <sup>-2</sup>	1.52 × 10 <sup>-2</sup>

The internal dose rate values for the scenario of airborne radioactive particulates leakage from the 9 kW beta-nuclide removal reactor are shown in Table 7. In this scenario, the internal dose rate for airborne radioactive particulates of 2.88 g/m<sup>3</sup>, corresponding to 9 kW, is 5.18 × 10<sup>-2</sup> mSv/h when internal exposure occurs.

For both the 1 kW and the 9 kW beta-nuclide removal reactor, the external and internal dose rates were most significantly influenced by <sup>137</sup>Cs, due to its high radioactivity concentration. The total dose rate combining the internal and external dose rates scenario of leakage of airborne radioactive particulates from the beta nuclide removal reactors is listed in Table 8.

The total dose rate, combining external and internal exposures, increased on average by 22 % per 0.05 g/m<sup>3</sup> interval for the 1 kW beta-nuclide removal reactor and similarly by 22 % per 0.5 g/m<sup>3</sup> interval for the 9 kW one, respectively. The total dose rate results for 1 kW and 9 kW are presented in Figs. 3 and 4, respectively. As no probabilistic methods are used in VISIPLAN, the linear interpolation applied may not fully reflect the physical relationships, which may lead to nonlinear results in the calculation of the external exposure dose. However, for the removal of beta-nuclides using 1 kW and 9 kW beta-nuclide removal reactors, dose rate increases ranged from 14.2 % to 33.2 % for concentration increases of 0.05 g/m<sup>3</sup> and 0.5 g/m<sup>3</sup>, respectively, which corresponds to an average increase rate of 22 %.

For the 1 kW reactor, when airborne radioactive particulates concentration reached 0.32 g/m<sup>3</sup>, internal exposure contributed 98.8 % of the total dose. This indicates that internal exposure is the dominant contributor to worker dose.

Similarly, at airborne radioactive particulates concentration of 2.88 g/m<sup>3</sup> for the 9 kW beta-nuclide removal reactor, the internal dose rate accounted for 99.8 % of the total dose rate, confirming that internal exposure is the most dominant concern for workers.

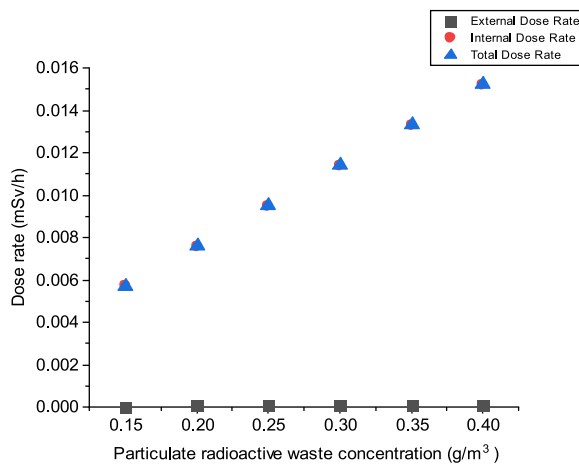
Based on the maximum airborne radioactive particulates concentrations, the maximum total dose rate for the 1 kW unit was 1.51 × 10<sup>-2</sup> mSv/h, whereas for the 9 kW unit it was 7.22 × 10<sup>-2</sup> mSv/h. Taking into account the average annual dose limit of 20 mSv over a five year period,

**Table 7**  
Internal dose rate for the 9 kW beta-nuclide removal reactor.

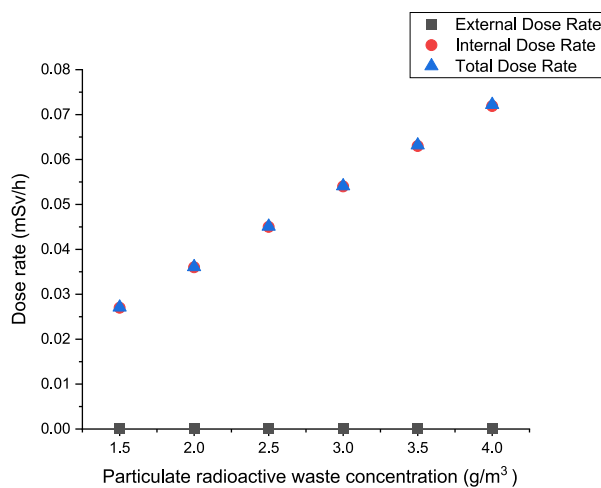
[mSv/h]						
Airborne radioactive particulates concentration [g/m <sup>3</sup> ]						
Nuclide	1.50	2.00	2.50	3.00	3.50	4.00
<sup>57</sup> Co	4.92 × 10 <sup>-6</sup>	6.56 × 10 <sup>-6</sup>	8.21 × 10 <sup>-6</sup>	9.85 × 10 <sup>-6</sup>	1.15 × 10 <sup>-5</sup>	1.31 × 10 <sup>-5</sup>
<sup>60</sup> Co	1.51 × 10 <sup>-3</sup>	2.02 × 10 <sup>-3</sup>	2.52 × 10 <sup>-3</sup>	3.02 × 10 <sup>-3</sup>	3.53 × 10 <sup>-3</sup>	4.03 × 10 <sup>-3</sup>
<sup>51</sup> Cr	1.67 × 10 <sup>-6</sup>	2.23 × 10 <sup>-6</sup>	2.79 × 10 <sup>-6</sup>	3.34 × 10 <sup>-6</sup>	3.90 × 10 <sup>-6</sup>	4.46 × 10 <sup>-6</sup>
<sup>134</sup> Cs	2.71 × 10 <sup>-5</sup>	3.62 × 10 <sup>-5</sup>	4.52 × 10 <sup>-5</sup>	5.43 × 10 <sup>-5</sup>	6.33 × 10 <sup>-5</sup>	7.23 × 10 <sup>-5</sup>
<sup>137</sup> Cs	2.07 × 10 <sup>-2</sup>	2.77 × 10 <sup>-2</sup>	3.46 × 10 <sup>-2</sup>	4.15 × 10 <sup>-2</sup>	4.84 × 10 <sup>-2</sup>	5.53 × 10 <sup>-2</sup>
<sup>54</sup> Mn	5.77 × 10 <sup>-6</sup>	7.69 × 10 <sup>-6</sup>	9.61 × 10 <sup>-6</sup>	1.15 × 10 <sup>-5</sup>	1.35 × 10 <sup>-5</sup>	1.54 × 10 <sup>-5</sup>
<sup>95</sup> Nb	1.26 × 10 <sup>-5</sup>	1.69 × 10 <sup>-5</sup>	2.11 × 10 <sup>-5</sup>	2.53 × 10 <sup>-5</sup>	2.95 × 10 <sup>-5</sup>	3.37 × 10 <sup>-5</sup>
<sup>125</sup> Sn	2.52 × 10 <sup>-4</sup>	3.37 × 10 <sup>-4</sup>	4.21 × 10 <sup>-4</sup>	5.05 × 10 <sup>-4</sup>	5.89 × 10 <sup>-4</sup>	6.73 × 10 <sup>-4</sup>
<sup>95</sup> Zr	1.24 × 10 <sup>-5</sup>	1.65 × 10 <sup>-5</sup>	2.06 × 10 <sup>-5</sup>	2.48 × 10 <sup>-5</sup>	2.89 × 10 <sup>-5</sup>	3.30 × 10 <sup>-5</sup>
<sup>152</sup> Eu	3.59 × 10 <sup>-3</sup>	4.79 × 10 <sup>-3</sup>	5.99 × 10 <sup>-3</sup>	7.19 × 10 <sup>-3</sup>	8.39 × 10 <sup>-3</sup>	9.58 × 10 <sup>-3</sup>
<sup>154</sup> Eu	3.90 × 10 <sup>-4</sup>	5.20 × 10 <sup>-4</sup>	6.50 × 10 <sup>-4</sup>	7.79 × 10 <sup>-4</sup>	9.09 × 10 <sup>-4</sup>	1.04 × 10 <sup>-3</sup>
<sup>3</sup> H	2.44 × 10 <sup>-4</sup>	3.25 × 10 <sup>-4</sup>	4.06 × 10 <sup>-4</sup>	4.87 × 10 <sup>-4</sup>	5.68 × 10 <sup>-4</sup>	6.49 × 10 <sup>-4</sup>
<sup>14</sup> C	1.80 × 10 <sup>-4</sup>	2.40 × 10 <sup>-4</sup>	3.00 × 10 <sup>-4</sup>	3.60 × 10 <sup>-4</sup>	4.20 × 10 <sup>-4</sup>	4.80 × 10 <sup>-4</sup>
SUM	2.70 × 10 <sup>-2</sup>	3.60 × 10 <sup>-2</sup>	4.50 × 10 <sup>-2</sup>	5.40 × 10 <sup>-2</sup>	6.30 × 10 <sup>-2</sup>	7.19 × 10 <sup>-2</sup>

**Table 8**  
Total dose rate by concentration interval for 1 kW and 9 kW beta nuclide removal reactors.

1 kW		9 kW	
Airborne radioactive particulates concentration	Total dose rate	Airborne radioactive particulates concentration	Total dose rate
0.15 g/m <sup>3</sup>	5.73 × 10 <sup>-3</sup>	1.50 g/m <sup>3</sup>	2.71 × 10 <sup>-2</sup>
0.20 g/m <sup>3</sup>	7.63 × 10 <sup>-3</sup>	2.00 g/m <sup>3</sup>	3.61 × 10 <sup>-2</sup>
0.25 g/m <sup>3</sup>	9.53 × 10 <sup>-3</sup>	2.50 g/m <sup>3</sup>	4.51 × 10 <sup>-2</sup>
0.30 g/m <sup>3</sup>	1.11 × 10 <sup>-2</sup>	3.00 g/m <sup>3</sup>	5.41 × 10 <sup>-2</sup>
0.35 g/m <sup>3</sup>	1.31 × 10 <sup>-2</sup>	3.50 g/m <sup>3</sup>	6.32 × 10 <sup>-2</sup>
0.40 g/m <sup>3</sup>	1.51 × 10 <sup>-2</sup>	4.00 g/m <sup>3</sup>	7.22 × 10 <sup>-2</sup>



**Fig. 3.** Total dose rate of a 1 kW beta-nuclide removal reactor.



**Fig. 4.** Total dose rate of a 9 kW beta-nuclide removal reactor.

workers can operate the 1 kW device for 1325 h and the 9 kW device for 277 h under the airborne radioactive particulates leakage scenario. This ensures that maintenance can be adequately performed in environments with airborne radioactive particulates leakage, ensuring safe operation of the beta-nuclide removal reactors.

**4. Conclusions**

In this study, the radiation safety of workers was assessed in beta-nuclide removal reactors based on possible airborne radioactive particulates leakage scenarios. At airborne radioactive particulates concentration of 0.32 g/m<sup>3</sup> for the 1 kW reactor, internal dose rates accounted for 98.8 % of the total dose rate, indicating that internal exposure is the primary concern for workers. Similarly, for the 9 kW reactor, at airborne radioactive particulates concentration of 2.88 g/m<sup>3</sup>, the internal dose rates also accounted for 99.8 % of the total dose rate. Both the external and internal dose rates for the 1 kW and 9 kW beta nuclide removal reactors were most influenced by the <sup>137</sup>Cs nuclide. Internal exposure was found to be the main concern for workers, as it was the largest contributor to the dose for both the 1 kW and 9 kW reactors.

Since the actual working time may vary depending on the amount of airborne radioactive particulates accumulated in the perforated mesh, a conservative approach was adopted for the dose assessment in this study. Accordingly, the allowable working time for personnel was calculated based on the total dose rate (mSv/h), which includes both external and internal exposures. This calculation was conducted with reference to the occupational dose limits recommended by the ICRP an average of 20 mSv per year and a cumulative dose of 100 mSv over five years.

A conservative assessment found that a working time of 1325 h for the 1 kW reactor and 277 h for the 9 kW met the annual average dose limit of 20 mSv for a radiation worker. The beta-nuclide removal reactor addressed in this study is currently planned for installation, and it is expected that field workers will directly conduct related operations once the system is operational. Based on the results of the demonstration experiment, the airborne radioactive particulates removal task associated with this system is anticipated to be completed within 10 min. Therefore, even after the system is installed and a work team is organized, the radiological safety of the radiation workers can be adequately ensured. These findings underscore the importance of optimizing the radiation safety of workers by considering different possible scenarios. It is believed that this is the crucial basis for improving radiation safety and waste management efficiency to ensure the radiation safety of the decommissioning process of heavy-water reactors.

**CRedit authorship contribution statement**

**Ja Yeong Yoon:** Writing – review & editing, Writing – original draft, Methodology, Investigation, Data curation, Conceptualization. **Sia Hwang:** Validation, Methodology, Investigation. **Hee Reyoung Kim:** Writing – review & editing, Supervision, Project administration.

**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.

**Acknowledgements**

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning and the Ministry of Trade, Industry, and Energy (MOTIE) of the Republic of Korea (Grant No. RS-2023-00233879 and grant no.2021400000410).

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