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Radionuclide-Specific Exposure Pathway Analysis of Kori Unit 1 Containment Building Surface

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Site characterization for decommissioning Kori Unit 1 is ongoing in South Korea after 40 years of successful operation. Kori Unit 1's containment building is assumed to be mostly radioactively contaminated, and therefore radiation exposure management and detailed contamination investigation are required for decommissioning and dismantling it safely. In this study, site-specific Derived Concentration Guideline Levels (DCGLs) were derived using the residual radioactivity risk evaluation tool, RESRAD-BUILD code. A conceptual model of containment building for Kori Unit 1 was set up and limited occupational worker building inspection scenario was applied. Depending on the source location, the maximum contribution source and exposure pathway of each radionuclide were analyzed. The contribution of radionuclides to dose and exposure pathways, by source location, is expected to serve as basic data in the assessment criteria of survey areas and classification of impact areas during further decommissioning and decontamination of sites.

Keywords: MARSSIM, Containment building, DCGL, RESRAD-BUILD, Pathway analysis

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1. Introduction

Currently, studies under relevant preparation and regulatory guidelines for the decommissioning of Kori Unit 1 are being prepared based on the U.S. standard decommissioning procedure guidance manual, MARSSIM (Multi-Agency Radiation Survey and Site Investigation Manual). Preliminary survey considerations and remedial action support survey in MARSSIM specify that DCGL should be derived to establish a site deregulation safety assessment method [1]. DCGL is the radionuclide-specific concentration when the maximum annual dose based on the release criteria is received. It is applied to the scoping survey, which is a step that checks for and confirms the absence of the contaminated area and complements the HSA (Historical Site Assessment). It confirms the decontamination activity according to the classification of contaminated areas and the level of contamination. Thus, DCGL is updated through iterative processes until the FSS (Final Status Survey), which is the ultimate goal of MARSSIM, to determine the eventual release of the contaminated site. In this study, the DCGL for the interior surface of the Kori Unit 1 containment building with the main system and equipment in the Kori site was derived using the RESRAD-BUILD v3.5 based on the industrial worker building inspection scenario. As the characterization survey of the Kori Unit 1 is underway, no specific dismantling schedule or reuse plan has been proposed yet. In the case of the land area, the area including the adjacent unit is included as the impacted area. Considering the specificity of the Kori site where multiple adjacent units are located, it is judged that the industrial worker building inspection scenario is appropriate that the containment building needs to be closed and managed during the dismantling of the remaining units. MARSSIM emphasizes that surveys of building surfaces require individual surveys of different locations and structural configurations within the building. For example, additional consideration should be given to the possibility of roof contamination if radioactive material is allowed to move or the facility is in close proximity to the air effluent discharge point, or when the walls are judged to be potential locations of external contamination according to process equipment, piping and ventilation equipment. Thus, for each radionuclide, the maximum contributing source and exposure pathway for the dose depending on the structural location within the containment building was identified.

2. Materials and Method

2.1 Selection of radionuclides for Kori Unit 1 containment building

The list of radionuclides applied to the Kori Unit 1 containment building was preliminarily determined in the same manner as those applied to the concrete and interior surfaces of the containment building of Zion and Rancho Seco NPP. It is difficult to obtain the accurate radionuclides data because the radionuclides data of Kori Unit 1 can be obtained through the decommissioning process of Kori Unit 1 ongoing now. Therefore, in order to select the radionuclides most likely to be detected in Kori Unit 1, radionuclides detected in 2 references PWRs were used. Sample dose calculations for the containment and auxiliary buildings of Zion NPP accounted for over 99.5% of the total dose of the normalized source term for ^{60}Co, ^{134}Cs, ^{137}Cs, ^{63}Ni and ^{90}Sr. In total, 8 radionuclides (^{60}Co, ^{134}Cs, ^{137}Cs, ^{152}Eu, ^{154}Eu, ^{3}H, ^{63}Ni, ^{90}Sr) were selected as the primary radionuclide of the containment underground concrete core sample via analysis of individual radionuclide concentrations and fractions for normalized mixtures on a specific date [2, 3]. Rancho Seco NPP selected 9 radionuclides (^{241}Am, ^{60}Co, ^{134}Cs, ^{137}Cs, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{90}Sr) by using the nuclide fraction (*nf*) by dividing the decayed concentration by the total sample radioactivity concentration based on the average result of the individual samples for the building [4, 5]. Thus, including all of them, a total of 13 radionuclides (^{241}Am, ^{60}Co, ^{134}Cs, ^{137}Cs, ^{152}Eu, ^{154}Eu, ^{3}H, ^{63}Ni, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{90}Sr) were preliminary set as a list of the application of the Kori Unit 1 containment building. It includes radionuclides (^{60}Co, ^{134}Cs) detected outside and at the Kori Unit 1 fuel building invasion site [6]. As the characterization survey is conducted, it is considered that a list update that reflects site-specific information is necessary.

2.2 RESRAD-BUILD modeling

RESRAD-BUILD code is a pathway analysis code for assessing the potential exposure dose of an individual residing or working in a building contaminated with radioactive materials, taking into account the diffusion of radioactive materials, mechanical decontamination activities, and emissions to indoor air through mechanisms such as erosion in the building structure. Code uses the exposure pathways of internal exposures through accidental ingestion and inhalation of contaminants, radon, airborne particulates, and external exposures of direct external radiation through sources, air submersion and deposited materials. The exposure mechanism of the RESRAD-BUILD code consists mainly of air quality due to the removal and transport of radioactive material inside the building and external exposure according to the various configurations of sources and receptors. The transport of radioactive materials inside the contaminated building is calculated as an indoor air quality model, which takes into account the exchange of air with outdoor air, the transport of radionuclides causing deposition and resuspension of particles, radioactive decay and ingrowth, and the release of indoor air through diffusion [7].

The Kori Unit 1 containment building is a cylindrical structure with a circular floor, a cylindrical wall and a domed ceiling, which was modeled as shown in Fig. 1 and applied to the RESRAD-BUILD code. The floor source was set to the basement of the containment building, and the dome floor located above 44.5 m above the ground level containment floor was set to the ceiling source. A total of six source areas were established by dividing the 5883 m^2 of cylindrical vertical wall area into four equal areas, each with four wall sources.

The receptor was located at the bottom center at ground level and each source was centered on all sides. The direct ingestion rate which is incidental ingestion rate of removable contaminated material directly from the source by any receptor in the room, for ceiling and floor entered as 1.15×10^{-7} h^{-1}, and 7.48×10^{-8} m^2·h^{-1} by dividing the area of each plane into the default value of building occupancy scenario, 1.1×10^{-4} m^2·h^{-1}. Indirect ingestion rate, is the rate at which an individual ingests deposited dust after it has transferred to hands, foods, or other items at each receptor location and receptor breathing rate was entered as log-uniform and triangular distribution, respectively, as a probability distribution rather than the default single value of the industrial worker occupancy scenario for the calculation of realistic dose estimates. The removable fraction was entered as 0.1 with a removable fraction of 10% as the basis for the industrial worker building occupancy scenario in NUREG/CR-5512 [8]. This is the same as the default

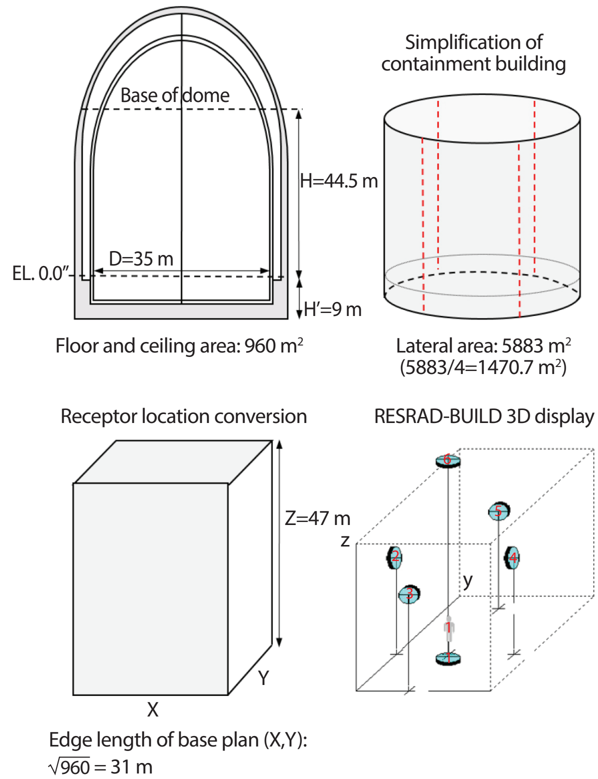

Fig. 1. Conceptual structural model of containment building.

Table 1. Interior surface DCGL for containment building of Kori site

Radionuclide	Dose conversion factor ($mSv\cdot yr^{-1}$ per 100 $dpm/100cm^2$)	DCGL ($dpm/100cm^2$)	
		20 $mSv\cdot yr^{-1}$	0.1 $mSv\cdot yr^{-1}$
^{241}Am	8.75×10^{-7}	2.29×10^{7}	1.14×10^{5}
^{60}Co	1.99×10^{-7}	1.01×10^{8}	5.03×10^{5}
^{134}Cs	1.91×10^{-7}	1.05×10^{8}	5.24×10^{5}
^{137}Cs	9.12×10^{-8}	2.19×10^{8}	1.10×10^{6}
^{152}Eu	9.80×10^{-8}	2.04×10^{8}	1.02×10^{6}
^{154}Eu	1.05×10^{-7}	1.90×10^{8}	9.52×10^{5}
^{3}H	1.48×10^{-10}	1.35×10^{11}	6.76×10^{8}
^{63}Ni	5.34×10^{-10}	3.75×10^{10}	1.87×10^{8}
^{238}Pu	1.00×10^{-6}	2.00×10^{7}	1.00×10^{5}
^{239}Pu	1.07×10^{-6}	1.87×10^{7}	9.35×10^{4}
^{240}Pu	1.09×10^{-6}	1.83×10^{7}	9.17×10^{4}
^{241}Pu	2.08×10^{-8}	9.62×10^{8}	4.81×10^{6}
^{90}Sr	1.09×10^{-7}	1.83×10^{8}	9.17×10^{5}

value for loss fraction, which is the ratio of surface contamination available for resuspension and ingestion that the DandD code uses in the same scenario [9].

3. Results and Discussion

3.1 DCGL derivation for interior surface of containment building

Through the RESRAD-BUILD modeling, DCGL for the interior surface of containment building based on industrial worker building inspection scenario was derived. The DCGL is the concentration of residual radioactivity distinguished from natural radioactivity which will result in the total effective dose equivalent to release criteria to the average member of the critical group. It is derived by dividing the dose conversion factor (DCF), which is the average total dose of the receptor calculated from RESRAD-BUILD, by the dose limit [4, 10]. DCF is a dose calculated by using a quantified value of the distribution of parameters via probabilistic analysis, which enables the identification of sensitive parameters through exposure pathway analysis in Section 3.2. Table 1 shows the DCGL derived by dividing DCF with the 20 $mSv\cdot yr^{-1}$ and 0.1 $mSv\cdot yr^{-1}$, which are effective dose limits for radiation workers and site release criteria for unlimited reuse of NPP sites in South Korea, respectively.

Based on the dose limit of the radiation worker, 20 $mSv\cdot yr^{-1}$, ^{240}Pu shows the lowest conservative DCGL value of 1.83×10^{7} $dpm/100cm^2$, which is 7,625 times the acceptable surface contamination of the Nuclear Safety Act for alpha emitting radionuclides of 0.4 $Bq\cdot cm^{-2}$. In other words, workers in building inspection scenario will be exposed to 20 $mSv\cdot yr^{-1}$ at 7,625 times and 0.1 $mSv\cdot yr^{-1}$ at 38 times the allowable surface contamination level. Rancho Seco NPP, based on the U.S. dose criteria for site release of 0.25 $mSv\cdot yr^{-1}$, also shows the most conservative DCGL of 11 $Bq\cdot cm^{-2}$ of ^{241}Am, which is 27 times the domestic allowable surface contamination level. The DCGL of the industrial worker building inspection scenario was derived as a less conservative value using a limited occupancy factor

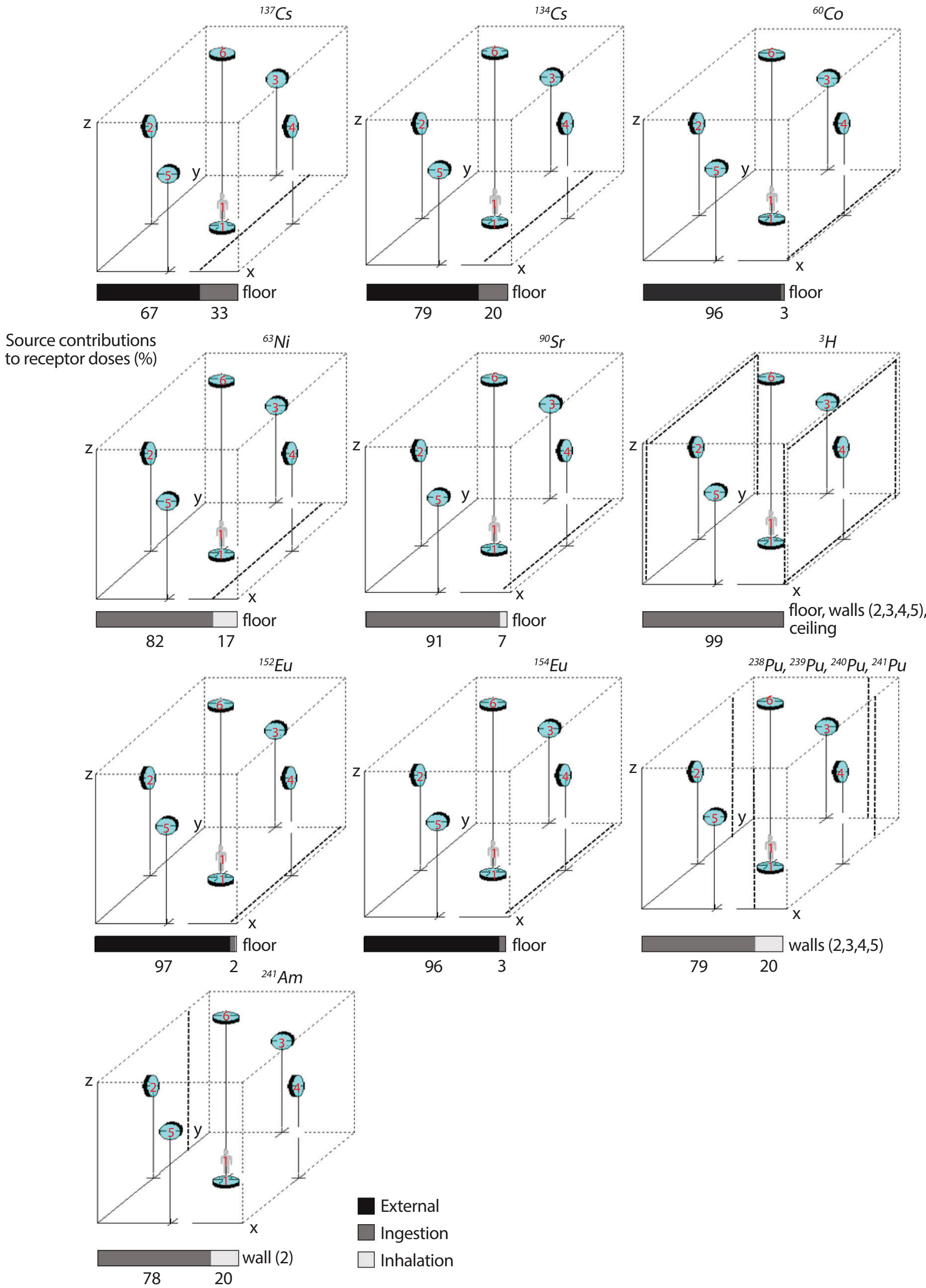

Fig. 2. Dose contribution source and exposure pathway by radionuclides in 0 year.

and considering only a single radionuclide compared to the domestic allowable surface contamination level. Unlike domestic allowable surface contamination level, DCGL is a concentration by radionuclide, calculated by applying scenarios and dose criteria using code, which is continuously updated as site-specific information is reflected via radiological surveys, and it would be desirable to use it as a degree identification and verification rather than a direct comparison with allowable surface contamination level. Further consideration of mixed contaminants will also need to be made using methodologies such as Unity rule using DCGL, radionuclide concentrations, and ratios through characterization survey samples for final dose assessment.

3.2 Analysis of radionuclide-specific exposure pathways

Instead of building occupancy scenario, a typical scenario of RESRAD-BUILD with long-term chronic exposure, a modified industrial worker building inspection scenario was used, taking into account the absence of ventilation, lighting, or power in the final state of the containment building. This is a realistic scenario with a more limited occupancy factor for industrial workers, and Rancho Seco NPP also used this scenario for containment building. 365.25 days of the maximum exposure duration in the code, and indoor fraction considering 4 days·yr^{-1} time in the containment building were applied. In addition, both Rancho Seco NPP and Hematite nuclear fuel factory derive DCGL of the building surface based on a year exposure duration for the evaluation period, which is the evaluation period recommended for probabilistic analysis by the Argonne National Laboratory (ANL). In the case of Hematite nuclear fuel factory, a one-year evaluation period was used for the general industrial worker occupancy scenario, and further assessed by applying the evaluation period considering the life of the building in accordance with the specific plan for reuse of the building [4, 11]. The evaluation period in this study was set to 0 year, which is the default value of RESRAD-BUILD, and 1 year for the industrial worker building inspection scenario, assuming four days of indoor occupancy based on one year. As Kori Unit 1 is currently undergoing a characterization survey, further studies will be needed as a specific reuse plan for the containment building is established. Radionuclide-specific dose was calculated by dividing into six major exposure pathways: direct exposure from contaminants, immersion of contaminated air, radiation exposure through the deposition, dust and radon inhalation, ingestion. Fig. 2 shows the source location and exposure pathway for each radionuclide that contributes most to the dose at 0 year. (Source 1: floor, source 2: west wall, source 3: south wall, source 4: east wall, source 5: north wall, source 6: ceiling)

Three exposure pathways, direct exposure, dust inhalation, accidental ingestion contributed most of the dose at 0 year. Radionuclides with large contributions of direct exposure to the entire dose based on radionuclides in which the maximum dose of the six sources appears from the floor sources were ^{137}Cs, ^{134}Cs, ^{60}Co, ^{152}Eu and ^{154}Eu, with ^{152}Eu contributing up to 97% and ^{137}Cs at least 67%. For ^{63}Ni and ^{90}Sr, ingestion pathway contributed 82% and 91% respectively, followed by inhalation pathway contributing less than 20% to dose. The ingestion pathway in the RESRAD-BUILD code considers secondary intake, which is the intake of contaminants deposited on walls or surface of equipment in the air, including direct intake of contaminants in the source area. The area considered for secondary intake is not limited to the source area, resulting in the largest contributing exposure pathway from many radionuclides. Since ^{3}H can be vaporized and diffused from building materials other than erosion to reach indoor air, the same dose is appeared from all surface sources, releasing only low energy beta radiation, resulting in zero external radiation [7]. In the case of ^{241}Am, a fine difference of 1.0×10^{-9} mSv with the walls in different directions showed greater doses from the western wall, but after a year, the same dose on all walls was greater than on the floor, and tended to be like Pu. Transuranic radionuclides, such as ^{241}Am, ^{238}Pu, ^{239}Pu, ^{240}Pu and ^{241}Pu were receiving

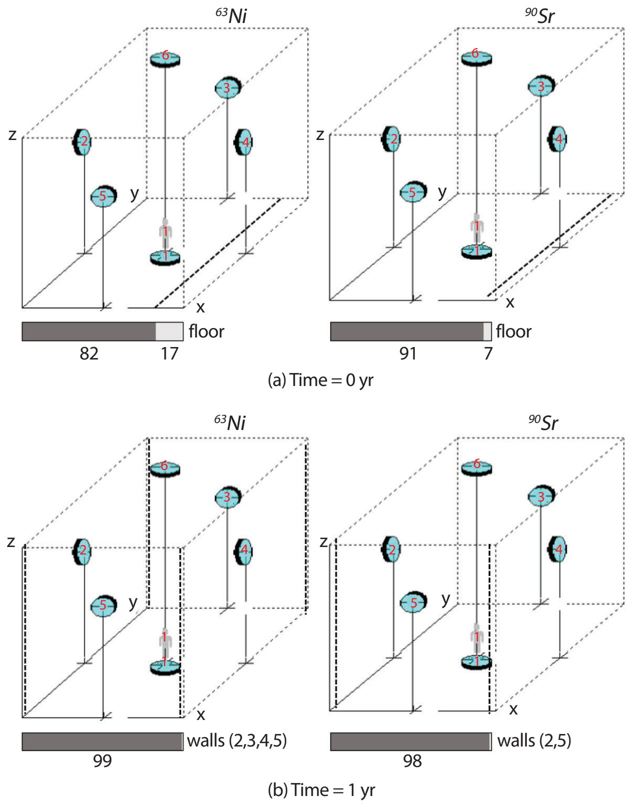

Fig. 3. Dose maximum contribution source change of ^{90}Sr and ^{63}Ni.

large doses from sources located on four walls than the floor. The floor source life derived from the parameter sensitivity analysis showed ^{137}Cs at 18,130 days and ^{240}Pu at 18,095 days. Depending on how long it takes to remove sources by radionuclide, indoor surface sources with higher doses vary. The source life is a physical and behavioral parameter. The physical parameter is source and site-specific parameter that does not change in value due to differences in receptor groups, and the behavioral parameter is parameters that depend on the scenario definition of the receptor. As in this study, if the source life is identified as a sensitive parameter through evaluation that reflects site-specific characteristics, it is judged that the removal period of surface contamination can be sufficiently adjusted according to the scenario change. At first year, the maximum contributing source and exposure pathway was equal to 0 year for most radionuclides. However, ^{90}Sr and ^{63}Ni have reduced their total dose by 23% in first year, and have changed their maximum contribution source and exposure pathway.

As shown in Fig. 3, the maximum contributing source and exposure pathway for ^{90}Sr at year 0 was the ingestion pathway from the floor source. Within a year, however, the floor source had zero doses for all exposure pathways except direct exposure pathway. Dose at 0 year of ^{63}Ni changed faster than ^{90}Sr, with dose from floor source, which are the largest contributing source, was all zero in a year. All surface sources except the floor were contributing to the overall dose with the same dose. Dose from all exposure pathways were shown to be zero, except for ingestion and inhalation pathways that contributed most to the overall dose. For beta emitters, ^{63}Ni and ^{90}Sr, where the ingestion pathway is dominant, nearly the same dose was calculated from wall sources, not from the floor in common. Although the ingestion and the inhalation pathways predominantly contribute to the dose, the source life, a behavioral and physical parameter rather than a metabolic parameter, was identified as a sensitive parameter. When applying the limited occupancy period as in this scenario, the source life, which is a physical parameter determined according to the site characteristics, was identified as the main sensitive parameter rather than the metabolic parameter characteristics that cannot be modified according to the site characteristics analysis.

4. Conclusion

In this study, industrial worker building inspection scenario was applied to the RESRAD-BUILD v3.5 code to derive DCGL for interior surface of containment building. The industrial worker building inspection scenario is being used as a realistic scenario for industrial workers who have applied a more limited occupancy factor to the building occupancy scenario, considering that there is no ventilation, lighting or power in the final state of the containment building. The containment building of Kori Unit 1 was designed as a conceptual model in the code and evaluated site-specifically. The most conservative ^{240}Pu DCGL based on both domestic site release criteria and the radiation dose limit of radiation workers, were higher than the domestic allowable surface contamination and were derived as safe values. The maximum contributing source and exposure pathway were analyzed radionuclide-specifically for a year recommended for deriving building surface DCGL using probabilistic analysis. Three exposure pathways, direct exposure from contaminants, dust inhalation, and accidental ingestion were predominant in 0 year. Depending on the source life value at 0 year, the transuranic radionuclides were making the greatest contribution to dose by the wall sources, unlike other radionuclides where the floor source was the largest source of dose contribution. In addition, except for ^{90}Sr and ^{63}Ni, the maximum contributing source and exposure pathway in 1 year was the same as 0 year. The ^{90}Sr and ^{63}Ni in one year, like the transuranic radionuclides, were all identified as radionuclides with dominant ingestion pathway, with the sources in the walls making the greatest contribution to dose. Containment buildings that are expected to be highly radioactive require various expected scenarios compared to other buildings, such as temporary look-up considering adjacent Kori Unit 2, 3 and 4 in the future as mentioned in introduction. Survey design will be developed later, using DCGL based on actual characterization survey information and scenario developed in this study, to establish criteria for classifying the impact areas according to contamination levels in the containment building and evaluating survey areas. In addition, it is expected that since the building is modeled in three dimensions and the dose is assessed, that it can be used as basic data to provide survey scope for scanning, measurement, sampling, etc. through identifying dose contribution by source locations.

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