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A Study on the Evaluation of Surface Dose Rate of New Disposal Containers Though the Activation Evaluation of Bio-Shield Concrete Waste From Kori Unit 1

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This study evaluates the radioactivity of concrete waste that occurs due to large amounts of decommissioned nuclear wastes and then determines the surface dose rate when the waste is packaged in a disposal container. The radiation assessment was conducted under the presumption that impurities included in the bio-shielded concrete contain the highest amount of radioactivity among all the concrete wastes. Neutron flux was applied using the simplified model approach in a sample containing the most Co and Eu impurities, and a maximum of 9.8×10^{4} $Bq\cdot g^{-1}$ ^{60}Co and 2.63×10^{5} $Bq\cdot g^{-1}$ ^{152}Eu was determined. Subsequently, the surface dose rate of the container was measured assuming that the bio-shield concrete waste would be packaged in a newly developed disposal container. Results showed that most of the concrete wastes with a depth of 20 cm or higher from the concrete surface was found to have less than 1.8 $mSv\cdot hr^{-1}$ in the surface dose of the new-type disposal container. Hence, when bio-shielded concrete wastes, having the highest radioactivity, is disposed in the new disposal container, it satisfies the limit of the surface dose rate (i.e., 2 $mSv\cdot hr^{-1}$) as per global standards.

Keywords: LILW, Radioactive evaluation, Decommissioned concrete waste, Disposal container

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1. Introduction

The first in Kori Unit 1 was permanently shut-down in June 2017, starting from that, a total of 12 NPPs are expected to be shut-down by 2030. Kori Unit 1, which is currently shut-down permanently, will begin decommissioning in 2026 and is scheduled to be dismantled completely in 2032. Currently, there is no experience of decommissioning commercial nuclear power plants in Korea, but only two experiences of decommissioning research nuclear power plants. Against in preparation for the decommissioning of nuclear power plants which will increase rapidly thereafter, many studies are being carried out on the characteristic evaluation of generated radioactive waste during decommissioning. During Kori 1 decommissioning, the total amount of generated radioactive concrete is expected to be 998.4 tons, of which 549.6 tons (55%) for very low level wastes and 448.8 (45%) tons for low level wastes [1].

There are advanced studies on the concrete radioactivity evaluation and neutron flux, and those were conducted several times. Based on the advanced studies, the study aims to calculate the nuclear inventory of nuclear wastes through the radioactive evaluation of the nuclear decommissioning concrete wastes, and as the results, we would want to evaluate the amount of the waste disposal containers currently being developed in the country, and the surface radiation dose rate. In addition, we would like to review whether it can satisfy the “less than 10 $mSv \cdot hr^{-1}$ on the surface” criteria presented in the current acceptance criteria, and whether it satisfies the surface dose limit (less than 2 $mSv \cdot hr^{-1}$) of overseas disposal containers.

2. Evaluation methods

2.1 Analysis on the characteristics of decommissioning wastes of overseas commercial nuclear power plants

First, overseas cases with experience in decommissioning commercial nuclear power plants are reviewed. The amount of concrete wastes accounted for 84% of the total amount of wastes at the Connecticut Yankee nuclear power plant. In the case of the Maine Yankee nuclear power plant, 63,485 tons of radioactive concrete waste were generated. Whereas in the case of Trojan nuclear power plant, it confirmed that only 284 tons were generated. As you can be seen in the case of Trojan, there is no significant correlation between the output of the reactor and the operating period of the amount of waste generated.

The above overseas cases shown can be confirmed that concrete accounts for the most of the wastes generated during the nuclear decommissioning. A number of metals are also generated as the concrete being coincided. it is difficult to predict the waste radionuclide inventory in disposal standpoint, because generated metal wastes during nuclear decommissioning will be melted by molten treatment. The radioactivity is dispersed and all residual nuclides are uniformly distributed and immobilized, so that radioactivity can be accurately measured from each ingot sample after the molten treatment.

Concrete is divided into surface-contaminated concrete and radioactive concrete. In the case of surface-contaminated concrete, it is partially reduced through decontamination before decommissioning. Therefore, the evaluation target of the study is conducted on radioactive concrete with less variation in the amount of nuclides depending on the treatment process.

2.2 Properties of radioactive concrete waste

According to domestic and overseas, among impurities contained in concrete, ^{60}Co, ^{152}Eu, and ^{154}Eu nuclides have a key effect on the production of radionuclides, so these must be considered when evaluating the radioactivity of concrete [8, 9]. Based on the analysis results of the Eu content in the domestic natural mineral aggregate conducted in Korea, the difference between the lowest and highest values was about 44 times from the lowest 0.036 ppm (Samcheok

Table 1. The overseas cases of decommissioning concrete

Reactor name	Power [MW]	Operation Year	Shut-down Year	Concrete Waste [ton]	Total Waste [ton]	Ratio [%]
Maine Yankee [2]	860	1972	1997	63,485	98,568	64%
Trojan [3]	1,095	1975	1992	284*	-	-
Haddam Neck [4]	560	1968	1996	100,539	120,388	84%
Kewaunee [5]	566	1974	2013	3,977	5,463	73%
Zion-1, 2 [6]	1040	1973	1998	139,327	223,840	62%
Oskarshamn-1 [7]	492	1972	2017	82,291	94,345	85%
Oskarshamn-2 [7]	661	1975	2016	135,445	159,422	88%

*Excluding activated volume of primary shield wall

Table 2. Impurity of bio-shield concrete

Nuclide	Impurity concentration [ppm]				
	No.1 [11]	No.2 [12]	No.3 [13]	No.4 [14]	No.5 [15]
^{59}Co	9.8	3.9	10	2.2	10
^{151}Eu	0.55	0.17	0.8	0.0019	1
^{153}Eu	0.55	0.17	0.8	0.0019	1

limestone, Gangwon) to the highest 1.587 ppm (Wando iron ore, Jeonnam), and 0.39 ppm on average [10]. As shown by this result, since the impurity content ratio varies greatly depending on the aggregate used, the radioactivity evaluation must be performed including the measured impurities as much as possible.

2.3 Radioactive evaluation method

As described in the previous section, it is necessary to evaluate the radioactivity by utilizing the impurity information of the measured concrete waste, but it is difficult to perform analysis by directly collecting samples at the current decommissioning planning stage. We used information on impurities in previous research cases, and evaluated activation of concrete using computer codes. MCNP used for calculating the neutron flux and the surface radiation dose rate, and the ORIGEN module of SACLE 6.2 used for conducting the radioactivity evaluation.

2.4 Evaluation method of surface radiation dose rate

Among the decommissioning waste disposal containers currently under development, there are two types of containers that can hold concrete wastes: medium-1 containers and large containers [16].

The medium-1 container is a metal container with a lid and floor of 5 mm thick and a side of 3 mm thick, and subject to concrete, metal, and miscellaneous low-level waste with weight below 35 tons. Radioactive concrete and miscellaneous wastes weight less than 16 tons even if 100% loaded in a medium-1 container, so the waste was modeled as packed no empty space in the container during the shielding evaluation.

The large container is a metal container with a lid and a floor of 5 mm and a side of 3 mm. Loadable wastes are subject to low-density decommissioning wastes without restriction, and the very low level waste. When the radioactivity level of the waste to be packaged is lower than the low level, such as the medium-sized container described above, the most conservative result can be obtained by conducting the shielding evaluation by assuming the radio-

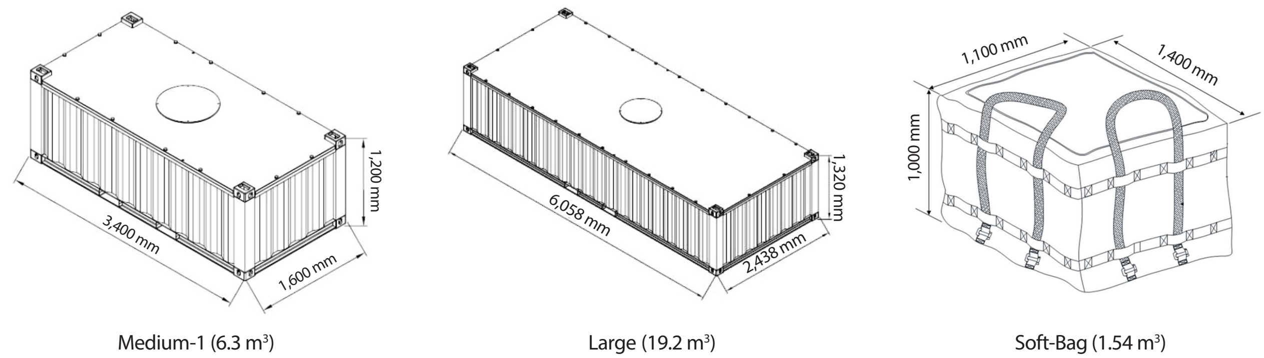

Fig. 1. The concept design of the new-type disposal containers and soft-bag.

activity concentration of the waste as limit of low level so that all low level radioactive waste can be packaged.

The soft bag was developed to be disposed of in a large type container for very low level concrete or soil waste, and its specifications are (W)1.1 m, (L)1.4 m, (H)1 m.

The MCNP 6.2 code used to evaluate the surface dose rate of the disposal container uses the Monte Carlo method, which is a stochastic/statistical method, and is a program that allows modeling and analysis similar to actual phenomenon for a three-dimensional geometric structure. The code contains a variety of nuclear reaction libraries, which are based on ENDF/B-VII, 1 libraries in MCNP 6.2. The resulting value of the MCNP code is calculated by the number of particles per unit area and the dose rate cannot be directly evaluated. Therefore, the radiation dose rate is derived using the flux to dose conversion factor after calculating the area of interest, the particle flux at the container surface and 2 m point through the calculation of the MCNP computer code. The conversion factor for the flux-dose ratio was based on the data in ICRP-74.

3. Evaluation result

3.1 Evaluation result of radioactivity

The purpose of the study is to estimate the surface radiation dose rate of a new disposal containers by conducting radiation assessment according to impurities content, so the specific MCNP modelling for Kori unit-1 was simplified and calculated. There can be differences in calculated values from other reference documents because many assumptions have been put in to simplify geometries. The inner diameter of the RV, RVIs, and bio-concrete is mostly composed of concentric circles, and only the nuclear fuel area and baffle are composed of a grid. Therefore, in this study, the model was set up by simplifying the nuclear fuel and baffle area into a circle structure equal to the area of the grid structure, similar to the method widely used when simplifying the reactor core model. As a result of the calculation, the highest 1.42×10^{12} flux to the lowest 6.73×10^{2} flux were calculated.

As such, the principal radioactive nuclei in the application of impurities in concrete materials are ^{152}Eu, ^{154}Eu, and ^{60}Co. 100% of ^{152}Eu nuclides are generated by the radiation of ^{151}Eu and 100% of ^{154}Eu nuclides are also generated by the radiation of ^{153}Eu. 100% of ^{60}Co nuclides are also generated by the radiation of ^{59}Co an impurity in concrete. In other words, the impact of impurities is dominant in the calculation of concrete radioactivity. Thus, the study was calculated by referring to the documents and applying the impurity content of the parent nuclide of the nuclide in question.

For the evaluation of radioactivity, it is necessary to review of the time when radionuclides are produced and when they are packaged and disposed of, taking into ac-

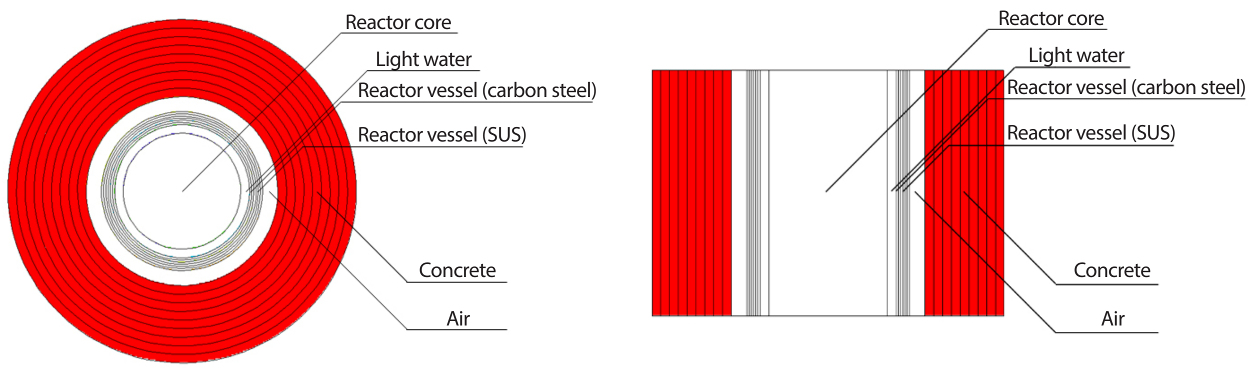

Fig. 2. Kori unit 1 RV, RVI, and bio-shield concrete geometry modeling.

Table 3. Evaluation result of neutron flux

[#·cm^{-2}·sec^{-1}]

Depth	0 cm	20 cm	40 cm	60 cm	80 cm	100 cm	120 cm	140 cm	160 cm
Flux	1.42×10^{12}	1.24×10^{11}	6.58×10^{9}	3.41×10^{8}	1.89×10^{7}	1.15×10^{6}	8.51×10^{4}	6.44×10^{3}	6.73×10^{2}

Table 4. Evaluation results of major nuclides according to impurities

	Total nuclide concentration (from internal-surface to 160 cm depth)					
Nuclide	^{60}Co		^{152}Eu		^{154}Eu	
Delay Time	P.S	+10 yr	P.S	+10 yr	P.S	+10 yr
No.1	3.59×10^{5}	9.64×10^{4}	2.41×10^{5}	1.44×10^{5}	6.82×10^{4}	3.05×10^{4}
No.2	1.44×10^{5}	3.86×10^{4}	7.45×10^{4}	4.46×10^{4}	2.11×10^{4}	9.42×10^{3}
No.3	3.66×10^{5}	9.84×10^{4}	3.50×10^{5}	2.10×10^{5}	9.93×10^{4}	4.43×10^{4}
No.4	8.06×10^{4}	2.16×10^{4}	8.32×10^{2}	4.99×10^{2}	2.36×10^{2}	1.05×10^{2}
No.5	3.66×10^{5}	9.84×10^{4}	4.38×10^{5}	2.62×10^{5}	1.24×10^{5}	5.54×10^{4}

* P.S: Permanently Shutdown

count half-life. Although Kori Unit 1 is scheduled to be approved for decommission in 2022, five years after its permanent shut-down, the decommissioning work will be carried out in earnest after the spent fuel is taken out even after the dismantlement approval is granted. Since the planned date for export of spent nuclear fuel is 2025, it is assumed that waste will be generated from 2027, 10 years after the permanent shut-down, not 2022. That is, 10 years after the permanent shutdown of Kori Unit 1, it is the concentration of the nuclide considering the half-life, which is the result calculated through ORIGEN-S.

The content of ^{60}Co, ^{152}Eu, and ^{154}Eu is as above.

3.2 Evaluation of disposal container surface dose ratio

For medium-sized containers was assumed that the total weight was 14.7 ton, large-sized containers (soft-bag) was 19.9 ton, and large-sized containers without soft-bag was 33 ton.

Table 5. Surface radiation dose rate of medium-1 container

[mSv·hr^{-1}]

Impurity	Depth								
	0 cm	20 cm	40 cm	60 cm	80 cm	100 cm	120 cm	140 cm	160 cm
No.1	6.75×10^{1}	1.33×10^{0}	6.65×10^{2}	6.04×10^{-3}	6.07×10^{-4}	6.93×10^{-5}	8.55×10^{-6}	1.03×10^{-6}	1.41×10^{-7}
No.2	2.35×10^{1}	4.50×10^{-1}	2.22×10^{-2}	2.02×10^{-3}	2.02×10^{-4}	2.31×10^{-5}	2.85×10^{-6}	3.44×10^{-7}	4.71×10^{-8}
No.3	8.57×10^{1}	1.76×10^{0}	8.87×10^{-2}	8.08×10^{-3}	8.11×10^{-4}	9.27×10^{-5}	1.14×10^{-5}	1.38×10^{-6}	1.89×10^{-7}
No.4	6.58×10^{0}	9.63×10^{-2}	4.34×10^{-3}	3.87×10^{-4}	3.84×10^{-5}	4.38×10^{-6}	5.33×10^{-7}	6.40×10^{-2}	9.01×10^{-9}
No.5	9.98×10^{1}	2.10×10^{0}	1.06×10^{-1}	9.68×10^{-3}	9.72×10^{-4}	1.11×10^{-4}	1.37×10^{-5}	1.66×10^{-6}	2.26×10^{-7}

Table 6. Surface radiation dose rate of large (soft-bag) container

[mSv·hr^{-1}]

Impurity	Depth								
	0 cm	20 cm	40 cm	60 cm	80 cm	100 cm	120 cm	140 cm	160 cm
No.1	3.98×10^{1}	7.86×10^{-1}	3.92×10^{-2}	3.56×10^{-3}	3.58×10^{-4}	4.08×10^{-5}	5.04×10^{-6}	6.09×10^{-7}	8.31×10^{-8}
No.2	1.38×10^{1}	2.65×10^{-1}	1.31×10^{-2}	1.19×10^{-3}	1.19×10^{-4}	1.36×10^{-5}	1.68×10^{-6}	2.03×10^{-7}	2.78×10^{-8}
No.3	5.06×10^{1}	1.04×10^{0}	5.23×10^{-2}	4.76×10^{-3}	4.78×10^{-4}	5.47×10^{-5}	6.74×10^{-6}	8.15×10^{-7}	1.11×10^{-7}
No.4	3.88×10^{0}	5.68×10^{-2}	2.56×10^{-3}	2.28×10^{-3}	2.27×10^{-5}	2.58×10^{-6}	3.14×10^{-7}	3.78×10^{-8}	5.31×10^{-9}
No.5	5.88×10^{1}	1.24×10^{0}	6.25×10^{-2}	5.71×10^{-3}	5.73×10^{-4}	6.54×10^{-5}	8.09×10^{-6}	9.78×10^{-7}	1.33×10^{-7}

Table 7. Surface radiation dose rate of large container

[mSv·hr^{-1}]

Impurity	Depth								
	0 cm	20 cm	40 cm	60 cm	80 cm	100 cm	120 cm	140 cm	160 cm
No.1	5.08×10^{1}	1.00×10^{0}	5.01×10^{-2}	4.55×10^{-3}	4.57×10^{-4}	5.21×10^{-5}	6.43×10^{-6}	7.78×10^{-7}	1.06×10^{-7}
No.2	1.77×10^{1}	3.39×10^{-1}	1.67×10^{-2}	1.52×10^{-3}	1.52×10^{-4}	1.74×10^{-5}	2.14×10^{-6}	2.59×10^{-7}	3.55×10^{-8}
No.3	6.45×10^{1}	1.32×10^{0}	6.67×10^{-2}	6.08×10^{-3}	6.10×10^{-4}	6.98×10^{-5}	8.60×10^{-6}	1.04×10^{-6}	1.42×10^{-7}
No.4	4.95×10^{0}	7.25×10^{-2}	3.27×10^{-3}	2.91×10^{-4}	2.89×10^{-5}	3.30×10^{-6}	4.01×10^{-7}	4.82×10^{-8}	6.78×10^{-9}
No.5	7.51×10^{1}	1.58×10^{0}	7.98×10^{-2}	7.28×10^{-3}	7.32×10^{-4}	8.35×10^{-5}	1.03×10^{-5}	1.25×10^{-6}	1.70×10^{-7}

The results of evaluating the surface dose rate of a new disposal container packed with a bio-shield concrete waste using MCNP are as above. In the case of impurities 1 to 4, it can be found that all depths except 20–40 cm depth satisfy the foreign reference value of 2 mSv·hr^{-1}. In the case of impurity 5, it was confirmed that the reference value was satisfied from 40 cm depth in the medium-1 container.

When composed of the activation waste generated in a section less than 0–40 cm from the inner surface of the bio-shield, the surface dose was analyzed to be 2 mSv·hr^{-1} or more in all containers. However, in the section of 40 cm or more, it was analyzed to be less than 2 mSv·hr^{-1}, and the deeper the depth, the very low dose was analyzed. According to reference [10], it is described that an additional shielding liner (made by concrete) is formed inside the medium-sized container according to the intensity of the surface dose, or the concentration is reduced for a sufficient time in the separate storage facility.

4. Conclusion

The study evaluated the radioactivity of concrete waste, which is expected to occur in the largest amount of nuclear power plant decommissioned waste and the surface dose rate when it is packaged in a disposal container.

First of all, the radiation assessment was conducted on the condition that impurities were included in the bio-shield concrete that is expected to contain the largest amount of radioactivity among concrete wastes. As a result, decommissioned concrete wastes except to bio-shield concrete is considered to clearance waste level. The amount of activated concrete waste generated by the decommissioning of Kori Unit 1 is determined by the evaluation of radioactivity based on the results of analysis through direct sampling collection. Accordingly, the number of new containers required will also be determined according to the amount of waste generated.

Subsequently, the surface dose rate of the container was measured assuming that the bio-shield concrete wastes generated during the decommissioning of the nuclear power plant would be packaged in a newly developed disposal container. As a result, the bio-shield concrete meeting the current domestic acceptance criteria is 247,412 kg, accounting for 52% of the total bio-shield concrete waste. In particular, it was confirmed that the outer bio-shield concrete after 120cm from the inner surface can be classified into clearance waste level.

Through the study, it was confirmed that most wastes were satisfied with the application of 2 $mSv \cdot hr^{-1}$ limit of surface dose rate of the foreign disposal containers to new disposal containers. In particular, it seems to be that the safety efficiency will be ensured if only a portion of excess waste is withdrawn for disposal after storage for a certain period of time considering half-life.

However, the inventory of impurities based on the actual measurement of wastes from Kori Unit 1 concrete, which is the most important part of the evaluation of radioactivity, has not been secured yet. Moreover, there has not been presented any ascertainable methods on the solidification requirements of radioactive wastes in the new disposal container and the internal homogeneity so far. It will thus be performed thereafter by the evaluation based on the actual measurement data on the nuclide composition of the structure and the homogeneous validity of the inside of the container due to the solidification. It is regarded as that the disposal process of nuclear decommissioning waste can be implemented safely and efficiently through above the suggestion.

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