Master's Thesis

Spatial Distribution of Radioactivity in Bioshield by using Monte Carlo Simulation for Reducing Waste Volume and External Dose during Kori unit 1 Nuclear Power Plant Decommissioning

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Abstract

Starting with Kori unit 1 on 2016, decontamination and dismantling projects (D&D project) of Hanbit 1, 2, Haneul 1, 2 and Wolsung 1, 2, 3, 4 are planned at the Republic of Korea. Specifically, since Kori unit 1 would be the first commercial nuclear power plant for the D&D project, it is more important to make adequate preparation in advance. In addition, currently, KHNP (Korea Hydro & Nuclear Power) aims for instant decommissioning (deferment with 5 years) instead of deferred decommissioning (deferment with 15 ~ 30 years). According to the raw timeline that has been announced, the preparation stage should be managed until 2024, which is the deadline for complication on SNF (spent nuclear fuel). These preparations are focused on minimization of radioactive waste during the main on-site decontamination and dismantling process, minimization of radiation exposure to workers in the facility, and radiation leakage minimization to the environment. Through this, the reduction in waste management budget can be achieved. Especially, radioactive wastes need to be tactfully managed to attain efficient management of budget requirements. Among the radioactive wastes that would be generated, concrete, which originates from the bioshield at the primary circuit that activates due to high neutron absorption, has the maximum contribution. Therefore, the exact estimation and minimization of management wastes can only be attained by assessing the radioactive inventory in the bioshield. Therefore, the exact estimation and minimization of the management waste could only be achieved from the assessment of radioactive inventory in bioshield. Looking at the case of Connecticut Yankee nuclear power plant that had been decommissioned in 1968, the United States of America which is where a large percentage of an error on pre-radioactive inventory analysis, the error caused the generation of additional 163,954 of 200 L of LLW (Low-Level Waste) that lead to increasing at 228 % of waste management budget and eventually concluded to delaying and change on whole D&D project. In order to prevent such failure, specific modeling on radioactivity inventory of Kori unit 1 has been done. Before the initiation of modeling on Kori unit 1, literature review and case study on radioactive inventory assessment in similar foreign nuclear power plants were made for identification of variables and information which should be identified on the modeling. Based on the case study, the research is objective to investigate the radioactivity inventory of the bioshield on Kori unit 1 and estimated the potential amount and cost of radioactive waste managing and give guidance to workers form external dose analysis. In addition, compared to studies on other foreign nuclear power plants, three-dimensional neutron flux distribution and nuclides behavior after the shutdown of the facility with time passes were also considered. Trojan nuclear power plant was used as benchmark for validation of the computation model for ensuring the reliability of it. On-site monitoring on radioactivity that had been initiated by IAEA (International Atomic Energy Agency) and result with MCNP 6 was compared and verified for its reliability. After the verification had been completed, specific Kori unit 1 bioshield assessment with MCNP 6 based on Monte Carlo probability theory was adopted with Boltzmann neutron transport scheme and activity of ^{60}Co, ^{152}Eu, ^{154}Eu, and ^{134}Cs which hold high sensitivity on regulation for clearance on Republic of Korea was assessed with MS-EXCEL adopting Bateman balance theory. Based on radioactivity inventory analysis, the regulation for clearance level was adopted and the amount of the potential LLW has been estimated with an additional change of them from 5 to 30 years after the shutdown of Kori unit 1. Finally, the potential external dose to workers on bioshield was classified using VISIPLAN 4.0 ALARA adopting governing balance scheme. As a result, the bioshield in Kori unit 1 showed average an 812 Bq/g of contamination with major radioactive nuclide as ^{60}Co. The clearance boundary was estimated as 425 cm from the reactor core with potential 3689 of LLW drums generation and 44 M USD would be required on managing. The average permittable working time was an average 14 hours. The research provides results within a reasonable amount of error and can be utilized as a basic tool to assess other domestic PWR nuclear power plants.

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I. Introduction

1.1 Decommissioning: The Emerging Challenges

With the sudden change in the extension of operating-time in nuclear power plant, 10 nuclear power plants are determined to be shut-down and decommissioned according to the instruction of the National Assembly Budget Office [1]. Which are Kori unit 1, 2, 3, 4, Hanbit unit 1, 2, Haneul unit 1, 2, and Wolsung unit 1, 2, 3, 4. Especially since Kori unit 1, which has been licensed on 1972.05.31 and operated since 1978.04.29 until 2017.06.18 would be the first domestic commercial nuclear power plant for undergoing decommission. According to **Figure 1**, raw timeline for Kori unit 1 decommissioning and dismantling project (D&D project) consists of four major parts. D&D basic plan design stage for handling schedule for the whole process, D&D plan licensing for taking authorization from the regulator, D&D initiating waste management with practical field works with on-site and off-site and finally the site restoration which evaluates radiation impact monitoring to an environment that completes D&D project concluded as license termination. Specific analysis for preparing decontamination and dismantling should be managed before 2024.12 that transportation of spent nuclear fuel finishes in order to get the license on starting of the D&D project [2]. Since the D&D project on Kori unit 1 demands for instant decommissioning not deferred decommissioning, most of the critical preparations are not yet been initiated. It is necessary to manage the required analysis on time to accomplish the minimization of radioactive waste, minimization of radiation impact on workers, and minimization of environmental impact with radiation which ultimately concludes to the minimization of management budget for D&D project [3].

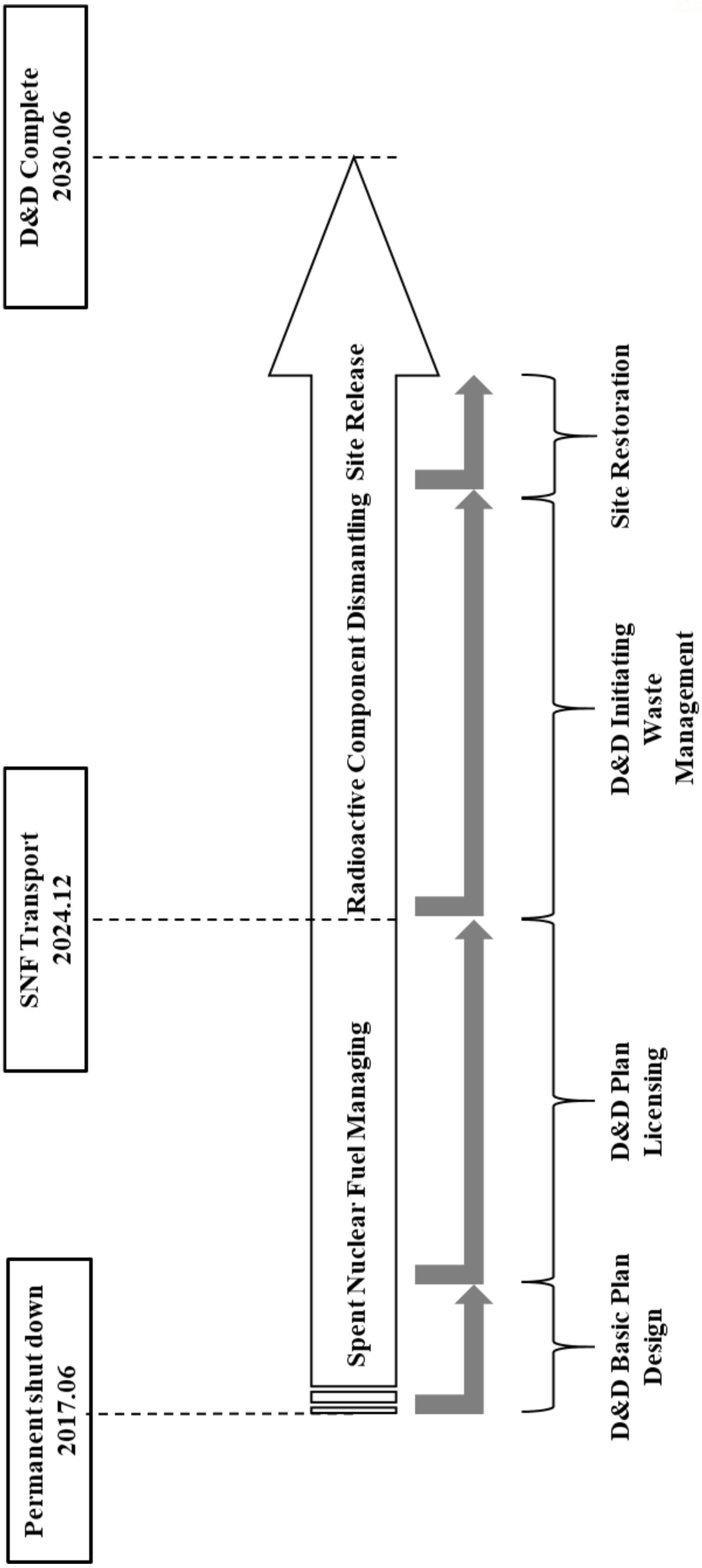


Figure 1. Raw timeline of Kori unit 1 nuclear power plant D&D project [2]

1.2 The Importance of the Radioactivity Inventory: Safety and economics

To minimize the budget for D&D project, it is crucial to determine the right scheme for each process of the project, which is consisted of dismantling, decontamination, and waste managing. For selecting the right scheme, adequate prediction is necessary, and this can be performed through pre-analysis using computational modeling before initiating the real onsite work.

1.2.1 Radioactivity Assessment on Kori unit 1 Bioshield

As shown on **Table 1**, most of the percentage of D&D project budget comprises radioactive waste managing budget [4]. Therefore, to minimize the D&D project budget, it is important to reduce the cost of waste management. As shown in **Figure 2**, the highest portion among radioactive waste, concrete contributes to a major portion, which is 75 % of the total. Most of this radioactive concrete is generated from the primary circuit of the nuclear power plant as shown in Fig 4, due to high neutron absorption along the duration of facility operating. Specifically, bioshield which holds the front defense line of the primary circuit including nuclear reactor core [5].

Table 1. Nuclear power plant D&D managing budget [4]

Process		Budget (M. USD)	Percentage (%)
Facility shut down	D&D license authorization SNF transport Facility bulkhead	122	43
Dismantling		93	34
Waste managing		62	23
Total		277	100

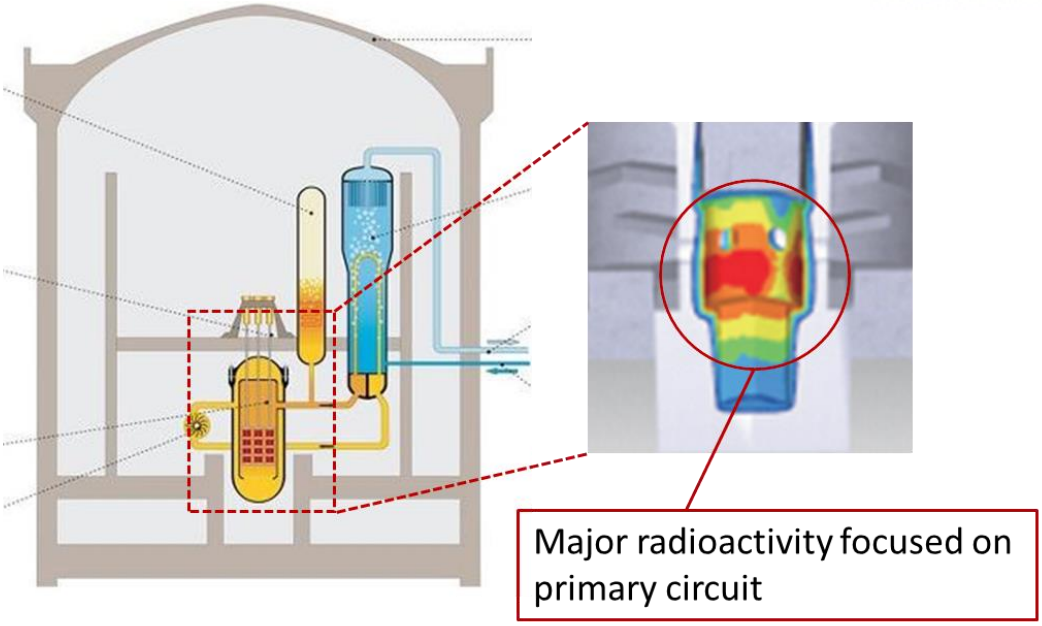


Figure 2. Major activation on the primary circuit of the nuclear power plant [5]

1.2.2 Computer Modeling on Spatial Radioactivity

As bioshield is highly activated from the neutron flux originated from the reactor core, it has a high potential risk for workers on accessing the site without pre-risk assessment. Therefore, pre-analysis using computer modeling is required. Undoubtedly, computational modeling itself requires a sensitivity test compare to that of on-site coring. However, computer modeling with reasonable state of error could enable the minimization of uncertainty on coring to targeted spot samples, which concludes the insurance of worker safety from the radioactive hazards, encouraging optimization of concrete dismantling and decontamination method, and minimizing waste generated from the process that ultimately enables minimization of the managing budget on radioactive waste handling [6]. Handling waste management budget is a crucial part in the whole D&D process, which could be found at the case for Connecticut Yankee nuclear power plant at United States of America that was decommissioned at 2006.06.17 [7]. For the Connecticut Yankee D&D project's pre-estimation of the amount of radioactive waste specifically Low-Level Waste (LLW), it occurred a huge portion of an error on prediction. Later it showed additional 163,954 of 200 L drums of LLW so that caused extra 228 % of total additional waste management budget so that delayed the entire D&D project.

II. Literature Review

2.1 Case of Radioactivity Estimation for Bioshield

As Kori unit 1 is the first commercial nuclear power plant to be decommissioned, there is limited information, from facility specification to analysis methods. To address this lack of information and determine the necessary variables for Kori unit 1 bioshield activation analysis, it is crucial to evaluate previous cases which were initiated on the globe. As shown in **Table 2,** by 2011, 129 nuclear power plants on the globe have been shut down and waiting for their decommissioning [8]. Although several nations hold experiences in the D&D project, a majority of them initiated D&D project based on experimental measurement before initiation of the dismantling and decontamination process not verifying with computer modeling. For ensuring worker safety and estimation of adequate budget on waste management, pre-analysis via computing modeling on activation is crucial. From the point of view on verification with computer modeling, major nations that hold background with Pressurized-Waster-Reactor (PWR) which is similar to Kori unit 1, are United States of America, United Kingdom, and Italy.

Table 2. Global shut down Nuclear Power Plant (2011) [8]

Country	Number of Nuclear power plant
United States of America	28
United Kingdom	26
Germany	19
France	12
Japan	9
Russian Federation	5
Ukraine	4
Italy	4
Bulgaria	4
Sweden	3
Slovak Republic	3
Canada	3
Spain	2
Lithuania	2
Switzerland	1
Netherlands	1
Kazakhstan	1
Belgium	1
Armenia	1

2.1.1 United States of America

Decommissioning in United States of America resembles most to that in the Republic of Korea. It went through the stage of changes on regulation in 1996, where the Nuclear Regulatory Commission (NRC) declared to submit a final decommissioning plan within two years after the shutdown of the facility. The final plan should indicate characterization data from the shutdown facility to planning of decommissioning activities and radioactive waste disposal activities. However, on 1996.07.29 such regulation changed for the practical application of the project initiation, and new requires does not stand for detailed characterization of the facility before the start of the deactivation activities. It only requires enough characterization for assisting worker safety during the deactivation and preparing it for decommission. The site characterization with major nuclides includes a description of remaining dismantling activities, plans for site remediation, and plans for the final license termination radiation survey [9].

United States of America has two major experiences with the D&D project of PWR with computing code pre-analysis. Trojan nuclear power plant which was shut down in 1992, and Rancho-Seco nuclear power plant which was shut down in 1989. Trojan nuclear power plant was operated from 1976 to 1992.11 with 9 Effective Full Power Year (EFPY) and holds 1095 MW(e) of generating capacity. The ANISN computing code was used for modeling the radioactivity inventory in the activated in core components, reactor vessel, and bioshield neutron flux distribution. For activation analysis, ORIGEN-2 code was adopted. Both neutron flux distribution and activation analysis were cross-checked with on-site monitoring during the decontamination procedure. As described in **Table 3**, activated levels of components on the Trojan nuclear plant were evaluated from 9 EFPY to 30 EFPY. Each EFPY cases were divided into 3 divisions which are 0 years, 10 years, and 100 years after shut-down of the facility. Case on 9 EFPY for 0 years after shut-down was evaluated as on-site monitoring of the samples and others were evaluated with computing code scheme. Each component was modeled using ANISN code as one-dimensional neutron transport analysis, which leads to only the result of bulk activation of each region of interest could be found. In addition, only total activation changes could be valid for the time period after shut-down. It is crucial to evaluate not only the total activation of the contaminated region but also the classification in each region via distance and height that evaluate clearance boundary for each component. Although Origen-2 code was used for the contribution of each radioactive nuclide level, only on the case for 9 EFPY 0 years after shut-down was analyzed. It is necessary to evaluate each major radioactive nuclide contribution via each year after shut-down in order to enhance the accuracy and reliability of classification regulatory approach [10].

Table 3. Radioactive inventory of materials in Trojan nuclear power plant (Bq) [10]

	9 EFPY			30 EFPY		
Activated Components	0 years	10 years	100 years	0 years	10 years	100 years
Core Shroud	1.11E+17	9.16E+15	1.63E+15	1.13E+17	9.32E+15	1.66E+15
Core Barrel	1.21E+16	9.98E+14	1.78E+14	2.17E+16	1.79E+15	3.19E+14
Thermal Shields	2.89E+15	2.38E+14	4.25E+13	4.82E+15	3.98E+14	7.09E+13
Vessel Inner Cladding	1.07E+15	8.83E+13	1.57E+13	4.20E+13	3.47E+12	6.17E+11
Vessel Wall	3.01E+14	2.19E+13	6.89E+11	4.33E+14	3.15E+13	9.92E+11
Upper Grid Plate	1.55E+15	1.28E+14	2.28E+13	8.03E+14	6.62E+13	1.18E+13
Lower Grid Plate	8.36E+15	6.90E+14	1.23E+14	1.82E+16	1.50E+15	2.68E+14
Bioshield	3.57E+13	2.32E+12	1.22E+11	4.45E+13	2.89E+12	1.52E+11
Contamination of Inner Surfaces	8.10E+13	1.30E+13	2.75E+11	1.80E+14	2.88E+13	6.12E+11
Totals	1.37E+17	1.13E+16	2.01E+15	1.59E+17	1.31E+16	2.33E+15

Rancho Seco nuclear power plant which was operated from 1975.3 to 1989.6 with 6 EFPY and holds 913 MW(e) of generating capacity. ANISN and ORIGEN-2 computing code were used as the same as the case of Trojan nuclear power plant. Each major component analyzed via time period after shut-down on 2 years, 11 years, 21 years, and 31 years as shown in **Table 4.** Since it was using the same one-dimensional analysis code ANSIN, the only bulk status of activation could be analyzed. Identical in Trojan nuclear power plant case, clearance boundary on each time period after shut-down was not been evaluated [11].

Table 4. Radioactivity inventory of materials in Rancho-Seco nuclear power plant (Bq) [11]

Activated Components	2 years	11 years	21 years	31 years
Core Shroud	2.74E+16	6.13E+15	2.38E+15	1.52E+15
Upper Core Barrel	2.27E+13	5.08E+12	1.97E+12	1.26E+12
Lower Core Barrel	5.56E+15	1.24E+15	4.83E+14	3.08E+14
Thermal Shields	1.76E+15	3.94E+14	1.53E+14	9.75E+13
Vessel Cladding	6.36E+12	1.42E+12	5.53E+11	3.52E+11
Vessel Wall	7.32E+13	1.64E+13	6.36E+12	4.06E+12
Control Rods/Guides	6.53E+15	1.46E+15	5.67E+14	3.62E+14
Top Grid/Plenum	1.84E+16	4.12E+15	1.60E+15	1.02E+15
Lower Forging	1.36E+16	3.04E+15	1.18E+15	7.53E+14
Orifice Rods/Retainers	6.38E+14	1.43E+14	5.54E+13	3.53E+13
Burnable Poison Rods	1.27E+16	2.84E+15	1.10E+15	7.04E+14
Bioshield	1.91E+13	3.53E+12	1.23E+12	6.53E+11
Contaminated Inner Surfaces	1.21E+14	1.25E+13	2.00E+12	8.30E+11
Total	8.72E+16	1.95E+16	7.57E+15	4.82E+15

2.1.2 United Kingdom

In the case of the United Kingdom, statutory and economic value dominate characterization program for decommissioning. Characterization of the program includes waste quantities estimation and classification under national categories VLLWs (Very Low-Level Wastes), LLWs (Low-Level Wastes), and ILWs (Intermediate Level Wastes) followed by selecting appropriate storage and disposal method. The classification of waste categories could be defined from pre-analysis categorization and hence control the costs and safety of packaging, storage, and disposal. The inventory of radioactive materials is determined in order to enable radiation fields to be estimated during the dismantling procedure so that such work can be conducted according to ALARA (As Low as Reasonably Achievable) principle [12].

Although the United Kingdom has no experience of D&D project with computer modeling pre-analysis on PWR but holds experience on the gas cooled reactor. Windscale Advanced Gas Cooled Reactor (WAGR) was operated from 1963 to 1981. ANSIN neutron transport code was used on neutron flux calculation on the void region [13]. Especially different form PWR, neutron streaming in the void region. Result of WAGR remains as overall components on activities. As shown in **Table 5**, ^{55}Fe and ^{60}Co were the majority on all components and ^{3}H, ^{60}Co, and ^{152}Eu were majored specifically, on reinforced concrete. Different from the case on Trojan and Ranco-Seco on the United States of America, analysis on each component was not been done but monitoring on whole waste was initiated. Due to using of one-dimensional analysis code ANSIN, which holds limit on geometry specification, leaded activity data on the only classification to which part holds the most impact.

Table 5. Radioactivity inventory of materials in WAGR (TBq) [13]

Radionuclides	Activity
^{3}H	44
^{14}C	4.7
^{36}Cl	0.088
^{41}Ca	0.121
^{54}Mn	0.004
^{55}Fe	1858
^{59}Ni	6.8
^{60}Co	692
^{63}Ni	698
^{93m}Nb	0.168
^{94}Nb	0.042
^{152}Eu	1.12
^{154}Eu	1.39
^{155}Eu	0.37
Total	3306

2.1.3 Italy

D&D project on Italy is mainly focused on safe containment of the facility which is different from the United States of America that holds focus on safe license termination of the site. Which is why the related regulation stands for minimum two-dimensional assessment for neutron flux and activity of the contaminated site with a reasonable amount of time duration after the shutdown and decommissioning of the facility [14]. Although most of the nuclear power plants on Italy are still on their process of extensive sampling, the D&D cases on Trino and Caorso nuclear power plant hold meaning that their usage of two-dimensional analysis approach which did not initiate in the United States of America and the United Kingdom.

Trino nuclear power plant is a pressurized water reactor type that operated from 1964 to 1987 with 870 MW(e). Its analysis on neutron flux was based on two-dimensional computing code DOT 3.5 and the activity of the contaminated site was evaluated with Origen-S computing code. As shown in **Table 6**, the total radioactivity of the waste was evaluated with diverse radioactive nuclides. In addition, as shown in **Table 7**, activity on bioshield was evaluated under ^{60}Co, ^{134}Cs, ^{152}Eu, and ^{154}Eu were used as comparison radioactive nuclides. Same as other activity analysis on the United States of America and the United Kingdom, no spatial analysis was done, which led no result on the identification of clearance boundary nor amount of the waste amount [15].

Table 6. Calculated activities from neutron activation in Trino nuclear power plant (TBq) [15]

Radionuclides	Activity
^{3}H	2.8
^{14}C	0.06
^{36}Cl	0.02
^{39}Ar	0.84
^{41}Ca	0.0002
^{54}Mn	0.49
^{55}Fe	835
^{60}Co	983
^{59}Ni	6.0
^{63}Ni	700
^{90}Sr	0.0005
^{93}Mo	0.0006
^{93m}Nb	0.03
^{94}Nb	0.02
^{108}Ag	0.04
^{108m}Ag	0.4
^{133}Ba	0.001
^{134}Cs	0.9
^{151}Sm	0.02
^{152}Eu	0.01
^{154}Eu	0.2
^{155}Eu	0.04
^{166m}Ho	0.000007
Total	2560

Table 7. Radioactivity inventory of a Trino for major components (Bq) [15]

Components	Activity
Internals	4.27E+15
Control rods	1.16E+15
Vessel	3.52E+14
Neutron shield	2.47E+12
Biological shield	9.39E+9
Total	5.7E+15

Caorso nuclear power plant is a Boiling-Water-Reactor (BWR) that operated from 1979 to 1986 with 2590 MW(e). Same with the analysis on Trino nuclear power plant, DOT 3.5 computing code was adopted as two-dimensional analysis on neutron flux and Origen-S computing code was used for activity assessment. As shown in **Table 8** and **9**, the average radioactivity of major nuclides was assessed under the condition of decades after the shutdown of the nuclear power plant [16].

Table 8. Calculated activities from neutron activation in Caorso nuclear power plant (TBq) [16]

Radionuclides	Activity
^{3}H	3.0
^{14}C	0.7
^{36}Cl	0.01
^{39}Ar	15.8
^{41}Ca	0.001
^{54}Mn	0.7
^{55}Fe	1971
^{60}Co	4375
^{59}Ni	4.7
^{63}Ni	554
^{90}Sr	0.004
^{93}Mo	0.01
^{93m}Nb	0.3
^{94}Nb	0.009
^{108}Ag	0.002
^{108m}Ag	0.06
^{133}Ba	0.09
^{134}Cs	17.2
^{151}Sm	0.09
^{152}Eu	0.4
^{154}Eu	6.5
^{155}Eu	1.7
^{166m}Ho	2.2
Total	7680

Table 9. Radioactive inventory of a Caorso for major components (Bq) [16]

Components	Activity
Internals	1.28E+16
Fuel cases	2.44E+15
Control rods	3.4E+15
Reactor pressure vessel	3.84E+13
Sacrificial shield	6.18E+11
Biological shield	3.33E+9
Dry well	1.51E+10
Total	1.8E+16

Overall, all nations showed a lack of spatial distribution of radioactivity. Since the regulations on D&D pre-analysis of the United States of America and the United Kingdom require only a one-dimensional approach on the region of interests, their result on radioactivity assessment only concluded to the summation of total radioactivity. Although the regulation on Italy demands two-dimensional analysis on neutron flux, there is limited information of radioactivity on the specific region of interests. Since the regulation on Italy focused on containment of the contamination, the radioactivity inventory was assessed after the decontamination and dismantling not on-site during the dismantling process or before the decontamination process.

2.2 Improvements Needed: Connection to Safety and Waste

Concluded from the case reviews on bioshield activation analysis of foreign major nuclear power plants decommissioning, it is clear that total radioactivity of the site is different from site to site as shown in **Table 10**. The radioactivity was diverse due to their difference in the specification such as geometry, reactor type, power density, and operation history. Therefore, it is possible to reference same PWR decommissioning case but specific modeling on Kori unit 1 is required. Especially spatial distribution of radioactivity assessment is necessary in order to estimate the clearance boundary which leads to optimization of the amount of the radioactive waste and estimate accurate external dose for ensuring worker safety. As shown in **Table 11**, specifically three-dimensional neutron transport, clearance boundary, external dose, waste managing budget, and time duration analysis should be targeted variables to be clarified from this research.

Table 10. Radioactivity inventory of bioshield on nuclear power plants [Bq]

Facility	Reactor type	Activity
Trino	PWR	9.39E+09
Trojan	PWR	4.45E+13
Rancho Seco	PWR	1.91E+13
Caorso	BWR	3.33E+09
Kori-1	GCR	3.30E+03

Table 11. Targeted analysis point on the research: improvements needed from previous studies

Facility	Researcher	3D neutron transport analysis	Time duration analysis	Clearance boundary analysis	External dose analysis	Waste managing budget analysis
Trino	IAEA	×	×	×	×	×
Trojan	Portland general electronic	×	○	×	×	×
Rancho Seco	IAEA	×	×	×	×	×
Caorso	IAEA	×	×	×	×	×
WAGR	IAEA	×	×	×	×	×
Kori-1	Donghyun Lee [UNIST]	○	○	○	○	○
Trino	IAEA	×	×	×	×	×

III. Research Design

3.1 Research Objectives and Approaches

Based on the literature reviews and case studies associated with bioshield activation analysis among major foreign nuclear power plants, it is possible to conclude the concept and design flow of the research. This study is objective to analyze the Kori unit 1 bioshield spatial activation using computational modeling. It is aimed to first identify and clarify spatial neutron flux distribution using MCNP 6 than calculate the spatial radioactivity using MS-EXCEL function with neutron flux result. In addition, external dose analysis using VISIPLAN 4.0 in order to assess the worker safety and finally leads to the estimation of the clearance boundary on bioshield with assessing the amount of the waste and its managing budget as well.

As shown in **Figure 3**, the flow of the research could be divided into three steps. First, the preparation of the research that includes a literature review and case study on pressurized water reactor (Trino, Trojan, and Rancho-Seco), boiling water reactor (Caorso), and gas-cooled reactor (WAGR) in order to identify the objective and aim of the research. In addition, for validation of the computational model which would be used on modeling of the Kori unit 1, the validation is made by comparing the bioshield radioactivity between on-site monitoring result and the result of MCNP 6 for Trojan nuclear power plant. After the validation of the modeling scheme finishes with reliability verified, the second part of the study is initiated. The second main work of the research includes three-dimensional modeling on Kori unit 1 with facility configurations and operation history, and activation analysis with targeted radioactive nuclides. Finally, the third part, three major variables that crucial on preparation for decommissioning would be discussed, which are the classification of clearance boundary, assessment of the radioactive waste amount, and evaluation of external dose rate to workers.

The targeted radioactive nuclides are ^{60}Co, ^{152}Eu, ^{154}Eu, and ^{134}Cs, which hold the highest sensitivity on the regulation of radioactive waste clearance level on Republic of Korea [18]. As shown in **Table 12**, except for ^{129}I, their limitation on clearance is 0.1 Bq/g which is the highest level of sensitivity interests. In addition, for the case of multiple nuclides exists at the same time, the total summation percentage should be below 1 as shown in **Equation 1** [19].

$$\sum_i \frac{C_i}{C_{L,i}} < 1 \tag{1}$$

Here, C_i is the radioactivity of nuclide i (Bq/g) and $C_{L,i}$ is the radioactivity limitation of nuclide i (Bq/g).

Table 12. Clearance level on radioactive nuclide in the Republic of Korea [18]

Radioactive nuclide	Clearance Level (Bq/g)
^{129}I	0.01
^{60}Co, ^{152}Eu, ^{154}Eu, ^{134}Cs	0.1
^{14}C, ^{59}Fe, ^{90}Sr	1
^{7}Be, ^{18}F	10
^{3}H, ^{35}S	100
^{31}Si, ^{32}P	1,000
^{58m}Co, ^{71}Ge	10,000

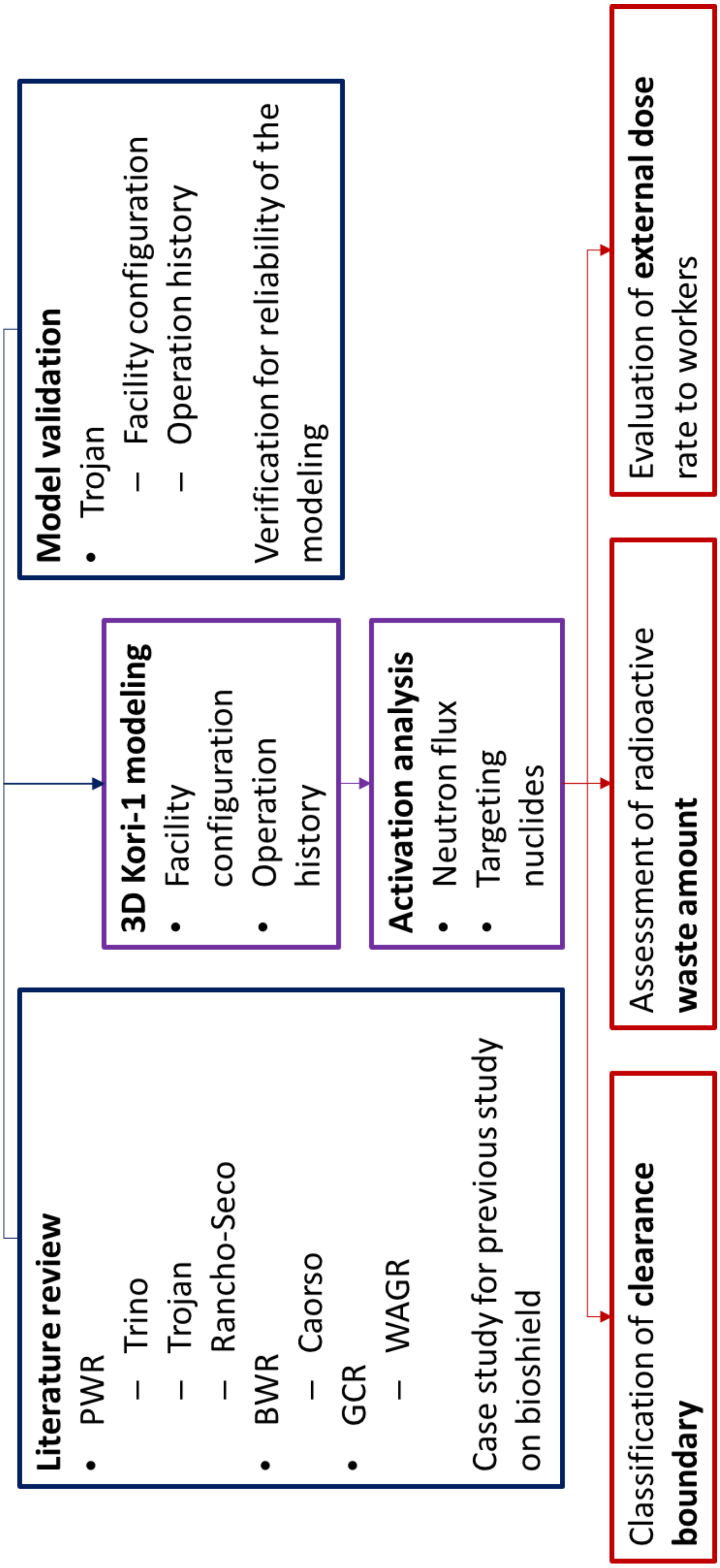


Figure 3. Research flow

3.2 Mathematical Equation and Numerical Methods

The main part of the research which is the three-dimensional modeling of Kori unit 1 and radioactivity inventory analysis requires three major parts of mathematical principles. First, neutron transport should be described from the defining of source term which is the reactor core shown in **Figure 4,** using the Boltzmann neutron transport equation [19]. The reactor core which is the ground zero of the neutron generation has been simplified as cylindrical volume. Watt fission spectrum was chosen for describing the flux of the particles as shown in **Equation 2**.

$$\left[\frac{1}{V}\frac{\partial}{\partial t} + \Omega\nabla + \sum(r,E,t)\right]\emptyset(r,\Omega,E,t) = \int_0^\infty dE' \int_{4\pi} d\Omega' \sum_s(r,\Omega'\Omega,E' \rightarrow E)\emptyset(r,\Omega',E',t) + \frac{\chi(E)}{4\pi}\int_0^\infty dE' \int_{4\pi} d\Omega' v(E') \sum_f(r,E',t)\emptyset(r,\Omega',E',t) + Q(r,\Omega,E,t) \quad (2)$$

As shown in **Figure 4**, in this stage of the work, MCNP 6 (General purpose Monte Carlo N-Particle code) was adopted. Kori unit 1 specification including geometry and material properties of each part on the structure, operation history which is EFPY (Effective Full Power Year), and power density were used as input of the code. From the usage of MCNP 6 spatial distribution of neutron flux could be detected. In order to match the minimum relative error 1% that is recommended for giving reliability under the level of possible usage on regulators, NPS (number of particles) was managed as 1×10^9.

Second, radioactivity inventory should be analyzed using the Bateman equation in **Equation 3** [19]. MS-EXCEL was adopted. For input of the stage, neutron flux which is generated from the MCNP 6, Bioshield specification including geometry and material properties were used.

$$\frac{dN_1}{dt} = -\lambda_1 N_1,\ \frac{dN_i}{dt} = \lambda_{i-1}N_{i-1} - \lambda_i N_i \quad (3)$$

Specifically, after modeling of neutron flux distribution analysis using MCNP 6 was completed, activation of nuclides on bioshield should be initiated. Bateman equation which stands equilibrium of radioactive nuclide on generation and decay is:

$$-\mathrm{dn(t)} = \lambda \mathrm{n(t)dt} \quad (4)$$

where -dn(t) is the rate of nuclides change on time t; λ is the decay constant (hr-1). From integration on both side of the balance on t, the equation could be changed as

$$n(t) = n_0 e^{-\lambda t} \quad (5)$$

radioactive nuclide's decay consists of decay term and production:

$$\frac{dn}{dt} = -\lambda n + R \tag{6}$$

where R is the term for generated nuclides, the total balance of the nuclides during the radioactive decay could be shown as:

$$n = n_0 e^{-\lambda t} + \frac{R}{\lambda}(1 - e^{\lambda t}) \tag{7}$$

From this stage of the work, it is possible to identify the radioactivity of the targeted radioactive nuclides which are ^{60}Co, ^{152}Eu, ^{154}Eu, and ^{134}Cs.

Third, the external dose rate should be identified using Governing equation on **Equation 8** [19]. VISIPLAN 4.0 computation code was used. The radioactivity converted into potential dose rate via dose conversion factor of each nuclide [20]. The input of the stage is the radioactivity of the targeted nuclides that were identified at the previous stage with the usage of MS-EXCEL.

$$\frac{\partial n}{\partial t} = s - \Sigma_a \phi - \nabla . J \tag{8}$$

3.3 Defining of Source Term

In order to describe neutron behavior, it is crucial to design adequate model for source term. The research contains two source terms. First, at the stage on radioactivity inventory assessment of the bioshield, it would be reactor core that has been simplified as volumetric cylinder. Effective full power years were chosen as input parameter instead of entire operation history in order to make conservative approach for reinforcing reliability of the result. Second, at the stage on assessing potential external dose to workers, the source term would be activated bioshield itself since the reactor core already should be removed before starting of dismantling and decontamination process on the bioshield. The bioshield was modeled as homogenous material property since exact blue print of reinforcing steel bar could not be identified. In addition, the impurity concentration was assumed in order to clarify the pre-material condition of the bioshield.

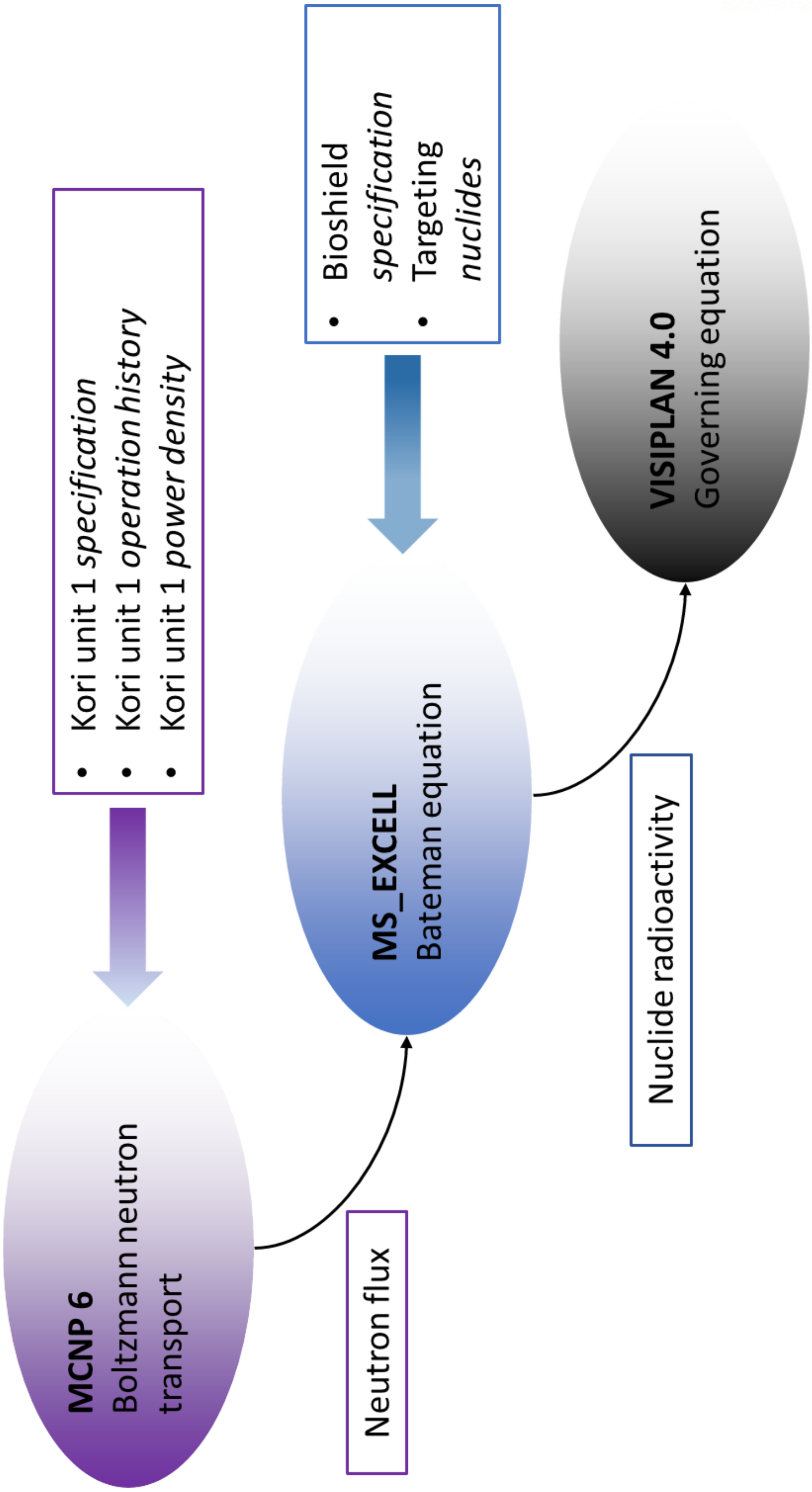


Figure 4. Numerical computation stage of the research

IV. Method Validation against Trojan Nuclear Power Plant

4.1 Trojan Modeling Condition

Since there is no pre-assessment case on domestic pressurized water reactor case study on foreign similar nuclear power plants had been done, as shown in **Table 13**, and **Figure 5**, from the case study, Trojan nuclear power plant showed the highest similarity with Kori unit 1. Specifically, reactor type as PWR, and geometry of the structures. Validation on MCNP 6 general-purpose Monte Carlo N-Particle code scheme was taken from the assessment on Trojan nuclear power plant. MCNP 6 was used for assessment of activation on bioshield. Structure from the reactor core to bioshield has been designed. Due to the lack of information on the exact blueprint of Trojan, rough design parameters were decided. Although the targeted area for evaluation is bioshield, whole structure on the primary circuit should be designed, since the neutron absorption effect should be considered on each part of the structure. The reactor core, barrel, bypass, pressure vessel, air, and bioshield was shaped as cylindrically. Specifically, bioshield was designed with two cylinders with two different diameters and height for more exact reflection of real geometry. Detailed input information on each structure was density, volume and nuclides composition as in material composition shown in **Table 14**, with geometry information shown in **Table 15**. Each structure was assumed as a homogeneous material status with no hole or gap inside [21].

Table 13. Modeling configurations on Trojan and Kori unit 1 [21]

Facility	Trojan	Kori unit 1
Bioshield radius	520 cm	530 cm
Bioshield height	1485 cm	1485 cm
EFPY	9 year	30 year
Power density	1095 Mw(e)	576 Mw(e)

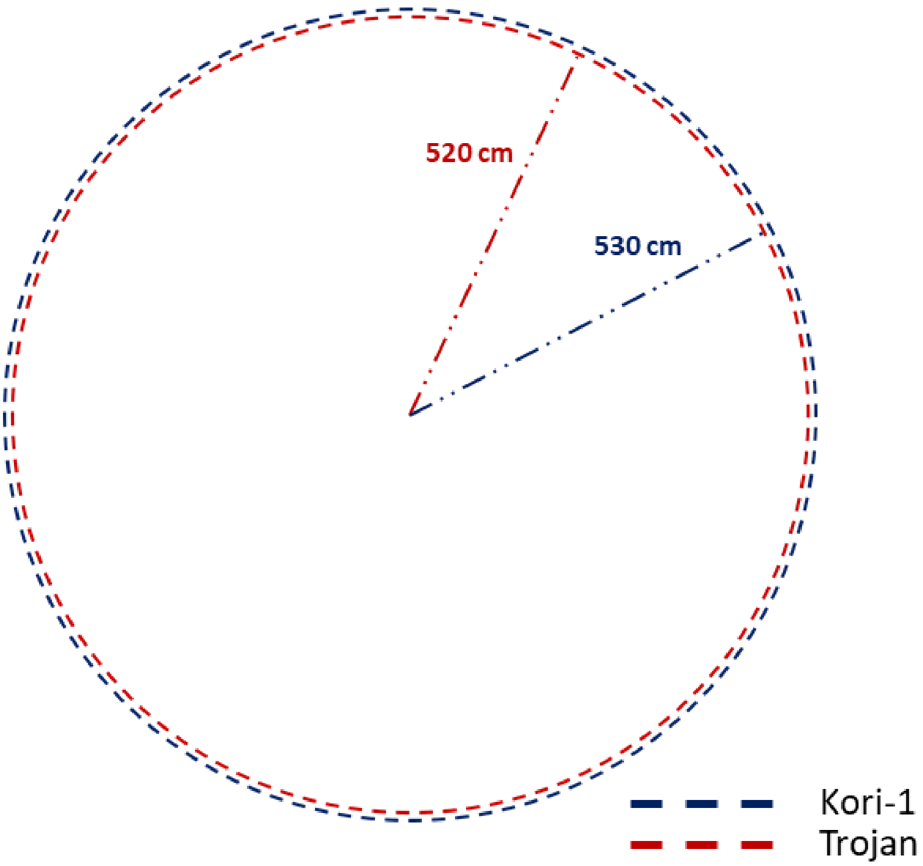


Figure 5. Radius comparison on Kori unit 1 and Trojan nuclear power plant [21]

Table 14. Trojan structure material/physical properties [g/g] [10]

Nuclei	Carbon Steel	Stainless Steel	Concrete	Nuclei	Carbon Steel	Stainless Steel	Concrete
U	2.00E-07	2.00E-07	2.70E-06	Ni	6.60E-03	6.00E-03	3.80E-05
Th	1.80E-07	1.80E-07	3.50E-06	Co	1.22E-04	1.90E-04	9.80E-06
Pb	8.20E-04	8.20E-04	6.10E-05	Fe	9.84E-01	9.84E-01	1.40E-02
W	5.50E-06	5.50E-06	1.40E-06	Mn	1.00E-02	1.27E-02	3.77E-04
Ta	1.30E-07	1.30E-07	4.40E-07	Cr	1.70E-03	4.80E-04	1.09E-04
Hf	2.10E-07	2.10E-07	2.20E-06	V	8.00E-05	2.00E-06	1.03E-04
Lu	2.00E-07	2.00E-07	2.70E-07	Ti	2.00E-06	2.00E-06	2.12E-03
Yb	1.00E-06	1.00E-06	1.40E-06	Sc	2.60E-07	2.60E-07	6.50E-06
Ho	8.00E-07	8.00E-07	9.00E-07	Ca	1.40E+05	1.40E-05	4.40E-02
Dy			2.30E-06	K	1.20E-05	1.20E-05	1.30E-02
Tb	4.50E-07	4.50E-07	4.10E-07	Cl	4.00E-05	4.00E-05	4.50E-05
Eu	3.10E-08	3.10E+08	5.50E-07	S	5.00E-04	1.60E-04	3.10E-03
Sm	1.70E-08	1.70E-08	2.00E-06	P	5.00E-04	1.10E-04	5.00E-03
Ce	1.00E-06	1.00E-06	2.43E-05	Si		2.00E-03	3.37E-01
La	1.00E-07	1.00E-07	1.30E-05	Al	3.30E-04	2.10E-04	3.40E-02
Ba	2.73E-04	2.73E-04	9.50E-04	Mg			2.00E-03
Cs	2.00E-07	2.00E-07	1.30E-06	Na	2.30E-05	2.30E-05	1.60E-02
Sb	1.10E-05	1.10E-05	1.80E-06	O			5.29E-01
Sn	7.00E-06	7.00E-06	7.00E-06	N	8.40E-05	7.00E-06	1.20E-04
Cd			3.00E-07	C	2.90E-03	2.10E-03	1.00E-03
Ag	2.00E-06	2.00E-06	2.00E-07	B			2.00E-05
Pd			3.00E-06	Li	3.00E-07	3.00E-07	2.00E-05
Mo	5.60E-07	4.30E-03	1.03E-05	H			1.00E-02
Nb	1.88E-05	1.88E-05	4.30E-06				
Zr	1.00E-05	1.00E-05	7.10E-05				
Y	2.00E-05	2.00E-05	1.82E-05				
Sr	1.50E-07	1.50E-07	4.38E-04				
Rb	4.80E-05	4.80E-05	3.50E-05				
Br	8.50E-07	8.50E-07	2.40E-06				
Se	7.00E-07	7.00E-07	9.20E-07				
As	5.32E-04	5.32E-04	7.90E-06				
Ga	8.00E-05	8.00E-05	8.80E-06				
Zn	1.00E-04	1.00E-04	7.50E-05				
Cu	2.00E-03	1.50E-03	2.50E-05				

Table 15. Specific design factors of Trojan [10]

Cell	Distance from the core (cm)
Core	138
Barrel	188
Bypass	198
Pressure vessel	219
Air	307
Concrete	520

4.2 Radioactivity Assessment Validation

After the modeling of the Trojan using MCNP 6, the radioactivity of the targeted nuclides on bioshield was assessed for the validation of the reliability of the MCNP 6 computation modeling. Before initiating the modeling on Kori unit 1 bioshield, it is necessary to verify the modeling with a reasonable margin of error [22, 23]. The difference ratio between computer modeling result at IAEA versus on-site monitoring and MCNP 6 modeling result versus on-site monitoring was compared.

As shown in **Figure 6**, and **Table 16**, the on-site monitoring result on ^{60}Co was maximum 7.03E+03 Bq/g on 304.75 cm region to minimum 7.03E-03 Bq/g on 516.75 cm point. IAEA computation modeling showed maximum 1.11E+04 Bq/g on 304.75 cm, and minimum 2.18E+01 Bq/g on 410.75 cm. Although the on-site monitoring result showed its minimum radioactivity at 516.75 cm, IAEA modeling was unable to achieve its radioactivity on the same region due to the limit of detection on ANSIN code. With the case of MCNP 6 the maximum radioactivity showed as 9.49E+03 Bq/g on 304.75 cm, and minimum 1.22E-02 Bq/g on 516.75 cm. The average difference ratio on IAEA modeling and MCNP 6 showed +89.6 % and +57.3 %.

As shown in **Figure 7**, and **Table 17**, the on-site monitoring result on ^{152}Eu was maximum 9.25E+03 Bq/g on 304.75 cm region to minimum 8.51E-03 Bq/g on 516.75 cm point. IAEA computation modeling showed maximum 1.07E+04 Bq/g on 304.75 cm, and minimum 2.55E+01 Bq/g on 410.75 cm. Although the on-site monitoring result showed its minimum radioactivity at 516.75 cm, IAEA modeling was unable to achieve its radioactivity on the same region due to the limit of detection on ANSIN code. With the case of MCNP 6 the maximum radioactivity showed as 1.17E+04 Bq/g on 304.75 cm, and minimum 1.35E-02 Bq/g on 516.75 cm. The average difference ratio on IAEA modeling and MCNP 6 showed +50.0 % and +33.2 %.

As shown in **Figure 8**, and **Table 18**, the on-site monitoring result on ^{154}Eu was maximum 9.99E+02 Bq/g on 304.75 cm region to minimum 3.37E-02 Bq/g on 463.75 cm point. IAEA computation modeling was unable to detect the radioactivity on the whole region of bioshield since its radioactivity is lower than the limit of detection. However, MCNP 6 showed maximum radioactivity as 1.26E+03 Bq/g on 304.75 cm, and minimum as 5.39E-02 Bq/g at 463.75 cm. Although the difference ratio with modeling on IAEA could not be compared, the average difference with MCNP 6 versus on-site monitoring showed +40.4 % which was lower than that of ^{60}Co case.

As shown in **Figure 9**, and **Table 19**, the on-site monitoring result on ^{134}Cs was maximum 3.52E+02 Bq/g on 304.75 cm region to minimum 7.40E-03 Bq/g on 463.75 cm point. IAEA computation modeling was unable to detect the radioactivity on the whole region of bioshield since its radioactivity is lower than the limit of detection. However, MCNP 6 showed maximum radioactivity as 4.26E+02 Bq/g on 304.75 cm, and minimum as 1.03E-02 Bq/g at 463.75 cm. Although the difference ratio with modeling on IAEA could not be compared, the average difference with MCNP 6 versus on-site monitoring showed +29.2 % which was lower than that of ^{60}Co case.

As shown in **Figure 10 and Table 20**, the maximum difference ratio on ^{60}Co, ^{152}Eu, ^{154}Eu, and ^{134}Cs were compared as 100 % criteria and compared with IAEA ORIGEN-2 and MCNP 6 computation modeling. Overall, MCNP 6 modeling showed higher sensitivity on all radioactive nuclides since IAEA ORIGEN-2 showed relatively conservative result on radioactivity compared to MCNP 6. Comparing from targeted radioactive nuclides, MCNP 6 showed a reasonable margin of difference so that could be converted on the modeling on Kori unit 1.

Table 16. ^{60}Co radioactivity concentration

	IAEA		Donghyun Lee	Difference ratio	
	ORIGEN-2 (1)	Measured (2)	MCNP 6 (3)	(1) vs (2)	(3) vs (2)
304.75	1.11E+04	7.03E+03	9.49E+03	+57.9	+33.2
357.75	9.25E+02	8.14/E+01	1.12E+02	+104.0	+38.0
410.75	2.18E+01	1.15E+01	1.71E+01	+89.6	+49.0
463.75	N/A	2.11E-01	3.25E-01	N/A	+54.5
516.75	N/A	7.03E-03	1.22E-02	N/A	+74.1

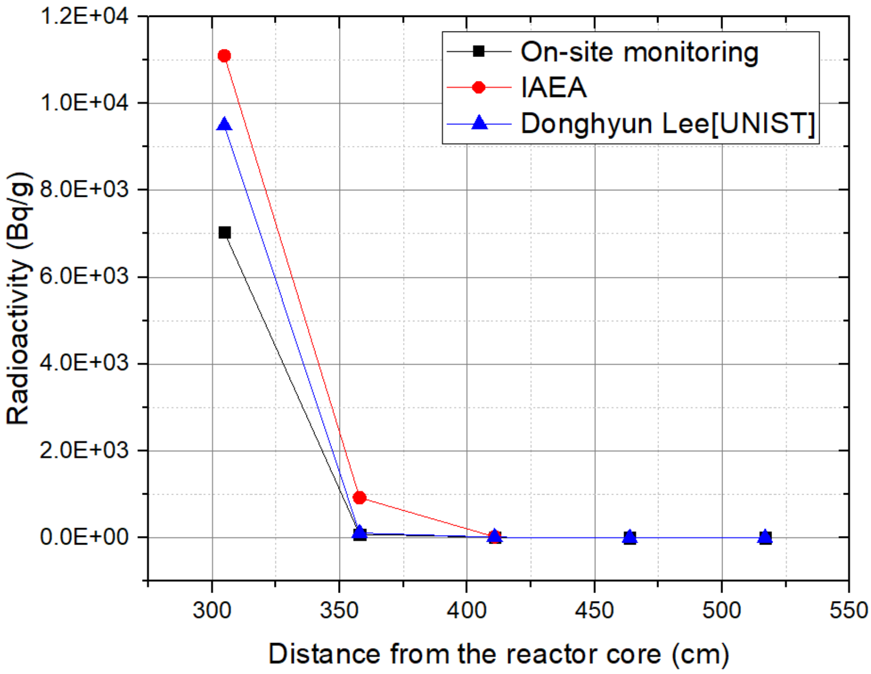


Figure 6. Radioactivity comparison on ^{60}Co of Trojan nuclear power plant

Table 17. ^{152}Eu radioactivity concentration

	IAEA		Donghyun Lee	Difference ratio	
	ORIGEN-2 (1)	Measured (2)	MCNP 6 (3)	(1) vs (2)	(3) vs (2)
304.75	1.07E+04	9.25E+03	1.17E+04	+15.7	+26.5
357.75	1.07E+03	1.04E+02	1.35E+02	+92.9	+30.3
410.75	2.55E+01	1.70E+01	2.31E+01	+50.0	+36.2
463.75	N/A	2.96E-01	4.32E-01	N/A	+46.0
516.75	N/A	8.51E-03	1.35E-02	N/A	+59.1

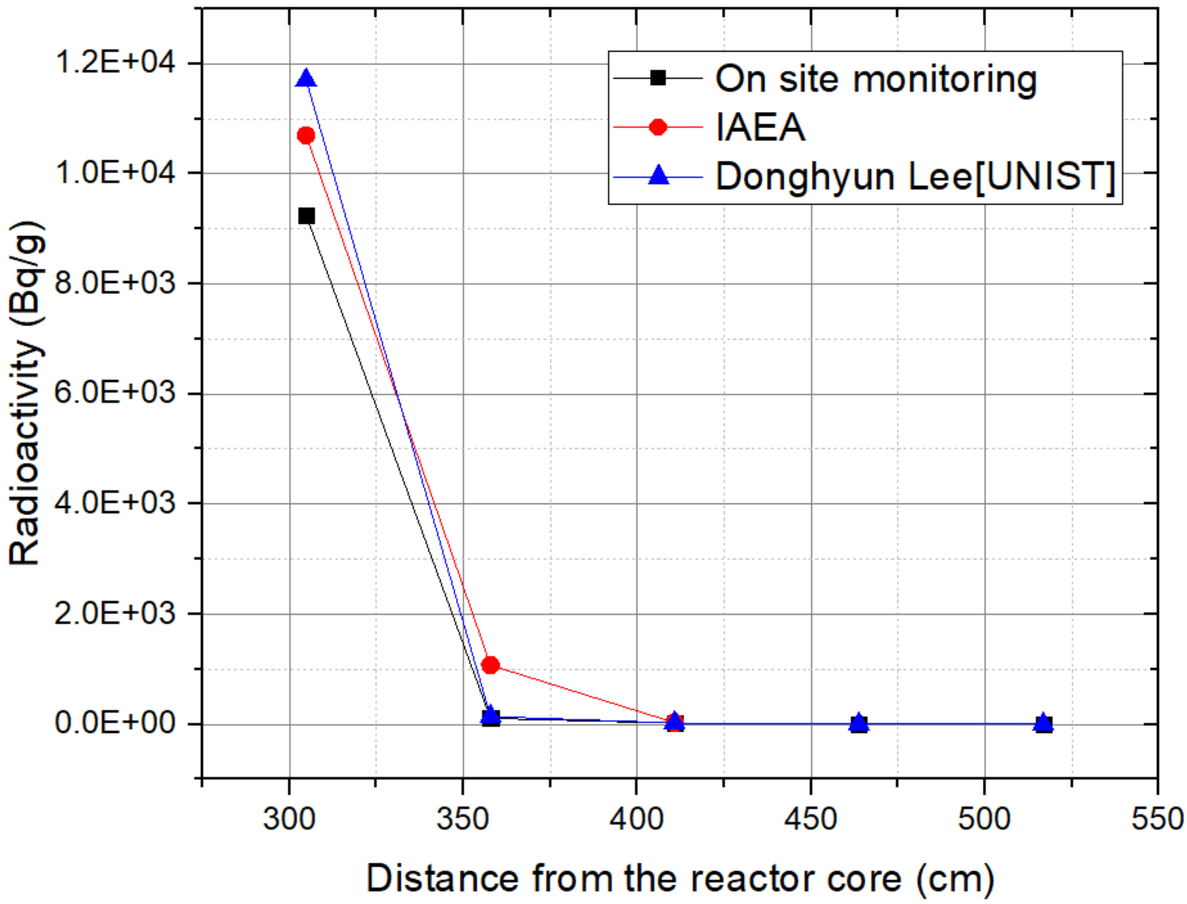


Figure 7. Radioactivity comparison on ^{152}Eu of Trojan nuclear power plant

Table 18. ^{154}Eu radioactivity concentration

	IAEA		Donghyun Lee	Difference ratio	
	ORIGEN-2 (1)	Measured (2)	MCNP 6 (3)	(1) vs (2)	(3) vs (2)
304.75	N/A	9.99E+02	1.26E+03	N/A	+26.0
357.75	N/A	1.04E+01	1.35E+01	N/A	+30.4
410.75	N/A	2.04E+00	2.86E+00	N/A	+40.3
463.75	N/A	3.37E-02	5.39E-02	N/A	+59.8
516.75	N/A	N/A	N/A	N/A	N/A

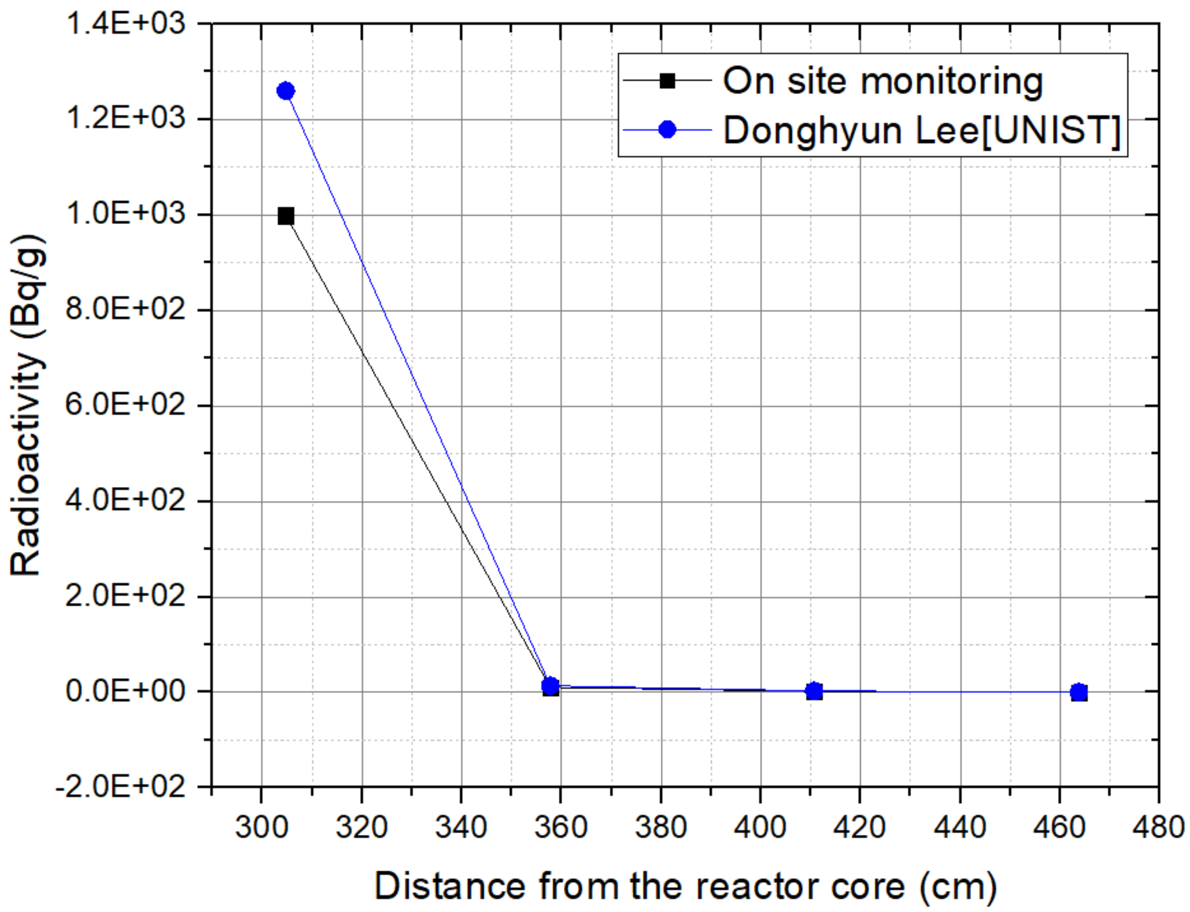


Figure 8. Radioactivity comparison on ^{154}Eu of Trojan nuclear power plant

Table 19. ^{134}Cs radioactivity concentration

	IAEA		Donghyun Lee	Difference ratio	
	ORIGEN-2 (1)	Measured (2)	MCNP 6 (3)	(1) vs (2)	(3) vs (2)
304.75	N/A	3.52E+02	4.26E+02	N/A	+21.0
357.75	N/A	1.85E+00	2.28E+00	N/A	+23.4
410.75	N/A	2.78E-01	3.59E-01	N/A	+29.3
463.75	N/A	7.40E-03	1.03E-02	N/A	+38.7
516.75	N/A	N/A	N/A	N/A	N/A

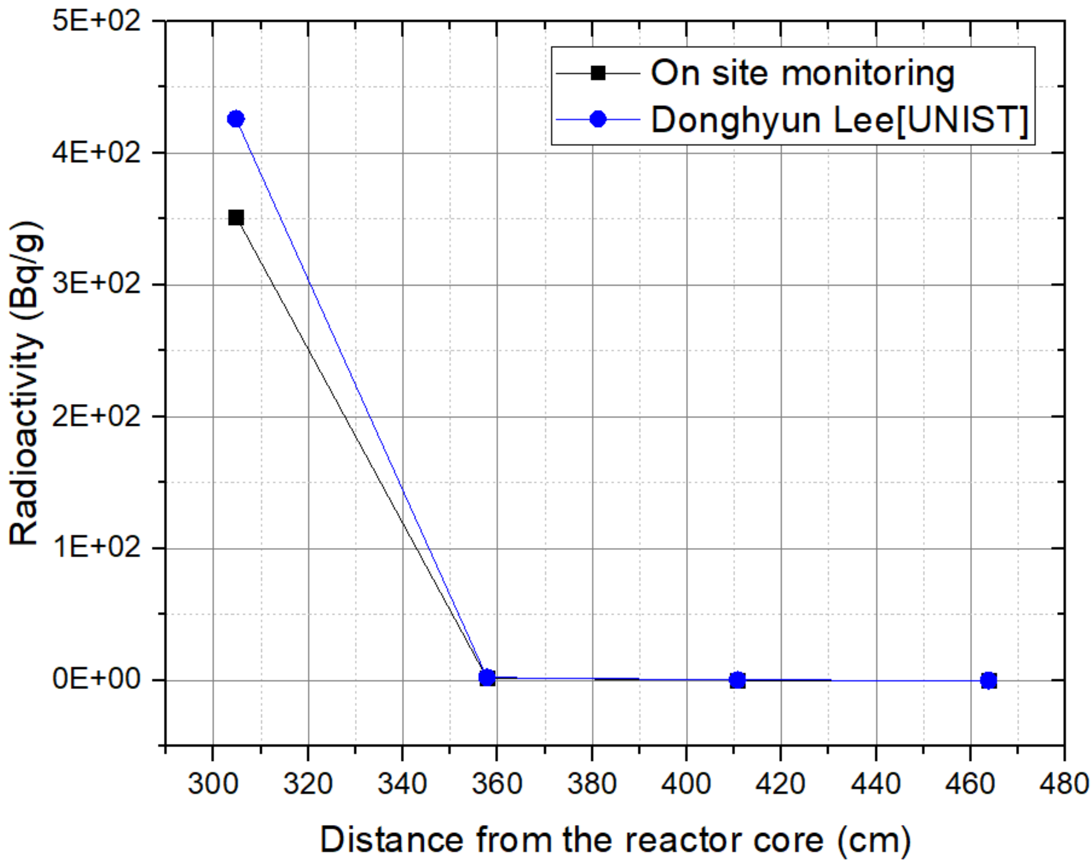


Figure 9. Radioactivity comparison on ^{134}Eu of Trojan nuclear power plant

Table 20. Difference ratio on major nuclides radioactivity

Radioactive nuclide	IAEA	Donghyun Lee [UNIST]
^{60}Co	+89.6 %	+57.3 %
^{152}Eu	+50.0 %	+33.2 %
^{154}Eu	N/A	+40.4 %
^{134}Cs	N/A	+29.2 %

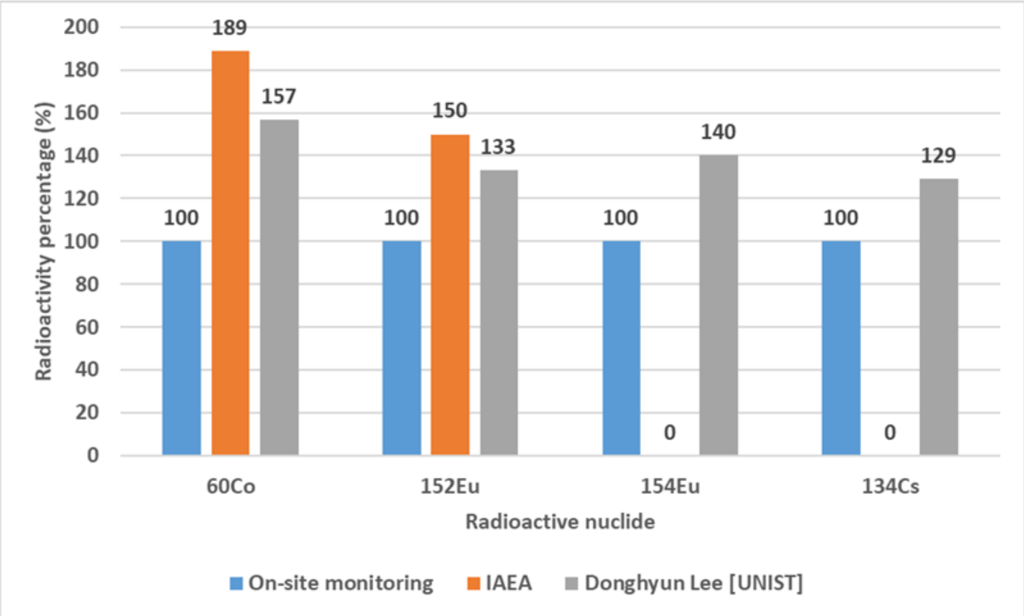


Figure 10. Difference ratio on major nuclides radioactivity

V. Radioactivity Inventory of Bioshield in Kori unit 1

5.1 Kori unit 1 Modeling Conditions

After the validation of the MCNP 6 computation modeling has been verified for its adequate on usage, three-dimensional Kori unit 1 modeling is initiated. As well as for the case on Trojan nuclear power plant, since the exact blueprint of the Kori unit 1 could not be opened to the public, rough design as shown in **Figure 11** and **12**, and parameters of each region on the primary circuit was managed which is clarified on **Table 20** and **21**.

In order to describe the flux of neutron which is generated from the reactor core to bioshield region, whole primary circuit of the nuclear power plant, which are reactor core, barrel, bypass, thermal shield, downcomer, pressure vessel, air, and bioshield was shaped as cylindrically. As shown in **Table 22**, each material properties and nuclide composition has been managed as the input of the MCNP 6.

Table 21. Design factor s of Kori unit 1 and Trojan nuclear power plant

	Reactor Pressure Vessel		Bioshield		EFPY
	Height (m)	Radius (m)	Height (m)	Radius (m)	(yr)
Trojan	13	4	14	3-5	30
Kori-1	13	4	14	3-5	27

Table 22. Specific design factors of Kori unit 1

Cell	Distance from the core
Core	138
Barrel	142
Bypass	146
Thermal shield	155
Downcorner	167
Pressure vessel	184
Air	316
Concrete	530

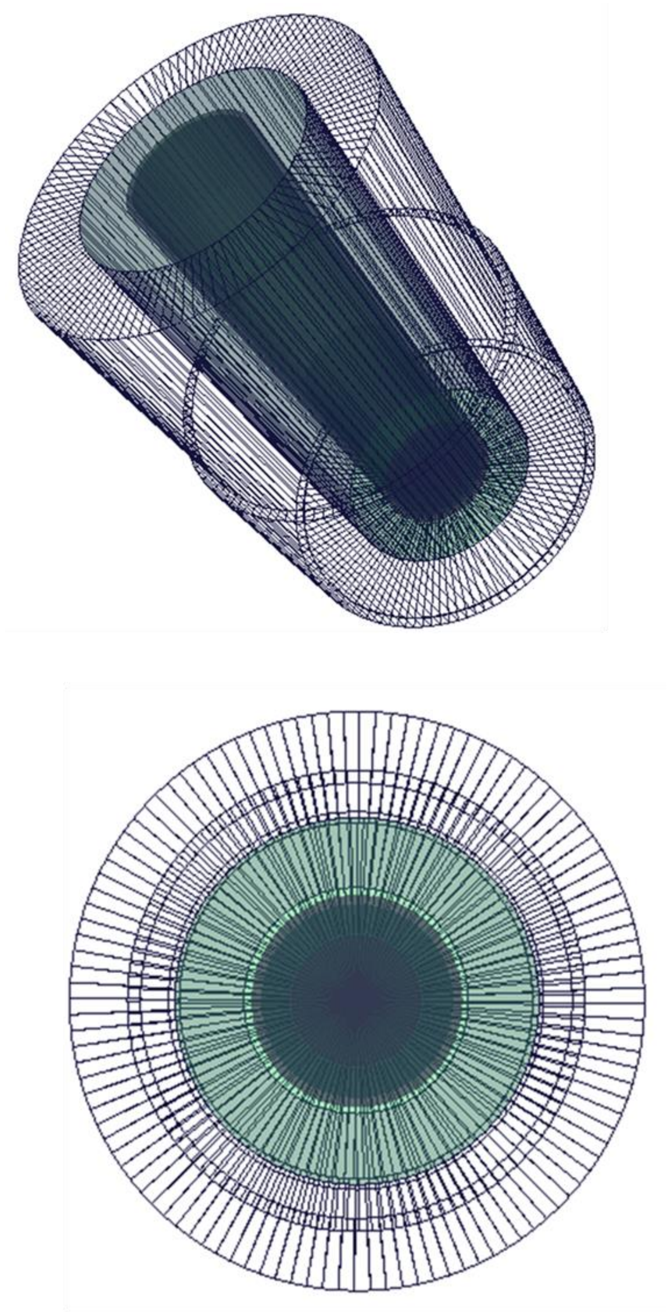


Figure 11. Basic three-dimensional structure of Kori unit 1 primary circuit

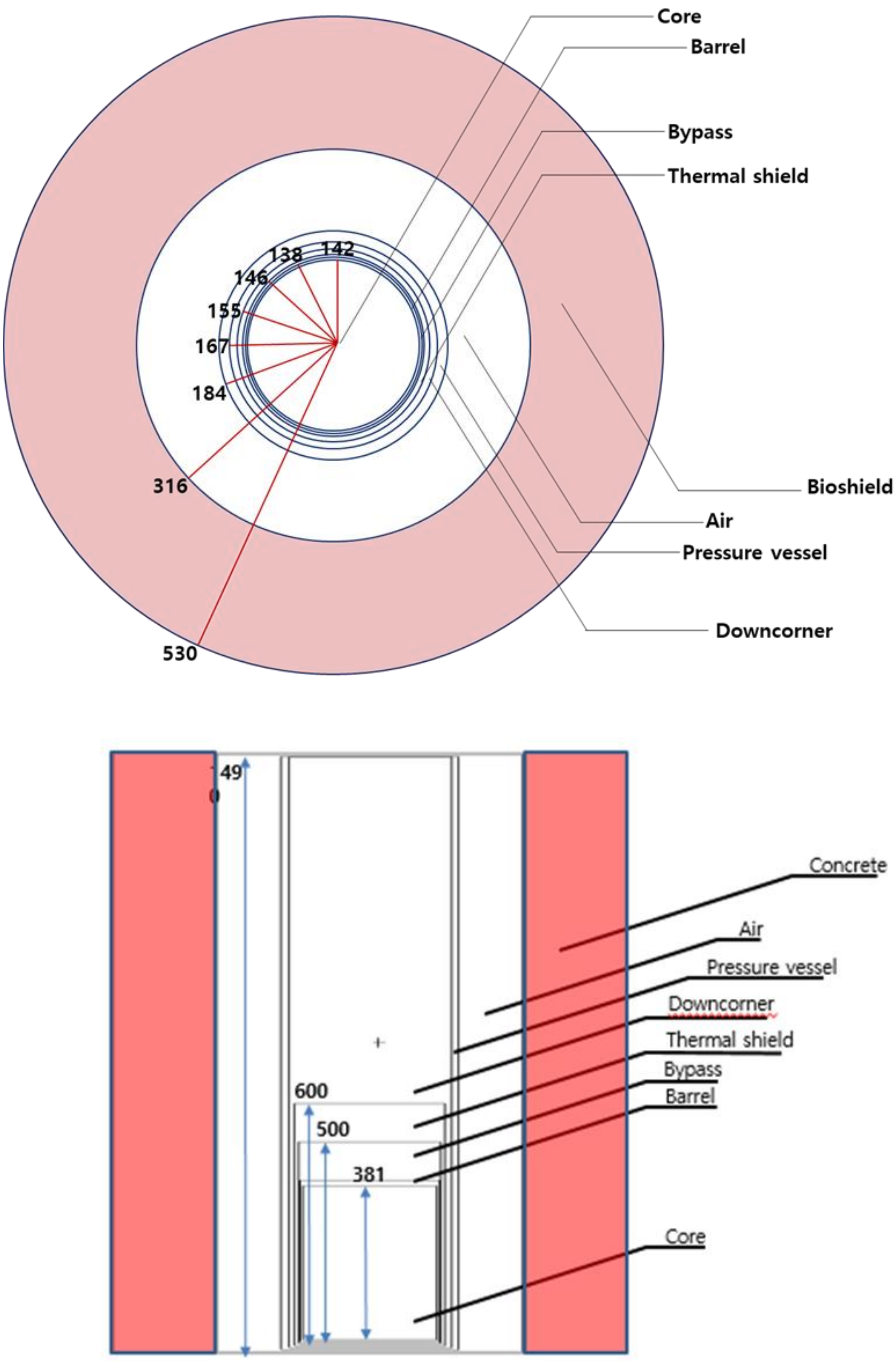


Figure 12. Simplified Kori unit 1 modeling configuration [21]

Table 23. Kori unit 1 structure material and physical properties [#/barn-cm]

Nuclide	Reactor Core	Stainless Steel	Pressure Vessel	Bypass Downcomer	Concrete	Air
^{235}U	1.15E-04					
^{238}U	6.64E-03					
^{239}Pu	3.70E-05					
^{240}Pu	8.86E-06					
^{241}Pu	3.57E-06					
^{133}Cs					1.30E-06	
^{151}Eu					2.25E-07	
^{153}Eu					2.25E-07	
^{1}H	2.76E-02			4.83E-02	7.41E-03	
^{16}O	2.68E-02			2.41E-02	4.21E-02	1.05E-03
^{10}B	2.30E-06			4.31E-06		
^{11}B				1.77E-05		
^{27}Al	1.13E-06				2.28E-03	
^{12}C	3.57E-06	3.17E-04	8.67E-04			7.49E-07
^{28}Si		1.69E-03	4.38E-04		1.52E-02	
^{50}Cr	5.51E-07	7.56E-04	1.27E-05			
^{52}Cr	1.06E-05	1.46E-02	2.44E-04			
^{53}Cr	1.21E-06	1.65E-03	2.77E-05			
^{54}Cr	3.00E-07	4.11E-04	6.89E-06			
^{55}Mn	2.16E-06	1.73E-03	5.43E-06			
^{54}Fe	3.60E-06	3.44E-03	4.86E-03			
^{56}Fe	5.60E-05	5.35E-02	7.55E-02			
^{57}Fe	1.28E-06	1.23E-03	1.73E-03			
^{58}Fe	1.71E-07	1.63E-04	2.31E-04		2.98E-04	
^{58}Ni	9.91E-05	5.10E-03	4.01E-04			
^{60}Ni	3.08E-05	1.97E-03	1.54E-04			
^{61}Ni	1.66E-06	8.55E-05	6.71E-06			
^{62}Ni	5.52E-06	2.72E-04	2.14E-05			
^{64}Ni	1.35E-06	6.94E-05	5.45E-06			
^{96}Mo			2.81E-04			
^{91}Zr	4.52E-03					
^{23}Na					1.00E-03	
^{24}Mg					1.42E-04	
^{32}S					5.38E-05	
^{39}K					6.61E-04	
^{40}Ca					2.78E-03	
Total	6.60E-02	8.70E-02	8.48E-02	7.24E-02	7.20E-02	1.05E-03

5.2 Radioactivity Assessment of bioshield in Kori unit 1

After the three-dimensional modeling on the primary circuit of the Kori unit 1 nuclear power plant, the neutron flux distribution has been analyzed in order to derive the radioactivity if ^{60}Co, ^{152}Eu, ^{154}Eu, and ^{134}Cs. The value showed minimum 1.37E-42 #/cm^{2}s on the farthest point from the reactor core and maximum 1.65E+07 #/cm^{2}s as shown in **Figure 13**. The tendency of the distribution showed exponential decreasing in the horizontal point of view and showed increasing tendency at the point of 300 cm from the bottom later decreasing since the reactor core is positioned at 300 cm from the bottom of the floor.

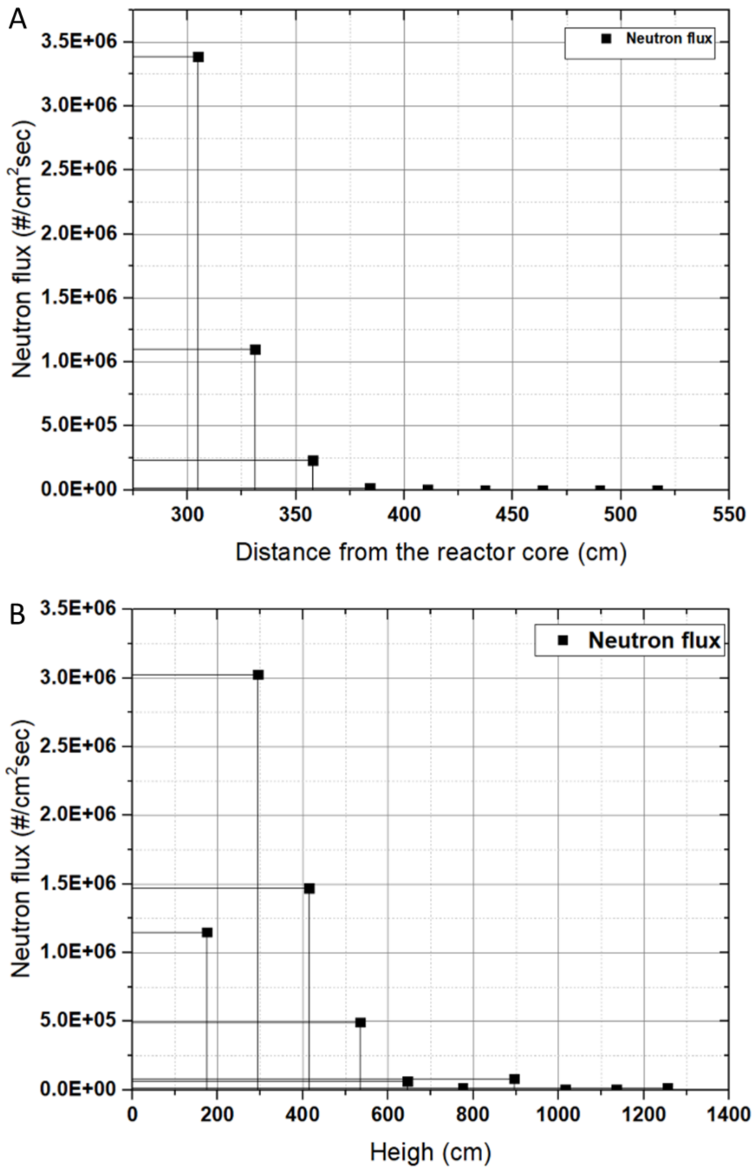
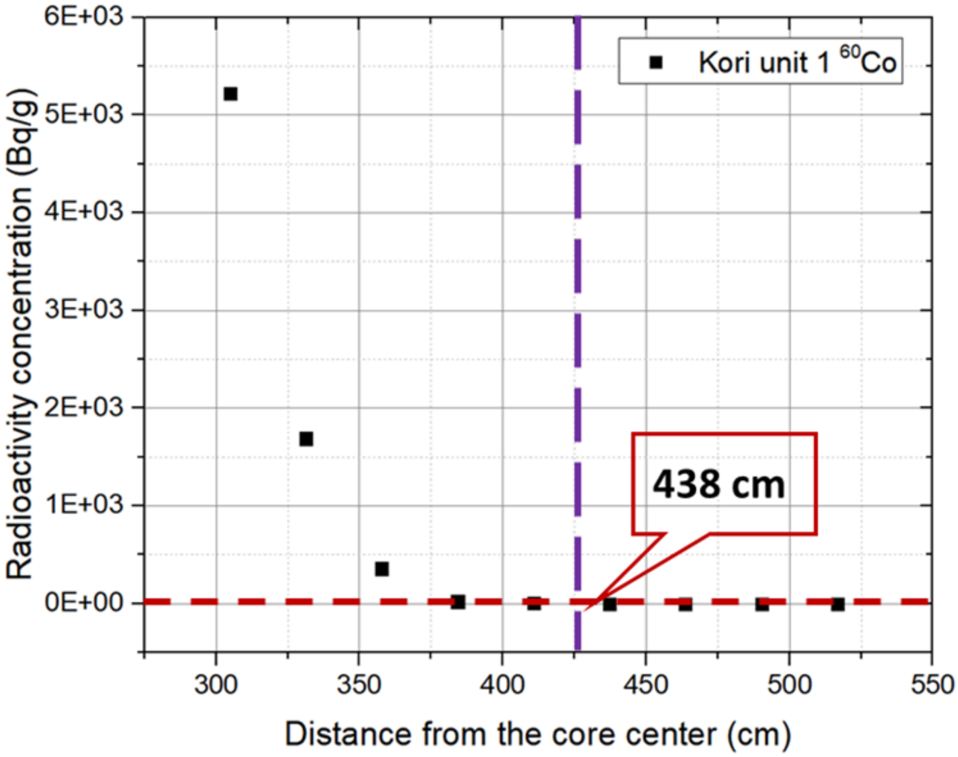


Figure 13. A: Neutron flux distribution on Kori unit 1 bioshield (horizontal viewpoint), B: Neutron flux distribution on Kori unit 1 bioshield (vertical viewpoint)

^{60}Co, ^{152}Eu, ^{154}Eu, and ^{134}Cs was chosen as target radioactive nuclides under considering initial concentration and external dose conversion factor, which effects on radiation workers. As shown in **Figure 14 and Table 23,** the radioactivity of Kori-1 ^{60}Co showed maximum 5.22E+03 Bq/g on 304.75 cm to minimum 3.43E-07 Bq/g on 516.75 cm. The tendency of the radioactivity showed correspondingly reducing along the radial distance same as neutron flux distribution. The clearance boundary assessed as 438 cm with the clearance level of 0.1 Bq/g.



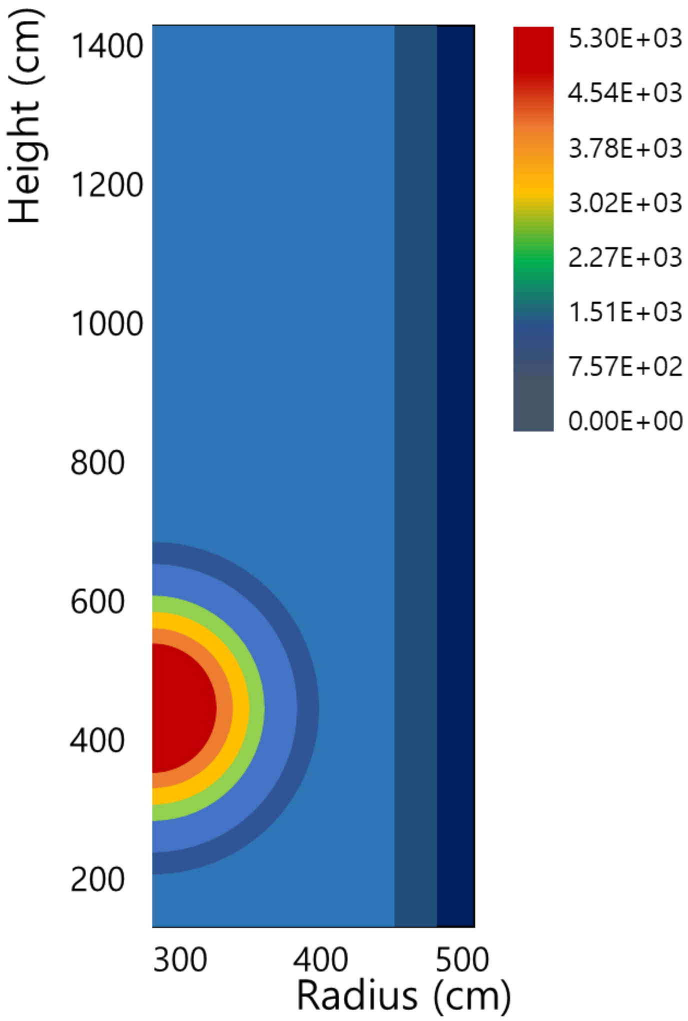


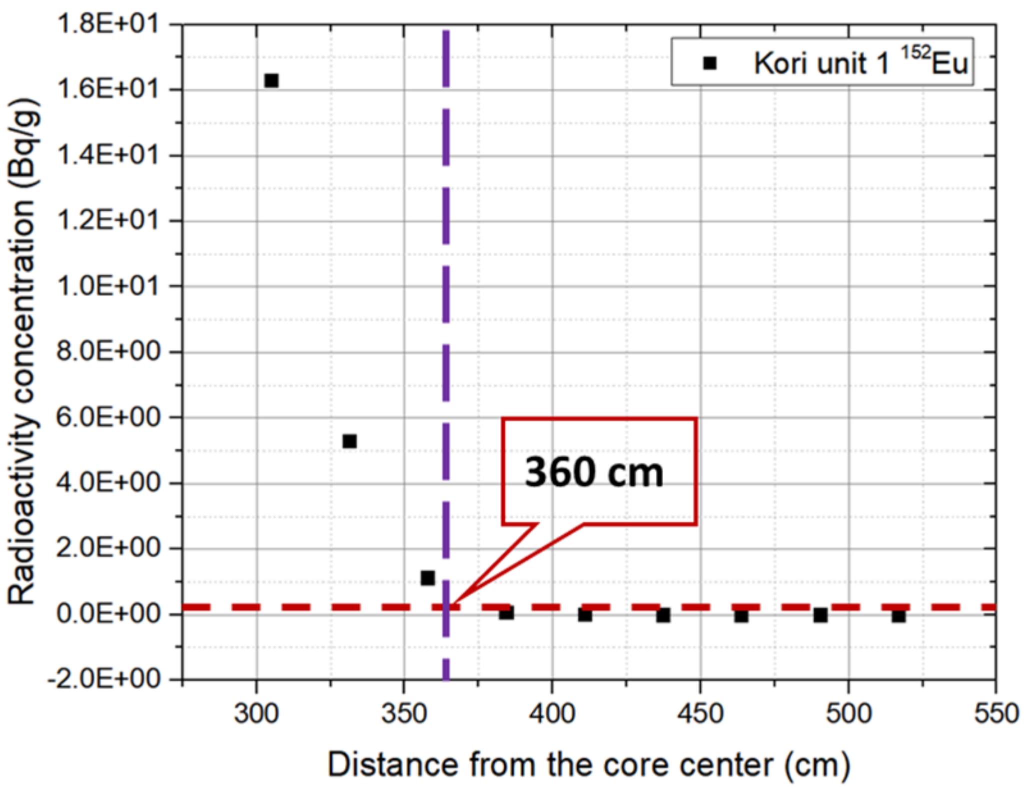
Figure 14. A: ^{60}Co radioactivity concentration

B: 3D ^{60}Co radioactivity

Table 24. ^{60}Co radioactivity concentration

Distance (cm)	Kori unit 1 (Bq/g)
304.75	5.22E+03
331.25	1.69E+03
357.75	3.60E+02
384.25	2.07E+01
410.75	2.08E+00
437.25	1.22E-01
463.75	5.86E-07
490.25	5.70E-07
516.75	3.43E-07

As identified in **Figure 15 and Table 24.**, ^{152}Eu showed highest 1.63E+01 Bq/g to lowest 1.07E-09 Bq/g. The area from 357.75 cm to 463.75 cm showed the same increasing tendency of difference ration which was confirmed on ^{60}Co as well. The clearance boundary assessed as 360 cm with the clearance level of 0.1 Bq/g.



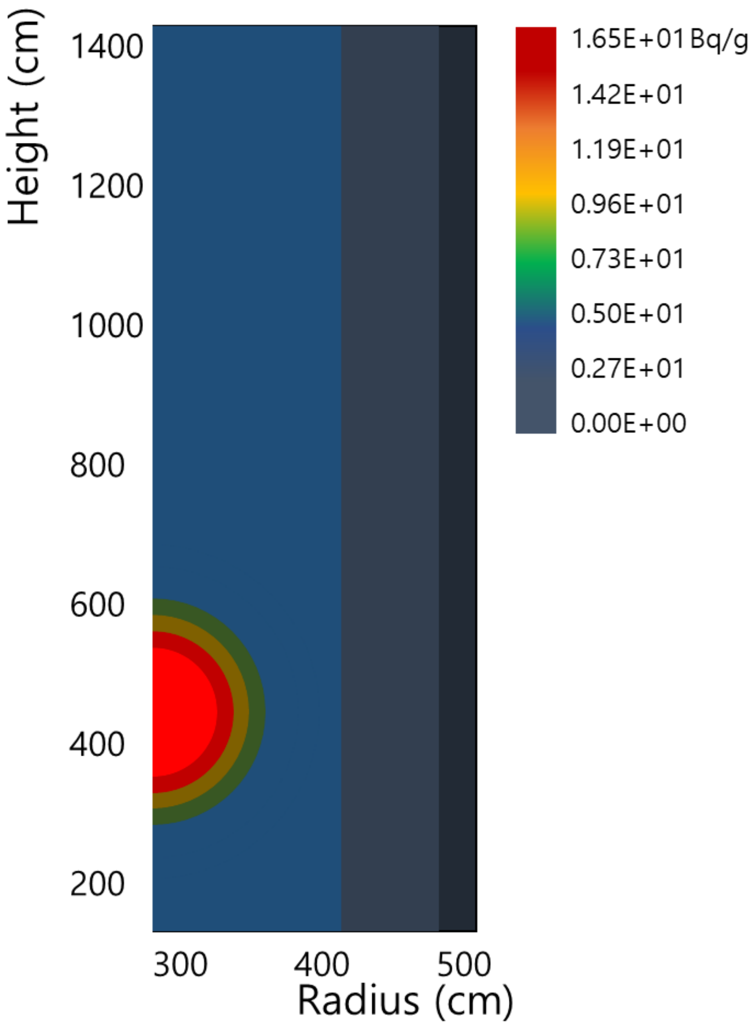


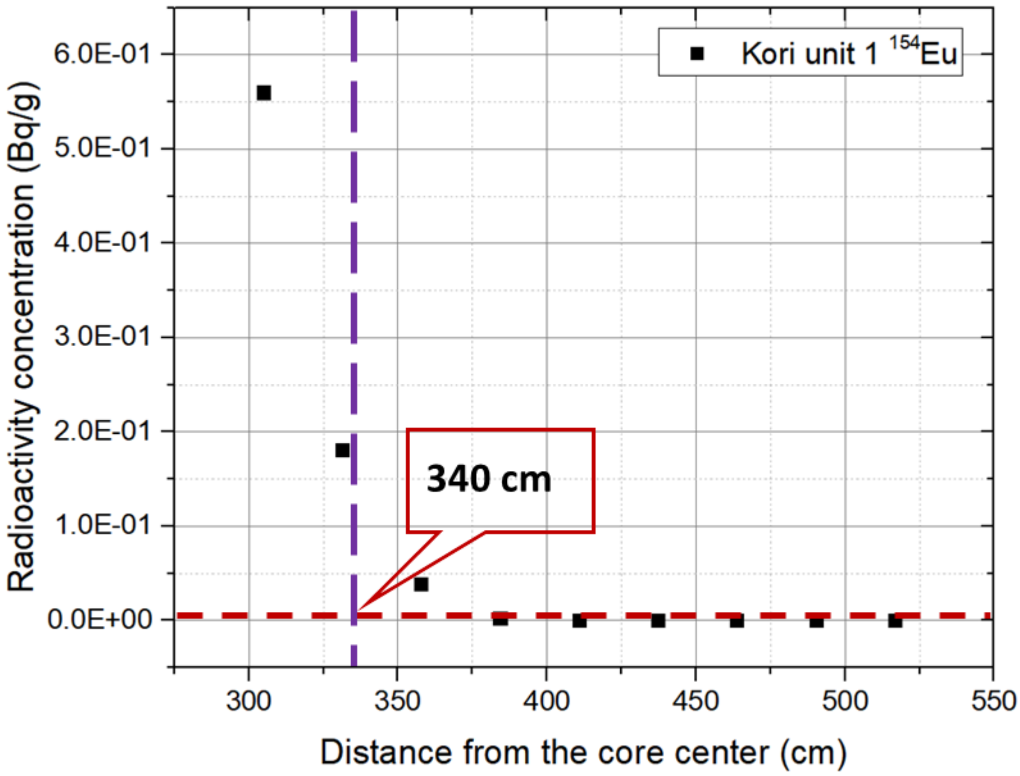
Figure 15. A: ^{152}Eu radioactivity concentration

B: 3D ^{152}Eu radioactivity

Table 25. ^{152}Eu radioactivity concentration

Distance (cm)	Kori unit 1 (Bq/g)
304.75	1.63.E+01
331.25	5.29.E+00
357.75	1.13.E+00
384.25	6.47.E-02
410.75	6.54.E-03
437.25	3.82.E-04
463.75	1.84.E-09
490.25	1.79.E-09
516.75	1.07.E-09

As identified in **Figure 16 and Table 25**, ^{154}Eu showed largest 5.60E-01 Bq/g to smallest 3.69E-11 Bq/g. As already identified on ^{60}Co and ^{152}Eu, ^{154}Eu also showed incline of difference ratio from region 357.75 cm to 463.75 cm area. The clearance boundary assessed as 340 cm with the clearance level of 0.1 Bq/g.



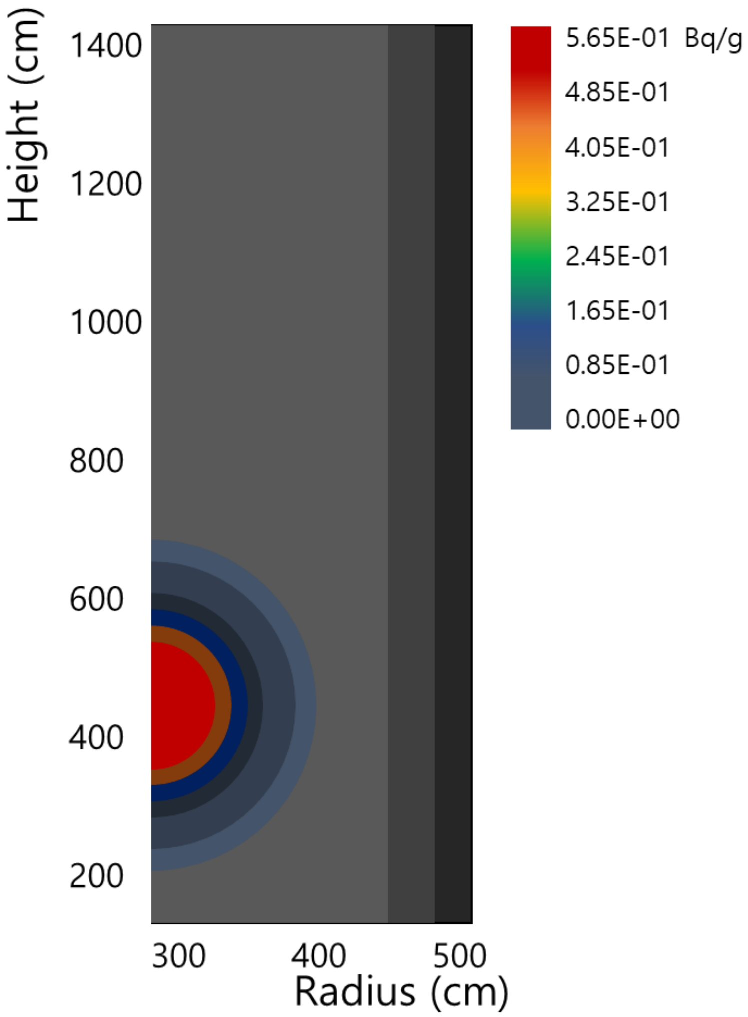
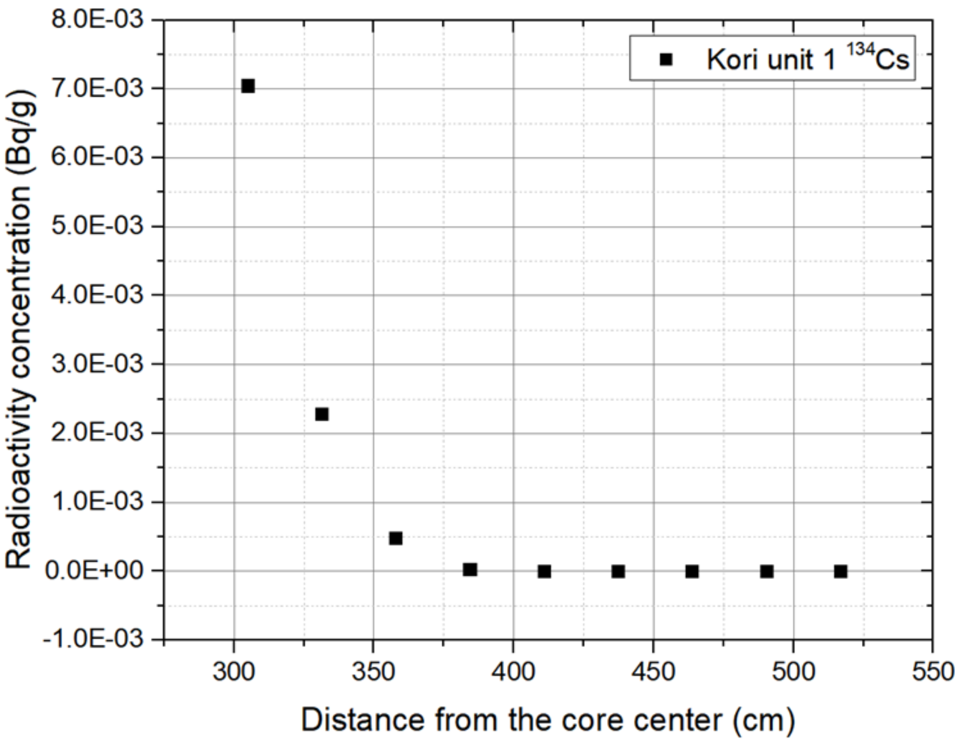


Figure 16. A: ^{154}Eu radioactivity concentration
B: 3D ^{154}Eu radioactivity

Table 26. ^{154}Eu radioactivity concentration

Distance (cm)	Kori unit 1 (Bq/g)
304.75	5.60.E-01
331.25	1.81.E-01
357.75	3.86.E-02
384.25	2.22.E-03
410.75	1.04.E-04
437.25	1.31.E-05
463.75	6.29.E-11
490.25	6.12.E-11
516.75	3.69.E-11

As identified in **Figure 17 and Table 26**, ^{134}Cs showed maximum 7.05E-03 Bq/g to minimum 4.64E-13 Bq/g Different from previous result as ^{60}Co, ^{152}Eu and ^{154}Eu, the comparison on ^{134}Cs between Kori unit 1 and Trojan should be carefully observed since the detecting of ^{134}Cs on Trojan was limited farthest to 463.75 cm as radial direction. Since the maximum radioactivity was lower than 0.1 Bq/g, ^{134}Cs alone showed entire clearance on bioshield.



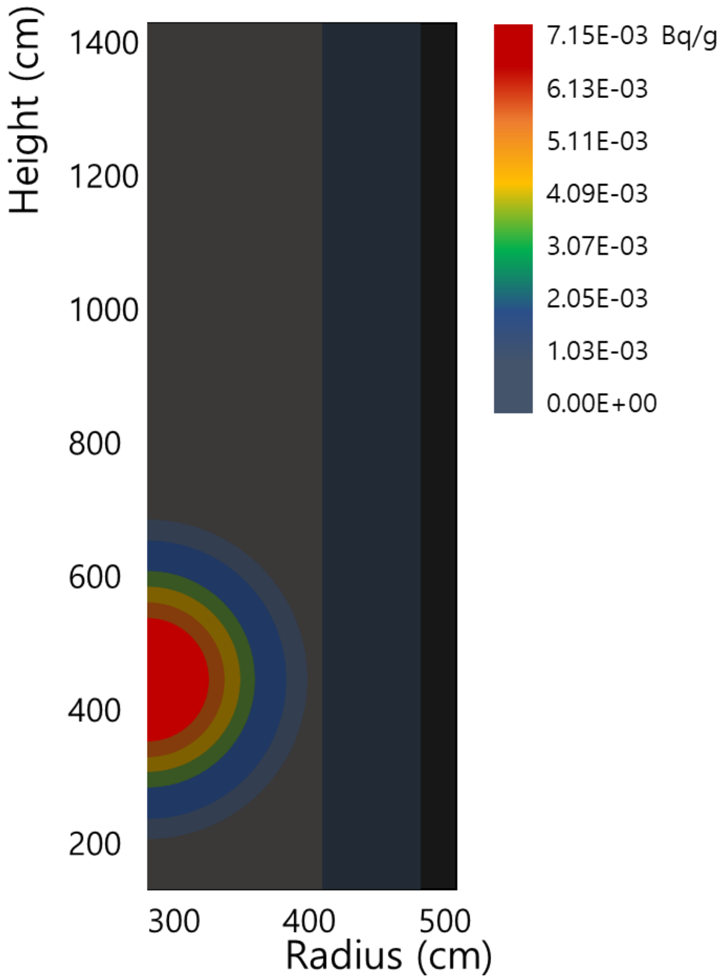


Figure 17. A: ^{134}Cs radioactivity concentration
B: 3D ^{134}Cs radioactivity

Table 27. ^{134}Cs radioactivity concentration

Distance (cm)	Kori unit 1 (Bq/g)
304.75	7.05.E-03
331.25	2.28.E-03
357.75	4.87.E-04
384.25	2.79.E-05
410.75	2.82.E-06
437.25	1.65.E-07
463.75	7.92.E-13
490.25	7.71.E-13
516.75	4.64.E-13

As shown in **Figure 18** and **Table 27**, total, ^{60}Co contributed more than 99% dominance on whole bioshield region followed by ^{152}Eu, ^{154}Eu, and ^{134}Cs. The maximum level was 5.22E+03 Bq/g on 304.75 cm from the reactor core and the minimum level was 3.43E-07 Bq/g on 516.75 cm from the reactor core. The clearance boundary assessed as 425 cm with the clearance level regulation with multiple nuclides.

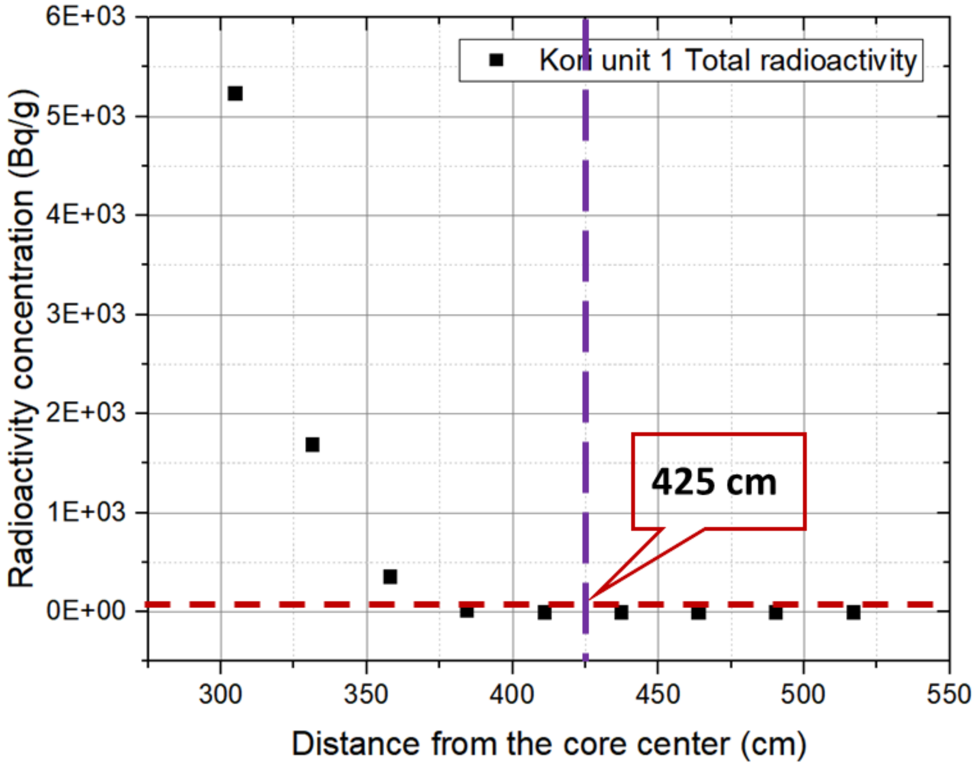


Figure 18. Total radioactivity concentration

Table 28. Total radioactivity concentration with impact factors

Distance (cm)	^{60}Co (Bq/g)	Impact (%)	^{152}Eu	Impact	^{154}Eu	Impact	^{134}Cs	Impact
304.75	5.22.E+03	9.97.E+01	1.63.E+01	3.11.E-01	5.60.E-01	1.07.E-02	7.05.E-03	1.35.E-04
331.25	1.69.E+03	9.97.E+01	5.29.E+00	3.12.E-01	1.81.E-01	1.07.E-02	2.28.E-03	1.34.E-04
357.75	3.60.E+02	9.97.E+01	1.13.E+00	3.13.E-01	3.86.E-02	1.07.E-02	4.87.E-04	1.35.E-04
384.25	2.07.E+01	9.97.E+01	6.47.E-02	3.12.E-01	2.22.E-03	1.07.E-02	2.79.E-05	1.34.E-04
410.75	2.08.E+00	9.97.E+01	6.54.E-03	3.13.E-01	1.04.E-04	4.98.E-03	2.82.E-06	1.35.E-04
437.25	1.22.E-01	9.97.E+01	3.82.E-04	3.12.E-01	1.30.E-05	1.06.E-02	1.65.E-07	1.35.E-04
463.75	5.86.E-07	9.97.E+01	1.84.E-09	3.13.E-01	6.29.E-11	1.07.E-02	7.92.E-13	1.35.E-04
490.25	5.70.E-07	9.97.E+01	1.79.E-09	3.13.E-01	6.12.E-11	1.07.E-02	7.71.E-13	1.35.E-04
516.75	3.43.E-07	9.97.E+01	1.07.E-09	3.11.E-01	3.69.E-11	1.07.E-02	4.64.E-13	1.35.E-04

VI. Safety, Economics and Waste Management for Bioshield

6.1 External Dose for Workers on Bioshield in Kori unit 1

From the activation level data, it is possible to generate safety guideline for the filed workers on the structure and give insight on decontamination method selection. The current limitation on the Republic of Korea regulates field worker's permission as 20 mSv/year [18]. Although radioactive inventory analysis was initiated based mainly focused on the horizontal distance from the reactor core, it is more adequate to focus on the height of the bioshield since the decontamination process would be done at the side of the structure with the vertical process. From **Table 28**, each radiation of nuclides was first assessed, and the result was converted into the vertical graphic of external dose which is shown in **Figure 19** and **Figure 20**. The maximum dose rate was shown on 165 cm with 5.55E+00 mSv/h and the lowest rate was on 1485 cm with 7.25E-01 mSv/h.

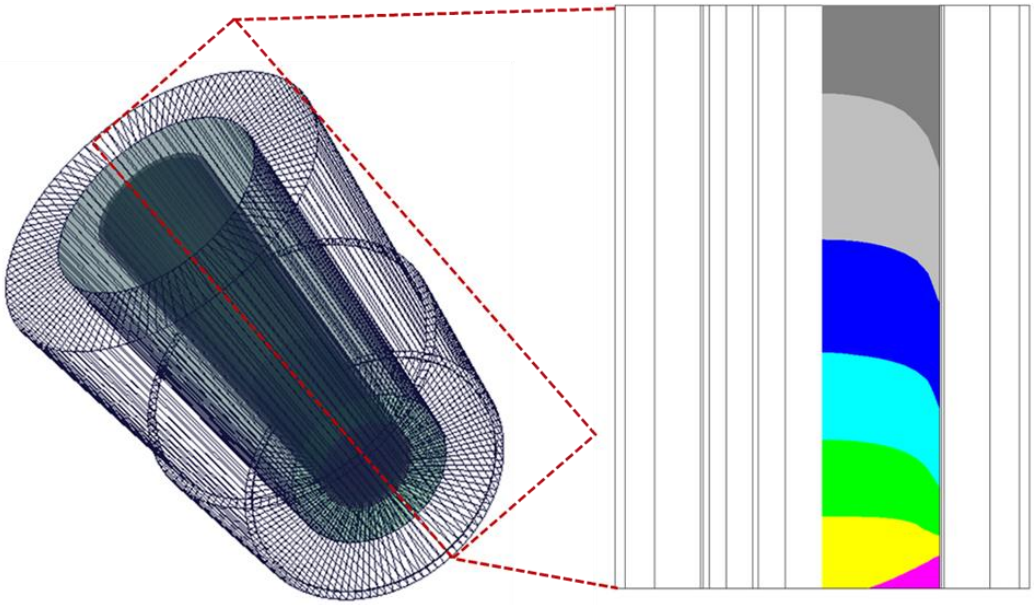


Figure 19. External dose rate on Kori unit 1 bioshield

Table 29. External dose rate on major nuclides [mSv/h]

Height (cm)	^{60}Co	^{152}Eu	^{154}Eu	^{134}Cs	Total
1485	6.89.E-01	2.90.E-02	4.35.E-02	7.25.E-03	7.25.E-01
1320	7.74.E-01	3.26.E-02	4.89.E-02	8.15.E-03	8.15.E-01
1155	8.69.E-01	3.66.E-02	5.49.E-02	9.15.E-03	9.15.E-01
990	1.19.E+00	5.00.E-02	7.50.E-02	1.25.E-02	1.25.E+00
825	1.52.E+00	6.40.E-02	9.60.E-02	1.60.E-02	1.60.E+00
660	2.00.E+00	8.40.E-02	1.26.E-01	2.10.E-02	2.10.E+00
495	2.61.E+00	1.10.E-01	1.65.E-01	2.75.E-02	2.75.E+00
330	3.52.E+00	1.48.E-01	2.22.E-01	3.70.E-02	3.70.E+00
165	5.27.E+00	2.22.E-01	3.33.E-01	5.55.E-02	5.55.E+00

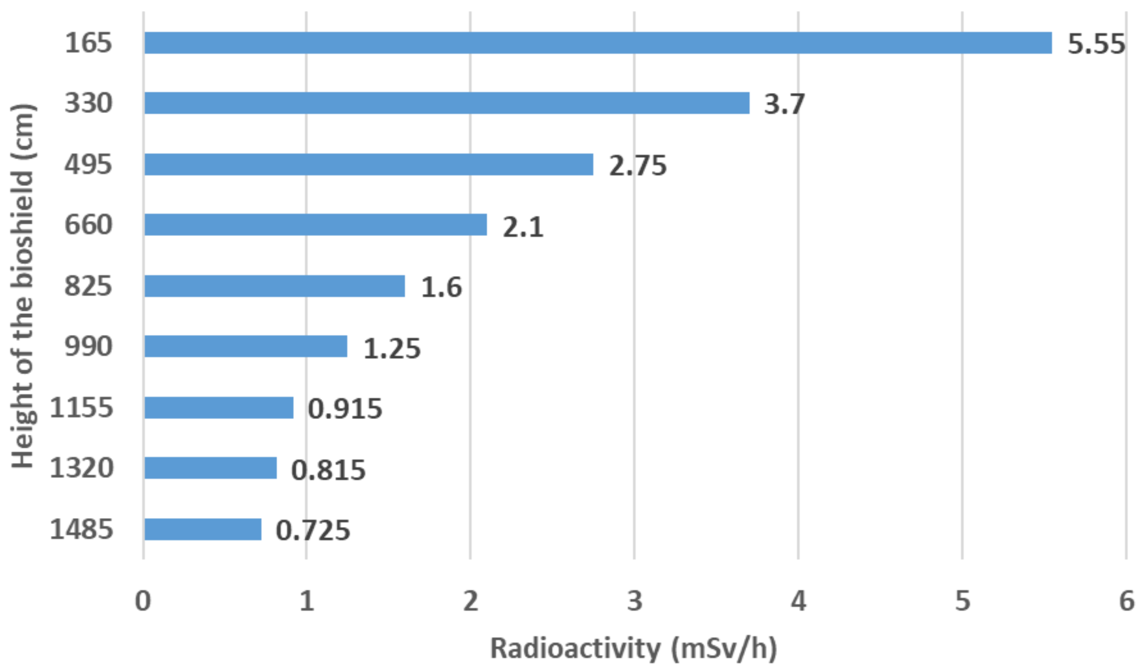


Figure 20. External dose for workers with major nuclides [hr]

Since current regulation on professional radiological workers on the field permits 20 mSv/year, the possible working hours in a year should be limited as shown in **Table 29**. As shown in **Figure 21**, the maximum working hour per personal is 2.76E+01 hr on 1485 cm region from the floor and the minimum hour is 3.60 hr on 165 cm region from the floor. Since most of the high duration process and the high-risk dose occur at the bottom of the structure, sensitivity test with added regional space analysis should be made from 330 cm height to 165 cm region for further concrete reliability.

Table 30. Permittable working hour on bioshield

Height (cm)	Working hour limit (hr)
1485	2.76.E+01
1320	2.45.E+01
1155	2.19.E+01
990	1.60.E+01
825	1.25.E+01
660	9.52.E+00
495	7.27.E+00
330	5.41.E+00
165	3.60.E+00

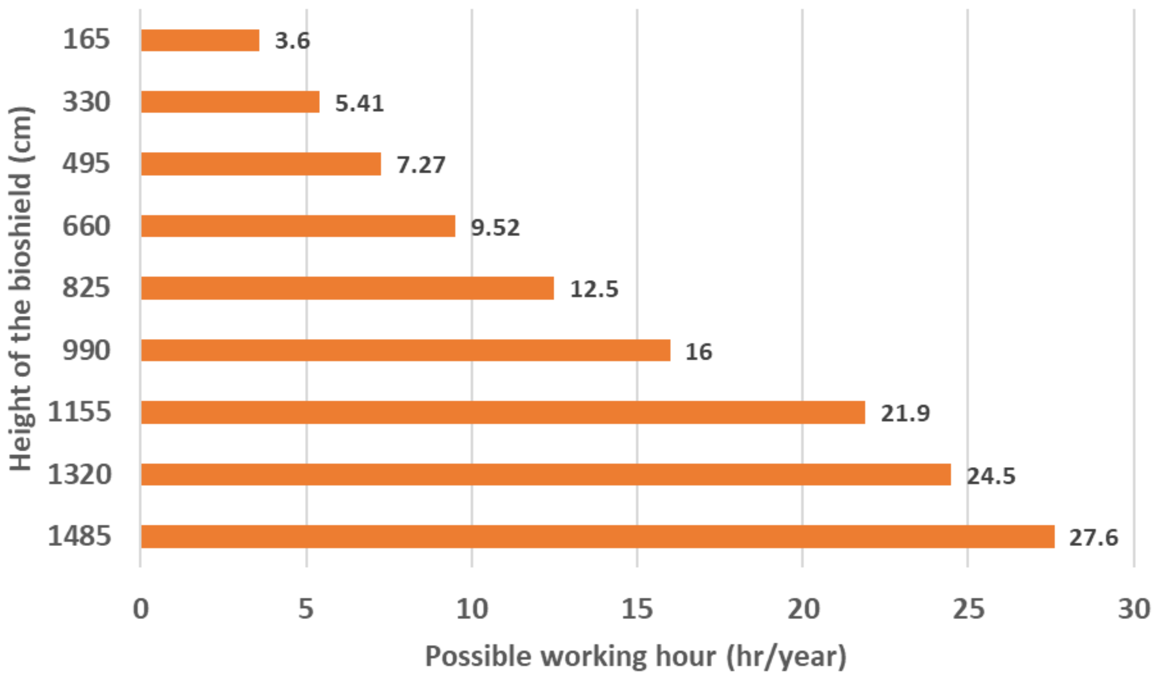


Figure 21. Bioshield region possible working hour in a year

From the permittable working hour limitation data, it is possible to clarify which method could be used on decontamination of the bioshield. The decontamination method is separated into two branches. First, chemical decontamination holds high purification capability of up to 99 %. However, it has difficulty in using on-site for real time. In addition, a large amount of secondary waste is generated from the solution that was used on the process.

Second, physical decontamination could be divided into two major schemes. Scabbling uses tungsten carbide head with 100 hours of a lifetime on it. Its efficiency is 0.25 ~ 0.5 m^2/h with 3 mm depth decontamination. Shaving uses the diamond head with 40 hours lifetime on it. Its efficiency is 15 ~ 25 m^2/h with 3 mm depth penetration [24, 25]. For comparing the usage of two heads, amendments on efficiency is required. In order to match both heads to 3 mm depth decontamination in 1 hour, scabbling requires an average 50 people and shaving requires average 1 person on the bioshield of Kori unit 1. Since the clearance boundary is 425 cm from the reactor core, it requires 363 hours of working time. As shown in **Figure 22**, scabbling needs steady supply total of 3522 people and shaving requires a total of 70 people on a year.

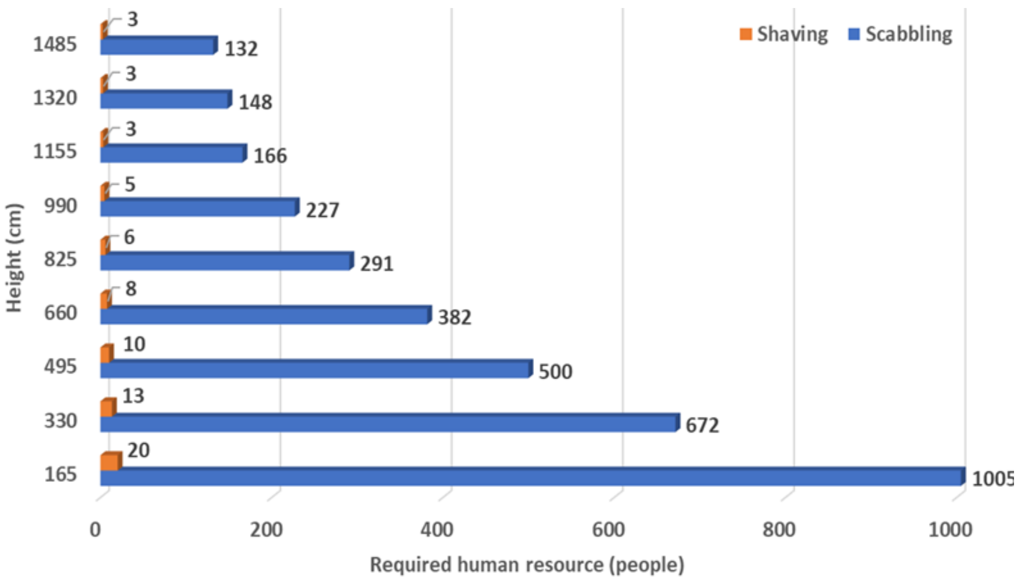


Figure 22. Human resource requirement on decontamination of the bioshield

6.2 Spatial Distribution for Radioactivity Waste

The D&D project on Kori unit 1 would be initiated with the immediate release of license. According to the brief timeline, the decontamination and dismantling of the bioshield would be done within 5 to 10 years after the shutdown of the facility. In order to identify classification, the change on the amount of the radioactive waste, as shown in **Figure 23**, the clearance boundary and waste amount is analyzed from 5 years after the shutdown to 30 years with a gap of 5 years. Since conservative analysis is required for estimation of the potential waste amount study, 200 cm through 400 cm height region which hold the highest radioactivity concentration was chosen as the criteria for assessment. As shown in **Figure 24**, it showed 2428 of 200 L drums at 5 years, and 1902 of 200 L drums at 30 years. No significance reduction after 10 years.

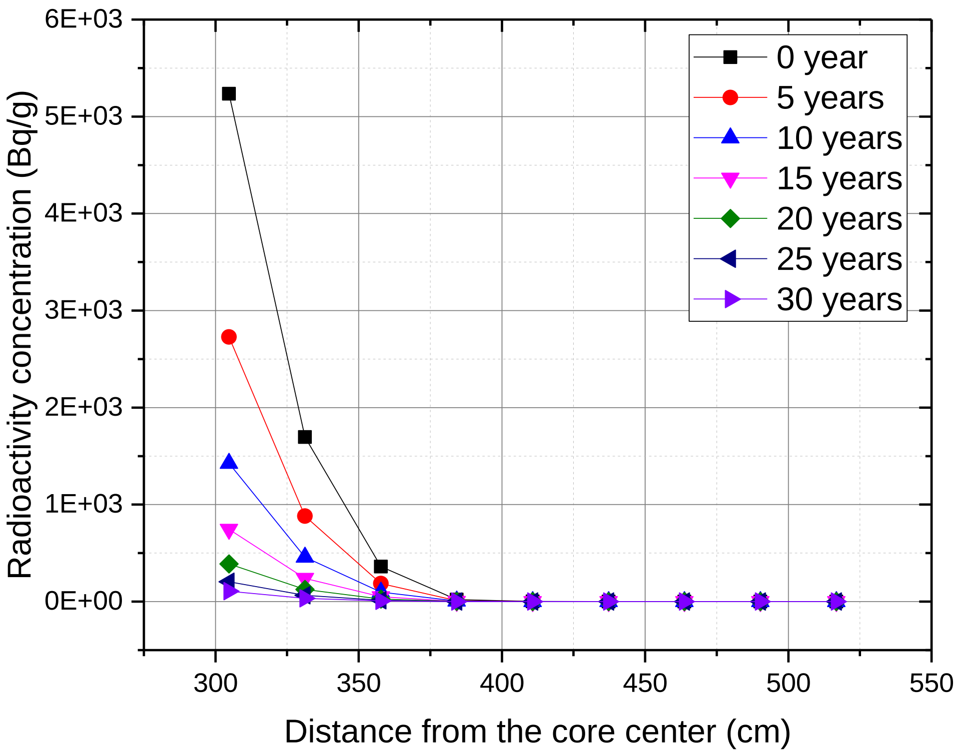


Figure 23. Total radioactivity on the timeline after the shutdown of the facility

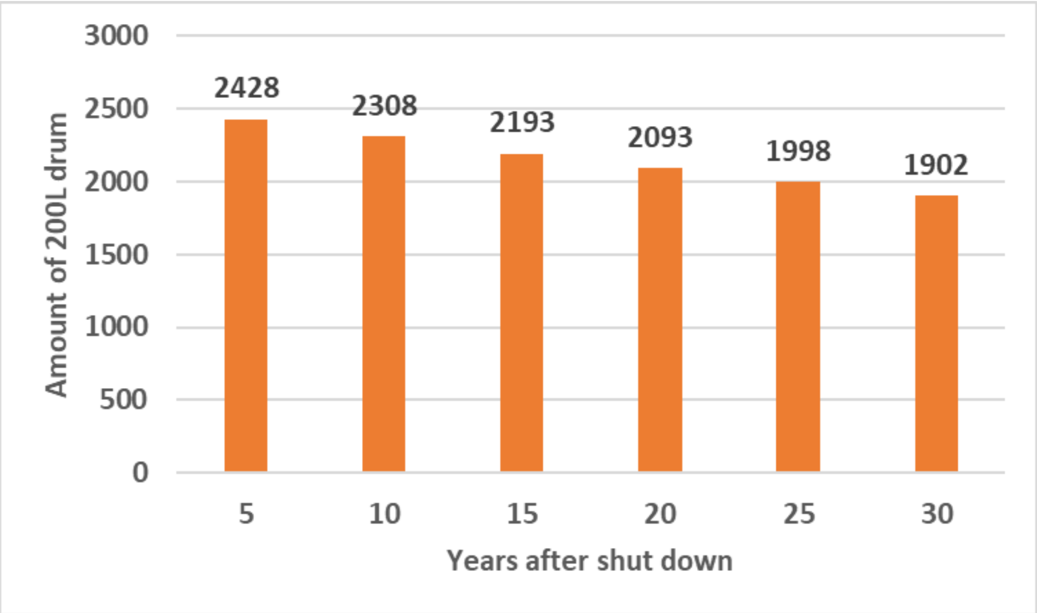


Figure 24. Generation of radioactive waste on timeline after the shutdown of the facility

6.3 Cost Reduction by Waste Volume Reduction

As the amount of the LLW (Low-Level Waste) has been analyzed, it is possible to estimate the budget on managing radioactive waste. Current regulation on the Republic of Korea regards 1 drum of 200 L LLW requires 12,080 USD. Under consideration as entire Bioshield as LLW without radioactive inventory investigation, it would cost 53 M USD. However as shown in **Table 30** and **Figure 25**, radioactive inventory assessment has been done it would cost 29 M USD at 5 years after the shutdown and 23 USD at the year of 30 after the shutdown. It reaches 42 % decreasing at 5 years and 56 % decrease at 30 years. Since the field decontamination and dismantling process would be initiated after 2024, which is 5 years after the shutdown of Kori unit 1 it requires comparison with other case studies. As shown in **Table 31**, Trojan, Haddam Neck, and Maine Yankee nuclear power plants showed each with 30 %, 25 %, and 25 % increasing between estimation and real cost [26], due to limitation of the analysis variables including 3D spatial distribution of radioactivity. Since the research has overcome the existing limitation on previous foreign nuclear power plants, the difference rate between estimated waste management budget and real budget would be optimized to the minimum.

Table 31. Kori unit 1 bioshield waste managing budget

Year	Waste managing budget (M. USD)
5	29
10	28
15	27
20	25
25	24
30	23

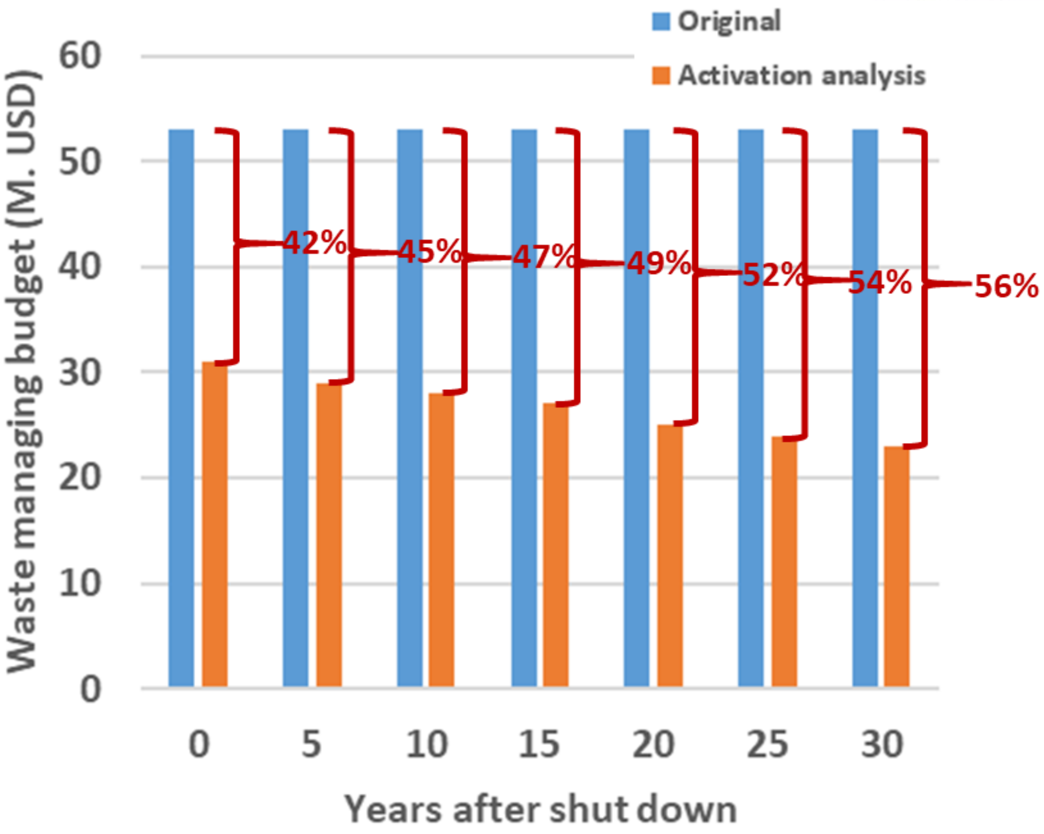


Figure 25. Kori unit 1 bioshield waste managing budget

Table 32. Waste management budget status

Plant	Reactor type	Electrical capacity (MW)	Estimated waste management cost (M USD)	Real waste managing budget (M USD)
Kori unit 1	PWR	576	58	N/A
Trojan	PWR	1095	37	52
Haddam Neck	PWR	603	75.8	112
Maine Yankee	PWR	900	82.5	110

VII. Conclusion

The research is aimed to assess the radioactivity of bioshield in Kori unit 1. The bioshield was contaminated by radioactive nuclides at an average of 812 Bq/g and ^{60}Co makes up the largest proportion of nuclides in the entire structure.

From the assessment of radioactive inventory, the clearance boundary was identified in order to clarify the amount of the LLW. It showed the bioshield structure can be considered as non-radioactive waste at the point 425 cm from the reactor core and a total of 2437 drums of LLW would be generated from the D&D. From the estimation of the waste generation, it is expected to be minimum 29 M USD would be required on managing LLW.

The radioactivity of major nuclides was converted into potential external dose exposure rate for assessing worker guideline and give insight on choosing of decontamination scheme. Since the Bioshield has been activated due to high neutron absorption, working hours should be limited. Average possible working hours according to dose levels should be 14 hours, with a maximum of 27.6 hours at the lowest dosage and a minimum of 3.6 hours at the highest.

This study has introduced a scheme to assess the Kori-1 Bioshield radioactive inventory for classification of its activation and amount of waste generated from the decommissioning process before the initiation of the real field dismantling and decommissioning process starts. The model provides results within a reasonable amount of error and can be utilized as a basic tool to assess other domestic PWR nuclear power plants.

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