https://doi.org/10.7733/jnfcwt.2023.013

Occupational Dose Analysis of Spent Resin Handling Accident During NPP Decommissioning

Hyunjin Lee, Chang-Lak Kim*, Sang-Rae Moon, and Sun-Kee Lee
KEPCO International Nuclear Graduate School, 658-91, Haemaji-ro, Seosaeng-myeon, Ulju-gun, Ulsan 45014, Republic of Korea

(Received October 5, 2022 / Revised November 4, 2022 / Approved December 5, 2022)

According to NSSC Notice No. 2021-10, safety analysis needs to be introduced in the decommissioning plan. Public and occupational dose analyses should be conducted, specifically for unexpected radiological accidents. Herein, based on the risk matrix and analytic hierarchy process, the method of selecting accident scenarios during the decommissioning of nuclear power plants has been proposed. During decommissioning, the generated spent resin exhibits relatively higher activity than other generated wastes. When accidents occur, the release fraction varies depending on the conditioning method of radioactive waste and type of radioactive nuclides or accidents. Occupational dose analyses for 2 (fire and drop) among 11 accident scenarios have been performed. The radiation doses of the additional exposures caused by the fire and drop accidents are 1.67 and 4.77 mSv, respectively.

Keywords: Occupational dose analysis, Decommissioning, Fire accident, Drop accident, AHP method

*Corresponding Author.
Chang-Lak Kim, KEPCO International Nuclear Graduate School, E-mail: clkim@kings.ac.kr, Tel: +82-52-712-7333

ORCID

Hyunjin Lee	http://orcid.org/0000-0002-3530-1247	Chang-Lak Kim	http://orcid.org/0000-0002-6931-9541
Sang-Rae Moon	http://orcid.org/0000-0003-3289-8591	Sun-Kee Lee	http://orcid.org/0000-0002-0481-068X

1. Introduction

Many nuclear power plants (NPP) around the world face decommissioning. More than 200 commercial, experimental, or prototype NPP and approximately 500 research reactors reached their operation license period and decommission phase is imminent [1]. Before commencing decommissioning, the owner of the plant must scrutinize these Structures, Systems, and Components (SSCs) and the site as each plant has its own unique physical and radiological characteristics. Workers will continue to be exposed to the decay of radioactive materials in the decommissioning stage and minimizing the personal dose as low as reasonably achievable will be an ongoing goal. In the phase of decommissioning, safety analysis is necessary. Safety analysis is one of the most crucial steps in decommissioning not only it is related to the safety of workers but also that of the public. This affects public acceptance. Thus, most research is focused on dose analysis for the public. However, in case of an accident during NPP decommissioning, radiological impacts on workers are far more severe than public. Therefore, in this paper, occupational dose analyses for a worker in decommissioning activity are conducted.

2. Safety Analysis

Safety analysis for decommissioning of a nuclear power plant is required to assess whether the radiological risk that occurred during decommissioning works meets the regulation for both workers and the public. The main purpose of the analysis is to guarantee the safety of workers and the public during the planned decommissioning activities and to grant legitimacy to a selection of decommissioning strategies. The period the analysis is carried out is from the defueling phase to the final site restoration phase. It is considered that the analysis should be done regularly because source terms are changed due to various dismantling activities. The radiological safety standard for workers suggested by the nuclear safety act is 20 mSv·year^{-1}.

Table 1. The scale of probability [3]

Scale of probability	Probability	Probability score
Very low	0–20%	1
Low	21–40%	2
Medium	41–60%	3
High	61–80%	4
Very high	81–100%	5

Table 2. The scale of radiological impact [3]

Radiological impact	Level of exposure	Impact score
Insignificant	<0.1 mSv·y^{-1} Onsite <0.01 mSv Offsite	1
Minor exposure	0.1–1 mSv·y^{-1} Onsite 0.01–0.1 mSv Offsite	2
Moderate exposure (Under dose limit)	1–20 mSv·y^{-1} Onsite 0.1–1 mSv Offsite	3
Major exposure (Above dose limit)	20–50 mSv·y^{-1} Onsite 1–5 mSv Offsite	4
Critical exposure	>50 mSv·y^{-1} Onsite >5 mSv Offsite	5

2.1 Safety Analysis Procedure

To conduct a safety analysis, quantifying risk is necessary. According to IAEA SRS No.77, a risk matrix is recommended [2]. The concept of risk matrix is evaluating the risk by product of the probability and radiological impact and they are scored in integers and described in Table 1 and Table 2, respectively [3].

2.2 Scenario Selection

Among the long procedure of decommissioning commercial NPP in Korea, the defueling operation is conducted in an earlier step. Thus accidents related to spent nuclear fuel are excluded. Since the purpose of this study is to find out the radiological impact of accidents during decommissioning NPP, non-radiological accidents are not

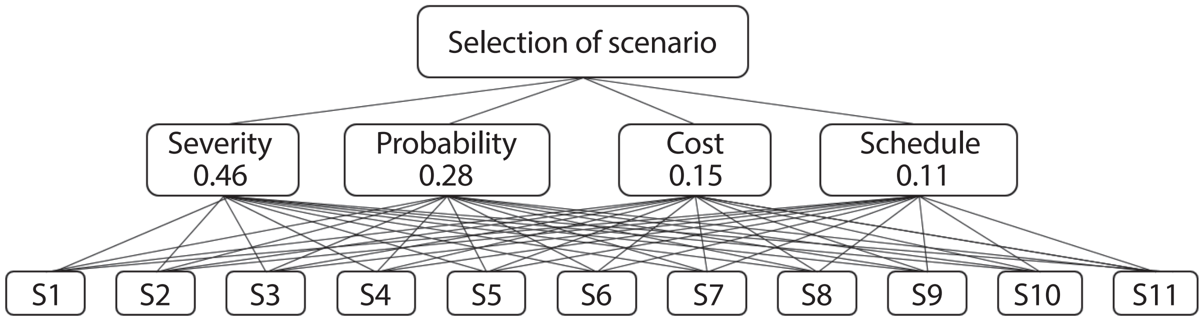

Fig. 1. Hierarchy for scenario selection [6].

Table 3. Final priority for accident scenarios [6]

Ranking	Scenario description
S1	Damage to the seal of waste resin drums due to a fire in the storage area of the radioactive waste drum, and leakage of some inventory
S2	Explosion accident during RV cutting
S3	Due to high temperature molten material leakage, worker burns, exposure, and workplace contamination
S4	Rupture of the vacuum filter bag during removal of activated concrete
S5	Contamination release during dismantlement of RCS
S6	The radioactive material in the drum diffuses into the air by dropping to the floor during transportation of the spent resin drum
S7	The decontamination waste liquid is dispersed into the air due to the damage to the decontamination waste liquid collection tank
S8	Concentrated contamination during decontamination of large component
S9	Accidental spraying of concentrated contamination with high-pressure spray during decontamination of large component
S10	Spread of contamination and exposure to workers during spent resin treatment
S11	Exposure of workers and spread of contamination due to leakage of concentrated waste fluid in the radioactive waste facility

in consideration. As a result, from 80 accident situations introduced in IAEA SRS 97 [4], 28 radiological accidents are chosen and four accident scenarios are presented in NUREG 0586 Appendix I [5]. As a result of the risk matrix, 11 scenarios are evaluated highest score of 12 [6].

2.3 Ranking of Scenarios

With risk matrix analysis, risk has been evaluated in only two criteria (radiological impact and probability) and it can be interpreted that 11 scenarios have the same amount of risk. To determine target scenarios to be analyzed, priority among the scenarios should be determined. Moreover, influences due to each accident scenario during decommissioning projects differ. In reality, accidents cause schedule delays of the work and additional costs. Thus, by taking into consideration of two additional criteria, a hierarchical structure for AHP (Analytic Hierarchy Process) can be built. AHP developed by Thomas Saaty in the 1970s is a method for experts in each hierarchy to determine the relative importance of complex problems through a pairwise comparison method and to derive the optimal situation. Criteria used in AHP are severity (radiological impact), frequency (probability), additional cost, and schedule delay. In order to determine the relative importance of the evaluation items, the evaluators composed of experts selected two evaluation items and performed a pairwise comparison. Experts who participated in the evaluation by the risk matrix method also participated in the evaluation by the AHP method. Each criterion has different importance and it is described in relative weighted value and they are shown in Fig. 1. And we can notice that this evaluation is valid by calculating the number of CR, a parameter that determines the validity of the analysis is below 0.1. The structure is described in Fig. 1.

As a result of the AHP method, the priority of each scenario can be found and Table 3 shows the result of the analysis.

3. Occupational Dose Analysis Method for Workers

It is remarkable that various accidents can occur during NPP decommissioning. And by conducting a risk matrix and AHP, we could find out the 11 most important accidents. Among them, 3 accidents were related to spent resin, and 2 of them are accidents during handling spent resin drums (S1, S6). One is a fire accident, and the other is a drop accident. It is worthwhile to choose two accident scenarios as a start of occupational dose analysis because they share similarities, for example, the source term of the scenarios and the size of the compartment.

3.1 General Analysis Approach&Calculation Process

The first thing to do in order to evaluate the dose for the worker is to find the concentration of the radioactivity leaked into the compartment (C_r). To perform the procedure, the necessary information is the working space (R), the amount of waste involved in the accident (V), the specific activity of each radionuclide (C_n), and the fraction of release of the radionuclide (f_w). The fraction of release is determined by accident conditions and the type of waste

Table 4. Value of the factors for activity concentration of compartment [7]

Factor	Value
Cn	3.612×10^{11} [Bq·m^{-3}]
V	Fire accident: 1.2 [m^3] (6 drums)
	Drop accident: 0.4 [m^3] (2 drums)
R	100 [m^3]
f_w	Drop accident: 1.00×10^{-4}
	Fire accident: 9.50×10^{-4}

form. The equation to acquire the concentration is written below:

$$C_r = \frac{C_n \times V \times f_w}{R}\ [\mathrm{Bq\cdot m^{-3}}] \quad (1)$$

Second, the effective external exposure dose (D_E) by immersion should be calculated. The effective dose conversion factor ($DCF_{E,i}$) is multiplied by the concentration of radioactivity of the compartment (C_r) and the exposure time (T). The equation to acquire the effective external exposure dose is written below:

$$DE = \textstyle\sum_i DCF_{E,i} \times C_r \times T\ [\mathrm{Sv}] \quad (2)$$

Finally, effective internal exposure (D_i) by inhalation dose should be obtained. The calculation method is similar to the effective external exposure dose, but the breathing rate (BR) must be additionally multiplied. For the conservative analysis, it is assumed that it is heavy load work, and the value of BR for such case is 1.2 m^3·h^{-1} [8]. The equation to acquire the effective internal exposure dose is written below:

$$D_I = BR \times \textstyle\sum_i DCF_{I,i} \times C_r \times T\ [\mathrm{mSv}] \quad (3)$$

3.2 Fraction of Release

A general analysis method of radionuclide transport cases is introduced in NUREG/CR-4370 [10]. According

Table 5. DCF value for internal and external exposure [9]

DCF	Value
Internal	1.70×10^{-8} [Sv·Bq^{-1}]
External	1.18×10^{-13} [Sv·s^{-1} per Bq·m^{-3}]

Table 6. Assumptions for the scenarios

No	Assumption
1	Only ^{60}Co exist in the waste
2	Temporary storage before transport to HIC (High Integrity Container)
3	Constant inventory of the radionuclide

to NUREG/CR-4370, the interaction factor (I_{air}) is related to the radionuclide release factor as follows:

$$I_{air} = f_o \times f_d \times f_w \times f_s \tag{4}$$

The interaction factor accounts for the transport of radionuclides to waste form to the environment. f_o, f_d, and f_s are the time delay factor, site design factor, and site selection factor respectively. f_w is the waste form and package factor. This factor describes a physical and chemical characteristics of the waste when the release of radionuclides is initiated. This factor varies depending on the type of accident scenario.

3.3 Assumption

In the analyses, spent resin that decontaminated the reactor coolant systems is chosen. In Table 6, assumptions are described.

4. Fire Accident Analysis

The fire scenario is modeled as 6 spent resin drums that are stored in decommissioning waste storage facility incinerated for 2 hours and after a worker discovered the accident, he puts on a protection mask whose protection factor is 10, and effective internal exposure dose become 1/10. And he tries to extinguish the fire for 10 minutes and evacuates (exposure duration: 10 minutes).

Table 7. Radionuclide-specific release fraction [10]

No	f_r
^{3}H	0.9
^{14}C	0.75
^{99}Tc	0.038
^{106}Ru	0.038
^{129}I	0.038
Particulates	0.019

4.1 Flammability Multiplier

The flammability multiplier is the number that varies depending on forms of waste and sorts of nuclides. According to NUREG/CR-4370 [10], waste form and package factor f_W in the case of fire accident is expressed as follows:

$$f_W = f_F = f_r \times 20^{-IFL} \tag{5}$$

f_r is a radionuclide-specific release fraction and accounts for nuclides release fraction when contaminated material is incinerated [10]. In this analysis ^{60}Co is selected as a nuclide, and 0.019 is chosen. Table 7 shows values of different f_r depending on nuclides.

IFL is the flammability index which is decided in the range of 0−3 according to its waste form and it is explained in Table 8. Since the source term of the accident is dewatered resin, the value of IFL is 1. As a result, f_F is 9.5×10^{-4}.

4.2 Calculation Result

The expected effective dose for a worker while a fire accident is written in Table 9.

Table 8. Flammability index [10]

IFL	Flammability tendency	Waste form
3	Non-flammable	Activated metal; Waste solidified in cement, etc
2	Low-flammability	Dewatered sludge; Calcined material solidified in synthetic polymer, etc
1	Burns if heat supplied	Dewatered ion exchange resin; Unsolidified filter cartridges, etc
0	flammable	Combustible trash; Liquid scintillation media, etc

Table 9. Calculation result of fire accident

Exposure type	Dose [mSv]
Internal	1.40×10^{0}
External	2.92×10^{-1}
Total	1.69×10^{0}

5. Drop Accident Analysis

The drop scenario is modeled as a spent resin drum which is stored in decommissioning waste storage facility is dropped upon another drum. Eventually, two drums have damage, which leads to radioactivity leakage. A worker finds out the incident and evacuates. Since the worker evacuate immediately without a protection mask after the accident, exposure time is assumed 10 minutes.

5.1 Release Fraction&Operational Dispersibility Factor

Waste form and package factor in the case of drop accident is defined as a product of release fraction (f_r) and operational dispersibility factor (f_c). According to NUREG/CR-4370 [10], waste form and package factor f_W in the case of drop accident is expressed as follows:

$$f_W = f_r \times f_c = f_r \times 10^{-ISC} \tag{6}$$

Table 10. Operational dispersibility factor [10]

ISC	Multiplier f_C	Waste form
3	0.001	Waste solidified using vinyl ester styrene, sealed sources
2	0.01	Waste solidified in cement
1	0.1	Trash, dewatered resins
0	1	Dewatered sludge, ash, dirt, miscellaneous powders

Table 11. Calculation result of drop accident

Exposure type	Dose [mSv]
Internal	4.76×10^{0}
External	1.02×10^{-2}
Total	4.77×10^{0}

The factor f_r is estimated based on work performed by the Department of Transportation (DOT) to determine requirements for the transport of radioactive assumed in which 0.1% of the contents of a waste package is released into the air [11]. And ISC is a measure of the potential for the contents of a waste container to be dispersed into the air due to an operational accident in which the waste container is severely damaged. ISC values according to waste form are written in Table 10. In this case, ISC is 1 and f_w is 1.0×10^{-4}.

5.2 Calculation Result

Expected effective dose for a worker while drop accident written in Table 11.

6. Conclusion

11 accident scenarios are evaluated as the most important among 32 scenarios by risk matrix considering radiological severity and probability. By introducing the AHP method with two additional criteria of cost and schedule delay, the priority of 11 accident scenarios is determined. Based on the

priority, target scenarios to analyze are determined.

The fire accident and the drop accident are selected as target scenarios for occupational dose analysis. Additional exposures due to the fire accident and the drop accident are 1.67 mSv and 4.77 mSv, respectively. Referring to the consequence of calculation, the drop accident can be considered more severe than the fire accident. However, a lower effective dose of the fire accident is due to wearing a protection mask. If it were not for a mask, the exposure dose would be increased to 14.2 mSv. Moreover, if the form of the waste or nuclide is more dispersible, or the activity of waste is higher, additional exposure is expected to be increased. For safety reason, wearing a mask at any operation should be recommended. By conducting the study, it is found that the radiological impact to workers in case of accident during decommissioning phase. In further study, the strategy to reduce the dose as low as reasonably achievable can be handled.

As a systematic approach, instead of temporary storage of spent resin in the 200 L drums, placing it directly in HIC is preferred. Thus, if waste is expected to be transported frequently, directly storing it in HIC can be an effective method for risk reduction.

Acknowledgements

This study was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP No. 20204010600130) and the Korea Foundation of Nuclear Safety (KOFONSD Project No. 2106023), Republic of Korea.

REFERENCES

[1] International Atomic Energy Agency-Power Reactor Information System. May 8 2022. “Permanent Shutdown Reactor-By Type.” IAEA homepage. Accessed May 10 2022. Available from: http://pris.iaea.org/PRIS/WorldStatistics/ShutdownReactorsByType.aspx.

[2] International Atomic Energy Agency, Safety Assessment for Decommissioning, Safety Reports Series No. 77 (2013).

[3] B. Park, J. Kim, and C. Kim, “Suggestion of Risk Assessment Methodology for Decommissioning of Nuclear Power Plant”, J. Nucl. Fuel Cycle Waste Technol., 17(1), 95-106 (2019).

[4] International Atomic Energy Agency, Management of Project Risks in Decommissioning, Safety Reports Series No. 97 (2019).

[5] United States Nuclear Regulatory Commission. Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities: Regarding the Decommissioning of Nuclear Power Reactor, U.S. NRC Report, 4-1~4-83, NUREG-0586 (2002).

[6] C.I. Ok, C.L. Kim, S.R. Moon, H.S. Koo, K.H. Yun, and S.K. Lee, “Prioritization of Radiological Accident Scenarios During Decommissioning of Nuclear Power Plants by Risk Matrix and AHP Method”, Ann. Nucl. Energy, 175, 109239 (2022).

[7] Electric Power Research Institute. Evaluation of the Decontamination of the Reactor Coolant Systems at Maine Yankee and Connecticut Yankee, EPRI Final Report, TR-112092 (1999).

[8] International Commission on Radiological Protection, The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Annals of ICRP (2007).

[9] K. Eckerman, J. Harrison, H.G. Menzel, and C.H. Clement, “ICRP Publication 119: Compendium of Dose Coefficients Based on ICRP Publication 60”, Ann. ICRP, 41(Supplement 1), 1-130 (2012).

[10] O.I. Oztunali and G.W. Roles. Updated of Part 61 Impact Analysis Methodology Vol. 1, U.S. NRC Report, NUREG/CR-4370 (1986).

[11] O.I. Oztunali and G.W. Roles. De Minimis Waste Impacts Analysis Methodology, U.S.NRC Report, NUREG/CR-3585 (1984).