On-site Dose Analysis in case of Spent Resin Handling Accident Process during NPP Decommissioning

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1. Introduction

The decommissioning of nuclear power plants requires adequate planning and demonstration so that it allows dismantling and decontamination activities to be conducted safely and on schedule. Existing safety standards require that an appropriate safety assessment be performed to support any activities related to the sitting, operation, modifications, and decommissioning of nuclear facilities [1]. All relevant hazards to workers, the public and the environment should be considered in the decommissioning safety assessment. But in this article, one specific handling accident scenario is selected and dose analysis for workers was performed.

2. Handling accidents Scenarios during Decommissioning Nuclear Power Plant and Dose Analysis

In this section, handling accidents cases occurred during decommissioning commercial nuclear power plants in the United States, will be handled. Furthermore, one of many handling accidents, one scenario is chosen that regarded very important and performed dose analysis for on-site workers.

2.1 Safety Assessment

In terms of safety assessment which is an essential part of the Final Decommissioning Plan, dose analyses for both ordinary dismantling activities such as cutting SSCs (reactor vessel, steam generator, etc.) and for handling accident cases are an essential part. Each dose analysis should include not only for workers but also public.

2.2 Handling accident

In this article, the lists of accidents included in NUREG-0586 were reviewed. More than 80 cases were introduced in the report. Among these accidents, leakage of radioactive material, fire, and container drop are frequently mentioned.

2.3 Spent Resin Packaging Process

While decommissioning nuclear facilities, a great amount of radioactive waste is generated and spent ion-exchange resin is of them. Not only it's categorized as Intermediate Level Waste but also its amount is not neglectable. A PWR-type reactor is expected to generate 6600L of spent resin per year. In order to safely handle spent resin, it needs to be immobilized and nuclear facilities have their own treatment system. For example, in APR1400, a polymer solidification system is installed as a part of the waste treatment system [2]. After polymerization, it is contained in 200L-drum and then stored in interim storage. However, various nuclear power plants in South Korea have different spent resin packaging facilities. In Table 1 various methods of packaging process of spent resin [1].

Table 1: Treatment Method for Spent Resin.

Treatment	Package Drum
Cement Solidification	200L Hannul C1 Concrete Hannul C2 Concrete
No treatment	200L (shielded)
Drying	200L Ferralium-HIC Polyethylene-HIC
Repacking	320L Kori Circle Concrete

2.4 Scenario Selection

In this scenario, it is assumed to occur in which a waste package (200L) drops from a significant height, and a portion of its contents is released into the air [3]. And the concentration of radioactive material in the container is assumed that meet the requirement for Low-Level-radioactive Waste limits. Table 2 shows the disposal limit of individual nuclides for LLW [4].

Table 2: LLW Disposal Limit of Individual Nuclides

Nuclide	Concentration(Ci/m^3)
H-3	1.68E+01
C-14	3.36E+00
Co-60	5.60E+02
Ni-59	1.12E+00
Ni-63	1.68E+02
Sr-90	1.12E+00
Nb-94	1.68E-03
Tc-99	1.68E-02
I-129	5.60E-04
Cs-137	1.68E+01

2.5 On-Site dose analysis

In the case of handling accidents during decommissioning, workers on sites are affected by its radiological impacts. Therefore, in this article, a dose analysis for workers on sites is conducted assuming that the above-selected scenario has occurred. The equation used to estimate exposure to workers inside the nuclear facility sites is as follow [3]:

$$H = \sum_n C_n \cdot f_r \cdot f_i \cdot f_c \cdot V_q \cdot f_q \cdot f_x \cdot PDCF_2$$

Where H is the 50-year dose commitment in mrem/yr.

C_n : Radionuclide concentration in the particular waste stream [Ci/m^3]

f_r : Fraction of the package contents released assuming a very dispersibility waste form

f_i : Fraction of dispersed into the air that is inhaled by an individual

f_c : Dimensionless correction factor, the relative dispersibility of various waste form in comparison with an extremely dispersible waste form

V_q : The volume of container [m^3]

f_q : Annual frequency of an accident of sufficient extent to result in the radioactive release.

f_x : Handling accident condition

$PDCF2$: Pathway Dose Conversion Factor 2 which is used for exposures that occur only once for considerably less than a year (acute exposure)

Radionuclide concentration in Table 2 is used for the concentration of the waste stream. According to reference [4], a handling accident of moderate severity, Factor f_r is estimated in which 0.1% of the contents of a waste package is released into the air. This is believed to be conservative as indicated by table 31 in reference [4]. Factor f_i represents the fraction of the material dispersed into the air that is inhaled by an individual. It is taken to be 0.001[5]. The factor f_c is estimated that 0.01 when waste is solidified in cement. Conservatively assuming that this incident occurs once a year, it is estimated to be unity. And Table 3 shows PDCF2 for nuclides corresponding to each organ.

Table 3: PDCF for Nuclides corresponding to Organs

PDCF2	Lungs	S.wall	Li. Wall	T.Body	Kidneys	Liver	Redmar	Bone	Thyriod
H-3	1.00E+09	1.00E+09	1.06E+09	1.00E+09	1.03E+09	9.92E+08	9.92E+08	7.88E+08	9.92E+08
C-14	4.94E+07	5.88E+07	5.78E+07	1.13E+08	6.34E+07	7.10E+07	1.94E+08	4.06E+08	5.18E+07
Ni-59	1.06E+10	1.02E+10	1.14E+10	1.06E+10	1.02E+10	1.00E+10	1.05E+10	1.04E+10	1.11E+10
Co-60	1.35E+12	4.27E+11	5.03E+11	4.61E+11	4.54E+11	5.21E+11	4.49E+11	4.45E+11	4.97E+11
Ni-63	2.58E+10	2.44E+10	2.82E+10	2.48E+10	2.44E+10	2.43E+10	2.43E+10	2.43E+10	2.34E+10
Sr-90	7.91E+10	1.58E+09	1.13E+11	1.92E+12	1.17E+11	1.17E+11	8.80E+12	7.60E+12	1.17E+11
Tc-99	7.70E+09	3.92E+09	4.07E+09	1.35E+09	2.78E+09	3.82E+09	1.95E+09	2.49E+09	8.56E+10
I-129	2.58E+10	1.46E+10	1.26E+10	5.32E+10	3.59E+10	1.91E+10	1.22E+10	3.86E+10	4.00E+13
Cs-137	2.06E+11	1.82E+11	1.97E+11	1.08E+11	4.85E+11	5.04E+11	4.71E+11	5.12E+11	4.54E+11

The calculation result is that workers will get 9.11mSv extra dose.

3. Conclusions

As a result, workers on-site face additional dose exposure. Furthermore, the methodology of diminishing the amount and activity of the ILW level spent resin is actively discussed. If auxiliary processes for waste ion-exchange resin are added during decommissioning, radiation workers are required more work safety.

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