RADIOECOLOGICAL ACTIVITY LIMITS FOR RADIOACTIVE WASTE DISPOSAL

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ABSTRACT

Near surface disposal is an option used by many countries for the disposal of radioactive waste containing mainly short lived radionuclides. Principally, disposal of radioactive waste requires the implementation of measures that will provide safety for human health and environment now and in the future. For this reason, preliminary activity limits should be determined to avoid radioecological problems. Radioactive waste has to be safely disposed in a regulated manner, consistent with internationally agreed principles and standards and with national legislations to avoid serious radioecological problems. The purpose of this study, presents a safety assessment approach to derive operational and post-closure radioecological activity limits for the disposal of radioactive waste. Radioecological activity concentration limits of each radionuclide in the waste (Bq/kg) were calculated. As a result of this study, radioecological activity limits are derived by calculating the highest dose for each selected radionuclide.

1. INTRODUCTION

Near surface disposal term includes broad range of facilities from simple trenches to concrete vaults. Disposal system has three components; the waste, the facility (incl. engineered barriers) and the site (natural barriers). Form of the waste (unconditioned or conditioned) is effective at the beginning of the migration scenerio. Existence of the engineered barriers in the facility will provide long term isolation of the waste from environment. The site characteristics (geology, groundwater, seismicity, climate etc.) are important for the safety of the system. Occupational exposure of a worker shall be controlled so that the following dose limits are not exceeded: an effective dose of 20mSv/y averaged over 5 consecutive years; and an effective dose of 50mSv in any single year. The effective dose limit for members of the public recommended by ICRP and IAEA is 1 mSv/y for exposures from all man-made sources [1,2]. Dose constraints are typically a fraction of the dose limit and ICRP recommendations (0.3 mSv/y) could be applied [3,4].

2. WASTE CHARACTERISTICS

It is possible to identify key radionuclides during the phases and over the timescales of disposal system. The list of radionuclides is presented in Table 1 for the post-closure phase which is considered to provide an illustrative list of disposed radionuclides of common interest. During the assessment, it might be necessary to take into account the in-growth of associated daughters. For a specific disposal facility, the list of radionuclides of interest will depend upon several factors. These factors are; source and nature of the radioactive waste streams, disposal facility specifications and duration of the institutional control time period.

Table 1. Selected radionuclides considered in post-closure assessment of near surface disposal.

H-3	Zr-93
C-14	Nb-94
Fe-55	Tc-99
Ni-59	I-129
Ni-63	Cs-134
Co-60	Cs-137
Sr-90	

In Table 1, certain radionuclides are excluded on the basis of their short half-life, in addition, near surface disposal facilities are based on minimum 30 year institutional control period. Ra-226 and Am-241 radionuclides are conditioned as retrievable method according to IAEA recommendations. These are long-lived radionuclides and they have to be disposed to deep geological repositories.

3. NEAR SURFACE DISPOSAL FACILITIES

According to their barrier system, near surface disposal facilities are constructed in the range of trench to vault. At one end, there is trench, which has minimally engineered barrier system. At the other end of this spectrum, there is vault, which has heavily engineered barrier system. Radioecological impacts have to be evaluated by using several scenarios. In this study leaching scenario was used for analyzing at the post-closure term of disposal system,

A) Trench

The trench is excavated into the ground without any engineered barriers. The upper surface of the trench is covered by uncontaminated soil or filling material. After filling of the trench, it is covered by further layer with drainage system for infiltrating rainwater (Fig.1).

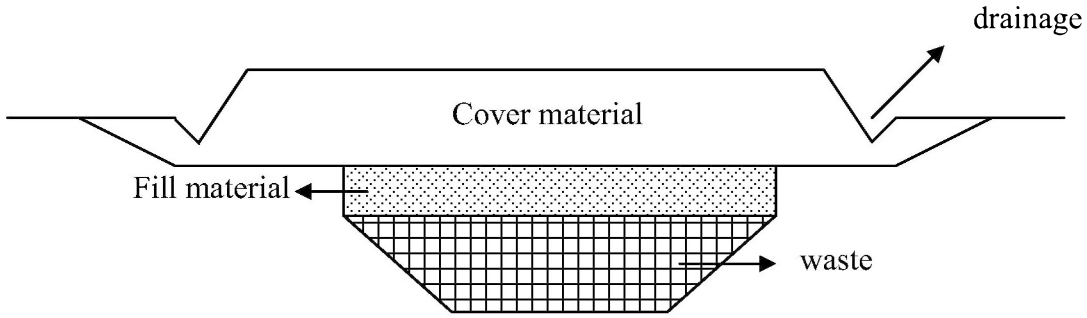

Figure 1. Cross-section of a closed trench.

B) Vault

The vault is excavated into the ground and lined with concrete. Waste packages are formed as concrete container. Radioactive wastes are emplaced into metallic drums and four drums are put into one concrete container. Vault is divided into sub-vaults by internal concrete walls. The waste packages are disposed regularly into each sub-vault. Each layer of waste packages is grouted for immobilization. After filling of vault, it is closed by placing of uncontaminated clay type filling material (Fig 2).

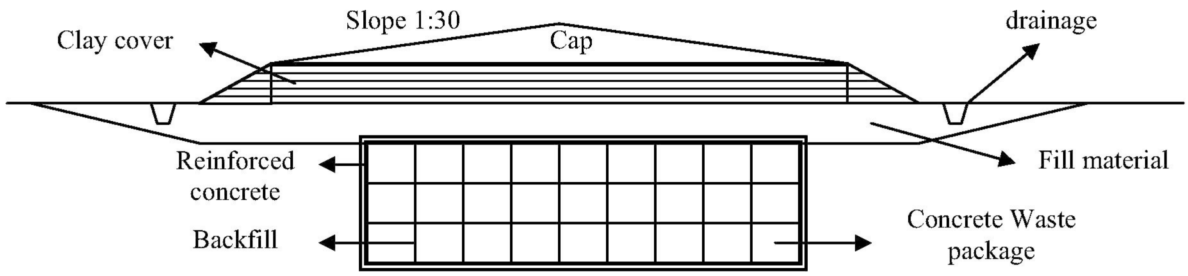

Figure 2. Cross-section of a closed vault.

Activity limits have been derived for disposal facilities (trench and vault) in different scenarios. These are; Clay and sandy materials in temperate and arid systems. Post closure leaching scenario has been developed for derivation of activity limits. Activity limits calculated using a dose limit of 1mSv/year for the public and 20 mSv/year for workers for each radionuclide. Activity limits calculated assuming a probability of unity for each scenario. Activity limits have been derived for trench with its minimally engineered features and vault with its heavily engineered features. Conceptual models are presented in Figure 1 and Figure 2.
In each case, conceptual model identifies;

1. the contaminant release mechanism causing release of radionuclides from the waste, and the media in which the radionuclides are released.
2. the contaminant transport media in which and through which the radionuclide move before reaching humans, and the associated transport mechanism.
3. the human exposure mechanism

4. DERIVATION OF ACTIVITY LIMITS

The leaching scenario is organized into a form that is amenable to mathematical representation. Basic assumptions are; facility dimensions, boundary conditions and transport mechanism. These assumptions comprise the conceptual model. The conceptual model of leaching scenario is expressed in mathematical form. Mathematical form has to be solved by using analytic and numerical techniques. Solution of this mathematical model is usually achieved by computer codes [5].

In post closure leaching scenario, dose limits of 1 mSv/year for members of the public is used according to radiation protection criteria. Calculation end points are determined as 1mSv/year. Institutional control period was taken as 300 years. Activity concentration limits of each radionuclide in the waste (Bq/kg) can be calculated by using;

$$Conc_i = Dose_{lim}\, C \,/\, Dose$$

$Conc_i$: radioecological activity concentration limit of radionuclide i (Bq/kg).
$Dose_{lim}$: is the dose limit (Sv/y).
Dose : is the dose resulting from the initial activity in the waste (Sv/y)
C : $A\, \rho\, V$
A : initial activity of the waste (Bq)
ρ : dry bulk density of the waste (kg/m^3)
V : volume of the waste (m^3)

Total activity limits of each radionuclide in the waste are calculated by using;

$$Amount_i = Dose_{lim} \,.\, A_i \,/\, Dose_i$$

$Amount_i$: total activity limits of radionuclide i (Bq).
$Dose_{lim}$: 1 mSv/year
A_i : initial activity of radionuclide i (Bq)
$Dose_i$: total dose resulting from initial activity A_i (Sv/year)

Activity limit for each radionuclide in the disposal facility have been established by using limiting condition and summation rule.

$$\Sigma_i = Q_i / Q_{iL} \leq 1$$

Q_i : actual activity of radionuclide i to be disposed (Bq/kg)
Q_{iL} : activity limit for radionuclide i if it were alone (Bq/kg)

Total activities (MBq) are presented in Table 2 for the post-closure leaching scenario for trench disposal.

Table 2. Total activities for the post-closure leaching scenario for the trench disposal

Radionuclide	Sand/Temperate	Sand/Arid	Clay/Temperate	Clay/Arid
H-3	10^5	10^7	10^{15}	10^{15}
C-14	10^5	10^8	10^{15}	10^{15}
Fe-55	10^{15}	10^{15}	10^{15}	10^{15}
Ni-59	10^8	10^{15}	10^{15}	10^{15}
Ni-63	10^{15}	10^{15}	10^{15}	10^{15}
Co-60	10^{15}	10^{15}	10^{15}	10^{15}
Sr-90	10^7	10^{11}	10^{15}	10^7
Zr-93	10^6	10^7	10^{15}	10^{15}
Nb-94	10^8	10^7	10^{15}	10^{15}
Tc-99	10^3	10^3	10^{11}	10^{10}
I-129	10^2	10^3	10^8	10^8
Cs-134	10^{15}	10^{15}	10^{15}	10^{15}
Cs-137	10^{15}	10^{15}	10^{15}	10^{15}

Similar calculations have been made for vault system and total activities (MBq) are presented in Table 3 for the post-closure leaching scenario for vault disposal.

Table 3. Total activities for the post-closure leaching scenario for the trench disposal

Radionuclide	Sand/Temperate	Sand/Arid	Clay/Temperate	Clay/Arid
H-3	10^7	10^7	10^{15}	10^{15}
C-14	10^7	10^8	10^{15}	10^{15}
Fe-55	10^{15}	10^{15}	10^{15}	10^{15}
Ni-59	10^8	10^{15}	10^{15}	10^{15}
Ni-63	10^{15}	10^{15}	10^{15}	10^{15}
Co-60	10^{15}	10^{15}	10^{15}	10^{15}
Sr-90	10^{10}	10^{11}	10^{15}	10^7
Zr-93	10^7	10^7	10^{15}	10^{15}
Nb-94	10^8	10^7	10^{15}	10^{15}
Tc-99	10^3	10^3	10^{11}	10^{10}
I-129	10^3	10^3	10^8	10^8
Cs-134	10^{15}	10^{15}	10^{15}	10^{15}
Cs-137	10^{15}	10^{15}	10^{15}	10^{15}

5. CONCLUSION

Radioecological activity limits are derived by calculating the highest dose for each selected radionuclide according to the radiological protection criteria. In this study only port-closure leaching scenario was evaluated. Although these activity limits are valid for conceptual designs and specific characteristics of the system as shown in this study, similar disposal systems will have similar results. These activities can be described as radioecological activity limits for near surface waste disposal systems. Because, these activities will not cause radiation doses higher than the effective dose limit for members of the public recommended by ICRP and IAEA in the post-closure leaching mechanism.

6. REFERENCES

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