Waste Segregation Based on Derived Clearance Levels

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ABSTRACT

This paper describes the methodology and results of a radiological modeling in support of an application to release very low level radiologically contaminated waste from regulatory control and allow its haulage and disposal in a hazardous waste landfill.

The Canadian regulatory body responsible for licensing operations involving nuclear materials (the Canadian Nuclear Safety Commission), has not yet formally defined clearance levels for free release of low level radiologically contaminated waste. The IAEA clearance levels have been derived for certain situations and receptor characteristics, which might be too conservative for an actual case. A site-specific pathways analysis was therefore completed to define conditional clearance levels using the concept of *de minimis* dose limit. Derived Conditional Clearance Levels were calculated for each radionuclide based on the maximally exposed hypothetical individuals to determine whether each waste stream can be “cleared” from regulatory controls.

The results showed that haulage of the waste from the station to the haulage/processing facility and transportation of waste or sludge from the haulage/processing facility to the disposal facility, handling of the waste or sludge at the haulage/processing facility, and incineration and/or disposal of waste or sludge at the disposal facility would not expose the workers to doses above 0.1 µSv/yr., which is less than the *de minimis* dose limit of 10 µSv/yr.

INTRODUCTION

In nuclear power stations, liquid chemical wastes originate from a variety of process systems and operations. These wastes include aqueous liquid industrial waste (commonly dilute solutions of acids, glycol, hydrazine, morpholine, and industrial cleaners), waste oils and fuels, laboratory and industrial solvents, boiler clean spent solvents and paint. Some of these liquid wastes meet criteria for treatment and release through the station Active Liquid Waste (ALW) systems. Other liquid wastes do not become contaminated with radionuclides, and are disposed of via external industrial liquid chemical waste services. The rest of the liquid chemical wastes become contaminated with radionuclides, and do not qualify for release through station ALW systems. These wastes are termed radioactive liquid chemical wastes.

These wastes may be generated in bulk, but are typically collected in smaller amounts and transferred to waste drums where they are segregated and stored in chemical waste collection areas. In some facilities, processing of chemical waste is performed, such as oil-water separation.

Once the wastes have been classified as either “radioactive” or “inactive”, disposition is initiated. For inactive chemical waste, disposal must be performed within 90 days or the Ontario Ministry of the Environment (MOE) is notified to justify continued storage.

The objective of this study was to conduct radiological modeling in support of an application to (i) allow the haulage from a nuclear power station (the station) of low level radiologically contaminated waste and its disposal in a hazardous waste landfill and (ii) increase the release limit for gross beta gamma activity for radiologically contaminated waste.

A pathways analysis was used in this study to calculate dose to hypothetical receptors including individuals such as truck drivers, incinerator workers, residue (ash) handlers, residents who live near the landfill, inadvertent intruders into the landfill after closure and residents who live near the outfall, to make sure that all exposure doses remain below a *de minimis* level.

A *de minimis* dose or dose rate represents a level of risk, which is generally accepted as being of no significance. Shipments of liquid chemical wastes with corresponding doses below *de minimis* can be sent to conventional (i.e., non-radioactive) landfills for incineration and disposal as the radioactive dose associated with them is much less than natural background. A similar approach was previously undertaken by Leung [1], Benovich [2], Garisto and Strain [3] and Garisto and Belanger [4].

This particular paper focuses on the methodology used to estimate the dose for each option of managing the liquid chemical waste. It illustrates the methodology with example calculations assuming 225 m^3 per year of waste is transported from the station. Other details are provided in Garisto and Parhizgari [5].

WASTE STREAMS

Typical radionuclide concentrations of the chemical liquid wastes are presented in Table I for the purposes of illustrative dose estimates presented below.

The total radionuclide activities in an annual volume of 225 m^3 of liquid chemical wastes exceed exemption activities (e.g., IAEA [6]). Therefore, the release of these wastes requires a pathways analysis to demonstrate that the associated dose to all potential receptors is below *de minimis* dose.

Table I. Typical Radionuclide Concentrations in Liquid Chemical Wastes

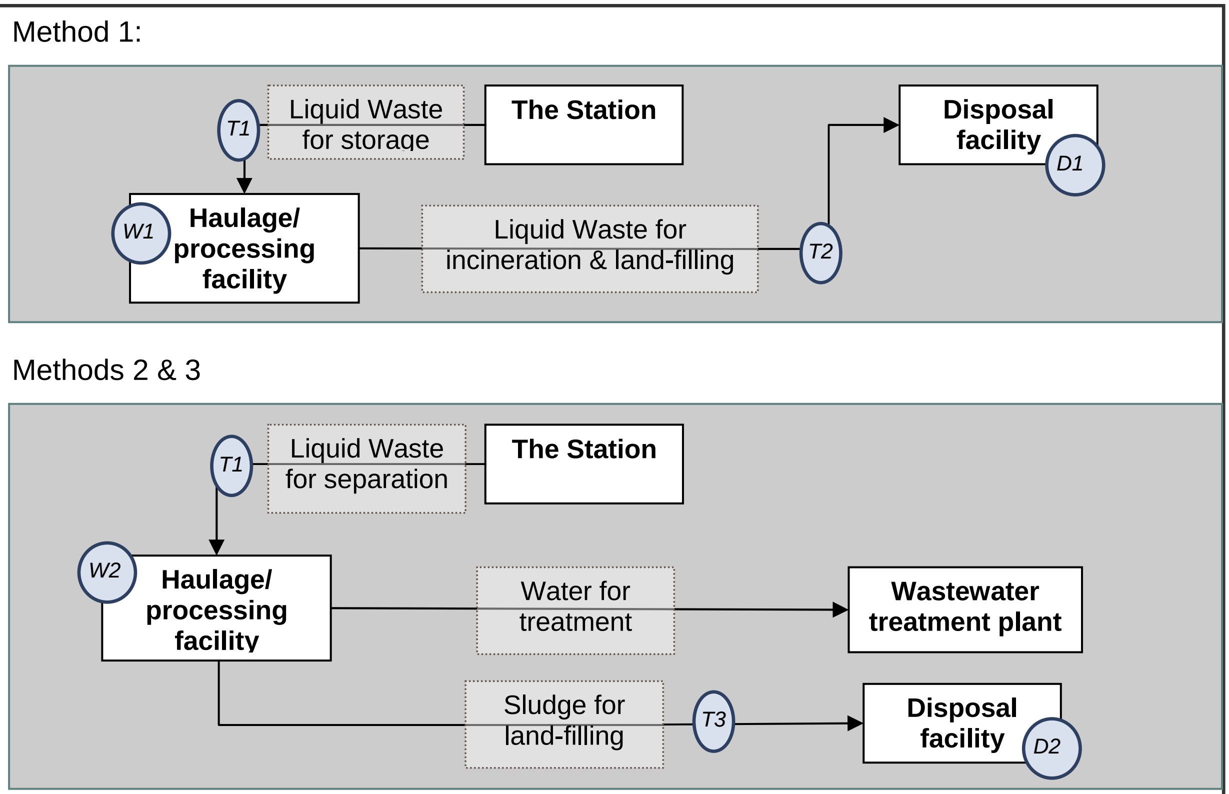
Radionuclide	Average concentration (Bq/g)
Co-60	0.0038
Cs-134	0.0013
Cs-137	0.0025
H-3	15.6

WASTE MANAGEMENT METHODS

The waste management methods assessed in this study are as follows (see Fig. 1):

- Transportation of waste from the station to the waste haulage/processing facility, storage for a short period of time and transportation to the disposal facility, where the waste can be incinerated and the ash disposed in a hazardous landfill, or
- Transportation of the waste from the station to the waste haulage/processing facility, storage for a short period of time, decanting water, transportation of the sludge to the

disposal facility for disposal in a hazardous landfill, and transportation of the decanted water to a wastewater treatment facility where it is chemically treated.



Notes: Receptor locations are shown in ovals (T_i: transport; W_i: waste haulage/processing; D_i: disposal)
Methods 2 and 3 involve different concentration ratios for liquid chemical waste: sludge

Fig. 1. Scenarios and receptors identification

DE MINIMIS DOSE CONCEPT

A *de minimis* dose or dose rate represents a level of risk, which is generally accepted as being of no significance. This level is derived from a small fraction of the annual dose due to natural background radiation, and thus represents an insignificant risk [7]. In Ontario, the individual dose from all natural sources is approximately 300 mrem (3 mSv) per year. The Atomic Energy Control Board (AECB) Regulatory Document, R-85, states that the AECB will use a *de minimis* dose criterion of 5 mrem (50 µSv) in a year to an individual provided that the potential for exposure to large populations is small [7]. Below this dose criterion, regulatory controls by the Canadian Nuclear Safety Commission (CNSC) are not required.

Since the issuance of R-85, the Advisory Committee on Radiological Protection (ACRP), an advisory body to the CNSC, have recommended a *de minimis* individual dose rate of 1 mrem (10 µSv) per year [8]. Their recommendation was based largely on that of the International Atomic Energy Agency (IAEA) regarding the principles to be used when exempting radiation sources and practices from regulatory control [9]. The recommended annual individual dose takes into account the potential exposure from multiple sources.

The CNSC (in draft) uses a dose of 10 μSv/y for the disposal requirements of nuclear substances [10]. It is interesting to note that the CNSC still considers a dose of 50 μSv/y to a member of the public to be a sufficiently low dose that below which an As Low As Reasonably Achievable (ALARA) assessment is not necessary [11].

For conservative purposes, this paper uses an individual dose rate of 1 mrem (10 μSv) per year as a *de minimis* level. Further, in order to take into account the cumulative impacts regarding other radioactive waste streams shipped to the haulage/processing facility and the disposal facility, a *de minimis* dose of 5 Sv/y was used in this analysis.

ESTIMATION OF EXPOSURE DOSES

This section describes the calculation of dose to hypothetical receptors from the transportation, storage and processing and disposal of liquid chemical waste. The dose calculations were carried out using several models as follows:

- The driver(s) transporting the wastes were assessed by MicroShield [12] and
- Receptors potentially impacted by the landfill disposal were assessed by IMPACTS-BRC [13]

The IMPACTS-BRC, Version 2.1 computer model [13] was used to estimate the dose to several hypothetical receptors. This code is a generic, radiological assessment code that was developed for use by the U.S. NRC to assist in the classification of waste streams as Below Regulatory Concern (BRC). The IMPACTS-BRC model estimates annual radiological doses to maximally exposed, hypothetical receptors as a result of transportation, treatment and disposal of wastes.

Dose coefficients for effective dose from inhalation and ingestion of radionuclides used in the IMPACTS-BRC model are based on tissue weighting factors recommended in ICRP 26 [14]. The effective dose coefficients were updated in this study to reflect ICRP 72 [15] recommendations.

The IMPACTS-BRC model was used to estimate doses associated with disposal of liquid chemical waste or the sludge produced from concentration this waste. However, the IMPACTS-BRC model does not provide much flexibility in terms of defining potential receptors or specifying the exposure scenarios. In particular, the transportation model in IMPACTS-BRC has pre-defined transportation parameters. Therefore, the driver dose was calculated using MicroShield v 6 [12]. MicroShield was also used to calculate external radiation doses from the liquid chemical waste during the waste storage and processing.

MicroShield is designed to incorporate the appropriate shielding with the corresponding geometry to calculate the gamma exposure at the specified distances. MicroShield reports dose rate for total radioactivity in a given shipment. This dose rate should be converted to the annual dose to the person exposed to the waste.

RECEPTORS

Several hypothetical receptors were defined in order to perform the pathways analysis. The selected receptors were expected to represent maximally exposed individuals (see Table II and Fig. 1).

Table II. Hypothetical Receptors for Pathways Analysis

Transportation-related	Haulage/ processing-related	Disposal-related
T1: Transportation of the liquid waste from the station to the haulage/processing facility	W1: Storage of liquid waste	D1: Incineration and landfill disposal of liquid waste
T2: Transportation of the liquid waste from the haulage/processing facility to the disposal facility	W2: Sludge separation	D2: Landfill disposal of sludge
T3: Transportation of the sludge from the haulage/processing facility to the disposal facility		

RELEASE LIMIT FOR GROSS BETA GAMMA ACTIVITY

The current release limit for gross beta gamma activity for radiologically contaminated waste from the station is 3.1E-7 µCi/ml, whereas the level recommended as acceptable for free release for the most restrictive gamma emitting radionuclide in the IAEA Safety Guide RS-G-1.7 is 2.7E-6 µCi/ml (or 0.1 Bq/g as shown in Table 2 of the Guide).

Among the radionuclides present in the waste, Co-60 is the most restrictive gamma emitter. The average concentration of Co-60 in the waste is 0.004 Bq/g (or 1.1E-7 µCi/ml). In order to estimate the acceptable release limit for gross beta gamma activity, the maximum dose calculated from exposure to such waste was compared to the *de minimis* dose and the activity acceptable for free release of the waste was back-calculated.

RESULTS

The dose calculations are discussed in detail by Garisto and Parhizgari [5]. The results of dose calculations are summarized in Table III and Table IV. Table III presents the calculated external dose to the driver and workers at the waste haulage/processing facility. Table IV presents the dose to the most exposed receptors for receptors at the waste disposal facility.

Table III. Calculated External Doses to Drivers and Workers Storing or Processing the Waste (Calculated by MicroShield)

Receptor	Source	External Dose (µSv/y)
Driver	Liquid chemical waste	0.074
Driver	Sludge (0.1)	0.091
Driver	Sludge (0.01)	0.087
Hose Operator	Liquid chemical waste	0.012
Sludge Handler	Sludge (0.1)	0.021
Sludge Handler	Sludge (0.01)	0.021

Table IV. Maximum Exposure Doses to Receptors at the Waste Disposal Facility (µSv/y)

Radionuclide	Exposure to Liquid Chemical Waste	Exposure to Sludge (0.1)	Exposure to Sludge (0.01 or 0.001)
H-3	3.7E-03	3.9E-03	2.3E-03
Co-60	1.2E-03	6.1E-06	6.3E-06
Cs-134	2.6E-04	1.4E-07	1.4E-07
Cs-137	2.0E-04	3.9E-05	4.0E-05
Total	0.005	0.004	0.002

The results show that for the illustrative calculations, all doses are much less than *de minimis*.

The Derived Conditional Clearance Limits for the most exposed receptors are shown in Table V. These DCLs were calculated based on the maximum doses over Table III and Table IV. The unity relationship (sum of concentrations over DCLs for all radionuclides) was checked for the three waste management methods and these results are shown as well.

Table V. DCLs Calculated for each Waste Management Method (Bq/g)

Radionuclide	Average concentration	Method 1	Method 2	Method 3
H-3	15.6	20919	20194	33700
Co-60	0.0038	0.36	0.29	0.30
Cs-134	0.0013	0.55	0.44	0.46
Cs-137	0.0025	1.4	1.1	1.2
Unity	-	0.015	0.019	0.018

The acceptable gross gamma activity was defined as the activity of Co-60 in the waste which will result in a dose of 5 µSv/yr to the most exposed receptor at 5.6E-6 µCi/ml.

CONCLUSIONS

A pathways analysis was conducted in support of an application to the CNSC to (i) allow the haulage and disposal from of low level radiologically contaminated waste by a specific contractor not licenced to haul or receive low level radioactive waste and (ii) increase the release limit for gross beta gamma activity for radiologically contaminated waste.

A site-specific pathways analysis was completed to define conditional clearance levels using the concept of *de minimis* dose limit. Derived Conditional Clearance Levels (DCLs) were calculated for each radionuclide based on the maximally exposed hypothetical individuals to determine whether each waste stream can be “cleared” from regulatory controls.

The lowest DCL for tritium (20,194 Bq/g) is higher than the currently approved limit of 74 Bq/g, demonstrating that the current limit results in much less than a *de minimis* dose. Nevertheless, it is not proposed to change the tritium limit.

For the available typical concentrations in waste, the results showed that:

- Haulage of the waste from the station and transportation of waste or sludge from the haulage/processing facility to the disposal facility will not expose the workers to doses above 0.091 µSv/yr
- Handling of the waste or sludge at the haulage/processing facility will not expose the workers to doses above 0.033 µSv/yr
- Incineration and/or disposal of waste or sludge at the disposal facility will not expose the workers and near-by resident to doses above 0.005 µSv/yr

The acceptable release limits for gross beta gamma activity for radiologically contaminated waste was estimated to be 5.6E-06, µCi/ml based on the most restricting limit for the three scenarios of (i) incineration of the liquid chemical waste and land disposal of the ash, and, (ii) and (iii) concentrating the waste with 1:10 and 1:100 sludge concentration ratios and landfill disposal of the sludge.

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