NRC Staff Analyses Identifying Potential Hazards Associated with the Disposal of Greater-Than-Class C and Transuranic Waste

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1. Introduction

The U.S. Nuclear Regulatory Commission's (NRC's) licensing requirements for the land disposal of Low-Level Radioactive Waste (LLRW) are provided in Part 61 of Title 10, "Energy," of the *Code of Federal Regulations* (10 CFR), "Licensing Requirements for Land Disposal of Radioactive Waste." Specifically, 10 CFR § 61.2 *waste* defines LLRW as "radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in paragraphs (2), (3), and (4) of the definition of Byproduct material set forth in § 20.1003 of this chapter." In 10 CFR § 61.55, the NRC has developed a classification system for LLRW which categorizes waste as Class A, B, C, or Greater-Than-Class C (GTCC). Based on this classification system GTCC waste is LLRW with concentrations of radionuclides that exceed the limits established by the Commission for Class C LLRW. GTCC waste is generated by nuclear power reactors, facilities supporting nuclear fuel cycle facilities, and other facilities and licensees outside of the nuclear fuel cycle. GTCC waste includes: (1) plutonium-contaminated nuclear fuel cycle wastes; (2) activated metals; (3) sealed sources; and (4) radioisotope product manufacturing wastes (i.e., wastes "occasionally generated as part of manufacture of sealed sources, radiopharmaceutical products and other materials used for industrial, education, and medical applications") [NRC,1987].

Recently, the Department of Energy (DOE) has described current and projected amounts of commercial GTCC waste and DOE GTCC-like waste in its Final Environmental Impact Statement (FEIS) [DOE, 2016]. DOE identified three general categories of waste (activated metals, sealed sources, and other) in estimating the quantity of waste from both current and potential (projected) facilities and activities. The source of the waste (e.g., commercial reactor and medical isotope production) defined the volumes and radionuclides associated with a particular waste stream. Volumes for specific waste streams and sources of the GTCC waste provided in DOE's FEIS are presented in Table 1-1 of this report. In addition to the variations in volume between waste streams, the characteristics of waste streams (e.g. composition of isotopes, concentrations, physical and chemical properties) can vary significantly depending on the specific waste stream (e.g., activated metals versus sealed sources). Thus, safety concerns with disposal of GTCC waste (e.g., offsite exposure, inadvertent intruder protection, security, and criticality) in addition to characteristics of the waste (e.g. heat output, gas generation) that are not typical of commercial LLRW are expected to vary significantly between specific waste streams.

The NRC staff has selected some of the waste streams in Table 1-1 to perform initial analyses to help identify potential hazards for disposal of GTCC waste using methods other than deep geological disposal. As the intent of these analyses is to identify potential hazards that should be evaluated as part of developing a regulatory basis, it was appropriate to evaluate a representative sample of different types of GTCC waste. As will be described in detail later, the NRC staff's analysis considered activated metals from commercial power reactors (Section 2), large and small sealed sources (Section 3), and GTCC other waste originating from medical isotope production (Section 4). The inventories considered in Sections 2-4 represent an appropriate range of radionuclides for identifying potential hazards with disposal of GTCC waste.

Relative to traditional LLRW, activated metals from commercial reactors contain a significant amount of heat generating radionuclides (relative to LLRW). Fission products and transuranic radionuclides may exist as surface contamination on the surfaces of activated metals. Some sealed sources contain a significant total quantity of the 30 year half-life radionuclide Cs-137 (i.e., approximately 63 billion MBq [1.7 million curies]), while other sealed sources contain primarily transuranic radionuclides. Mo-99 (medical isotope) production in the 'Other' waste stream category contains a significant amount of fissile radionuclides. Thus, the analyses considers characteristics of GTCC wastes such as transuranic radionuclides, heat generating wastes, fissile materials, and the quantity/volume of high activity waste. The analysis did not use waste streams in the DOE FEIS that were described as "GTCC-like" waste because these waste streams did not provide radionuclides or quantities significantly different from those in the GTCC waste streams.

These evaluations do not represent a safety determination for a particular waste stream nor are they an endorsement of any particular disposal method or disposal depth. The waste streams for these initial analyses were selected:

- To provide a range of characteristics that could be significant for evaluating the disposal of GTCC and transuranic wastes,
- To improve staff understanding of potential concerns with disposal of GTCC, and
- To provide input for and enhance NRC staff and stakeholder interactions for identifying potential concerns with disposal of GTCC waste.

Table 1-1 Estimated Volumes for Waste Streams Associated with GTCC Waste

Waste Stream Sources	Estimated Waste Volumes (cubic meters)		
	Activated metals	Sealed Sources	Other Waste
Existing Facilities and Activities			
Commercial Reactors	880		
Large Sealed Sources		1,000	
Small Sealed Sources		1,800	
Potential Facilities and Activities			
West Valley NDA exhumation	210		1,900
West Valley SDA exhumation	520		400
West Valley SNAP			1,200
Mo-99 Production			390
Overall TOTAL	1,610	2,800	3,890

NDA = NRC-licensed disposal area, SDA = State-licensed disposal area, SNAP = systems for nuclear auxiliary power

2. Disposal of Activated Metals (Commercial Reactors)

Activated metal waste from commercial reactors consists primarily of steel, stainless-steel, and a number of specialty alloys used in nuclear reactors. Metal materials used in nuclear power plants can become radioactive as a result of neutron irradiation 'activating' the iron, cobalt and nickel atoms in the materials of reactor internal components (e.g., baffle plates) that are subject to high neutron radiation fields (thermal neutron capture is the primary source of the activation) – hence the term 'activated metals' is used to characterize this waste stream. Most of the initial activated metal inventory comes from radionuclides (e.g., Co-60 and Ni-63, which have half-lives of approximately 5 and 100 years, respectively) with relatively short half-lives relative to the timeframes of 1,000 years and longer that are typically considered for radioactive waste disposal. Additionally, the Co-60 and Ni-63 inventories are activated radionuclides that occur throughout the metal materials and are not just on the metal surface. Generally, these short-lived radionuclides are the key contributors to the thermal output of the activated metal waste stream from commercial reactors.

Surface contamination may also contain transuranic isotopes and fission products from damaged fuel contaminating the reactor components. This surface contamination can contain radionuclides (e.g., Tc-99 [half-life of 210,000 years] and Pu-239 [half-life of 24,400 years]) more significant to disposal than the aforementioned short-lived radionuclides. The surface contamination, which is not a result of activation processes, must be considered when estimating the inventory for activated metals due to the potential for increasing radiation exposure.

2.1 Offsite Radiation Exposures from Activated Metals

The performance of a disposal facility in limiting potential public exposures is associated with the characteristics of the inventory (e.g., amount, half-life of the radionuclides, daughter products of radionuclides), the amount of dilution and dispersion, and the time delay it would take radionuclides to arrive at receptor locations (e.g., retardation along the groundwater pathway). One can assume that regardless of how robust the design of engineered barriers, radionuclides will eventually be released from the engineered facility and transported (typically via ground water) from a disposal site to a location of a potential receptor. A disposal facility is sited and designed to reduce/delay/impede the movement of radionuclides to potential receptor locations. The more effective the barriers, the longer it takes the wastes to migrate to potential receptors, thereby, reducing the hazard of the waste because of the reduction of the activities due to radioactive decay. The delay in reaching potential receptor locations will be dependent on the timing of a breach in a waste package and the potential for retardation along groundwater pathways.

A simplified calculation has been performed to determine the annual fractional release rate (FRR) that would result in an annual dose of 0.01 mSv (1 mrem) for each radionuclide at selected times representing the potential delay in arrival to an assumed receptor. Here the term

FRR is used as a measure of the performance a disposal facility would need to achieve to limit the annual dose to 0.01 mSv for that specific radionuclide, including the daughter products. The selection of an annual dose of 0.01 mSv to calculate the FRR for a single radionuclide is appropriate because a value of 0.01 mSv is easily scalable to a different annual dose for stakeholders that consider a different value to be more appropriate or are interested in comparison with a different annual dose, however, the purpose of the analysis is to assist the identification of potential hazards and not to draw any conclusions regarding compliance with a specific regulatory requirement. At two extremes a FRR of 1 would mean the entire inventory could be released in one year and the annual dose would be 0.01 mSv, whereas, a FRR of 1 in one-million (or 10^{-6}) represents a facility that needs to limit the release to one-millionth of the inventory per year to limit the dose to 0.01 mSv. This first set of FRRs are estimates of what a system would need to achieve to reduce doses to 0.01 mSv. Later in this document, ranges of FRRs are estimated based on assumed ranges of potential environmental conditions and disposal system designs. This second set of FRRs are ranges of values that disposal systems may be able to achieve for different designs and conditions.

The simplified calculations were developed using the following assumptions:

1.) Characteristics of the inventory

- Radionuclide amounts are based on the values in Table 2-1
- Disposal volume of 450 m^3

The radionuclides in Table 2-1 are representative of radionuclides that could be important to offsite ground water exposures (e.g., long-lived radionuclides) with different geochemical characteristics (e.g., solubility and sorption). These radionuclide inventories are almost entirely the result of surface contamination on activated metals. Ni-63 is one of the activated radionuclides that is present throughout the metal material and is included in Table 2-1 because it is estimated as having a concentration exceeding the Class C limit. Other examples of other activated radionuclides that may also exceed Class C limits are C-14, Ni-59, Nb-94.

NRC has used information available from measurements made at commercial reactor sites to estimate surface contamination inventories for the activated metal waste stream (Aber, 1986). To determine the inventory from the surface contamination it is assumed that the activated metals waste stream had an average surface area to volume ratio of approximately 150 m^2/m^3. This average surface area to volume ratio is based on 90% of the activated metals being comprised of components with a somewhat low surface to volume ratio (i.e., 80 m^2/m^3 representative of a baffle plate that is comprised of rectangular plates that are approximately 1 inch thick) and the remainder of the activated metals being comprised of a much higher surface area to volume ratio components (i.e., 800 m^2/m^3 representative of in-core instrument thimble tubes that has a tube wall thickness of approximately 1/20 of an inch). Thus, the inventories presented in Table 2-1 were determined by multiplying the surface concentration by the average surface area to volume ratio of 150 m^2/m^3 estimated for the activated metal components (i.e., 90% baffle plates and 10% tubes).

A waste volume of 450 m^3 was used in the analysis because it:

(i) represents a disposal volume that could be accommodated by a variety of disposal methods;
(ii) eliminates potential bias for waste streams with a large volume that might not be expected to go to a single disposal facility; and
(iii) is more consistent with releases from disposal unit sizes that might be captured by small- and intermediate-sized water wells.

In particular, a waste unit volume of 450 m^3 could be accommodated by: a near surface trench disposal unit with a footprint of 3 meters wide and 30 meters long and waste disposal occurs from 5 to 10 meters below the ground surface; an intermediate depth disposal unit with a footprint of 3 meters wide and 15 meters long and waste disposal occurs from 30 to 40 meters below the ground surface; or an intermediate depth borehole with a footprint of 4.5 meters and waste disposal occurs from 30 to 130 meters below the ground surface. These values assume a 100% packing efficiency; in practice packing efficiency would likely be in the 50-70% range.

2.) Delay time of radionuclides reaching the receptor location

- Delay times of 500; 5,000; and 50,000 years.

A delay of the arrival of radionuclides at the receptor location is based on a combination of degradation of the waste package, degradation of the waste form, and the time for radionuclides to be transported via ground water to the receptor location. The selected variation in delay times is useful for providing perspective of a variety of radionuclides (e.g., mobile versus immobile) as well as to identify the importance of facility characteristics. The shortest time of 500 years is intended to represent potentially extreme events that result in significant degradation of barriers of a robust facility (i.e., engineering and site) and transport of the more mobile radionuclides. A 500 year delay time may be representative of the expected delay associated with a LLW facility of more limited performance. The longest delay time of 50,000 years would represent a site with specific characteristics that isolates radionuclides from the environment very well.

3.) Characteristics of the receptor and dose calculation

- Capture well extracts 2 million liters of water per year
- Ingestion dose based on drinking 2 liters of well water per day

A small community well serving 25 people at 225 liters/day per person would require approximately 2 million liters of water to be withdrawn annually. Contaminant concentrations in ground water at the receptor location can be reduced by hydro-geologic conditions at a specific site, however, the current analysis only accounts for the dilution provided by the annual water demand of the well. DOE's FEIS assumed a farmer would use 2.5 million liters of water per year (DOE, 2016; page E-16). Internal radiation dose is calculated based on the assumption that the receptor drinks 2 liters of contaminated water per day and dose coefficients from Federal Guidance Report 13 [EPA, 2002] are applied.

The results of this calculation are presented in Table 2-2. The dose calculation included long-lived daughter products for the transuranic radionuclides (e.g., Np-237 as a daughter product of Am-241). Where a daughter product is listed in Table 2-2, the FRR is determined by the combined release of parent and daughter radionuclides present in the inventory. Consideration of the daughter product can be important to the FRR at longer time periods when a shorter half-life parent is substantially gone but the daughter product is still present. The daughter product is identified in Table 2-2 when it is the more significant radionuclide in determining the FRR. These FRRs are used as a measure of the performance a disposal facility would need to achieve to limit the annual dose to 0.01 mSv. The annual dose of 0.01 mSv does not represent a performance standard for the facility, rather it is a benchmark that can be used for comparison and is easily scalable to other values.

Generally, the FRR results (see Table 2-2) can be divided into two distinct groupings: (1) large FRRs such as values near 0.01 and above (a FRR of 0.01 indicates that a release of 1% of the entire inventory in a single year would result in an average annual dose of 0.01 mSv); and (2) small FRRs such as values near 10^{-4} and lower (i.e., annual release of 0.01% or less of the inventory the facility would be necessary to limit the average annual exposure to 0.01 mSv). The FRRs for the transuranic radionuclides present the most significant challenge, however, only Pu-239 was significant at 5,000 years and beyond. Although the transuranic radionuclides considered in this analysis (i.e., Pu and Am) require the lowest FRRs, release of these radionuclides to potential receptor locations are anticipated to be more limited by their solubility and retardation in the geosphere than the more mobile radionuclides such as Tc-99 and I-129.

Table 2-1 Key Radionuclides Associated with Surface Contamination on GTCC Activated Metals from Commercial Reactors.

Radionuclide and Half-life	**Average Surface Activity** (Ci/cm^2)	**Inventory for 100 m^3** (Ci)	**Concentration[a]** (Ci/m^3) [nCi/g]	**Relevant Part 61 Tables 1 and 2 Class C Limits**
Tc-99 (210,000 years)	1.5×10^{-10}	0.023	2.3×10^{-4}	3 Ci/m^3
I-129 (16,000,000 years)	1.4×10^{-12}	1.9×10^{-4}	1.9×10^{-6}	0.08 Ci/m^3
Sr-90 (28.5 years)	3.2×10^{-9}	0.49	4.9×10^{-3}	7,000 Ci/m^3
Cs-137 (30 years)	5.7×10^{-8}	8.6	0.086	4,600 Ci/m^3
Pu-238 (88 years)	1.1×10^{-9}	0.17	1.7×10^{-3} [0.85]	100 nCi/g
Pu-239 (24,000 years)	9.7×10^{-10}	0.15	1.5×10^{-3} [0.75]	100 nCi/g
Am-241 (430 years)	1.8×10^{-9}	0.27	2.7×10^{-3} [1.4]	100 nCi/g
Ni-63 (96 years)	8.0×10^{-7}	120	1.2	7,000 Ci/m^3
Activated Radionuclide[b]				
Ni-63 (96 years)	Not applicable	2.0×10^{6}	2.0×10^{4}	7,000 Ci/m^3

Note: 1 Ci = 37,000 MBq

a – Concentrations in nCi/g based on an assumed density of 2,000 kg per cubic meter of packaged waste (void space taken up by a solid such as cement).

b – Value for activated radionuclide throughout the metal taken from Table B-4 (DOE, 2016).

TABLE 2-2 Fractional Release Rates (FRRs) for Radionuclides in Activated Metal Wastes (identification of daughter product in table indicates it as primary contributor to FRR).

Radionuclides (half-life)	Release Rate at 500 years	Release Rate at 5,000 years	Release Rate at 50,000 years
Tc-99 (210,000 years)	0.01	0.01	0.01
I-129 (16 M years)	0.007	0.007	0.007
Sr-90 (28.5 years)	>1	>1	>1
Cs-137 (30 years)	0.1	>1	>1
Pu-238 & U-234 (88 years & 247,000 years)	2×10^{-4}	0.06 (U-234)	0.07 (U-234)
Pu-239 & U-235 (24,400 years & 700 M years)	5×10^{-6}	5×10^{-6}	2×10^{-5}
Am-241 & Np-237 (433 years & 2.1 M years)	7×10^{-6}	0.007	0.03 (Np-237)
Ni-63 (surface) (96 years)	3×10^{-4}	>1	>1
Activated Radionuclide			
Ni-63 (96 years)	2×10^{-8}	>1	>1

2.2 Performance of Facility Barriers for Activated Metal Wastes

To provide further understanding of the Fractional Release Rates (FRRs) presented in Table 2-2, the staff has generated FRRs for a variety of scenarios of facility characteristics and the inventory information of Table 2-1. These release rate calculations identify values of release rates that would result from various combinations of site and engineered barrier conditions. The main variables considered were the infiltration rates into the disposal units, the geochemistry of the disposal units, and the performance of the waste form (e.g., cementitious and glass waste forms). Emphasis was placed on release from the waste by leaching with water. Staff realized that, for some disposal facilities, other pathways and phenomena may contribute to the release. However, for most facilities, these secondary phenomena and pathways do not typically constitute significant contributions to release from the waste. It was assumed that releases via water infiltration and leaching would be most limiting and would be a good proxy for situations where other processes may be the main contributors to release. FRRs were not estimated for other pathways, such as erosional release.

Infiltration rates, geochemical parameters, and waste form performance are the assumed primary drivers of release rates to groundwater. Infiltration rates were assigned three states: low, medium, and high. Infiltration rates utilized were consistent with an arid to very arid site, a semi-arid site, and a humid site, respectively. Geochemical parameters were assigned two states: geochemistry "off" or geochemistry "on." When geochemistry was "off" there was no sorption and solubility limits were assigned very large (effectively infinite) values. When geochemistry was "on," the distribution coefficients of the waste form and the solubility limits assigned to fluids in the waste form environment were assigned values consistent with sand to sandy-loam soils (i.e., more acidic). Waste form performance was evaluated by assigning different normalized release rates (g/m^2-yr) to the waste form matrix. Waste form performance was represented by four states: none, minimal, moderate, and superior where none represented an instantaneous release, minimal represented a waste form with approximately 100 years of performance, moderate represented performance of an engineered cementitious waste form, and superior represented performance associated with a high-performance cementitious or glass waste form or other advanced waste form of similar normalized release rate (the performance period for the moderate and superior waste forms will depend on the specific parameters selected – periods can range from 100s of years to 100,000s of years for a superior waste form).

Table 2-3 provides the data assigned to the main variables. The values assigned are not representative of any particular site nor should they be used in licensing analysis for a particular facility. The values are generally representative of a range of conditions and levels of performance suitable for this analysis. The isotopes for which results were developed included short- and long-lived isotopes as well as mobile, moderately-mobile, and relatively immobile isotopes.

Table 2-4 provides the FRRs calculated for key radionuclides over a variety of disposal facility characteristics. Very limited release rates (i.e., less than one part in a million per year) are generally limited to disposal facility conditions with either low infiltration rates, superior waste forms, or both. As discussed above, the calculations were limited to releases by leaching in water. Calculations were run for 10,000 years, and the peak release rate was selected irrespective of the time of occurrence. Uncertainty/variability was included in the calculations to produce mean FRRs as well as 5^{th} and 95^{th} percentiles. In general, the uncertainty/variability corresponded to approximately a factor of +/- 2 to +/- 5 for the median result compared to the 5^{th} and 95^{th} percentiles. Only the mean results are shown below in Table 2-3. The first column provides the case analyzed by infiltration/geochemical/waste form performance state of the system. For example, a value of **low/off/none** corresponds to low infiltration, no geochemistry, and no waste form performance.

Table 2-2 (i.e., values a system would need to achieve to reduce doses to 0.01 mSv) shows the transuranic radionuclides present the greatest need for achieving low FRRs. Table 2-4 (i.e., values a disposal system may be able to achieve for different designs and conditions) shows a variety of site and design characteristics (5 of the 9 scenarios considered) were estimated to have FRRs that were lower (e.g., better performance) than what was estimated to be necessary

to achieve an annual dose of 0.01 mSv (i.e., FRRs in Table 2-2) for the most restrictive transuranic radionuclide considered in the activated metals analysis (i.e., Pu-239).

Table 2-3 Values Assigned to Key Variables for Fractional Release Rate (FRR) Calculations

Variable	Setting1	Setting2	Setting3	Setting4
Infiltration (mm/yr)	LU [0.1, 3] Low[1]	LU [5, 30] Medium	LU [50, 200] High	Not applicable
Geochemistry Kd (ml/g) Solubility (mol/L)	Kd's = 0 ml/g Solubility = unlimited GEO-Off	Sr: Kd=9; Sol=1.5x10^{-3} Tc: Kd=0.1; Sol=0.01 I: Kd=0.4; Sol=2 Cs: Kd=30; Sol=15 U: Kd=2; Sol=1 x10^{-3} Np: Kd=2; Sol=1.5 x10^{-2} Pu: Kd=300; Sol=2 x10^{-5} Am:Kd=100; Sol=1.5 x10^{-6} GEO-On	Not applicable	Not applicable
Waste form performance (g/m^2-day)	No waste form	LU [10, 100] Minimal	LU [0.1, 1] Moderate	LU [1x10^{-4}, 1 x10^{-3}] Superior

[1] The descriptor for the scenario is found under the distribution
LU = loguniform

Table 2-4 Disposal Facility Fractional Release Rates (FRRs) Based on Assumed Characteristics of a Potential Disposal Facility (infiltration/geochemistry/waste form)

#	Case*	Tc-99	I-129	Sr-90	Cs-137	Pu-238	Pu-239	Am-241
1	low/off/none	2x10^{-3}	2x10^{-3}	2x10^{-3}	2x10^{-3}	2x10^{-3}	2x10^{-3}	2x10^{-3}
2	high/on/none	3x10^{-2}	2x10^{-2}	8x10^{-4}	2x10^{-4}	3x10^{-5}	4x10^{-5}	1x10^{-4}
3	med/on/none	7x10^{-3}	3x10^{-3}	1x10^{-4}	3x10^{-5}	4x10^{-6}	5x10^{-6}	1x10^{-5}
4	low/on/none	5x10^{-4}	2x10^{-4}	6x10^{-6}	2x10^{-6}	2x10^{-7}	3x10^{-7}	8x10^{-7}
5	high/off/minimal	3x10^{-3}	3x10^{-3}	2x10^{-3}	2x10^{-3}	2x10^{-3}	3x10^{-3}	3x10^{-3}
6	high/off/moderate	3x10^{-5}	3x10^{-5}	2x10^{-5}	2x10^{-5}	3x10^{-5}	3x10^{-5}	3x10^{-5}
7	high/off/superior	3x10^{-8}	3x10^{-8}	2x10^{-8}	2x10^{-8}	3x10^{-8}	3x10^{-8}	3x10^{-8}
8	low/on/moderate	2x10^{-5}	2x10^{-5}	4x10^{-9}	1x10^{-9}	4x10^{-10}	5x10^{-8}	6x10^{-9}
9	low/on/superior	3x10^{-8}	2x10^{-8}	4x10^{-12}	1x10^{-12}	4x10^{-13}	7x10^{-11}	6x10^{-12}

* The case designators are for infiltration/geochemistry/waste form performance. For example, case 8 represents an arid site in combination with an engineered waste form. Differences in results between isotopes of the same element can be mostly attributed to round off.

2.3 Fissile Material in Activated Metals

For disposal of GTCC waste that contains a significant amount of fissile material, criticality safety and security may need to be assessed. The major material characteristics important to criticality safety include the total quantity, concentration, enrichment (applicable to uranium-235 only), physical and chemical form, potential for accumulation and concentration of fissile material, and the availability of moderator material, including special moderator materials such as beryllium, carbon, heavy water or combination of these material in various forms.

The total quantity of the fissile materials is the most important parameter in assessing the criticality safety and security of the disposal of GTCC waste containing fissile material. A total quantity of fissile material not exceeding the respective regulatory exemption quantity does not need an assessment on the criticality and security. For example, an Agreement State normally can license Special Nuclear Material (SNM) quantities that are less than or equal to the quantities in 10 CFR Part 150. A critical mass is defined at §150.11(a) as:

> For the purposes of this part, special nuclear material in quantities not sufficient to form a critical mass means uranium enriched in the isotope U-235 in quantities not exceeding 350 grams of contained U-235; uranium-233 in quantities not exceeding 200 grams; plutonium in quantities not exceeding 200 grams; or any combination of them in accordance with the following formula: For each kind of special nuclear material, determine the ratio between the quantity of that special nuclear material and the quantity specified above for the same kind of special nuclear material. The sum of such ratios for all kinds of special nuclear materials in combination shall not exceed unity. For example, the following quantities in combination would not exceed the limitation and are within the formula, as follows: (175 (grams contained U-235/350)+(50 grams U-233)/200)+(50 grams Pu/200)=1.

If the total fissile material exceeds the exempted quantity for the specific application, an assessment of the material concentration must be evaluated. The overarching concern for criticality safety of a GTCC disposal site is whether a sufficient amount of fissile materials can be accumulated to impose a criticality safety hazard. For example, Pu-239 is a fissile radionuclide present as surface contamination for activated metals. Using the activity value for Pu-239 in Table 2.2 for the volume of activated metal waste used in the off-site dose calculation, the total activity for Pu-239 is 0.68 Ci and equals approximately 40 milligrams of Pu-239, which is an amount that does not present a criticality concern.

2.4 Thermal Output from Activated Metals

Heat generation due to radioactive decay for activated metals from commercial reactors is principally due to decay heat from Ni-63. The heat generated per cubic meter of waste from radioactive decay of Ni-63 is approximately 20 watts per cubic meter and decreases to approximately 0.5 watts per cubic meter at 500 years (see Table 2-5). Heat will be dissipated to the environment from the disposed waste. However, depending on the design and site

conditions, there could be an appreciable increase in temperature over a period of decades that could result in the degradation of waste form performance, adverse impacts to other engineered barriers, and changes to the environmental conditions within the disposal units. Temperatures could be elevated more than 20 degrees centigrade over ambient temperatures, and possibly much more, depending on waste loading and thermal design of the facility.

Temperature rise in the waste or disposal facility can, in most cases, be managed by proper thermal design. Proper sizing of the waste forms and spacing of the waste forms in the disposal facility can limit the temperature rise. A temperature rise to near or above the boiling point of water would need to be avoided, as the uncertainties associated with disposal system performance could increase substantially.

Table 2-5 Thermal Output for Ni-63 in the Activated Metal Waste Stream for Commercial Reactors

Radionuclide (half-life) [Initial Activity]	Volume (cubic meters)	Initial Thermal Output (watts)	Thermal Output at 100 years (watts)	Thermal Output at 500 years (watts)
Ni-63 (96 years) [9.2 MCi]	880	18,000	8,800	490

Note: 1 Ci = 37,000 MBq

2.5 Gas Generation from Activated Metals

Radioactive waste has the potential to generate hydrogen through radiolysis, chemical reactions, thermal degradation, and biological activity. Because of limitations and restrictions provided by the waste characteristics requirements in 10 CFR § 61.56, hydrogen gas generation by other mechanisms may be more limited than hydrogen gas generation by radiolysis. For disposal of low activity waste in a traditional trench-type facility, gas generation is unlikely to be a significant phenomenon because the propensity to generate the gas is low and if hydrogen gas is generated the gas is generally free to flow into the surrounding media. However, for GTCC waste that may be disposed in a more highly-engineered facility, because both the generation rate may be higher and the sealing of the waste containers and engineered barriers may be tighter, there is an elevated potential to generate high concentrations of hydrogen gas in the facility.

The amount of gas that is generated is driven by the composition of the waste, and the amount and type of radiation that is emitted. Radiolytic generation of hydrogen occurs when ionizing radiation interacts with hydrogenous materials. The metric used to describe hydrogen generation from a particular material undergoing radiolysis is the G-value, which has units of molecules of gaseous hydrogen product per 100 eV of radioactive decay energy absorbed. G-values are experimentally measured. NUREG/CR-6673 “Hydrogen Generation in TRU Waste Transportation Packages” summarizes reasonably bounding $G(H_2)$ for materials generally representative of those commonly present in TRU or transuranic waste (LLNL, 2000). Some

types of GTCC waste (i.e., activated metals, sealed sources) are not anticipated to generate significant amounts of gas from the waste itself because of the limited amounts, if any, of water or organic substances likely to be present in the waste form. However, these wastes could generate gases outside the waste form depending on the intensity of radiation emitted and the shielding of the waste and disposal containers/engineered barriers.

3. Disposal of Sealed Sources

DOE's FEIS describes two categories of sealed sources, namely small sealed sources and large sealed sources (Cs-137 irradiators). DOE's analyses assumed that the small sealed sources would be packaged in approximately 8,700 drums (208 liter [55 gallon]) for disposal. The large sealed sources (1,435 Cs-137 irradiators) in the waste inventory were assumed to each have a total volume of 0.71 m^3 (25 ft^3) and would be disposed individually in their original shielded devices (i.e., these irradiators cannot be packaged in 55-gallon drums). DOE stated that for these large irradiators, the Cs-137 source is contained within a very robust shielded device, which is expected to retain its integrity for many years following disposal (DOE, 2016; page B-21). Although Cs-137 has a 30 year half-life, the initial inventory for the large sealed sources is more than 37 billion MBq (one-million curies). The small sealed sources contain transuranic radionuclides, which include plutonium and americium. Table 3-1 provides the characteristics (i.e., inventory and half-lives) for the significant radionuclides associated with the disposal of sealed sources.

3.1 Offsite Radiation Exposures from Sealed Sources

The staff calculated FRRs for sealed sources based on the same approach described for the activated metal waste in Section 2 (i.e., based on limiting the annual offsite dose to no more than 0.01 mSv [1 mrem] for each radionuclide and an assumed disposal volume of 450 m^3 for the large sealed sources and 450 m^3 for the small sealed sources). Table 3-2 provides the release rates for each of the radionuclides considered for the sealed sources. Clearly, the small sealed sources provide the greatest challenge for disposal based on the calculated FRR for Pu-239 of 6 parts in one billion after 50,000 years of decay.

3.2 Performance of Facility Barriers for Sealed Sources

As described in Section 2.2, infiltration rates, geochemical parameters, and waste form performance are expected to be the primary drivers of release rates. Table 2-4 provides the calculated FRRs for the key radionuclides over a variety of disposal facility characteristics. Very limited release rates (i.e., less than one part in a million per year) are generally limited to disposal facility conditions with either low infiltration rates, superior waste forms, or both (as noted in Section 2.2, the calculations were limited to releases by leaching in water). The release rates estimated for sealed sources to achieve an annual dose of 0.01 mSv tend to be very low for the small sealed sources. A combination of barriers would be needed to limit facility releases to these low values (see Table 2-4).

A comparison of the FRRs in Table 2-4 (i.e., values a disposal system may be able to achieve for different designs and conditions) and the FRRs in Table 3-2 (i.e., values a system would need to achieve to reduce doses to 0.01 mSv) shows that transuranic radionuclides in the small sealed sources waste stream can be a significant concern, especially Pu-239 (i.e., FRR on the order of 6 in one billion after 50,000 years of decay in Table 3-2). Only one of the scenarios estimated in Table 2.4 had a FRR for Pu-239 that was less than the very low FRR estimated to be necessary to limit the annual dose to 0.01 mSv (see Table 3-2). The one scenario represented the best site and facility characteristics (i.e., low infiltration, geochemistry considered, and a superior waste form).

Table 3-1 Key Radionuclides in the Sealed Sources Waste Stream

Radionuclide and half-life	**Inventory** (Ci)	**Concentration[a, b]**	**Relevant Part 61 Tables 1 and 2 Class C Limits**
large sealed sources			
Cs-137 (30 years)	1,700,000	1,700 Ci/m^3	4,600 Ci/m^3
small sealed sources			
Pu-238 (88 years)	120,000	33,000 nCi/g	100 nCi/g
Pu-239 (24,000 years)	8,400	2,300 nCi/g	100 nCi/g
Am-241 (430 years)	150,000	42,000 nCi/g	100 nCi/g

Note: 1 Ci = 37,000 MBq

a - based on 1,000 cubic meters for Cs-137 and 1,800 cubic meters for Pu and Am isotopes

b - values assume single radionuclide present; and grams per cubic meter based on an assumed density of 2,000 kg per cubic meter.

TABLE 3-2 Fractional Release Rates (FRRs) for Radionuclides in Sealed Sources (the daughter radionuclide is identified when it is the dominant contributor to the FRR)

Radionuclides (half-life)	**Release Rate at 500 years**	**Release Rate at 5,000 years**	**Release Rate at 50,000 years**
Cs-137 (30 years)	7×10^{-6}	>1	>1
Pu-238; U-234 (88 years; 247,000 years)	5×10^{-9}	1×10^{-6} (U-234)	2×10^{-6} (U-234)
Pu-239; U-235 (24,400 years; 700 M years)	1×10^{-9}	2×10^{-9}	6×10^{-9}
Am-241; Np-237 (433 years; 2.1 M years)	2×10^{-10}	2×10^{-7}	9×10^{-7} (Np-237)

3.3 Fissile Material in Sealed Sources

As described in Section 2.3, the total quantity of the fissile materials is an important factor in assessing the criticality safety of a disposal site. Based on the total quantity of a single fissile material or equivalent quantity of combined fissile materials, a GTCC disposal site/facility can be characterized as exempted or non-exempted from requirements of criticality safety consideration. The overarching concern for criticality safety of a GTCC disposal site is whether a sufficient amount of fissile materials can be accumulated to impose a criticality safety hazard. Accumulation may occur in a disposal facility as a result of natural processes that may mobilize and concentrate radionuclides. Pu-239 is a fissile radionuclide present in small sealed sources. Based on the activity value for Pu-239 in Table 3.1, the amount of Pu-239 at 100 years contained in a volume of 450 m^3 is approximately 30 kilograms.

3.4 Thermal Output from Sealed Sources

Heat generation due to radioactive decay in sealed sources is due to a variety of radionuclides (see Table 3-3). Initially, the thermal output from small sealed sources is approximately 3 watts per cubic meter (primarily heat generated by the decay of Pu-238 and Am-241) and decreases to approximately 1 watt per cubic meter at 500 years. The thermal output for the large sealed sources is initially 2 watts per cubic meter and decreases to effectively zero after 500 years as the primary source of the heat is from the decay of Cs-137, which has a 30 year half-life. The thermal output from sealed sources is not considered significant.

Table 3-3 Thermal Output for Key Radionuclides in Sealed Sources Waste Stream

Radionuclide (half-life) [Initial Activity]	Volume (m^3)	Initial Thermal Output (watts)	Thermal Output at 100 years (watts)	Thermal Output at 500 years (watts)
large sealed sources				
Cs-137 (30 years) [1.7 MCi]	1,000	1,900	190	<<1
small sealed sources				
Pu-238 (88 years) [0.12 MCi]	1,800	3,900	1,800	<1
Pu-239 (24,000 years) [8,400 Ci]	1,800	250	250	250
Am-241 (430 years) [0.15 MCi]	1,800	4,900	4,200	2,200

Note: 1 Ci = 37,000 MBq

3.5 Gas Generation from Sealed Sources

As discussed briefly in Section 2.5, radioactive waste has the potential to generate hydrogen through radiolysis, chemical reactions, thermal degradation, and biological activity. The characteristics of certain GTCC waste and the type of facility (e.g., more highly-engineered facility) has the potential to generate high concentrations of hydrogen gas should sufficiently high radiation fields be present. However, sealed sources are not anticipated to generate significant amounts of gas from the waste itself because limited to no water or organic substances are likely to be present. Sealed sources do contain sufficient quantities of radionuclides (e.g., Cs-137, Am-241, and Pu-238) that have the potential to generate gas outside of the waste form depending on the shielding of the waste and disposal containers/engineered barriers.

4. Disposal of Other Waste (Mo-99 Production)

DOE's FEIS included a category entitled GTCC 'other' waste in its current and projected amounts of GTCC waste. The GTCC 'other' waste category included waste streams associated with a potential exhumation of radioactive waste from the West Valley site in New York (DOE, 2016; page B-4) as well as radioactive waste resulting from the production of radioisotopes for nuclear imaging procedures. The staff's analysis of the GTCC 'other' waste considered only the Mo-99 production waste stream because Mo-99 production offered a different type of waste stream not considered in the other two categories. DOE estimated the total volume for the Mo-99 production waste stream was 390 m^3 that would be generated over a 71-year period (DOE, 2016; page B-5). The current analysis does account for the radioactive decay that would occur over the 71-year period. Thus, the estimated inventory is conservative for those radionuclides with short half-lives relative to a 71-year period.

Table 4-1 provides the characteristics (i.e., inventory, half-lives, and waste concentrations) for the significant radionuclides associated with the Mo-99 production waste stream and the relevant Class C limits in 10 CFR Part 61. For the set of radionuclides shown in Table 4-1, Pu-239 is the one radionuclide that exceeds the Class C limit of 100 nCi/gm for transuranic radionuclides with half-lives greater than 5 years.

4.1 Offsite Radiation Exposures from Mo-99 Production Waste

The staff's analysis of offsite ground water releases for the Mo-99 production waste stream uses the same approach as described for the activated metal waste in Section 2 (i.e., based on limiting the annual offsite dose to 0.01 mSv [1 mrem] for each radionuclide). However, the analysis for the Mo-99 production uses a volume of 390 m^3, which is the total volume DOE provided for this waste stream in its FEIS, rather than the larger volume of 450 m^3 that was assumed in the analyses for the activated metals and sealed sources waste streams. Table 4-2 provides the FRRs for each of the radionuclides considered for the Mo-99 production waste

stream. Generally, the FRR results (see Table 4-2) at early times (i.e., 500 years) are small values (e.g., on the order of 10^{-4} per year and lower) indicating the facility needs to perform well over a wide spectrum of radionuclides. At later times (e.g., 5,000 years), the FRRs are lowest for U-238, Pu-239, and Pu-240. Pu-239 is quite low with a release of rate of 3×10^{-8} per year. As with other waste streams evaluated above, some of the transuranic radionuclides (e.g., Pu-239) considered in the analyses require the lowest FRRs. Release of these radionuclides to potential receptor locations are anticipated to be more limited by their solubility and retardation in the geosphere than the more mobile radionuclides such as Tc-99 and I-129.

4.2 Performance of Facility Barriers for Mo-99 Production Waste

As described in Section 2.2, infiltration rates, geochemical parameters, and waste form performance are expected to be the primary drivers of release rates. Table 2-4 provides the calculated FRRs for the key radionuclides over a variety of disposal facility characteristics. Very limited release rates (i.e., less than one part in a million per year) are generally limited to disposal facility conditions with either low infiltration rates, superior waste forms, or both (as noted in Section 2.2 the calculations were limited to releases by leaching in water). The release rates estimated for the Mo-99 production waste stream to achieve an annual dose of 0.01 mSv tend to be quite low for transuranic radionuclides. A combination of barriers would be needed to limit facility releases to these low values (see Table 2-4).

A comparison of the FRRs in Table 2-4 (i.e., values a disposal system may be able to achieve for different designs and conditions) and the FRRs in Table 4-2 (i.e., values a system would need to achieve to reduce doses to 0.01 mSv) shows that transuranic radionuclides present the largest concern and primarily Pu-239 presents a concern for the long term (i.e., beyond 500 years). A limited set of the site and design characteristics used in Table 2-4 (2 of the 9 scenarios considered) were estimated to have FRRs that were lower (e.g., better performance) than what was estimated to be necessary to achieve an annual dose of 0.01 mSv (i.e., FRRs in Table 4-2) for Pu-239.

4.3 Fissile Material in Mo-99 Production Waste

As described in Section 2.3 the total quantity of the fissile materials is an important parameter in assessing the criticality safety of a disposal site. Based on the total quantity of a single fissile material or equivalent quantity of combined fissile materials, a GTCC disposal site/facility can be characterized as exempted or non-exempted from requirements of criticality safety consideration. The overarching concern for criticality safety of a GTCC disposal site is whether a sufficient amount of fissile materials can be accumulated to impose a criticality safety hazard. Accumulation may occur in a disposal facility as a result of natural processes that may mobilize and concentrate radionuclides. Fissile radionuclides present in the Mo-99 production waste stream include U-235 and Pu-239. Based on the activity value for these fissile radionuclides in Table 4.1, the amount of fissile material at 100 years for the estimated volume of 390 m^3 is approximately 2 kilograms of Pu-239 and 1,700 kilograms of U-235. Although the quantity of

U-235 is a large amount of fissile material, the U-235 will be present as low-enriched uranium targets.

Table 4-1 Key Radionuclides Associated with the Mo-99 Production Waste Stream (inventory values from Argonne, 2010; Table 4-4)

Radionuclide and Half-life	**Total Activity** [Ci]	**Inventory for 100 m^3** [Ci]	**Concentration** [a] (Ci/m^3) [nCi/g]	**Relevant Part 61 Tables 1 and 2 Class C Limits**
Tc-99 (210,000 years)	11.2	2.9	0.029	3 Ci/m^3
I-129 (16M years)	0.02	5.1×10^{-3}	5.1×10^{-5}	0.08 Ci/m^3
Sr-90 (28.5 years)	9.1×10^{4}	2.3×10^{4}	230	7,000 Ci/m^3
Cs-137 (30 years)	9.5×10^{4}	2.4×10^{4}	240	4,600 Ci/m^3
Pu-238 (88 years)	0.6	0.15	1.5×10^{-3} [0.75]	100 nCi/g
U-234 (247,000 years)	8.5×10^{-3}	2.2×10^{-3}	2.2×10^{-5} [0.01]	
Pu-239 (24,000 years)	110	28	0.28 [140]	100 nCi/g
U-235 (700 million years)	3.6	0.92	9.2×10^{-3} [4.6]	
Am-241[b] (430 years)	0.16*	0.069	6.9×10^{-4} [0.34]	100 nCi/g
Np-237 (2.1 million years)	6.8×10^{-3}	1.7×10^{-3}	1.7×10^{-5} [0.0085]	100 nCi/g
Pu-240 (6,600 years)	1.6	0.41	4.1×10^{-3} [2.0]	100 nCi/g
U-236 (23 million years)	0.28	0.072	7.2×10^{-4} [0.36]	
U-238 (4.5 billion years)	2.3	0.59	5.9×10^{-3} [3.0]	

Note: 1 Ci = 37,000 MBq

a – Concentrations in nCi/g based on an assumed density of 2,000 kg per cubic meter of packaged waste (void space taken up by a solid such as cement).

b - Inventory of Am-241 is based on complete decay of Pu-241 (14.4 year half-life)

TABLE 4-2 Fractional Release Rates (FRRs) of Radionuclides from Mo-99 Production

Radionuclides (half-life)	Release Rate at 500 years	Release Rate at 5,000 years	Release Rate at 50,000 years
Tc-99 (210,000 years)	1×10^{-4}	1×10^{-4}	1×10^{-4}
I-129 (16 M years)	4×10^{-4}	4×10^{-4}	4×10^{-4}
Sr-90 (28.5 years)	5×10^{-5}	>1	>1
Cs-137 (30 years)	6×10^{-5}	>1	>1
Pu-238 & U-234 (88 years & 247,000 years)	2×10^{-4}	2×10^{-3} (U-234)	2×10^{-3} (U-234)
Pu-239 & U-235 (24,400 years & 700 M years)	3×10^{-8}	3×10^{-8}	1×10^{-7}
Am-241 & Np-237 (433 years & 2.1 M years)	5×10^{-5}	1×10^{-3} (Np-237)	1×10^{-3} (Np-237)
Pu-240 & U-236 (6,600 years & 23 M years)	2×10^{-6}	3×10^{-6}	5×10^{-5} (U-236)
U-238 (4.5 B years)	7×10^{-6}	7×10^{-6}	7×10^{-6}

4.4 Thermal Output from Mo-99 Production Waste

Heat generation due to radioactive decay from radionuclides in the Mo-99 production waste stream is due principally to the short-lived radionuclides Sr-90 and Cs-137 (see Table 4-3). The thermal output from these two radionuclides after 100 years is quite small (i.e., 0.05 W/m^3). Given that this waste stream is estimated to take approximately 70 years to be generated, the thermal output from the Mo-99 production waste stream is not expected to be a concern.

Table 4-3 Thermal Output for Key Radionuclides in the Mo-99 Production Waste Stream

Radionuclide (half-life) [Initial Activity]	Volume (m^3)	Initial Thermal Output (watts)	Thermal Output at 100 years (watts)	Thermal Output at 500 years (watts)
Sr-90 (29 years) [91,000 Ci]	390	110	10	<<1
Cs-137 (30 years) [95,000 Ci]	390	110	11	<<1

Note: 1 Ci = 37,000 MBq

4.5 Gas Generation from Mo-99 Production Waste

As discussed briefly in Section 2.5, radioactive waste has the potential to generate hydrogen through radiolysis, chemical reactions, thermal degradation, and biological activity. Mo-99 production is not anticipated to generate significant amounts of gas from the waste itself because of the limited amounts of water or organic substances likely to be present and the amount of radiation emitted is rather limited especially considering the waste inventory builds up over a 70 year period that allows some significant decay of radionuclides with short half-lives (i.e., Cs-137 and Sr-90). The limited activity also is not expected to result in a significant generation of gases outside the waste form.

5. Intruder Analysis

To assess the potential impacts to the inadvertent intruder that would be required by 10 CFR §61.42 for disposal of GTCC and transuranic waste in a commercial LLRW disposal facility, the staff performed screening analyses. The screening analyses compared the concentrations of the different types of GTCC and transuranic waste with the waste classification values found in Table 1 and Table 2 of §61.55. The comparison used the volumes and radionuclide inventory values presented in the Sections 2 thru 4 and also included a few additional radionuclides considered to be significant in an intruder analysis (e.g., C-14, Ni-59, and Nb-94).

The NRC used a limited number of intruder scenarios to inform the development of the 10 CFR Part 61 waste classification criteria (NRC, 1981, 1982, 1986). The intruder scenarios were based on expected normal activities. The receptor scenarios assumed both direct and indirect contact with disposed waste, including consumption of contaminated food. The receptor scenarios involved a single, acute exposure as well as long-term, chronic exposures. The NRC used the direct contact receptor scenarios to develop the 10 CFR Part 61 waste classification and segregation criteria (NRC, 1981, 1982) and, later, to update the analysis (NRC, 1986). Appendix G of NUREG-2175 “Guidance for Conducting Technical Analyses for 10 CFR Part 61, Draft Report for Comment” provides a summary of the intruder scenarios and the calculations used to develop the waste classification values (NRC, 2015). The pathways considered in this analysis are shown in Table 5-1.

The pathways and scenarios considered in this analysis are similar but not identical to the pathways and scenarios considered in the development of 10 CFR Part 61. In this analysis, deeper disposal is an option that may be evaluated for disposal of GTCC waste. The same chronic pathways were evaluated following disruption of the site for either a home excavation (shallow) scenario or a well drilling (deep) scenario. After the home is built or a well is installed, the intruder resident is assumed to live at the site and potentially be exposed to radiation as reflected by the pathways provided in Table 5-1.

5.1 Intruder Analysis Approach

To evaluate the safety of an inadvertent intruder the staff used the average concentration of the different types of GTCC waste and then compared the concentrations to the §61.55 Table 1 and Table 2 Class C waste concentrations (Table 5-2 and 5-3 below, respectively).

A GoldSim ® model was created by the staff to evaluate the potential impacts to inadvertent intruders relative to the Part 61 dose limits for the intruder. The variables considered included different potential intruder exposure scenarios, waste types (volumes and concentrations), facility designs and disposal configurations. The analysis process involved the following steps:

1) Analyze the intruder-resident for waste type *i*.
2) Determine if the 10 CFR 61.42 performance objective is exceeded, if "no" then stop. If "yes", then proceed to step 3.
3) Analyze the intruder-driller for waste type *i*.
4) Determine if the10 CFR 61.42 performance objective is exceeded, if "no" then stop. If "yes", then proceed to step 5.
5) Waste type *i* exceeds Part 61 limits.
6) Repeat steps 1) to 5) for all waste types *i*+*n*.

The staff recognizes that other intruder scenarios may be appropriate at a particular site. However, the intruder-resident and intruder-driller are generic scenarios tied to basic human behaviors that are reasonable to evaluate at most disposal sites. These scenarios are expected to provide information to assist the identification and understanding of potential concerns with disposal of GTCC waste.

As described in the 6 steps, the scenario initially considered was the intruder-resident. In this scenario, it is assumed that an intruder excavates a foundation and constructs a home on the cover of the disposal site resulting in the exhumation of cover material and waste. Knowledge that waste has been disposed at the location of the excavation is lost.

At the time of excavation, the waste is assumed be indistinguishable from soil. The conservatism of this assumption is tied to the amount of time that has elapsed between when the waste was disposed and when the excavation occurs. In addition, the disposal environment will play a role. Sites that are more humid would generally result in faster corrosion of the metallic components of the disposal system compared to more arid sites.

The concentration of radionuclides that an intruder may be exposed to is a function of the initial concentration in the waste and the assumed parameters describing the volume of waste exhumed in relation to the non-waste material (e.g., backfill, cover material) that may be exhumed. A common excavation depth for residential construction is approximately 3 meters. If waste is disposed deeper than 3 meters then disturbance from common excavation would not occur. It is possible, for some intruder scenarios and certain waste types, that indirect radiation

exposures, such as external radiation or inhalation of gaseous releases (e.g., radon), are larger than direct radiation exposures (e.g., consumption of contaminated produce). However, based on the types of waste and the radiological composition considered in this analysis, the direct exposure scenarios (intruder-resident for waste buried at depths 3 m or less and intruder-driller for wastes deeper than 3 m) were more limiting.

Figure 5-1 is a screen capture of example output for the intruder for the GoldSim model. The output summarizes for different groups of radionuclides and waste types the ratio of the projected dose to an inadvertent intruder compared to a 5 mSv/yr (500 mrem/yr) standard. A value of 1 or less in the output boxes indicates acceptable results. The main variables analyzed were the required time of isolation and the burial depth. Different facility designs (e.g. trench disposal, borehole, deep geologic repository) can be considered within this framework because they would impact the likelihood of disturbance and the volume of material disrupted if disturbed. For the example provided and from the perspective of intruder safety, all of the short-lived GTCC waste would be suitable to be disposed in a commercial LLW disposal facility. However, GTCC waste containing the long-lived isotopes identified in Table 1 of §61.55 would likely exceed the criteria for activated metals and for GTCC other waste. For the example, GTCC waste containing long-lived alpha emitting radionuclides would be significantly above the criteria for almost all types of GTCC waste for shallow disposal (< 3 m disposal depth) and a 500-year intruder barrier. The example places all GTCC waste in a single disposal facility.

Table 5-1 Exposure Pathways of Intruder Receptor Scenarios

Receptor Scenario	Exposure Pathway							
	Inhalation[#]		Ingestion[†]			Direct/External[‡]		
	Air	Soil	Food (Air)	Food (Soil)	Food (Water)	Air	Soil Surface	Soil Volume
Acute Exposures								
Intruder-Construction	●		●			●		●
Intruder-Drilling	●		●			●		●
Chronic Exposures								
Intruder-Resident	●		●	●		●		●

[#] Inhalation includes pathways originating via breathing contaminated air due to suspension of soil particles caused by human activity (air) and caused by natural suspension and volatilization of surface soil (soil).

[†] Ingestion includes pathways for plant-to-human, plant-to-animal-to-human, and plant-to-animal-to-product-to-human uptake. Food (air) considers food pathways originating via atmospheric deposition on plant surfaces and surrounding soil leading to soil-to-root transfer. Food (soil) considers food pathways originating via soil-to-root transfer from contaminated soil. Food (water) considers food pathways originating via irrigation deposition on plant surfaces and the surrounding soil as well as uptake of radionuclides originating from ingestion of contaminated water (i.e., water-to-human; water-to-animal-to-human; and water-to-animal-to-product-to-human).

[‡] Direct/External includes exposure to gamma rays from standing in homogeneously contaminated air (air), standing on a homogeneously contaminated surface area (surface), and standing on homogeneously contaminated ground (volume).

Table 5-2 Class C Concentrations for Long-lived Isotopes (from Table 1 in 10 CFR Part 61)

Radionuclide	Concentration (Ci/m^3)
C-14	8
C-14 in activated metal	80
Ni-59 in activated metal	220
Nb-94 in activated metal	0.2
Tc-99	3
I-129	0.08
Radionuclide	**Concentration (nCi/g)**
Long-lived alpha-emitting nuclides	100
Pu-241	3,500
Cm-242	20,000

Note: 1 Ci = 37,000 MBq

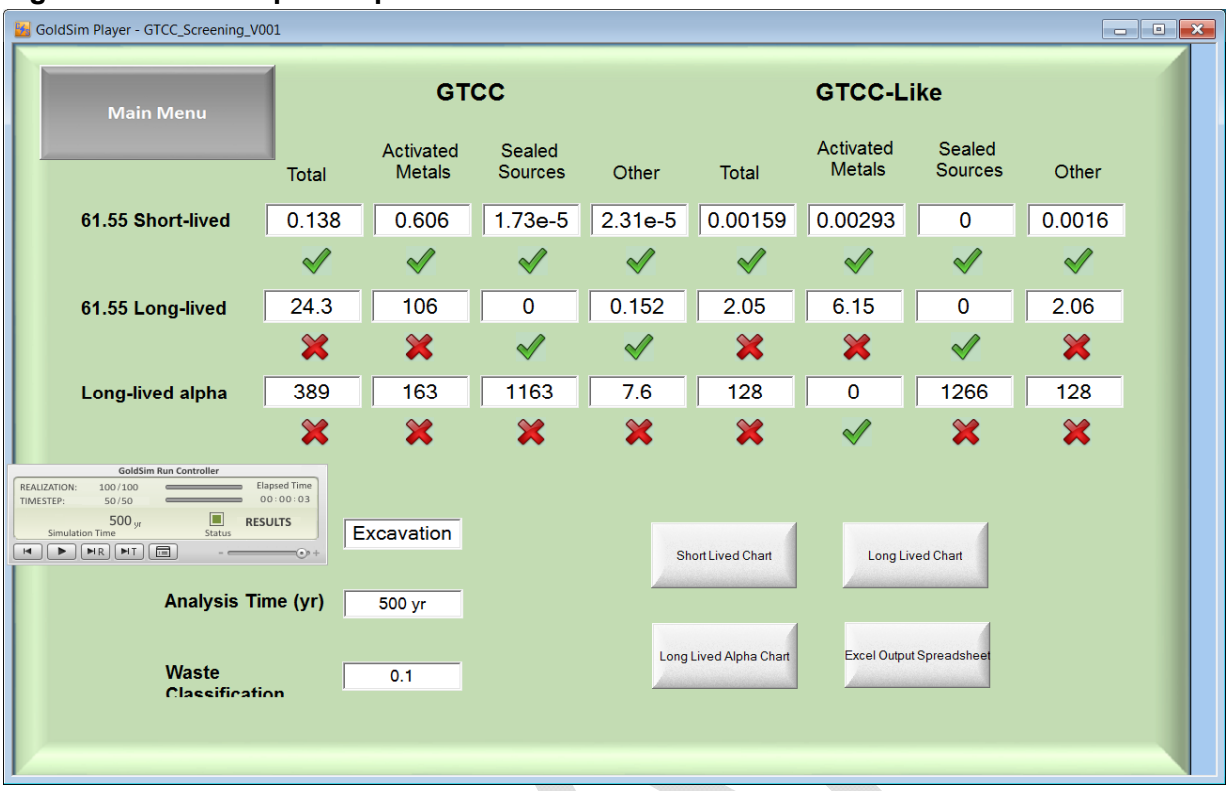
Table 5-3 Class C Concentrations for Short-lived Isotopes (from Table 2 in 10 CFR Part 61)

Radionuclide	Concentration (Ci/m^3)
Total of all radionuclides with less than a 5 year half-life	*
H-3	*
Co-60	*
Ni-63	700
Ni-63 in activated metal	7,000
Sr-90	7,000
Cs-137	4,600

Note: 1 Ci = 37,000 MBq

* There are no limits established for these radionuclides in Class B or C wastes. Practical considerations such as the effects of external radiation and internal heat generation on transportation, handling, and disposal will limit the concentrations for these wastes. These wastes shall be Class B unless the concentrations of other nuclides in Table 2 of 10 CFR Part 61 determine the waste to be Class C independent of these nuclides.

Figure 5-1 Example Output for the GTCC GoldSim Model



5.2 Intruder Results of Shallow Disposal

Table 5-4 provides a summary of the results for the intruder resident scenario for different waste types disposed in a traditional (trench-type) facility design at different times after disposal. The time after disposal can be interpreted as the time an intruder barrier (or the facility design) would prevent an inadvertent intruder from undertaking the activity associated with the scenario. For the excavation scenario, the activated metals and GTCC other (Mo-99 production) waste types present the greatest concern for disposal with respect to the long-lived isotopes identified in Table 1 of § 61.55 (Table 5-2 in this report). For shallow disposal, all waste types are not suitable for disposal with respect to the long-lived, alpha-emitting radionuclides covering timeframes of 500 years up to 50,000 years. This result is consistent with the framework used to develop Part 61; GTCC waste will cause radiological impacts to the inadvertent intruder when it is disposed shallowly.

Table 5-4 Summary of Results for the Inadvertent Intruder Resident Scenario (either below or above the limit)

Time (years)	Radionuclide Group	Total	Activated Metals	Sealed Sources	Other Waste
500	61.55 short-lived	Below	Below	Below	Below
	61.55 long-lived	Above	Above	Below	Above
	Long-lived alpha	Above	Below	Above	Above
5,000	61.55 short-lived	Below	Below	Below	Below
	61.55 long-lived	Above	Above	Below	Above
	Long-lived alpha	Above	Below	Above	Above
50,000	61.55 short-lived	Below	Below	Below	Below
	61.55 long-lived	Above	Above	Below	Above
	Long-lived alpha	Above	Below	Above	Above

5.3 Intruder Results for Deeper Disposal

The NRC staff also analyzed the intruder-driller for facility designs associated with deeper disposal. Table 5-5 summarizes the results for the intruder-driller evaluation. For the intruder-driller scenario and a 1,000-year or greater barrier effectiveness period the sealed sources was the only type of GTCC waste that presented concern for the intruder. The sealed source results are sensitive to the barrier effectiveness time because of the long-lived alpha emitting isotopes that drive the results (Pu-238, Pu-239, Am-241). The results for individual waste types are based on the particular inventory for that waste type and its associated volume.

The results are sensitive to the particular assumptions and parameter values assigned. However, the parameters assigned were consistent with the values used in the original analysis to develop 10 CFR Part 61. Site-specific analysis could be used to evaluate inadvertent intruder safety at a particular site to better understand limitations and concerns related to a particular design and site characteristics.

Table 5-5 Summary of Results for the Inadvertent Intruder Drilling Scenario (either below or above the limit)

Time (years)	Radionuclide Group	Total	Activated Metals	Sealed Sources	Other Waste
500	61.55 short-lived	Below	Below	Below	Below
	61.55 long-lived	Below	Below	Below	Below
	Long-lived alpha	Below	Below	Above	Below
5,000	61.55 short-lived	Below	Below	Below	Below
	61.55 long-lived	Below	Below	Below	Below
	Long-lived alpha	Below	Below	Above	Below
50,000	61.55 short-lived	Below	Below	Below	Below
	61.55 long-lived	Below	Below	Below	Below
	Long-lived alpha	Below	Below	Above	Below

6. Summary

Although all GTCC waste can be described as LLRW with concentrations of radionuclides that exceed the limits established by the Commission for Class C LLRW, there are a variety of waste streams that are grouped under GTCC waste. The DOE, in its FEIS (DOE, 2016), identified three general categories of waste (activated metals, sealed sources, and other waste) in estimating the amount of waste from both current and potential (projected) facilities and activities. The NRC has used information from DOE's FEIS to conduct analyses to better understand potential hazards associated with the disposal of GTCC and transuranic waste. NRC did not evaluate all the waste streams in the DOE FEIS but, rather, selected specific waste streams that had inventories of radionuclides that would provide a spectrum of potential hazards associated with disposal of GTCC and transuranic waste. The NRC staff evaluations do not represent a safety determination for a particular waste stream nor are they an endorsement of any particular disposal method. The evaluations were performed to improve staff understanding of potential concerns with disposal of GTCC and transuranic waste and to provide input for and enhance NRC staff and stakeholder interactions regarding disposal of GTCC and transuranic waste.

The NRC staff evaluations consider each of the three waste stream categories included in DOE's FEIS. Table 6-1 provides the characteristics of the specific waste streams in terms of the activities for the key radionuclides and the thermal output that was assumed to be initially present and what remained after 100 years of radioactive decay. Table 6-1 shows the large variability between specific waste streams, for example: Mo-99 production waste stream has a very large amount of fissile material (U-235); activated metals from commercial reactors and small sealed sources have the highest thermal load, and large sealed sources have a large inventory of Cs-137.

The NRC staff analyses considered (i) exposure to an off-site individual, (ii) inadvertent intruder, (iii) fissile material, (iv) thermal output, and (v) gas generation in evaluating the characteristics of the inventories for the specific waste streams. Table 6-2 provides a summary of the hazards and the associated radionuclides identified for each of the waste streams considered in the NRC staff analyses. As stated at the beginning of this report the NRC staff evaluations do not represent a safety determination for a particular waste stream nor are they an endorsement of any particular disposal method; however, a comparison of the waste streams indicate that Pu-239 is often a radionuclide of concern common to all three categories.

Table 6-1 Activities and Thermal Output Initially Present and After 100 years for GTCC Waste Streams Considered in the NRC Staff Analyses[1]

Radionuclide and Half-life	Activated Metals (Commercial Reactors)		Sealed Sources		Other Waste (Mo-99 Production)	
	Initial	100 yrs	Initial	100 yrs	Initial	100 yrs
Volume:	450 m^3		450 m^3		390 m^3	
	Inventory (Ci)		Inventory (Ci)		Inventory (Ci)	
Ni-63 (96 years)	9.0×10^6	4.5×10^6				
Sr-90 (28.5 years)	2.2	0.22			9.1×10^4	8.3×10^3
Tc-99 (210,000 years)	0.1	0.1			11	11
I-129 (16 M years)	8.7×10^{-4}	8.7×10^{-4}			0.02	0.02
Cs-137 (30 years)	39	3.8	7.7×10^5	7.6×10^4	9.5×10^4	9.4×10^3
Pu-238 (88 years)	0.77	0.35	3×10^4	1.4×10^4	0.6	0.27
Pu-239 (24,000 years)	0.66	0.66	2.1×10^3	2.1×10^3	110	110
Pu-240 (6,600 years)					1.6	1.6
Am-241 (430 years)	1.2	1.0	3.8×10^4	3.2×10^4	0.16*	0.14
U-235 (700 M years)					3.6	3.6
U-238 (4.5 B years)					2.3	2.3
	Thermal Output (watts)		Thermal Output (watts)		Thermal Output (watts)	
	9,300	4,500	2,300 (small sources) 860 (large sources)	1,600 (small sources) 85 (large sources)	220	24

Note: 1 Ci = 37,000 MBq

[1] where no table entry exists the radionuclide was not present or of minor significance in that waste stream

* inventory of Am-241 includes complete decay of 4.9 Ci of Pu-241 (14.4 year half-life)

Table 6-2 Radionuclides of Interest Associated with Disposal of GTCC Waste

Hazard	Activated Metals (Commercial Reactors)		Sealed Sources		Other Waste (Mo-99 Production)	
	500 yrs	5,000 yrs	500 yrs	5,000 yrs	500 yrs	5,000 yrs
Off-site Dose	Pu-239	Pu-239	Am-241, Pu-239 Cs-137	Pu-239, Am-241	Pu-239	Pu-239
Thermal Output	Ni-63	None	Am-241	None	None	None
Fissile Material	None	None	Pu-239	Pu-239	U-235	U-235
Gas Generation	Ni-63	None	Am-241	None	None	None
Intruder (shallow)	C-14, Ni-59, Nb-94, Ni-63	C-14, Ni-59, Nb-94, Ni-63	Am-241	Pu-239	Pu-238, Pu-239, Pu-240, Am-241	Pu-238, Pu-240
Intruder (deep)	None	None	Am-241	Pu-239	None	None

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