TECHNICAL ANALYSIS OF THE HAZARDS OF DISPOSAL OF GREATER-THAN-CLASS C (GTCC) AND TRANSURANIC WASTE

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1. Introduction

The purpose of this document is to present U.S. Nuclear Regulatory Commission (NRC) staff and contractor analyses of the hazards associated with disposal of GTCC waste in a land disposal facility. These analyses support development of a regulatory basis for the disposal of Greater-Than-Class-C[1] and transuranic waste. As a result of GTCC waste disposal, members of the public may be impacted during operations (i.e. offsite) and after operations cease (i.e. onsite – an inadvertent intruder, and offsite). The U.S. Department of Energy (DOE), in its Final Environmental Impact Statement for the Disposal of Greater-Than-Class C (GTCC) Low-Level Radioactive Waste and GTCC-Like Waste[2] issued in January 2016, evaluated the hazards associated with disposal of GTCC waste in a land disposal facility for Low-level Radioactive Waste (LLRW) (DOE, 2016). Though there may be important legal distinctions associated with GTCC and GTCC-like waste, for this technical analysis these wastes are not treated differently and are hereafter referred to as GTCC waste. The DOE evaluated potential impacts to offsite members of the public both during operations and after facility closure. However, DOE did not evaluate impacts to an inadvertent intruder.

NRC regulations that specify requirements for commercial LLRW disposal are provided in 10 CFR Part 61 (NRC, 1982a). The regulation was promulgated in 1982, with subsequent minor revisions throughout the years. The main topics of the regulation with respect to technical analysis were:

- Site suitability
- Waste characteristics
- Waste classification
- Performance objectives
- Stability

Some features, events, and processes may not be amenable to reliable modeling and technical analysis given current technology and understanding. The site suitability and waste characteristics requirements are designed to ensure that the site that is selected and the waste that is disposed will likely meet the performance objectives. The waste classification system was designed to ensure that the waste likely to be disposed would not unduly impact the safety of a member of the public who may inadvertently disturb the waste after closure of the disposal facility. Three classes of waste were defined: Class A, B, and C.[3] At the time the regulations were developed, LLRW was envisioned to be waste that decayed to acceptable levels by either 100 years (Class A or B[4]) or 500 years (Class C). The structure of the regulations was developed around this concept.

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[1] GTCC waste is commercially generated waste that exceeds the concentrations limits in 10 CFR 61.55. GTCC-like waste is waste similar in characteristics to commercially generated waste but is waste that is owned or generated by the DOE. Some GTCC waste is also transuranic waste depending on the concentrations of long-lived alpha-emitting radionuclides. Therefore, transuranic waste is also included in the assessment.

[2] Hereafter referred to in this document as the FEIS.

[3] GTCC is not explicitly defined in 10 CFR Part 61, rather it is inferred from the definition of waste provided in 10 CFR 61.2 and the limits on concentrations for Class C waste provided in 10 CFR 61.55.

[4] Although the limits for Class A and B waste were both developed based on the same assumed decay time, the limits for Class B waste also accounted for waste stabilization that is not required for Class A waste.

Limits on allowable concentrations in waste (Table 1 and 2 of 10 CFR 61.55) were developed by calculating the projected doses to inadvertent intruders for prescribed exposure scenarios. Those exposure scenarios were originally referred to as “construction” and “agriculture”. The term “construction” was used to refer to the acute dose to an intruder who unknowingly excavates a home foundation and constructs a home at the disposal site. The term “agriculture” was used to refer to the chronic dose to an intruder who lives onsite and consumes plants, animals, and animal products grown onsite[5].

In this report the staff evaluates potential performance impacts resulting from disposal of GTCC waste. Staff considered potential disposal methods different from disposal methods for Class A, B, and C waste as well as the potential for features, events, and processes (e.g., heat generation) that are typically not applicable to Class A, B, and C waste. First, the characteristics of GTCC waste are discussed including the different types of waste, quantities of waste, radionuclides in the waste, and other pertinent characteristics. Next, the potential hazards that may result from disposal are evaluated, primarily using calculations and computer models. The data used to parameterize the calculations are provided. The NRC and contractor staff independently evaluated potential impacts from GTCC disposal. A comparison and discussion of the different results is provided.

2. Characteristics of GTCC Waste

GTCC waste is more hazardous than Class C waste and some GTCC waste contains transuranic (TRU) radionuclides (e.g., isotopes of plutonium) that contaminate nuclear fuel cycle wastes (NRC, 1987). GTCC wastes can vary considerably in volume, radionuclide content, and the form of the waste (e.g., activated metal, sealed sources, exhumed waste and soil). The form of the waste can impact the rate at which radioactivity is released to the environment as well as the likelihood of disturbance at some time in the future.

DOE’s Final Environmental Impact Statement (FEIS) divided the GTCC waste streams into two groups based on if the waste has been already produced or will be produced in the future from currently operating facilities (Group 1) or if the waste is anticipated to be generated in the future by facilities that are not yet operating (Group 2). Some of the Group 1 GTCC wastes have already been generated and are in storage awaiting disposal. Group 2 consisted of projected wastes from proposed actions or planned facilities not yet in operation. Some or all of the Group 2 waste may never be generated, depending on the outcomes of proposed actions that are independent of the FEIS. In this document, Group 1 are referred to as “Existing GTCC” whereas Group 2 are referred to as “Potential GTCC”.

GTCC waste is LLRW exceeding the limits for Class C prescribed by 10 CFR 61.55 and arising from activities licensed by the NRC. In the FEIS, DOE identified an additional type of waste, termed GTCC-like waste. DOE described GTCC-like waste as radioactive waste that has similar characteristics as GTCC waste but is owned or generated by DOE (DOE, 2016; pages 1-1&2). DOE stated it identified GTCC-like waste as waste for which there may not be a path for disposal at this time. DOE indicated that they were not creating a new DOE classification of radioactive waste. A majority of the GTCC-like waste is associated with decontamination of the West Valley Demonstration Project (WVDP) site located in Western New York. The WVDP site

[5] The exposure scenario referred to as an “agriculture” exposure scenario in the original development of the 10 CFR Part 61 waste classification limits is generally consistent with the characteristics of an exposure scenario commonly referred to as a “resident farmer” scenario today. The term “agriculture” is used in this document for consistency with the original calculations.

was the location of a former commercial spent nuclear fuel reprocessing facility. The analyses in this report evaluated both GTCC and GTCC-like waste.

Figure 2-1 provides an overview of the GTCC waste volumes, including the distribution of waste as a function of the concentrations of transuranic radionuclides. As will be discussed later, the concentration of transuranic radionuclides is an important technical consideration when evaluating the hazards of GTCC waste disposal. Based on DOE's FEIS, there is approximately 5,000 m^3 (180,000 ft^3) of 'existing' GTCC and GTCC-like wastes that is either already generated or will be generated by currently operating facilities; and approximately 7,000 m^3 (250,000 ft^3) of 'potential' GTCC and GTCC-like wastes associated with proposed facilities or activities. DOE's FEIS (DOE, 2016) described three broad categories for GTCC waste: activated metals, sealed sources, and "other waste." The activity levels of these various waste categories range from concentrations well below to concentrations significantly above their respective Class C concentration limits; however, at least one radionuclide in a specific waste stream (e.g., activated metals from commercial reactors, sealed sources from Cs-137 irradiators) will exceed the Class C concentration limit. In addition, though DOE did not explicitly address variability in radionuclide concentrations, based on staff's experience with commercial LLRW it is anticipated that there is likely to be significant variability in radionuclide concentrations within a waste category (i.e. intra-waste variability). Transuranic nuclides can be found in each one of these three categories of GTCC waste.

Table 2-1 provides an overview of waste volumes and provides the information used to create Figure 2-1. The FEIS provided information from a variety of sources in compiling an inventory of GTCC and GTCC-like waste. Although the FEIS discusses GTCC and GTCC-like waste primarily according to the three broad categories of activated metals, sealed sources, and other; the NRC staff used the information in the FEIS and supporting documents to develop the inventory for specific waste streams (e.g., activated metals from commercial waste and activated metals from potential exhumation of the NRC licensed disposal area as part of the WVDP). The NRC staff developed 17 GTCC waste streams with a total volume of approximately 11,285 m^3 (Table 2-1). This value differs from the total in the FEIS (volume of 12,000 m^3) because DOE elected to round-off certain values prior to summing the totals, then rounded the numbers again to arrive at the value of 12,000 m^3.

The NRC staff used the information in the FEIS to separate the GTCC waste into 17 GTCC waste streams as a means of identifying the hazards associated with each waste stream and to facilitate comparisons by stakeholders (Table A-1 provides a crosswalk between the 17 waste streams and the three categories of waste described in the FEIS). Table 2-2 provides the volume, average transuranic radionuclide concentration, waste stream identification (ID), and identifies whether the waste stream would be contact-handled (CH) or remote-handled (RH). The waste stream ID is the terminology used by the NRC to reference the various waste streams in the text and figures that follow. There are multiple ways the information on the different GTCC waste streams could be presented (e.g., existing vs. potential, GTCC vs. GTCC-like, main waste type by generating process). The staff elected to summarize the waste characteristics by how the waste was generated to allow stakeholders to better relate the conclusions of the report to different waste types.

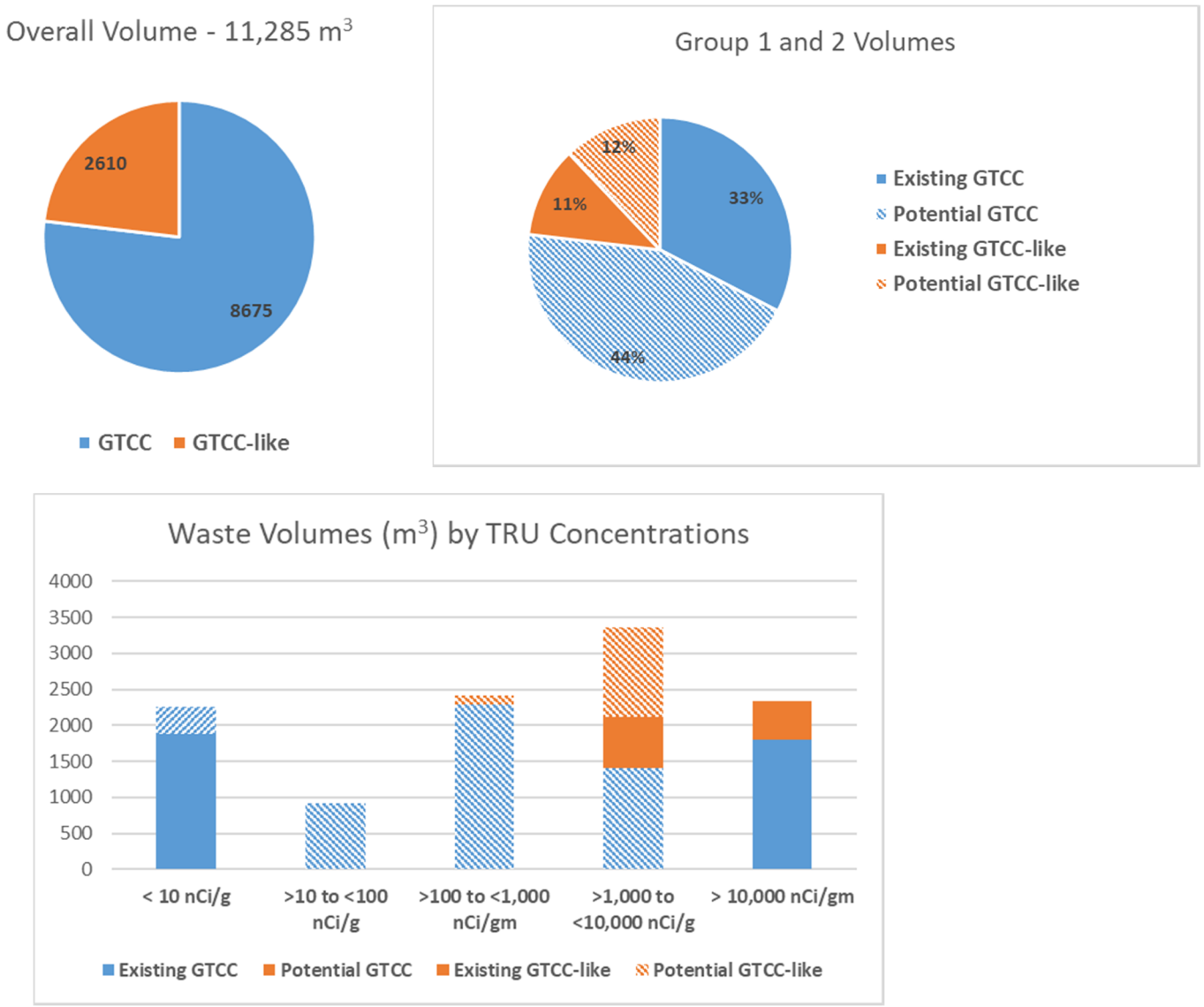


Figure 2-1 Summary of GTCC Waste Volumes

Table 2-1 Estimated Volumes for GTCC Waste Streams

Waste Stream Sources	Estimated Waste Volumes (m^3)			
	GTCC			GTCC-Like
	Activated metals	Sealed Sources	Other Wastes	Other Wastes
Existing Facilities and Activities				
Commercial Reactors	880			
WVDP Decontamination				1,250
Cs-137 irradiators		1,000		
Small irradiators		1,800		
Potential Facilities and Activities				
Commercial Reactors	370			
Exhumation West Valley SDA (State-Licensed Disposal Area)	525		1,600	
WWDP Exhumation NDA (NRC-Licensed Disposal Area)	210		1,900	
WVDP Decommissioning				980
Mo-99, and Pu-238 Production			390	380

Table 2-2 Characteristics of GTCC Waste Streams

Waste Stream Description	GTCC -like?	Waste Stream ID	Volume (m^3)	TRU Concentrations[1] [half-lives > 5 years] (nCi/g)
Small Sealed Sources (CH)		NeutronIRCH	1,800	85,900 (Am-241:54%, Pu-238: 43%)
WVDP Decontamination – Other (RH)	yes	LikeWVDeconRH	540	13,300 (Am-241: 41%)
West Valley Exhumation – SDA-SNAP (CH)		WVSNAPOCH	1,200	8,400 (Pu-238: 100%)
WVDP Decommissioning – Other (CH)	yes	LikeWVDecomOCH	220	6,700 (Am-241: 52%)
WVDP Decontamination – Other (CH)	yes	LikeWVDeconCH	710	5,700 (Am-241: 60%)
WVDP Decommissioning – Other (RH)	yes	LikeWVDecomORH	760	3,500 (Am-241: 53%)
West Valley Exhumation – NDA Activated Metals (RH)		WVNDAAMRH	210	3,200 (Am-241: 57%)
Pu-238 Production – Other (RH)	yes	LikePu238ORH	260	1,900 (Pu-238: 99%)
West Valley Exhumation – NDA Other (RH)		WVNDAORH	1,900	520 (Am-241: 56%)
Mo-99 Production – Other - MURR (RH)		Mo99MurrRH	35	300 (Pu-239: 100%)
Pu-238 Production – Other (CH)	yes	LikePu238OCH	120	160 (Pu-239: 37%, Am-241: 32%)
Mo-99 Production – Other - MIPS (RH)		Mo99MipsRH	355	150 (Pu-239: 97%)
West Valley Exhumation – SDA Other (RH)		WVSDAOCH	400	310 (Pu-238: 70%)
West Valley Exhumation – SDA Activated Metals (RH)		WVSDAAMRH	525	24 (Pu-238: 45%)
Reactors – Activated Metals (RH) [existing waste stream]		ReactorAMRH	880	3
Reactors – Activated Metals (RH) [potential waste stream]		ReactorAMRH1	370	3
Large Sealed Sources (CH)		Cs137SSCH	1,000	0

[1] Concentrations of TRU radionuclides are based on (DOE, 2016) except for reactors activated metals. The source term for those values are derived in Appendix A.

2.1 Activated Metals

Activated metal wastes have been produced at commercial reactors and will continue to be produced by operating reactors. In addition, activated metal waste is currently buried at the NRC-licensed Disposal Area (NDA) and the State-licensed Disposal Area (SDA) in western New York at the West Valley site. The activated metal waste was generated when a commercial spent nuclear fuel reprocessing facility was in operation from 1966 to 1972 at West Valley (the only commercial nuclear fuel reprocessing plant to operate in the United States).

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 (see Appendix A for specific inventories). Radionuclides in the activated metals are comprised of both short half-lived radionuclides that decay substantially over a 100-year period and other radionuclides with much longer half-lives that will be present for thousands of years and longer. Most of the initial activated metal inventory comes from radionuclides with a relatively short half-life.

DOE has estimated that approximately 880 m^3 of activated metal waste would be generated by the current fleet of commercial reactors (primarily during decommissioning) and another 380 m^3 of activated metal waste would be generated by 33 new proposed commercial reactors (DOE 2016; page 1-15). Currently, a small portion (i.e., 10%) of the activated metal waste is in storage with the remainder to be generated in the future from the decommissioning of the commercial reactors (DOE 2016; page 1-15). The short-lived radionuclides are the key contributors to the thermal output of the activated metal waste stream from commercial reactors. Activated metals from commercial reactors can contain a significant amount of heat generating radionuclides (e.g., Ni-63). Short-lived, gamma-emitting radionuclides (e.g., Co-60) in activated metal waste will need a significant amount of shielding to reduce the levels of radiation to acceptable levels, or, as described in the FEIS the canisters would be handled remotely to protect workers from unacceptable exposure. For some isotopes, such as Co-60, there is no concentration limit established for Class C waste because over the 500-year timeframe envisioned for intruder protection there would be significant decreases in concentrations resulting from radioactive decay. However, it was noted in 10 CFR Part 61 that practical considerations such as the effects of external radiation and internal heat generation on transportation, handling, and disposal will limit the concentrations of these wastes.

Table 2-3 provides information on some of the key radionuclides contained in activated metals wastes and for surface contamination on activated metal waste. Although rare, activated metals from commercial reactors can be impacted by the occurrence of damaged fuel, which has the potential to contaminate metal surfaces with fission products and transuranic radionuclides. Unlike the activated radionuclides that occur throughout the metals, fission products and transuranic radionuclides exist only as surface contamination (i.e., radioactive contamination deposited on corrosion films and surfaces of the various plant systems). This surface contamination may contain long-lived radionuclides that are typically included in safety assessments (e.g., Tc-99 with a half-life of 210,000 years) and TRU radionuclides (e.g., Pu-239 with a half-life of 24,000 years). Generally, surface contamination is expected to be low (see Appendix A) such that the radionuclides associated with surface contamination would be well below the Class C limits in 10 CFR 61.55. As discussed in Appendix A, DOE assigned surface contamination values that were unreasonably large which has the impact of misrepresenting the

Table 2-3 Average Contents of a Canister[a] Containing Activated Metals from Commercial Reactors

Radionuclide and Half-life	Canister Inventory (Ci)	Dose Rate at 1 ft[c] (R/hr)	Concentration[b]	Part 61 Tables 1 and 2 Class C Limits
C-14 (5,700 years)	9.7	0	26 Ci/m^3	80 Ci/m^3
Fe-55 (2.7 years)	17,000	0	46,000 Ci/m^3	No limit
Ni-59 (75,000 years)	55	0	149 Ci/m^3	220 Ci/m^3
Ni-63 (96 years)	7,600	0	20,500 Ci/m^3	7,000 Ci/m^3
Co-60 (5.3 years)	21,000	85,000	57,000 Ci/m^3	No limit
I-129 (16 million years)	0.0008	0	0.002 Ci/m^3	0.08 Ci/m^3
Pu-238 (88 years)	6.3×10^{-4}	0	0.85 nCi/g	100 nCi/g
Pu-239 (24,000 years)	5.6×10^{-4}	0	0.74 nCi/g	100 nCi/g
Am-241 (430 years)	1×10^{-3}	0	1.4 nCi/g	100 nCi/g

Note: 1 Ci = 37,000 MBq

a – Based on an internal volume of 0.37 m^3

b – Concentrations in nCi/g based on an assumed density of 1,800 kg per cubic meter of packaged waste (void space taken up by a solid such as cement).

c – dose rate calculated for gamma radiation after 10 years of decay no attenuation due to canister or shielding

concentrations of transuranic radionuclides. Thus, NRC staff used available information from commercial reactors to assign more realistic values for surface contamination on GTCC activated metal waste.

Radioactive wastes were disposed at two separate areas at the West Valley site: (1) the NDA and (2) the SDA. A future decision will be made whether to remove or to close in-place this buried waste as part of future decommissioning activities (Argonne 2010; page 45). Activated metal waste buried at the NDA is comprised primarily of hardware (210 m^3) and much smaller amounts of failed spent fuel (0.3 m^3) (DOE 2016, Table B-1)). Activated metal waste buried at the SDA is from metal components associated with special purpose reactors such as small research and naval reactors. The estimated volume of the activated metal waste at the SDA is about 525 m^3 (DOE 2016, Table B-1). Table 2-4 provides information on some of the key

Table 2-4 Average Contents of a Canister[a] Containing Activated Metals from the NDA and SDA Disposal Areas at West Valley

Radionuclide and Half-life	Canister Inventory (Ci)	Dose Rate at 1 foot[c] (R/hr)	Concentration[b]	Part 61 Tables 1 and 2 Class C Limits
Co-60 (5.3 years)	1,000	3,800	2,700 Ci/m^3	700 Ci/m^3 (Class A limit)
Pu-238 (88 years)	0.065	0	100 nCi/g	100 nCi/g
Pu-239 (24,000 years)	0.1	0	160 nCi/g	100 nCi/g
Pu-240 (6,500 years)	0.08	0	120 nCi/g	100 nCi/g
Am-241 (430 years)	0.35	0	520 nCi/g	100 nCi/g

Note: 1 Ci = 37,000 MBq

a – Concentrations in nCi/g based on an assumed density of 1,800 kg per cubic meter of packaged waste (void space taken up by a solid such as cement).

b – dose rate calculated for gamma radiation after 10 years of decay, no attenuation due to canister or shielding

radionuclides contained in these wastes using an average concentration of the combined NDA and SDA inventories. DOE assumed that all activated metal GTCC waste would be remote-handled and would be packaged in unshielded, stainless steel canisters with an internal volume of 0.37 m^3 each (DOE 2016; pages B-18 and 26).

2.2 Sealed Sources

The GTCC sealed sources can be divided into two groups: small sealed sources (i.e., commercial plutonium, americium and curium sources generally referred to as neutron irradiators in this report) and large sealed sources (i.e., industrial Cs-137 sources). Small sealed sources typically consist of concentrated radioactive material encapsulated in relatively small containers made of titanium, stainless steel, or other metals. These sources are commonly used to sterilize medical products, detect flaws and failures in pipelines and metal welds, determine the moisture content in soil and other materials, and diagnose and treat illnesses such as cancer. Sealed sources can encompass several physical forms, including ceramic oxides, salts, or metals. DOE estimated that there are 1,435 individual sealed sources using Cs-137, each with an assumed volume of 0.71 m^3 (25 ft^3). Because of their size, these irradiators cannot be packaged in 0.2-m^3 (55-gallon) drums and are assumed to be disposed of individually in their original shielded devices. In these sealed sources, the Cs-137 source is contained within a very robust shielded device, which would be expected to help the wasteform retain its integrity and reduce the worker dose levels during processing and disposal (FEIS, page B-21). Cesium chloride salt was generally used in older Cs-137 sources. Newer sources typically have the radionuclide bonded in a ceramic. Of these two forms, cesium chloride salt is much more water-soluble.

Table 2-5 provides information for key radionuclides contained in sealed sources. The small sealed sources are associated with neutron irradiators and contain transuranic radionuclides in concentrations that are significantly above the 10 CFR 61.55 Class C limits (e.g., Pu-239 was estimated to be present at 2,300 nCi/g – see Appendix A). Pu-239 is a fissile radionuclide with a total mass of 135 kilograms for the sealed sources associated with neutron irradiators, which results in approximately 15 grams of Pu-239 in a 0.2-m^3 (55-gallon) drum (based on a total volume of 1,800 m^3 for sealed sources associated with neutron irradiators). Additionally, the half-lives for some of transuranic radionuclides are long (e.g., Am-241 half-life of 430 years, Pu-239 half-life of 24,000 years).

2.3 Other Waste

DOE's FEIS included a category entitled 'other' waste that 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 (i.e., Mo-99 production). Other waste from the Mo-99 production stream is a different type of waste not considered in the other two categories (i.e., activated metals and sealed sources). DOE estimated the total volume generated over a 71-year period for the other waste from Mo-99 production would be 390 m^3 (DOE, 2016; page B-5).

A variety of radionuclides are present in the other waste from Mo-99 production, with the isotopes of various actinides (uranium, neptunium, plutonium, americium, and curium) being of most concern for long-term management. Other waste from Mo-99 production contains long-lived transuranic radionuclides (e.g., Pu-239) as well as isotopes of uranium. DOE estimated that 71 years of production would result in approximately 1,700 kg of U-235. An individual 0.2-m^3 (55-gallon) container would hold nearly 1 kg of U-235. This amount of U-235 in a single container is well above the 10 CFR 150.11 critical mass limit of 350 grams.

Table 2-6 and 2-7 provide information averaged over all the other waste streams for key radionuclides contained in CH and RH waste containers, respectively.

Table 2-5 Representative Contents of a Waste Container for Sealed Sources[a] (based on an internal volume of 0.7 m3 for Cs-137 and 0.2 m3 for Pu and Am)

Radionuclide and Half-life	Canister Inventory (Ci) [g]	Dose Rate at 1 foot[c] (R/hr)	Concentration[b]	Part 61 Tables 1 and 2 Class C Limits
Cs-137 (30 years)	1,200	1,100	1,700 Ci/m^3	4,600 Ci/m^3
Pu-238 (88 years)	13	0	37,000 nCi/g	100 nCi/g
Pu-239 (24,000 years)	0.9	0	2,600 nCi/g	100 nCi/g
Am-241 (430 years)	17	0	46,000 nCi/g	100 nCi/g

Note: 1 Ci = 37,000 MBq

a – Based on an internal volume of 0.7 m^3 for sealed sources associated with Cs-137 and 0.2 m^3 for sealed sources associated with neutron irradiators.

b – Concentrations in nCi/g based on an assumed density of 1,800 kg per cubic meter of packaged waste (void space taken up by a solid such as cement for sealed sources associated with neutron irradiators).

c – dose rate calculated for gamma radiation after 10 years of decay, no attenuation due to canister or shielding

Table 2-6 Representative Contents of a CH Waste Container for Other Waste Sources (based on an internal volume of 0.2 m^3)

Radionuclide and Half-life	Canister Inventory (Ci)	Dose Rate at 1 foot[c] (R/hr)	Concentration[b]	Part 61 Tables 1 and 2 Class C Limits
Pu-238 (88 years)	1.7	0	4,800 nCi/g	100 nCi/g
Pu-239 (24,000 years)	0.1	0	280 nCi/g	100 nCi/g
Pu-240 (6,500 years)	0.4	0	220 nCi/g	100 nCi/g
Am-241 (430 years)	2.2	0	1,200 nCi/g	100 nCi/g

Note: 1 Ci = 37,000 MBq

b – Concentrations in nCi/g based on an assumed density of 1,800 kg per cubic meter of packaged waste

c – dose rate calculated for gamma radiation after 10 years of decay, no attenuation due to canister or shielding

Table 2-7 Representative Contents of a RH Waste Container for Other Waste Sources (based on an internal volume of 0.37 m^3)

Radionuclide and Half-life	Canister Inventory (Ci)	Dose Rate at 1 foot[c] (R/hr)	Concentration[b]	Part 61 Tables 1 and 2 Class C Limits
Cs-137 (30 years)	50	45	130 Ci/m^3	4,600 Ci/m^3
Pu-238 (88 years)	0.9	0	520 nCi/g	100 nCi/g
Pu-239 (24,000 years)	1.0	0	580 nCi/g	100 nCi/g
Pu-240 (6,500 years)	0.7	0	370 nCi/g	100 nCi/g
Am-241 (430 years)	2.3	0	1,300 nCi/g	100 nCi/g

Note: 1 Ci = 37,000 MBq

b – Concentrations in nCi/g based on an assumed density of 1,800 kg per cubic meter of packaged waste

c – dose rate calculated for gamma radiation after 10 years of decay, no attenuation due to canister or shielding

3. Technical Analysis of GTCC Hazards

In this technical analysis, the staff evaluates three types of potential impacts: 1) impacts to an inadvertent intruder (onsite individual) after the facility has closed, 2) impacts to an offsite member of the public (offsite individual) after the facility has closed, and 3) impacts to an offsite member of the public during operations. As previously discussed, the staff selected 17 GTCC waste streams to evaluate. The inventory assigned to the waste streams is provided in Table 3-1a and b. GTCC waste can contain a large variety of different radionuclides. The radionuclides included in Table 3-1a and b represent a subset of the radionuclides contained in the waste. This subset was selected by considering the radionuclides considered for the technical analysis to develop 10 CFR Part 61 (NRC, 1981a) and the results provided by DOE in the FEIS (DOE, 2016). In addition, staff experience with performance and intruder assessments was used to evaluate the list of radionuclides and select the specific radionuclides to analyze.

3.1 Inadvertent Intruder

Inadvertent intruder protection is one of the main safety components of 10 CFR Part 61. The regulations at § 61.42 require protection of an inadvertent intruder for disposal of low-level waste in the near surface.[6] NRC and contractor staff performed analyses to evaluate potential doses to an inadvertent intruder that may result from the disposal of GTCC waste.

[6] The LLRW regulations have been proposed to be revised. Currently, the regulation does not explicitly require an intruder dose assessment to demonstrate that protection can be achieved with reasonable assurance.

Figure 3-1 outlines the components of an inadvertent intruder dose assessment. The waste and its emplacement configuration must be known or assumed. Next, potential disturbance mechanisms are assumed. This is followed by the assignment of pathways of exposure to the disturbed waste (radioactivity). Finally, resulting doses are estimated using models that have been parameterized for the scenario. Commercial software products are capable of accurately accounting for decay and ingrowth. The most challenging parts of the intruder assessment are determining the appropriate exposure scenarios to analyze and assigning some of the exposure parameter values.

The calculations to develop the waste classification tables (i.e. Tables 1 and 2 provided in 10 CFR Part 61.55) assumed an intruder excavation scenario (NRC, 1981a). The types of intruder exposure scenarios considered were only possible after active institutional control of the disposal site is no longer active. Licensees are required to have active institutional controls for a period of up to 100 years after operations cease,[7] and it is envisioned that the passive performance of controls will likely continue to be effective at preventing future use of the disposal site. However, it was also envisioned that future site usage could not be ensured. Therefore, impacts to potential inadvertent intruders are evaluated and provide an additional defense in depth for protection of public health and safety. The waste was assumed to be disposed shallowly, consistent with operational practices at the time, such that construction of a home was a credible disturbance mechanism. For the analysis described in this document, the staff evaluated an excavation scenario, consistent with the original analyses, as well as drilling of a well for retrieval of natural resources (e.g. water). The well drilling exposure scenario was not evaluated explicitly when 10 CFR Part 61 was developed because the excavation exposure scenario is much more limiting. However, for waste that may be disposed more deeply, the home excavation scenario is not a credible disturbance mechanism. Adjustments to the analyses must be made to account for the differences in a drilling scenario compared to an excavation scenario.

The disturbance exposure scenarios are classified as two types: acute and chronic. Figure 3-2 is a representation of the inadvertent intruder scenarios. The potential radiological doses to the person(s) doing the disturbing are evaluated in the acute exposure scenarios. The chronic exposure scenarios are used to evaluate the impacts to a long-term receptor (e.g., resident or resident farmer) after the disturbance has taken place and a home and water well have been installed. The exposure times are typically much longer for the chronic scenarios, though the concentrations of radionuclides in the environment are smaller compared to the acute scenarios.

[7] The original Part 61 calculations credited 102 years of decay for Class A and Class B waste to account for 2 years of closure activities after waste emplacement ceased followed by 100 years of institutional control.

Table 3-1a Inventory of Radionuclides by GTCC Waste Streams

Nuclide	ReactorAMRH	Cs137SSCH	NeutronIRCH	LikeWVDeconCH	LikeWVDeconRH	ReactorAMRH	Mo99MurrRH	Mo99MipsRH	WVNDAAMRH
H3	7.30E+00	0.00E+00	0.00E+00	2.26E-04	2.80E-02	7.30E+00	0.00E+00	4.53E-01	4.16E-02
C14	2.61E+01	0.00E+00	0.00E+00	1.83E-02	1.85E-01	2.61E+01	1.60E-06	4.22E-01	2.45E+00
Ni59	1.48E+02	0.00E+00	0.00E+00	1.07E-04	2.96E-01	1.48E+02	0.00E+00	0.00E+00	5.28E+00
Co60	4.99E+04	0.00E+00	0.00E+00	5.07E-06	1.95E+00	4.99E+04	0.00E+00	0.00E+00	1.18E+01
Ni63	2.03E+04	0.00E+00	0.00E+00	3.50E-05	1.73E+01	2.03E+04	0.00E+00	0.00E+00	4.82E-02
Sr90	4.77E-03	0.00E+00	0.00E+00	9.07E-02	6.51E+01	4.77E-03	1.26E+03	1.26E+02	3.52E+01
Tc99	2.27E-04	0.00E+00	0.00E+00	4.51E-04	3.15E-01	2.27E-04	1.20E-01	1.97E-02	2.04E-02
I129	2.16E-03	0.00E+00	0.00E+00	1.37E-07	5.00E-03	2.16E-03	2.60E-04	3.10E-05	4.30E-05
Cs137	8.44E-02	1.70E+03	0.00E+00	8.95E-02	7.06E+01	8.44E-02	1.37E+03	1.27E+02	4.61E+01
U233	0.00E+00	0.00E+00	0.00E+00	1.32E-02	1.46E+00	0.00E+00	0.00E+00	0.00E+00	1.80E-02
U234	0.00E+00	0.00E+00	0.00E+00	6.20E-02	2.96E-03	0.00E+00	0.00E+00	0.00E+00	9.30E-04
U235	0.00E+00	0.00E+00	0.00E+00	2.25E-04	6.48E-04	0.00E+00	0.00E+00	0.00E+00	2.28E-04
U236	0.00E+00	0.00E+00	0.00E+00	7.61E-05	1.46E-03	0.00E+00	0.00E+00	0.00E+00	5.05E-04
U238	0.00E+00	0.00E+00	0.00E+00	1.28E-04	2.04E-02	0.00E+00	0.00E+00	0.00E+00	3.07E-03
Np237	0.00E+00	0.00E+00	0.00E+00	1.55E-03	2.78E-03	0.00E+00	1.40E-06	1.89E-05	3.20E-04
Pu238	1.69E-03	0.00E+00	6.65E+01	1.82E+00	2.76E+00	1.69E-03	4.82E-07	1.68E-03	5.69E-01
Pu239	1.48E-03	0.00E+00	4.66E+00	1.27E+00	5.37E+00	1.48E-03	5.43E-01	2.56E-01	1.16E+00
Pu241	2.71E-02	0.00E+00	0.00E+00	1.88E+01	3.00E+01	2.71E-02	2.64E-04	1.32E-02	1.08E+01
Am241	2.72E-03	0.00E+00	8.35E+01	6.19E+00	9.80E+00	2.72E-03	1.85E-09	6.19E-06	3.27E+00
Cm243	0.00E+00	0.00E+00	0.00E+00	1.04E-04	4.98E-03	0.00E+00	0.00E+00	0.00E+00	6.35E-04
Cm244	0.00E+00	0.00E+00	1.22E-02	2.44E-03	1.96E+00	0.00E+00	0.00E+00	0.00E+00	1.53E-02

Table 3-1b Inventory of Radionuclides by GTCC Waste Streams

Nuclide	WVNDAORH	WVSDAAMRH	WVSDAOCH	WVSNAPOCH	LikeWVDecomOCH	LikeWVDecomORH	LikePu238OCH	LikePu238ORH
H3	1.05E-02	1.32E+00	4.73E-01	0.00E+00	4.73E-04	2.11E-04	0.00E+00	0.00E+00
C14	0.00E+00	0.00E+00	1.10E-02	0.00E+00	2.68E-02	1.18E-02	0.00E+00	0.00E+00
Ni59	1.11E-03	0.00E+00	8.25E-05	0.00E+00	1.50E-04	6.71E-05	0.00E+00	0.00E+00
Co60	2.53E-02	3.30E+03	1.43E-02	0.00E+00	7.98E-07	3.46E-07	0.00E+00	0.00E+00
Ni63	9.47E-02	0.00E+00	9.18E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr90	4.74E+00	1.05E+00	6.83E-03	0.00E+00	2.71E-02	5.75E+01	0.00E+00	0.00E+00
Tc99	3.05E-03	6.97E-03	2.50E-06	0.00E+00	5.91E-04	4.21E-03	0.00E+00	0.00E+00
I129	1.78E-05	2.46E-03	7.25E-06	0.00E+00	0.00E+00	5.00E-06	0.00E+00	0.00E+00
Cs137	7.89E+00	1.46E+01	5.37E-02	0.00E+00	1.47E-02	4.14E+02	9.77E-10	6.76E+01
U233	3.89E-03	6.03E-07	0.00E+00	0.00E+00	1.86E-02	8.42E-03	0.00E+00	0.00E+00
U234	2.01E-04	2.09E-08	2.42E-05	0.00E+00	8.64E-02	3.82E-02	0.00E+00	0.00E+00
U235	5.26E-05	4.59E-05	1.20E-06	0.00E+00	3.64E-05	1.84E-05	0.00E+00	0.00E+00
U236	8.42E-05	1.18E-10	0.00E+00	0.00E+00	1.09E-04	4.74E-05	0.00E+00	0.00E+00
U238	4.21E-04	3.73E-04	2.50E-05	0.00E+00	1.77E-04	9.61E-05	0.00E+00	0.00E+00
Np237	4.86E-05	7.85E-10	8.50E-12	0.00E+00	1.00E-04	7.89E-04	0.00E+00	6.54E-03
Pu238	1.10E-01	1.91E-02	0.00E+00	1.51E+01	2.54E+00	1.33E+00	4.96E-02	3.36E+00
Pu239	1.79E-01	0.00E+00	8.50E-02	1.25E-02	1.76E+00	8.41E-01	1.08E-01	1.88E-03
Pu241	2.05E+00	3.19E+00	2.75E+00	5.16E-01	3.98E+01	1.88E+01	8.74E-01	1.43E-01
Am241	5.26E-01	1.39E-02	3.00E-05	0.00E+00	6.30E+00	3.40E+00	9.15E-02	4.61E-02
Cm243	1.26E-04	1.24E-05	1.81E-08	0.00E+00	1.73E-04	5.01E-03	0.00E+00	0.00E+00
Cm244	2.79E-03	8.78E-03	1.18E-05	0.00E+00	4.37E-03	1.15E-01	0.00E+00	0.00E+00

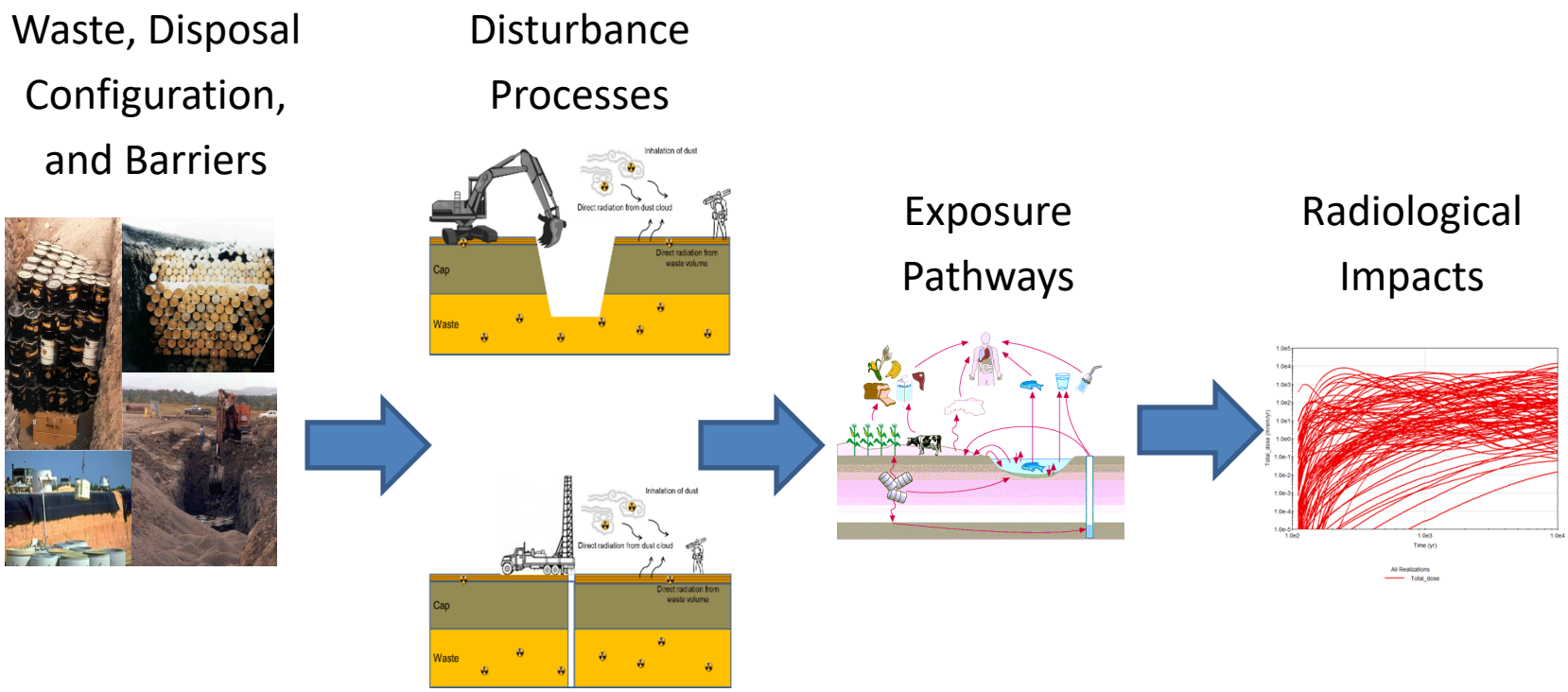


Figure 3-1 Major Components of an Inadvertent Intruder Assessment

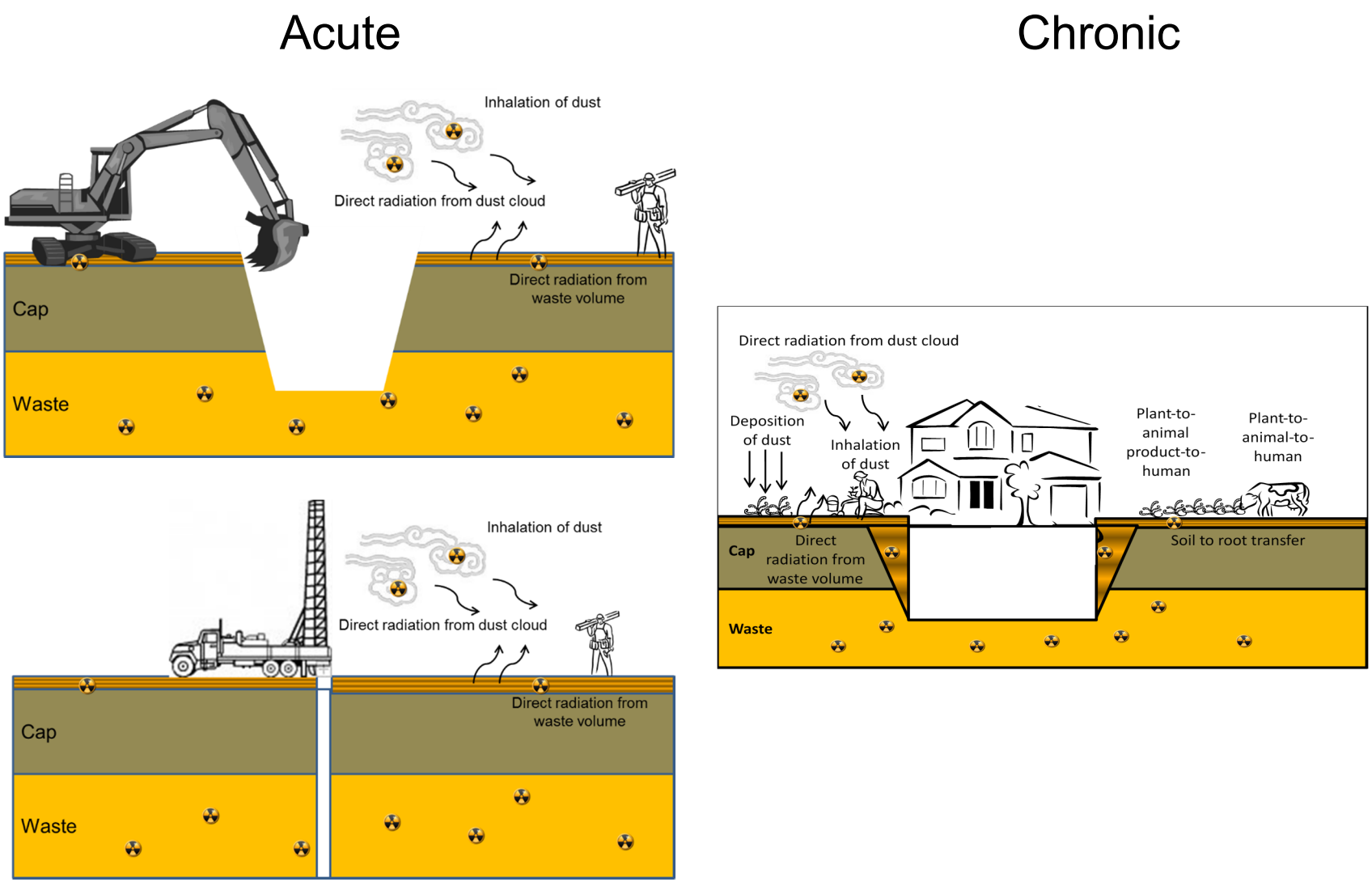


Figure 3-2 Inadvertent Intruder Exposure Scenarios (NOTE: Waste thickness evaluated in calculations thinner than depicted in the figures)

One of the main challenges to the inadvertent intruder assessment is selecting which exposure scenarios should be evaluated. Most disposal facilities are purposely located in areas with low population density and with limited natural resources that may be exploited. Therefore, based solely on present day observations, one may argue that direct disturbance scenarios are not credible. However, this viewpoint may suffer from recency bias, and may not reflect the dynamic nature of societal development over long timeframes. Technology to identify and recover natural resources is continually evolving. Technologies employed today (e.g. laser-guided sonic drilling) were not even imagined 100 years ago. In addition, disposal facilities may be subject to disruptive processes and events (e.g., earthquakes, floods, erosion) that may release or disturb waste. The limitations on acceptable waste concentrations imposed based on the results of an intruder assessment may serve to mitigate the potential impacts from disruptive processes and events that are much more difficult to assess. Cautious but reasonable intruder scenarios play an important role in the management of LLRW. The intruder assessment is not a prediction of what is going to happen in the future, rather it is a regulatory tool to ensure, should disturbance occur, that public health and safety will be protected.

The original 10 CFR Part 61 intruder analysis assumed that inadvertent intrusion occurred following an institutional control period, and that the intruder occupied the disposal facility and engaged in normal activities, such as agriculture or dwelling construction, resulting in direct contact with the waste (NRC, 1981a). Waste is disposed underground to limit contact with man and biota. GTCC waste may have many different forms, ranging from the composition of ordinary municipal trash to components of stainless steel. The form of the waste can be an important consideration when evaluating potential impacts to an inadvertent intruder. Doses to an inadvertent intruder are directly dependent on exposure time. If the waste material, when disturbed, is clearly distinguishable as something different (and dangerous) then exposure times may be less. This type of exposure scenario is termed a discovery scenario.

Figure 3-3 provides an overview of potential pathways that could be considered in an intruder dose assessment. The relevance of the pathways to an assessment can be site-specific. In general (dependent on the specific radionuclides in the waste), only a limited number of pathways contribute significantly to inadvertent intruder doses. For acute intruder scenarios those pathways are generally inhalation of resuspended waste/soil mixture, direct radiation exposure, and inadvertent soil ingestion. Due to the short duration of acute scenarios, ingestion of contaminated food or water is usually not evaluated. For chronic scenarios, the aforementioned pathways are included but the ingestion pathways can be more significant.

Table 3-2 provides the pathways considered in the original 10 CFR Part 61 analyses. Exposure pathways are generally grouped into three major types: inhalation, ingestion, and direct radiation exposure. The inadvertent intruder analysis considered exposure to radionuclides through inhalation of contaminated soil and air, direct radiation, and ingestion of contaminated food and water. It is important for the reader to understand that while a larger number of exposure pathways were initially considered and implemented in the calculations, the waste classification tables are only based on select exposure scenarios and pathways. For example, the intruder analysis included the capability to consider contaminated water, however contaminated water-related pathways were not included in development of the waste classification tables.

The concentrations and fluxes of radionuclides through and within environmental pathways are converted to radiological doses using dose conversion factors based on a specific dose methodology. The dose methodology is how individual dose factors (for example: external, ingestion, or inhalation) are calculated for each radionuclide. The dose limits used to develop

the waste classification tables[8] were based on organ doses using different limits for different organs and International Commission on Radiological Protection (ICRP) 2 dose methodology (NRC, 1982a; ICRP, 1960). Figure 3-4 shows the different ICRP methodologies that could be applied to the intruder dose assessment. To develop the 10 CFR Part 61 waste classification tables, the doses to different organs (i.e., total body, bone, liver, kidney, lung, gastrointestinal tract/lower large intestine, thyroid) were calculated for a particular scenario. The limiting organ, based on comparing the dose to the dose limit provided for that organ, was then identified for each radionuclide. In this manner, the waste classification values in 10 CFR Part 61 reflect the limiting organ and the limiting exposure scenario (i.e. the limiting value of construction or chronic scenarios). The historical approach described here is quite a bit different than what is done in a modern dose assessment. As discussed in the next section, the staff used both the historical approach and modern methods to examine the potential impacts to an inadvertent intruder from the disposal of GTCC waste.

[8] Table 1 and 2 of 10 CFR Part 61.55

Table 3-1 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-Discovery	●		●			●		●
Chronic Exposures								
Intruder-Agriculture	●		●	●		●		●

[#] 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).

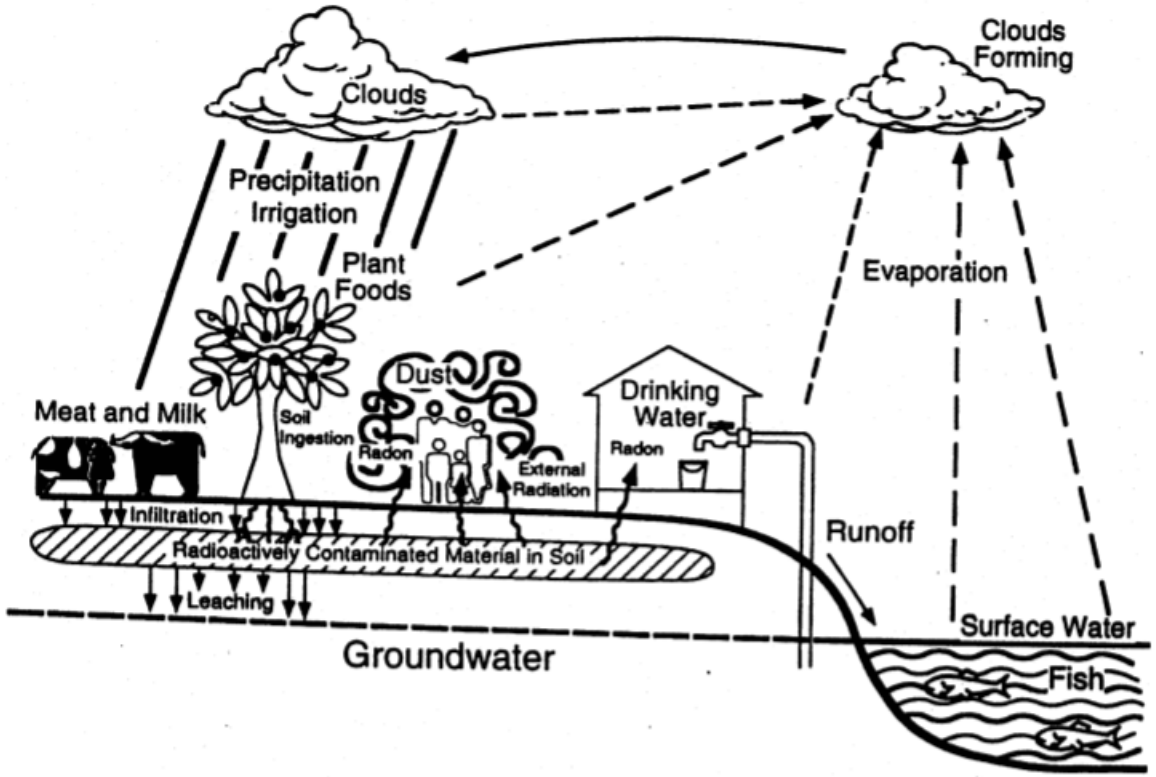


Figure 3-3 Potential Pathways of Exposure (Generic Dose Assessment)

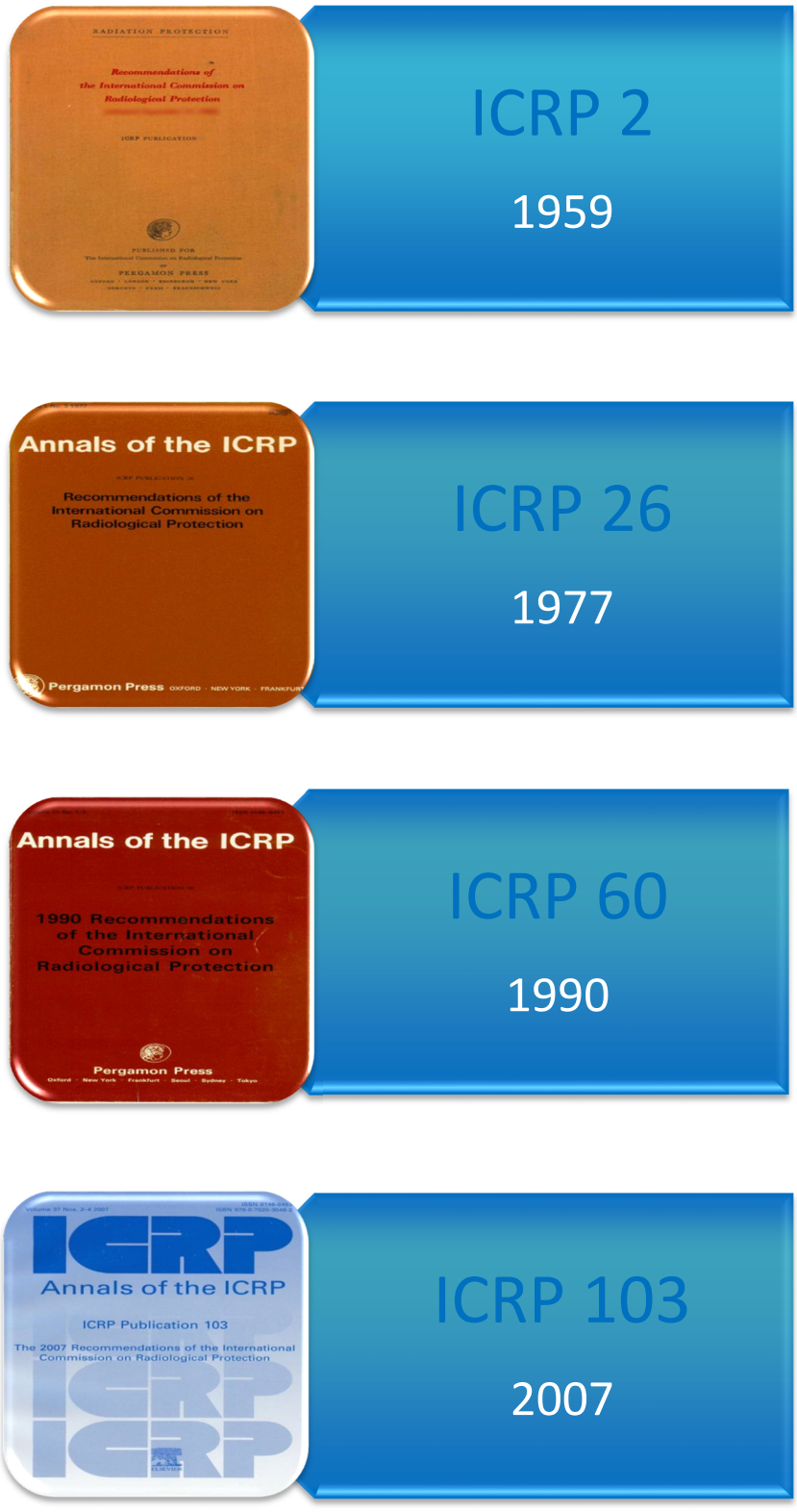


Figure 3-4 ICRP Dose Methodologies that Could Be Used in Intruder Assessments

3.1.1 Models and Data

The NRC (and contractor) staff elected to pursue two different calculation approaches to estimate the potential impacts to an inadvertent intruder from the disposal of GTCC waste. The staff used an approach based on the original impacts analysis performed for development of 10 CFR Part 61 (NRC, 1981a; NRC, 1982b). The contractor staff developed a modern intruder dose assessment using the BDOSE[9] model developed with the commercial software package GoldSim.[10] The primary reason different approaches were used is that there can be differences in the output resulting from the methods used. Staff wanted to understand and explore the full uncertainty space associated with the outcomes to support more robust and risk-informed decision-making. While it is common to examine variability in the output resulting from different waste streams, exposure scenarios and pathways, and input data selected, it is far less common to evaluate the impact of calculation or model uncertainty. From multiple decades of experience performing reviews of models, the staff has an appreciation for the potential importance of model uncertainty. Initially, the two groups proceeded independently. Then as results were being produced, NRC staff and the contractor met to discuss models, input data, and other considerations to understand similarities and differences in the results. Results are provided in Section 4 of this document. This section provides a description of the NRC staff models and data. Although comparisons are made between NRC and contractor model output in this document, most of the discussion of the contractor models and data can be found in (LaPlante, 2019; Wittmeyer, 2019).

The staff approach started with the calculations used to develop the waste classification tables. The basis for using the original calculations is to provide an “apples to apples” comparison of results for GTCC with Class A, B, and C LLRW. Those calculations are documented in various reference documents (NRC, 1981a; NRC, 1981b; NRC, 1982b; NRC, 1983). This report does not attempt to replicate the full description from the reference reports; the key details are summarized to allow the reader to understand the results that follow. The analyses used to support the development of 10 CFR Part 61 in the early 1980s relied on six FORTRAN computer codes, which are documented in the following references:

- NUREG-0782 Appendix H (NRC, 1981a),
- NUREG-0945 Appendix D (NRC, 1982b),
- NUREG/CR-1759, Vol. 3 - Section 6 and Appendix D (NRC, 1981b), and
- NUREG-0959 Appendix 1 (NRC, 1983).

Two of the six codes, called “DOSE” and “INVERSI,” were used to develop the radionuclide concentrations used as the class limits for Class A, B, and C LLRW, which are given in Tables 1 and 2 of 10 CFR Part 61.55. DOSE was run first to generate pathway dose conversion factors (PDCFs) that converted radionuclide concentrations in a single contaminated medium to critical organ doses for eight sets of exposure pathways. The PDCFs are summarized in Table 3-3. Next INVERSI was run using the PDCFs generated by DOSE with assumptions about radionuclide transport and exposure scenarios to project inadvertent intruder doses in 20 hypothetical scenarios. The PDCFs were used or combined in different ways to represent scenarios.

[9] BDOSE – Biosphere Dose Model developed by staff at the Southwest Research Institute (SwRI)
[10] GoldSim is a commercial software package developed by GoldSim Technology Group of Issaquah WA

Table 3-3 Description of Pathway Dose Conversion Factors Generated by the DOSE Code

PDCF	Contaminated Medium	Description
1	Air	Inhalation, direct radiation from submersion in air, and direct radiation from radionuclides deposited on the ground
2	Air	Inhalation, direct radiation from contaminated air, and ingestion of food contaminated by non-equilibrium deposition of airborne particles
3	Air	Inhalation, direct radiation from contaminated air, and ingestion of food contaminated by equilibrium deposition of airborne particles
4	Soil	Ingestion of food contaminated by uptake from contaminated soil (including animal pathways)
5	Soil	Direct radiation from a volumetric soil source
6	Well Water	Ingestion of contaminated well water, ingestion of food contaminated through irrigation, inhalation, direct radiation from an area source, and direct radiation from submersion in contaminated air
7	Surface Water	The exposure pathways are the same as the exposure pathways for PDCF-6 except fish and seafood ingestion are included
8	Air	Inhalation, direct radiation from radionuclides deposited on the ground, direct radiation from submersion in air, and ingestion of food contaminated by deposition from air

Out of the 20 hypothetical exposure scenarios, the results of 5 were considered to develop concentration limits for Class A, B, or C waste. Characteristics of those five scenarios are given in Table 3-4. The construction scenarios represent acute exposure (i.e., 500 hours) during excavation and construction to build a home. The agriculture scenarios represent chronic exposure (i.e., a total of 6180 hours indoors and outdoors onsite). The remaining 15 scenarios were discussed but the outputs calculated by INVERSI were not considered as potential waste class limits. Some of those remaining 15 scenarios are discussed in later sections of this report.

For each radionuclide and each class of waste, the result from the more limiting scenario applicable to that waste class was used to develop the concentration limit. For example, in the original analyses for Class C waste, the results from GEN5-CON and GEN5-AGR were considered for each radionuclide. The concentration limits defining Class C waste are a mix of results from those two scenarios because the relative risk significance of the different exposure pathways depends on the radionuclide. In general, the construction scenarios were more limiting for radionuclides for which inhalation is a particularly significant dose concern and the agriculture scenarios were more limiting in other cases.

As shown in Table 3-4, the only difference between the exposure scenarios used to develop the Class A limits (i.e., UNSI-CON and UNSI-AGR) and the scenarios used to develop the Class C limits (i.e., GEN5-CON and GEN5-AGR) is the assumed decay time (i.e., 102 years versus 502 years). Although Class C waste must be stabilized, the “GEN5” exposure scenarios did not

assume credit for waste stabilization because the stabilization was assumed to have failed by the time of intrusion. In the original analyses, separate exposure scenarios were used to account for different decay times because the original computer codes accounted for decay with analytical expressions and the codes could only generate results for one time of intrusion. To perform the GTCC analyses described in this report, the staff used commercial software that accounted for radioactive decay and in-growth and results are shown as a function of time rather than point estimates. Therefore, for the analyses described in this report, one model run could represent both "UNSI-CON" and "GEN5-CON" by taking the results corresponding to 102 years and 502 years, respectively. Similarly, one model run could represent both "UNSI-AGR" and "GEN5-AGR."

Table 3-4 Key Features of the INVERSI Exposure Scenarios Used to Develop the Waste Classification Limits of 10 CFR 61.55

Exposure scenario and INVERSI designation	Credit for Waste Stabilization	Credit for Layered Disposal	Credit for Intrusion Barrier or 5 meters or deeper disposal depth	PDCFs Used (media)	Decay Time (years) (includes 2-year post-operational period)	Applicable Waste Class (radionuclides)
Agriculture UNSI-AGR	No	No	No	3 (air) 4 (soil) 5 (soil)	102	Class A H-3, C-14, Fe-55, Ni-59, Co-60, Ni-63, Nb-94, Sr-90, Tc-99, I-129, Cs-135, and Cs-137
Construction UNSI-CON	No	No	No	2 (air) 5 (soil)	102	Class A U-235, U-238, Np-237, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Am-243, Cm-243, Cm-244
Construction (Discovery) STAI-CON	Yes	No	No	2 (air) 5 (soil)	102	Class B Ni-63, Sr-90, Cs-137
Agriculture GEN5-AGR	No (fails prior to intrusion)	No	Yes	3 (air) 4 (soil) 5 (soil)	502	Class C H-3, C-14, Fe-55, Ni-59, Co-60, Ni-63, Nb-94, Sr-90, Tc-99, I-129, Cs-135, and Cs-137
Construction GEN5-CON	No (fails prior to intrusion)	No	Yes	2 (air) 5 (soil)	502	Class C U-235, U-238, Np-237, Pu-238, Pu-239/240, Pu-241, Pu-242, Am-241, Am-243, Cm-243, Cm-244

3.1.1.1 Intruder Excavation

When waste is disposed less than 5 meters from the land surface the scenario evaluated is a potential excavation scenario. Potential doses to an individual performing the excavation and subsequent home construction are evaluated, followed by potential doses to an individual who lives in the home after it is constructed and consumes plants, animals, and animal products grown onsite.

Intruder Construction Exposure Scenario

The intruder construction scenario is used to represent an exposure scenario where a residence is inadvertently constructed on a disposal site at some time in the future. The intruder excavates a volume of material necessary to install a foundation for a moderately-sized home. The receptor in the intruder construction scenario is the reasonably maximally exposed person involved in the home construction. The dominant exposure pathway is inhalation of resuspended dust. The agriculture exposure scenario is used to represent the reasonably maximally exposed individual who lives in the residence after it has been constructed. The exposure times for the agriculture scenarios are much longer and the pathways of exposure more varied, with groundshine, consumption of contaminated food, and inhalation of resuspended dust contributing to intruder doses. The key parameters for the two exposure scenarios are summarized below.

The intruder construction scenario includes the pathways of inhalation of contaminated dust, immersion in a contaminated dust cloud, groundshine from contaminated soil, and ingestion of contaminated food from deposition of contaminated dust. However, the most dominant pathway for those isotopes with concentrations limited by the intruder construction scenario is the inhalation pathway. The inhalation pathway accounts for 90-99% of the dose with groundshine contributing most of the remainder (Figure 3-5a). The key parameters associated with inhalation and groundshine are summarized in Figure 3-6 for the intruder construction scenario.

The inhalation pathway calculation can be broken down into four principle areas:

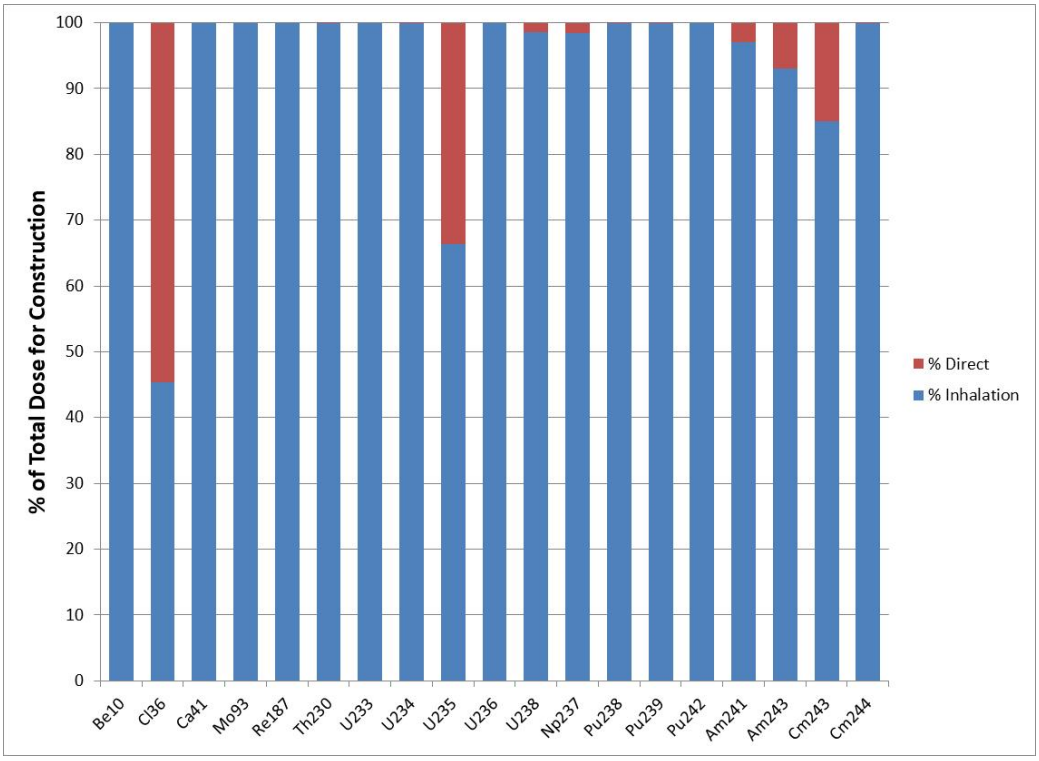
- Dilution or mixing of waste with non-waste materials
- The amount of material suspended in air
- The inhalation rate
- The exposure time

Given the parameters shown in Figure 3-6, the inhalation dose to the intruder who constructs a home can be calculated. Not shown are the dose conversion factors (DCFs) that define the amount of dose received per unit concentration of a radionuclide for a particular exposure type. The DCFs used in the analysis to develop the waste classification tables were based on ICRP 2. Those DCFs are organ-specific and are provided in Appendix B. For exposures to contaminated soil, the depth of the contamination must be considered as well as any shielding factors associated with the habits and actions of the receptor. For the intruder construction scenario used to develop the waste classification tables, the contaminants were assumed to be associated with an infinite, unshielded source. For the intruder agriculture scenario average shielding factors were developed for indoor and outdoor exposures.

Intruder Agriculture Exposure Scenario

For the intruder agriculture scenario the calculations are a bit more complex and include more potential pathways. In the agricultural exposure scenarios, waste is assumed to be excavated during the construction of a home and mixed with soil surrounding the home. The mixture is assumed to be 25 percent waste and 75 percent soil. The major pathways for the intruder agriculture scenario include groundshine, consumption of contaminated food, and inhalation of resuspended dust. The groundshine and inhalation calculations are functionally similar to the intruder construction, however the exposure time is longer and the mass loading values are smaller. In addition, shielding of the individual by the home while the person is indoors and accounting for the finite size of the source of radiation are accounted for through calculation of indoor (0.24) and outdoor (0.74) shielding factors. Figure 3-7 summarizes the main parameters used in the intruder agriculture scenario. Many of the parameters in the ingestion pathway are radionuclide-specific (vectors). The parameter description and units are provided in Figure 3-7, but the vectors are too long to provide in the figure. The values assigned for the vectors are listed in Table 3-5.

a)



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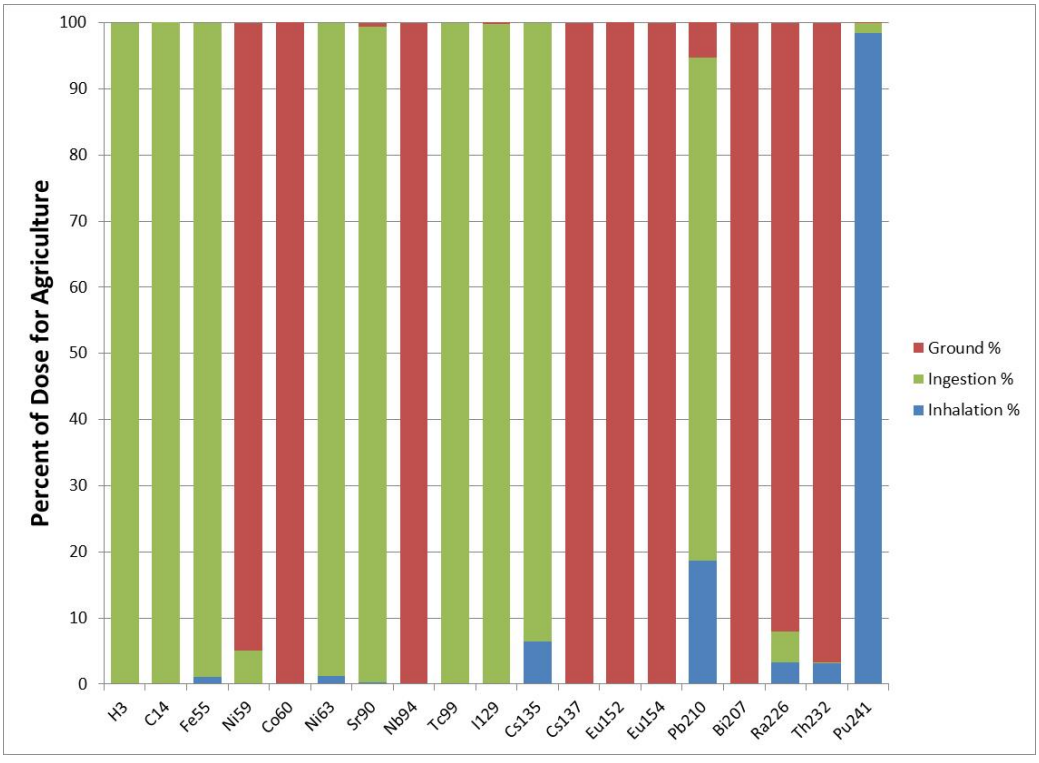

Figure 3-5 a) Contributions to Total Dose for the Intruder Construction Scenario, b) Contributions to Total Dose for the Intruder Agriculture Scenario.

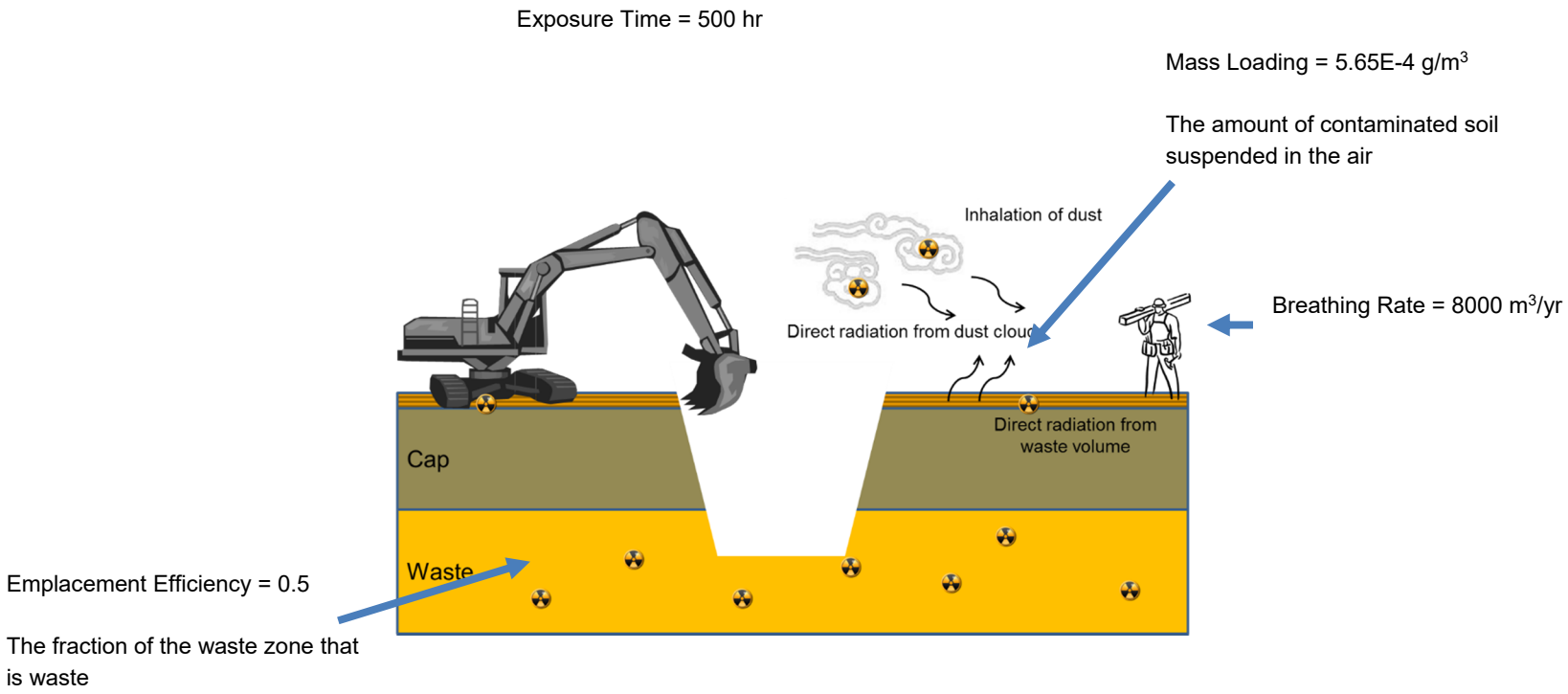

Figure 3-6 Main Parameters for the Intruder Construction Exposure Scenario

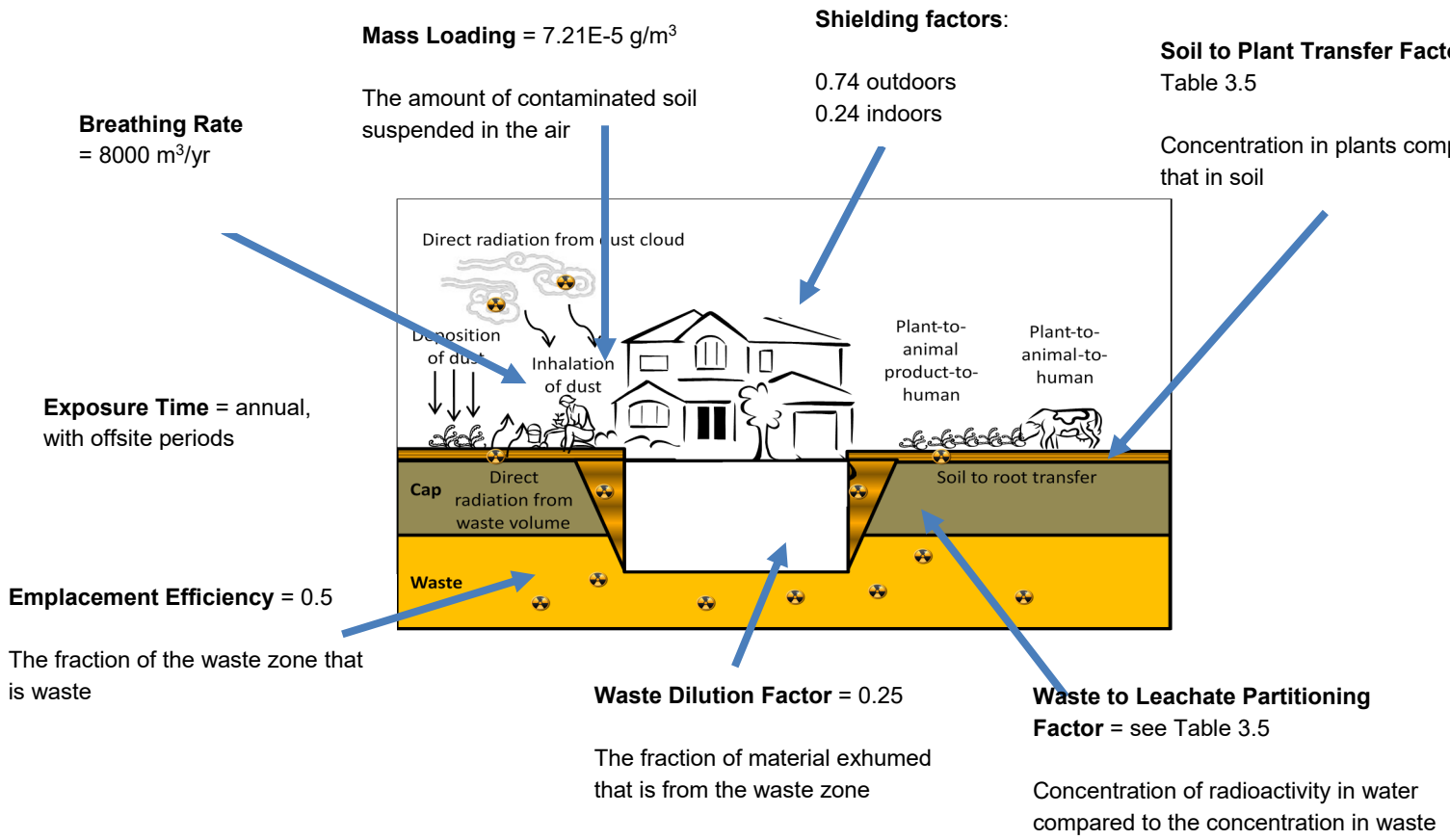

Figure 3-7 Main Parameters for the Intruder Agriculture Exposure Scenario

Table 3-5 Transfer and Partitioning Factors Used in the Intruder Agriculture Scenario

Radionuclide	Soil to Plant Transfer Factor	Waste to Leachate Partitioning Factor (pCi/L leachate per pCi/L waste)
H-3	4.80E+00	1.15E+00
C-14	5.50E+00	5.76E-03
Fe-55	6.60E-04	1.48E-02
Ni-59	1.90E-02	1.48E-02
Co-60	9.40E-03	1.48E-02
Ni-63	1.90E-02	1.48E-02
Nb-94	9.40E-03	1.11E-02
Sr-90	2.90E-01	9.86E-03
Tc-99	1.10E+00	1.15E-01
I-129	5.50E-02	1.15E-01
Cs-135	9.30E-03	1.62E-04
Cs-137	9.30E-03	1.62E-04
U-235	2.50E-03	1.25E-04
U-238	2.50E-03	1.25E-04
Np-237	2.50E-03	4.67E-04
Pu-238	5.60E-04	4.67E-04
Pu-239	5.60E-04	4.67E-04
Pu-241	5.60E-04	4.67E-04
Pu-242	5.60E-04	4.67E-04
Am-241	5.60E-03	4.11E-03
Am-243	5.60E-03	4.11E-03
Cm-243	2.50E-03	4.67E-04
Cm-244	2.50E-03	4.67E-04

Table 3-6 Annual Fractional Availability of Radionuclides to Plant Uptake for Waste Mixed with Soil (factors applied in addition to soil-to-plant uptake factors).

Waste type	Activated Metals [stainless]	Sealed Source	Sealed Source	Other [decon waste]	Other [decon waste]	Activated Metals [stainless]	Other [no metal]	Other [no metal]	Activated Metals [stainless & carbon]	Other [mostly trash]	Activated Metal [stainless-reactors]	Other [mainly MOX waste]	Other [mainly Pu generators]	Other [WVTF decom]	Other [WVTF decom]	Other [Pu targets]	Other [Pu targets]
Nuclide	ReactorAM RH	Cs137SS CH	NeutronIR CH	LikeWVDecon CH	LikeWVDecon RH	ReactorAM RH	Mo99Murr RH	Mo99Mips RH	WVNDAAM RH	WVNDAO RH	WVSDAAM RH	WVSDAO CH	WVSNAPO CH	LikeWVDecomO CH	LikeWVDecomO RH	LikePu238O CH	LikePu238O RH
H3	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
C14	$1x10^{-5}$	1.0	1.0	1.0	1.0	$1x10^{-5}$	1.0	1.0	1.0	1.0	$1x10^{-5}$	1.0	1.0	1.0	1.0	1.0	1.0
Ni59	$1x10^{-5}$	1.0	1.0	1.0	1.0	$1x10^{-5}$	1.0	1.0	1.0	1.0	$1x10^{-5}$	1.0	1.0	1.0	1.0	1.0	1.0
Co60	$1x10^{-5}$	1.0	1.0	1.0	1.0	$1x10^{-5}$	1.0	1.0	1.0	1.0	$1x10^{-5}$	1.0	1.0	1.0	1.0	1.0	1.0
Ni63	$1x10^{-5}$	1.0	1.0	1.0	1.0	$1x10^{-5}$	1.0	1.0	1.0	1.0	$1x10^{-5}$	1.0	1.0	1.0	1.0	1.0	1.0
Sr90	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Tc99	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
I129	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Cs137	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
U233	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
U234	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
U235	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
U236	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
U238	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Np237	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Pu238	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Pu239	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Pu241	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Am241	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Cm243	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Cm244	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

The ingestion exposure pathways in the agricultural exposure scenario account for two sources of contamination: (1) soil contamination from mixing with waste and (2) soil and plant contamination from deposition of contaminated dust. Food can be contaminated by deposition of radioactivity onto plants from the air or root uptake of radioactivity by plants from contaminated soil.

Soil contaminated by being mixed with exhumed waste is modeled as contaminating plants through root uptake. The pathways that begin with deposition of contaminated dust consider two methods of plant contamination: (1) dust deposition onto plant leaves and (2) root uptake of radionuclides from soil contaminated by dust deposition. Soil and plant contamination from contaminated irrigation water are not included in the agricultural scenario.[11] Although root uptake is considered for both contamination sources (i.e., waste mixing with soil and dust deposition), there is a difference in how the root uptake is treated depending on the contamination source. For the pathways that begin with waste mixed with soil, the radionuclides in the waste are modeled as needing to transfer (partition) into the liquid phase before they can be taken up by plant roots (Table 3.5). For all low-level waste (stabilized and unstabilized) credit is assigned for the wasteform. That is, for the waste-mixed-with-soil source, root uptake is calculated as:

$$C_{WS} \cdot F_{MF} \cdot F_{SP} = C_{WSUP} \qquad \text{Eqn. 1}$$

Where C_{WS} is the effective concentration of radionuclides in the soil mixed with waste, F_{MF} is a leachate factor (i.e., the ratio of the volumetric concentration of radionuclides in leachate to the volumetric concentration of radionuclides in the waste), F_{SP} is the soil-to-plant transfer factor, and C_{WSUP} is the radionuclide concentration in the plant from root uptake of radionuclides from waste mixed with soil.

In contrast, for the pathways that begin with the deposition of contaminated dust onto soil and plants, the root uptake portion of the contamination is calculated with a soil-to-plant transfer factor applied directly to the soil concentration (Table 3.5). The equation for this calculation is:

$$C_{DS} \cdot F_{SP} = C_{DSUP} \qquad \text{Eqn. 2}$$

Where C_{DS} is the concentration of radionuclides in soil from dust deposition, F_{SP} is the soil-to-plant transfer factor, and C_{DSUP} is the concentration in the plant from root uptake of radionuclides in soil from deposition of contaminated dust. The soil-to-plant transfer factors used in Equations 1 and 2 are the same and given in Table 3-5. In addition to root uptake of soil contaminated by dust deposition, dust deposition directly on to plants is modeled. However, this pathway is generally of minor significance compared to root uptake directly from contaminated surface soil.

Conceptually, the difference between Equations 1 and 2 is that contamination in the soil-waste mixture is assumed to be less available to plants than radionuclides deposited in the soil from contaminated dust. Whereas the availability of contamination in soil from deposited dust is assumed to be equal to the availability of radionuclides in the soils in the experiments done to determine the soil-to-plant transfer factors, the availability of the contamination in the soil-waste mixture is assumed to be restricted by the availability of the radionuclides in the waste. The

[11] Contamination from soil and foliar deposition of contaminants in irrigation water are included in the "INT_WAT" and "ERO_WAT" scenarios in the original code. Those exposure scenarios did not lead directly to any waste classification values.

validity of that assumption depends on the waste type and possibly the exposure of the waste to degradation and partitioning processes prior to being distributed in the soil. For example, radionuclides within activated metal are expected to be less available to plants than radionuclides present as surface contamination on cloth or paper products.

Although waste-to-leachate factors used in the original Part 61 analysis were derived from disposal site data, they are difficult to apply to an analysis of GTCC waste because they were derived from measurements of leachate from trenches of waste containing multiple waste types and the applicability of the data to GTCC waste is difficult to determine. Furthermore, the factors were developed from concentrations in the leachate after it migrated out of the waste trench, such that the concentrations incorporated both release from the wasteform and transport through the waste. That factor may not be conceptually consistent with a model in which the factor is applied in addition to a soil-to-plant uptake factor because a soil-to-plant uptake factor already accounts for the tendency of radionuclides to adsorb to surfaces (i.e., the sorptivity may be double counted).

As shown in Table 3-5, the effect on the projected plant uptake in the original Part 61 analysis varied by radionuclide. For example, the waste-to-leachate partitioning factor reduces the projected dose from the plant uptake pathway by approximately an order of magnitude for Tc-99 and I-129 and by approximately four orders of magnitude for isotopes of cesium, uranium, and most TRU. The values used were the same for many radionuclides because the original measurements were made for H-3, C-14, C-60, Sr-90, Cs-137, U-238, Pu-339, and Am-241 and were assigned to other radionuclides based on expected chemical similarity.[12]

In the original Part 61 analysis, radionuclide unavailability is also credited in two other ways. First, for C-14, Ni-59, Ni-63, and Nb-94 in activated metals, the waste class limits were increased by a factor of 10 to credit the activated metal wasteform. Second, the Class C limits for all radionuclides were increased by a factor of 10 to account for the inaccessibility of Class C waste and the unavailability of radionuclides in the types of waste likely to be represented in Class C waste (i.e., a net increase of 100 for C-14, Ni-59, Ni-63, and Nb-94 in activated metals). For example, for Cs-137 in Class C waste, which is not predominantly expected to occur in activated metals, the original calculation provides approximately four orders of magnitude reduction in the exposure pathways based on plant root uptake and another order of magnitude credit for the combination of radionuclide unavailability and waste inaccessibility in Class C waste.

Because the assumptions underlying the use of the waste-to-leachate factor in the original analysis may not be applicable to GTCC waste, in this assessment, the NRC staff considered both results using the original leachate-to-waste factors as well as results from calculations using an alternative representation of radionuclide availability. Instead of a vector by radionuclide that is applied to all waste types (i.e., the leachate-to-waste partitioning factor in Table 3-5), the alternative is a matrix of values that varies by radionuclide and waste type. For trash, soils, and “other” waste, radionuclides were assumed to be as available as they are in soil. That is the waste-to-leachate partitioning factor was set to 1 (see Table 3-6) and the soil-to-plant uptake factor (Table 3-5) was applied directly to the concentration in the waste mixed with soil. Similarly, radionuclides present as surface contamination on activated metals (i.e., not radionuclides within activated metals) are also assumed to be as available to plants as they would be if they were adsorbed to soil. In contrast, radionuclides within activated metals in

[12] NURE/CR-1759 Vol. 3, Table 3-7 indicates that the value of 1.11×10^{-2} for Nb-94 was based on the value of 1.48×10^{-2} for Co-60, but does not provide an explanation for the difference between the values.

stainless steel were assumed to have a release fraction of 1 x 10^{-5} per year. That value was based on the DOE value of 1.19 x 10^{-5} yr^{-1} release fraction for stainless steel activated metals waste streams as described in Appendix E of the DOE FEIS (2016) and the assumption that radionuclides within activated metal would become available for plant uptake as the metals corrode. Sealed sources were assumed to be disrupted by the intrusion and were not credited with limiting plant root uptake of radionuclides.

The staff has not performed a comprehensive evaluation of all the 17 waste streams to determine appropriate release fractions and has relied primarily on conservative values (i.e., values of 1 for fractional availability in Table 3-6). Release fractions for radionuclides within activated metals of stainless steel were assigned a significantly lower rate (i.e. 1x10^{-5}) due to the properties of stainless steel and the clear identification of the waste form for certain waste streams as predominately being stainless steel. The use of this release fraction is used to demonstrate the importance of certain waste forms for reducing potential exposures. Although effort for characterizing and understanding the waste forms for other waste streams was not done for this report, further effort in characterizing release fractions for other waste streams could also be significant in reducing estimates for potential exposures.

For both sources of plant contamination (i.e., waste mixing with soil and dust deposition), the subsequent contamination of meat and milk and the resulting exposure of humans from ingestion of contaminated plants, milk, and meat is calculated in the same way. Irrespective of the source of the contamination in the plants, the same transfer factors are used to project the contamination of milk and meat from the concentration in the plants. Human consumption of milk and meat are also projected with annual consumption factors. Incidental soil ingestion is not included in the calculations. Results of the contractor analysis indicated that incidental ingestion is unlikely to comprise more than 15% of the projected dose.

3.1.1.2 Intruder Drilling

Assumptions were made in the analyses performed to develop the waste classification tables found in 10 CFR Part 61.55. Some of these key assumptions were:

- Intrusion was based on excavation of the waste.
- The waste would be indistinguishable from soil.
- The waste would be readily dispersed.
- Limiting concentrations were based on only two exposure scenarios: intruder-construction or intruder-agriculture.
- Contaminated water was not used by the intruder.
- Environmental conditions were those of a humid, Southeastern location.

For GTCC waste disposal, some of these assumptions may not be valid. If the regulations for Class C waste require either deeper disposal (> 5 m) in order to avoid an excavation exposure scenario or implementation of an intruder barrier that prevents intrusion for at least 500 years (thereby allowing for decay) it would be anticipated that requirements for GTCC waste would need to be at least as stringent and probably more stringent. The analyses for GTCC waste disposal needed to take into account deeper disposal or the presence of an intruder barrier.

Staff started with the TableCalculator tool (NRC, 2019), which replicates the 10 CFR Part 61 calculations used to develop the waste classification requirements. Next, modifications were made to account for the smaller volume of material disturbed, increased mixing with non-waste,

and the shorter exposure time of a driller compared to the intruder-construction receptor (i.e., an individual excavating a foundation and then building a residence). Additionally, the calculations made use of the decay and ingrowth of radionuclides such that the dose vs. time can be calculated. It can be important to consider the temporal profile of dose projections to determine if barriers to intrusion may need to be implemented. In the 10 CFR Part 61 calculations, the concentrations (dose) are calculated at a user-defined point in time. Radioactive decay was taken into account through analytical expressions. Ingrowth for select radionuclides was considered, although full decay chains were not evaluated. Figure 3-8 summarizes the key parameters for the intruder drilling exposure scenario that is used to model intrusion into waste placed at depths greater than 5 m from the land surface.

The form and dispersibility of the waste after excavation or drilling could be important to consider in the dose assessment. Some GTCC waste represents material that has been exhumed following decommissioning or remediation and may have been exposed to the natural environment for multiple decades. Other GTCC wastes, such as sealed sources, have been stored in controlled conditions and may involve use of robust materials (e.g., stainless steel) for encapsulation or packaging. In some cases, the source material may be in a dispersible form contained in a robust capsule. Staff considered the form and dispersibility of the waste for the intruder-drilling scenario. One special case looked at different release fractions (using the factor FMF, described later). It is likely that the range of potential disposal conditions could significantly affect the state of the material at the time of drilling. In addition, the type of drilling employed (e.g. rotary vs. sonic) could impact the waste differently. These aspects are better addressed on a site-specific basis for a disposal application rather than in a generic regulatory analysis.

The IMPACTS codes included the capability to assign credit for the form of the waste. If stabilization was implemented, the waste was assumed to be less dispersible and therefore effective release rates would be lower. In addition, Class B waste was required to be stabilized and therefore the exposure time was significantly decreased because it was assumed the waste would be recognizable. The excavation exposure scenario generally involves less energetic disturbance compared to a drilling scenario. Waste that is recognizable to an excavator may not be recognizable to a driller. For instance, sonic drilling is used to drill through very hard rock and may disperse most waste types.

Intruder Drilling Exposure Scenario

The drilling exposure scenario involves the exhumation of approximately 1 m^3 of soil and waste for a 15 cm diameter well drilled to an average depth of approximately 55 m. The IMPACTS calculations examined an intruder-excavation scenario where 232 m^3 of waste (and approximately 3 times that volume of soil) was exhumed. When that amount of material is spread in the vicinity of a newly constructed home the dilution factor used (0.25) in combination with the exposure times (100 hours gardening, 1700 hours outdoors, 4380 hours indoors) weren't unduly conservative. The waste/soil mixture was assumed to be spread in an area of approximately 2000 m^2, which corresponds to a thickness of approximately 0.3 m. The exposure times and shielding factors were consistent with a contaminated area of this size. However, in the current analysis where a drilling scenario is evaluated for more deeply disposed waste, the volume of material exhumed is so small (~ 1 m^3) that the combination of chronic exposure times and waste concentrations are inconsistent. If the same area was used (2000 m^2) then the contaminated layer would only be 0.5 mm thick and could not persist for a year. NRC staff assumed that the drill cuttings would be spread in a circle of radius 5-15 m. Then after the spreading of the cuttings, the waste soil mixture would be further mixed in a layer of 5

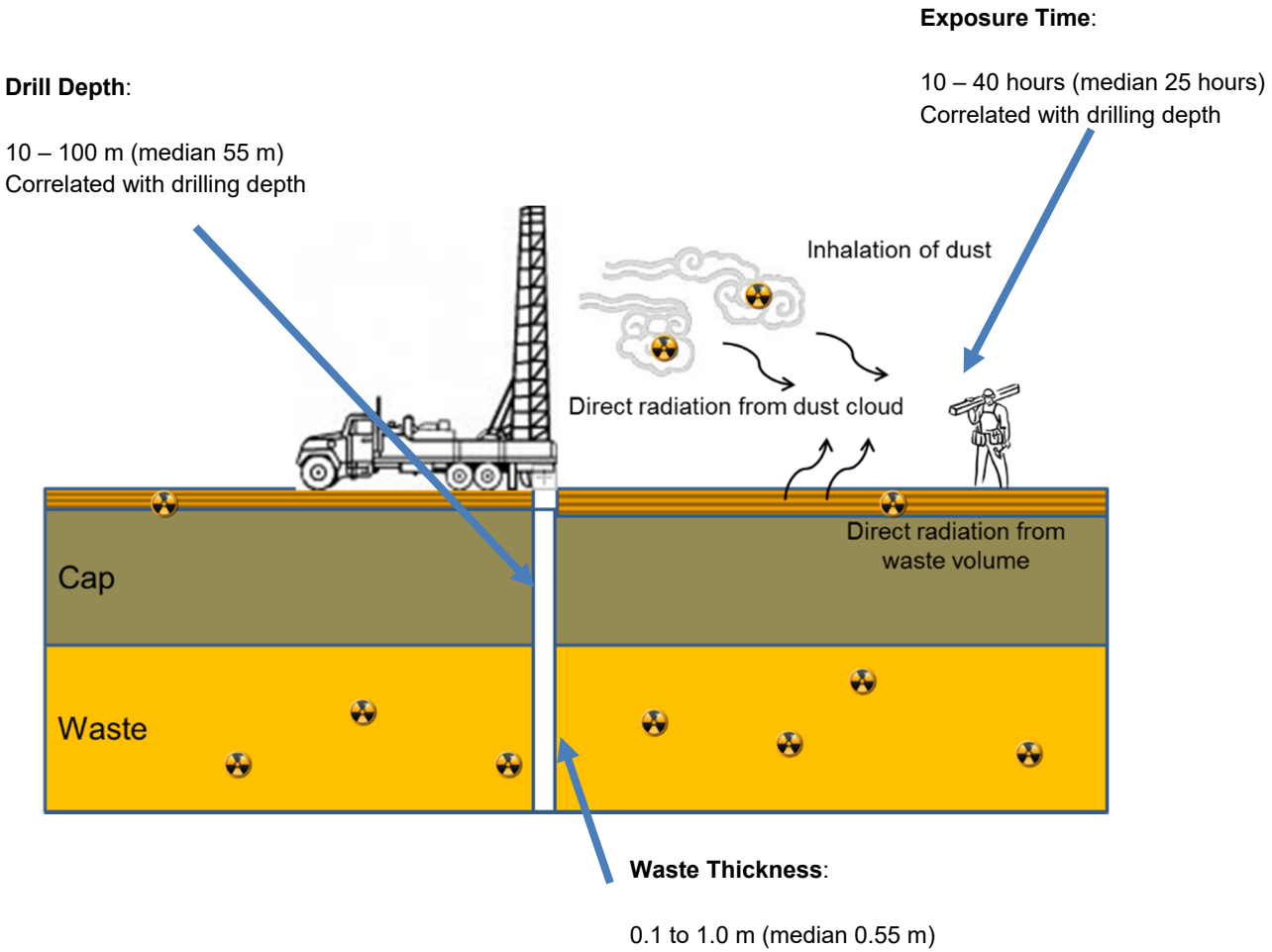

Figure 3-8 Main Parameters for the Intruder Drilling Exposure Scenario (Unless shown, the parameters used are the same as for the intruder construction scenario)

cm thickness. The gardening activity has the highest mass loading values by an order of magnitude, and it is typical to assume a till depth of 15 cm for agricultural activities. However, not all the spread material would be in the garden plot. As will be discussed in the results section, this approach reduces the dose for chronic exposure from an inadvertent intruder drilling scenario (chronic drilling scenario) such that they are similar in magnitude to the doses from acute exposure from an inadvertent intruder drilling scenario (acute drilling scenario).

Though many exposure pathways could be included, consideration of contaminated water ingestion and domestic water usage provides a good first-order estimate of the radiological impacts to an offsite member of the public. Accidents are considered separately in a later section.

The modeling to develop 10 CFR Part 61 considered potential impacts to offsite members of the public. That modeling is discussed in the previously cited references (NRC, 1981a; NRC, 1981b; NRC, 1982b; NRC, 1983). Calculations to estimate radionuclide concentrations in groundwater were performed and erosional release scenarios were considered. However, it was recognized that the results were significantly influenced by site-specific characteristics. If the results of the analyses were used to set concentration limits, the concentration of radionuclides that would be acceptable would be limited to that acceptable for the "poor" disposal sites. A highly-performing site would be unnecessarily restricted in the waste that would be suitable for disposal. For this reason, the requirements that were developed were for a licensee to perform site-specific pathway analyses. In this analysis, staff evaluated groundwater impacts to determine if sites with certain characteristics should be prohibited from receiving GTCC waste.

The three main components of this analysis are highlighted in Figure 3-9 in the dashed boxes: release from the source term, transport through geologic materials, and ingestion of contaminated drinking water. The assessment of impacts to an offsite member of the public can be much more complex than evaluated here. But to perform a generic screening assessment of GTCC waste streams the components highlighted in Figure 3-9 are generally sufficient. Any specific impacts for select waste streams at particular sites would be evaluated in the analyses developed to support licensing of the specific disposal facility.

Figure 3-10 is a screenshot of the dose assessment model for the offsite member of the public. The main elements are a source term, vadose zone flow and transport, flow and transport through an aquifer, and conversion of radionuclide concentrations in ingested water to a radiological dose. The representations and parameter values (distributions) assigned are described below. The description provided here assumes some basic working knowledge of the software package GoldSim. The software package has detailed user manuals that explain the various elements and how they work. The model simulates the decay, in-growth, release, and environmental transport through the subsurface of up to 38 different species (radionuclides). Table 3-6 provides the half-lives and modeled daughter products. The model was built with the capability for the user to switch between the half-lives used in NUREG/CR-1759 Vol.3 and more modern values. Both sets of values are shown in the table. Tables 3-7 to 3-11 provide other key inputs used in the simulations.

3.2 Offsite Individual – Post-closure

Protection of members of the public who live near, but not on, the disposal site is one of the main safety components of 10 CFR Part 61. The regulations at § 61.41 require protection of

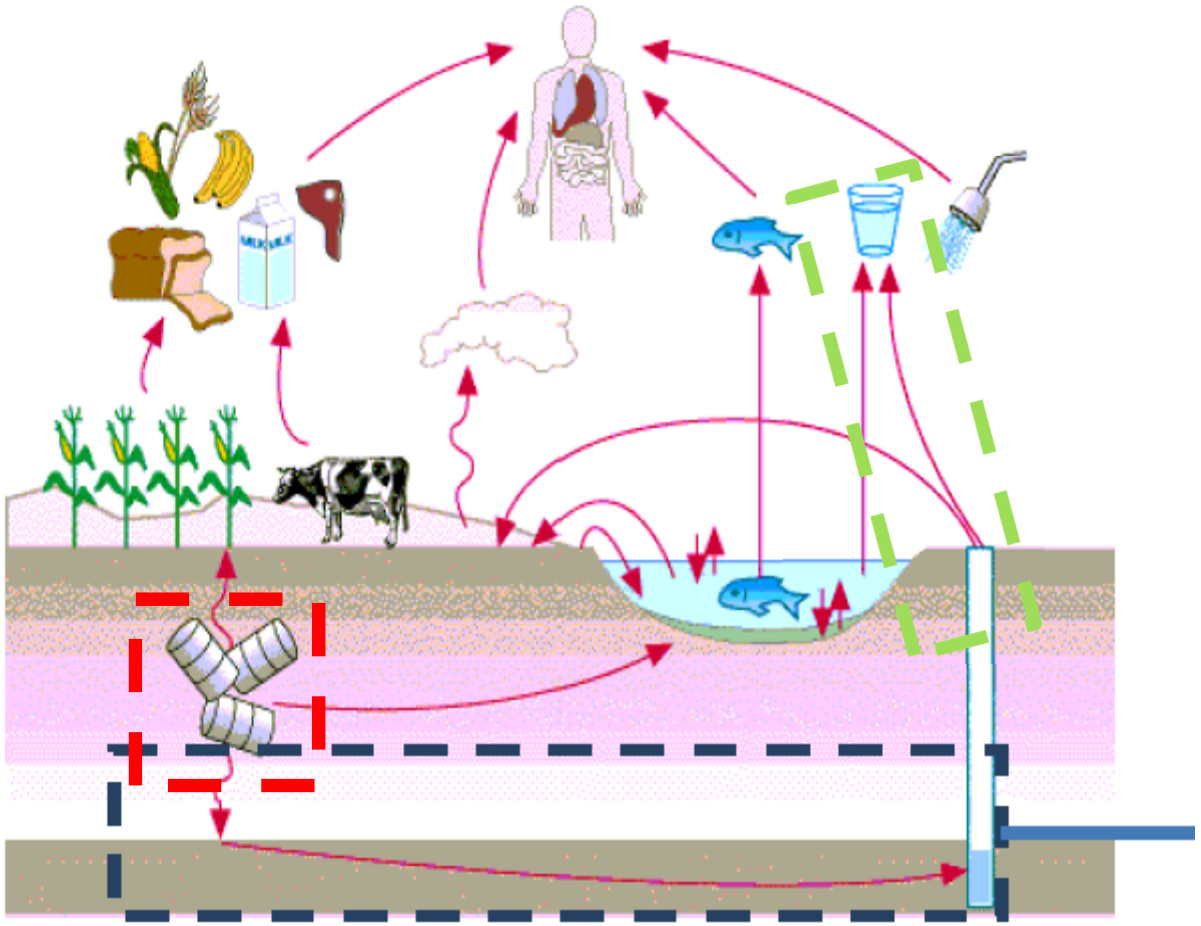

Figure 3-9 Overview of the Offsite Member of the Public Dose Assessment (boxed areas show aspects explicitly considered in this analysis)

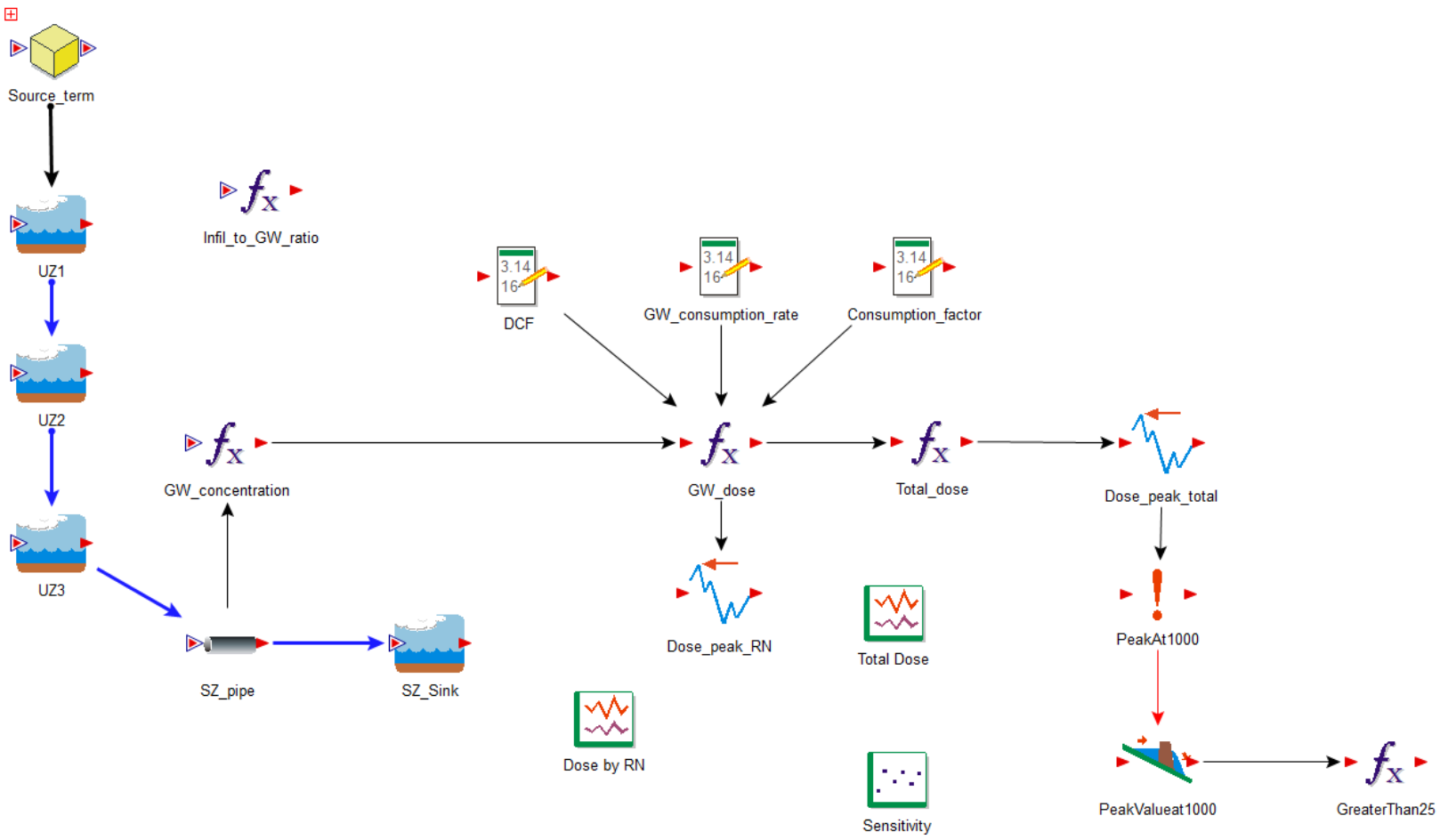

Figure 3-10 Screen Snapshot of the Offsite Dose Calculation

Table 3-6 Species List and Half Lives

Isotope	NUREG/CR-1759 Vol. 3 Half Life (yr)	Modern Half Life (yr)	Modeled Daughter Isotopes
H3	12.32	12.32	
Be10	1.39E+06	1.39E+06	
C14	5730	5700	
Cl36	3.01E+05	3.01E+05	
Ca41	1.03E+05	1.03E+05	
Fe55	2.6	2.744	
Ni59	8.0E+04	7.60E+04	
Co60	5.26	5.26	
Ni63	92	100	
Sr90	28.1	28.79	
Mo93	4000	4000	
Nb94	20000	20300	
Tc99	2.12E+05	2.11E+05	
I129	1.17E+07	1.57E+07	
Cs135	3.00E+06	2.30E+06	
Cs137	30	30.167	
Eu152	13.537	13.537	
Eu154	8.593	8.593	
Re187	4.12E+10	4.12E+10	
Pb210	22.2	22.2	
Bi207	31.55	31.55	
Ra226	1600	1600	Pb210
Th230	75380	75380	Ra226
Th232	1.41E+10	1.41E+10	
U233	1.59E+05	1.59E+05	
U234	2.46E+05	2.46E+05	Th230
U235	7.10E+08	7.04E+08	
U236	2.34E+07	2.34E+07	
U238	4.51E+09	4.47E+09	U234
Np237	2.14E+06	2.14E+06	U233
Pu238	86.4	87.7	U234
Pu239	24400	24100	U235
Pu241	13.2	14.35	Am241, Np237
Pu242	2.79E+05	3.75E+05	U238
Am241	458	432.2	Np237
Am243	7950	7370	Pu239
Cm243	32	29.1	Pu239, Am243
Cm244	17.6	18.1	

Table 3-7 Waste Stream Volumes

Waste Stream	Volume (m^3)
ReactorAMRH	880
Cs137SSCH	1000
NeutronIRCH	1800
LikeWVDeconCH	710
LikeWVDeconRH	540
ReactorAMRH1	370
Mo99MurrRH	355
Mo99MipsRH	35
WVNDAAMRH	210
WVNDAORH	1900
WVSDAAMRH	525
WVSDAOCH	400
WVSNAPOCH	1200
LikeWVDecomOCH	220
LikeWVDecomORH	760
LikePu238OCH	120
LikePu238ORH	260
CHTotal	5450
RHTotal	5835
Total	11285
Test	1000

Table 3-8 Stochastic Parameters Used for the Offsite Dose Calculation

Parameter Name	Distribution Type	Units	Minimum	Maximum
Source_porosity	uniform	-	0.25	0.4
Source_density	uniform	kg/m^3	1200	1600
Source_liquid_sat	uniform	-	0.3	0.5
Infiltration_rate	Log uniform	mm/yr	1	300
UZ_porosity	uniform	-	0.15	0.4
UZ_density	uniform	kg/m^3	1400	1600
UZ_thickness	uniform	m	5	100
UZ_liquid_sat	uniform	-	0.25	0.8
SZ_density	uniform	kg/m^3	1600	1800
SZ_porosity	uniform	-	0.1	0.3
SZ_thickness	uniform	m	5	5.01
SZ_length	uniform	m	100	100.1
AquiferFlowRate	Log uniform	m/yr	0.1	1000
SZ_dispersivity_factor	uniform	-	0.05	0.15
bound_waste_rate	Log uniform	1/yr	0.0001	1
WasteThickness	uniform	m	0.1	1.0

any member of the public from the disposal of low-level waste in the near surface. NRC and contractor staff performed analyses to evaluate potential doses to members of the public that may result from the disposal of GTCC waste.

3.2.1 Models and Data

Figure 3-9 outlines the components of a dose assessment for an offsite member of the public. The dose assessment for an offsite member of the public is generally more complex than the inadvertent intruder dose assessment. For the intruder dose assessment, the intruder produces the disturbance and release of waste to the environment due to a discrete event and simple relationships can be used to derive concentrations. For an offsite member of the public, natural phenomena produce releases at gradual and variable rates, and then a variety of transport processes must occur for the radioactivity to reach the potential receptor.[13] The main exposure scenario typically analyzed is leaching of radioactivity into subsurface media followed by transport in water to a well. Water withdrawn from the well is assumed to be used for domestic purposes. Other exposure scenarios that can be evaluated include, but are not limited to, release and transport in air, erosional release of waste, and natural disruptive events (e.g., floods, earthquakes).

The behaviors and characteristics of the potential offsite receptors can be different from and, in general, more varied than, the characteristics and behaviors of inadvertent intruders. To avoid excessive speculation, the characteristics of the intruders are recommended to be constrained and defined in a reasonable but protective way. People are expected to need shelter and food in the distant future. However, the behavior and characteristics of offsite members of the public are anticipated to be site-specific and driven by the release pathway(s) and phenomena. For example, if flooding were to disrupt a disposal site it would also likely disturb nearby receptors. For this analysis, the offsite receptor is defined to be a member of the public living 100 m from the disposed waste. The offsite receptor has a well that is used to provide both drinking water and irrigation water for a garden that supplies a portion on the intruder's annual vegetable intake.

The source term representation is shown in Figure 3-11. The source definition container contents shown on the right side of the figure provide the ability to specify which source is being evaluated (e.g., ReactorAMRH) using the *SourceIndex*[14] parameter. The volume is then selected from a vector by waste streams defined with *GTCCStreamVolumes* (Table 3-7). The user may specify a rapid release fraction (*RapidFraction*) which is a fractional amount of each species that would be immediately available to be released from the source term at the beginning of the simulation. The rapidly released fraction is input directly into the first cell of the vadose zone or unsaturated zone (*UZ1*). The other parameters shown on the right side are the calculated inputs to the source term and are generally self-explanatory. The *input_mass* element is the amount of grams of each isotope provided as input to the GoldSim source element *source_term_input*. For the source element, many features are available in GoldSim to represent release from a wasteform. The only parameter used in this model is a parameter to define release from a waste matrix (Note: Other parameters (such as FMF described later) can be used to limit the availability of material that has been released). The choices available are "none", "specified lifetime", "specified degradation rate", or "congruent dissolution". The "specified degradation rate" option is used which is a fractional degradation rate per unit time.

[13] A receptor is an individual who could be exposed to radiation from the modeled facility.
[14] Parameter names are provided in italic

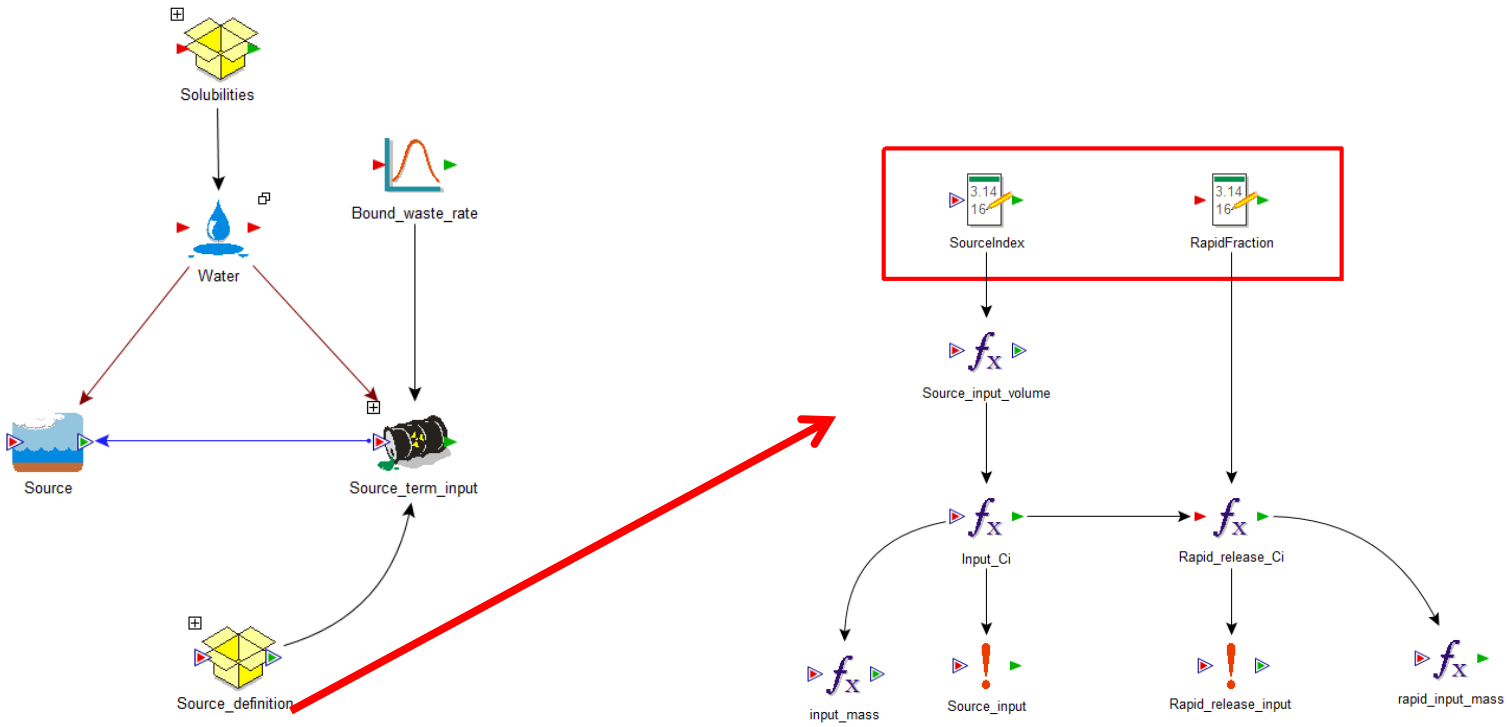

Figure 3-11 Source Term Representation in the Offsite Dose Calculation

The stochastic parameter *bound_waste_rate* was assigned a log-uniform distribution with a range of 0.0001 1/yr to 1 1/yr. The range encompasses very good wasteforms to almost no credit for a wasteform. The median value of 0.01 1/yr corresponds to approximately a 100-year wasteform. It should be noted that a constant fractional rate per unit time does not mean that the wasteform lasts exactly the inverse of the rate because the source is exponentially depleted. The GoldSim source element can be thought of as making available radionuclides that can then speciate in the fluid phase and adsorb onto solid materials, which occurs in the *Source* cell. The source cell contains water and source material. The amount of water is based on the user-defined source material porosity and source liquid saturation (Table 3-8). There is also a parameter (*release_delay*) that prevents any release until the time specified for the parameter is exceeded. The parameter was set at 100 years to represent a well-maintained system during the institutional control period.

After radioactivity is released from the wasteform, it can then partition to solids in the source element and will be limited by the solubility limits applied within the source element. Table 3-9 provides the distribution coefficients and Table 3-10 provides the solubility limits applied in the model. Distribution coefficients can vary widely from site to site because of differences in minerology and geochemistry. For instance, the observed values for uranium range from 0.03 to 395,100 ml/g for different soil types in one commonly cited reference (Thibault et al., 1990). For sand, the observed values for uranium ranged from 0.03 to 2200 ml/g. The right most column of Table 3-9 provides a comparison to the geometric mean values for sand for (Thibault et al., 1990). The values used in this study were consistent with the reference values but were not set the same as the reference values because an attempt was made to account for different material types. None of the existing disposal facilities are in locations where the disposal units are in sand. The important aspect of this evaluation was to understand the significance of the variability in geochemical parameters, which was accomplished by assigning probability distributions rather than point estimates. Solubility limits were only applied within the source element (i.e., disposal trench). Once radioactivity enters the geological environment solubility limits are assigned very large values in the model such that, in effect, a solubility limit is not applied. In general, advection is the dominant release mode in most near-surface disposal systems. For diffusion-dominated transport systems the radiological risks tend to be low. Only advective release was explicitly included in the screening model.

For each waste stream, the area of the source term is calculated based on the volume of the waste stream and the sampled value of the waste thickness defined by the *WasteThickness* parameter. The product of the area and the infiltration rate (*Infiltration_rate*) determined the amount of water flowing through the waste disposal trench. Released radioactivity is then transported through a series of three transport cells (*UZ1, UZ2, UZ3*) to represent the vadose zone (Figure 3-10). The vadose zone cells contain a mass of soil or geologic media and a volume of water defined by the parameters provided in Table 3-8. The use of three cells, based on the numerical representation of cells in the model, provides for numerical dispersion that is comparable to typical hydrodynamic dispersion. For this screening calculation, dispersion in the vadose zone is not a significant process.

Radioactivity that exits in the vadose zone enters a pipe element which is used to simulate an underlying aquifer. The pipe element is a GoldSim element specifically designed to simulate flow and transport phenomena. Figure 3-12 shows the element interface window for the pipe element as implemented in the screening model. The length of the pipe is the sum of the parameter *SZ_length* and the *SourceLength*. The first parameter is set at 100 m to represent a receptor 100 m from the edge of the disposal trench. The second parameter is a calculated parameter to represent the average length of the source. It is the square root of the area of the

source. The area through which contamination is flowing is the product of the source length and the thickness of the aquifer (*SZ_thickness*). The thickness of the aquifer was fixed at 5 m to represent the well screen length over which contaminated water would be extracted. Dispersivity is the product of the *SZ_dispersivity_factor* parameter and the total length of the pipe element.

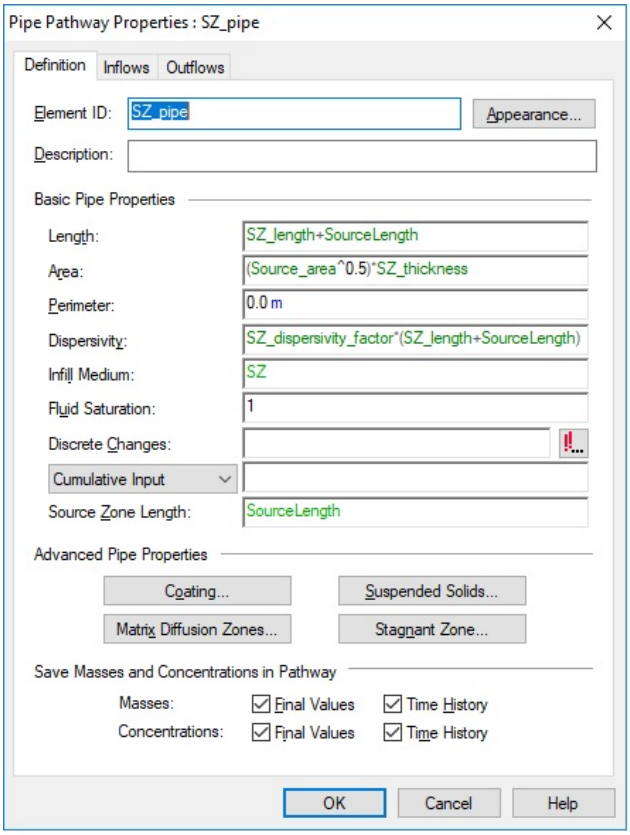

Figure 3-12 Element Interface for the Pipe Element (Aquifer)

The calculated concentrations of radioactivity exiting the aquifer are multiplied by three parameters: dose conversion factor (*DCF*), groundwater ingestion rate (*GW_consumption_rate*), and a factor to account for ingestion of contaminated food, which is not explicitly calculated (*Consumption_factor*). The groundwater ingestion rate was fixed at 730 L/yr (2 L/day) and the consumption factor was set at 2. The consumption factor represents the ratio of the receptor's total ingestion of radionuclides (i.e., including consumption of plant and animal products that were contaminated by contaminated water) to the consumption of contaminated water alone. The assumed value of the consumption factor is based on the staff's experience with many different performance assessment models and the associated results. The ratio of the dose from the ingestion of plant and animal products compared to the groundwater ingestion dose is strongly dependent on the particular radionuclides and the consumption rates of various products. It is appropriate for a generic screening calculation but is not appropriate for a site-specific analysis. The DCFs are provided in Table 3-11. The source of the DCFs is Federal Guidance Report 11 (EPA, 1988), which provides the amount of radiological dose per unit of radioactivity ingested.

Table 3-9 Geochemical Parameters

Isotope	Source Kd Geometric Mean (mL/g)	Source Kd Geometric Standard Deviation	UZ & SZ Kd Geometric Mean (mL/g)	UZ & SZ Kd Geometric Standard Deviation	Thibault Geometric Mean for Sand (mL/g)
H3	0.000001	1.5	20	1.5	0.06
Be10	23	1.4	250	1.4	250
C14	5	1.4	20	1.4	5
Cl36	0.000001	1.4	0	1.4	
Ca41	5	1.4	5	1.4	5
Fe55	165	1.4	25	1.4	220
Ni59	1000	1.4	300	1.4	400
Co60	1000	1.4	1300	1.4	60
Ni63	1000	1.4	300	1.4	400
Sr90	15	1.9	20	1.9	15
Mo93	20	1.4	10	1.4	10
Nb94	500	1.4	550	1.4	160
Tc99	1	1.6	0.1	1.6	0.1
I129	4	2	5	2	1
Cs135	10	1.8	4600	1.8	280
Cs137	10	1.8	4600	1.8	280
Eu152	600	1.4	1	1.4	
Eu154	600	1.4	1	1.4	
Re187	100	1.4	100	1.4	
Pb210	600	1.4	1200	1.4	270
Bi207	100	1.4	100	1.4	100
Ra226	70	2.4	36000	2.4	500
Th230	500	1.5	3300	1.5	3200
Th232	500	1.5	3300	1.5	3200
U233	2200	2.4	1	2.4	35
U234	2200	2.4	1	2.4	35
U235	2200	2.4	1	2.4	35
U236	2200	2.4	1	2.4	35
U238	2200	2.4	1	2.4	35
Np237	50	1.5	25	1.5	5
Pu238	1000	1.4	1800	1.4	550
Pu239	1000	1.4	1800	1.4	550
Pu241	1000	1.4	1800	1.4	550
Pu242	1000	1.4	1800	1.4	550
Am241	600	1.4	9600	1.4	1900
Am243	600	1.4	9600	1.4	1900
Cm243	600	1.4	18000	1.4	4000
Cm244	600	1.4	18000	1.4	4000

Table 3-10 Solubility Limits# for the Disposal Zone

Isotope	Minimum (ml/g)	Maximum (ml/g)
H3		
Be10		
C14	7	70
Cl36		
Ca41	100	1000
Fe55	1E-3	60
Ni59	5E-3	20
Co60		
Ni63	5E-3	20
Sr90	9E-2	300
Mo93		
Nb94	50	100
Tc99		
I129		
Cs135		
Cs137		
Eu152		
Eu154		
Re187		
Pb210		
Bi207		
Ra226		
Th230	2E-5	2E-3
Th232	2E-5	2E-3
U233	2.4E-1	240
U234	2.4E-1	240
U235	2.4E-1	240
U236	2.4E-1	240
U238	2.4E-1	240
Np237	2.4	24
Pu238	2E-4	2E-1
Pu239	2E-4	2E-1
Pu241	2E-4	2E-1
Pu242	2E-4	2E-1
Am241	7E-3	7E-2
Am243	7E-3	7E-2
Cm243	7E-3	7E-2
Cm244	7E-3	7E-2

Log uniform distributions. If values are not given they are set very large. For normal ranges of distribution coefficients the solubility limits are a redundant geochemical barrier for many isotopes.

Table 3-11 Dose Conversion Factors for Ingestion

Isotope	DCF (mrem/Ci)
H3	6.40E+04
Be10	4.66E+06
C14	2.09E+06
Cl36	3.03E+06
Ca41	1.27E+06
Fe55	6.07E+05
Ni59	2.10E+05
Co60	2.69E+07
Ni63	5.77E+05
Sr90	1.43E+08
Mo93	1.35E+06
Nb94	7.14E+06
Tc99	1.46E+06
I129	2.76E+08
Cs135	7.07E+06
Cs137	5.00E+07
Eu152	6.48E+06
Eu154	9.55E+06
Re187	9.51E+03
Pb210	7.29E+09
Bi207	5.48E+06
Ra226	1.33E+09
Th230	5.48E+08
Th232	2.73E+09
U233	2.89E+08
U234	2.83E+08
U235	2.68E+08
U236	2.69E+08
U238	2.68E+08
Np237	4.44E+09
Pu238	3.20E+09
Pu239	3.54E+09
Pu241	6.85E+07
Pu242	3.36E+09
Am241	3.64E+09
Am243	3.63E+09
Cm243	2.51E+09
Cm244	2.02E+09

The other elements shown in Figure 3-10 are used to calculate the peak doses for each radionuclide (with time) and the overall total dose. Peak doses within 1,000 years and whether they exceed 25 mrem/yr are also tracked. Additional elements are discussed in the results section of this document.

Impacts to water were considered as part of the original assessment for development of the regulatory requirements for 10 CFR Part 61. Other scenarios such as erosional release were also considered. However, they were not carried through to development of the waste classification limits because the impacts from water-related pathways can be highly site-specific. The original assessment used analytical expressions to represent advection and dispersion of contaminants in an aquifer (NRC, 1981b). Modern tools allow for a much more complete assessment of contaminant fate and transport. The groundwater impacts assessment performed to support this report used the more modern tools. The ranges of key hydrologic parameters (i.e. thickness of the vadose zone, infiltration rate, groundwater flow rate) were selected to encompass a broad range of potential site conditions (e.g., shallow/deep, dry/wet, small dilution/large dilution). Although the geochemical parameter values assigned were reasonable for the type of generic screening calculations being performed, those parameters could have a much larger range of values than were assessed in this evaluation. The values selected are consistent with different soil types but did not attempt to encompass every possible observation, some of which represent extreme conditions. In addition, although a first-order range of performance (uncertainty) for the wasteform was provided through the bound waste release parameter, individual waste stream performance would be expected to be represented in much more detail for site-specific licensing decisions.

3.2.2 Other Technical Considerations

Many aspects of the problem of assessing impacts to an offsite individual can be strongly site-specific. Although impacts to water are usually the most important exposure pathway for offsite receptors, this is not always the case. For very arid sites with deep water tables, releases from the air pathway, biota, or disruptive events may be the limiting scenario. For complex waste streams such as GTCC waste streams resulting from remediation activities, the characteristics of the waste could be different from the characteristics of traditional Class A, B, and C wastes. These characteristics could make normal ranges for geochemical parameters inapplicable.

The assessment did not explicitly evaluate discrete disruptive events (e.g., earthquakes, volcanos, floods) because the assumption was made that the 10 CFR 61.50 disposal site suitability requirements would apply to GTCC waste streams. Because GTCC waste streams have higher concentrations of radionuclides than Class A, B, and C wastes, events of lower likelihood could be more significant to performance (risk or dose being the product of probability and consequences). Those requirements do not specify a probability threshold to limit the consideration of very unlikely events. However, those requirements do say that most types of disruptive events may be considered with respect to their impacts on the performance objectives:

(9) Areas must be avoided where tectonic processes such as faulting, folding, seismic activity, or vulcanism may occur with such frequency and extent to significantly affect the ability of the disposal site to meet the performance objectives of subpart C of this part, or may preclude defensible modeling and prediction of long-term impacts.

(10) Areas must be avoided where surface geologic processes such as mass wasting, erosion, slumping, landsliding, or weathering occur with such frequency and extent to significantly affect the ability of the disposal site to meet the performance objectives of subpart C of this part, or may preclude defensible modeling and prediction of long-term impacts.

The potential dose to an inadvertent intruder evaluated in the previous section is expected to bound the projected dose from a disruptive event because the assumed contact time and exposure route parameters (e.g., mass loading for inhalation) used in the inadvertent intruder analysis would yield greater doses than the values the NRC staff expects to be applicable to a disruptive event scenario. This conclusion is based on the intruder scenarios evaluated but would not be applicable if intruder scenarios involving very little disturbance were the primary intruder scenarios considered.

Special phenomena related to GTCC waste disposal could include but not be limited to heat generation, radiolytic gas generation, and criticality. These types of phenomena could impact the release rates from the disposal facility and therefore the impacts to an offsite member of the public or an inadvertent intruder. This analysis used a broad range of waste release rates to account for disposal systems where these processes were and were not significant. Because of the complexity associated with these phenomena they are not amenable to assessment in a generic evaluation but would need to be considered in a site-specific evaluation if the phenomena are not prohibited by waste characteristics requirements.

3.3 Offsite Individual - Operations

The original analysis for development of 10 CFR Part 61 considered two operational accident exposure scenarios: container rupture and disposal trench fire (NRC, 1981b). Those exposure scenarios were not used in establishing the waste classification tables. In general, the accident scenarios were much more limiting than the intruder scenarios (i.e., resulted in lower concentration limits) because the accident scenarios did not include the probability of the accident occurring and some of the parameters selected were very conservative. The overall philosophy for operational doses is to avoid accidents if possible and reduce the likelihood of them occurring through operational procedures and training. This section describes the models used to represent these two types of accidents to assess the potential operational impacts from GTCC waste disposal. The staff used models similar to the original assessment in the analysis to provide a direct comparison to the results from the original assessment.

3.3.1 Models and Data

The accident-container exposure scenario assumes a waste container is dropped from a height sufficient to result in a container break. A portion of the waste inside the container is released and is transported offsite in air where it can lead to human exposure. The release is modeled as a “puff” with short-term exposures. The form of the waste is a very important variable in this type of scenario. Waste that is in a solid form will have very little release even with a container breach whereas waste that is in a dispersible form would have a much larger fractional release. A primary difference between the operational exposure scenarios and the intruder and post-closure offsite exposure scenarios is that the operational scenarios are assumed to occur before significant radioactive decay can take place.

The accident-container scenario assumed a 1 second release with 1 breath of air taken by a member of the public as the contaminated puff passes the person. Longer release periods

would involve more breaths but at lower concentrations for the same quantity of material released. The IMPACTS code used a site selection factor, f_s, to account for the atmospheric dilution, dispersion, air intake, and exposure time associated with the container drop accident. The site selection factor was given by:

$$f_s = 1.56 \times 10^{-7}\, f_r\, V \left(\frac{X}{Q}\right)$$

Where the value 1.56×10^{-7} is the fraction of air inhaled during the release compared to the total annual air inhaled (1.25 L compared to 8000 m^3). The parameter f_r is the fraction released. It is fixed at 0.1% of the total waste volume, V (4.8 m^3 [170 ft^3]), for all types of waste. The Chi over Q factor (X/Q) is used to account for atmospheric dispersion. The value used[15] was 4.42×10^{-3} s/m^{-3}.[16] Exposures occur primarily from inhalation, though direct radiation from submersion in the puff release and deposition on the ground surface are also included. An additional reduction of 0.1 is applied to account for the waste being unsolidified but having moderate dispersibility. This factor combined with the 0.1% release fraction results in an overall reduction factor of 0.0001. The product of all of these factors produced a unitless site selection factor for the reference facility of 3.32×10^{-12}.

The accident-fire exposure scenario is calculated in a nearly identical way except an intruder is exposed to a fire in a disposal trench for 10 minutes instead of being exposed to a puff release for 1 second. The fire is assumed to take two hours to burn 100 m^3 of waste, or a portion of a trench. The release is assumed to be constant with a release rate of 1/7200 s^{-1}. The fire exposure scenario assumes a release fraction of 0.1 for the most flammable waste (i.e., the type of waste assumed in setting the waste classification limits), as compared to a release fraction of 0.001 in the container accident puff release. The receptor is assumed to be located 100 m from the fire and remain in the centerline of the plume for 10 minutes before relocating. The person is assumed to inhale 0.29 m^3 of air during that 10-minute period (compared to 0.00125 m^3 for the container drop). The report (NUREG/CR-1759 Vol.3) stated that X/Q was equal to $3.62 \times 10^{-3} s/m^3$, though that value could not be verified based on the stated inputs and equations. The unitless site selection factor for the accident-fire exposure scenario was 1.83×10^{-9}.

Calculation of the dose impacts, or derived concentration limits associated with the accident exposure scenarios, is straightforward and uses the same equations and approaches described earlier for the acute intruder.

3.3.2 Other Safety Considerations

Operational dose assessment, or an operational safety assessment, could consider a number of accident exposure scenarios and initiating event sequences. The approach taken in NUREG/CR-1759 Vol 3 (and replicated here) was to examine two representative types of accidents that may be reasonably bounding. The analysis does not take into account the probability of those accidents occurring.

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[15] The value used could not be verified based on the text provided in the report.

[16] For a puff release, it is common to use Q to represent the amount released rather than a release rate. In that case, the dimensions of *X*/Q are $1/m^3$ rather than s/m^3. Equation 3-30 in NUREG/CR-1759 Vol. 3 corresponds to that interpretation of *X*/Q. The units of *X*/Q reported in the accompanying text in that NUREG were adjusted to s/m^3 by dividing the mass of the puff release, Q, by the assumed 1-second puff duration.

The radionuclides in DOE's FEIS includes large quantities of special nuclear materials (see Appendix A for specific inventories) that could result in quantities in excess of the §70.24 limits. For example, U-235 and Pu-239 are fissile radionuclides in wastes resulting from Molybdenum-99 (Mo-99) production. Based on the activity values for U-235 and Pu-239 in DOE's FEIS, there is approximately 1,700 kg of U-235 and approximately 2 kg of Pu-239 for the Mo-99 waste stream. The amount of U-235 and the amount of Pu-239 in this waste stream is significant and highly concentrated relative to the 10 CFR 150.11 limits (e.g., one 0.2-m^3 [55-gallon] container would contain nearly 1 kg of U-235) that would exempt an agreement state from the critical mass limits currently set forth in 10 CFR 150.11. In addition, special moderators may exist in the GTCC waste, particularly in the sealed neutron sources that use beryllium to produce neutrons. Accidents involving potential criticality would need to be evaluated if GTCC waste that could result in a critical mass was disposed.

Criticality safety evaluation involves three major factors: (1) the quantity of the special nuclear material (SNM) and enrichment (U-235 only), (2) the concentration, and (3) the availability of moderators, particularly special moderators. When the disposal of GTCC and TRU waste contains a significant quantity of SNM, the disposal facility must be designed and constructed, and the SNM limited, such that it would be subcritical if the SNM and available moderators, including special moderators, were accumulated and concentrated by manmade or natural processes resulting in the maximum reactivity of the fissile material over the period of storage.

The GTCC waste may need to be handled remotely due to the high dose rate at the surface of the containers. This is especially true for activated metal waste, which has a high external dose rate similar to spent nuclear fuel (DOE 2016, page B-26). Part 61 requires applicants to provide information about the design features of the land disposal facility and the disposal units related to occupational exposures [§61.12(b)]. Part 61 also requires applicants to provide a description of the radiation safety program [§61.12(k)] that would include items such as the procedures, instrumentation, facilities, and equipment used to control occupational exposures and ensure compliance with 10 CFR Part 20 requirements. Accidents involving incomplete or improper shielding or failure of shielding may be more credible than a container drop or trench fire, but those types of accidents are significant to workers not to an offsite member of the public. Sky shine may need to be evaluated. Radiation safety programs are successfully implemented at commercial power reactors and independent spent fuel storage facilities that handle highly-radioactive material such as activated metal waste. Thus, it is reasonable to expect that a radiation safety program that would be protective of the workers can be implemented at a disposal facility.

4. Results

This section provides the results of NRC staff and contractor analyses to examine the potential impacts from the disposal of GTCC waste to inadvertent intruders and to offsite members of the public during operations and post-closure. The contractor analyses are documented in (Laplante, 2019) and (Wittmeyer, 2019). Only select results and inputs are discussed here. The reader is referred to the aforementioned references for the full details of the analyses.

Sets of projected doses to three different receptors are presented: an inadvertent intruder, an offsite member of the public after closure of the disposal facility (i.e., post-closure), and an offsite member of the public during operation of the disposal facility. Neither the intruder impacts nor the impacts to an offsite member of the public during operations are expected to occur. Both active and passive institutional controls are applied at low-level waste disposal facilities to reduce the likelihood of future usage or disturbance of the disposal site. During operations, active and passive management controls and safety systems are used to limit the likelihood of an accident that could impact an offsite member of the public.

4.1 Inadvertent Intruder

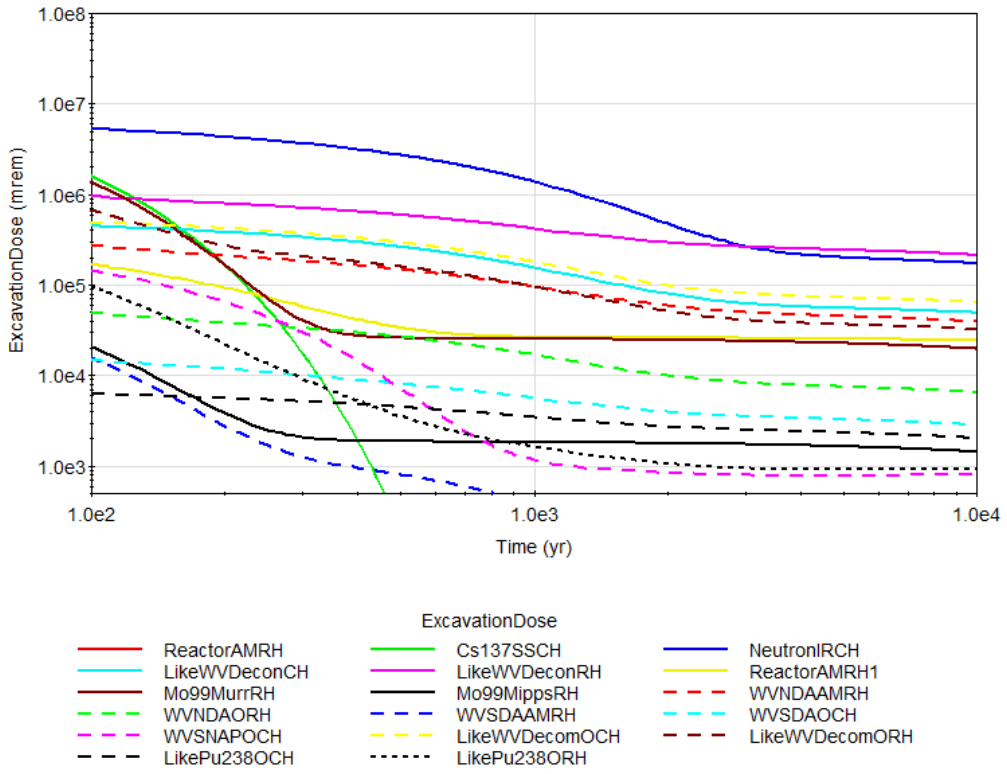
The inadvertent intruder exposure scenarios evaluated in this assessment considered the potential excavation for construction of a home on the disposal site after the institutional control period (100 years) or the potential installation of a well for domestic water usage. Projected doses to both acute and chronic receptors were evaluated for each exposure scenario (excavation or drilling).

4.1.1 NRC Staff Results

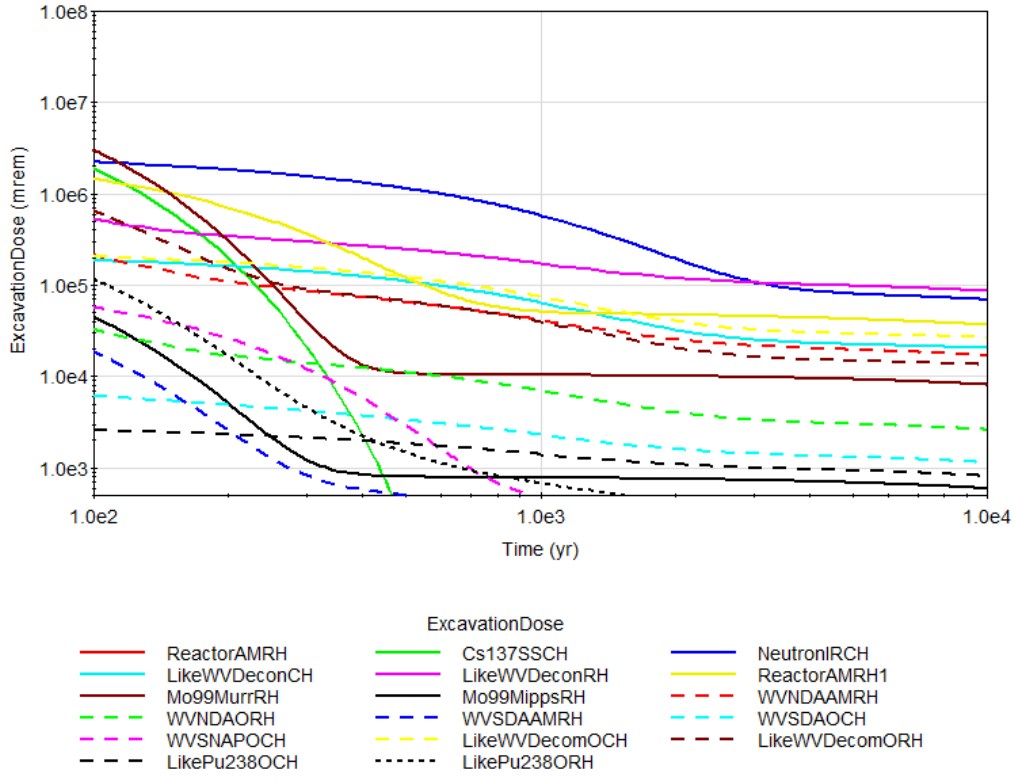
Figures 4-1(a) and 4-1(b) provide the overall dose results for acute and chronic exposures from the inadvertent intruder excavation scenario. Figures 4-2(a) and 4-2(b) provide the overall dose results for acute and chronic exposures from the inadvertent intruder drilling scenario. The excavation results show large impacts (> 10,000 mrem doses) persisting for more than 10,000 years for over half of the GTCC waste streams. Only two of the waste streams (Cs137SSCH, WVSDAAMRH) have doses less than 500 mrem within 10,000 years for the acute excavation scenario. Because the excavation doses are so large, they are not examined in greater detail.

The drilling exposure scenario doses show similar temporal profiles but much lower overall magnitudes compared to the excavation exposure scenario doses because of the much smaller amount of waste exhumed and the larger dilution factor. Only two of the waste streams (*sealed sources associated with neutron irradiators* [NeutronIRCH] and *remote-handled other waste from decontamination activities* at the WVDP [LikeWVDecomRH]) have acute drilling scenario doses larger than 500 mrem by 1,000 years.

Table 4-1 provides the doses at different times for the intruder drilling (acute) scenario. At 100 years over half the waste streams produce acute drilling scenario doses that are over 500 mrem. By 500 years only three waste streams exceed 500 mrem, and by 10,000 years no waste streams exceed 500 mrem. Also shown in the right-hand column is the fraction of the dose at 500 years that is coming from inhalation. All GTCC waste streams except the Cs-137 sealed sources and the reactor activated metals have high concentrations of isotopes that cause impacts predominantly through the inhalation pathway. Table 4-2 provides a comparison of the acute and chronic intruder drilling scenario doses at 500 years for each of the 17 waste

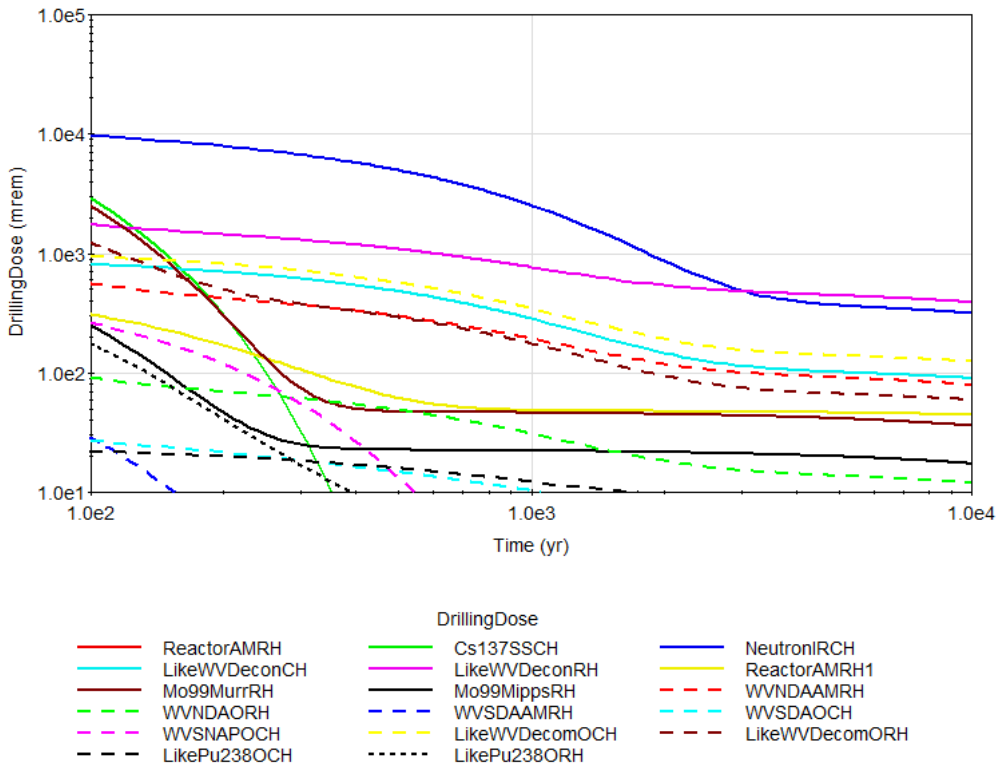


(a)

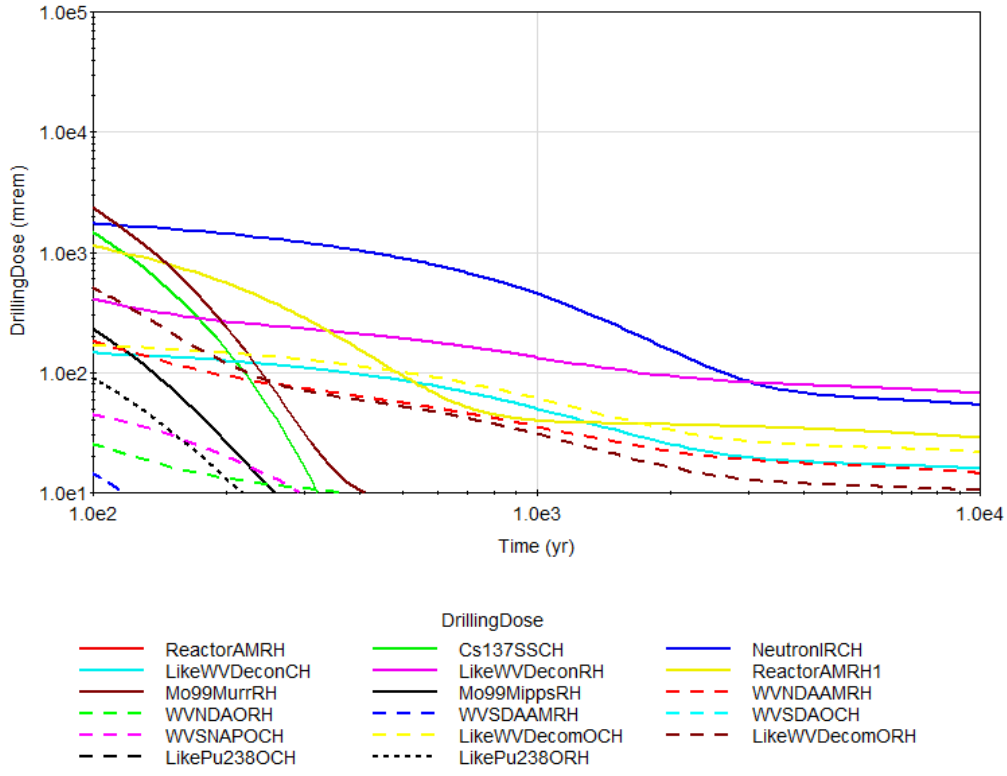


(b)

Figure 4-1 (a) Inadvertent Intruder Excavation (acute) and (b) Inadvertent Intruder Excavation (Chronic) Dose Time Histories



(a)



(b)

Figure 4-2 (a) Inadvertent Intruder Drilling (acute) and (b) Inadvertent Intruder Drilling (Chronic) Dose Time Histories

Table 4-1 Acute Intruder Doses for the Drilling Exposure Scenario

Waste Stream	Total – 100 yr	Total – 500 yr	Total – 10,000 yr	Inhalation fraction @ 500 yr
ReactorAMRH	307	61	44	0.07
Cs137SSCH	2,904	0	0	0.00
NeutronIRCH	9,648	4,957	314	0.96
LikeWVDeconCH	813	481	90	0.96
LikeWVDeconRH	1,736	1,082	388	0.97
ReactorAMRH1	307	61	44	0.07
Mo99MurrRH	2,494	47	36	0.99
Mo99MipsRH	251	23	17	0.99
WVNDAAMRH	554	298	79	0.96
WVNDAORH	91	49	12	0.97
WVSDAAMRH	29	1	0	0.94
WVSDAOCH	27	15	5	0.97
WVSNAPOCH	261	13	1	0.93
LikeWVDecomOCH	942	565	124	0.97
LikeWVDecomORH	1,217	291	59	0.96
LikePu238OCH	22	16	7	0.98
LikePu238ORH	176	6	2	0.95

Table 4-2 Comparison of the Acute and Chronic Doses for the Drilling Exposure Scenario (at 500 years)

Waste Stream	Acute (mrem)	Chronic (mrem)
ReactorAMRH	61	94
Cs137SSCH	<1	<1
NeutronIRCH	4,957	889
LikeWVDecomCH	481	85
LikeWVDecomRH	1,082	191
ReactorAMRH1	61	94
Mo99MurrRH	47	8
Mo99MipsRH	23	4
WVNDAAMRH	298	56
WVNDAORH	49	9
WVSDAAMRH	1	<1
WVSDAOCH	15	3
WVSNAPOCH	13	2
LikeWVDecomOCH	565	100
LikeWVDecomORH	291	52
LikePu238OCH	16	3
LikePu238ORH	6	1

streams. The acute drilling scenario impacts are larger than the chronic impacts for all but the reactor activated metals waste stream. Although the exposure time for the acute exposure scenario is much shorter than the exposure time for the chronic scenario, the concentrations are much higher with the overall result being the acute and chronic drilling scenario doses are of similar magnitude. As discussed in Section 3.1.2, the amount of material involved in the chronic drilling scenario is small such that the combination of concentrations and exposure times could not be supported as implemented in the original IMPACTS calculations for the excavation scenario.

The NRC staff also performed sensitivity analyses for the effects of modern dosimetry, the effects of the leachate-to-waste transfer factor in the plant root uptake pathways (see Section 3.1.1), and the effects of correcting a unit conversion error found in the plant ingestion pathway in the original IMPACTS analysis. Table 4-3 shows the effect on projected dose of using DCFs from Federal Guidance Report (FGR) 11 and 12 instead of the original ICRP-2 based DCFs used in the original IMPACTS analysis for the 17 waste streams considered in this analysis. Similarly, Table 4-4 shows the effect of using ICRP-72 based DCFs. Using more modern dosimetry resulted in decreases in the projected dose (shown are results for the acute drilling scenario).

The original IMPACTS analysis was completed in the early 1980's with the computer codes written in FORTRAN. When the NRC staff attempted to replicate the codes results using modern tools the staff identified unit conversion errors in the plant ingestion pathway calculation where some inputs were provided on a per day basis where other inputs were provided on a per year basis but the inputs were added without the proper unit conversion. Table 4-5 provides a comparison of the results for the different GTCC waste streams (showing the percent increase). The significance of the error is dependent on how much of the expected dose comes from the plant ingestion pathway, therefore the significance is also dependent on time. The results in Table 4-5 are for the chronic drilling scenario at 500 years. The impact on the commercial activated metals waste stream is an increase of over 100% because by 500 years most of the short-lived activity has decayed. Otherwise the increase in projected doses for all other waste streams is less than 10%.

4.1.2 Contractor Results

Staff from the Southwest Research Institute (SwRI), an NRC contractor, performed analyses similar to the NRC staff analyses to evaluate the impacts to inadvertent intruders from the disposal of GTCC waste. Whereas the NRC staff used a methodology similar to the original IMPACTS methodology, the contractor used the BDOSE code[17] to perform the analyses. The contractor documented these analyses in (Laplante, 2019) and (Wittmeyer, 2019). This section only provides a summary of the results and some key assumptions and observations.

The key assumptions applicable to the intruder analyses were:

- Institutional controls are assumed to be effective for 100 years, therefore, no intrusion dose calculations are done prior to 100 years.
- The waste inventory evolves over time due to radioactive decay and ingrowth, but losses are not considered.

[17] BDOSE reference

- All intercepted waste material is assumed to be uniformly available for exhumation to the surface at the specified inventory concentration with no credit for waste form resilience (e.g., expected to be conservative for metal or sealed source waste forms).
- Inhalation and ingestion dose calculations use ICRP 72 effective dose coefficients and external dose calculations use Federal Guidance No 12 effective dose equivalent dose coefficients.

Table 4-3 Impact of Using ICRP 26/30 (Percent Decrease in Doses)

Waste Stream	Total – 100 yr	Total – 500 yr	Total – 10,000 yr
ReactorAMRH	-99	-99	-99
Cs137SSCH	-43	-43	-43
NeutronIRCH	-46	-48	-28
LikeWVDecomCH	-46	-45	-30
LikeWVDecomRH	-42	-39	-26
ReactorAMRH1	-99	-99	-99
Mo99MurrRH	-43	-28	-27
Mo99MipsRH	-42	-26	-26
WVNDAAMRH	-45	-43	-29
WVNDAORH	-45	-43	-27
WVSDAAMRH	-43	-49	-36
WVSDAOCH	-38	-39	-28
WVSNAPOCH	-22	-25	-41
LikeWVDecomOCH	-45	-44	-30
LikeWVDecomORH	-44	-44	-30
LikePu238OCH	-39	-37	-27
LikePu238ORH	-37	-40	-49

Table 4-4 Impact of Using ICRP 72 (Percent Decrease in Doses)

Waste Stream	Total – 100 yr	Total – 500 yr	Total – 10,000 yr
ReactorAMRH	-99	-99	-99
Cs137SSCH	-43	-43	-43
NeutronIRCH	-55	-57	-27
LikeWVDecomCH	-54	-52	-30
LikeWVDecomRH	-49	-46	-30
ReactorAMRH1	-99	-99	-99
Mo99MurrRH	-43	-26	-26
Mo99MipsRH	-42	-26	-26
WVNDAAMRH	-51	-49	-28
WVNDAORH	-51	-49	-27
WVSDAAMRH	-44	-60	-59
WVSDAOCH	-43	-44	-26
WVSNAPOCH	-20	-24	-46
LikeWVDecomOCH	-53	-51	-30
LikeWVDecomORH	-47	-52	-29
LikePu238OCH	-44	-40	-26
LikePu238ORH	-37	-49	-70

Table 4-5 Impact of Unit Conversion Errors, Chronic Drilling Scenario Dose at 500 Years

Waste Stream	Dose (mrem)	% Increase Over Base
ReactorAMRH	194	106
Cs137SSCH	<1	0
NeutronIRCH	890	0.1
LikeWVDecomCH	86	0.1
LikeWVDecomRH	191	0.3
ReactorAMRH1	194	106
Mo99MurrRH	8	1.2
Mo99MipsRH	4	6
WVNDAAMRH	59	6.2
WVNDAORH	9	0.1
WVSDAAMRH	<1	7.4
WVSDAOCH	3	0.3
WVSNAPOCH	2	<0.1
LikeWVDecomOCH	100	0.1
LikeWVDecomORH	52	0.1
LikePu238OCH	3	<0.1
LikePu238ORH	1	0.1

The key assumptions specific to the drilling analyses were:

- The driller receptor constructs a residential water well and is exposed to drill cuttings containing exhumed waste for the duration of the well construction activity (25 hours).
- The residential water well is assumed to be 55 meters deep and the thickness of the waste is 0.5 meters.
- A waste package is assumed to be directly and completely intercepted by the drilling.
- All material exhumed by the well drilling activity is assumed to be uniformly mixed (in the drill cuttings pile) and spread over the soil at same concentration prior to dose calculations with no further dilution or loss of material.
- External dose calculations assume exposure to a 1 cm plane of exhumed material at the concentration of the uniformly mixed drill cuttings.
- The resuspension factor was 2.5E-9 1/m for acute exposures.

The key assumptions for the chronic intruder analyses were:

- The receptor is an individual who resides on top of a LLRW disposal cell after the disturbance activity has taken place.
- The chronic intruder receptor is assumed to have a garden to grow crops in soil that is not known to be contaminated with exhumed waste material.

- The garden crops include leafy green vegetables, root vegetables, fruits, and grains.
- All material exhumed by the well drilling activity is assumed to be uniformly mixed (in the drill cuttings pile) and spread over and plowed into the top 15 cm of soil prior to the assumed gardening activities.
- External dose calculations assume exposure to a 15 cm-thick plane of exhumed material at the concentration of the uniformly mixed drill cuttings that have been uniformly plowed into the top 15 cm of garden soil.
- Inhalation and external exposure durations for the chronic intruder (resident) are based on typical indoor and outdoor activity pattern assumptions and attenuation of both external radiation levels and air concentrations when the chronic intruder is indoors.
- The volume of air inhaled by the chronic intruder is determined by human activity-specific inhalation rates and the duration of indoor, outdoor, and offsite activities.

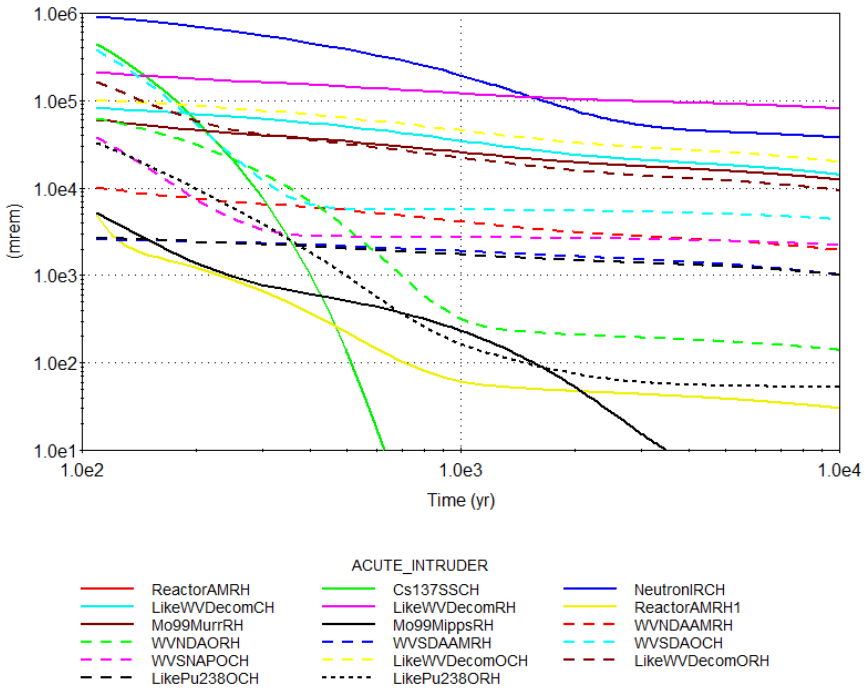
Figures 4-3(a) and 4-3(b) provide the overall dose results for the acute and chronic excavation scenarios. Figures 4-4(a) and 4-4(b) provide the overall dose results for the acute and chronic drilling scenarios. Similar to the NRC staff results, the excavation exposure scenario results show large impacts for most GTCC waste streams. The drilling scenarios results are much lower than the excavation scenario results because less material is exhumed and the dilution factor is larger. Table 4-6 provides a comparison of the acute and chronic drilling scenario results at 500 years after the end of the institutional control period.

4.1.3 Comparison and Discussion

The results from the intruder-excavation exposure scenario demonstrate that GTCC waste cannot be disposed shallowly (i.e. within 5 m of the land surface). Doses for some waste streams are on the order of 100,000 mrem well into the distant future (i.e. 10,000 years). This is because many GTCC waste streams have high concentrations of long-lived radionuclides.

The calculations for the excavation scenarios did not take credit for an intruder recognizing the waste form as anything other than soil. However, depending on the waste stream and the disposal site, the waste may degrade very slowly in the disposal facility. In that case, an intruder may recognize that the waste should not have been exhumed or that it requires investigation and may limit his exposure time. In the original analysis, that type of exposure scenario was referred to as a “discovery” scenario. A Discovery scenario was used in the development of the Class B limits to account for stabilization of Class B waste. Whereas the doses would be much lower than those shown in Figures 4-1(a), 4-1(b), 4-3(a), and 4-3(b), the doses would still exceed a 500 mrem performance objective for an inadvertent intruder for many waste streams because of the high radiation levels of those wastes.

The challenge in the generic analysis is that the full range of waste types and future disposal site conditions should be accounted for and the more limiting case is that the waste is no longer recognizable. Because of the persistence of the potential impacts and the uncertainty associated with future site conditions and human actions, it was reasonable to use the assumption that the waste was unrecognizable. Both the NRC staff and contractor results are



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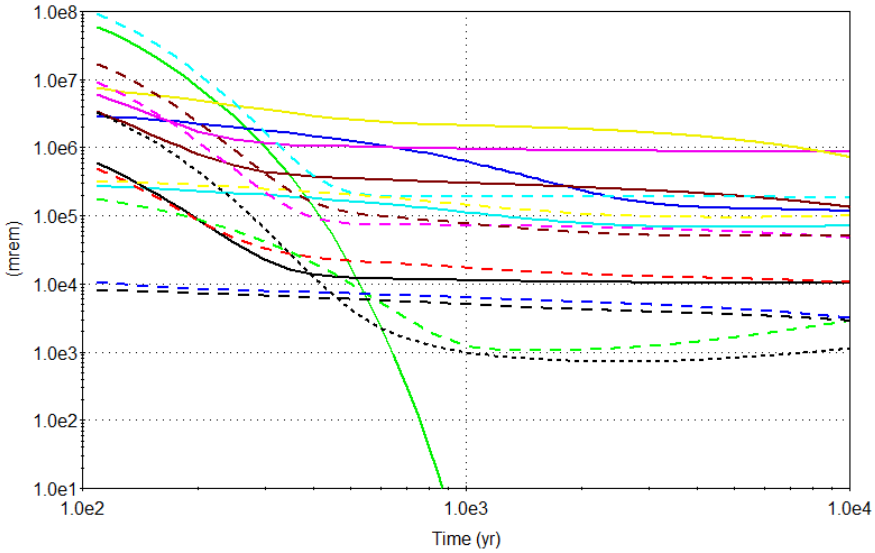

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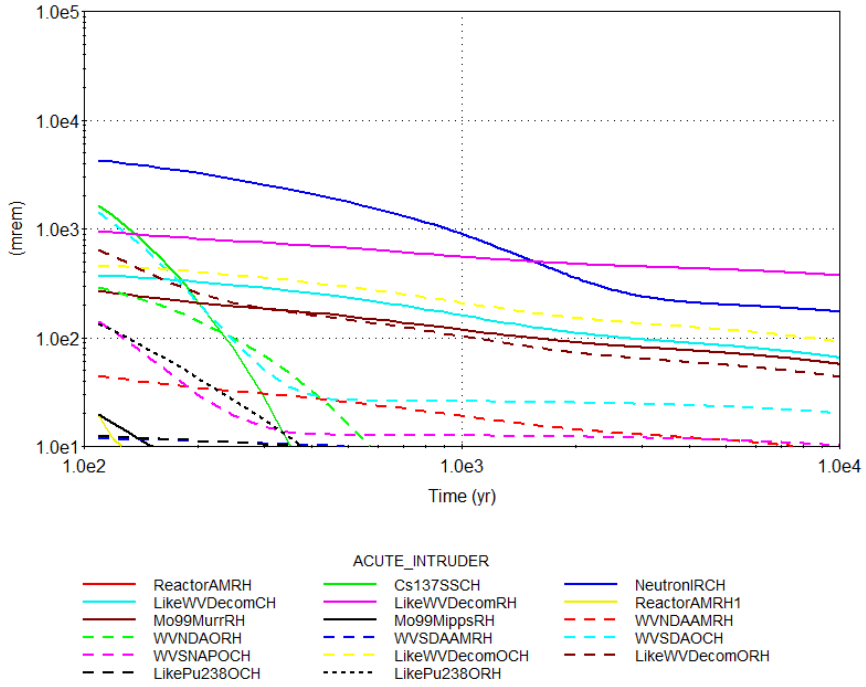
Figure 4-3 Inadvertent Intruder Excavation Scenario Dose Time Histories for Acute (a) and Chronic (b) Exposures



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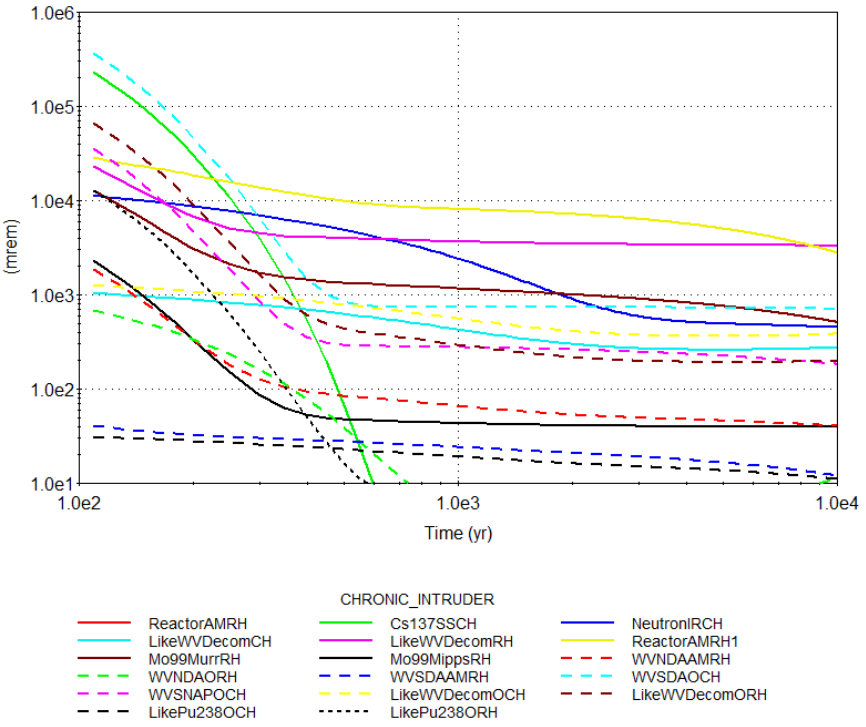

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Figure 4-4 Inadvertent Intruder Drilling Scenario Dose Time Histories for Acute (a) and Chronic (b) Exposures

Table 4-6 Comparison of the Contractor Results for the Acute and Chronic Doses (Drilling Exposure Scenario at 500 years)

Waste Stream	Acute (mrem)	Chronic (mrem)
ReactorAMRH	0.011	120
Cs137SSCH	0.52	72.3
NeutronIRCH	1650	4347
LikeWVDecomCH	217	557
LikeWVDecomRH	616	3477
ReactorAMRH1	0.011	120
Mo99MurrRH	24.8	756
Mo99MippsRH	11.8	278
WVNDAAMRH	146	1297
WVNDAORH	23.8	75.0
WVSDAAMRH	0.52	37.8
WVSDAOCH	10.1	27.3
WVSNAPOCH	17.6	40.1
LikeWVDecomOCH	272	695
LikeWVDecomORH	136	388
LikePu238OCH	8.7	20.3
LikePu238ORH	4.3	15.1

in agreement that intruder-excavation doses are likely to be unacceptably large for most GTCC waste streams over most timeframes.

The NRC staff and contractor results for the intruder drilling scenarios are in general agreement but with some key differences. The analyses were done with different tools by different groups. The groups did meet periodically to discuss progress and technical issues, but there was not an attempt to perform a rigorous calibration of the models. There was an attempt to align some key parameters associated with the intruder exposure scenarios (e.g., thickness of the waste, drill depth). Both the NRC staff and contractor results show that the doses for the acute drilling scenario are likely to be large if drilling were to occur shortly after a 100 year institutional control period ended. Use of a robust intruder barrier that would deter drilling for some time may allow much of the radioactivity to decay in place. It is anticipated that institutional controls will continue to be effective after 100 years but the effectiveness of passive institutional controls cannot be assured over very long periods of time.

Both the NRC staff and contractor results for the acute drilling scenario identified two waste streams, (*sealed sources associated with neutron irradiators* [NeutronIRCH] and *remote-handled other waste from decontamination activities* at the WVDP [LikeWVDecomRH]), that would be difficult to dispose in the near-surface. Because of the mechanical impacts of the drilling process, the driller may not identify that he has drilled through something dangerous. Based on the NRC staff results for the chronic drilling scenario, all GTCC waste streams except the NeutronIRCH sources would likely be acceptable for near-surface disposal. In the NRC staff analysis, the projected doses to an individual chronically exposed to drill cuttings were generally lower than the projected acute doses to a driller. This result was primarily due to two factors. First, the mass loading (amount of material that ends up in the air) was lower on an

annually averaged basis by approximately an order of magnitude. Most typical activities don't create as much disturbance as construction, and those activities that do are generally of very short duration. Second, as discussed in the previous section, the amount of material exhumed in the drilling process is only about 1 m^3, and a small fraction of that 1 m^3 is waste. The amount of material exhumed during drilling is not large enough to support someone being exposed for a year at the exhumed waste concentrations present during drilling (e.g., radioactive material suspended in the air will continue to be dispersed after drilling). The exposure duration is much larger in the chronic drilling scenario compared to the acute drilling scenario (6180 hours vs. 25 hours). The increased exposure time is offset by lower mass loading values and lower concentrations.

For the results being discussed and presented here, the assumed median thickness of the waste for the Intruder Drilling Exposure Scenario is 0.55 m, described in Figure 3-8, with a minimum and maximum distribution of 0.1 m and 1.0 m, respectively. This range of waste thickness is comparable to a single layer of waste packages. However, some of the waste disposal configurations discussed in the DOE's Final Environmental Impact Statement (DOE, 2016) involved trench disposal configurations of up to five layers of 0.2-m^3 (55-gallon) drums or standard waste boxes containing contact-handled waste. Waste thickness would then approximate 4 m and, assuming an emplacement efficiency of 50 percent, increase the amount of waste in 1 m^3 of exhumed material four-fold. Consequently, if GTCC waste is disposed in multiple layers of waste packages, the estimated doses can be expected to increase due to the additional amount of waste that would be brought to the surface by the drilling activity. In addition to the previously identified waste stream (i.e., NeutronIRCH and LikeWVDecomRH) additional waste streams could be more difficult to dispose of in the near surface, so that intended waste thicknesses would be an important design variable for any site-specific analysis. For example, a disposal configuration of two layers of drums containing either one of the two waste streams associated with the contact-handled other waste from the decontamination and decommissioning of West Valley's Main Plant Process Building (i.e., waste streams LikeWVDeconCH and LikeWVDecomOCH) would double the median thickness of the waste to approximately 1 m. Staff performed an additional sensitivity analysis to examine waste thickness. Figure 4-5 provides the variability in dose for one waste stream (LikeWVDeconRH) as a function of waste thickness. The waste thickness range was increased to go from 0.1 m to 5.5 m. Other variables are also sampled, but the overall impact is a strong correlation between waste thickness and acute intruder doses. A demonstration of the acceptability of near-surface disposal for some waste streams could be more difficult with thicker waste thicknesses. The waste thickness range used in the analyses was set to be what may be the minimum thickness that may be technically and economically practical. Waste thickness would likely be a design variable and can be easily controlled (above the minimum practical values).

The contractor results show higher doses for the chronic drilling scenario when compared to the acute drilling scenario. The primary reason is that the contractor model accounted for a reduction in concentrations the acute intruder is exposed to when the soil that is exhumed is placed in the environment before it could be resuspended which the NRC staff model did not. The NRC model accounted for a reduction for the chronic drilling scenario (because the amount of waste is limited and can be inconsistent with long-duration exposure over large areas) whereas the contractor model did not. Even with these differences the NRC staff and contractor results were in generally good agreement considering all the potential sources of uncertainty and different parameter values that could be selected.

The primary difference between the contractor and NRC staff results is that the contractor results for the chronic drilling scenario identified other waste streams (LikeWVDecomCH,

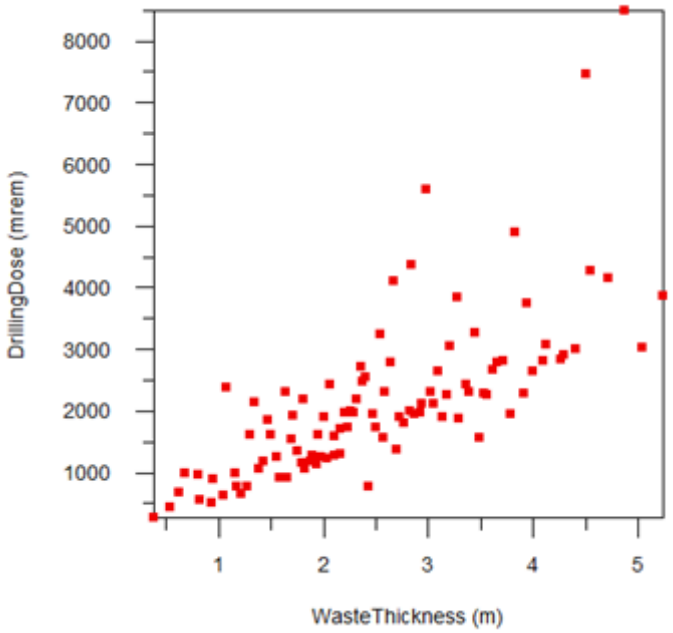


Figure 4-5 The Impact of Waste Thickness on Acute Drilling Scenario Doses (for LikeWVDeconRH)

Mo99MurrRH, WVNDAAMRH, LikeWVDecomOCH, LikeWVDecomORH) as potentially resulting in significant impacts to the intruder. The source of the difference is the plant ingestion pathway from the isotopes C-14 and Fe-55. The chronic exposure scenario adds a plant ingestion pathway that doesn't apply to the acute exposure pathway. This pathway can be significant depending on the assumptions about the form of the waste and availability of the waste for uptake of radioactivity by plants. The primary reason for a difference in the NRC staff and contractor results is that an additional factor was not applied by the contractor to take into account the availability of radioactivity in the waste for uptake by plants after the waste is exhumed by the intruder and mixed with soil in the near-surface environment. The IMPACTS assessment applied a factor for all waste (termed FMF) to account for the partitioning of waste to leachate (discussed previously). This reduction factor greatly impacted the estimated plant ingestion doses, especially for the aforementioned isotopes. Because most of the activity in activated metals is in stainless steel, it would be reasonable to account for the relative limited availability of radioactivity when embedded in steel for uptake by plants. However, it should be noted that this factor was applied to all forms of low-level waste in the IMPACTS analysis, which may be non-conservative for waste which is in more bio-available forms.

4.2 Offsite Individual – Post-Closure

NRC staff and the contractor assessed the potential impacts to an offsite receptor from the release of radioactivity from the disposal facility into water and the transport of radioactivity to a drinking water well where the contaminated water is extracted for domestic uses. The behaviors and characteristics of the potential receptors can be different and, in general, more varied for the offsite individuals compared to the inadvertent intruders. For this analysis, the offsite receptor is defined to be a member of the public living 100 m from the site boundary. The offsite receptor uses a well for drinking water and irrigation of a garden where he grows a portion of his annual vegetable intake.

4.2.1 NRC Staff Results

The simulation of potential radiological doses to an offsite receptor used a probabilistic model to directly assess the impacts of uncertainty and variability. When the original analysis was done for IMPACTS, the NRC staff recognized that groundwater impacts would be highly site-specific but the staff did not have the computational tools that are available today. Even for a particular site type (e.g., humid), the impacts would be highly variable as a result of the different disposal system designs, site hydrology and geochemistry, and other factors. The analysis performed for disposal of GTCC waste streams was a generic analysis of potential groundwater impacts across all types of disposal sites and environments. Figure 4-6 shows results for a single realization of the probabilistic model to illustrate how radioactivity is released and transported through the modeled domain. It can be difficult to evaluate the doses to an offsite individual with a generic (i.e. not site-specific) analysis because the magnitude of doses, their timing, and the key radionuclides are strongly influenced by the site-specific hydrogeological conditions, especially the site mineralogy and geochemistry (which controls transport times), the infiltration rate (i.e., the rate at which precipitation enters the subsurface), and the properties of the aquifer (i.e., how much dilution and dispersion there is during transport).

Hydrogeologic processes can affect the movement of radionuclides through air, water, and soil pathways. As previously mentioned, releases through the water pathways tend to be the predominant release from land disposal facilities and differences in the flow velocity profile of groundwater at various spatial scales will affect the rate of the radionuclide release. In addition, various physical mechanisms can influence the rate of release including advection, whereby the transport of radionuclides occurs with the general movement of water, and dispersion, whereby different flow paths of the water cause mixing and dilution of the radionuclides, and diffusion, whereby radionuclides move from areas of high concentrations to areas of low contaminant concentration.

Geochemical processes include physical-chemical interactions between liquid, solid, and gas phases within the disposal site and surrounding environment, and can include dissolution-precipitation, sorption-desorption, oxidative-reductive, and gas-solution interactions (NRC, 2015). A radionuclide's rate of movement out of the wasteform and through the surrounding environment is strongly dependent on its phase (solid, liquid, or gas), which is influenced by geochemical processes. Distribution coefficients, also called the solid-water partition coefficient, represent the mass of chemical absorbed per mass of soil compared to the mass of the chemical dissolved in a volume of water. Distribution coefficients are strongly influenced by the acidity of the water, among other chemical properties (e.g., carbonate concentration, redox potential). The capacity of hydrogeological units to adsorb radionuclides moving with the groundwater can have a significant effect on how quickly contaminants can reach areas beyond the disposal facility.

Releases are assumed to begin at the end of the institutional control period, for instance to represent the end of active maintenance of an engineered cover. The delay in releases can be assigned different values but unless the value is very large it is not a significant parameter in the analyses. Radioactivity is released from the wasteform and is gradually depleted as it is released. The radioactivity is then dispersed and delayed as it is transported through the geologic system. Figure 4-6 shows mass fluxes for Tc-99, which generally is subject to very little sorption in geologic systems. It may appear that the saturated zone doesn't have a very large impact on the results. That is because transport times through 100 m of aquifer material are generally much faster than transport through even tens of meters of vadose zone, especially for a radionuclide like Tc-99 that is lightly sorbed relative to other radionuclides (e.g., isotopes of

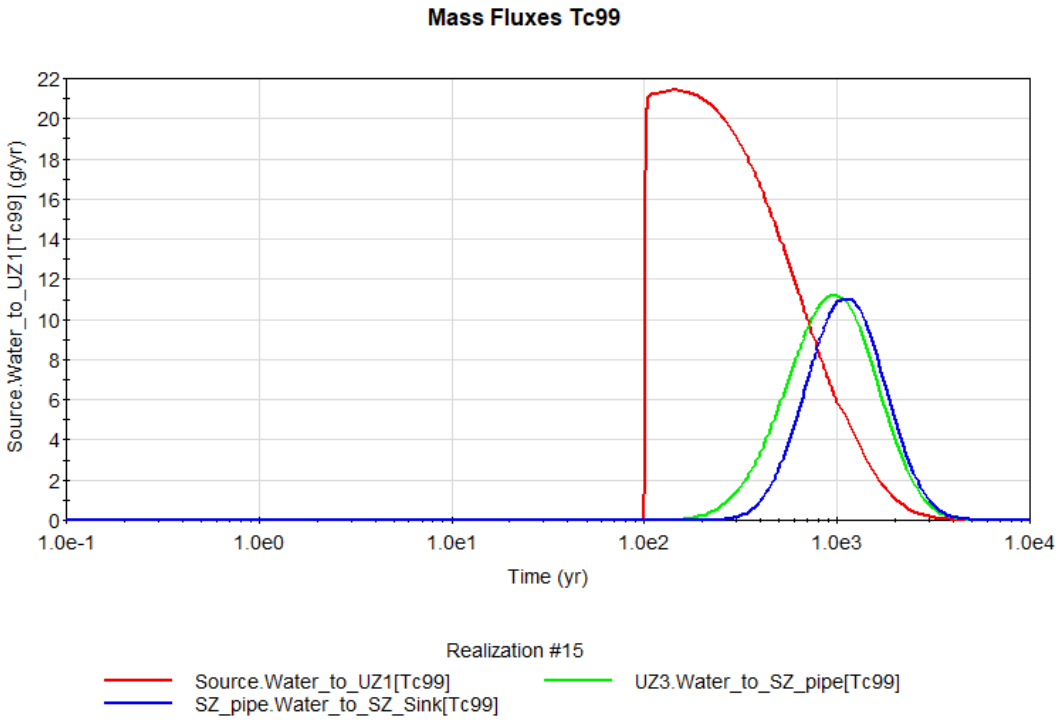


Figure 4-6 Illustrative Example of Transport Through the Modeled Domain

Pu). However, the aquifer, depending on the flow rates, can be a source of significant dilution of concentrations which results in a direct reduction of radiological doses. In the example, the aquifer doesn't significantly affect the timing of when a radionuclide such as Tc-99 arrives but it can significantly impact the concentrations.

Figure 4-7 is an example horsetail plot of the probabilistic output for one waste stream. Each line on the graph is a single realization of the probabilistic simulation. A realization is one set of sampled values of the uncertainty parameters. When plotted together the set of curves can appear similar to a horse's tail. For the ranges of parameters considered, the range in peak dose spans approximately six orders of magnitude. This large range in output is being driven by the ranges in hydrologic and geochemical parameters. The chart illustrates the importance of good site selection.

Figure 4-8 is a multivariate comparison of the simulated peak dose and the log base 10 of the ratio of infiltration rate to groundwater flow rate. A multivariate chart can illustrate the dependence of an output on an input. This ratio represents an effective dilution factor when contaminants flow from the waste through the vadose zone and that flux of radionuclides enters an underlying aquifer. The largest peak doses are associated with sites with relatively high infiltration rates and relatively low aquifer flow rates. Conversely the smallest peak doses are associated with low infiltration rates and high aquifer flow rates (large dilution factors). This was the most important parameter (combination of parameters) in the generic assessment.

Figure 4-9 is a plot of a single realization of one waste stream (LikeWVDecomRH) to illustrate the contribution of different radionuclides to the dose. Because the different radionuclides can have considerably different transport characteristics, their arrival times at the withdrawal well

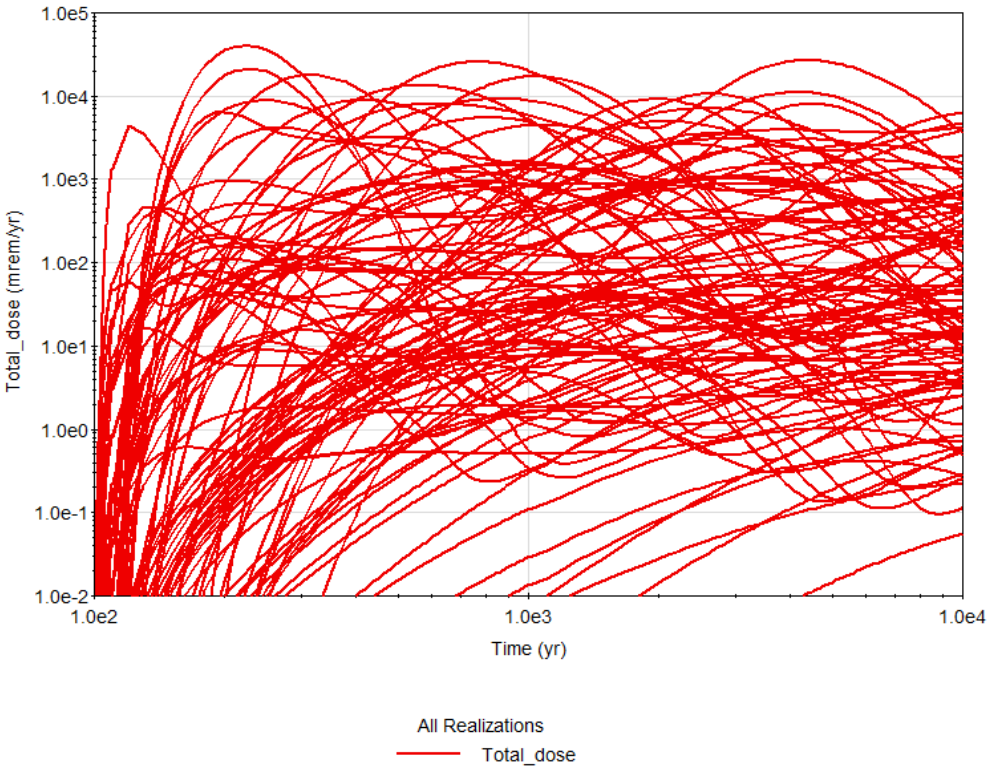


Figure 4-7 Example Horsetail Plot Shows Output of the Probabilistic Simulation

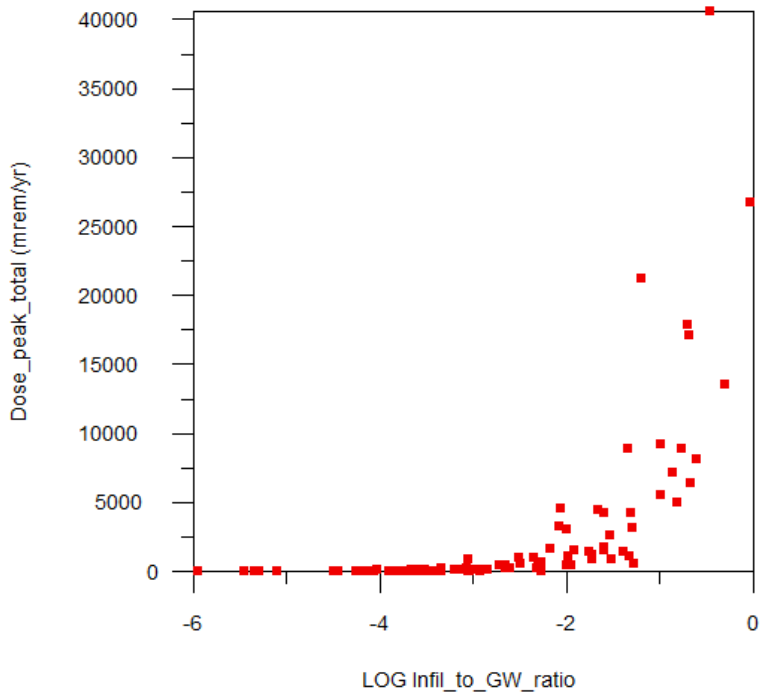


Figure 4-8 Multivariate Chart of the Impact of the Aquifer Dilution Factor on Peak Dose

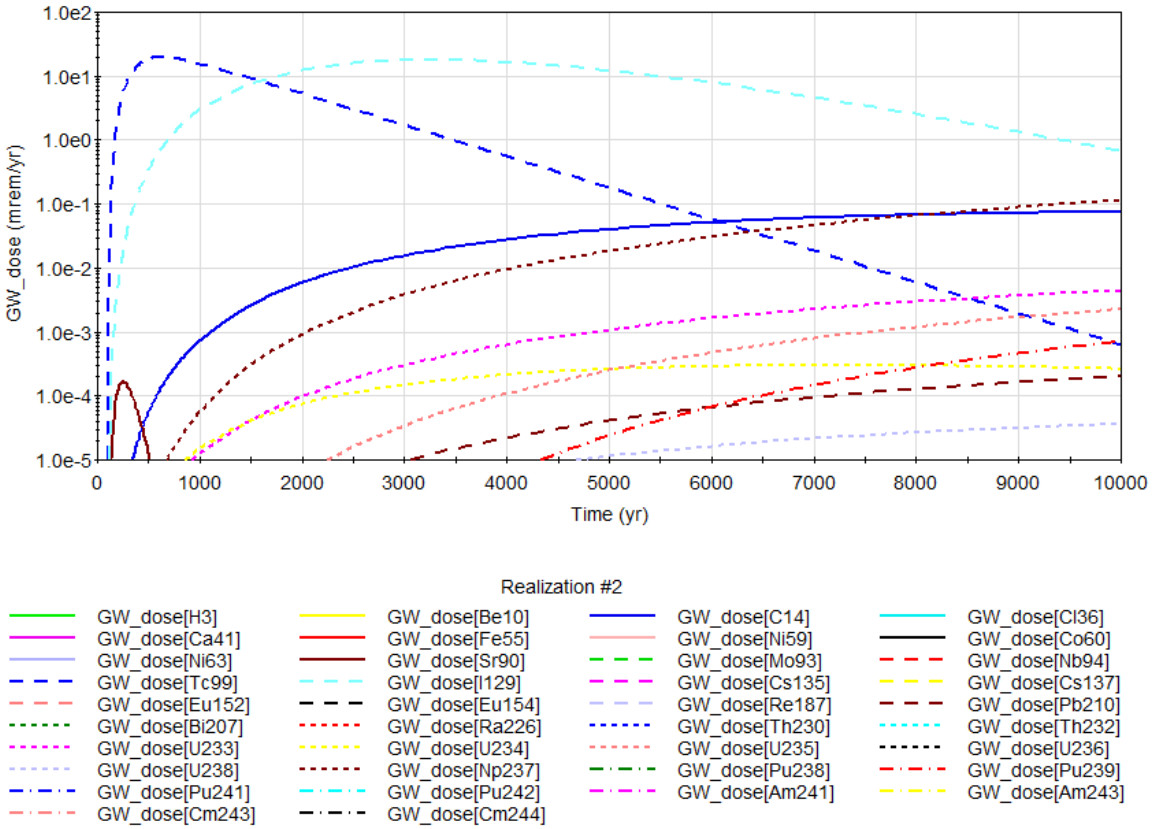
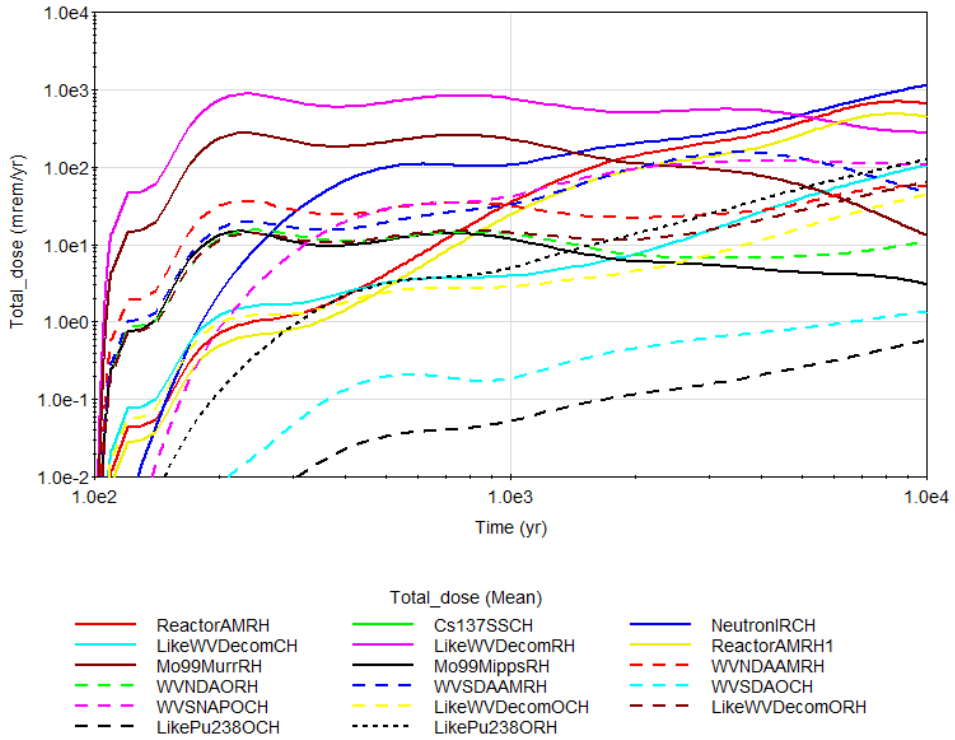


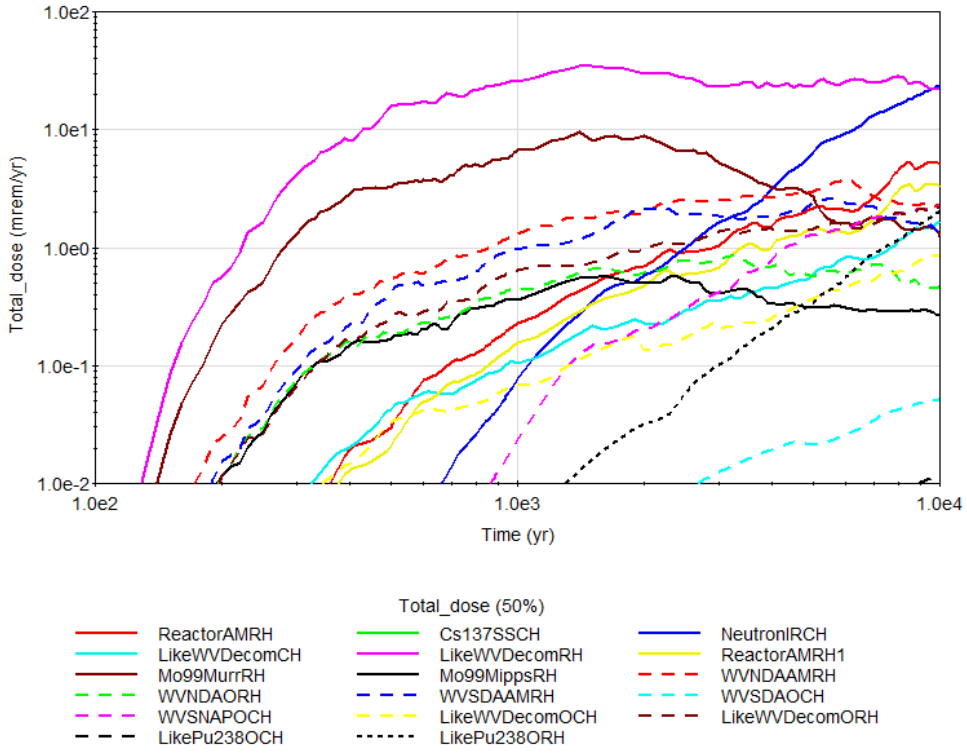
Figure 4-9 Example Output Showing the Contribution of Different Radionuclides for One Realization of One Waste Stream

can vary significantly. The output metric commonly used to evaluate the impacts is the peak of the mean dose curve. The peak of the mean dose curve is useful because it accounts for the variability in the different contaminant arrival times and the relatively short intervals a single receptor can be present during the very long time histories. However, a problem with using the peak of the mean metric in an analysis such as this is that site variability (e.g. characteristics of different sites) has been represented by the broad ranges in parameter values assigned. Those broad ranges can overstate the variability at any single site, leading to a phenomenon called "risk dilution." Risk dilution occurs when the timing of projected dose peaks is sufficiently variable such that there is little overlap in the projected dose peaks, leading to a peak of the mean result that is lower than the peak in any single dose projection. If risk dilution occurs, the mean result isn't meaningful by itself. Instead, the full range of outputs (represented by different percentiles) must be examined, though the peaks in those percentiles are useful to consider.

Figure 4-10(a) is a comparison of the mean result from a probabilistic simulation for each different waste type and Figure 4-10(b) is the 50th percentile result for each different waste type. Each probabilistic simulation for each waste stream used 100 realizations. The range in peak doses span roughly three orders of magnitude. The inventory of radioactivity associated with different waste types is an important source of variability in the results. However, the variability resulting from hydrogeological parameters is larger. When developing the waste classification system of 10 CFR Part 61, the staff applied concentration limits to protect the inadvertent intruder. The staff determined it was not practical to develop generic waste concentration limits that would ensure protection of an offsite member of the public. A licensee or applicant must



(a)



(b)

Figure 4-10 (a) Mean Dose Result to an Offsite Member of the Public from Each GTCC Waste Stream and (b) 50th Percentile Dose Result to an Offsite Member of the Public

demonstrate the protection of an offsite member of the public by performing site-specific pathways analysis.

For GTCC waste disposal, if a dose limit of 25 mrem total effective dose equivalent (TEDE) is used and based on the mean result for each waste stream shown in Figure 4-10, approximately one third of the waste streams would meet the criteria, approximately one third would be above the criteria but not by a significant margin, and the remainder would have mean peak doses exceeding hundreds of mrem. By comparison, at the 50 percentile result all but one waste stream would meet 25 mrem per year TEDE. These results highlight the importance of site-specific analysis to demonstrate compliance with 61.41. The key radionuclides were Tc-99, I-129, C-14, and Np-237. At longer timeframes, the isotopes of uranium contributed to the results. From the perspective of radiological doses to an offsite member of the public, GTCC waste disposal in the near surface is appropriate when done in disposal sites with favorable conditions. GTCC waste disposal isn't appropriate in just any site and likewise it isn't appropriate in all sites.

4.2.2 Contractor Results

Staff from the Southwest Research Institute, an NRC contractor, performed analyses similar to the NRC staff analyses to evaluate the impacts to offsite members of the public from the disposal of GTCC waste. The contractor staff completed technical analyses to improve NRC's understanding of how differences in disposal design and configuration as well as site hydrogeologic conditions may affect potential groundwater pathway doses from the near-surface disposal of GTCC waste. The contractor documented those analyses in (Wittmeyer, 2019). This section only provides a summary of the results and some key assumptions and observations.

Contractor staff developed a probabilistic groundwater dose assessment model to examine the potential impacts from disposal of 17 GTCC waste streams. The model was created in GoldSim Version 12.0 and up to six different disposal scenarios were evaluated. The six scenarios included:

1) Near-surface trench in an arid to semi-arid location with depth to water of 135 to 200 m,
2) Near-surface trench in a semi-arid to sub-humid location with depth to water of 50 to 135 m,
3) Near-surface trench in a sub-humid to humid location with depth to water of 20 to 40 m,
4) Moderate depth disposal unit located in an arid to semi-arid location where the depth to water is 135 to 200 m,
5) Moderate depth disposal unit located in a semi-arid to sub-humid location where the depth to water is 50 to 135 m, and
6) Disposal in a deep borehole in an arid to semi-arid location where the depth to the water table is 135 to 200 m.

The scenarios evaluated the main variables of disposal environment, disposal depth, and depth to water table. The main parameter uncertainties considered were for the infiltration rate, depth to water table, waste container failure time, waste form dissolution rate, and distribution coefficients. There were 33 different radionuclides considered (including decay chains). In general, the radionuclides that cause groundwater impacts from disposal of low-level waste are

well-known and simulation of a much larger set of radionuclides does not significantly enhance understanding but does increase computational burden.

The disposal system was envisioned to contain concrete, which would modify the disposal cell conditions for release and degradation of metal components. The form of each of the 17 GTCC waste streams was considered and the container failure model and waste form release models were parameterized accordingly (Wittmeyer, 2019). Infiltration rates ranged from a low of 0.1 mm/yr for an arid western site to a high of 200 mm/yr for a humid eastern site.

Table 4-7 provides the projected peak of the mean annual dose for each of the 17 GTCC waste streams for each of the six scenarios analyzed. The results show a strong dependence on the scenario and the waste stream type. The scenario encompasses different disposal system designs and site conditions. Some waste streams are more sensitive to the scenario (LikeWVDeconRH) than others (ReactorAMRH). Waste streams associated with decommissioning of the West Valley site are most limiting. For scenario 3, disposal of 8 of the waste streams resulted in a peak dose above 25 mrem. By comparison, for scenarios 1, 4, and 6, disposal of all of the waste streams resulted in a peak dose below 25 mrem.

For disposal scenarios with long hydrologic travel times, the most important radionuclides were C-14, I-129, and Np-237. For disposal scenarios with relatively rapid hydrologic travel times, various isotopes of plutonium and uranium were important. The contractor generated many illustrative figures to explain the simulation results. Figure 4-11 provides the peak of the mean dose occurring before 10,000 years as a function of disposal technology (trench, moderate depth disposal, deep borehole) and the site conditions (infiltration rate range, water table depth). For a poorly-performing site the impacts can be quite high (> 1,000 mrem/yr) whereas for a well-performing site the impacts are very low (< 1 mrem/yr). For a well-performing site the impacts would meet the established limits for doses to an offsite member of the public for all GTCC waste streams. Whereas for a poorly-performing site only select waste streams would likely meet established dose limits. Figure 4-12 provides the mean dose curve for all 17 waste streams for trench disposal with a site with moderate infiltration and moderate depth to the water table. The median dose curves were less than the mean dose curves but not by the large amount seen in the NRC staff analyses because site condition variability was more limited when a particular site type was being evaluated as in Figure 4-12 compared to the NRC staff analyses, which included overall site variability in one analysis.

4.2.3 Comparison and Discussion

Although the NRC staff and contractors developed model inputs independently, the results agree reasonably well. Both analyses demonstrate the importance of site-specific conditions in the determination of potential dose impacts to an offsite member of the public. The most limiting waste stream appears to be the LikeWVDeconRH waste stream. The radiological constituents and the form and characteristics of different waste streams can be very important to determining the radiological doses to offsite members of the public. Differences in the results between particular waste streams can be attributed to the assignment of different release parameters and different parameter values defining geochemical conditions for release and transport. Some geochemical parameters have large observed ranges in their values.

Table 4-7 Peak of the Mean Doses to the Offsite Receptor from Releases to Groundwater for Different Scenarios (Contractor Results)

Waste Stream	#1	#2	#3	#4	#5	#6
ReactorAMRH	7.8	54	62	6.8	32	0.00
Cs137SSCH	0.00	0.00	0.0	0.00	0.00	0.00
NeutronIRCH	0.02	20	390	0.01	8.8	0.00
LikeWVDeconCH	0.16	14	130	0.15	4.7	0.00
LikeWVDeconRH	4.7	140	1,300	3.8	58	0.00
ReactorAMRH1	3.0	22	26	2.5	13	0.00
Mo99MurrRH	0.02	0.13	2.5	0.01	0.11	0.00
Mo99MipsRH	0.39	4.5	8.9	0.33	3.2	0.00
WVNDAAMRH	0.16	1.2	1.4	0.13	0.71	0.00
WVNDAORH	0.12	3.3	43	0.11	2.0	0.00
WVSDAAMRH	0.25	1.4	1.5	0.20	0.96	0.00
WVSDAOCH	0.02	0.16	3.0	0.01	0.12	0.00
WVSNAPOCH	0.00	1.0	7.7	0.00	0.43	0.00
LikeWVDecomOCH	0.03	3.9	59	0.02	1.1	0.00
LikeWVDecomORH	0.04	6.4	110	0.03	2.0	0.00
LikePu238OCH	0.00	0.00	0.89	0.00	0.00	0.00
LikePu238ORH	0.00	1.8	11	0.00	0.61	0.00

There are numerous uncertainties that can impact offsite dose calculations. Some of these uncertainties could include, but are not limited to:

- Waste characteristics impacts on release and transport such as the presence of organic substances (e.g. chelating agents),
- Biotic transport of waste,
- Natural disruptive processes or events,
- Long-term climate change,
- Discrete failure modes of disposal containers or disposal cells (e.g., bath tub release),
- Impacts of radiation on release rates and material properties,
- Impacts of heat-generating waste on release rates and material properties,
- Complex hydrogeology (e.g., fast pathways or focused flow) in the vadose zone, and
- Atypical exposure scenarios or biosphere characteristics.

While every attempt was made to account for the significant sources of uncertainty that generally apply to most offsite receptor dose assessments, many uncertainties that could apply on a site-specific basis weren't evaluated in the NRC staff and contractor analyses. The analyses to support licensing of a disposal facility would be more comprehensive than the analyses documented in this report and would generally employ a model development step to determine the features, events, and processes that are relevant at a site. It is not practical to perform a generic offsite dose assessment that includes all features, events, and processes that could be relevant at any site.

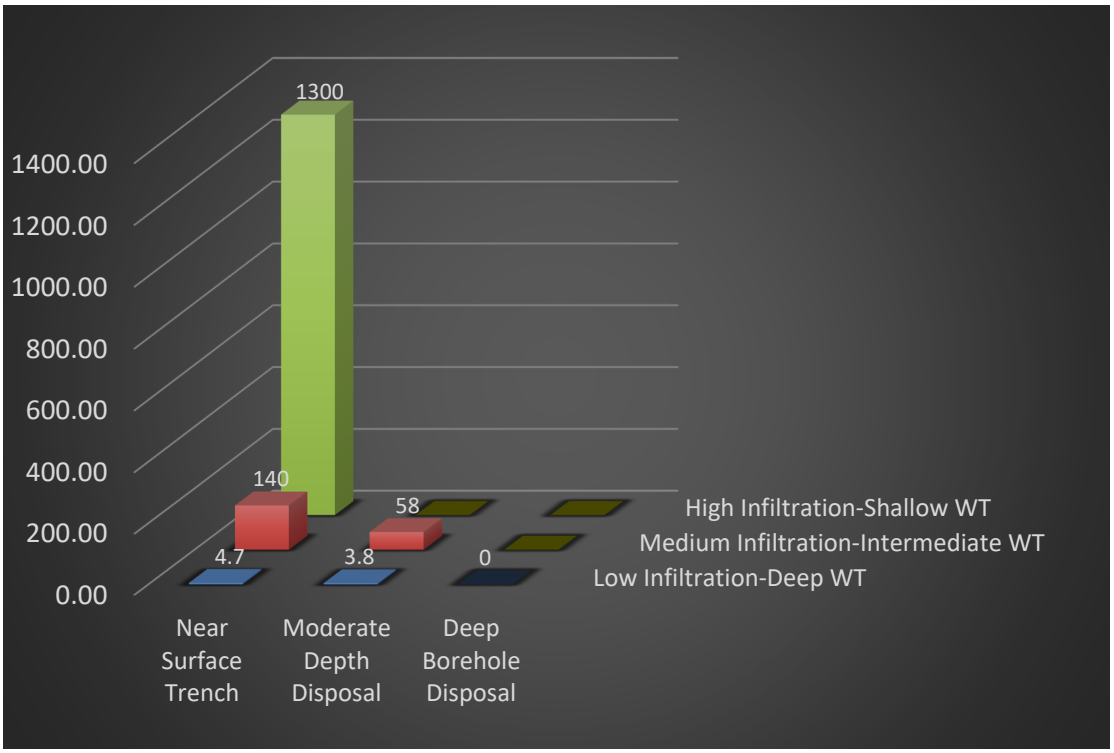


Figure 4-11 Peak Mean Dose for the LikeWVDeconRH Waste Stream (values shown are in mrem/yr)

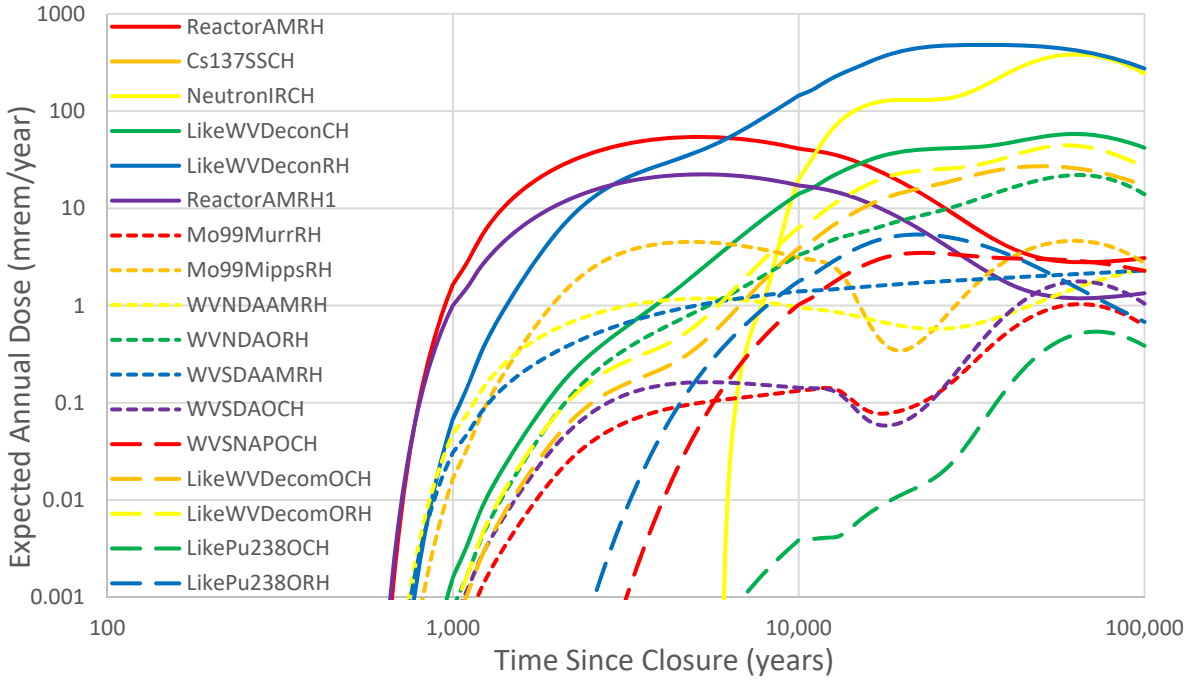


Figure 4-12 Peak Mean Dose for the Offsite Receptor for All 17 GTCC Waste Streams (Trench disposal, moderate infiltration rates, moderate water table depth)

4.3 Offsite Individual - Operations

The results of the two operational accidents evaluated are summarized in the sections that follow. As previously discussed, the original analysis for development of 10 CFR Part 61 considered two operational accident exposure scenarios: Container rupture and disposal trench fire. Those exposure scenarios were not used in establishing the waste classification tables. In general, the accident scenarios were much more limiting than the intruder exposure scenarios (i.e. they resulted in lower concentration limits) because the accident scenarios did not include the probability of the accident occurring and some of the parameters selected were very conservative. The NRC staff and contractor results are presented below.

4.3.1 NRC Staff Results

Release of radioactivity resulting from the drop of a container and a trench fire were the two scenarios analyzed by the NRC staff. Table 4-8 provides the potential dose to an offsite member of the public located near the boundary of the disposal facility (nominally 100 m from the dropped container or burning trench). The doses shown are the results for the average waste package of each GTCC waste stream.

The doses are quite large because in the original IMPACTS analysis the waste was assumed to be dispersible and, for the trench fire, flammable. For the container drop exposure scenario 0.1% of the material in the container was assumed to be released into the air following the drop. For a 0.2-m^3 (55-gallon) drum and a metallic waste with a density of approximately 3000 kg/m^3, this would result in 600 grams of material being released into the air, which could then be inhaled by a downwind receptor. This amount of material being dispersed for most forms of GTCC waste is likely to be unrealistic. The IMPACTS code can account for different forms of waste and different disposal facility or management characteristics. For example, by changing the ACC parameter from a default value of 1 (accessible) to a value of 3 (accessible with difficulty), an extra reduction factor of 0.01 is applied to the accident releases. When waste is switched from being flammable (FLAM=3) to non-flammable (FLAM=0) the factor that accounts for releases to the air during the fire changes from 1.83E-11 to 2.29E-15. Because GTCC waste is not anticipated to be highly dispersible or flammable, alternate cases were analyzed and the results are shown in Table 4-8. Instead of doses from a trench fire of thousands of rem, the doses are now on the order of single rem which is relative agreement with the results from DOE. The IMPACTS code applies the same magnitude reduction to all waste types when the waste parameter settings are changed.

Table 4-8 Doses for an Offsite Receptor from Container Drop and Trench Fire Accident Exposure Scenarios (NRC Results in Rem)

Waste Stream	Container Drop - Dispersible	Container Drop - Low Dispersibility	Trench Fire - Flammable	Trench Fire - non-Flammable
ReactorAMRH	1.52E+02	1.52E-02	2.77E+04	3.47E+00
Cs137SSCH	3.58E-01	3.58E-05	2.42E+01	3.03E-03
NeutronIRCH	2.95E+02	2.95E-02	1.63E+04	2.03E+00
LikeWVDeconCH	1.96E+01	1.97E-03	1.09E+03	1.36E-01
LikeWVDeconRH	4.18E+01	4.18E-03	2.39E+03	2.98E-01
ReactorAMRH1	1.52E+02	1.52E-02	2.77E+04	3.47E+00
Mo99MurrRH	4.14E+01	4.14E-03	2.34E+03	2.92E-01
Mo99MipsRH	4.46E+00	4.46E-04	2.54E+02	3.18E-02
WVNDAAMRH	1.18E+01	1.18E-03	6.65E+02	8.31E-02
WVNDAORH	1.89E+00	1.89E-04	1.09E+02	1.36E-02
WVSDAAMRH	9.81E+00	9.81E-04	1.82E+03	2.27E-01
WVSDAOCH	2.04E-01	2.04E-05	4.25E+01	5.32E-03
WVSNAPOCH	2.05E+01	2.05E-03	1.30E+03	1.63E-01
LikeWVDecomOCH	2.22E+01	2.22E-03	1.23E+03	1.54E-01
LikeWVDecomORH	1.38E+01	1.38E-03	7.68E+02	9.60E-02
LikePu238OCH	4.78E-01	4.78E-05	2.64E+01	3.30E-03
LikePu238ORH	4.71E+00	4.71E-04	3.03E+02	3.79E-02

4.3.2 Contractor Results

The contractor analyzed the potential impacts resulting from a fire that releases some of the waste to the air (Laplante, 2019). The fraction of the contents released (0.1%) is assumed to be uniformly dispersed over the 30-minute duration of the fire. A member of the public is located 100 m downwind from the location of the fire. The downwind concentration is estimated based on applying a X/Q value from a Gaussian plume calculation that involves conservative dispersion assumptions (NRC, 1988). The X/Q value is based on NUREG-1140 and is used by DOE to estimate accident consequences from DOE facilities. The public receptor is assumed to be exposed to the full duration of the plume passage (30 minutes).

The calculation is a conservative screening style calculation to gain insights into the level of hazard presented by the materials in different waste streams. Key assumptions for the hypothetical accident are:

- Waste material is assumed to be of uniform concentration and readily dispersible (no consideration of waste form in mitigating release of the portion of the waste released by the fire (0.1%)).
- The DOE/NRC X/Q value (3.5E-3 s/m^3) for dispersion modeling conservatively assumes F stability class, 1 m/s wind speed, flat terrain, and a continuous 30 minute release.

- Building wake effect is assumed (less conservative because it enhances plume dispersion but LLRW facilities have some buildings so it is considered applicable).
- Inhalation is the only dose pathway considered.

Figure 4-13 provides the estimated doses to a member of the public as a result of the 30 minute fire. The impacts are on the order of tens of mrem to tens of rem depending on the waste stream. Because the release fraction for the fire was set at 0.1%, the contractor was effectively assuming that the amount of flammable waste corresponded to 0.1% of the waste and the flammable fraction was 100% released.

To compare the results to the NRC staff results shown in "trench fire – flammable" column of Table 4-8 multiply the results in Figure 4-12 by a factor of 440 (to account for assuming the intruder was exposed to 0.00188 m^3 released waste[18] as compared to 0.83 m^3 released waste in the IMPACTS analysis[19]). To refine the analysis, the release fraction would likely need to be waste stream- and radionuclide-specific. For example, the activated metal waste stream is comprised of isotopes that are almost 100% embedded in metal (e.g., C-14, Fe-55, Co-60) and others that would be associated mainly with degradation products (e.g., Pu isotopes). The relative release rates of these isotopes during a fire could be considerably different and would likely need to be accounted for in a site-specific accident analysis.

4.3.3 Comparison and Discussion

Both the NRC staff and contractor results identified potentially large impacts to an offsite member of the public as the result of operational accidents if the waste is dispersible or flammable. Both the NRC staff and contractor determined that it would be necessary to account for waste characteristics for the offsite dose impacts to be acceptable. The amount of waste that is dispersible, flammable, or both is a key variable to the offsite dose assessment. Management controls to reduce the potential for the occurrence of an accident and to reduce the impacts should an accident occur would likely be necessary without additional regulatory requirements on waste characteristics to limit the impacts of accidents. It is likely that waste-specific release factors could be developed to support site-specific licensing of GTCC waste disposal, and the results of those analyses could be used to identify management controls or other actions necessary to ensure that offsite members of the public are protected from accidental releases of radioactivity during operations. Waste specific release factors are likely to be waste stream and accident scenario specific.

The fire and container drop exposure scenarios were reasonable to consider for a facility that disposes of GTCC waste. Though the results of the accident consequence analyses assume the accident has occurred, these types of events are not anticipated. Without credit for waste form characteristics that reduce dispersivity of radioactivity released in an accident, the results are likely to be bounding. Likewise, assuming all the waste is flammable and can be suspended in the air during a fire is also likely to be a bounding assumption. An operational safety assessment can properly account for the waste characteristics.

[18] The intruder is assumed to be present for the entire 30 minute interval during which 1.88 m^3 waste is burned with a release fraction of 0.001.

[19] See Section 3.3.1. Volume of waste intruder is exposed to is 100 m^3 x 0.1 release fraction x 10 minutes inhalation time / 120 minutes fire duration = 0.83 m^3

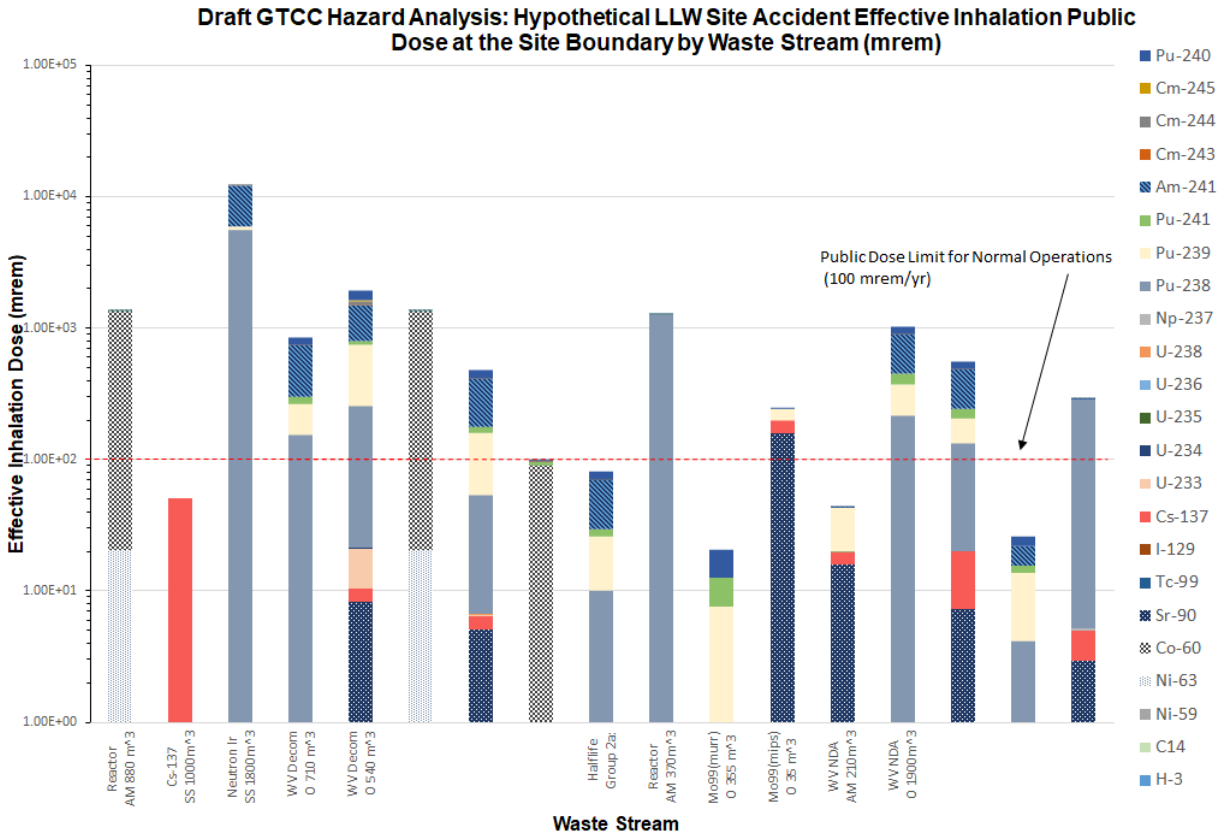


Figure 4-13 Contractor Results for Doses to an Offsite Member of the Public During a Hypothetical Trench Fire Accident

5. Conclusions

The NRC staff and contractor analyzed the impacts to members of the public from the disposal of 17 different GTCC waste streams. For the protection of the inadvertent intruder, the NRC staff determined that two waste streams may not be suitable for near-surface disposal (sealed sources associated with neutron irradiators and remote-handled other waste from decontamination activities at the WVDP). Those waste streams could potentially be found suitable for disposal with a site-specific intruder assessment. However, when the potential impacts are above 500 mrem to the acute intruder from a drilling event even after 500 years, the analysis shows it is difficult to safely manage the material with near-surface disposal and requires methods that are generally much more stringent than are normally used for disposal of Class A, B, and C waste.

The analyses of the potential doses to an offsite member of the public resulting from disposal of 17 different GTCC waste streams showed that while the characteristics of individual waste streams are important, the characteristics of the disposal site are arguably more important. When developing 10 CFR Part 61, the NRC recognized that the mobility of certain radionuclides in the environment can vary significantly from site to site. Site-specific analyses would be most appropriate to assess the dose resulting from releases of radioactivity from a land disposal facility (rather than fixed inventory limits generated by the regulator). Thus, the regulations require an analysis of pathways to demonstrate that the general population will be protected from releases of radioactivity. The NRC recognized that various facilities may differ in the type of waste received for disposal, the methods used to dispose of the waste, and that various facilities may be sited in different locations and subject to different natural processes. Thus, the NRC recognized that specific design precautions, waste form specifications, or both might be necessary to protect the public from more hazardous, long-lived LLRW. The analyses did not identify any changes that may be necessary to the existing regulatory requirements with respect to protecting an offsite member of the public after closure of the disposal facility.

The technical analysis required at § 61.13 is considered appropriate for demonstrating compliance, however, it is worth noting that GTCC wastes, which can have much higher activity levels than Class A, B, and C LLRW, may require the consideration of certain processes that have not previously been considered in performance assessments for LLRW disposal. For example, the generation of heat in certain sealed sources and activated metals can be significantly larger than Class A, B, and C LLRW and the generation of hydrogen gas may also need to be accounted for if the radiation flux is significant (see Appendix A for further details) (LLNL, 2000) (NRC,2000). Additionally, criticality could be a concern if sufficient fissile material is placed in a single disposal unit. The impacts will need to be considered and could have an impact on facility design or waste acceptance criteria. However, those phenomena can be considered under the existing regulatory requirements.

The analyses of potential doses to an offsite member of the public resulting from select accident scenarios during operations identified that the dose impacts could be very large if the GTCC waste had poor wasteform characteristics (e.g., high dispersivity, flammability). With proper characteristics, the accident impacts are likely to be acceptable. A site-specific operational safety assessment can be used to identify operational practices and management controls, as well as possible physical safety systems, necessary to mitigate the impacts to a member of the public that could result from operational accidents.

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Appendix A: Description of GTCC Waste Streams

Contents

A.1 Background

In conducting this regulatory analysis, the staff used the most comprehensive information available on the Greater Than Class C (GTCC) waste streams, which is DOE's FEIS ["Final Environmental Impact Statement for the Disposal of Greater-Than-Class C (GTCC) Low-Level Radioactive Waste and GTCC-Like Waste," DOE/EIS-0375] issued in February 2016. The U.S. Department of Energy (DOE) uses the term GTCC-like waste to describe radioactive waste that has similar characteristics to GTCC waste except the GTCC-like waste is owned by DOE. DOE issued the FEIS to consider the potential environmental impacts associated with constructing and operating a new facility or facilities, or using an existing facility, for the disposal of an estimated total volume of 12,000 m^3 (420,000 ft^3) of GTCC and GTCC-like waste anticipated to be generated through 2083 (DOE 2016; pages 1-3).

DOE divided the waste streams in its FEIS by the following characteristics/attributes:

1) WASTE GENERATOR: *GTCC and GTCC-like waste*

GTCC waste is waste that is generated from commercial facilities or activities, whereas, GTCC-like waste is owned or generated by DOE. The NRC staff has considered both GTCC and GTCC-like waste streams in its hazards evaluation without a distinction. In this document, the term GTCC waste is used as inclusive of both GTCC and GTCC-like waste and, as appropriate, the term GTCC-like waste is used when a distinction between these two types of waste is relevant to the discussion.

2) WASTE AVAILABILITY: *Existing (Group1) and Potential (Group 2)*

DOE separated the waste streams into two different groups as means of identifying the likelihood that the waste would be generated. DOE's Group 1 waste (termed 'existing' in this report) represents waste that is already generated or anticipated to be generated by existing facilities. DOE's Group 2 waste (termed 'potential' in this report) represents waste that may be generated due to either facilities that have yet to be built (e.g., potential new reactor construction) or activities that are dependent on future decisions (e.g., cleanup activities associated with the West Valley Demonstration Project - WVDP). DOE estimated the volume of GTCC waste out to the year 2083 (DOE 2016, pages 1-3). The NRC staff has also considered existing and potential waste streams in its hazards evaluation without a distinction. In this document, the term GTCC waste is used as inclusive of existing and potential waste and, as appropriate, the terms existing and potential waste are used when such a distinction is relevant.

3) WASTE TYPES: *Activated Metals, Sealed Sources, and Other*

The FEIS categorized the GTCC waste into activated metals, sealed sources, and other waste. As for the waste types, *activated metals* are largely generated from the decommissioning of nuclear reactors while *sealed sources* are widely used in equipment to diagnose and treat illnesses, sterilize medical devices, irradiate blood for transplant patients, nondestructively test structures and industrial equipment, and explore geologic formations to find oil and gas. The remaining GTCC waste is referred to as *other waste* and consists of contaminated equipment, debris, scrap metal, and decommissioning waste. Sources for other waste can include those from production of Mo-99 to the environmental cleanup of West Valley in the State of New York.

4) WASTE HANDLING: *Contact-Handled (CH) and Remote-Handled (RH)*

The hazards of various waste streams are related to the contents of the waste package and the container type. For example, the external exposure at the surface of a waste package affecting operational handling activities is a direct consequence of the waste package contents (e.g., gamma emitting radionuclides) and the consequences of an inadvertent intruder exposure scenario drilling through one or more waste packages is also affected by the concentration of waste within the waste package. DOE's FEIS differentiated between wastes that were contact-handled (CH) and remote-handled (RH) for those waste streams that were expected to have a higher exposure rate at the surface of the waste package. DOE stated a common container for storage and disposal of CH and RH waste is a 0.2-m^3 (55-gallon) drum and that standard waste boxes (SWB), with an internal volume of approximately 2 m^3, could also be used for CH waste (DOE 2016, page B-24). One exception is for RH activated metal waste that would use a specific container that has an internal volume of approximately 0.4 m^3.

The NRC staff used the information in the FEIS to separate the GTCC waste into seventeen GTCC waste streams as a means of identifying the hazards associated with each waste stream. The separation of the GTCC waste into seventeen waste streams helped ensure the hazards evaluation appropriately associates the hazards with the specific waste streams. Table A-1 provides the information for each of the 17 waste streams based on the four characteristics and attributes described above. Additionally, Table A-1 assigns a waste stream number to each of the waste streams that is used as an identifier in some of the tables and figures in this document.

Table A-1 presents the estimated volumes for these 17 waste streams, which represents over 99% of the waste considered in DOE's FEIS. The overall waste volume in Table A-1 is 11,285 m^3. DOE's overall estimate of 12,000 m^3 of waste material was based on rounding off volume estimates to two significant figures prior to adding specific waste stream amounts and explains the discrepancy between the Table A-1 overall waste volume (i.e., 11,285 m^3) and the FEIS value. The largest amount of waste is from GTCC waste (8,675 m^3) with approximately 40% of this amount comprised of existing waste (i.e., waste already existing or to be generated by currently operating facilities). The single largest amount of waste comes from the WVDP, which is estimated to generate 4,340 m^3 of waste and includes 2,110 m^3 of waste from potential exhumation of waste at the NRC licensed disposal area (NDA). Potential exhumation of the State-Licensed Disposal Area (SDA) at the West Valley site is estimated to generate another 2,125 m^3 of GTCC waste if a decision was made to undertake this activity. The second largest source of waste is from sealed sources and represents 2,800 m^3 of waste. Activated metal waste from commercial reactors totals 1,250 m^3 of waste, which includes both existing and future volumes of waste.

The waste streams comprising the GTCC, GTCC-like, and transuranic (TRU) wastes can vary considerably in volume, radionuclide content, and the form of the waste (e.g., activated metal, sealed sources, exhumed waste and soil). The remainder of this appendix provides a detailed compilation of the characteristics of the inventory of the 17 waste streams. This appendix groups the waste stream into the two broad designations. One designation is 'Existing waste streams' and represents waste generated from existing activities and facilities. The second designation is 'Potential waste streams' and represents waste that may be generated if future decisions are made to move forward on certain activities and facilities. The inventories of radionuclides presented in this report are considered potentially important to the performance of a disposal facility both during operations (i.e., safety and security) and after closure of the facility (i.e., inadvertent intruder and offsite individuals). The inventories do not include all the

Table A-1 Description of the Waste Streams in NRC's Hazards Evaluation

Waste Stream Type	Waste Generator	NRC Hazards Evaluation		
		Volume (m^3)	Waste Stream Source	Waste Stream Number
DOE FEIS Group 1 (existing activities and facilities)				
Activated Metals	GTCC	880	Commercial reactors (remote-handled)	1
Sealed Sources	GTCC	1000	Cs-137 irradiators (contact-handled)	2
		1800	Neutron irradiators (contact-handled)	3
Other	GTCC-like	710	WVDP Decontamination of the MPPB (contact-handled)	4
		540	WVDP Decontamination of the MPPB (remote-handled)	5
DOE FEIS Group 2 (potential activities and facilities)				
Activated Metals	GTCC	370	Commercial reactors (remote-handled)	6
		210	WVDP Exhumation of NRC Disposal Area (remote-handled)	7
		525	Exhumation of State Disposal Area (remote-handled)	8
Other	GTCC	1,900	WVDP Exhumation of NRC Disposal Area (remote-handled)	9
		1,200	Exhumation of State Disposal Area (SNAP) (contact-handled)	10
		400	Exhumation of State Disposal Area (contact-handled)	11
		35	Mo-99 Production - MURR (remote-handled)	12
		355	Mo-99 Production - MIPS (remote-handled)	13
	GTCC-like	220	WVDP Decommissioning of MPPB and WTF (contact-handled)	14
		760	WVDP Decommissioning of MPPB and WTF (remote-handled)	15
		120	Pu-238 Production (contact-handled)	16
		260	Pu-238 Production (remote-handled)	17

Note: MURR – Missouri University Research Reactor; MIPS – Medical Isotope Production System; and SNAP – Systems for Nuclear Auxiliary Power

radionuclides in the FEIS but are considered appropriate for providing reasonable inventories for the assessment of the potential hazards (e.g., no significant underestimate of the hazards). The set of radionuclides is the same for each of the waste streams; however, the hazard significance will vary significantly over the waste streams as will be evident based on the variation in the specific inventories.

A.2 Existing Waste Streams

Existing waste streams represent waste that is either already generated or waste that will be generated that is not dependent on a future decision to proceed (e.g., waste generated by facilities currently operating, specific activities at the WVDP for which a decision to proceed has already been made). These waste streams include both GTCC waste and GTCC-like waste.

A.2.1 GTCC Activated Metal Waste

Activated metal GTCC waste generated from existing activities and facilities are associated with commercial reactors. DOE has estimated that approximately 880 m^3 of activated metal waste would be generated by the current fleet of commercial reactors (primarily during decommissioning). 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 GTCC waste stream. Radionuclides in the activated metals are comprised of both short half-lived radionuclides that decay substantially over a 100 year period and other radionuclides with much longer half-lives that will be present for thousands of years and longer. Most of the initial activated metal inventory comes from radionuclides (e.g., Ni-63, half-life of approximately 100 years) with a relatively short half-life relative to the timeframes of 1,000 years and longer that are typically considered for radioactive waste disposal. As shown in Table A-3, Ni-63 is estimated to have a concentration approximately 3 times the Class C limit.

The short-lived radionuclides are the key contributors to the thermal output of the activated metal waste stream from commercial reactors. Compared to Class A, Class B, and Class C categories of LLRW, activated metals from commercial reactors contain a significant amount of heat generating radionuclides (e.g., Ni-63). Additionally, short-lived, gamma-emitting, radionuclides (e.g., Co-60) in activated metal waste will need a significant amount of shielding to reduce the levels of radiation to acceptable levels and/or will have to be handled remotely to protect workers from unnecessary exposure (e.g., waste with a dose rate of 2 millisievert per hour (200 millirem per hour [mrem/h]) at the surface of the waste package) [DOE, 2016. Page 1-13].

Occasionally, activated metals from commercial reactors are impacted by the occurrence of damaged fuel, which has the potential to result in fission products and transuranic radionuclides contaminating the metal surfaces via contact with water containing radionuclides from the damaged fuel. Unlike the activated radionuclides that occur throughout the metal materials, fission products and transuranic radionuclides exist only as surface contamination (i.e., radioactive contamination deposited on corrosion film and crud surfaces of the various plant systems). This surface contamination will contain long-lived radionuclides that are typically included in safety assessments (e.g., Tc-99 with a half-life of 210,000 years) and TRU radionuclides (e.g., Pu-239 with a half-life of 24,000 years). Generally, surface contamination is expected to be low due to the increasingly lower rates for damaged fuel in currently operating

reactors; therefore, the radionuclides associated with surface contamination are estimated to be well below the Class C limits in Part 61.

Table A-2 provides the characteristics of the activated metal GTCC waste to be generated from the closed and currently operating commercial reactors. The basis for the information provided in Table A-2 is:

- Volume of GTCC Activated Metal waste is 880 m^3 (FEIS, Table B-1)

- DOE assumed activated metal GTCC waste would be remote-handled and packaged in activated metal canisters (ACMs) that are unshielded stainless steel canisters with internal volume of 0.37 m^3 (FEIS; pages B-26)[20]

- Inventory from FEIS Table B-5 except for selected radionuclides on surface contamination for which NRC used available data from commercial reactors (see average surface activity reported in Table V, McCartin et al, 2018) – the selected radionuclides are: Co-60, Sr-90, Tc-99, I-129, Cs-137, Pu-238, Pu-239, and Am-241 (The NRC elected to use available data from commercial reactors for specific radionuclides due to inconsistencies in the FEIS inventory for certain surface contamination radionuclides)

[20] For simplicity, the analysis in this regulatory basis uses an internal volume of 0.2 m^3 for contact-handled waste containers and an internal volume of 0.4 m^3 for remote-handled waste containers.

Table A-2 Inventory for 880 m^3 of GTCC Activated Metal Waste (Remote-Handled) Generated by Operating and Closed Commercial Reactors (Waste Stream #1 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	5×10^{7}	44.7	57,000		None
H-3	12	6,800	7×10^{-4}	7.7		None
Pu-241	14	25	2.4×10^{-4}	0.028	16 nCi/g	3,500 nCi/g
Cm-244	18	0	0			100 nCi/g
Sr-90	29	4.3	3.1×10^{-5}	0.005		7,000 Ci/m^3
Cm-243	29	0	0			100 nCi/g
Cs-137	30	76	8.8×10^{-4}	0.09		4,600 Ci/m^3
Pu-238	88	1.5	8.8×10^{-5}	0.0017	0.95 nCi/g	100 nCi/g
Ni-63	96	1.8×10^{7}	306	20,500		7,000 Ci/m^3
Am-241	430	2.4	7×10^{-4}	0.0027	1.5 nCi/g	100 nCi/g
C-14	5,700	23,000	5.18	26		80 Ci/m^3
Pu-240	6,500	0	0			100 nCi/g
Cm-245	8,500	0	0			100 nCi/g
Pu-239	24,000	1.3	0.021	0.0015	0.82 nCi/g	100 nCi/g
Ni-59	75,000	1.3×10^{5}	1,620	148		220 Ci/m^3
U-233	160,000	0	0			None
U-234	240,000	0	0			None
Tc-99	250,000	0.2	0.012	2.3×10^{-4}		3 Ci/m^3
Np-237	2.1×10^{6}	0	0			100 nCi/g
I-129	1.6×10^{7}	1.86×10^{-3}	0.011	2.1×10^{-6}		0.08 Ci/m^3
U-236	2.3×10^{7}	0	0			None
U-235	7.0×10^{8}	0	0			None
U-238	4.5×10^{9}	0	0			None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

A.2.2 GTCC Sealed Source Waste

GTCC sealed sources can be divided into two groups: small sealed sources (i.e., commercial plutonium, americium and curium sources) and large irradiators (i.e., industrial cesium sources). The small sealed sources are neutron irradiators typically consisting of concentrated radioactive material encapsulated in relatively small containers made of titanium, stainless steel, or other metals. These sources are commonly used to sterilize medical products, detect flaws and failures in pipelines and metal welds, determine the moisture content in soil and other materials, and diagnose and treat illnesses such as cancer. Sealed sources can encompass several physical forms, including ceramic oxides, salts, or metals. DOE estimated that there are 1,435 Cs-137 irradiators (identified as large sealed sources in DOE's FEIS) in the waste inventory, each with an assumed volume of 0.71 m^3 (25 ft^3). Because of their size, the Cs-137 irradiators are not easily packaged and are assumed to be disposed of individually in their original shielded devices. DOE estimated the total volume of the neutron irradiators (small sealed sources) is 1,800 m^3 and the total volume of the Cs-137 irradiators is 1,000 m^3. This waste is considered GTCC waste.

A.2.2.1 Large Sealed Sources (Cs-137 irradiators)

Cs-137 irradiators result in a total inventory of 1.7 million curies of Cs-137 (see Table A-12). In these irradiators, the Cs-137 source is contained within a very robust shielded device, which would be expected to help the waste form retain its integrity and reduce the worker dose levels during processing and disposal (FEIS, page B-21). Cesium chloride salt was generally used in older Cs-137 sources, and newer small sources typically have the radionuclide bonded in a ceramic. Of these two forms, cesium chloride salt is much more water-soluble. Cs-137, with a half-life of 30 years, will show significant radioactive decay over a few hundred years. Table A-3 provides the characteristics of the GTCC sealed source waste to be generated from the Cs-137 irradiators.

The basis for the information provided in Table A-3 is:

- Volume of GTCC sealed source waste (Cs-137) is 1,000 m^3 (FEIS, Table B-1)
- DOE assumed Cs-137 sealed sources would be contact-handled and would be disposed individually in their original shielded devices due to the large volume – a single Cs-137 source occupies a volume of 0.71 m^3 (FEIS; pages B-21)
- Inventory from the FEIS Table B-4

A.2.2.2 Small Sealed Sources (neutron irradiators)

The neutron irradiators contain transuranic radionuclides in concentrations that are significantly above the Part 61 Class C limits (e.g., Pu-238 concentration of 37,000 nCi/g and Pu-239 concentration of 2,600 nCi/g – see Table A-12) and thus are considered TRU waste. Pu-239 is a fissile radionuclide with an inventory of 135 kilograms for all of the neutron irradiators, which results in approximately 15 grams of Pu-239 in a 0.2 m^3 container. Isotopes of plutonium can be a concern for the intruder, especially because some of these radionuclides have half-lives that are very long (e.g., Pu-239 half-life of 24,000 years). The encapsulated nature of sealed sources does provide some limitations on exposure while the capsule remains intact.

Table A-4 provides the characteristics of the GTCC sealed source waste to be generated from the neutron irradiators. The basis for the information provided in Table A-4 is:

- Volume of GTCC sealed source waste (neutron irradiators) is 1,800 m^3 (FEIS, Table B-1)
- DOE assumed neutron sealed sources would be contact-handled and would be packaged in a 0.2-m^3 (55-gallon) drum (FEIS; pages B-21 and B-26)
- Inventory from FEIS Table B-4

Table A-3 Inventory for 1,000 m^3 of GTCC Sealed Source Waste (Contact-handled) from Cs-137 Irradiators (Waste Stream #2 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3					None
H-3	12					None
Pu-241	14					3,500 nCi/g
Cm-244	18					100 nCi/g
Sr-90	29					7,000 Ci/m^3
Cm-243	29					100 nCi/g
Cs-137	30	1.7×10^6	20	1,700		4,600 Ci/m^3
Pu-238	88					100 nCi/g
Ni-63	96					7,000 Ci/m^3
Am-241	430					100 nCi/g
C-14	5,700					80 Ci/m^3
Pu-240	6,500					100 nCi/g
Cm-245	8,500					100 nCi/g
Pu-239	24,000					100 nCi/g
Ni-59	75,000					220 Ci/m^3
U-233	160,000					None
U-234	240,000					None
Tc-99	250,000					3 Ci/m^3
Np-237	2.1×10^6					100 nCi/g
I-129	1.6×10^7					0.08 Ci/m^3
U-236	2.3×10^7					None
U-235	7.0×10^8					None
U-238	4.5×10^9					None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

Table A-4 Inventory for 1,800 m^3 of GTCC Sealed Source Waste (Contact-handled) from Neutron Irradiators (Waste Stream #3 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3					None
H-3	12					None
Pu-241	14					3,500 nCi/g
Cm-244	18	22	2.7×10^{-4}	0.012	6.8	100 nCi/g
Sr-90	29					7,000 Ci/m^3
Cm-243	29					100 nCi/g
Cs-137	30					4,600 Ci/m^3
Pu-238	88	1.2×10^5	7.1	67	37,000	100 nCi/g
Ni-63	96					7,000 Ci/m^3
Am-241	430	1.5×10^5	44	83.3	46,300	100 nCi/g
C-14	5,700					80 Ci/m^3
Pu-240	6,500					100 nCi/g
Cm-245	8,500					100 nCi/g
Pu-239	24,000	8,400	136	4.7	2,600	100 nCi/g
Ni-59	75,000					220 Ci/m^3
U-233	160,000					None
U-234	240,000					None
Tc-99	250,000					3 Ci/m^3
Np-237	2.1×10^6					100 nCi/g
I-129	1.6×10^7					0.08 Ci/m^3
U-236	2.3×10^7					None
U-235	7.0×10^8					None
U-238	4.5×10^9					None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

A.2.3 Other Waste

DOE's FEIS includes a category entitled 'other' waste that includes a variety of waste streams, however, only one of the waste streams is considered an existing waste stream, which is associated with decontamination activities at the WVDP.

A.2.3.1 GTCC Other Waste from Decontamination Activities at the WVDP

DOE's decontamination activities for the WVDP are estimated to involve the generation of GTCC-like waste from past and ongoing activities related to decontamination of the Main Plant Process Building (MPPB). The existing GTCC-like 'other waste' is estimated to represent a volume of 1,250 m^3.

Table A-5 provides the characteristics of the contact-handled GTCC-like other waste from activities associated with decontamination of the Main Plant Process Building (MPPB) at the WVDP. The basis for the information provided in Table A-5 is:

- Volume of contact-handled GTCC-like other is 710 m^3 (FEIS, Table B-3)
- DOE assumed that other waste would be stabilized with grout or another matrix prior to being shipped for disposal (FEIS, page B-22); DOE assumed contact-handled waste would be packaged in a 0.2-m^3 (55-gallon) drum or a standard waste box (SWB) having an internal volume of 1.88 m^3 (FEIS, B-22, -24, and -26)
- Inventory from FEIS Table B-4 (note: the inventory reported in Table B-4 contains approximately 35 m^3 of other waste primarily from Idaho National Laboratory that for purposes of the NRC staff evaluation the radionuclide amounts from that waste was simply included as part of the WVDP inventory – a conservative assumption)

Table A-6 provides the characteristics of the remote-handled GTCC-like other waste from activities associated with decontamination of the MPPB at the WVDP. The basis for the information provided in Table A-6 is:

- Volume of remote-handled GTCC-like other is 540 m^3 (FEIS, Table B-3)
- DOE assumed that other waste would be stabilized with grout or another matrix prior to being shipped for disposal (FEIS, page B-22); DOE assumed remote-handled waste (other than activated metal waste) would be packaged in 0.2-m^3 (55-gallon) drums (FEIS, B-22 and -26)
- Inventory from FEIS Table B-4 (note: the inventory reported in Table B-4 contains approximately 165 m^3 of other waste primarily from the Oak Ridge Reservation that for purposes of the NRC staff evaluation the radionuclide amounts from that waste was simply included as part of the WVDP inventory – a conservative assumption)

Table A-5 Inventory for 710 m^3 of GTCC-like Other Waste (Contact-handled) from Decontamination Activities at the WVDP (Waste Stream #4 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	0.0041	3.7×10^{-9}	5.8×10^{-6}		None
H-3	12	0.17	1.8×10^{-8}	2.4×10^{-4}		None
Pu-241	14	14,000	0.14	20	11,000	3,500 nCi/g
Cm-244	18	1.8	2.2×10^{-5}	0.0025	1.4	100 nCi/g
Sr-90	29	66	4.8×10^{-4}	0.093		7,000 Ci/m^3
Cm-243	29	0.076	1.5×10^{-6}	1.1×10^{-4}	0.06	100 nCi/g
Cs-137	30	65	7.5×10^{-4}	0.092		4,600 Ci/m^3
Pu-238	88	1,300	0.077	1.8	1,000	100 nCi/g
Ni-63	96	0.025	4.2×10^{-7}	3.5×10^{-5}		7,000 Ci/m^3
Am-241	430	4,400	1.3	6.2	3,400	100 nCi/g
C-14	5,700	13	0.0029	0.018		80 Ci/m^3
Pu-240	6,500	710	3.1	1	560	100 nCi/g
Cm-245	8,500	2×10^{-9}	1.2×10^{-11}	2.8×10^{-12}	1.6×10^{-9}	100 nCi/g
Pu-239	24,000	900	14	1.3	700	100 nCi/g
Ni-59	75,000	0.076	9.4×10^{-4}	1.1×10^{-4}		220 Ci/m^3
U-233	160,000	9.4	0.97	0.013		None
U-234	240,000	44	6.9	0.062		None
Tc-99	250,000	0.32	0.019	4.5×10^{-4}		3 Ci/m^3
Np-237	2.1×10^6	1.1	1.6	0.0016	0.86	100 nCi/g
I-129	1.6×10^7	9.7×10^{-5}	5.5×10^{-4}	1.4×10^{-7}		0.08 Ci/m^3
U-236	2.3×10^7	0.054	0.84	7.6×10^{-5}		None
U-235	7.0×10^8	0.16	74	2.2×10^{-4}		None
U-238	4.5×10^9	0.091	270	1.3×10^{-4}		None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

Table A-6 Inventory for 540 m^3 of GTCC-like Other Waste (Remote-handled) from Decontamination Activities at the WVDP (Waste Stream #5 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	1,200	1.1×10^{-3}	2.2		None
H-3	12	16	1.7×10^{-6}	0.03		None
Pu-241	14	17,000	0.17	32	17,800	3,500 nCi/g
Cm-244	18	1,100	0.014	2.0	1,100	100 nCi/g
Sr-90	29	36,000	0.26	67		7,000 Ci/m^3
Cm-243	29	2.2	4.3×10^{-5}	4.1×10^{-3}	2.3	100 nCi/g
Cs-137	30	39,000	0.45	72		4,600 Ci/m^3
Pu-238	88	1,500	0.088	2.8	1,550	100 nCi/g
Ni-63	96	9,400	0.16	17		7,000 Ci/m^3
Am-241	430	5,300	1.6	9.8	5,440	100 nCi/g
C-14	5,700	100	0.022	0.18		80 Ci/m^3
Pu-240	6,500	1,800	7.9	3.3	1,830	100 nCi/g
Cm-245	8,500	340	2.0	0.63	350	100 nCi/g
Pu-239	24,000	2,900	47	5.4	3,000	100 nCi/g
Ni-59	75,000	160	2.0	0.30		220 Ci/m^3
U-233	160,000	790	82	1.5		None
U-234	240,000	1.6	0.25	3.0×10^{-3}		None
Tc-99	250,000	170	10	0.32		3 Ci/m^3
Np-237	2.1×10^6	1.5	2.1	2.8×10^{-3}	1.5	100 nCi/g
I-129	1.6×10^7	2.7	15	5.0×10^{-3}		0.08 Ci/m^3
U-236	2.3×10^7	0.79	12	1.5×10^{-3}		None
U-235	7.0×10^8	0.35	160	6.5×10^{-4}		None
U-238	4.5×10^9	11	33,000	0.020		None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

A.3 Potential Waste Streams

Potential waste streams represent waste that may be generated in the future. The generation of these waste streams is dependent on future decisions that have not been made (e.g., decision to build additional commercial nuclear power reactors, decision to exhume GTCC waste from the NRC licensed disposal area at the West Valley site). These waste streams include both GTCC waste and GTCC-like waste. Although the waste streams include both activated metal and other waste streams, there are no sealed sources that were estimated in DOE's FEIS.

This section presents the inventory estimated for waste streams that may be generated in the future based on the potential activities described in DOE's FEIS.

A.3.1 GTCC Activated Metal Waste

DOE has estimated that approximately 380 m^3 of activated metal waste would be generated by 33 new proposed commercial reactors (DOE 2016; page 1-15). In addition, activated metal waste is currently buried at the West Valley site and is estimated to result in a total volume of 740 m^3 (DOE 2016; Page 1-15) associated with a potential exhumation of the NDA associated with the WVDP and with the SDA at the West Valley site but not associated with the WVDP. All the activated metal waste is considered GTCC waste. DOE assumed activated metal GTCC waste would be remote-handled and packaged in unshielded stainless steel canisters with internal volume of 0.37 m^3 each (DOE 2016; pages B-18 and 26).[21]

A.3.1.1 Commercial Reactors

Table A-7 provides the characteristics of the activated metal GTCC waste to be generated from the commercial reactors to be built in the future. The basis for the information provided in Table A-7 is:

- Volume of GTCC Activated Metal waste is 370 m^3 (FEIS, Table B-1).
- DOE assumed activated metal GTCC waste would be remote-handled and packaged in activated metal canisters (ACMs) that are unshielded stainless steel canisters with internal volume of 0.37 m^3 (FEIS; pages B-26).
- Inventory is obtained by scaling the inventory in Table A-2 to 370 m^3.

A.3.1.2 Activated Metal Waste from Exhumation of the NDA

The WVDP in western New York is the location of the only commercial nuclear fuel reprocessing plant to operate in the United States. Radioactive waste that is now characterized as GTCC activated metal waste was disposed at the NDA. A future decision will be made whether to remove or to close in-place this buried waste as part of future decommissioning activities (Argonne 2010; page 45). Activated metal waste buried at the NDA is comprised primarily of hardware (210 m^3) and a very small amount of failed spent fuel (0.3 m^3) (Argonne 2010, Table 4-2).

[21] For simplicity, the analysis in this regulatory basis uses an internal volume of 0.2 m^3 for contact-handled waste containers and an internal volume of 0.4 m^3 for remote-handled waste containers.

Table A-7 Inventory for 370 m^3 of GTCC Activated Metal Waste (Remote-handled) Generated by Potential Commercial Reactors (Waste Stream #6 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	2.1×10^7	18.8	57,000		None
H-3	12	2,900	3.0×10^{-4}	7.7		None
Pu-241	14	10.5	1.0×10^{-4}	0.028	16 nCi/g	3,500 nCi/g
Cm-244	18	0	0			100 nCi/g
Sr-90	29	1.8	1.3×10^{-5}	0.005		7,000 Ci/m^3
Cm-243	29	0	0			100 nCi/g
Cs-137	30	32	3.7×10^{-4}	0.09		4,600 Ci/m^3
Pu-238	88	0.63	3.7×10^{-5}	0.0017	0.95 nCi/g	100 nCi/g
Ni-63	96	7.6×10^6	129	20,500		7,000 Ci/m^3
Am-241	430	1.0	3×10^{-4}	0.0027	1.5 nCi/g	100 nCi/g
C-14	5,700	9,700	2.18	26		80 Ci/m^3
Pu-240	6,500	0	0			100 nCi/g
Cm-245	8,500	0	0			100 nCi/g
Pu-239	24,000	0.55	0.0088	0.0015	0.82 nCi/g	100 nCi/g
Ni-59	75,000	5.5×10^4	680	148		220 Ci/m^3
U-233	160,000	0	0			None
U-234	240,000	0	0			None
Tc-99	250,000	0.084	0.0049	2.3×10^{-4}		3 Ci/m^3
Np-237	2.1×10^6	0	0			100 nCi/g
I-129	1.6×10^7	7.8×10^{-4}	0.0044	2.1×10^{-6}		0.08 Ci/m^3
U-236	2.3×10^7	0	0			None
U-235	7.0×10^8	0	0			None
U-238	4.5×10^9	0	0			None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

Table A-8 provides the characteristics of the activated metal GTCC waste to be generated from the NDA should a decision be made to exhume the waste. The basis for the information provided in Table A-8 is:

- Volume of GTCC Activated Metal waste at the NDA is 210 m^3 (FEIS, Table B-1, footnote h)

- DOE assumed activated metal GTCC waste would be remote-handled and packaged in activated metal canisters (ACMs) that are unshielded stainless steel canisters with internal volume of 0.37 m^3 (FEIS; pages B-26)

- Inventory from Table 2-2 (Wild, 2000) [note: Inventory values Table 2-2 were reported for the years 2000 – these values were decay corrected to the year 2019 to be consistent with times used in the FEIS (Argonne 2010, page 45)]

A.3.1.3 Activated Metal Waste from Exhumation of the SDA

Radioactive waste that is now characterized as GTCC waste was disposed at the SDA at the West Valley site in western New York. A future decision will be made whether to remove or to close in-place this buried waste as part of future decommissioning activities (Argonne 2010; page 45). Activated metal waste buried at the SDA is from metal components associated with special purpose reactors such as small research and naval reactors.

Table A-9 provides the characteristics of the activated metal GTCC waste to be generated from the SDA should a decision be made to exhume the waste. The basis for the information provided in Table A-9 is:

- Volume of GTCC Activated Metal waste at the NDA is 210 m^3 (FEIS, Table B-1, footnote h).

- DOE assumed activated metal GTCC waste would be remote-handled and packaged in activated metal canisters that are unshielded stainless steel canisters with internal volume of 0.37 m^3 (FEIS; pages B-26).

- FEIS Table B-7 provides a combined inventory for GTCC activated metal waste that might be generated in the future. The specific waste streams that together add up to the inventory values in Table B-7 are for commercial reactors to be built in the future (Waste Stream 6), exhumation of the NDA (Waste Stream 7), and exhumation of the SDA (Waste Stream 8). The inventory for activated metal waste from just the SDA (Waste Stream 8) is estimated by subtracting the appropriate values for waste streams 6 and 7 from the combined inventory amounts in Table B-7. Waste Stream 7 values can be taken directly Table A-4, however, Waste Stream 6 values could not be taken directly from Table A-2 because the values in Table A-2 include some values from the FEIS and some values developed by NRC staff. To estimate Waste Stream 6 values (commercial reactors to be built in the future) based on the FEIS only is needed to allow appropriate subtraction of values from the combined FEIS values in Table B-7. The Waste Stream 6 values for the subtraction are based on scaling the values in FEIS Table B-5 for GTCC activated metal waste from operating and closed commercial reactors and scaling those values, which are based on a volume of 880 m^3, to the 370 m^3 assigned to Waste Stream 6.

Table A-8 Inventory for 210 m^3 of GTCC Activated Metal Waste (Remote-handled) from Potential Exhumation of the NDA (Waste Stream #7 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	2,500	2.2×10^{-3}	12		None
H-3	12	8.7	9.1×10^{-7}	0.042		None
Pu-241	14	2,300	0.022	11	6,000	3,500 nCi/g
Cm-244	18	3.2	4.0×10^{-5}	0.015	8	100 nCi/g
Sr-90	29	7,400	0.054	35		7,000 Ci/m^3
Cm-243	29	0.13	2.6×10^{-6}	6.4×10^{-4}	0.4	100 nCi/g
Cs-137	30	9,700	0.11	46		4,600 Ci/m^3
Pu-238	88	120	7.0×10^{-3}	0.57	320	100 nCi/g
Ni-63	96	1.0×10^5	1.7	480		7,000 Ci/m^3
Am-241	430	690	0.20	3.3	1,800	100 nCi/g
C-14	5,700	515	0.12	2.45		80 Ci/m^3
Pu-240	6,500	160	0.70	0.76	420	100 nCi/g
Cm-245	8,500	8×10^{-4}	4.7×10^{-6}	3.8×10^{-6}	2.0×10^{-3}	100 nCi/g
Pu-239	24,000	240	3.9	1.2	640	100 nCi/g
Ni-59	75,000	1,100	14	5.3		220 Ci/m^3
U-233	160,000	3.8	0.39	0.018		None
U-234	240,000	0.2	0.031	9.3×10^{-4}		None
Tc-99	250,000	4.3	0.25	0.02		3 Ci/m^3
Np-237	2.1×10^6	0.067	0.096	3.2×10^{-4}	0.2	100 nCi/g
I-129	1.6×10^7	9.0×10^{-3}	0.051	4.3×10^{-5}		0.08 Ci/m^3
U-236	2.3×10^7	0.11	1.6	5.0×10^{-4}		None
U-235	7.0×10^8	0.048	22	2.3×10^{-4}		None
U-238	4.5×10^9	0.64	1,900	3.1×10^{-3}		None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

Table A-9 Inventory for 525 m^3 of GTCC Activated Metal Waste (Remote-handled) from Potential Exhumation of the SDA (Waste Stream #8 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	1.97×10^{6}	1.76	3.76×10^{3}		None
H-3	12	730	7.6×10^{-5}	1.39		None
Pu-241	14	220	2.2×10^{-3}	0.42	230	3,500 nCi/g
Cm-244	18	4.8	5.9×10^{-5}	9.1×10^{-3}	5.1	100 nCi/g
Sr-90	29	550	0.004	1.0		7,000 Ci/m^3
Cm-243	29	6.4×10^{-3}	1.2×10^{-7}	1.2×10^{-5}	7.0×10^{-3}	100 nCi/g
Cs-137	30	7,900	0.09	15		4,600 Ci/m^3
Pu-238	88	10	5.9×10^{-4}	0.019	10.6	100 nCi/g
Ni-63	96					7,000 Ci/m^3
Am-241	430	7.3	2.0×10^{-3}	0.014	7.8	100 nCi/g
C-14	5,700					80 Ci/m^3
Pu-240	6,500	.32	1.4×10^{-3}	6.1×10^{-4}	0.3	100 nCi/g
Cm-245	8,500	4.2×10^{-6}	2.5×10^{-8}	8.1×10^{-9}	4.5×10^{-6}	100 nCi/g
Pu-239	24,000					100 nCi/g
Ni-59	75,000					220 Ci/m^3
U-233	160,000	0.03	3.0×10^{-3}	5.8×10^{-5}		None
U-234	240,000	5.0×10^{-3}	7.9×10^{-4}	9.5×10^{-6}		None
Tc-99	250,000	3.7	0.22	7.0×10^{-3}		3 Ci/m^3
Np-237	2.1×10^{6}					100 nCi/g
I-129	1.6×10^{7}	1.3	7.4	2.5×10^{-3}		0.08 Ci/m^3
U-236	2.3×10^{7}	4.0×10^{-3}	0.06	7.6×10^{-6}		None
U-235	7.0×10^{8}	0.024	11	4.6×10^{-5}		None
U-238	4.5×10^{9}	0.20	585	3.7×10^{-4}		None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

A.3.2 Sealed Source Waste Streams

There were no sealed sources reported in DOE's FEIS for future activities and facilities.

A.3.3 Other Waste Streams

DOE's FEIS includes a category entitled 'other' waste that included waste streams associated with a potential exhumation of radioactive waste from the SDA at the West Valley site in New York, activities associated exhumation of radioactive waste from the NDA associated with the WVDP, decommissioning the facilities associated with the WVDP, waste resulting from the production of radioisotopes for nuclear imaging procedures (i.e., Mo-99 production), and waste associated with a potential new DOE program for the production of Pu-238. The waste streams in the 'other waste' category include GTCC and GTCC-like waste. Additionally, the inventory for each waste stream is quite variable due to differences in the quantities of radionuclides present and the form of the waste (e.g., exhumation of a disposal area versus waste from a Mo-99 production facility).

A.3.3.1 GTCC Other Waste from Exhumation of the NDA and SDA

Should a decision be made to exhume the NDA and SDA, GTCC 'other waste' would result from this activity amounting to 3,500 m^3 of waste (1,900 m^3 associated with the NDA and 1,600 m^3 associated with the SDA). One part of the exhumation activities of the SDA is exhumation associated with SNAP, which accounts for 1,200 m^3 of potential waste exhumed from the SDA.

Table A-10 provides the characteristics of the remote-handled GTCC other waste from activities associated with the potential exhumation of the NDA as part of WVDP. The basis for the information provided in Table A-10 is:

- Volume of remote-handled GTCC other waste is 1,900 m^3 (FEIS, Table B-1, Footnote h).
- DOE assumed that other waste would be stabilized with grout or another matrix prior to being shipped for disposal (FEIS, page B-22); and DOE also assumed remote-handled waste (other than activated metal waste) would be packaged in 0.2-m^3 (55-gallon) drums (FEIS, B-22 and 26)
- Inventory from Table 2-2 (Wild 2000) [note: Inventory values Table 2-2 were reported for the years 2000 – these values were decay corrected to the year 2019 to be consistent with times used in the FEIS (Argonne 2010, page 45)]

Table A-10 Inventory for 1,900 m^3 of GTCC Other Waste (Remote-handled) from Potential Exhumation of the NDA (Waste Stream #9 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	4.8	4.3×10^{-3}	2.5×10^{-3}		None
H-3	12	14	1.4×10^{-3}	7.1×10^{-3}		None
Pu-241	14	3,900	38	2.0	1,100	3,500 nCi/g
Cm-244	18	5.0	0.062	2.6×10^{-3}	1.5	100 nCi/g
Sr-90	29	11,000	79	5.7		7,000 Ci/m^3
Cm-243	29	0.24	4.6×10^{-3}	1.2×10^{-4}	0.069	100 nCi/g
Cs-137	30	14,000	160	7.4		4,600 Ci/m^3
Pu-238	88	210	12	0.11	60	100 nCi/g
Ni-63	96	170	2.9	0.09		7,000 Ci/m^3
Am-241	430	1,000	310	0.55	305	100 nCi/g
C-14	5,700	0.87	0.20	4.6×10^{-4}		80 Ci/m^3
Pu-240	6,500	240	1,000	0.13	70	100 nCi/g
Cm-245	8,500	1.3×10^{-3}	7.5×10^{-3}	6.7×10^{-7}	3.7×10^{-4}	100 nCi/g
Pu-239	24,000	340	5,400	0.18	98	100 nCi/g
Ni-59	75,000	1.8	23	9.7×10^{-4}		220 Ci/m^3
U-233	160,000	7.5	770	3.9×10^{-3}		None
U-234	240,000	0.38	60	2.0×10^{-4}		None
Tc-99	250,000	6.0	350	3.1×10^{-3}		3 Ci/m^3
Np-237	2.1×10^6	0.093	130	4.9×10^{-5}	0.027	100 nCi/g
I-129	1.6×10^7	0.012	71	6.5×10^{-6}		0.08 Ci/m^3
U-236	2.3×10^7	0.17	2,600	9.0×10^{-5}		None
U-235	7.0×10^8	0.072	34,000	3.8×10^{-5}		None
U-238	4.5×10^9	0.81	2.4×10^6	4.3×10^{-4}		None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

Table A-11 provides the characteristics of the contact-handled GTCC other waste from SNAP generated from potential exhumation of the SDA at the West Valley site. The basis for the information provided in Table A-11 is:

- Volume of contact-handled GTCC other waste is 1,200 m^3 (Argonne 2010 Table 4-3)

- DOE assumed that other waste would be stabilized with grout or another matrix prior to being shipped for disposal (FEIS, page B-22); DOE assumed contact-handled waste would be packaged in a 0.2-m^3 (55-gallon) drum or a standard waste box (SWB) having an internal volume of 1.88 m^3 (FEIS, B-22, -24, and -26)

- Inventory from Table 4-3 (Argonne 2010) and were decay corrected to the year 2019 to be consistent with times used in the FEIS (see footnote c in Table 4-3)

Table A-12 provides the characteristics of the contact-handled GTCC other waste from potential exhumation of the SDA at the West Valley site but excluding the SNAP waste. The basis for the information provided in Table A-12 is:

- Volume of contact-handled GTCC other waste (excluding SNAP) is 400 m^3 based on a total SDA volume of 1,600 m^3 for CH waste resulting from exhumation of the SDA in FEIS Table B-1 and subtracting the volume of 1,200 m^3 for SNAP (see basis for Table 11 above), which leaves 400 m^3 for the non-SNAP portion for exhumation of the SDA

- DOE assumed that other waste would be stabilized with grout or another matrix prior to being shipped for disposal (FEIS, page B-22); DOE assumed contact-handled waste would be packaged in a 0.2-m^3 (55-gallon) drum or a standard waste box (SWB) having an internal volume of 1.88 m^3 (FEIS, B-22, -24, and -26)

- FEIS Table B-7 provides the inventory for all CH other waste resulting from the exhumation of the SDA. The non-SNAP portion was determined by subtracting out that portion of the inventory due to the SNAP as represented in Table 11 of this appendix

A.3.3.2 Other GTCC Waste from Mo-99 Production

The production of Mo-99 results in GTCC ‘other waste’ with an estimated volume that would be generated over a 71-year period of 390 m^3 (DOE, 2016; page B-5). There are two potential processes suggested for Mo-99 production (Argonne 2010, page 53). One process uses the Missouri University Research Reactor (MURR), which uses low-enriched uranium and results in an annual volume of waste of 0.5 m^3 (i.e., 35 m^3 over a 71-year period). The other process is the Medical Isotope Production System (MIPS), which is estimated to generate an annual volume of 5 m^3 (i.e., 355 m^3 over a 71-year period).

Table A-11 Inventory for 1,200 m^3 of GTCC Other Waste (Contact-handled) from Potential Exhumation of the SDA (SNAP) (Waste Stream #10 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3					None
H-3	12					None
Pu-241	14	780	7.6×10^{-3}	0.65	360	3,500 nCi/g
Cm-244	18					100 nCi/g
Sr-90	29					7,000 Ci/m^3
Cm-243	29					100 nCi/g
Cs-137	30					4,600 Ci/m^3
Pu-238	88	21,000	1.2	17	9,600	100 nCi/g
Ni-63	96					7,000 Ci/m^3
Am-241	430					100 nCi/g
C-14	5,700					80 Ci/m^3
Pu-240	6,500	12	0.055	0.010	5.8	100 nCi/g
Cm-245	8,500					100 nCi/g
Pu-239	24,000	18	0.29	0.015	8.2	100 nCi/g
Ni-59	75,000					220 Ci/m^3
U-233	160,000					None
U-234	240,000					None
Tc-99	250,000					3 Ci/m^3
Np-237	2.1×10^6					100 nCi/g
I-129	1.6×10^7					0.08 Ci/m^3
U-236	2.3×10^7					None
U-235	7.0×10^8					None
U-238	4.5×10^9					None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

Table A-12 Inventory for 400 m^3 of GTCC Other Waste (Contact-handled) from Potential Exhumation of the SDA (Waste Stream #11 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	6.5	5.8×10^{-6}	0.016		None
H-3	12	200	2.1×10^{-5}	0.50		None
Pu-241	14	1,900	0.019	4.8	2,700	3,500 nCi/g
Cm-244	18	4.9×10^{-3}	6.0×10^{-8}	1.2×10^{-5}	6.8×10^{-3}	100 nCi/g
Sr-90	29	2.8	2.0×10^{-5}	7.0×10^{-3}		7,000 Ci/m^3
Cm-243	29	7.4×10^{-6}	1.4×10^{-10}	1.8×10^{-8}	1.0×10^{-5}	100 nCi/g
Cs-137	30	22	2.5×10^{-4}	0.55		4,600 Ci/m^3
Pu-238	88	160	9.2×10^{-3}	0.39	220	100 nCi/g
Ni-63	96	3.7	6.3×10^{-5}	9.2×10^{-3}		7,000 Ci/m^3
Am-241	430	0.012	3.5×10^{-6}	3.0×10^{-5}	0.017	100 nCi/g
C-14	5,700	4.4	9.9×10^{-4}	0.011		80 Ci/m^3
Pu-240	6,500	32	0.14	0.081	45	100 nCi/g
Cm-245	8,500	0.0	0.0			100 nCi/g
Pu-239	24,000	31	0.50	0.078	43	100 nCi/g
Ni-59	75,000	0.033	4.1×10^{-4}	8.2×10^{-5}		220 Ci/m^3
U-233	160,000	0.0	0.0			None
U-234	240,000	9.7×10^{-3}	1.5×10^{-3}	2.4×10^{-5}		None
Tc-99	250,000	1.0×10^{-3}	5.9×10^{-5}	2.5×10^{-6}		3 Ci/m^3
Np-237	2.1×10^{6}	3.4×10^{-9}	4.8×10^{-9}	8.5×10^{-12}	4.7×10^{-9}	100 nCi/g
I-129	1.6×10^{7}	2.9×10^{-3}	0.016	7.2×10^{-6}		0.08 Ci/m^3
U-236	2.3×10^{7}	0.0	0.0			None
U-235	7.0×10^{8}	4.8×10^{-4}	0.22	1.2×10^{-6}		None
U-238	4.5×10^{9}	0.010	30	2.5×10^{-5}		None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

The GTCC 'other waste' from the MIPS waste stream contains Pu-239 slightly above the Part 61 concentration limit (i.e. Pu-239 concentration of 140 nCi/g) and the remainder of the radionuclides are in concentrations typical of Class A waste. However, the MIPS waste stream contains a very large amount of U-235 (i.e., overall amount of 1,700 kg of U-235 representing an average 1.9 kg in a waste package). This is LEU but is estimated at 18% enrichment. The GTCC 'other waste' from the MURR process is similar to the waste produced by the MIPS process (i.e., Pu-239 exceeding the Part 61 limit and remaining radionuclides typical of Class A waste), with the one exception that the MURR process does not produce uranium waste (see Tables A-13 and 14).

Table A-13 provides the characteristics of the remote-handled GTCC other waste from the potential production of Mo-99 using the MURR process. The basis for the information provided in Table A-13 is:

- Volume of remoted handled GTCC other waste using the MURR process for Mo-99 production is 35 m^3 (Argonne 2010, page 53)

- DOE assumed that other waste would be stabilized with grout or another matrix prior to being shipped for disposal (FEIS, page B-22); DOE assumed remote-handled waste (other than activated metal waste) would be packaged in 0.2-m^3 (55-gallon) drums (FEIS, B-22 and -26)

- Inventory based on Table 4-4 (Argonne 2010)

Table A-14 provides the characteristics of the remote-handled GTCC other waste from the potential production of Mo-99 using the MIPS process. The basis for the information provided in Table A-14 is:

- Volume of remoted handled GTCC other waste using the MIPS process for Mo-99 production is 355 m^3 (Argonne 2010, page 53)

- DOE assumed that other waste would be stabilized with grout or another matrix prior to being shipped for disposal (FEIS, page B-22); DOE assumed remote-handled waste (other than activated metal waste) would be packaged in 0.2-m^3 (55-gallon) drums (FEIS, B-22 and -26)

- Inventory based on Table 4-4 (Argonne 2010)

Table A-13 Inventory for 35 m^3 of GTCC Other Waste (Remote-handled) from Potential Mo-99 Production (MURR process) (Waste Stream #12 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	0.0	0.0			None
H-3	12	0.0	0.0			None
Pu-241	14	9.7×10^{-3}	9.5×10^{-8}	2.8×10^{-4}	0.15	3,500 nCi/g
Cm-244	18	0.0	0.0			100 nCi/g
Sr-90	29	45,000	0.33	1,300		7,000 Ci/m^3
Cm-243	29	0.0	0.0			100 nCi/g
Cs-137	30	49,000	0.57	1,400		4,600 Ci/m^3
Pu-238	88	1.7×10^{-5}	1.0×10^{-9}	4.9×10^{-7}	2.7×10^{-4}	100 nCi/g
Ni-63	96	0.0	0.0			7,000 Ci/m^3
Am-241	430	6.5×10^{-8}	1.9×10^{-11}	1.9×10^{-9}	1.0×10^{-6}	100 nCi/g
C-14	5,700	5.6×10^{-5}	1.3×10^{-8}	1.6×10^{-6}		80 Ci/m^3
Pu-240	6,500	0.038	1.7×10^{-4}	1.1×10^{-3}	0.60	100 nCi/g
Cm-245	8,500	0.0	0.0			100 nCi/g
Pu-239	24,000	19	0.31	0.54	300	100 nCi/g
Ni-59	75,000	0.0	0.0			220 Ci/m^3
U-233	160,000	0.0	0.0			None
U-234	240,000	0.0	0.0			None
Tc-99	250,000	4.2	0.25	0.12		3 Ci/m^3
Np-237	2.1×10^6	4.9×10^{-5}	7.0×10^{-5}	1.4×10^{-6}	7.8×10^{-4}	100 nCi/g
I-129	1.6×10^7	9.1×10^{-3}	0.052	2.6×10^{-4}		0.08 Ci/m^3
U-236	2.3×10^7	0.0	0.0			None
U-235	7.0×10^8	0.0	0.0			None
U-238	4.5×10^9	0.0	0.0			None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

Table A-14 Inventory for 355 m^3 of GTCC Other Waste (Remote-handled) from Potential Mo-99 Production (MIPS process) (Waste Stream #13 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	0.0	0.0			None
H-3	12	170	1.8×10^{-5}	0.48		None
Pu-241	14	4.9	4.8×10^{-5}	0.014	7.7	3,500 nCi/g
Cm-244	18	0.0	0.0			100 nCi/g
Sr-90	29	46,000	0.34	130		7,000 Ci/m^3
Cm-243	29	0.0	0.0			100 nCi/g
Cs-137	30	46,000	0.53	130		4,600 Ci/m^3
Pu-238	88	0.60	3.5×10^{-5}	1.7×10^{-3}	0.94	100 nCi/g
Ni-63	96	0.0	0.0			7,000 Ci/m^3
Am-241	430	2.2×10^{-3}	6.4×10^{-7}	6.2×10^{-6}	3.4×10^{-3}	100 nCi/g
C-14	5,700	150	0.034	0.42		80 Ci/m^3
Pu-240	6,500	1.6	7.0×10^{-3}	4.5×10^{-3}	2.5	100 nCi/g
Cm-245	8,500	0.0	0.0			100 nCi/g
Pu-239	24,000	91	1.5	0.26	144	100 nCi/g
Ni-59	75,000	0.0	0.0			220 Ci/m^3
U-233	160,000	0.0	0.0			None
U-234	240,000	8.5×10^{-3}	1.3×10^{-3}	2.4×10^{-5}		None
Tc-99	250,000	7.0	0.41	0.020		3 Ci/m^3
Np-237	2.1×10^{6}	6.7×10^{-3}	9.6×10^{-3}	1.9×10^{-5}	0.010	100 nCi/g
I-129	1.6×10^{7}	0.011	0.063	3.1×10^{-5}		0.08 Ci/m^3
U-236	2.3×10^{7}	0.28	4.4	7.9×10^{-4}		None
U-235	7.0×10^{8}	3.6	1,700	0.010		None
U-238	4.5×10^{9}	2.3	6,900	6.5×10^{-3}		None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

A.3.3.3 Other GTCC-like Waste from Decommissioning Activities at the WVDP

There is the potential for decommissioning activities at West Valley site that would result in the generation of GTCC-like other waste from the MPPB, the West Valley Tank Farm (WVTF), and TRU waste from debris associated with cleanout of mechanical processing cells of the former Nuclear Fuel Services, Inc. reprocessing plant (DOE 2016, page B-4). This GTCC-like "other waste" is estimated to represent 980 m^3 (220 m^3 is contact-handled and 760 m^3 is remote-handled) and would be generated if a decision is made to initiate these decommissioning activities.

Table A-15 provides the characteristics of the contact-handled GTCC-like other waste from activities associated with decommissioning of the MPPB and the WVTF at the WVDP. The basis for the information provided in Table A-15 is:

- Volume of contact-handled GTCC-like other waste is 220 m^3 (FEIS, Table B-3)

- DOE assumed that other waste would be stabilized with grout or another matrix prior to being shipped for disposal (FEIS, page B-22); DOE assumed contact-handled waste would be packaged in a 0.2-m^3 (55-gallon) drum or a standard waste box having an internal volume of 1.88 m^3 (FEIS, B-22, -24, and -26)

- Inventory from FEIS Table B-7 for other GTCC-like waste is the total for the decommissioning waste from activities at the WVDP and waste generated from a potential program at Oak Ridge National Laboratory for the production of Pu-238, which is listed under the ORR site in Table B-3 of the FEIS. The inventory for the ORR other GTCC-like waste is provided in Table 4-5 (Argonne, 2010). The inventory for WVDP decommissioning activities is obtained by subtracting the inventory for the ORR site (PU-238 production) as represented in Table 17 of this appendix from the overall inventory amount provided in Table B-7 of the FEIS.

Table A-16 provides the characteristics of the remote-handled GTCC-like other waste from activities associated with decommissioning of the MPPB and the WVTF at the WVDP. The basis for the information provided in Table A-16 is:

- Volume of remote-handled GTCC-like other waste is 760 m^3 (FEIS, Table B-3)

- DOE assumed that other waste would be stabilized with grout or another matrix prior to being shipped for disposal (FEIS, page B-22); DOE assumed remote-handled waste (other than activated metal waste) would be packaged in 0.2-m^3 (55-gallon) drums (FEIS, B-22 and -26)

- Inventory from FEIS Table B-7 for other GTCC-like waste is the total for the decommissioning waste from activities at the WVDP and waste generated from a potential program at Oak Ridge National Laboratory for the production of Pu-238, which is listed under the ORR (Oak Ridge Reservation) site in Table B-3 of the FEIS. The inventory for the ORR other GTCC-like waste is provided in Table 4-5 (Argonne, 2010). The inventory for WVDP decommissioning activities is obtained by subtracting the inventory for the ORR site (PU-238 production) as represented in Table 18 of this appendix from the overall inventory amount provided in Table B-7 of the FEIS.

Table A-15 Inventory for 220 m^3 of GTCC-like Other Waste (Contact-handled) from Potential Decommissioning Activities at the WVDP (Waste Stream #14 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	2×10^{-4}	1.8×10^{-10}	9.1×10^{-7}		None
H-3	12	0.11	1.1×10^{-8}	5.0×10^{-4}		None
Pu-241	14	9,200	0.090	42	23,000	3,500 nCi/g
Cm-244	18	1.0	1.2×10^{-5}	4.6×10^{-3}	2.5	100 nCi/g
Sr-90	29	6.1	4.4×10^{-5}	0.028		7,000 Ci/m^3
Cm-243	29	0.039	7.6×10^{-7}	1.8×10^{-4}	0.1	100 nCi/g
Cs-137	30	3.3	3.8×10^{-5}	0.015		4,600 Ci/m^3
Pu-238	88	560	0.033	2.6	1,400	100 nCi/g
Ni-63	96	0.0	0.0			7,000 Ci/m^3
Am-241	430	1,400	0.4	6.3	3,500	100 nCi/g
C-14	5,700	5.9	1.3×10^{-3}	0.027		80 Ci/m^3
Pu-240	6,500	310	1.4	1.4	780	100 nCi/g
Cm-245	8,500	0.0	0.0			100 nCi/g
Pu-239	24,000	390	6.2	1.8	1,000	100 nCi/g
Ni-59	75,000	0.033	4.1×10^{-4}	1.5×10^{-4}		220 Ci/m^3
U-233	160,000	4.1	0.42	0.019		None
U-234	240,000	19	3.0	0.086		None
Tc-99	250,000	0.13	7.6×10^{-3}	5.9×10^{-4}		3 Ci/m^3
Np-237	2.1×10^6	0.022	0.031	1.0×10^{-4}	0.056	100 nCi/g
I-129	1.6×10^7	0.0	0.0			0.08 Ci/m^3
U-236	2.3×10^7	0.024	0.37	1.1×10^{-4}		None
U-235	7.0×10^8	8.0×10^{-3}	3.7	3.6×10^{-5}		None
U-238	4.5×10^9	0.039	120	1.8×10^{-4}		None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

Table A-16 Inventory for 760 m^3 of GTCC-like Other Waste (Remote-Handled) from Potential Decommissioning Activities at the WVDP (Waste Stream #15 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3	3.0×10^{-4}	2.7×10^{-10}	4.0×10^{-7}		None
H-3	12	0.17	1.8×10^{-8}	2.2×10^{-4}		None
Pu-241	14	15,000	0.15	20	11,000	3,500 nCi/g
Cm-244	18	91	1.1×10^{-3}	0.12	66	100 nCi/g
Sr-90	29	45,000	0.33	59		7,000 Ci/m^3
Cm-243	29	3.9	7.6×10^{-5}	5.1×10^{-3}	2.8	100 nCi/g
Cs-137	30	320,000	3.7	420		4,600 Ci/m^3
Pu-238	88	1,000	0.060	1.3	720	100 nCi/g
Ni-63	96	0.0	0.0			7,000 Ci/m^3
Am-241	430	2,600	0.76	3.4	1,900	100 nCi/g
C-14	5,700	9.0	2.0×10^{-3}	0.012		80 Ci/m^3
Pu-240	6,500	510	2.2	0.67	370	100 nCi/g
Cm-245	8,500	0.0	0.0			100 nCi/g
Pu-239	24,000	640	10	0.84	470	100 nCi/g
Ni-59	75,000	0.051	6.3×10^{-4}	6.7×10^{-5}		220 Ci/m^3
U-233	160,000	6.4	0.66	8.4×10^{-3}		None
U-234	240,000	29	4.6	0.038		None
Tc-99	250,000	3.2	0.19	4.2×10^{-3}		3 Ci/m^3
Np-237	2.1×10^{6}	0.60	0.86	7.9×10^{-4}	0.44	100 nCi/g
I-129	1.6×10^{7}	3.8×10^{-3}	0.022	5.0×10^{-6}		0.08 Ci/m^3
U-236	2.3×10^{7}	0.036	0.56	4.7×10^{-5}		None
U-235	7.0×10^{8}	0.014	6.5	1.8×10^{-5}		None
U-238	4.5×10^{9}	0.073	220	9.6×10^{-5}		None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

A.3.3.4 Other GTCC-like Waste from Pu-238 Production

DOE is planning a new program for producing Pu-238, a radionuclide that is used to produce heat and electricity in harsh and remote environments, including in unmanned spacecraft and satellites (Argonne 2010, page 54). DOE estimated that the volume of GTCC-like "other waste" generated from the Pu-238 production would be 380 m^3 (i.e., 120 m^3 contact-handled waste and 260 m^3 remote-handled waste) [Argonne 2010, page 55].

Table A-17 provides the characteristics of the contact-handled other GTCC-like waste from the future production of Pu-238. The basis for the information provided in Table A-17 is:

- Volume of contact-handled other GTCC-like waste is 120 m^3 (Argonne 2010, page 55)
- DOE assumed that other waste would be stabilized with grout or another matrix prior to being shipped for disposal (FEIS, page B-22); DOE assumed contact-handled waste would be packaged in a 0.2-m^3 (55-gallon) drum or a standard waste box (SWB) having an internal volume of 1.88 m^3 (FEIS, B-22, -24, and -26)
- Inventory based on Table 4-5 (Argonne 2010)

Table A-18 provides the characteristics of the remote-handled other GTCC-like waste from the future production of Pu-238. The basis for the information provided in Table A-18 is:

- Volume of remote-handled other GTCC-like waste is 260 m^3 (Argonne 2010, page 55)
- DOE assumed that other waste would be stabilized with grout or another matrix prior to being shipped for disposal (FEIS, page B-22); DOE assumed remote-handled waste (other than activated metal waste) would be packaged in 0.2-m^3 (55-gallon) drums (FEIS, B-22 and -26)
- Inventory based on Table 4-5 (Argonne 2010)

Table A-17 Inventory for 120 m^3 of GTCC-like Other Waste (Contact-handled) from Potential Production of Pu-238 (Waste Stream #16 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3					None
H-3	12					None
Pu-241	14	110	1.1×10^{-3}	0.92	510	3,500 nCi/g
Cm-244	18					100 nCi/g
Sr-90	29					7,000 Ci/m^3
Cm-243	29					100 nCi/g
Cs-137	30	1.2×10^{-7}	1.4×10^{-12}	1.0×10^{-9}		4,600 Ci/m^3
Pu-238	88	6.0	3.5×10^{-4}	0.050	28	100 nCi/g
Ni-63	96					7,000 Ci/m^3
Am-241	430	11	3.2×10^{-3}	0.092	51	100 nCi/g
C-14	5,700					80 Ci/m^3
Pu-240	6,500	5.0	0.022	0.042	23	100 nCi/g
Cm-245	8,500					100 nCi/g
Pu-239	24,000	13	0.21	0.11	60	100 nCi/g
Ni-59	75,000					220 Ci/m^3
U-233	160,000					None
U-234	240,000					None
Tc-99	250,000					3 Ci/m^3
Np-237	2.1×10^6					100 nCi/g
I-129	1.6×10^7					0.08 Ci/m^3
U-236	2.3×10^7					None
U-235	7.0×10^8					None
U-238	4.5×10^9					None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.

Table A-18 Inventory for 260 m^3 of GTCC-like Other Waste (Remote-handled) from Potential Pu-238 Production (Waste Stream #17 in Table A-1)

Nuclides	Half-Life (years)	Inventory		Concentration[a]		Class C Limits
		Ci	kg	Ci/m^3	nCi/g	
Co-60	5.3					None
H-3	12					None
Pu-241	14	39	3.8×10^{-4}	0.15	83	3,500 nCi/g
Cm-244	18					100 nCi/g
Sr-90	29	6,200	0.045	24		7,000 Ci/m^3
Cm-243	29					100 nCi/g
Cs-137	30	18,000	0.21	69		4,600 Ci/m^3
Pu-238	88	880	0.052	3.4	1,900	100 nCi/g
Ni-63	96					7,000 Ci/m^3
Am-241	430	12	3.5×10^{-3}	0.046	26	100 nCi/g
C-14	5,700					80 Ci/m^3
Pu-240	6,500	0.21	9.3×10^{-4}	8.1×10^{-4}	0.45	100 nCi/g
Cm-245	8,500					100 nCi/g
Pu-239	24,000	0.49	7.9×10^{-3}	1.9×10^{-3}	1.0	100 nCi/g
Ni-59	75,000					220 Ci/m^3
U-233	160,000					None
U-234	240,000					None
Tc-99	250,000					3 Ci/m^3
Np-237	2.1×10^6	1.7	2.4	6.5×10^{-3}	3.6	100 nCi/g
I-129	1.6×10^7					0.08 Ci/m^3
U-236	2.3×10^7					None
U-235	7.0×10^8					None
U-238	4.5×10^9					None

a – Concentrations for transuranic radionuclides in nCi/g based on an assumed density of 1,800 kg/m^3 for packaged waste.