ABSTRACT

PLATFOOT, ELIZABETH ANNE LONG. Greater Than Class C Low Level Radioactive Waste Management Strategies. (Under the direction of Dr. Man-Sung Yim).

Greater Than Class C (GTCC) waste management is not a new problem, but the issue of spent, or used, nuclear fuel has been the main focus of radioactive waste management for many years. Relatively small volumes of GTCC waste and the lower longevity and activity levels of GTCC have caused used nuclear fuel management to take precedence. GTCC waste is to be stored in "…a geologic repository … unless proposals for disposal of such waste in a disposal site licensed pursuant to this part are approved by the Commission", as stated in 10 CFR 61. Since the future of the Yucca Mountain Project has become less certain, and the currently allowed storage space practically all accounted for, a new way to manage GTCC waste may be worth investigating.

The goal of this research is to provide a new look at GTCC waste management strategies. This was accomplished by first determining the composition and constituents of GTCC waste. GTCC waste is modeled as entirely activated metals for this study. Currently used treatment methods for spent fuel, mixed waste, and other low level waste are studied and applied to GTCC waste, and alternative disposal options are explored. GTCC activated metals are internal reactor components which have been activated throughout by the neutron flux in operating reactors. The high neutron flux creates activation products throughout the material, and concentrations of carbon-14, nickel-59, nickel-63, and niobium-94 cause these components to be classified as GTCC waste. The goal of treatment application to GTCC waste is to reduce the impact of the waste by lowering the volume of waste for disposal. The treatments studied as possibilities in this study were determined to be not technologically feasible for GTCC activated metals waste, leaving disposal as the only option for this type of waste. Alternative disposal options, enhanced near surface burial or intermediate depth boreholes, were determined to be significantly cheaper than a geologic repository. The safety of these potential alternatives is examined through conservative risk assessment calculations for hypothetical reference facilities, which are chosen based on their ability to represent a variety of areas throughout the United States. The conservative modeling approach determined that engineered barriers are a necessary part of the waste disposal facility. Results from this conservative modeling approach and a sensitivity analysis show that enhanced near surface burial and intermediate depth boreholes result in the expected public dose from the reference disposal facilities fall well below the EPA limits for radionuclides in drinking water. Both enhanced near surface burial and intermediate depth boreholes are estimated to be feasible alternatives disposal options for GTCC waste.

Greater Than Class C Low Level Radioactive Waste Management Strategies

by
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DEDICATION

For my dad.

BIOGRAPHY

Elizabeth (Beth) Platfoot was born in Dallas, TX and grew up in Blountville, TN. She graduated from Sullivan Central High School in 2003, and then continued on to Tennessee Technological University as a chemical engineering major. While at Tennessee Tech, she participated in the TTU chapter of the American Institute of Chemical Engineers and served as Vice President for Omega Chi Epsilon, the chemical engineering honor society. Beth spent 14 months after her sophomore year in Tennessee Tech's co-operative education program at the Weyerhaeuser Papermill in Hawesville, KY, where she learned valuable laboratory skills as well as practical work experience. After taking a radiochemistry class and attending the American Chemical Society and Department of Energy's Radiochemistry Summer Program at San Jose State University, she decided to pursue graduate work in nuclear engineering.

Beth was chosen as a 2008 fellow for the Advanced Fuel Cycle Initiative's University Research Fellowship program, and continued on to North Carolina State University as a nuclear engineer to complete research work under this fellowship. During this time, she completed two summer internships at Oak Ridge National Laboratory.

In June 2010 Beth began full time work as a nuclear engineer at Northrop Grumman Shipbuilding in Newport News, VA as part of the Radiological and Waste Group of the Reactor Services division.

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AISI	American Iron and Steel Institute
BWR	Boiling Water Reactor
CARLA	Centrale Anlage zur Recyklieren Leichtaktiver Abfälle (Central Plant to Recycle Slightly Radioactive Wastes)
CEP	Catalytic Extraction Process
CFR	Code of Federal Regulations
CH	Contact Handled
CPU	Catalytic Processing Unit
D&D	Decontamination and Decommissioning
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor II
EPA	Environmental Protection Agency
FTE	Full Time Effective Workers
GE	General Electric
GTCC	Greater Than Class C
HLW	High Level Waste
ICRP	International Commission on Radiological Protection
INL	Idaho National Laboratory

LLW	Low Level Waste
MMT	Molten Metal Technologies
MSRE	Molten Salt Reactor Experiment
MTHM	Metric Tons Heavy Metal
NRC	Nuclear Regulatory Commission
O&M	Operation and Maintenance
ORNL	Oak Ridge National Laboratory
PLCC	Planning Life Cycle Cost
PWR	Pressurized Water Reactor
Q-CEP	Quantum - Catalytic Extraction Process
RH	Remote Handled
S&M	Surveillance and Maintenance
SEG	Scientific Ecology Group
SNF	Spent Nuclear Fuel
SNL	Sandia National Laboratory
TRU	Transuranic
UNF	Used Nuclear Fuel
USDA	United States Department of Agriculture
WIPP	Waste Isolation Pilot Plant

1. Introduction

The question of how to manage Greater Than Class C (GTCC) Low Level radioactive waste has been around for a number of years. Much of the research done on GTCC waste management was done in the 1980s, along with high level waste (HLW) studies, but soon the issue of HLW and spent (or used) nuclear fuel disposal took precedence. One reason for this is the small volumes of GTCC waste when compared to HLW or even other classes of low level waste. Another is the lower longevity and activity levels of GTCC waste as compared to HLW. When a repository site (Yucca Mountain) was located for HLW for storage in 1987, the default for GTCC waste was also set for storage in the repository. However, as the future of the Yucca Mountain Project becomes less certain, and the currently allowed storage space practically all accounted for, a new way to manage GTCC waste may be worth investigating.

The goal of this research is to provide GTCC waste management strategies and to propose alternatives to deep geologic disposal. This goal will be met through three major Objectives: Objective one – the determination of the composition and constituents of GTCC waste, Objective two – the application of possible treatment methods to GTCC waste and the determination of their feasibility and practicality, and Objective three – the study of disposal options for GTCC waste which could replace the default geologic repository disposal. Chapters one and two will address Objective one and determine the GTCC composition and constituents as well as the scope of the thesis. These chapters will primarily consist of data from previous studies, and will combine all the data useful for GTCC waste management studies into one location. Chapter three focuses on Objective two by examining possible treatment techniques. This chapter will study previously and currently used methods for radioactive waste treatments, and apply them to TCC waste. Chapter four defines the alternative disposal options which are currently being considered by the DOE, and Chapters five and six explore the alternative disposal options through the analysis of costs and potential risk to the public through expected radiation exposure. Overall, this study will be a useful addition to GTCC waste management strategies, as it will combine data from previous studies with information from new calculations to help aid in GTCC waste management decisions.

1.1 Waste Classification System

In order to create a waste management scheme for GTCC waste, it is first necessary to understand how radioactive waste is classified as well as what types of material are expected in this classification of waste.

The radioactive waste classification scheme in the United States can be somewhat difficult to understand. There are four basic radioactive waste categories: high level waste (HLW), transuranic (TRU) waste, uranium mill tailings, and low level waste (LLW). HLW includes by definition all waste from reprocessing spent fuel, but can also be extended to include the spent, or used, fuel itself. TRU waste contains elements with atomic number greater than that of uranium (92) in amounts greater than 100 nCi/g. Uranium mill tailings are the by-product of uranium extraction from uranium ore. Finally, LLW is defined as anything that does not fit into the other three categories [1].

This broad definition means there is large variety of materials and thus of a variety of corresponding nuclides which make up LLW. In order to help in determining storage and disposal options, LLW is divided into four classes: A, B, C, and Greater-than-Class-C. These classes are divided according to the presence of specific nuclides in the waste and their concentrations. These limits are given in 10 CFR 61.55 in two tables, shown below in Table 1 and Table 2.

Table 1: NRC LLW Concentration Limits [2]

Radionuclide	**Concentration**
C-14	8 Ci/m^3
C-14 in activated metal	80 Ci/m^3
Ni-59 in activated metal	220 Ci/m^3
Nb-94 in activated metal	0.20 Ci/m^3
Tc-99	3 Ci/m^3
I-129	0.08 Ci/m^3
Alpha emitting transuranics with a half-life > 5 years	100 nCi/g
Pu-241	3,500 nCi/g
Cm-242	20,000 nCi/g

Table 2: Additional NRC LLW Concentration Limits [2]

Radionuclide	**Concentration (Ci/m^3)**		
	Column 1	**Column 2**	**Column 3**
All nuclides with < 5 year half-life	700	No limit	No limit
H-3	40	No limit	No limit
Co-60	700	No limit	No limit
Ni-63	3.5	70	700
Ni-63 in activated metal	35	700	7,000
Sr-90	0.04	150	7,000
Cs-137	1	44	4,600

The following lists show the limits for each LLW class A, B, and C based on the values given in Table 1 and Table 2.

For waste to be classified as Class A, one of the following must be true:

1) No nuclides from Table 1 or Table 2 are present
2) Concentrations of nuclides in Table 1 are less than or equal to 0.1 times the value in Table 1
3) Concentrations of nuclides in Table 2 are less than or equal to 0.1 times the value in Column 1
4) Any combination of 2) and 3)

For waste to be classified as Class B, one of the following must be true:

1) Concentrations of nuclides in Table 2 are greater than Column 1 but less than Column 2
2) Concentrations of nuclides in Table 1 are less than or equal to 0.1 times the value in Table 1 in addition to condition 1)

For waste to be classified as Class C, one of the following must be true:

1) Concentrations of nuclides in Table 1 are greater than 0.1 times the value, but less than or equal to the value in Table 1
2) The only nuclides present are those in Table 2 with concentrations greater than Column 2 but less than or equal to Column 3
3) Case 2 is true in addition to nuclides present with concentrations in Table 1 less than or equal to 0.1 times the value in Table 1

GTCC waste exceeds the values in either Table 1 or Column 3 of Table 2, and thus contains the highest concentrations of radioactive nuclides in LLW. These concentration limits were established based on half lives of the nuclides and the hazards associated with potential exposure to humans. Due to larger concentrations of these nuclides, GTCC is the most potentially hazardous to humans of all LLW. It is because of this that GTCC management in the United States is currently limited to disposal in "…a geologic repository … unless proposals for disposal of such waste in a disposal site licensed pursuant to this part are approved by the Commission" [2].

The NRC classification system dictates what limiting nuclides can be present in GTCC waste. The next step in GTCC waste characterization is to determine what types of materials are most likely to be present and typical concentrations of the limiting nuclides in these materials.

1.2 Greater-Than-Class-C Low-Level Waste Categories

Commercial GTCC waste is generally divided into three categories based on its source: nuclear utility waste, sealed sources, and other generator waste. There is another category of waste which is similar to GTCC, known as DOE GTCC-like waste. NRC rulings, and thus the LLW classification scheme, do not apply to government wastes. The DOE GTCC-like category results from applying the limits of the NRC radioactive waste classification to DOE waste.

1.2.1 Nuclear Utility Waste

Nuclear utility waste is the GTCC waste that results from normal nuclear power plant operations or from the decommissioning of nuclear reactors. This waste includes process wastes, such as water filter cartridges and ion-exchange resins, and activated metals. Most process wastes can be classified as Class C, with only a small amount of GTCC process waste produced. Activated metals, consisting of reactor internals, make up the majority of GTCC nuclear utility waste [3].

1.2.2 Sealed Sources

Sealed sources are small, high-activity radioactive materials which are typically encased in metal containers [3]. These sources are used in industrial and medical settings, and can be used for many applications including oil well logging, density and moisture content measurements, thickness measurements for pipes and welds, fill-level gauges, and medical irradiation. Some nuclides used for these sources are Am-241, Am-243, Cm-244, Pu-238, Pu-239, Pu-240, and Cs-137 [3]. Other nuclides are also used, but these are the most common nuclides which are in sources that will become GTCC waste. The majority of the volume of these GTCC sealed sources can be contributed to high-activity Cs-137 irradiator sources, which represent about 60% of the volume of sealed sources. This is primarily due to activity limitations in packaging requirements. In sheer numbers, sources made from three isotopes: Am-241, Pu-238, and Pu-239, make up 98.5% of the GTCC seal source inventory.

1.2.3 Other Generator Waste

Other generator GTCC waste is waste produced by any commercial generator other than nuclear utilities and sealed source licensees. This can include sealed source manufacturers, industrial R&D, fuel fabrication and irradiation research laboratories, and research reactors. Only two generators reported having GTCC waste, according to a 2007 study by Sandia National Laboratory (SNL). One industrial R&D firm, and one sealed source manufacturer in Texas reported GTCC waste production. Only the industrial R&D firm planned to continue producing GTCC waste, which they reported as miscellaneous TRU waste. Stored inventory consists mostly of TRU waste from industrial R&D and Am-241 and beryllium contaminated dry active waste from sealed source manufacturing, with a very small amount of contaminated lead and sealed source waste.

1.2.4 DOE GTCC-like Waste

DOE GTCC-like waste is generated by the national laboratories in various forms. About of 82% stored and 75% of projected DOE GTCC-like waste volumes is TRU or mixed TRU waste which is not eligible for storage in WIPP. The WIPP facility in Carlsbad, NM is only available for defense waste storage. Much of the DOE GTCC-like waste is in the form of remote handled (RH) or contact handled (CH) waste. These wastes are primarily from glovebox and hot cell work. The only difference in CH and RH waste is the external radiation dose on contact with the item. RH waste has a much higher gamma radiation dose than CH waste. Selected stored and projected DOE waste inventories are shown in Table 3 and Table 4, respectively. These were selected from a more comprehensive list in the SNL GTCC waste study, and represent nearly 100% of the volume of both the stored and projected wastes [3].

Table 3: Highest Volume Stored DOE GTCC Waste [3]

DOE Site	Waste Type
West Valley	TRU waste
ORNL	RH Process waste
INL	Radioactive only waste items
West Valley	Mixed TRU waste
INL	Mixed TRU waste
BWXT-Lynchburg, VA (DOE waste at a commercial facility)	TRU waste
ORNL	RH Scrap metal
ORNL, INL	Activated metals
ORNL	MSRE salts and charcoal canisters

Table 4: Highest Volume Projected DOE GTCC Waste [3]

DOE Site	Waste Type
Proposed Radioisotope Power Systems (RPS) Project	CH and RH TRU waste
West Valley	D&D Project
West Valley	D&D Project: TRU waste
West Valley	RH waste operations
West Valley	Process Plant Demolition
INL	Activated metals

It can be observed that even though the majority of the waste is TRU waste, the form of waste varies from TRU contaminated scrap metals to TRU mixed waste, with many of the wastes in an unidentified form. DOE GTCC-like waste certainly consists of a variety of different waste forms.

1.3 GTCC Waste Generation

Estimates for GTCC stored and projected amounts by generator were based on the best available information in 2007 in the study completed for the DOE by SNL [3]. The data is summarized below in Table 5 and shown also in Figure 1 (by activity of waste produced) and Figure 2 (by volume of waste produced), which were generated using the supplied data.

Table 5: Waste Generation by Type [3]

Waste Type	Stored		Projected		Total Stored and Projected	
	Volume (m^3)	Activity (MCi)	Volume (m^3)	Activity (MCi)	Volume (m^3)	Activity (MCi)
Activated metals	58	3.5	810	110	868	114
Sealed Sources	0	0	1700	2.4	1700	2.4
Other Generator	76	0.0076	1	0.00023	77	0.00783
DOE GTCC-like	870	11	2100	20	2970	31
Total	***1004***	***15***	***4611***	***132***	***5615***	***147***

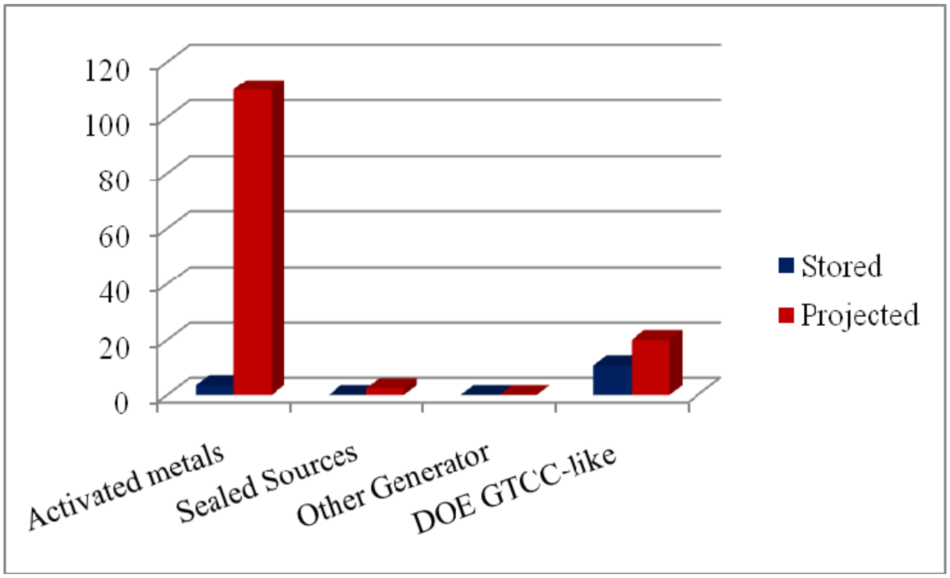

Figure 1: GTCC Waste Generation by Activity

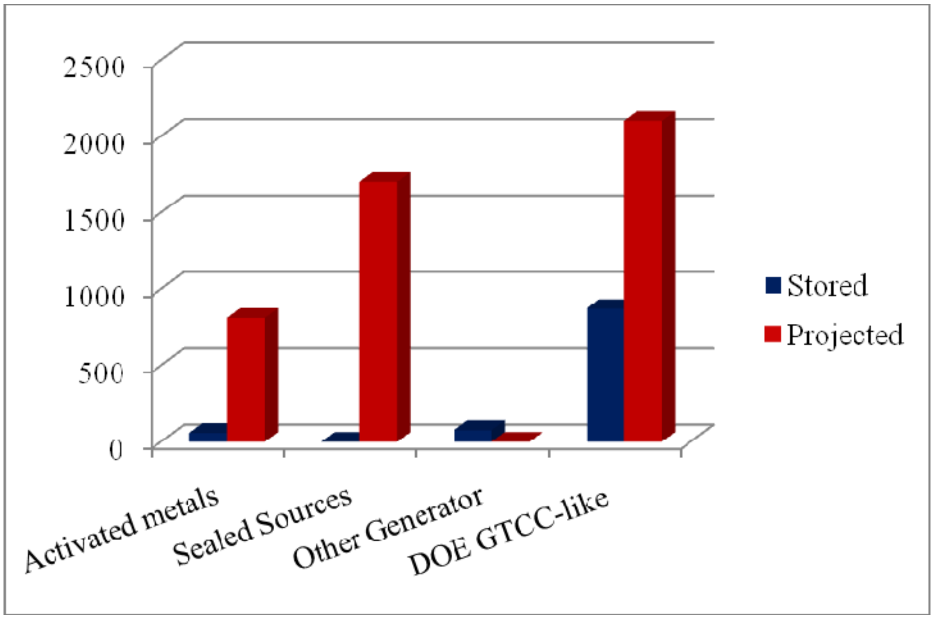

Figure 2: GTCC Waste Generation by Volume

Projections for volumes and activities for all GTCC waste are through 2062. Assumptions in this data are primarily in the estimates for nuclear utility activities and sealed source volumes. Data provided in the SNL report assumes that all nuclear utility waste is activated metals, and that these activated metals have decayed for six years after reactor shutdown, or to 2007, for those wastes already in storage. This lowers the total stored and projected activity from 465 MCi initially after shutdown to the 114 MCi shown in Table 5. Nuclear utility waste projections account for likely 20-year license extensions for current reactors and the six year cooling period. Waste would begin becoming available for disposal by 2035 [3].

Another assumption in this data is that there are no currently stored GTCC sealed sources, but all sealed sources held by NRC licensees that may become GTCC waste are included in the projected inventories. This is due to a lack of data from sealed source licensees on the status of the sealed sources at their facilities: whether they were in use or in onsite storage after being used [3]. Sealed sources are not considered for reuse or recycling in this data – all of them are disposed of after first use. Finally, the volumes shown for sealed sources are packaged volume of the sealed sources themselves, which are much higher than the actual volumes due to activity limitations on packaging. For example, only one Cs-137 irradiator source will be placed into a 55-gallon drum [3].

2. Generation of Activated Metal GTCC Waste

This study will focus on nuclear utility waste, which has been determined to be primarily activated metals. As shown in Figure 2, DOE GTCC-like waste has the highest volume of both stored and projected waste. However, Table 3 and Table 4 have shown that DOE GTCC waste contains a variety of materials and objects, and any stored waste may not be well characterized. For these reasons, DOE GTCC-like waste must be treated on a case-by-case basis. Sealed sources also contribute a large volume to projected GTCC waste, but large uncertainties exist in sealed source inventories, due to limited tracking in the past as well as limited projections of when these sources will realistically become available for disposal [3].

Figure 1 shows that activated metals supply by far the majority of the projected activity of GTCC waste, making a solution for managing this source of GTCC waste a concern. Volumes of activated metals could drastically increase in the future if the “nuclear renaissance” is indeed at hand. Current studies only take into account the currently operating reactors. Any new reactors would produce additional GTCC waste at the end of their lifetime, thus playing a larger part in the volume generation of GTCC waste. Because activated metal waste contains only waste which is in the metal form, stored and projected inventories can be analyzed together, instead of studying each piece on a case-by-case basis as would be necessary for DOE GTCC-like waste.

2.1 Reactor Components

Activated metal waste is produced as a result of exposure of the nuclear reactors’ internal components to a neutron flux over the lifetime of the reactor plant [4]. Both pressurized water reactors (PWRs) and boiling water reactors (BWRs) are used in the United States, and each type of reactor uses slightly different components which will become GTCC waste. Typical BWR components which become GTCC waste at decommissioning are the top fuel guide, the core shroud, fuel support castings, and the core support plate. PWR components generally considered GTCC waste are the upper core support plate, the core shroud (baffle), the lower core barrel, and the lower core support plate. The location of these components in the reactor vessel can be seen in Figure 3 [3,4].

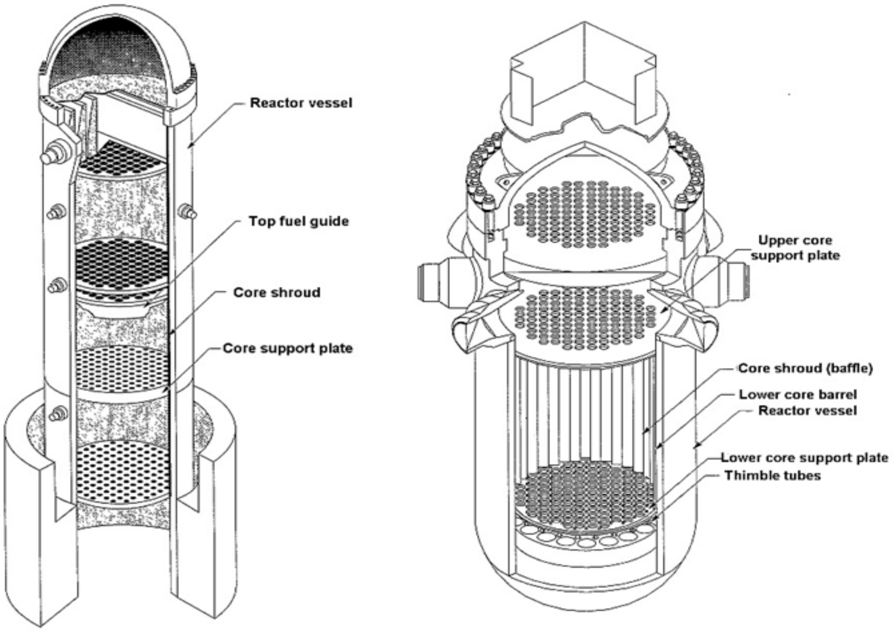

Figure 3: Location of Reactor Components for BWR (Left) and PWR (Right) [3,4]

In their 1994 GTCC study, the DOE considered components of two BWR models along with four different PWR models. The BWR models were General Electric BWR-4 and BWR-6, which are two of the larger operating BWR models. PWR models considered were Westinghouse intermediate and large size reactors, a Combustion Engineering model, and a Babcock and Wilcox model. In each model considered, all components that are expected to become GTCC waste are made of type 304 stainless steel [4].

2.2 Stainless Steel

Given that stainless steel is the primary component of GTCC activated metal waste, it is important to consider the properties of stainless steel. Stainless steel is more resistant to rusting and staining than carbon steels due to the addition of chromium. Other elements such as copper, aluminum, silicon, nickel, and molybdenum also increase corrosion resistance, but are much more useful when combined with chromium. According to the American Iron and Steel Institute (AISI), a “stainless steel” is greater than 4% chromium, while less than 4% chromium is considered to be an “alloy” [5]. The variety of additional elements which can be used to create stainless steel has led to many different alloys which are all stainless steels. The AISI has designated type numbers for each different alloy. Type 304 stainless steel is a common type of stainless steel. The composition of type 304 stainless steel can be seen in Table 6, where the remaining 66.35-70.85% is iron.

Table 6: Stainless Steel Type 304 Composition (%) [5]

Chromium Range	**Nickel Range**	**Manganese (max)**	**Silicon (max)**	**Carbon (max)**	**Sulfur (max)**	**Phosphorus (max)**
18.00-20.00	8.00-10.50	2.00	1.0	0.08	0.030	0.045

However, there are other trace elements which can be found in stainless steel which can become important to consider when the steel is exposed to a neutron flux. Elements most commonly found in trace amounts in type 304 stainless steel are lithium, nitrogen, sodium, aluminum, chlorine, potassium, calcium, scandium, titanium, vanadium, cobalt, copper, zinc, gallium, arsenic, selenium, bromine, rubidium, strontium, yttrium, zirconium, niobium, molybdenum, silver, antimony, cesium, barium, lanthanum, cerium, samarium, europium, terbium, dysprosium, holmium, ytterbium, lutetium, hafnium, tungsten, lead, thorium, and uranium [6]. These elements are present on parts per million scales in varying orders of magnitude, and the amounts will vary with individual metals. Of this comprehensive list of trace elements, only a few will be a considerable concern for GTCC waste: niobium and nitrogen primarily, with copper and molybdenum playing a lesser role.

2.3 Activation Products

Important radionuclides in activated metal components include C-14, Mn-54, Fe-55, Co-60, Ni-59, Ni-63, and Nb-94 [3]. Of these nuclides Fe-55 and Co-60 contribute the majority of the total initial activity, but the longer-lived nuclides with 10 CFR 61 concentration limits will cause the metals to be GTCC LLW.

2.3.1 Nuclide Decay

Although they are important activation products in stainless steel reactor components, iron-55, cobalt-60, and manganese-54 are not GTCC nuclides. The reason for this is their short half-lives, which can be seen along with the previously discussed GTCC activation products in Table 7 for comparison.

Table 7: Half-lives of Activation Products of Stainless Steel

Nuclide	Half-life (years)
Fe-55	2.73
Co-60	5.27
Mn-54	0.855
Ni-59	76,000
Ni-63	100.1
C-14	5730
Nb-94	20,300

To help understand the difference in these nuclides, DOE data for reactor component activities after reactor decommissioning can be studied. This data includes activities for each GTCC component in both PWRs and BWRs [3]. Data for a GE BWR-4 top fuel guide plate was used to generate Figure 4 and Figure 5, helps to illustrate the reason for more closely controlling GTCC nuclides.

Figure 4: Short Term Activation Products' Decay

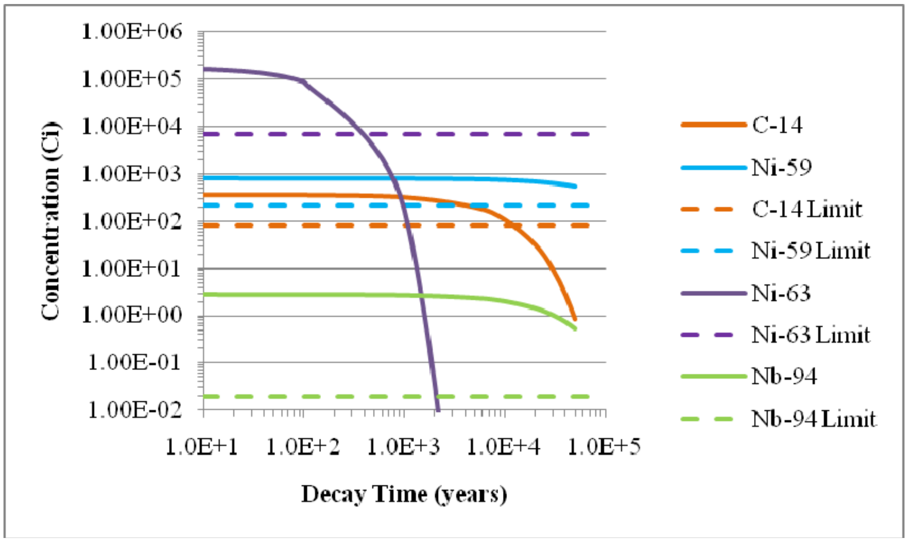

Figure 5: Long Term Activation Products' Decay with GTCC Limits

It can be seen from Figure 4 why Fe-55, Co-60, and Mn-54 are not considered GTCC nuclides. In comparison with the other activation products, these nuclides decay very quickly. Figure 5 shows GTCC activation products' decay, along with their GTCC limits. It can be seen that even after one hundred thousand years of decay, some of these nuclides will never drop below their limit to be classified as Class C waste. Thus, the components present too large of a health and safety risk to humans, and cannot be left to decay naturally. These reactor components must be treated if possible and safely disposed of if treatment is not feasible.

2.3.2 Formation of GTCC Nuclides

Perhaps the most important activation reaction in a nuclear reactor is thermal neutron capture (n-γ) in which an isotope of mass A captures a neutron to form a new isotope of mass

A+1. This reaction is often notated $^{A}X(n,\gamma)^{A+1}X$. Sometimes this product isotope will be a short-lived parent nuclide which will decay into a longer-lived daughter of a different atomic number. Generally, neutron capture rates are maximal in the thermal region [6].

Other interactions of interest which occur are $^{A}X(n,2n)^{A-1}X$ reactions and charged particle reactions. Most n-2n reactions involve target isotopes which are present in low natural abundances, making these reactions generally far less dominating than n-γ reactions. Charged particle reactions primarily include n-p, n-d, and n-α reactions, which typically occur when target elements have low atomic numbers.

Two isotopes of nickel are considered a GTCC waste concern: Ni-59 and Ni-63. Nickel is present as a large percentage of stainless steel, and the abundance of each naturally occurring isotope is important when considering the activation products which are created. For nickel, the natural abundance is 68.1% Ni-58, 26.2% Ni-60, 3.63% Ni-62, 1.14% Ni-61, and 0.926% Ni-64. Ni-63 is primarily produced via neutron capture from Ni-62, and the smaller natural abundance of Ni-62 helps to minimize Ni-63 production [6]. However, it can also be produced from Cu-63 through $^{63}Cu(n,p)^{63}Ni$ reactions. Cu-63 is present in trace amounts in stainless steel (Table 8) and is 69.1% of natural copper. Ni-59 can be produced through several mechanisms, with the most important being neutron capture from Ni-58. It can also be produced via n-2n reactions from Ni-60, but n-γ reactions are expected to dominate by about 3-4 times [6]. Of all the activation products, Ni-63 is one of the most abundant during dismantlement, and Ni-59 is one of the most important due to its extremely long half life.

Carbon-14 is produced primarily through the charged particle reaction $^{14}N(n,p)^{14}C$. Nitrogen is found in trace quantities in steel. Nitrogen-14 is 99.6% abundant, making it practically the sole naturally occurring isotope of nitrogen. Quantities of nitrogen in steel can be highly variable.

Niobium-94 is produced via thermal neutron capture from Nb-93, which is 100% naturally abundant. Although niobium is present only in trace amounts in stainless steel, as seen in Table 8, the 100% abundance of Nb-93 results in a significant production of Nb-94. On a much smaller scale, Nb-94 can also be produced from Mo-94, which is 9.1% of natural molybdenum and is also present in trace amounts in stainless steel as seen in Table 8 [6]. Additionally, Nb-94 is very long-lived and is one of the principal contributors of personnel exposure during dismantlement of nuclear power plants, making even small amounts important to monitor.

Table 8: Trace Elements of Importance: Amounts in Type 304 Reactor Stainless Steels [6]

Element	Average Amount (ppm)	Standard Deviation (%)	Range (ppm)	Number of Measurements Used
N	452 ± 64	14	402 to 525	3
Nb	89 ± 90	101	<5 to 300	13
Cu	3080 ± 2270	74	300 to 8150	13
Mo	2600 ± 1500	58	80 to 5500	13

2.3.3 Type of Contamination

Determining the type and location of radioactive contamination in materials is important when making decisions for treatment possibilities. In the case of activated metals, the activation products are present in any part of the metal reached by the neutrons which cause the interactions. Typical reactors have lives of 30 years or longer, and by the time of their decommissioning, the reactor internals have been exposed to a high neutron flux long enough for the entire piece to contain the GTCC nuclides.

Average neutron fluxes were computed for a typical PWR core shroud and core barrel and a typical BWR core shroud at the core mid-plane by the DOE for use in estimating the components' activities. Neutron fluxes of about 10^{10} to 10^{11} n/cm^{2}s or greater were found to generate activities over the Class C limit throughout the component being exposed to these fluxes. The expected neutron fluxes for these components are shown in Table 9 [7].

Table 9: Average Neutron Flux at the Core Mid-Plane for Selected BWR and PWR GTCC Reactor Components [7]

Component	Neutron Energy Range	Average Neutron Flux (n/cm^{2}s)
PWR Core Shroud	Thermal	1.3E+13
	Epithermal	3.5E+13
	Fast (> 1MeV)	7.0E+12
PWR Core Barrel	Thermal	1.4E+12
	Epithermal	3.9E+12
	Fast (> 1MeV)	7.7E+11
BWR Core Shroud	Thermal	2.5E+13
	Epithermal	3.0E+12
	Fast (> 1MeV)	1.0E+12

Thermal neutrons have been determined to be the most important in creating activation products which will cause the component to become GTCC waste. This data shows that neutron fluxes at the mid-plane of the reactor internals are high. Therefore, nuclides over the Class C limit will be produced throughout the reactor components, meaning the radioactivity is not only surface on the surface. The technological feasibility of treatment possibilities will depend in part on the location of the contamination in the waste to be treated, which in this case is bulk activation.

3. Exploring Treatment Possibilities

Wastes are typically recycled if possible, especially in recent years, due to less land available for waste disposal, resulting in increased disposal costs. For GTCC waste, treatment will be preferred if the treatment methods can be shown to be technologically feasible and cost effective (less than disposal costs). In order to make these determinations, it is important to explore some of the options which may be available for treating the waste. In this study, four different options for treatment were explored: laser cutting, electrochemical processing (pyroprocessing), metal melting, and use of the quantum-catalytic extraction process. These treatment methods were chosen because of their previous or current successful use as LLW, HLW, or mixed waste treatment methods. Each treatment will first be discussed, with an analysis of the applicability to GTCC waste at the end of the section.

3.1 Laser Cutting

Laser cutting is most often used industrially to cut sheet metal to desired thicknesses or pipes to desired lengths. In fact, laser cutting is the most common industrial application of the laser [8]. Lasers have several advantages to other types of cutting techniques. Cuts performed with lasers have a very narrow kerf - the width of the cutting opening. This is advantageous due to the substantial savings of the material which is being cut [8]. Additionally, laser cutting is fast, nearly all materials can be cut with lasers, and tool wear is minimal. Another attribute of laser cutting that is of particular interest when working with radioactive materials is the absence of a cutting edge, making the process "non-contact". This means minimal equipment contamination will occur in the cutting process. Lasers can also be transmitted over long distances, meaning equipment could be placed far enough away from radioactive material so that it will not become radioactive [8].

Methods of laser cutting have developed over the years into seven different methods: vaporization, fusion, reactive fusion, thermal stress cracking, scribing, "cold cutting", and burning stabilized laser cutting (LASOX) [8]. For metals, vaporization seems to be the simplest (and perhaps most likely to be used) of these processes, most of which are not even useful for metal cutting. The vaporization cutting method has only two basic steps involved. First, the laser beam is directed onto the surface of the work-piece to heat it rapidly, which results in melting or vaporization. This molten metal/vapor is then blown away using an assist gas [9].

Some disadvantages in using lasers include the relatively high capital cost of equipment, low efficiencies, and possibly high reflectivity of laser beams on some metals [9].

3.2 Electrochemical Processing

Electrochemical processing, or pyroprocessing, was developed as a spent fuel reprocessing technology, and has been used at Idaho National Laboratory for treatment of EBR-II fuel since 1996. The name "pyroprocessing" came from the very high temperatures (typically around 500°C) of operation used throughout the process. Pyroprocessing uses a molten salt bath in which the spent fuel is dissolved and uranium and plutonium are separated from the other elements present in the spent fuel with an electrorefiner, shown in Figure 6 [10]. Molten salts are used for their fast reaction kinetics as well as their high radiation resistance [11].

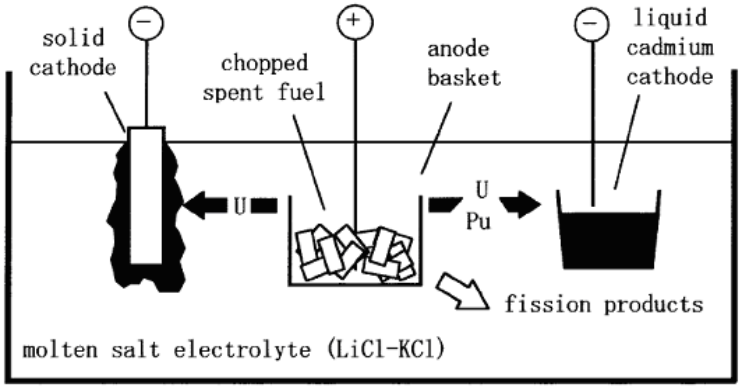

Figure 6: Pyroprocessing Electrorefiner Vessel [10]

Before entering the electrorefiner, the spent fuel is chopped into segments and placed in a cathode basket where it is reduced and converted from oxide form into metallic form, producing oxygen and carbon dioxide at the anode [11]. In the electrorefiner, the chopped spent fuel is placed in a steel basket which represents the anode. The impure spent fuel is transported through the molten salt, which acts as an electrolyte, to the cathode, where it is deposited in a more pure form. The molten salt used is typically a LiCl-KCl eutectic. A liquid cathode is used to collect the uranium-plutonium mixture due to its non-adherence on the solid cathode [11].

Pyroprocessing separation is based on the thermodynamic free energies (ΔG's) of chloride formation at the operating temperature of 500°C. There are three thermodynamic groups used in the electrochemical process for spent nuclear fuel, shown in Figure 7: metal waste, fuel fabrication, and ceramic waste [11].

The elements which make up the metal waste, shown at the top of the chart, have an unstable standard free energy of formation, meaning chlorides will not readily form, and they will remain in the anode basket. Those elements in the ceramic waste, shown at the bottom of Figure 7, have very stable chlorides which will form completely, accumulating in the molten salt. The fuel fabrication group, shown at the center, consists of the elements which will be deposited on either the solid or liquid cathode [11].

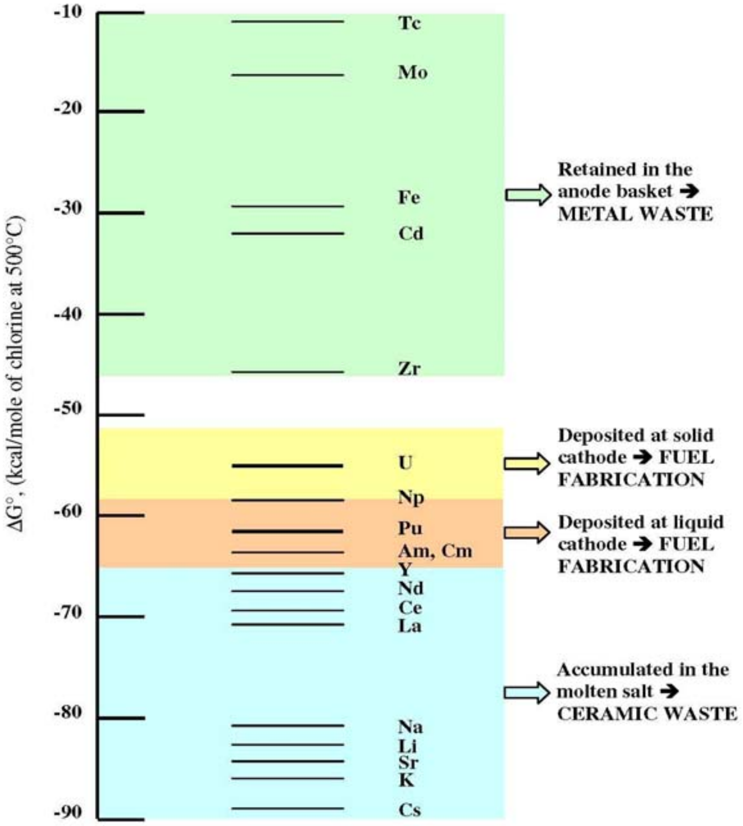

Figure 7: Free Energies Used for Spent Nuclear Fuel Pyroprocessing [11]

In order to use pyroprocessing for GTCC waste, data for free energy of chloride formation at 500°C for each element must be used to determine the thermodynamic group it will fall into on a chart such as Figure 7.

3.3 Melting Plant Processing

Metal melting plants have been used in several different countries for recycling or reusing low level radioactive scrap metal (usually steel). Examples of such refineries are

France's INFANTE plant, the United States' SEG plant, Sweden's Studsvik plant, and Germany's CARLA plant [12]. Additionally, the United Kingdom recycled aluminum and copper during the decommissioning of their Capenhurst diffusion plant, which was released and returned to the scrap metal market [12]. The United States also performed decontamination of metal process plant equipment as part of the National Conversion Pilot Project in Rocky Flats, CO, where slightly contaminated scrap was processed and converted into waste casks [12,13]. Metal melting plants still remain in operation around the world.

One such metal melting facility is Germany's CARLA plant. The CARLA plant currently accepts waste with a total specific activity of 1,000 Bq/g or less. Additionally, the nuclides, H-3, C-14, Fe-55, and Ni-63 must be present less than 10,000 Bq/g total [14]. Again using data from the 1994 DOE report, the example GE BWR-4 top fuel guide plate and core shroud will still have a total specific activity between 10^8 Bq/g and 10^{10} Bq/g, respectively - far above the limit for this particular plant. Major changes would need to be made to the CARLA plant (greatly enhanced shielding, for example) in order to accommodate GTCC waste activity levels, which may or may not be reasonable.

This melting process is also referred to as smelt purification. In the smelt purification process, metals are melted to help with both decontamination and volume reduction. Oxidizing materials can be added to the melted scrap metals in order to react with and remove impurities in the metals [15]. Elements such as uranium and other alpha-emitters will form oxides easily under these conditions, and thus will be removed from the metal with relative ease, while other metals such as technetium are not as easily removed [15]. A major benefit of applying metal melting techniques to LLW is the removal of cesium-137. Because cesium is very volatile, it is almost entirely removed from the metal into dust and ventilation filters [14]. Less reactive transition metals (such as cobalt, iron, and manganese) cannot be removed by the melting process, and will remain in the molten form [16].

3.4 Quantum Catalytic Extraction Processing (Q-CEP)

The Catalytic Extraction Process (CEP) was developed by Molten Metal Technology (MMT) in the mid-1990s. CEP was designed to convert hazardous wastes into marketable commercial products, and results from the process showed “success in minimizing both waste and emissions” [17]. The Quantum Catalytic Extraction Process (Q-CEP) was designed to apply the CEP process to mixed waste. A three-year project involving DOE LLW and mixed waste demonstrated recycling ferrous and non-ferrous metals, immobilization of radionuclides, destruction of hazardous organics, and recovery of volatile metals. Hopes at the time were to further expand the technology to decontamination of scrap metal [18].

The process chemistry of CEP is driven by thermodynamics, solution equilibria, and metal catalysts [17]. Before beginning the waste treatment process, the reactor, called a Catalytic Processing Unit (CPU), is loaded with iron (or sometimes nickel), which is then melted by induction heating. Lime and silica are then added to form an initial ceramic layer. After the ceramic layer is formed, feed materials (wastes) are also added to the CPU. One benefit of this process is that the feed can be gaseous, solid, liquid, or slurry in form [17]. A representation of the process can be seen in Figure 8 [17].

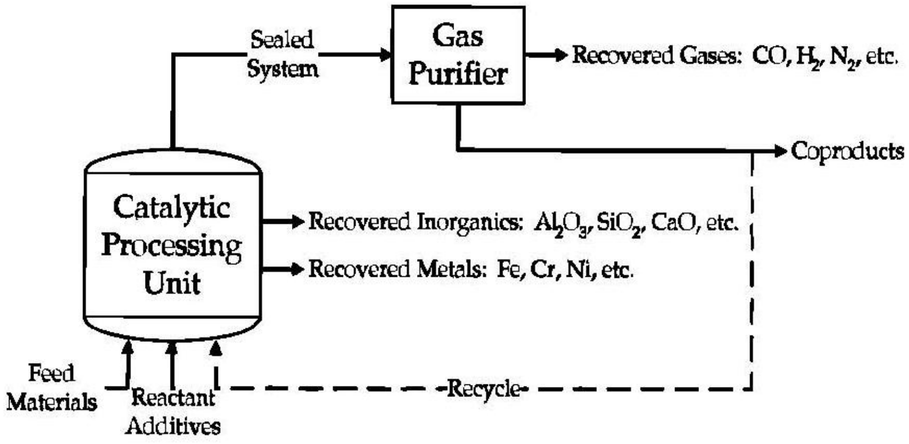

Figure 8: The Catalytic Extraction Process [17]

Once the feed material enters the bath, it dissociates into its elemental form in the molten metal and begins to form products. These products will separate into three phases: ceramic, metal, or a gas, which usually consisting of CO, H_2, and N_2. Selecting co-reactants and controlling operating conditions can dictate the final products that these elemental intermediates form, at least to some extent [18].

For typical mixed waste, oxygen would be added as the reactant additive, or co-reactant, which would then result in two possible reactions:

1. $2C + M + O_2(g) \rightarrow 2\,CO(g) + M$
2. $C + xM + yO_2(g) \rightarrow M_xO_{2y} + C$

The favored reaction will be determined by the thermodynamics of the molten metal solution, through the calculation of the Gibbs' free energy of the system. A typical operating temperature is around 1500°C, so for reaction 2 to occur, the Gibbs' free energy of the metal must be more negative than that of carbon at this temperature. From Figure 9, it is easy to see that carbon monoxide formation from carbon and oxygen can be an estimated -500 kJ/mol at 1500°C. This means elements with more positive free energies, such as iron, nickel, and copper will remain in the molten metal phase, while metals with a more negative free energy, such as uranium, plutonium, and aluminum, are more likely to form oxides and enter into the ceramic phase.

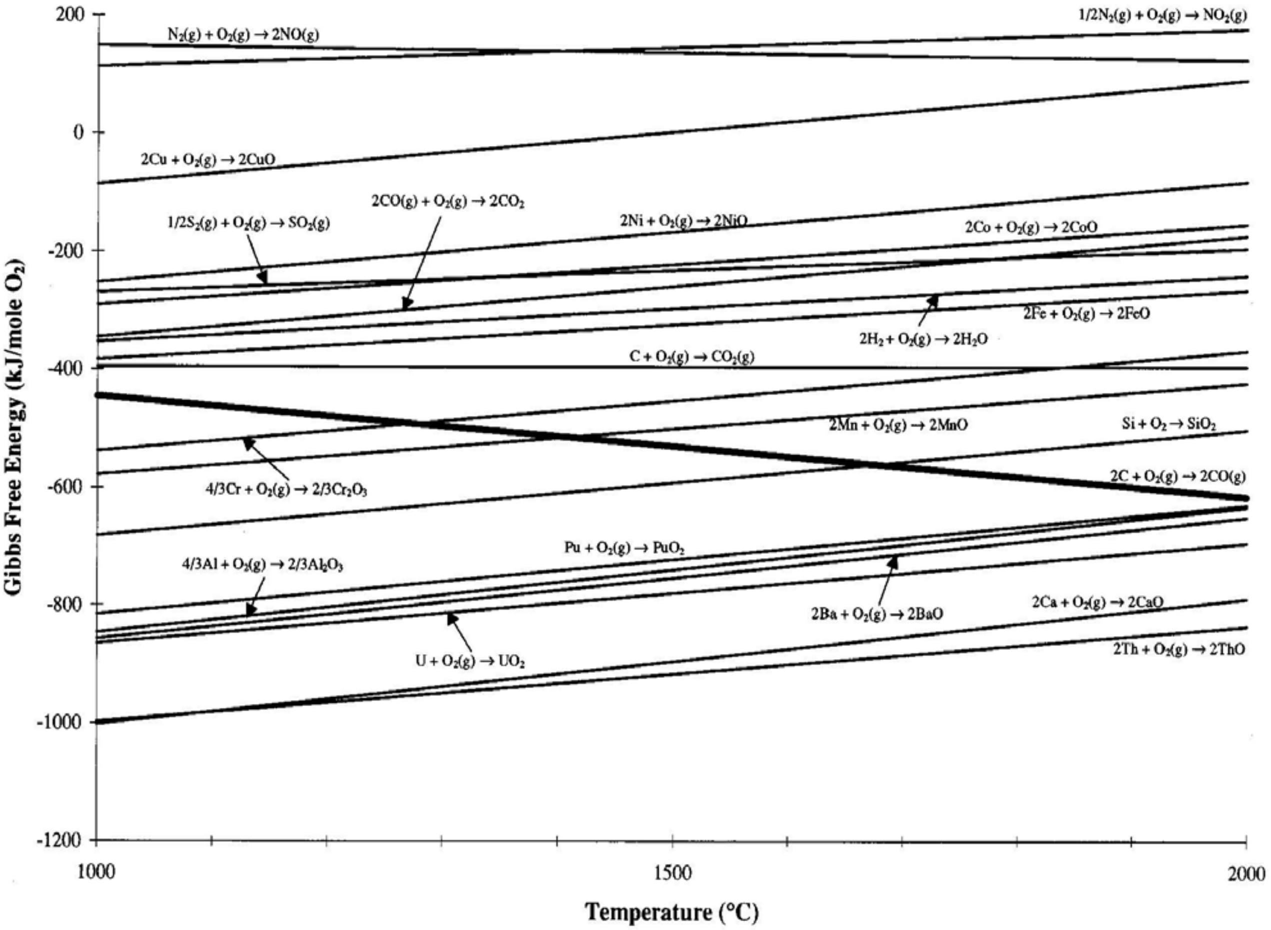

Figure 9: Gibbs' Free Energy for the Formation of Select Oxides [17]

An additional advantage of Q-CEP is the production of synthetic gas as a product, without any hazardous gases included. This led the EPA to designate this process a nonincineration technology. Incineration, or combustion, produces products in high oxidation states (e.g., H_2O and CO_2), whereas CEP produces products in lower oxidation states, making production of undesirable products like SO_2 and NO_x thermodynamically unfavorable [19].

These synthetic gases can be used for commercial applications, making it a desirable product, which could be sold to other industries, increasing profits.

3.5 Treatment Feasibility for Activated Metals

The possible treatments just discussed may be effective for some types of GTCC but not others. The scope of this thesis is primarily GTCC activated metals, so each treatment's applicability to activated metals will be the focus of this section.

3.5.1 Laser Cutting Effectiveness

Laser cutting is only useful for decreasing the volume of metals which are found to be contaminated predominantly by surface contamination. Laser cutting could then be used to remove the contaminated surface, leaving a "clean" or lower level radioactive waste portion of the metal and a higher level radioactive waste portion, which could be disposed of or possibly treated by other means.

In the case of GTCC activated metals, the nuclides of concern are distributed throughout the material rather than only on the surface (as discussed in Section 2.3.3). This makes laser cutting an ineffective method of treatment for activated metal GTCC waste. However, other GTCC metals, such as DOE GTCC-like scrap metals, could have primarily surface contamination, making laser cutting a useful method of treatment.

3.5.2 Electrochemical Processing Effectiveness

Electrochemical processes are only useful when the elements to be separated have significantly different free energies of chloride formation at the operating temperature.

Reactions with positive free energies will not occur spontaneously at that operating temperature.

At the typical 500°C operating temperature, niobium chloride exhibits a ΔG value of about -15 kcal/mol, similar to that of technetium or molybdenum, shown in Figure 7. If activated metals were treated by this method, niobium would be retained in the anode basket to become part of the metal waste. Nickel chloride has a ΔG value of about -28 kcal/mol (between that of molybdenum and iron, shown in Figure 7), which is still well within the region to be retained in the anode basket. Carbon also does not readily react with halogens, including chlorine. For example, carbon tetrachloride has a positive ΔG even at 500°C and higher temperatures.

Due to the thermodynamics of the reactions which would take place in molten salt, the nuclides of concern in GTCC activated metals would not be separated from other metals in stainless steel through electrochemical processing. As shown in Figure 7, iron (the main component of stainless steel) is within the metal waste group along with the other GTCC nuclides. Therefore, electrochemical processing is not a viable option for activated metals GTCC waste. However, it is possible that DOE TRU contaminated wastes could be treated by pyroprocessing, depending on the types of waste with TRU contamination. Additionally, sealed sources containing americium, curium, and plutonium may also be treated through this process by separation of the GTCC nuclides from their containment metals, which are typically not a GTCC concern.

3.5.3 Melting Plant Processing Effectiveness

The metal melting plant process is also based on the thermodynamics of the system, by using oxidizing materials to remove impurities which easily form oxides. As with chloride formation in the electrochemical process, nickel and niobium have lower ΔG's of oxide formation than the nuclides which are typically separated by this method, such as uranium, plutonium, and cesium. Carbon will remain a part of the stainless steel alloy, and will not react with oxygen in its elemental form.

Materials which contain contaminants with similar ΔG's, such as activated stainless steel, will not be successfully treated by this process. Therefore, melting plant processing or smelt refining should not be used as a treatment for GTCC activated metals.

As previously mentioned, a current use for the metal melting processes is in removing the very volatile cesium from metal components. This process has been shown to be very effective in cesium removal [14]. Using this method of treatment for sealed sources containing Cs-137 could prove to be very effective with further investigation. Since cesium-containing sealed sources are projected to be at least 60% of the packaged volume of sealed sources classified as GTCC, treating these sources could prove to be a valuable volume reduction option for GTCC waste.

3.5.4 Q-CEP Effectiveness

After breaking down the waste, the Q-CEP process forms oxides from the elemental intermediates for compounds with a Gibbs free energy of formation that is more negative than that of carbon monoxide formation. Using a typical operating temperature of 1500°C, it is apparent from Figure 9 that nickel will be present in the metal phase. Niobium can be expected to react similarly to other transition metals, such as chromium and magnesium, which will also be present in the metal phase according to Figure 9. Carbon, however, will react to form CO. If carbon were the only concern, this method could be a good option. However, with activated metals, nickel and niobium are additional concerns, as Ni-59 and Nb-94 will not decay below their Class C limits even in 100,000 years, as was shown in Figure 5. Additionally, CO is a constituent of the marketable synthetic gas product. Creating a new activated product in addition to leaving much of the GTCC activity in the waste makes the Q-CEP method very unattractive for treating GTCC activated metals.

4. Examination of Disposal Options

Even when treatments may be technologically feasible, disposal options could still prove to be more cost effective and/or safe. Because none of the treatment methods analyzed in the previous chapters of this study appeared promising, disposal is the only option for activated metal GTCC waste. Currently the DOE is considering three different disposal possibilities for GTCC waste. These possible facilities include enhanced near surface burial, intermediate depth boreholes, and deep geologic repository [20].

4.1 Enhanced Near Surface Burial

Enhanced near surface burial is based on low level waste disposal practices which are currently in use, where the waste is disposed of in shallow trenches or holes and covered with a backfill. For higher classes of waste, barriers such as concrete are used between the waste and the outside environment. GTCC waste will require even more monitoring and containment than any other class of LLW, which would most likely be increased thicknesses of the engineered barriers, greater depths of waste disposal, and longer observation periods after closure of the disposal area. Waste packaging may also be enhanced. A conceptual enhanced near surface facility is shown in Figure 10 [20].

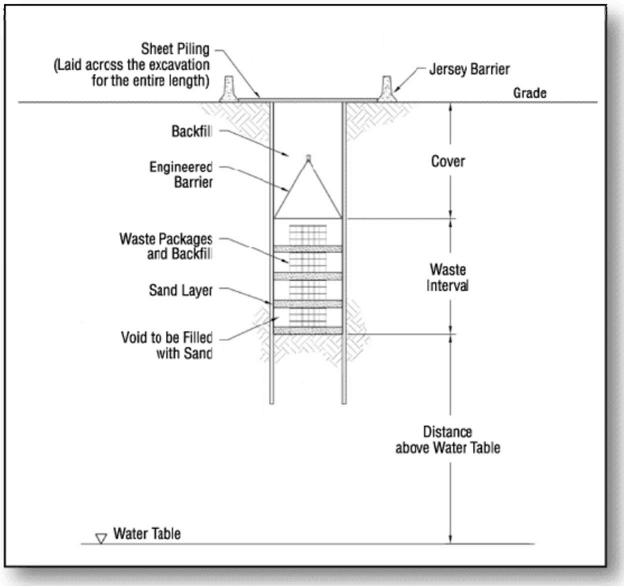

Figure 10: Conceptual Enhanced Near Surface Disposal [20]

4.2 Intermediate Depth Boreholes

Intermediate depth boreholes were used for GTCC waste disposal in the late 1980s [21]. Boreholes are created by digging deep holes into the ground, filling them with radioactive waste to a certain predetermined height, then using a backfill to completely fill the hole.

The boreholes used in the past for GTCC waste were about 3 meters in diameter and 36 meters deep, with the bottom 15 meters were used to hold the waste. Current intermediate depth boreholes restrictions are expected to allow the boreholes to hold waste no closer to the surface than 30 meters [20]. In order to take advantage of the previously used 15 meters of waste disposal space, the boreholes of current consideration would need to be 45 meters deep. A conceptual intermediate depth borehole facility under consideration is shown in Figure 11 [20].

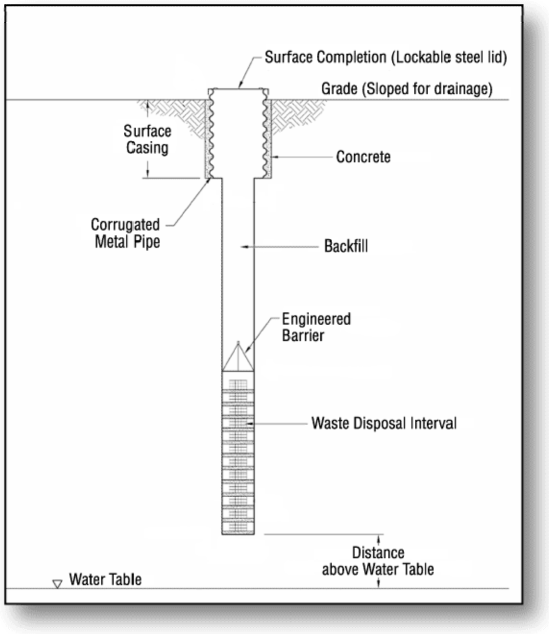

Figure 11: Conceptual Intermediate Depth Borehole Facility [20]

4.3 Deep Geologic Repository

According to the Nuclear Waste Policy Act, the default for GTCC waste is a deep geologic repository, which is in place unless another method can be proven to be effective [2]. Geologic repositories are considered a good method of isolating waste, because their depths help to delay groundwater access to the waste and slow the dissolution and sorption of radionuclides, along with other desirable properties of a waste storage and disposal facility [1]. Typical depths of a deep geologic repository are from 300 to 1,000 meters. In the United States, salt, basalt, tuff, and granite have been considered for repository sites in the past, and other countries have considered primarily salt, clay, and granite, with the most commonly studied host rock being granite [1]. Each rock type has its own set of advantages and disadvantages. Geologic repositories can also be a variety of designs, but the most common designs usually include excavated tunnels and sometimes boreholes (either vertical or horizontal) extending out from the tunnels where the waste is buried so the boreholes can be sealed individually [1].

5. Modeling Cost

The cost model developed for activated metal GTCC waste disposal uses previously calculated costs for HLW and LLW disposal to determine estimates for all three of the considered disposal methods. Previous studies have been completed for a variety of low level waste disposal options. One such study was completed by Idaho National Laboratory in 1994. This study calculated “Planning Life Cycle Cost” (PLCC) estimations for burial options, which included disposal methods similar to those being considered for GTCC waste. The PLCC for each of these burial options was broken down into 4 categories, Pre-Operations, Construction, Operation & Maintenance (O&M), and Decontamination and Decommissioning (D&D) [22].

Pre-operation costs include studies, bench scale tests, demonstrations, and "operations-budget-funded activities". "Operations-budget-funded activities" include conceptual designs, safety assurance, efforts for compliance with the National Environmental Policy Act, permitting, preparation for operations, and project management costs [22]. Pre-operation costs for the studied facilities were estimated through predictions of necessary research manpower and essential equipment.

Construction costs consist of two key components: major equipment costs and building costs. These costs were generally estimated by obtaining costs from similar existing facilities, suppliers, or by using engineering judgment [22].

Operation and Maintenance costs are estimated by dividing the total costs into four subcategories: operating labor, utilities, consumable material, and maintenance. Maintenance costs include parts, equipment, and labor, and were estimated as a percentage of the original equipment costs [22]. Costs for operating labor, utilities, and consumable materials were predicted through unit operations analyses [22].

Decontamination and decommissioning are costs associated with the facility's closure, and were determined based on the square footage of the facility [22]. Surveillance and Monitoring costs are included in D&D costs.

The total PLCC for the studied facilities can be calculated by summing the four categories.

5.1 Estimating Enhanced Near Surface Burial Cost

One burial option described in the cost estimations report by INL is engineered disposal. Engineered disposal is a near surface disposal which consists of disposal units with concrete cells where the waste canisters are stacked. Once the cells are full, they are backfilled with sand and covered with concrete to seal them [22]. Estimated costs for engineered disposal facilities of various capacities are given by this report. Associated costs given for a facility with a 50,000 m^3 capacity are shown in Table 10.

Table 10: Estimated Costs for a Near Surface Facility [22]

Pre-Operations	**Construction**	**O&M (1 year)**	**S&M**
(\$/m^3)	(\$/m^3)	(\$/m^3)	(\$/m^3)
360	3200	120	1700

Costs vary depending on the size of the facility, and the cost per cubic meter for an enhanced near surface facility increases exponentially for smaller volume capacities. Total GTCC stored and projected volumes are expected to be about 5,600 m^3. A capacity of 10,000 m^3 should provide extra storage to include projections past 2062 as well as allow for additional nuclear power plants which are not currently operating but could be built and decommissioned in the future. The cost per volume of various sizes of facilities is shown in Figure 12. Data from INL's report was extrapolated to extend to 5,000 m^3, as the data given only extends to 41,000 m^3 [22].

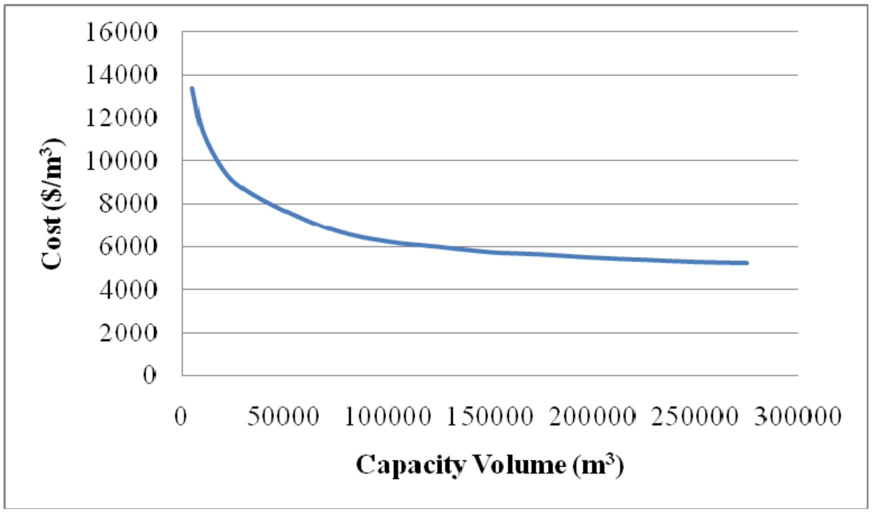

Figure 12: Total Cost of an Engineered Disposal Facility per Unit Volume

Additional cost increases for GTCC enhanced near surface burial will come from the necessary added containment and monitoring. Construction, O&M, and D&D (primarily S&M) costs are expected to increase by 20% to account for these changes. Taking these changes into account for 10,000 m^3 facility, cost values for each stage of the facility's life can be estimated, and are shown in Table 11.

Table 11: Estimated Costs for an Enhanced Near Surface Facility

Pre-Operations	Construction	O&M (20 years)	D&D	Total
($M)	($M)	($M)	($M)	($M)
7.6	81.0	60.8	43.0	192.4

The total cost for a 10,000 m^3 facility with 20 years of operation and maintenance would be about $190 million 1994 USD, or about $285 million 2010 USD.

5.2 Estimating Intermediate Depth Borehole Cost

INL has also estimated costs for shallow boreholes (25 meters deep), which were previously considered for LLW disposal. These estimations were determined for both “low sensitivity” and “high sensitivity” disposal sites. Low sensitivity areas were considered to be remote, existing DOE sites that are far above the water table, while the high sensitivity areas were described as non-DOE facilities that are near populated areas and close to the water table [22].

As a DOE facility, the low sensitivity site data are that the site is already characterized, has a permit for operations and is in compliance with the National Environmental Policy Act. Due to higher risks in operations, an extra 100 to 200 years of monitoring is included for high sensitivity areas, which is adds to D&D costs [22]. Data from INL for low and high sensitivity areas for 25 meter boreholes is shown in Table 12 [22].

Table 12: Shallow Borehole Cost per Excavation (1994 USD) [22]

	Low Sensitivity	**High Sensitivity**
	PLCC	**PLCC**
Pre-Operations	32	7,059
Construction	52	437
O&M (20 years)	568	8,200
D&D	1,757	14,770
Total	2,409	30,466

Current considerations for GTCC burial sites by DOE are only sites which are owned by the DOE, including the Hanford site, Oak Ridge Reservation, Savannah River Site, and others [20]. Because intermediate depth boreholes will almost certainly be at a DOE site, Pre-operations costs are estimated to be similar to low sensitivity site costs.

Construction costs for intermediate boreholes will increase due to the greater drilling depths. As with near surface burial costs per volume, borehole cost per meter of depth decreases with increasing depths. HLW boreholes of 4 km have a cost of $5.3 million per excavation (2006 USD) [22]. This corresponds to be a cost of about $1,460 (2010 USD) per meter excavated. Costs for shallow boreholes (low-sensitivity) are about $3,080 (2010 USD) per meter excavated. High sensitivity data for construction includes additional support and characterization costs which would not be necessary for GTCC boreholes at a DOE site. However, additional materials required for GTCC waste would include a 20% increase in costs of a shallow borehole, bringing the cost to about $3,690 per meter excavated. Interpolating between these values, the cost for 45 meter boreholes is $3,680 per meter for each excavation.

More stringent and extended lengths of facility monitoring will be necessary for GTCC waste disposal sites, which increases D&D and O&M costs, making these costs similar to those of a high sensitivity site.

Stored and projected GTCC waste volumes are expected to be around 5,600 m^3. Using 15 meters of depth for waste placement along with a 3 meter diameter, one intermediate depth borehole is expected to hold slightly more than 100 m^3 of waste. Fifty-three boreholes are needed for the currently stored and projected waste. For cost comparisons with enhanced near surface burial, 95 boreholes would be needed to store 10,000 m^3 of GTCC waste. Cost estimations for these two possible numbers of boreholes are shown in Table 13, in billions of 2010 USD.

Table 13: Estimations for Total Costs Intermediate Depth Boreholes

Number of Boreholes	**Billions of Dollars (2010)**
53	\$1.82
95	\$3.26

5.3 Estimating Deep Geologic Disposal Cost

Due to the deep disposal depths and the greatly enhanced barriers, geologic disposal is likely to have much greater costs than the other two options which are being considered. In 2007, INL estimated disposal costs for three different deep geologic repositories: Yucca Mountain, and recent repository models in Sweden and Switzerland. Cost predictor formulas were constructed for each repository to represent costs. These formulas were constructed using existing cost data for different repository capacities, and are shown below [21]. Costs are calculated in billions of USD.

Yucca Mountain: $\text{Cost} = 36.2 + 2.47\cdot10^{-4} \times \text{Capacity (in MTHM)}$

Sweden: $\text{Cost} = 2.706 + 2.36\cdot10^{-4} \times \text{Capacity (in MTHM)}$

Switzerland: $\text{Cost} = 1.96 + 3.35\cdot10^{-4} \times \text{Capacity (in MTHM)}$

The coefficients in these equations are similar for all the sites, but the Yucca Mountain predictor equation has a significantly larger constant. This higher number is thought to be due to the prolonged planning and preconstruction stage that Yucca Mountain has undergone [21]. A capacity of 10,000 m^3, which translates to about 36,000 MTHM, was used to determine the associated total costs for each repository. These values can be seen in Table 14. Although Yucca Mountain costs are inflated, this could be an accurate estimation of additional repositories which might be constructed in the United States and should be considered when considering building any new U.S. repositories.

Table 14: Estimated Costs for Deep Geologic Repository (Billions of USD)

	2010
Yucca Mountain	$48.30
Sweden	$12.04
Switzerland	$15.07

6. Characterization of Risk from GTCC Waste Disposal

In studying potential repository sites, an important consideration is the impact the site will have on the surrounding public. To determine the impact of these potential sites, the risks and possibilities of adverse effects are determined. Risks are typically measured by using data gathered from environmental monitoring or modeling which can be translated into exposure or dose levels to the public. From these exposure calculations, the expected adverse effects can be estimated, typically in terms of increased chance of cancer or lost years of life.

Useful environmental modeling data for radioactive waste repository sites will generally consist of expected concentration levels of the radionuclides of concern in the environment. Radionuclide concentration data for GTCC reactor components immediately after shutdown was generated in the 1994 by the DOE, from compositions of reactor components and the operating neutron flux, and is the data used in this study. From these initial concentrations, predicted radionuclide concentrations at a site nearby the repository in contact with the public can be determined.

6.1 Exposure Pathways to the Public

Regulatory limits on radionuclide intake have been established in order to protect individuals in the general public from receiving a radiation dose which could result in a greater possibility of cancer or death. Radiation doses can be obtained through a variety of different means. Humans are exposed to radionuclides primarily from their transportation through water. Radionuclides will be leached from their containment in the disposal facility by water, and will then reach humans through a variety of means, shown in Figure 13 [23].

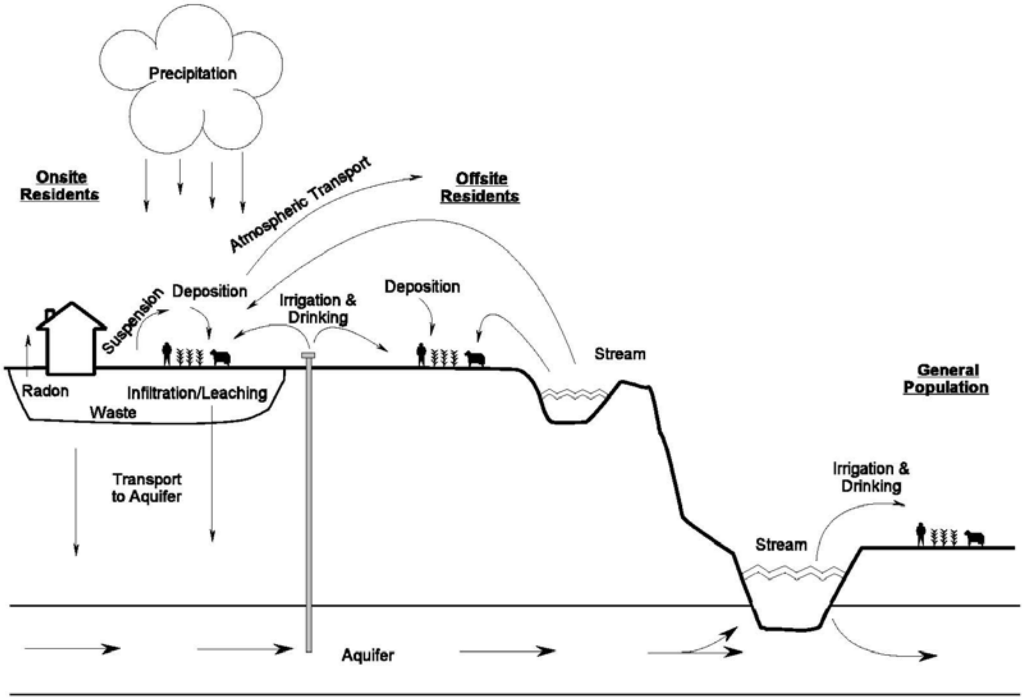

Figure 13: Pathways for Radionuclide Transport by Water [23]

6.1.1 Water Cycle for a Waste Disposal Site

The main source of the water entering the disposal site is precipitation. Precipitation falls on the top layer of soil on the site, and then either infiltrates the soil or leaves the site as runoff or evaporation. The water which infiltrates the soil then reaches the contamination zone and leaches out radionuclides. The majority of the water then percolates through the soil under the site into an aquifer below. During this percolation, some of the radionuclide contamination is lost through sorption into the soil. The remaining radionuclide concentration is transported with the water through the aquifer to a well. Once the contaminated water reaches the well, it will be used by humans for drinking water, irrigation, and/or animal feed [23]. For this study, it is assumed that the majority of human exposure will result from drinking the contaminated groundwater. Exposure from eating plants irrigated by the water or animals fed contaminated water and plants will be smaller due to the indirect path to human consumption. Drinking the groundwater is a direct pathway to the body and is the exposure pathway of the most concern.

Considering the water's pathway, major factors for determining the exposure to the general public are the amount of precipitation to the site, the amount of that precipitation which infiltrates through the topsoil into the waste (which will determine the amount of waste which will leach into the water), and the transportation of the groundwater through the soil and the aquifer to the well. Precipitation amounts will be determined based on the climate of the disposal site's region, and the infiltration and groundwater transport will depend on soil properties of the site.

6.1.2 Soil Retention of Radionuclides

The primary soil property of interest will be the soil's retention of the radionuclides, which will determine the amount of the radionuclides which will remain in the groundwater. A common way of measuring the soil retention is by calculation of the soil distribution coefficient, K_d. The soil distribution coefficient is defined as the concentration of radionuclide sorbed on the soil divided by the concentration of the radionuclide which remains in the fluid phase. This model represents soil retention mechanisms as a linear relationship between the soil and the groundwater, and assumes the radionuclide concentrations in the groundwater and soil are at equilibrium [24]. This means that larger K_d values indicate a slower migration of the radionuclide relative to groundwater flow. Because the soil distribution coefficient is a chemical property, each element behaves the same regardless of the isotope.

The soil sorption coefficients for elements are typically calculated in a chemical laboratory through batch adsorption techniques [25]. This experiment is performed by combining a solution of a known radionuclide concentration with a known mass of the soil being studied, as seen in Figure 14. After mixing and equilibrating, the concentration of the radionuclide in the resulting solution is measured, and the difference in this concentration from the original solution concentration due to sorption onto the soil.

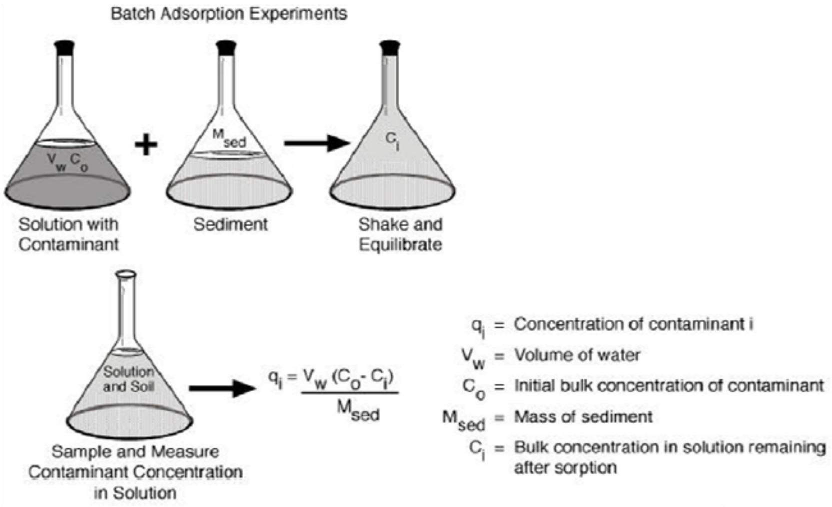

Figure 14: Calculating the Soil Sorption Coefficients [25]

Typically, carbon-14 is a concern in the environment because it is especially mobile in groundwater [24]. Carbon as a constituent of stainless steel is in elemental form, and will not be available for microbial activity, which would typically result in the formation of CO_2 and CH_4 from organics. Formation of $^{14}CO_2$ would be of great concern due to potential inhalation as well as significant uptake by plants [24]. In groundwater systems, carbon is generally found in carbonate or bicarbonate forms. The negatively charged carbonates and bicarbonates do not generally absorb into soils due to the net negative charge found on most soil surfaces [24]. Thus, the soil distribution coefficient (K_d) is often conservatively assumed to be zero or near zero. Additionally, these carbon compounds have been found to dissociate completely in the groundwater, meaning that solubility will not be a limiting factor in concentration calculations.

Nickel tends to be easily absorbed by soils, but strongly depends on pH and soil type, which varies significantly [24]. Generally, nickel is found in the Ni^{2+} state, but can also be found in +1, +3, or +4 oxidation states. Nickel is found naturally in soils and groundwater, in much larger concentrations than the trace amounts expected to be leached from waste disposal sites. Nickel K_d values have been measured in a wide range of values from less than 100 to 5,000 mL/g [24]. Typically, when nickel in groundwater comes into contact with soil from groundwater, it will be deposited in the soil and not migrate a great extent from the original site. Nickel 2+ in groundwater has a solubility of 5.87E+06 Ci/m^3, which will be a non-limiting property for calculations of radionuclide concentrations in the groundwater.

Like nickel, niobium is not easily leached into the groundwater, and will be mostly absorbed by soils, meaning it will also not noticeably travel far from the original site. Niobium is typically found in compounds in an oxidation state of +5, such as in Nb_2O_5, but it also often forms compounds in the +4 or +2 oxidation states. Measured values K_d for niobium are usually high, and similar to those for nickel [24]. Niobium compounds will completely dissociate in the groundwater, so that solubility will not be a limiting property in radionuclide concentrations in groundwater.

Although the general behavior of radionuclides in the environment is important to keep in mind when performing calculations, climate and soil properties will depend heavily on the site's location. For this reason, reference disposal facilities are used in this study in order to accurately and effectively represent a range of possible disposal site locations.

6.2 Reference Facility Selection

Possible locations for a waste disposal facility should not be limited to one area of the country. In order to represent the entire United States as accurately as possible, disposal facility locations were chosen which will best describe general regions of the country. Knowing the location of the disposal facility is necessary to determine how the radionuclides of concern will be transported from the facility to a well downstream. The two factors of consideration for disposal facility location are the climate of region and the soil type (to determine soil transport properties). These two factors are not independent; soil type depends heavily on climate. Reference locations will help to determine general results which can describe the expected performance of the disposal site in each location. Once an actual disposal site is known, calculations such as the ones performed in this study should be performed for the specific site. All reference facility calculations are for determining general site performance, and cannot be applied to predict the performance of specific sites.

6.2.1 Climate of the Area

Although climate can be divided into many categories, there are two basic climate divisions: humid and arid. Humid climates are characterized by hot, humid summers with cooler winters. Precipitation is significant throughout each season. In the United States, this climate is found in the northeastern and southeastern states as well as the Deep South. An arid climate is characterized by small amounts of precipitation and fast evaporation, meaning the water maintained in the ground is very low. These climates are often desert or desert-like. In the United States, these areas are throughout the western part of the country, especially the southwestern states. On average, humid sites are expected to have precipitation amounts of 118 cm/year, while an arid site will typically have about 25 cm of precipitation per year.

6.2.2 Soil Textures

Soil textures can be classified into a wide variety of very specific categories, but there are three basic types of soil textures: clay, silt, and sand. The differences in the properties of clay, silt, and sand are determined by the particle size. The soil texture is determined based on the percentages of each material which is in the soil. Clay is defined as material with 40% or more clay, less than 45% sand, and less than 40% silt. Sand contains 85% or more sand, with the remaining material no more than 10% clay. Silt contains 80% or more silt and less than 12% clay. Loam, another common texture classification, contains 7-27% clay, 28-50% silt, and less than 52% sand. Similarly, soils which are a combination of these types are classified according (e.g., loamy sand or silty clay loam). The U.S. Department of Agriculture (USDA) classifies soil by using their soil texture triangle, shown in Figure 15 [26]. This triangle shows the relationship of each texture with the others, and makes it easy to classify soils with known percentages of each soil type.

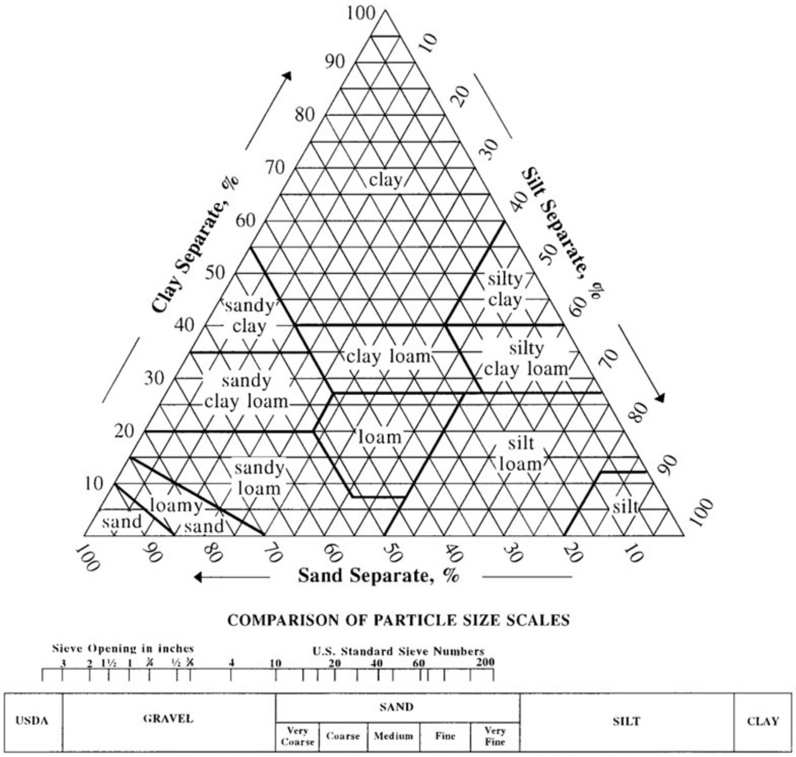

Figure 15: USDA Soil Texture Triangle [26]

Soil textures help in determining other soil properties, such as porosity and permeability, which in turn will help to determine the sorption of radionuclides onto the soil. Soil sorption is an important property when considering groundwater transport, because as more of a radionuclide is absorbed into the soil, less of that radionuclide is available for transportation with the groundwater to the well, causing eventual public exposure.

6.2.3 Soil Sorption Properties

Soil sorption characteristics have been determined and characterized by using these basic texture types. A compendium by Sheppard and Thibault gives K_d values for sand, loam, clay, and organic soil for many elements which are a concern in various nuclear wastes, including carbon, nickel, and niobium. Data for elements which are relevant to GTCC waste is shown in Table 15 [27].

Table 15: Average K_d values for GTCC elements of concern (mL/g) [27]

Element	**Sand**	**Loam**	**Clay**	**Organic**
C	***5***	20	1	70
Ni	***400***	300	***650***	***1,100***
Nb	160	550	900	2,000

Note: Values in bold italics are literature values

Data shown in Table 15 was collected from literature and averaged to provide a single value. For literature values which had time dependence, the authors used the longest time in order to best represent equilibrium conditions [27]. Values for studied soil textures which had no data found in the literature were determined from a soil-to-plant concentration ratio method to indicate mobility, which was determined by the authors to be an effective technique [27].

Sheppard and Thibault also included ranges of K_d values for the studied elements. These ranges will be useful when determining conservative concentration estimates as well as performing sensitivity analyses, so this compiled data for GTCC elements of concern is shown in Table 16 [27].

Table 16: Ranges of K_d values (mL/g) [27]

Element	No. of Observations*	Average	Data Range
	Sand Soil		
C	3	5	1.7 to 7.1
Nb	---	160	---
Ni	11	400	60 to 3,600
	Loam Soil		
C	---	20	---
Nb	---	550	---
Ni	---	300	---
	Clay Soil		
C	---	1	---
Nb	---	900	---
Ni	10	650	305 to 2,467

*Note: Values with no observations are those with average values calculated by the authors using the concentration ratio technique [27]

6.2.4 Reference Facilities Chosen for This Study

In the study completed by the EPA for LLW disposal sites, reference facilities which were used were arid permeable, humid impermeable, and humid permeable soils [28]. Arid soils have very low organic content, and are typically sandy. Clay is the most impermeable soil due to its small particle size, and is found commonly in the humid areas of the United States. Humid permeable soils could be sand, silt or loam, so for this study, a humid permeable soil will be considered to be loam. Modeled after the EPA's study, the three reference facilities used in this study will be: arid permeable (sand), humid impermeable (clay), and humid permeable (loam). Thus the arid permeable site will be located in the southwestern part of the United States, the humid impermeable site will be in the northeast, and the humid impermeable site will be in the southeast.

Due to the high variability in parameters used for hydrogeologic modeling, when well concentration for a specific site are determined, variables which correspond to actual measured values of that site should be used. These reference facilities were chosen in order to represent a range of possible data and do not represent specific sites. General locations were chosen in order to represent a range of possible locations for GTCC waste storage sites in the United States. Data for these reference facilities was based primarily on data from the EPA's study of LLW disposal sites. Specific sites were used by the EPA to help characterize the reference facilities. These sites included Beatty, NV (arid permeable), West Valley, NY (humid impermeable), and Barnwell, SC (humid permeable) [28].

6.3 Reference Waste Package

When radioactive wastes are disposed of at many facilities, they are first placed into a waste package. For this study, any waste package used for GTCC activated metals will be assumed to fail as soon as it is placed into the facility. This is a conservative estimate because the waste package must fail in order for the waste to leach out of it. This assumption could also be viewed as assuming there is no waste package and the waste is placed into the disposal site as is. This failure will release the waste into the environment at a rate which is dependent on waste package parameters, primarily the surface area of the waste.

In order to determine waste package release rates, a waste package size must be assumed. The size of the waste package is necessary in determining the surface area which will be exposed to the soil and water which will corrode the material. For this study, the size of the waste package used is based on the waste package used for a Korean pyroprocessing metal waste stream. This metal waste is classified as Intermediate Level Waste – an international classification with properties very similar to GTCC. The waste package used for this waste has a volume of 4.275 m^3, and is 1.5 m (L) × 1.5 m (W) × 1.9 m (H) [29]. Using this size of waste package, 1,314 packages are needed to store all currently stored and projected GTCC waste (5,615 m^3). This yields a surface area of 15.9 m^2 for one waste package, and a total surface area of 20,893 m^2 for all the waste packages. This surface area represents the area which is exposed and available for corrosion to occur. Release rates will be determined by assuming that the entire surface area of each package is exposed to the surrounding environment. Additionally, these release rates will be dependent on the amount of water entering the site, which is dependent on the specifics of the waste disposal site. Therefore, before calculating the release rates of the waste package, the movement of the groundwater in and around the site must be considered.

6.4 Groundwater Transport Model

As water enters the disposal facility by infiltration, it causes radioactivity to be released through corrosion of the waste. These radioactive contaminants then leach into the groundwater, moving with it. As groundwater moves through the soil, it disperses in all directions. Thus, the general three-dimensional mass balance equation given can be applied to calculations for transport in an aquifer containing radionuclides and is shown in Equation (1).

$$R\frac{\partial C}{\partial t} - \nabla \cdot (\underline{D} \cdot \nabla C) + V\nabla C + \lambda_d RC = 0 \qquad (1)$$

Subject to the boundary conditions:

$C = 0$ for all x, y, and z when $t=0$

$C = C_0(t)$ at $x = 0$ for $t>0$

$C =$ finite at $x = \infty$ for $t>0$

These variables are defined as: R is the retardation factor, C is the concentration of radionuclide in the groundwater, $\underline{D}$ is the dispersivity tensor, V is the interstitial velocity, and λ_d is the radionuclide decay constant [30]. The interstitial velocity is defined as $V = \underline{v}/\varepsilon$, where $\underline{v}$ is the velocity vector and ε is the porosity of the soil. The retardation factor is given by $R = 1 + \frac{\rho_b K_d}{\varepsilon}$, where ρ_b is the bulk density of the soil, and K_d is the soil distribution coefficient. Radionuclides are assumed to be at equilibrium in the groundwater and the soil, meaning that the linear K_d model for soil absorption is valid.

The EPA's PRESTO code for modeling LLW movement in groundwater applies this equation in one-dimensional form, which is valid when assuming that the flow of groundwater is steady and uniform and no heat is being lost or absorbed [31]. This one-dimensional model is safe to assume because it represents a conservative approximation as opposed to a three-dimensional model. The 1-D model concentrates the radionuclides in the groundwater into a uniform, narrow stream, as opposed to the dispersed plume of the 3-D model, making a higher concentration of radionuclide reach the well in a shorter amount of time. This will make dose estimations higher than those which will actually occur, making the method conservative. Thus, the three-dimensional Equation (1) simplifies to the one-dimensional Equation (2).

$$D\left(\frac{\partial^2 C}{\partial x^2}\right)-\frac{\partial C}{\partial t}-\frac{V}{R}\frac{\partial C}{\partial x}-\lambda_d C=0 \qquad (2)$$

Subject to the boundary conditions:

$C(x,0) = 0$

$C(0,t) = C_0(t)$ for $t>0$

$C(\infty,t) =$ finite for $t>0$

This one-dimensional equation has an undefined boundary condition at x=0, so a solution is not analytically obtainable. To simplify the model, longitudinal dispersion effects can be neglected, leading to a solvable equation, shown as Equation (3). Additionally, concentration can be converted to the rate of radionuclide transport, Q (curies per year), by multiplying each term by the groundwater flow. Q_0 is defined as the rate of radionuclide transport at the bottom of the contaminant source (the disposal site), which is typically an easier variable to determine than the actual concentration at this point.

$$\frac{\partial Q}{\partial t}+\frac{V}{R}\frac{\partial Q}{\partial x}+\lambda_d Q=0 \qquad (3)$$

Subject to the boundary conditions:

$$Q(x,0) = 0$$

$$Q(0,t) = Q_0(t) \text{ for } t>0$$

$$Q(\infty,t) = \text{finite for } t>0$$

Equation (3) can then be solved by using the convolution method. This method is an integral transform which expresses each function in the partial differential equation in terms of a new variable. The resulting function can be thought of as an impulse response function. The convolution for Equation (3) is shown as Equation (4), where τ is the dummy variable and u(τ) is the response function, given by Hung and shown in Equation (5) [31]. The release of the radionuclide is at x = 0 and τ = 0.

$$Q(t) = \int_0^t Q_0(t-\tau)u(\tau)d\tau \qquad (4)$$

$$u(\tau) = \exp\left(\frac{-R\lambda_d}{V}\right)\delta\left(\tau - \frac{RL}{V}\right) \qquad (5)$$

The integration can then be performed, yielding a solution to the one-dimensional model, shown in Equation (6), where $\tau = {}^{LR}\!/_{V}$, L is the length of the transport path, and V is the groundwater flow velocity. This equation is the final form applied in LLW modeling used in PRESTO.

$$Q(L,t) = Q_0(t-\tau)\exp(-\lambda_d \tau) \qquad (6)$$

In order to calculate the concentration of the radionuclides at the well, the rate of radionuclide transport to the well must be determined. The Hung model, shown in Equation (6), can be applied throughout the contaminant's pathway. After the contaminant is leached from the waste into the water, it will continue to travel with the groundwater along its pathway. For this study, two areas were identified as important in modeling the pathway of groundwater transportation: the unsaturated, or vadose zone, and the aquifer. The groundwater will travel vertically through the vadose zone, then horizontally through the aquifer. A representation of this model can be seen in Figure 16, which is a slightly simplified version of the model used in the EPA PRESTO groundwater modeling code. The PRESTO model includes conceptual "collection reach" which takes into account mixing and dispersion effects in the groundwater's transition from the vadose zone to the aquifer [30].

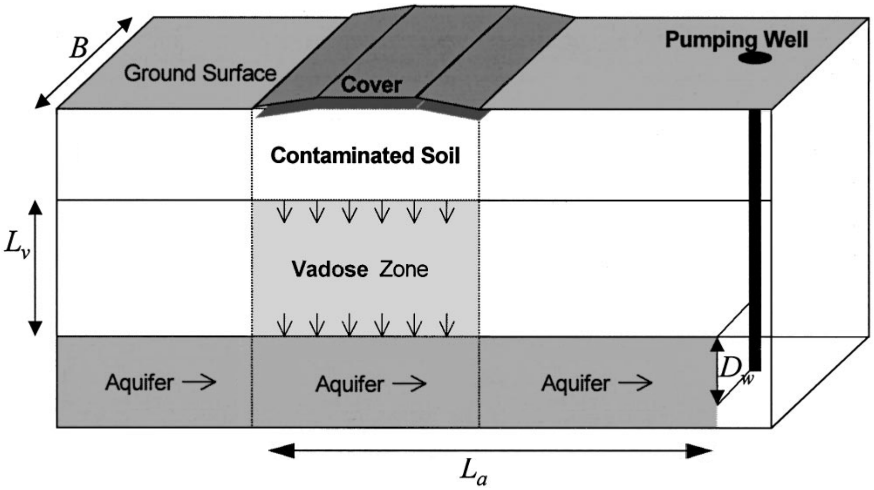

Figure 16: Measuring Groundwater Flow Pathways (modified from Hung) [30]

The total distance traveled by the groundwater is then $L_v + L_a$. Because Equation (6) is valid for each of these regions, the rate of radionuclide transfer for the total distance traveled can be modeled by Equation (7), where subscripts refer to the vadose zone (v) and the aquifer (a).

$$Q(t) = Q_0(t-\tau)\exp\left(\frac{-\lambda_d L_v R_v}{V_v}\right)\exp\left(\frac{-\lambda_d L_a R_a}{V_a}\right) \qquad (7)$$

The concentration of the radionuclide in the well water, C_{well} (Ci/m^3), can be determined from the rate of radionuclide transfer at this point by dividing by the rate of groundwater flow at the well (in m^3/yr). The rate of groundwater flow can be found by considering the lateral dispersion at the well along with the well screen location. Generally, the well screen is installed near the bottom of the aquifer, according to drilling practices and various state well water regulations in the United States [23]. Additionally, the well is assumed to withdraw water uniformly from the water surface to the well screen. The rate of groundwater flow at the well, as determined by the EPA for PRESTO, is shown by Equation (8) [23].

$$W_a = V_a \varepsilon_a D_w (B + 2\tan(\alpha / 2) D_h) \qquad (8)$$

Where V_a is the velocity of the groundwater in the aquifer, ε_a is the porosity of the aquifer, D_w is the depth of the well penetrating the aquifer, B is the site width, D_h is the distance from the center of the site to the well, and α is the dispersion angle.

Finally, the overall equation which is used for the calculation of radionuclide concentration at the well for this study can be shown by Equation (9).

$$C_{well}(t) = \frac{Q(t)}{W_a} = \frac{Q_0(t-\tau)\exp\left(\frac{-\lambda_d L_v R_v}{V_v}\right)\exp\left(\frac{-\lambda_d L_a R_a}{V_a}\right)}{V_a \varepsilon_a D_w (B + 2\tan(\alpha / 2) D_h)} \qquad (9)$$

6.5 Converting Well Concentrations to Expected Public Dose

After calculating the concentration of the radionuclides at the well, the expected dose to the public from these concentration levels should be determined. The EPA has established limits for radionuclides in drinking water in 40 CFR 141.66. The GTCC radionuclides of concern are primarily beta emitters, and concentrations of these radionuclides in drinking water must not cause a public dose of more than 4 mrem/year [32]. Concentrations below these limits were determined by the EPA to result in no increased cancer risk, while concentrations causing public doses above this limit may lead to an increased cancer risk.

In order to compare the radionuclide concentrations reaching the well from the disposal site to the 4 mrem/year limit, dose conversion coefficients must be used. Values for these coefficients are determined by the International Commission on Radiological Protection in ICRP 72 for radionuclides which present a concern when either ingested or inhaled. For GTCC nuclides in activated metals, the concern is from the ingestion of these radionuclides. Values for the dose coefficients in ICRP 72 are shown in Table 17 in the given units (Sv/Bq), as well as in into rem/Ci, which will be more useful units for this study [33].

Table 17: Ingestion Dose Conversion Coefficients [33]

	Sv/Bq	**rem/Ci**
C-14	5.8E-10	2.1E+03
Ni-59	6.3E-11	2.3E+02
Ni-63	1.5E-10	5.6E+02
Nb-94	1.7E-09	6.3E+03

Water drawn from the well can be conservatively assumed to be ingested immediately. The EPA directs in 40 CFR 141 that all calculations of dose acquired from drinking water to assume a 2 liter per day consumption of water per person. This corresponds to 730.5 liters consumed per person per year. The dose received to the public can be found from the well concentrations by using Equation (10), where D is the dose received, C is the concentration of the radionuclide, and IDCC is the ingestion dose conversion coefficient.

$$D(mrem/yr) = C(Ci/m^3) \times IDCC(mrem/Ci) \times 0.7305(m^3) \qquad (10)$$

The calculated dose can then be compared to the 4 mrem/year EPA limit. If the dose is above this limit, then the concentrations of the radionuclide are too high for the disposal facility to be considered safe for operations, and other facilities should be considered.

6.6 Preliminary Well Concentration Calculations

In order to gaingeneral idea of the activated metal's performance in the reference waste disposal facilities, preliminary calculations were performed by modeling individual activated metal components. Data for the total activity of these components and the activity for each radionuclide immediately after shutdown were determined by the DOE's GTCC Inventory report. Four BWR components: the top fuel guide plate, core shroud, fuel support castings, and core support plate were studied, along with four PWR components: the upper core support plate, core baffle, lower core support barrel, and lower core support plate. Six different reactor models were used in the DOE's study: two GE BWRs (models 4 and 6), two

Westinghouse PWRs (an intermediate and large size reactor), a Combustion Engineering PWR, and a Babcock and Wilcox PWR.

In order to take into account the expected decay storage time, this data is decayed for six years and is shown in Appendix A. These items were considered to be placed into the disposal facility as-is in order to determine the radionuclide concentrations leached from the each component into the groundwater.

No barriers were assumed to be present for the facilities in these calculations, resulting in a very high infiltration rate. The near surface facility is modeled with only a topsoil cover above the waste. At each site the waste disposal location will be above the vadose zone, and the radionuclides will travel through the vadose zone to the aquifer, as was shown in Figure 16. The vertical length of the vadose region varies depending on the disposal site for the near surface facility.

Intermediate and deep facilities are also considered for each reference site. The EPA's model for deeper disposal facilities assumes the waste is placed in an aquiclude, confined by an upper and lower aquifer, as shown in Figure 17 [28]. This model considers the vertical movement of groundwater through the waste facility to the upper aquifer.

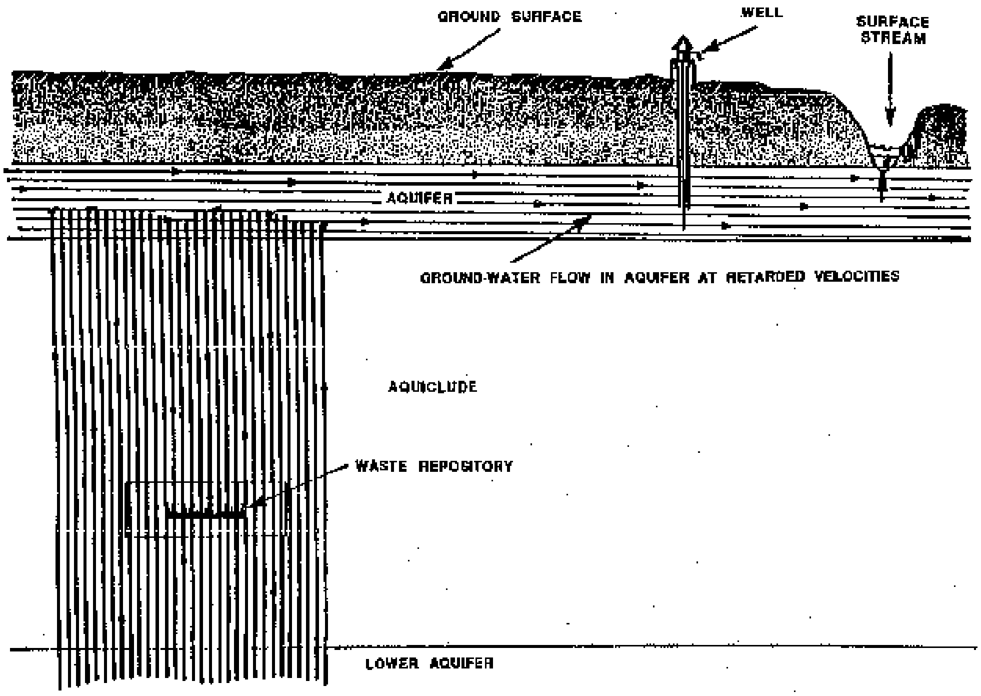

Figure 17: Groundwater Model for Intermediate Depth and Deep Facilities [28]

The EPA's intermediate depth facility is located 50 meters from the surface, and the distance from the waste repository to the lower aquifer is estimated to be almost 430 meters. A deep waste repository is 300 meters from the surface, and about 180 meters to the lower aquifer [28]. Due to the relatively shallow depth of the upper aquifer in both the humid models, both intermediate depth and deep facilities will be located in the aquiclude region.

Determining the infiltration rate for each facility at near surface, intermediate, and deep depths is important in determining the rate of radionuclide transfer at the edge of the site (Q_0). The infiltration rates depend on the type of soil at the site, due to differences in permeability and porosity. The amount of water that reaches the waste disposal area determines the amount of the radionuclides that will be leached from the waste into the groundwater.

Waste release rates depend not only on the the infiltration amount, but also the waste form and disposal method. The EPA (in their LLW Draft Environmental Impact Statement) utilizes an approach for determining waste release rates which calculates the percentage of waste released each year from each site. Three factors are considered in the EPA’s study: a Site Factor, which takes into account the site climate and infiltration, a Waste Factor, which takes into account the waste corrosion and release rates, and a Disposal Factor, which takes into account the disposal depth of the facility by determining the infiltration through the soil and any engineered barriers to the aquifer [28]. The fraction of waste released can then be determined by Equation (11).

$$RELFRAC = Site\ Factor \times Waste\ Factor \times Disposal\ Factor \tag{11}$$

The site factor takes into account the climate and the amount of rainwater that infiltrates through the topsoil of the reference facility. A steady, uniform infiltration is assumed, resulting in higher infiltration rates than are actually observed, making this a conservative estimate. Due to differences in the soil types, each reference facility will have a unique site factor. The humid permeable site receives 118 cm/year of rainfall and a large percentage of the rainfall infiltrates the sandy loam topsoil. Although the humid impermeable site receives the same amount of annual rainfall, much less water is allowed to infiltrate the clay soil. The sandy soil of the arid site allows for significant infiltration, but this site receives a very small amount of rainfall, about 25 cm/year. The EPA determined the site factors to be: 1.0 for the humid permeable site, 0.25 for the arid site, and 0.5 for the humid impermeable site [28]. These values represent fractions of infiltration and are unitless.

Waste corrosion and leaching will depend heavily on the waste form as well as amount of water passing through the waste as well as the rate of the water movement. A multi-phase leaching model can be used to determine waste corrosion, and assumes that infiltration is not uniform, but is concentrated in conduits due to heterogeneity of the waste [23]. The waste modeled in this study is all in the form of activated metals, which is a homogeneous waste, so a simplified leaching model can be used which assumes a uniform leaching rate. As previously discussed, this is a conservative estimate. By using this simplified model and assuming that the waste will not retain any of the radionuclides which have leached into groundwater (the waste K_d value is zero), the waste factor can be calculated by dividing the effective annual infiltration to the waste by the soil permeability. Using these assumptions, and the EPA's study of LLW activated metals, the waste factor determined for activated metal waste is approximately 1.0 [28].

The disposal factor is a representation of the fraction of precipitation which will infiltrate the disposal site and reach the aquifer, and is calculated by taking into account the permeability and specific yield of the soil. The specific yield is simply the soil porosity with the added effects of surface tension, cohesion, and adhesion of the groundwater to the soil particles, and is generally very close to the value of porosity, but is always lower in value.

Additionally, specific yield takes the grain size of the particles into account, unlike porosity. These values, shown in Table 18, can be used to determine the amount of the annual precipitation which will infiltrate the site and reach the aquifer each year.

Table 18: Soil Properties for Disposal Factor Calculations

	Permeability (cm/s)	**Average Specific Yield (%)**
Sand	10^{-3} to 10^{-1}	26
Silty sands, fine sand	10^{-5} to 10^{-3}	21
Silt, sandy silts, clayey sands	10^{-6} to 10^{-4}	18
Clay	10^{-9} to 10^{-6}	2
Concrete	10^{-8} to 10^{-7}	5

The EPA found in their study that for near surface disposal, the disposal factor is 2.25×10^{-4}, for intermediate depth disposal, 6.0×10^{-7}, and 1.0×10^{-7} for deep disposal [28]. Using Equation (11) and the values determined for each factor, the annual fraction of waste released can be calculated at each reference facility at the varying disposal depths. The results are shown in Table 19.

Table 19: Annual Fraction of Waste Released for Facilities at Different Depths (yr^{-1})

	Humid Impermeable	**Humid Permeable**	**Arid Permeable**
Near Surface	1.13E-04	2.25E-04	5.63E-05
Intermediate	3.00E-07	6.00E-07	1.50E-07
Deep	5.00E-08	1.00E-07	2.50E-08

The annual amount of radionuclides released from the waste into the environment can be calculated from these waste release fractions, which is equal to the rate of radionuclide transport at the bottom of the site.

Additional data used to determine well concentrations was primarily specified in the EPA's LLW study for each reference facility. However, specifics on the size of a GTCC waste facility are determined in this study by the projected volumes of waste and the size of the waste package. The waste packages were assumed to be placed into the facility one layer deep, yielding a height of 1.9 meters and a width and length of about 54 meters.

In the EPA's study, values for the distance from the facility to the well were based on site specific data. For this study, each facility will be set at 500 meters from the well, allowing results for the well concentrations to be easily compared to one another and to help determine any differences in the facilities. This value corresponds to the length of radionuclide transport in the aquifer, which is measured from the edge of the waste facility. Using the values of the site length and the distance from the site to the well, the distance from the center of the site to the well, D_h, is calculated to be 527 meters.

Reference facility data which was used from the EPA's study included the length of the vadose zone for each facility, the velocity in both the vadose zone and the aquifer, the porosity of the vadose zone and the aquifer, the bulk density of the site, the dispersion angle, α, and the depth of the well penetrating the aquifer. These variables vary among sites, and are shown in Table 20 [28]. Additionally, values for the vadose region length and velocity vary with site depth, and will be different for the near surface, intermediate, and deep facilities. These values can also be seen in Table 20.

Table 20: Data to Calculate Well Concentrations for Each Reference Facility [28]

		Humid permeable	**Arid permeable**	**Humid impermeable**
Distance traveled in the vadose region, L_v (m)	Near Surface	12.9	78.3	19.3
	Intermediate	50.0	50.0	50.0
	Deep	300	300	300
Average linear velocity in vadose region, V_v (m/yr)	Near Surface	1.0	0.1	1.0
	Intermediate	5.0	5.0	5.0
	Deep	0.5	0.5	0.5
Aquifer velocity, V_a (m/yr)		27.8	90	0.03
Aquifer porosity, ε_a		0.39	0.4	0.25
Vadose region porosity, ε_v		0.35	0.4	0.32
Soil bulk density, ρ_b (g/cm^3)		1.6	1.55	1.49
Depth of well penetrating the aquifer, D_w (m)		30.5	37	11
Dispersion angle, α (rad)		0.3	0.3	0.1

Data for the soil distribution coefficient, K_d, is not only site specific but, as previously discussed, is also specific to each element. Therefore, different K_d values are used for each element in the vadose zone and the aquifer at each site. From these K_d values, along with the bulk density and porosity, the retardation factor can then be calculated by $R = 1 + \frac{\rho_b K_d}{\varepsilon}$. Values for K_d (from the EPA's study) and R (calculated) are shown in Table 21.

Table 21: Soil Sorption Coefficients [28] and Retardation Factors for Radionuclides

	Humid permeable	**Arid permeable**	**Humid impermeable**
	Carbon-14		
$K_{d,v}$ (mL/g)	0.01	0.01	0.01
R_v	1.0457	1.0388	1.0466
$K_{d,a}$ (mL/g)	0.01	0.01	0.01
R_a	1.0410	1.0388	1.0596
	Nickel-59 and Nickel-63		
$K_{d,v}$ (mL/g)	150	3000	150
R_v	686.7	11626.0	699.4
$K_{d,a}$ (mL/g)	150	3000	150
R_a	616.4	11626.0	895.0
	Niobium-94		
$K_{d,v}$ (mL/g)	350	350	350
R_v	1601.00	1357.25	1630.69
$K_{d,a}$ (mL/g)	350	350	350
R_a	1436.90	1357.25	2087.00

These K_d values are within the ranges expected, based on the study by Sheppard and Thibault for related soil types, and are consequently assumed to be a valid representation for each site.

Now each variable in Equation (9) has been defined, and well concentrations can be calculated for each facility depth at the three representative locations. These values are then converted to expected public doses by using the dose conversion factors. Waste burial at near surface depths will result in the highest expected dose to the public. However, a less intuitive result is that the humid impermeable yields highest dose rates of the three reference facility locations. This is shown in results below in Table 22, with doses exceeding the EPA limit shown as highlighted values.

Table 22: Expected Dose to the Public (mrem/yr) for Reactor Components for Near Surface Disposal Facilities at Humid Impermeable Site

Humid Impermeable				
GE BWR-4				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	**2.32E+02**	**2.18E+02**	2.39E+00	1.96E+00
Ni-59	1.12E-56	2.48E-56	3.05E-58	2.50E-58
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00
GE BWR-6				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	**1.60E+02**	**2.00E+02**	1.49E+00	1.24E+00
Ni-59	9.13E-57	2.44E-56	1.90E-58	1.58E-58
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Westinghouse PWR (1)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	**1.55E+01**	**1.35E+03**	**7.10E+01**	**1.20E+02**
Ni-59	1.80E-57	7.01E-56	8.69E-57	8.65E-57
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 22 (continued)

<table>
<tr><th colspan="5"><i>Westinghouse PWR (2)</i></th></tr>
<tr><th></th><th>Upper Core Support Plate</th><th>Core baffle</th><th>Lower Core Support Barrel</th><th>Lower Core Support Plate</th></tr>
<tr><td>C-14</td><td>1.38E+01</td><td>1.62E+03</td><td>1.90E+02</td><td>7.47E+01</td></tr>
<tr><td>Ni-59</td><td>1.62E-57</td><td>1.35E-55</td><td>2.29E-56</td><td>5.48E-57</td></tr>
<tr><td>Ni-63</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td></tr>
<tr><td>Nb-94</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td></tr>
<tr><th colspan="5"><i>Combustion Engineering PWR</i></th></tr>
<tr><th></th><th>Upper Core Support Plate</th><th>Core baffle</th><th>Lower Core Support Barrel</th><th>Lower Core Support Plate</th></tr>
<tr><td>C-14</td><td>1.03E+00</td><td>1.79E+03</td><td>7.17E+02</td><td>5.08E+01</td></tr>
<tr><td>Ni-59</td><td>1.24E-58</td><td>1.17E-55</td><td>7.59E-56</td><td>3.77E-57</td></tr>
<tr><td>Ni-63</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td></tr>
<tr><td>Nb-94</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td></tr>
<tr><th colspan="5"><i>Babcock and Wilcox PWR</i></th></tr>
<tr><th></th><th>Upper Core Support Plate</th><th>Core baffle</th><th>Lower Core Support Barrel</th><th>Lower Core Support Plate</th></tr>
<tr><td>C-14</td><td>1.73E+00</td><td>6.66E+02</td><td>1.43E+02</td><td>1.31E+01</td></tr>
<tr><td>Ni-59</td><td>1.99E-58</td><td>4.80E-56</td><td>1.62E-56</td><td>8.12E-58</td></tr>
<tr><td>Ni-63</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td></tr>
<tr><td>Nb-94</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td><td>0.00E+00</td></tr>
</table>

Data for the two additional sites can be found in Appendix B. The radionuclide of greatest concern from activated metal waste is clearly carbon-14. All the modeled PWRs show expected doses over the limit for C-14 for the core baffle, lower core support barrel, and lower core support plate. Also, both of the Westinghouse PWRs' upper core support plates are over the regulatory limit for carbon-14. Both GE BWRs exceed the C-14 limit with their top fuel guide plate and core shroud. The humid permeable site also presents some concerns, but instead of C-14, the radionuclide in this case is Ni-59. Two PWRs, a Westinghouse (2) and Combustion Engineering model, were found to exceed the EPA limit by a small amount. The arid permeable site had no radionuclides which were over the limit. Well concentrations and expected public dose for intermediate and deep disposal depths are calculated in the same manner, and are shown in Appendix C and Appendix D, respectively.

These preliminary calculations show that waste disposal in near surface facilities with no barriers is not feasible at all the possible GTCC waste disposal facility locations. Some of the intermediate depth facilities will yield public doses near to the limit without any barriers. Expected public dose should always be kept as low as possible. If a facility filled with one type of component will leach amounts of waste into the environment that will exceed the allowed concentrations of the radionuclides, the average of these components in the facility together can also be expected to exceed the EPA limits. Because of this, additional barriers should be explored as an option for lowering these radionuclide concentrations.

6.7 Well Concentration Calculations for Sites with Engineered Barriers

Engineered barriers will limit the amount of water reaching the waste disposal area, and thus decrease the amount of radionuclides which will be leached from the waste. Due to high public exposures without barriers, using engineered barriers is a necessity. For near surface disposal of Class C radioactive waste, a typical barrier utilizes several different soil types, concrete, and rocks. An example of one such barrier can be seen in Figure 18.

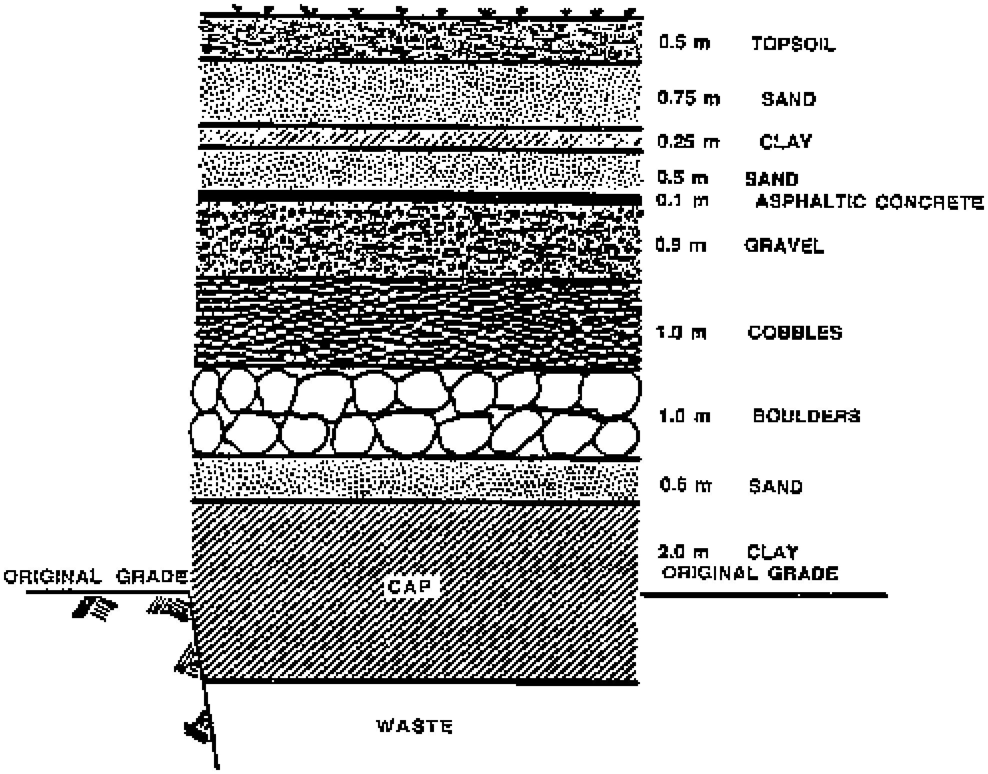

Figure 18: A Typical Class C Engineered Barrier [28]

Each soil type or rock size used in this barrier exhibits a unique permeability and porosity, and each layer will help to decrease the amount of water which will infiltrate the soil above the waste. Infiltration through the barriers can be calculated at as percentage of the original water which will allowed through each layer of the barrier. These percentages can be calculated using the permeability and specific yield of the soil, which are shown in Table 23.

Table 23: Soil and Rock Properties

	Permeability (cm/s)	**Average Specific Yield (%)**
Boulders	High	75
Cobbles	High	35
Gravel	10^{-2} to 1	23
Sand	10^{-3} to 10^{-1}	26
Silty sands, fine sand	10^{-5} to 10^{-3}	21
Silt, sandy silts, clayey sands	10^{-6} to 10^{-4}	18
Clay	10^{-9} to 10^{-6}	2
Concrete	10^{-8} to 10^{-7}	5

For an engineered barrier such as the one in Figure 18 along with the corresponding permeability and specific yield values, the water which will infiltrate into the waste disposal site is 6.04×10^{-4} % of the original annual precipitation, which is clearly a significant decrease.

Intermediate depth boreholes generally have an engineered barrier similar to the Class C barrier shown in Figure 18 with an additional 2 meters of concrete. Serving as another barrier between the surface and the waste, backfill is added to fill the otherwise void space. The added concrete and backfill lower the infiltration to 1.51×10^{-5} % of the annual precipitation.

Deep disposal depths are one of the main barriers used to help isolate deep geologic repositories and keep radionuclides from reaching the public. When considering adding barriers in deep geologic repositories, barriers used in Yucca Mountain barriers can be studied. These barriers are extremely engineered, and include a titanium drip shield and 5 ½ meters of backfill surrounding the waste. For this study, 5 ½ meters of sand backfill and 2 meters of concrete (used to represent a cheap version of the titanium drip shield) are assumed to be an acceptable barrier for GTCC waste. These barriers result in an infiltration rate that is 7.9×10^{-6} % of the initial precipitation.

These infiltration percentages for the different engineered barriers for each disposal depth are used to determine new release rates for each disposal facility considered These calculations assume that the release rate is directly proportional to the infiltration rate controlling the metal surface available for corrosion. The values for these release rates are shown in Table 24.

Table 24: Annual Fraction of Waste Released for Facilities with Engineered Barriers at Different Disposal Depths (yr^{-1})

	Humid Impermeable	**Humid Permeable**	**Arid Permeable**
Enhanced Near Surface	1.36E-11	1.36E-09	3.40E-10
Intermediate Depth Boreholes	9.07E-16	9.07E-14	2.27E-14
Deep Geologic Repository	7.88E-17	7.88E-15	1.97E-15

In order to obtain more accurate results for the reference facilities with engineered barriers, the entire volume of GTCC waste is modeled as activated metal. The total activity was determined from the data for the individual activated reactor components. To calculate this, the average radionuclide activity concentration for each component was determined for both BWR and PWR components. These activities were totaled, and the production of GTCC waste was assumed to be 60% by volume from PWR reactors, and 40% from BWR reactors. These percentages were determined from the approximate percentage of each type of operating reactor in the United States. The weighted-average total activity concentration of GTCC waste was then multiplied by the total GTCC volume of both stored and projected waste, around 5,600 m^3. The total activity for each nuclide as well as the overall total activity is shown in Table 25.

Table 25: GTCC Total Activity for 5,615 m^3 Activated Metals

Total Activity (Ci)	
C-14	2.14E+06
Ni-59	6.86E+06
Ni-63	1.19E+09
Nb-94	3.03E+07
Total	1.23E+09

With the total GTCC activity and the newly calculated release rate fractions, the expected dose to the public can be determined for each disposal facility location at each of the considered depths. The results can be seen in Table 26 (near surface disposal), Table 27 (intermediate disposal), and Table 28 (deep disposal). It can be seen from these tables that carbon-14 continues to be a major player in the expected public dose, although the EPA limits will not be exceeded with the engineered barriers intact.

Table 26: Expected Dose to the Public (mrem/yr) for Total GTCC Waste Volume for Enhanced Near Surface Disposal Facilities at each Facility Location

	Humid Impermeable	**Humid Permeable**	**Arid Permeable**
C-14	6.08E-01	6.43E-02	3.63E-03
Ni-59	1.17E-58	1.73E-01	6.25E-39
Ni-63	0.00E+00	3.75E-61	0.00E+00
Nb-94	0.00E+00	1.87E-03	7.63E-18
Total	*6.08E-01*	*2.39E-01*	*3.63E-03*

Table 27: Expected Dose to the Public (mrem/yr) for Total GTCC Waste Volume for Intermediate Depth Boreholes at each Facility Location

	Humid Impermeable	**Humid Permeable**	**Arid Permeable**
C-14	3.94E-05	4.29E-06	2.67E-07
Ni-59	4.52E-77	1.11E-05	5.60E-08
Ni-63	0.00E+00	2.67E-77	0.00E+00
Nb-94	0.00E+00	9.07E-08	1.15E-06
Total	*3.94E-05*	*1.55E-05*	*1.47E-06*

Table 28: Expected Dose to the Public (mrem/yr) for Total GTCC Waste Volume for Deep Geologic Repositories at each Facility Location

	Humid Impermeable	**Humid Permeable**	**Arid Permeable**
C-14	3.65E-05	3.97E-06	2.47E-07
Ni-59	1.15E-79	2.81E-08	7.33E-60
Ni-63	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	1.94E-30	2.46E-29
Total	*3.65E-05*	*4.00E-06*	*2.47E-07*

6.8 Sensitivity Analysis

Sensitivity analysis is a methodical approach to changing input parameters in a model in order to assess how the model's output changes. For this study, a single parameter sensitivity analysis was performed – one in which each parameter was varied at a time in order to assess the change in the expected public dose. Any time models or methods are used in a new application, performing a sensitivity analysis can help to determine the parameters which are the most sensitive to change, in order to more accurately characterize them. In this study, low level radioactive waste modeling has been applied to GTCC waste through the use of reference facilities. Determining important parameters in this study can help to better characterize a specific GTCC facility in the future. Modeling the most sensitive parameters indentified by the sensitivity analysis can be focused on for specific site characterization, so that they are the most accurately determined, and the most accurate results are calculated.

In the single variable sensitivity analysis, each parameter is tested for each reference facility. The parameters evaluated were the soil distribution coefficients, the distance traveled in the vadose region, the vadose region velocity, the aquifer velocity, the distance from the site to the well, the depth of the aquifer penetrating the well, and the site width. After determining quantitative changes in the public dose results, a qualitative sensitivity was assigned to each of these variables based on the percentage of change to the expected public dose caused by the input variable change. The qualitative sensitivity was characterized as high, medium, or low.

The sensitivity analysis was performed on each site with engineered barriers, for the carbon-14 radionuclide. Because the dose from carbon-14 was the most limiting, it was the radionuclide chosen to use for performing the sensitivity analysis. Any sensitivity of the parameters to change will be most apparent in the changes in the results for C-14. Numerical results from the sensitivity analysis can be found in Appendix E. The qualitative results are tabulated below in Table 29.

Table 29: Qualitative Results from Sensitivity Analysis

High sensitivity parameters:	- Distance from site to well - Soil distribution coefficient (K_d) values
Medium sensitivity parameters:	- Site width - Depth of well penetrating aquifer - Velocity in the aquifer
Low sensitivity parameters	- Average linear velocity in the vadose region

Although the sites behaved qualitatively the same with regards to the input parameter sensitivity, the humid impermeable site is the most sensitive to changes in these variables. For example, the humid impermeable site must be at least 250 meters from well for the expected dose to public to always be below the EPA's public dose limit. Other sites did not have this concern, and the expected public dose remained below the EPA limits for even the worst case scenarios.

Based on this sensitivity analysis, the greatest care should be taken in characterizing a humid impermeable site which might be used for waste disposal. Also, well-characterized soil distribution coefficients and accurate measurements of the distance between the disposal facility and the well are very important, as small changes in these values can result in large changes in the output.

7. Discussion of Results and Conclusions

The goal of this research was to provide new GTCC waste management strategies. This goal was pursued through three major Objectives, which were identified as Objective one – determining the composition and constituents of GTCC waste, Objective two – applying treatment methods to GTCC waste and determining their feasibility and practicality, and Objective three – studying disposal options for GTCC waste which could replace the default geologic repository disposal.

The first Objective was to determine the composition and constituents of GTCC waste, and was met through the compilation of data from previous DOE studies. Greater Than Class C waste is the highest category of low level radioactive waste, and is generally categorized by its origin of production. Commercial GTCC categories are nuclear utility waste, sealed sources, and other generator waste. DOE GTCC-like waste is a separate category which is DOE waste that exceeds the NRC's radionuclide limits for Class C LLW. Based on the review of the published literature, projected volumes for sealed sources and DOE GTCC-like waste are estimated to be higher than those for activated metals waste, but their uncertainties and difficulties in modeling these wastes and the potential to generate more activated metal waste through additional reactor decommissioning drove the choice to consider nuclear utility waste for potential alternative management in this study. Nuclear utility waste is primarily activated metals, which are generally type 304 stainless steel internal reactor components. These internal components are activated by the high neutron flux in operating reactors, creating activation products throughout the material, not only on the surface. The radionuclides which exist in these internal components over the Class C limits and cause them to be classified as GTCC waste are carbon-14, nickel-59, nickel-63, and niobium-94.

Objective two goals were to examine the possibility of reducing the impact of GTCC waste by lowering the volume of waste for disposal through possible treatments for GTCC waste. The proposed treatments – laser cutting, electrochemical processing, metal melting plant processing, and Q-CEP – were chosen for possible application to GTCC waste based on their success with LLW, UNF, and/or mixed waste. Due to the bulk-type contamination of the waste, laser cutting was not a good treatment option. An additional consideration which became very important in the application of chemically-based treatments was the Gibbs' free energy of formation for oxide and chloride compounds of nickel and niobium. Because these metals have free energies which are very similar to the other metal elements in the activated metal waste (specifically iron), they cannot be separated chemically by electrochemical processing, metal melting plant processing, and Q-CEP so these treatments are not feasible methods. This left disposal as the only remaining option for GTCC activated metal waste.

Regardless of the feasibility of proposed treatment options, any type of GTCC waste will require a disposal site. Even the best treatment technologies will never remove all the contamination from a material; no chemical separation process can be 100% efficient. In the case of activated metals, however, disposal was found to be the only option due to the technological limitations of applying the proposed treatment methods. Disposal options in consideration by the DOE were studied to determine if a geologic repository is necessary for GTCC waste. Objective three was met by applying data from INL reports on LLW disposal methods for cost estimates, and by calculating risks which can be expected from GTCC waste facilities based on parameters used for EPA LLW facilities.

Geologic repositories have been shown to be very difficult (if not impossible in the United States) to establish. Other disposal options, such as enhanced near surface burial or intermediate depth boreholes, were found to be much cheaper than a geologic repository, with results showing that, when possible, an enhanced near surface facility would be the most logical for this reason. Total costs were calculated for a 10,000 m^3 facility for each disposal type, and are summarized in Table 30.

Table 30: Total Costs for Each Facility for a 10,000 m^3 Capacity

Enhanced Near Surface	**Intermediate Depth Boreholes**	**Deep Geologic Repository**
$285 million	$3.26 billion	$25.14 billion (average)

The safety of these potential alternatives was examined through risk assessment calculations of hypothetical reference facilities. These reference facilities were chosen based on their ability to represent a variety of areas throughout the United States. These three facilities were differentiated by their climate and soil properties. The sites were chosen based on their success when used by the EPA for LLW disposal, and are humid permeable (loam), humid impermeable (clay), and arid permeable (sand). The EPA determined the use of these reference facilities to be an adequate method of estimating expected public doses due to drinking groundwater downstream of a potential disposal facility. Given the complexity of groundwater hydrology and radionuclide transport in the groundwater, a conservative approach was used for the estimation of models and parameters. A one-dimensional groundwater transport model was used for this reason, as well as conservative values for the soil distribution coefficients for the GTCC radionuclides of concern.

Based on this conservative modeling, preliminary estimations for the expected public dose showed that engineered barriers are a necessary part of the waste disposal facility. Shallow burial without engineered barriers showed that the humid impermeable site will result in expected public dose one to two orders of magnitude above the EPA's limit for carbon-14 in drinking water. Nickel-59 was found to be the important radionuclide for the humid permeable site, causing expected public doses to be very close to the EPA's limit. Engineered barriers in the form of a waste cap between the surface of the facility and the waste disposal location were studied to show that shallow burial is a feasible option with barriers to decrease the groundwater flow through the waste. The values expected dose to the public from a total GTCC activated metals waste volume with engineered barriers are summarized in Table 31.

Table 31: Expected Public Dose for 5,615 m^3 of GTCC Waste Modeled as Activated Metals for Different Waste Facility Locations and Depths with Engineered Barriers

	Humid Impermeable	**Humid Permeable**	**Arid Permeable**
Enhanced Near Surface Burial	6.08E-01	2.39E-01	3.63E-03
Intermediate Depth Boreholes	3.94E-05	1.55E-05	1.47E-06
Deep Geologic Repository	3.65E-05	4.00E-06	2.47E-07

Through the use of credible information from the DOE and EPA for modeling, and sensitivity analysis which was performed, the results from this study are determined to be in good standing. The sensitivity analysis showed that the humid impermeable site is the most sensitive to changing variables, and is the only site location where a minimum distance between the disposal facility and the well must be maintained to stay below the EPA limit. Because of this, humid permeable or arid permeable sites may be better location choices.

Therefore, while the deep geologic repository disposal option will always be the safest, as it has the lowest expected public dose associated with it for each site location, the process of establishing the facility will certainly be both lengthy and costly, as it has been in the past. Enhanced near surface burial and intermediate depth boreholes were both shown in this study to be cheaper options for disposal, and the expected public dose from these disposal facilities is estimated to be well within the EPA limits, making them a feasible alternative to deep geologic repositories for GTCC waste disposal.

8. Future Work Recommendations

This study only considered activated metal GTCC waste. Additional studies of interest would be those which considered sealed sources or DOE GTCC-like waste. Treatment possibilities may be feasible for either of these waste forms. A metal melting technique applied to the cesium-137 sealed sources could result in the cesium portion of the waste being separated from the metal portion. DOE GTCC-like waste would be a complex and difficult problem to approach, since they must be considered on a case-by-case basis. However, some of the scrap metals in DOE GTCC-like waste are TRU contaminated, and thus electrochemical processing could prove to be a feasible treatment option.

Additionally, in this study the waste package was assumed to fail immediately, alternatively meaning that the waste was placed directly into the disposal facility with no waste package. Adding a waste package would reduce the risk associated with the near surface and intermediate disposal sites decrease, and different materials could be explored in order to see the differences in this decreased dose. This waste package should focus on managing the dose from carbon-14, which could include a concrete waste package with a clay backfill. Costs and effectiveness of engineered barriers and waste packages could then be compared separately or as a combined risk management effort.

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APPENDICES

Appendix A. Comprehensive Set of Data Adapted from DOE’s 1994 report

The data for radionuclide activities in GTCC waste reactor components which is used in this study for the calculation of well concentrations and the expected dose to public was from the 1994 DOE GTCC waste inventory report. The data is listed here in tables by reactor type. Table A.1 is data for GE BWR-4. Table A.2 shows GE BWR-6 data. Tables A.3 and A.4 show data for the Westinghouse PWRs. Table A.5 contains data which describes the Combustion Engineering reactor, and Table A.6 contains data for the Babcock and Wilcox reactor.

Table A.1: GE BWR-4 Internal Components

GE BWR-4				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
Volume (m^3)	0.28	2.85	0.5	0.44
	Activity (Ci) After 6 years decay			
C-14	1.01E+02	9.49E+01	1.04E+00	8.54E-01
Ni-59	2.34E+02	5.16E+02	6.35E+00	5.21E+00
Ni-63	4.74E+04	6.76E+04	7.22E+02	5.93E+02
Nb-94	7.98E-01	9.41E-01	3.02E-02	2.48E-02
Total	4.77E+04	6.82E+04	7.30E+02	5.99E+02

Table A.2: GE BWR-6 Internal Components

GE BWR-6				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
Volume (m^3)	0.46	4.92	0.64	0.53
	Activity (Ci) After 6 years decay			
C-14	6.94E+01	8.71E+01	6.50E-01	5.38E-01
Ni-59	1.90E+02	5.07E+02	3.96E+00	3.28E+00
Ni-63	3.50E+04	6.33E+04	4.51E+02	3.73E+02
Nb-94	6.68E-01	9.60E-01	1.88E-02	1.56E-02
Total	3.53E+04	6.39E+04	4.56E+02	3.77E+02

Table A.3: Westinghouse PWR (1) Internal Components

Westinghouse PWR (1)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
Volume (m^3)	0.35	1.26	3.49	0.38
	Activity (Ci) After 6 years decay			
C-14	6.76E+00	5.88E+02	3.09E+01	5.23E+01
Ni-59	3.75E+01	1.46E+03	1.81E+02	1.80E+02
Ni-63	4.80E+03	3.16E+05	2.21E+04	2.97E+04
Nb-94	9.14E-02	8.65E+00	5.34E-01	6.73E-01
Total	4.84E+03	3.18E+05	2.23E+04	3.00E+04

Table A.4: Westinghouse PWR (2) Internal Components

Westinghouse PWR (2)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
Volume (m^3)	0.52	1.99	3.49	0.33
	Activity (Ci) After 6 years decay			
C-14	5.99E+00	7.05E+02	8.26E+01	3.25E+01
Ni-59	3.37E+01	2.82E+03	4.76E+02	1.14E+02
Ni-63	4.23E+03	4.34E+05	5.85E+04	1.85E+04
Nb-94	9.57E-02	1.28E+01	1.41E+00	4.25E-01
Total	4.27E+03	4.37E+05	5.91E+04	1.87E+04

Table A.5: Combustion Engineering PWR Internal Components

Combustion Engineering PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
Volume (m^3)	0.57	2.48	8.11	0.33
	Activity (Ci) After 6 years decay			
C-14	4.48E-01	7.79E+02	3.12E+02	2.21E+01
Ni-59	2.58E+00	2.43E+03	1.58E+03	7.85E+01
Ni-63	3.15E+02	4.38E+05	2.11E+05	1.27E+04
Nb-94	8.75E+03	1.26E-01	5.16E+00	2.96E-01
Total	9.07E+03	4.42E+05	2.13E+05	1.28E+04

Table A.6: Babcock and Wilcox PWR Internal Components

Babcock and Wilcox PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
Volume (m^3)	0.42	1.45	2.51	0.41
	Activity (Ci) After 6 years decay			
C-14	7.54E-01	2.90E+02	6.23E+01	5.71E+00
Ni-59	4.14E+00	1.00E+03	3.37E+02	1.69E+01
Ni-63	5.36E+02	1.63E+05	4.34E+04	2.75E+03
Nb-94	8.64E+03	4.52E+00	9.80E-01	5.91E-02
Total	9.18E+03	1.64E+05	4.38E+04	2.78E+03

Appendix B. Results for Expected Public Dose for Near Surface Facilities

Expected dose to the public for each individual GTCC waste reactor component results are shown here for near surface disposal facilities at each reference facility location. Table B.1 shows the calculated values for a Humid Permeable site. Values in Table B.2 correspond to the expected dose received from an Arid Permeable site, and Table B.3 displays results for the Humid Impermeable site.

Table B.1: Expected Dose to the Public (mrem/yr) for Reactor Components for Near Surface Disposal Facilities at a Humid Permeable Site

Humid Permeable				
GE BWR-4				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	4.95E-01	4.65E-01	5.09E-03	4.19E-03
Ni-59	4.71E-01	1.04E+00	1.28E-02	1.05E-02
Ni-63	2.23E-298	3.05E-298	3.25E-300	2.67E-300
Nb-94	7.13E-07	8.41E-07	2.70E-08	2.22E-08
GE BWR-6				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	3.40E-01	4.27E-01	3.18E-03	2.63E-03
Ni-59	3.82E-01	1.02E+00	7.96E-03	6.60E-03
Ni-63	1.58E-298	2.85E-298	2.03E-300	1.68E-300
Nb-94	5.97E-07	8.58E-07	1.68E-08	1.39E-08
Westinghouse PWR (1)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	3.31E-02	2.88E+00	1.51E-01	2.56E-01
Ni-59	7.54E-02	2.94E+00	3.64E-01	3.62E-01
Ni-63	2.16E-299	1.42E-297	9.94E-299	1.34E-298
Nb-94	8.17E-08	7.73E-06	4.77E-07	6.01E-07
Westinghouse PWR (2)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	2.93E-02	3.46E+00	4.05E-01	1.59E-01
Ni-59	6.78E-02	**5.67E+00**	9.57E-01	2.29E-01
Ni-63	1.91E-299	1.95E-297	2.64E-298	8.34E-299
Nb-94	8.55E-08	1.14E-05	1.26008E-06	3.80E-07

Table B.1 (continued)

Combustion Engineering PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	2.19E-03	3.82E+00	1.53E+00	1.08E-01
Ni-59	5.19E-03	**4.89E+00**	3.18E+00	1.58E-01
Ni-63	1.42E-300	1.97E-297	9.51E-298	5.70E-299
Nb-94	7.82E-03	1.13E-07	4.61E-06	2.65E-07
Babcock and Wilcox PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	8.80E-03	9.79E-01	1.22E-01	6.82E-02
Ni-59	1.98E-02	1.39E+00	2.70E-01	8.29E-02
Ni-63	2.42E-300	7.35E-298	1.95E-298	1.24E-299
Nb-94	7.72E-03	4.04E-06	8.76E-07	5.28E-08

Table B.2: Expected Dose to the Public (mrem/yr) for Reactor Components for Near Surface Disposal Facilities at an Arid Permeable Site

Arid Permeable				
GE BWR-4				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	2.83E-02	2.66E-02	2.92E-04	2.40E-04
Ni-59	3.53E-38	7.78E-38	9.57E-40	7.85E-40
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	3.33E-20	3.92E-20	1.26E-21	1.03E-21

Table B.2 (continued)

GE BWR-6				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	1.95E-02	2.44E-02	1.82E-04	1.51E-04
Ni-59	2.86E-38	7.64E-38	5.97E-40	4.94E-40
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	2.78E-20	4.00E-20	7.83E-22	6.50E-22
Westinghouse PWR (1)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	1.89E-03	1.65E-01	8.66E-03	1.47E-02
Ni-59	5.65E-39	2.20E-37	2.73E-38	2.71E-38
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	3.81E-21	3.60E-19	2.23E-20	2.80E-20
Westinghouse PWR (2)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	1.68E-03	1.98E-01	2.32E-02	9.11E-03
Ni-59	5.08E-39	4.25E-37	7.18E-38	1.72E-38
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	3.99E-21	5.33E-19	5.876E-20	1.77E-20
Combustion Engineering PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	1.26E-04	2.19E-01	8.75E-02	6.20E-03
Ni-59	3.89E-40	3.66E-37	2.38E-37	1.18E-38
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	3.65E-16	5.25E-21	2.15E-19	1.23E-20

Table B.2 (continued)

Babcock and Wilcox PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	2.12E-04	8.13E-02	1.75E-02	1.60E-03
Ni-59	6.24E-40	1.51E-37	5.08E-38	2.55E-39
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	3.60E-16	1.88E-19	4.08E-20	2.46E-21

Table B.3 Expected Dose to the Public (mrem/yr) for Reactor Components for Near Surface Disposal Facilities at Humid Impermeable Site

Humid Impermeable				
GE BWR-4				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	**2.32E+02**	**2.18E+02**	2.39E+00	1.96E+00
Ni-59	1.12E-56	2.48E-56	3.05E-58	2.50E-58
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table B.3 (continued)

GE BWR-6

	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	**1.60E+02**	**2.00E+02**	1.49E+00	1.24E+00
Ni-59	9.13E-57	2.44E-56	1.90E-58	1.58E-58
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Westinghouse PWR (1)

	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	**1.55E+01**	**1.35E+03**	**7.10E+01**	**1.20E+02**
Ni-59	1.80E-57	7.01E-56	8.69E-57	8.65E-57
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Westinghouse PWR (2)

	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	**1.38E+01**	**1.62E+03**	**1.90E+02**	**7.47E+01**
Ni-59	1.62E-57	1.35E-55	2.29E-56	5.48E-57
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Combustion Engineering PWR

	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	1.03E+00	**1.79E+03**	**7.17E+02**	**5.08E+01**
Ni-59	1.24E-58	1.17E-55	7.59E-56	3.77E-57
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Appendix C. Results for Expected Public Dose for Intermediate Depth Facilities

Expected dose to the public for each individual GTCC waste reactor component results are shown here for intermediate depth facilities at each reference facility location. Table C.1 shows the calculated values for a Humid Permeable site. Values in Table C.2 correspond to the expected dose received from an Arid Permeable site, and Table C.3 displays results for the Humid Impermeable site.

Table C.1 Expected Dose to the Public (mrem/yr) for Reactor Components for Intermediate Depth Facilities at a Humid Permeable Site

Humid Permeable				
GE BWR-4				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	6.20E-04	5.83E-04	6.38E-06	5.25E-06
Ni-59	3.17E-06	6.99E-06	8.60E-08	7.06E-08
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	8.47E-30	9.98E-30	3.20E-31	2.63E-31
GE BWR-6				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	4.27E-04	5.35E-04	3.99E-06	3.30E-06
Ni-59	2.57E-06	6.87E-06	5.37E-08	4.44E-08
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	7.09E-30	1.02E-29	1.99E-31	1.65E-31
Westinghouse PWR (1)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	4.15E-05	3.61E-03	1.90E-04	3.21E-04
Ni-59	5.08E-07	1.98E-05	2.45E-06	2.44E-06
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	9.70E-31	9.18E-29	5.66E-30	7.14E-30
Westinghouse PWR (2)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	3.68E-05	4.33E-03	5.08E-04	1.99E-04
Ni-59	4.57E-07	3.82E-05	6.45E-06	1.54E-06
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.02E-30	1.36E-28	1.50E-29	4.51E-30

Table C.1 (continued)

Combustion Engineering PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	2.75E-06	4.79E-03	1.91E-03	1.36E-04
Ni-59	3.50E-08	3.29E-05	2.14E-05	1.06E-06
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	9.28E-26	1.34E-30	5.47E-29	3.14E-30
Babcock and Wilcox PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	4.63E-06	1.78E-03	3.82E-04	3.50E-05
Ni-59	5.61E-08	1.36E-05	4.57E-06	2.29E-07
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	9.17E-26	4.79E-29	1.04E-29	6.27E-31

Table C.2 Expected Dose to the Public (mrem/yr) for Reactor Components for Intermediate Depth Facilities at an AridPermeable Site

Arid Permeable				
GE BWR-4				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	3.08E-04	2.90E-04	3.17E-06	2.61E-06
Ni-59	6.62E-57	1.46E-56	1.80E-58	1.47E-58
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.71E-29	2.02E-29	6.49E-31	5.33E-31

Table C.2 (continued)

GE BWR-6				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	2.12E-04	2.66E-04	1.98E-06	1.64E-06
Ni-59	5.38E-57	1.43E-56	1.12E-58	9.28E-59
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.44E-29	2.06E-29	4.04E-31	3.35E-31
Westinghouse PWR (1)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	2.06E-05	1.79E-03	9.43E-05	1.60E-04
Ni-59	1.06E-57	4.13E-56	5.12E-57	5.09E-57
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.96E-30	1.86E-28	1.15E-29	1.45E-29
Westinghouse PWR (2)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	4.57E-06	5.39E-04	6.31E-05	2.48E-05
Ni-59	2.38E-58	2.00E-56	3.37E-57	8.07E-58
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	5.14E-31	6.88E-29	7.58E-30	2.28E-30
Combustion Engineering PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	1.37E-06	2.38E-03	9.52E-04	6.74E-05
Ni-59	7.30E-59	6.88E-56	4.47E-56	2.22E-57
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.88E-25	2.71E-30	1.11E-28	6.36E-30

Table C.2 (continued)

Babcock and Wilcox PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	2.30E-06	8.85E-04	1.90E-04	1.74E-05
Ni-59	1.17E-58	2.83E-56	9.54E-57	4.78E-58
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.86E-25	9.71E-29	2.11E-29	1.27E-30

Table C.3 Expected Dose to the Public (mrem/yr) for Reactor Components for Intermediate Depth Facilities at a Humid Impermeable Site

Humid Impermeable				
GE BWR-4				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	2.85E-01	2.68E-01	2.93E-03	2.41E-03
Ni-59	6.47E-76	1.43E-75	1.76E-77	1.44E-77
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00
GE BWR-6				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	1.96E-01	2.46E-01	1.83E-03	1.52E-03
Ni-59	5.25E-76	1.40E-75	1.09E-77	9.07E-78
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table C.3 (continued)

Westinghouse PWR (1)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	1.91E-02	1.66E+00	8.72E-02	1.48E-01
Ni-59	1.04E-76	4.04E-75	5.00E-76	4.98E-76
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Westinghouse PWR (2)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	1.69E-02	1.99E+00	2.33E-01	9.17E-02
Ni-59	9.31E-77	7.79E-75	1.32E-75	3.15E-76
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Combustion Engineering PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	1.26E-03	2.20E+00	8.80E-01	6.23E-02
Ni-59	7.13E-78	6.72E-75	4.37E-75	2.17E-76
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Babcock and Wilcox PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	2.13E-03	8.18E-01	1.76E-01	1.61E-02
Ni-59	1.14E-77	2.76E-75	9.31E-76	4.67E-77
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Appendix D. Results for Expected Public Dose for Deep Facilities

Expected dose to the public for each individual GTCC waste reactor component results are shown here for deep facilities at each reference facility location. Table D.1 shows the calculated values for a Humid Permeable site. Values in Table D.2 correspond to the expected dose received from an Arid Permeable site, and Table D.3 displays results for the Humid Impermeable site.

Table D.1 Expected Dose to the Public (mrem/yr) for Reactor Components for Deep Facilities at a Humid Permeable Site

Humid Permeable				
GE BWR-4				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	1.03E-04	9.72E-05	1.06E-06	8.74E-07
Ni-59	5.28E-07	1.17E-06	1.43E-08	1.18E-08
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.41E-30	1.66E-30	5.34E-32	4.38E-32
GE BWR-6				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	7.11E-05	8.92E-05	6.65E-07	5.50E-07
Ni-59	4.29E-07	1.15E-06	8.94E-09	7.41E-09
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.18E-30	1.70E-30	3.32E-32	2.76E-32
Westinghouse PWR (1)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	6.91E-06	6.01E-04	3.16E-05	5.35E-05
Ni-59	8.47E-08	3.30E-06	4.09E-07	4.07E-07
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.62E-31	1.53E-29	9.44E-31	1.19E-30
Westinghouse PWR (2)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	6.13E-06	7.22E-04	8.46E-05	3.32E-05
Ni-59	7.61E-08	6.37E-06	1.08E-06	2.57E-07
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.69E-31	2.26E-29	2.49E-30	7.51E-31

Table D.1 (continued)

Combustion Engineering PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	4.58E-07	7.98E-04	3.19E-04	2.26E-05
Ni-59	5.83E-09	5.49E-06	3.57E-06	1.77E-07
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.55E-26	2.23E-31	9.12E-30	5.23E-31
Babcock and Wilcox PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	7.72E-07	2.97E-04	6.37E-05	5.84E-06
Ni-59	9.35E-09	2.26E-06	7.61E-07	3.82E-08
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	1.53E-26	7.99E-30	1.73E-30	1.04E-31

Table D.2 Expected Dose to the Public (mrem/yr) for Reactor Components for Deep Facilities at an Arid Permeable Site

Arid Permeable				
GE BWR-4				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	5.14E-05	4.83E-05	5.29E-07	4.35E-07
Ni-59	1.10E-57	2.43E-57	3.00E-59	2.46E-59
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	2.86E-30	3.37E-30	1.08E-31	8.88E-32

Table D.2 (continued)

GE BWR-6				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	3.54E-05	4.44E-05	3.31E-07	2.74E-07
Ni-59	8.96E-58	2.39E-57	1.87E-59	1.55E-59
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	2.39E-30	3.44E-30	6.73E-32	5.59E-32
Westinghouse PWR (1)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	3.44E-06	2.99E-04	1.57E-05	2.66E-05
Ni-59	1.77E-58	6.89E-57	8.54E-58	8.49E-58
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	3.27E-31	3.10E-29	1.91E-30	2.41E-30
Westinghouse PWR (2)				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	7.62E-07	8.98E-05	1.05E-05	4.13E-06
Ni-59	3.97E-59	3.33E-57	5.61E-58	1.34E-58
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	8.57E-32	1.15E-29	1.26E-30	3.81E-31
Combustion Engineering PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	2.28E-07	3.97E-04	1.59E-04	1.12E-05
Ni-59	1.22E-59	1.15E-56	7.45E-57	3.70E-58
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	3.13E-26	4.51E-31	1.85E-29	1.06E-30

Table D.2 (continued)

Babcock and Wilcox PWR				
	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	3.84E-07	1.48E-04	3.17E-05	2.90E-06
Ni-59	1.95E-59	4.72E-57	1.59E-57	7.97E-59
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	3.09E-26	1.62E-29	3.51E-30	2.12E-31

Table D.3 Expected Dose to the Public (mrem/yr) for Reactor Components for Deep Facilities at a Humid Impermeable Site

Humid Impermeable				
GE BWR-4				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	4.75E-02	4.47E-02	4.89E-04	4.02E-04
Ni-59	1.08E-76	2.38E-76	2.93E-78	2.40E-78
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00
GE BWR-6				
	Top Fuel Guide Plate	**Core Shroud**	**Fuel Support Castings**	**Core Support Plate**
C-14	3.27E-02	4.10E-02	3.06E-04	2.53E-04
Ni-59	8.75E-77	2.34E-76	1.82E-78	1.51E-78
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table D.3 (continued)

Westinghouse PWR (1)

	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	3.18E-03	2.76E-01	1.45E-02	2.46E-02
Ni-59	1.73E-77	6.73E-76	8.34E-77	8.29E-77
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Westinghouse PWR (2)

	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	2.82E-03	3.32E-01	3.89E-02	1.53E-02
Ni-59	1.55E-77	1.30E-75	2.19E-76	5.25E-77
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Combustion Engineering PWR

	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	2.11E-04	3.67E-01	1.47E-01	1.04E-02
Ni-59	1.19E-78	1.12E-75	7.28E-76	3.62E-77
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Babcock and Wilcox PWR

	Upper Core Support Plate	**Core baffle**	**Lower Core Support Barrel**	**Lower Core Support Plate**
C-14	3.55E-04	1.36E-01	2.93E-02	2.68E-03
Ni-59	1.91E-78	4.61E-76	1.55E-76	7.78E-78
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-94	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Appendix E. Quantitative Sensitivity Analysis Results

Quantitative results from the sensitivity analysis are shown in Tables E.1 through E.3. Table E.1 shows changes in the soil distribution coefficient (K_d) and the site width (B). Results from changing the distance traveled in the vadose zone, the vadose zone velocity, and the aquifer velocity are shown in Table E.2. In Table E.3, values for changing the distance to the well and the well penetrating depth in the aquifer are determined. The values in the column “Current Values” in Table E.1 are the base case values from the EPA’s LLW study. The cells shaded in gray in each table are the changing cells for that specific scenario.

Table E.1: Quantitative Sensitivity Analysis for Soil Distribution Coefficient and Site Width

Scenario Summary		Current Values:	Kd high	Kd low	Site Width
Changing Cells:					
$K_{d,v}$ (mL/g)	**HP**	0.01	150	0	0.01
	AP	0.01	150	0	0.01
	HI	0.01	150	0	0.01
$K_{d,a}$ (mL/g)	**HP**	0.01	150	0	0.01
	AP	0.01	150	0	0.01
	HI	0.01	150	0	0.01
Distance to	**HP**	500	500	500	500
Well (m)	**AP**	500	500	500	500
	HI	500	500	500	500
Site Width	**HP**	5.44E+01	5.44E+01	5.44E+01	2.70E+01
(m)	**AP**	5.44E+01	5.44E+01	5.44E+01	2.70E+01
	HI	5.44E+01	5.44E+01	5.44E+01	2.70E+01
Vadose distance	**HP**	12.9	12.9	12.9	12.9
(m)	**AP**	78.3	78.3	78.3	78.3
	HI	19.3	19.3	19.3	19.3
Aquifer Velocity	**HP**	27.8	27.8	27.8	27.8
(m/yr)	**AP**	90	90	90	90
	HI	0.03	0.03	0.03	0.03
Vadose Velocity	**HP**	1.00E+00	1.00E+00	1.00E+00	1.00E+00
(m/yr)	**AP**	1.00E-01	1.00E-01	1.00E-01	1.00E-01
	HI	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Well Pentrating	**HP**	30.5	30.5	30.5	30.5
Depth (m)	**AP**	37	37	37	37
	HI	11	11	11	11
Result Cells:					
Public Dose (mrem/year)	**HP**	6.43E-02	5.78E-03	6.43E-02	7.54E-02
	AP	3.63E-03	3.03E-27	3.64E-03	4.26E-03
	HI	6.08E-01	0.00E+00	6.86E-01	8.31E-01

Table E.2: Quantitative Sensitivity Analysis for Vadose Length, Vadose Velocity, and Aquifer Velocity

Scenario Summary		Lv low	Lv high	Va high	Vv high
Changing Cells:					
$K_{d,v}$ (mL/g)	**HP**	0.01	0.01	0.01	0.01
	AP	0.01	0.01	0.01	0.01
	HI	0.01	0.01	0.01	0.01
$K_{d,a}$ (mL/g)	**HP**	0.01	0.01	0.01	0.01
	AP	0.01	0.01	0.01	0.01
	HI	0.01	0.01	0.01	0.01
Distance to Well (m)	**HP**	500	500	500	500
	AP	500	500	500	500
	HI	500	500	500	500
Site Width (m)	**HP**	5.44E+01	5.44E+01	5.44E+01	5.44E+01
	AP	5.44E+01	5.44E+01	5.44E+01	5.44E+01
	HI	5.44E+01	5.44E+01	5.44E+01	5.44E+01
Vadose distance (m)	**HP**	10	500	12.9	12.9
	AP	10	500	78.3	78.3
	HI	10	500	19.3	19.3
Aquifer Velocity (m/yr)	**HP**	27.8	27.8	100	27.8
	AP	90	90	100	90
	HI	0.03	0.03	100	0.03
Vadose Velocity (m/yr)	**HP**	1.00E+00	1.00E+00	1.00E+00	1.00E+02
	AP	1.00E-01	1.00E-01	1.00E-01	1.00E+02
	HI	1.00E+00	1.00E+00	1.00E+00	1.00E+02
Well Pentrating Depth (m)	**HP**	30.5	30.5	30.5	30.5
	AP	37	37	37	37
	HI	11	11	11	11
Result Cells:					
Public Dose (mrem/year)	**HP**	6.43E-02	6.04E-02	1.79E-02	6.44E-02
	AP	3.95E-03	2.14E-03	3.27E-03	4.00E-03
	HI	6.09E-01	5.72E-01	1.54E-03	6.09E-01

Table E.3: Quantitative Sensitivity Analysis for Distance from Disposal Site to Well and the Well Penetrating Depth in the Aquifer

Scenario Summary		La low	La 250	Dw low
Changing Cells:				
$K_{d,v}$ (mL/g)	**HP**	0.01	0.01	0.01
	AP	0.01	0.01	0.01
	HI	0.01	0.01	0.01
$K_{d,a}$ (mL/g)	**HP**	0.01	0.01	0.01
	AP	0.01	0.01	0.01
	HI	0.01	0.01	0.01
Distance to Well (m)	**HP**	100	250	500
	AP	100	250	500
	HI	100	250	500
Site Width (m)	**HP**	5.44E+01	5.44E+01	5.44E+01
	AP	5.44E+01	5.44E+01	5.44E+01
	HI	5.44E+01	5.44E+01	5.44E+01
Vadose distance (m)	**HP**	12.9	12.9	12.9
	AP	78.3	78.3	78.3
	HI	19.3	19.3	19.3
Aquifer Velocity (m/yr)	**HP**	27.8	27.8	27.8
	AP	90	90	90
	HI	0.03	0.03	0.03
Vadose Velocity (m/yr)	**HP**	1.00E+00	1.00E+00	1.00E+00
	AP	1.00E-01	1.00E-01	1.00E-01
	HI	1.00E+00	1.00E+00	1.00E+00
Well Pentrating Depth (m)	**HP**	30.5	30.5	5
	AP	37	37	5
	HI	11	11	5
Result Cells:				
Public Dose (mrem/year)	**HP**	1.48E-01	9.96E-02	3.92E-01
	AP	8.36E-03	5.61E-03	2.68E-02
	HI	**5.36E+00**	2.31E+00	1.34E+00