BNL-107250-2014-IR

EVALUATION OF MAXIMUM RADIONUCLIDE GROUNDWATER CONCENTRATIONS FOR BASEMENT FILL MODEL

Zion Station Restoration Project

Terry Sullivan

December 2, 2014

Informal Report

Biological, Environmental & Climate Sciences Department

Brookhaven National Laboratory
P.O. Box 5000
Upton, NY 11973-5000
www.bnl.gov

Table of Contents

Figures

Tables

Evaluation of Maximum Radionuclide Groundwater Concentrations for Radionuclides of Concern
Zion Station Restoration Project

1. Introduction

Zion*Solutions* is in the process of decommissioning the Zion Nuclear Power Station (ZNPS). After decommissioning is completed, the site will contain two reactor Containment Buildings, the Fuel Handling Building and Transfer Canals, Auxiliary Building, Turbine Building, Crib House/Forebay, and a Waste Water Treatment Facility that have been demolished to a depth of 3 feet below grade. Additional below ground structures remaining will include the Main Steam Tunnels and large diameter intake and discharge pipes. These additional structures are not included in the modeling described in this report but the inventory remaining (expected to be very low) will be included with one of the structures that are modeled as designated in the Zion Station Restoration Project (ZSRP) License Termination Plan (LTP). The remaining underground structures will be backfilled with clean material. The final selection of fill material has not been made.

Remaining structures will contain residual radioactive material to varying extents. The bulk of the source term will be contained in the concrete floors. Current interior demolition plans are to remove all concrete inside the steel liner in the Unit 1 and Unit 2 Containment Buildings. Based upon concrete characterization data, the highest end state source term is anticipated to be contained in the Auxiliary Building floor located approximately 50 feet below grade. The end state source term will be at least 3 feet below grade in all remaining structures eliminating conventional pathways such as direct radiation and inhalation rendering groundwater related pathways the most significant potential sources of future exposure.

An important component of the decommissioning process is the demonstration that any remaining activity will not cause a hypothetical individual (average member of the critical group) to receive a dose in excess of 25 mrem/y as specified in 10 CFR Part 20 Subpart E. To demonstrate compliance with 10 CFR Part 20 Subpart E requires modeling of the fate and transport of radioactive material to a receptor. This involves characterization of the building basements to remain on site to quantify the amount of residual radioactivity, modeling the release of radioactivity from the concrete, and mixing with the water contained in the fill material. Transport away from the fill to a receptor well located outside of the basements may also be a relevant pathway.

A previous study (Sullivan, 2014a) performed screening calculations for the Auxiliary Building for 26 radionuclides. The Auxiliary Building was used for the screening calculations because it is expected to contain the majority of the residual contamination inventory at the time of license termination. This analysis was used by ZSRP along with characterization data and RESRAD modeling to screen out low dose significance radionuclides and identify eight radionuclides of concern (ROC's) Co-60, Ni-63, Sr-90, Cs-134, Cs-137, Eu-152, Eu-154, and H-3 for detailed assessment.

This report addresses the release of a given radionuclide inventory, for each of the ROCs, to the interstitial water of the fill material and calculates the equilibrium concentration at a well located in the middle of the subsurface remains of the seven buildings. The ratio of the resulting equilibrium water concentration in units of picocuries per liter (pCi/L) to the assumed inventory in units of Curies (Ci) for each building is used by ZSRP, in conjunction with the RESRAD code, to demonstrate compliance with 10 CFR 20 Subpart E.

Calculation of the fill interstitial water concentration requires site-specific information on the hydrogeologic properties (effective porosity and bulk density) and chemical transport properties (sorption). Conestoga-Rovers & Associates (CRA) has collected a substantial amount of site-specific hydrogeologic data (CRA, 2014).

Brookhaven National Laboratory (BNL) has determined site-specific sorption data for five nuclides that are ROCs with four soil types, two concrete types of construction demolition debris, two cinder block materials, and one grout material that are under consideration for the fill (Yim, 2012, Milian, 2014). Two ROCs, Eu-152 and Eu-154 have not had site-specific sorption measurements. A report (Sullivan, 2014) provided recommended values to use for dose assessment based on measured values, when available, and literature values in other cases. For nuclides with site-specific measured values, the lowest measured distribution coefficient in any of the media tested was recommended for use.

The objectives of this report are:

a) To present a simplified conceptual model for release from the buildings with residual subsurface structures that can be used to provide an upper bound on contaminant concentrations in the fill material.
b) Provide maximum water concentrations and the corresponding amount of mass sorbed to the solid fill material that could occur in each building for use in dose assessment calculations.
c) Estimate the maximum concentration in a well located outside of the fill material.
d) Perform a sensitivity analysis of key parameters.

2. Conceptual Models of Release

2.1 Site Overview

Figure 1 provides the site layout at ZNPS located on the shores of Lake Michigan. Major features include two reactor Containment Buildings (Unit-1 and Unit-2 in Figure 1, a Fuel Handling Building, Auxiliary Building, Turbine Building, Crib House, and Waste Water Treatment Facility (WWTF).

The proposed decommissioning approach involves removal of regions with high-levels of contamination through a remediation process. There will be some surface contamination and volumetric contamination left in place. This contamination will provide a potential source of radioactivity to the groundwater. These structures will be filled with non-contaminated material. Fills that have been under consideration include:

- Clean concrete construction debris (CCDD);
- Clean cinder block material;
- Clean Sand
- Clean Grout

Recently, grout has been eliminated from consideration for fill material. The fill may contain a combination of the three remaining choices or it could only include sand. Cinder block or CCDD will be blended with sand to reduce the available pore space. The total capacity of the underground structures (basements) for placement of fill is approximately 6 million cubic feet.

There are seven buildings (Figure 1) that will have residual structures beginning three feet below grade. Contaminated concrete from inside the liner in the Containment Buildings will be removed and this will substantially decrease the amount of contamination in the Containment Buildings. Characterization data indicates there is no significant liner contamination or concrete activation past the liner, leaving the Auxiliary Building with the highest residual contamination. Low-levels of contamination were found in the Turbine Building. The below grade concrete to remain in the Fuel Handling Building and Transfer Canals has not yet been characterized.

2.2 Modeling Overview

The Disposal Unit Source Term – Multiple Species (DUST-MS) computer code has been selected to calculate the source term release and equilibrium water concentration at the receptor well which is assumed to be in the center of the backfilled building. DUST-MS has received wide-spread use in subsurface radionuclide release calculations and undergone model validation studies (Sullivan, 1993; 2006). The equilibrium model can be easily calculated by hand. However, DUST-MS is necessary when simulating diffusion controlled release or transport to a receptor well. To maintain consistency between all calculations DUST-MS was used for all simulations.

An important parameter is the volume of water available to mix with released radionuclides. Another important parameter defines how the release of contaminants will be modeled. In many buildings the contamination is expected to be loosely bound or near the surface of the remaining structure. In these buildings, the release is assumed to occur instantly, such that the entire inventory is available immediately after license termination. In some buildings the contamination is expected to have diffused into the concrete resulting in volumetrically contaminated concrete. For these buildings, a diffusion controlled release model is used. The Auxiliary Building has been characterized and shown to be contaminated to a depth of at least the first inch of the concrete. The concrete in the Fuel Handling Building and Transfer Canals is also expected to be volumetrically contaminated below the liner but the extent of this contamination will not be characterized until the liner is removed. Diffusion controlled release is assumed for the Auxiliary and Fuel Handling Building/Transfer Canals.

Table 1 summarizes the total fill volume available for mixing and the release assumptions for each building. The mixing volume is calculated assuming that the water level in the basements is equal to the natural water table elevation outside of the basements (i.e., 579 feet), which is the minimum long term level that could exist in the basements. The amount of water available for mixing will be the total fill volume multiplied by the porosity of the backfill. For conservatism it was assumed that the backfill had only 25% porosity. This is believed to be a minimum value for porosity because it will be difficult to achieve this packing density. For example, the native sand has total porosity greater than 30%.

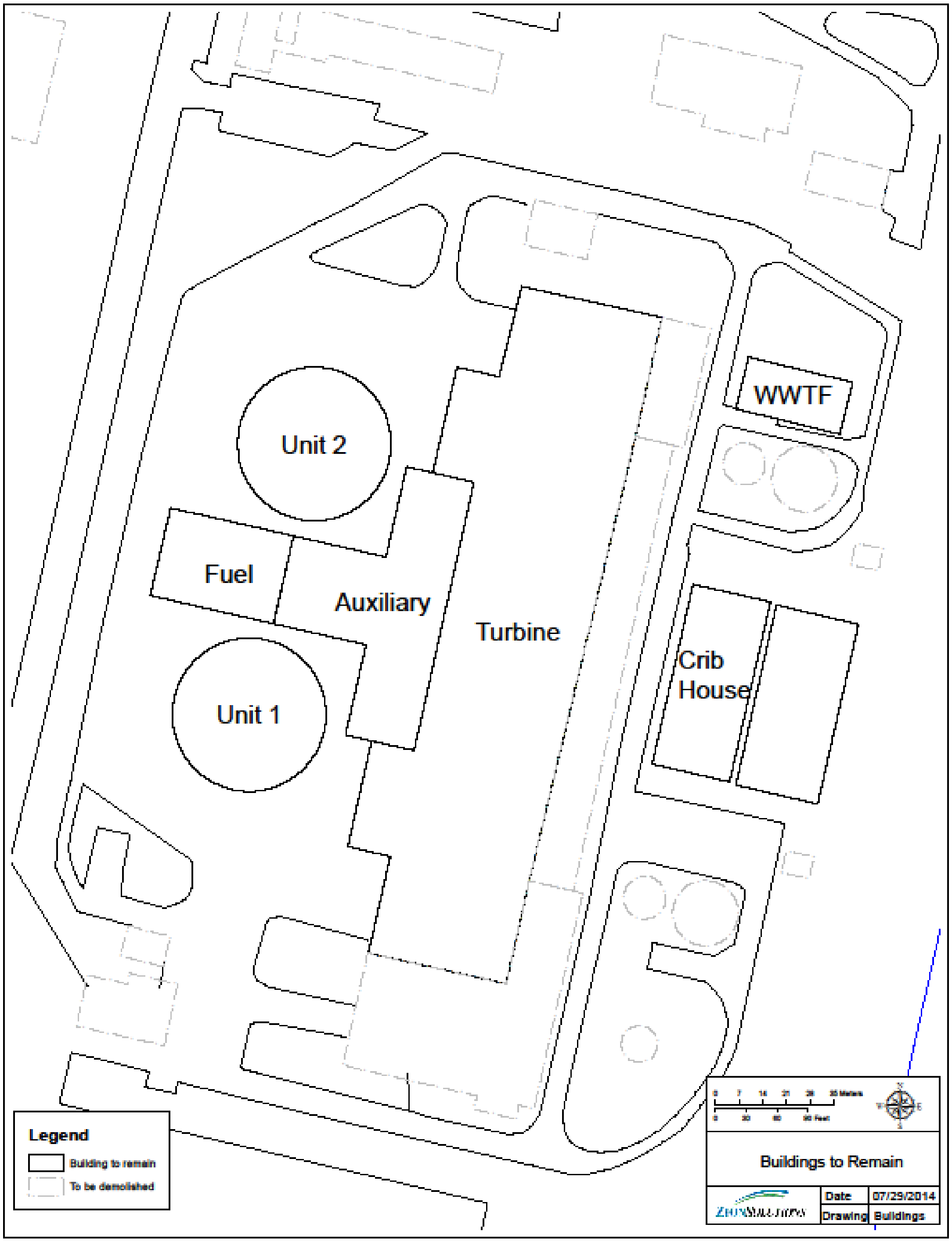

Figure 1 Zion Site building layout.

Table 1 Mixing volume and release rate assumption

Building	Volume* (m^3)	Release Rate Assumption
Unit 1 Containment	6.54E+03	Instant Release (loose surface contamination)
Unit 2 Containment	6.54E+03	Instant Release (loose surface contamination)
Auxiliary	2.84E+04	Diffusion Controlled Release (concrete contamination at depth in concrete)
Turbine	2.61E+04	Instant Release (the limited contamination present is at the concrete surface with very limited contamination at depth.)
Crib House and Forebay	3.05E+04	Instant Release (limited or no surface contamination)
Waste Water Treatment Facility	1.44E+02	Instant Release (limited or no surface contamination)
Spent Fuel Pool and Transfer Canals	2.08E+02	Diffusion Controlled Release (Concrete contamination expected at depth under the liner)

* (From Farr, 2014)

In the Containment Buildings only loose surface contamination is expected to remain. The distribution of the surface source term is generally expected to be uniform over the remaining liner surface. The release mechanism is therefore Instant Release (e.g. 100% of the inventory is assumed to be instantly released) because the source term is surface contamination only on the remaining steel liner.

The contamination in the Auxiliary Basement is found at depth in the concrete, predominantly in the floor. Diffusion Controlled Release was therefore used to estimate the rate of radionuclide release for the Auxiliary Basement.

The Turbine Basement source term is very limited and associated with surface contamination in concrete and embedded piping in the Turbine Building foundation. The inventory in the concrete and embedded piping is assumed to be instantly released.

There is very little, if any, contamination in the Crib House/Forebay and Waste Water Treatment Facility. The minimal contamination present is assumed to be on the concrete surfaces and instantly released.

Diffusion Controlled Release was used to estimate the source term release rate for the Fuel Handling Building Basement and Fuel Transfer Canals due to expected contamination at depth in concrete after the liners are removed.

In addition to the primary modeling used for 10 CFR 20 Subpart E compliance, a check calculation was performed to determine the water concentration in a well assumed to be placed outside of the building basements at the downstream (eastern) edge of the Turbine Building. The check calculation applies transport modeling to confirm the expectation that the concentration in water outside of the Basements would be lower than inside and that assuming the well is placed inside the Basements is conservative for dose assessment. The area for flow was calculated using the width of the building perpendicular to the primary direction of water flow (from west to east to the Lake in Figure 1) and the mixing height. The contaminated zone in the flow model is the fill material. Outside of the contaminated zone (i.e., outside of the basements) a mixture of fill sand and native soil is simulated. Table 2 contains flow areas for the calculations.

The inventory for each building was based on a uniform contamination level of 1 pCi/m^2 on the wall and floor surfaces. This contamination level was used for modeling convenience only. The total inventory used in the simulation is the value of interest because the total inventory will be used for scaling with the final inventory measured in each basement after remediation is completed. For example, the Auxiliary Building has 6503 m^2 of total wall and floor surface area that leads to a total of 6503 pCi in this simulation. To scale to the actual inventory obtained by measurement after remediation is completed, the results of the simulations presented in this report should be multiplied by the ratio of the measured inventory to simulated inventory.

Material properties were chosen to match site-specific values to the extent possible. Sorption coefficient, K_d, values were based on the measured values for Zion soils, concrete, cinder block, and grout (Yim, 2012, Milian, 2014) when available and literature values when site-specific values were not available. A review of literature values and rationale for selecting K_d for dose assessment was performed (Sullivan, 2014). The K_d values selected from the literature were chosen to give a conservative estimate of water concentration (highest value) for dose assessment. When site-specific values are available, the lowest K_d value measured in any fill material or soil was selected.

The compliance assessment requires prediction of the release and transport of contaminants to the hypothetical individual. Characterization studies and assessments by Zion*Solutions* have identified the following ROCs (Table 3). All nuclides in Table 3 were used in the simulation of maximum groundwater concentration.

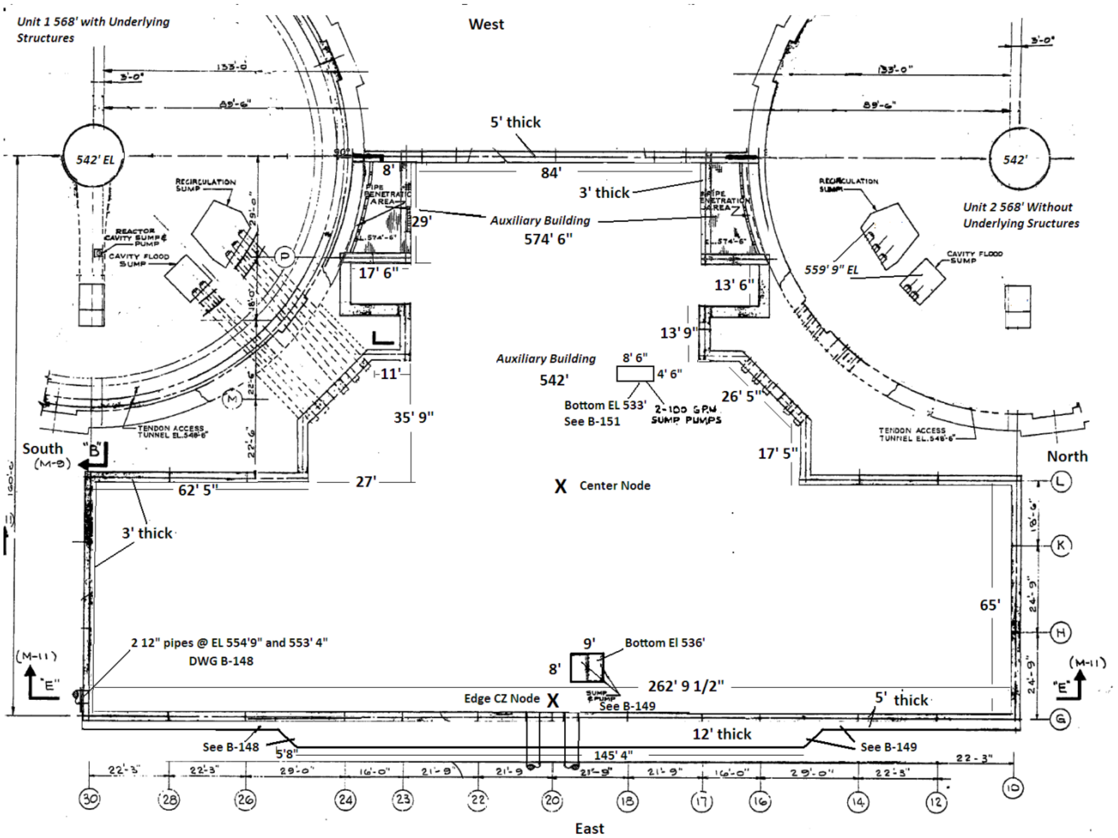

Figure 2. Geometry of the Auxiliary Building.

Table 2 Geometric Parameters and Unit Inventory for Residual Structures (Farr, 2014)

Structure	Basement Floor Elevation (feet)	Distance to Water Table meters	Structure Total Surface Area (m^2)	Inventory (Ci)
Auxiliary Building	542	11.28	6503	6.50E-09
Unit 1 Containment	565	4.27	2759	2.76E-09
Unit 2 Containment	565	4.27	2759	2.76E-09
Crib House & Forebay	537	12.80	6940	6.94E-09
Turbine Building, Main Steam, Diesel Gen Oil Storage	560	5.79	14679	1.468E-08
Spent Fuel Pool and Transfer Canals	576	0.91	780	7.80E-10
Waste Water Treatment Facility	577	0.61	1124	1.124E-09

Table 3 Potential Radionuclides of Concern at the Zion Nuclear Power Station

Radionuclides
H-3
Co-60
Ni-63
Sr-90
Cs-134
Cs-137
Eu-152
Eu-154

2.3 Release Models

2.3.1 Instant Release

For the instant release model the key parameters are the distribution coefficient (K_d), porosity and bulk density of the fill material. The Containment Buildings, Crib House/Forebay, Turbine Building, and the Waste Water Treatment Facility (WWTF) are modeled using an instant release.

2.3.2 Release Rate: Diffusion Controlled Release from the concrete

In two of the buildings, Auxiliary and Fuel, there is volumetric contamination in the concrete floors and walls that will release over time as the nuclides diffuse out from the concrete into the water. Therefore, the time-dependent diffusion controlled release rates are used to calculate the maximum water concentrations for the Auxiliary and Fuel Buildings.

Studies have been conducted for the diffusion in concrete of the radionuclides under consideration at Zion (H-3, Co-60, Ni-63, Sr-90, Cs-134, Cs-137, Eu-152, and Eu-154). The diffusion coefficient from concrete will depend on the water to cement ratio used in forming the concrete and the aggregate. A typical range from the literature is presented in Table 4. The maximum in the range was selected for use in the analysis.

Table 4 Typical diffusion coefficients in cement for radionuclides of concern

Nuclide	Diffusion Coefficient Range (cm^2/s)	Selected Diffusion Coefficient (cm^2/s)	Reference
H-3	6.0E-09 – 5.5E-07	5.5E-07	Szanto, 2002
Co-60	5.0E-12 – 4.1E-11	4.1E-11	Muurinen,1982
Ni-63	8.7E-10 – 1.1E-09	1.1E-09	Jakob, 1999
Sr-90	1.0E-11 – 5.2E-10	5.2E-10	Sullivan, 1988
Cs-134; Cs-137	4.0E-11 – 3.0E-09	3.0E-09	Atkinson, 1986
Eu-152; Eu-154	1.0E-12 – 5.0E-11	5.0E-11	Serne, 1992; Serne, 2001

In the conceptual model for diffusion controlled release it is assumed that the concrete is uniformly contaminated over a 0.5 inch thickness and that all of the material is released at the surface (i.e. it does not diffuse further into the concrete). This assumption is equivalent to having one side of the contaminated zone as a no flow boundary. In practice, some of the nuclides would continue to diffuse deeper into the concrete initially and thereby increase the time before being released to the water. The assumption that everything is released into the water is modeled with an analytical solution for diffusion from a slab. To simulate release at the surface, the slab is modeled as being one inch thick and allowed to flow out of both sides of the slab. Using the principle of symmetry, the centerline is a no flow boundary and this is equivalent to having a slab 0.5 inch thick but preventing diffusion further into the cement. This is accomplished in DUST-MS by modeling a slab with a thickness of one inch, which reduces the calculated waste form concentrations from the assumed inventory by a factor of 2 as compared to a one inch thickness. The contributions from both sides of the slab are then summed to calculate the maximum release from one surface of the 0.5 inch slab. Using symmetry, the release from this model, which has two sides, is equivalent to release from a 0.5 inch thick contaminated zone.

2.4 Receptor Well Outside the Turbine Building

If CCDD or crushed cinder block is used as fill material, the pH of the water in the fill region will rise to levels that make it non potable. Notwithstanding the high pH condition, the conceptual model assumes that this water will be used as a residential water supply, livestock water supply and for irrigation. This section addresses a more credible scenario where the well is located outside of the basements.

The Auxiliary Building will have the highest levels of residual contamination. The Auxiliary Building is adjacent to the Turbine Building and there are penetrations that will remain in place and connect these buildings. The Containment Buildings are also connected to the Auxiliary Building by penetrations but Containment will have minimal contamination after removal of all internal concrete.

The closest place to put a well in the shallow aquifer outside of the Auxiliary Building is just outside and to the east of the Turbine Building. The Auxiliary Building foundation rests on the clay aquitard and a well located directly to the east of the Auxiliary building, and under the Turbine Building floor would not flow. To examine the maximum concentration that could be obtained from a well in the soil, DUST-MS was used to predict the concentrations 2 meters outside of the eastern edge of the Turbine Building, Figure 1. Therefore, the modeled domain contains the Auxiliary Building and the section of the Turbine Building that aligns with the Auxiliary Building and groundwater flow direction. A schematic representation of the model domain is presented in Figure 3. The dotted rectangular region is the modeled region and consists of clean soil upstream from the Auxiliary Building, the Auxiliary and Turbine Buildings and clean soil downstream of the Turbine Building. A hypothetical well located 2 m from the edge of the Turbine Building is shown. To address the higher contamination levels anticipated in the Auxiliary Building, the Turbine Building contamination level was reduced by a factor of 1000 to 0.001 pCi/m^2. The groundwater flow rate through the buildings is assumed to be at the rate determined by the local flow conditions at the site.

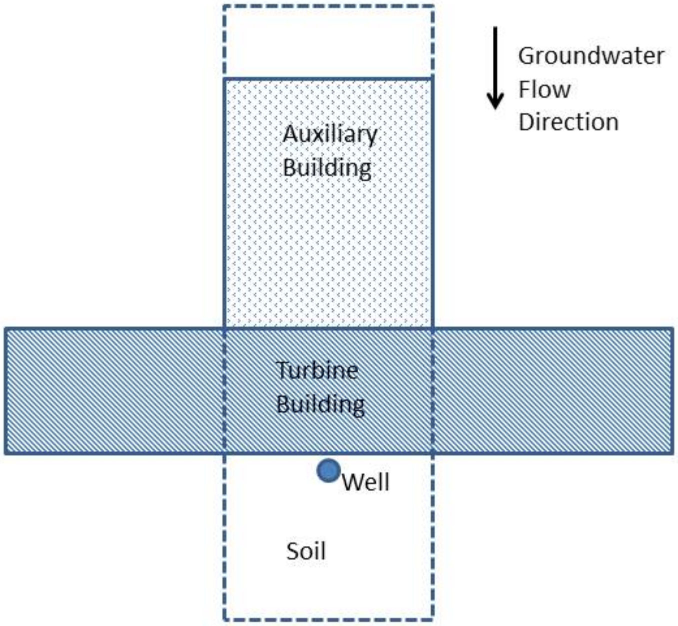

Figure 3 Schematic Representation of Flow the geometry used to assess flow to a well outside the Turbine Building.

3. Analysis Parameters

All release models are established using the unit source term and grounded in conservative estimates of site-specific measured values for the model parameters where available. The instant release model was used in buildings with minimal inventory or with only surface contamination expected. The instant release model is meant to provide a conservative upper bound estimate for groundwater concentration. A diffusion release model is used in buildings with volumetric contamination of the concrete.

3.1 Parameters

Initial conditions assumed that the groundwater concentration of each contaminant was zero everywhere. The source term is modeled such that the results can be scaled to the actual inventory of the various buildings on site. For this modeling scenario, each building was modeled with the assumption of uniform contamination across the floor of the entire building.

The exact constitution of the backfill has not been decided yet. Therefore, the bulk density and porosity are unknown. A bulk density of 1.5 grams per cubic centimeter (g/cm^3) and an effective porosity of 0.25 were selected for the screening model. With any of the fill materials it is difficult to conceive of reducing the packing material below this value. The effective porosity helps determine the amount of water available for mixing and through selecting a low value for this parameter the estimates of concentration in the water will be biased high (e.g. conservative with respect to dose estimates).

The distribution coefficients (K_d) are important parameters in controlling the equilibrium concentrations and transport (if modeled). A study (Sullivan, 2014) reviewed the literature and site-specific data to provide conservative values for K_d in assessing groundwater dose. In selecting values from the literature, environmental conditions with high pH (cement sorption data) as well as environmental data (soil sorption) data were considered. For conservatism the minimum value from these conditions was selected. For nuclides with measured site-specific K_d values, the lowest measured K_d in any backfill or soil was selected. Selected values are in Table 5.

For the base case model it is assumed that there is no flow through the system. This leads to the highest concentrations possible and is conservative. To accomplish this in DUST-MS the flow velocity is set to zero.

Table 5 Selected distribution coefficients (Sullivan, 2014)

Radionuclide	Half Life (years)	Basement Fill K_d to Be Used cm^3/g
H-3	12.3	0
Co-60	5.27	223
Ni-63	96	62
Sr-90	29.1	2.3
Cs-134	2.06	45
Cs-137	30	45
Eu-152	13.4	95
Eu-154	8.2	95

3.1.1 Diffusion Controlled Release Model

For the diffusion release model the selected diffusion coefficients were presented in Table 4. The base case model assumes that contamination is uniformly distributed over 0.5 inch in the concrete and all contamination migrates out of the concrete into solution. Additional diffusion into the concrete is not allowed in the model. This maximizes the release rate.

3.1.2 Model Geometry

DUST-MS is a one dimensional model. The conceptual model contains a contaminated floor in the direction of flow. DUST-MS model requires a flow area to calculate the correct concentrations above the floor. The flow area is defined as the area perpendicular to the transport direction. In these simulations, the transport direction is towards the Lake. Therefore, the flow is the product of the height of the water table above the floor and the width of the building that is parallel to the Lake. Table 6 provides the height to the water table based on a 579 foot elevation, effective distance parallel to the Lake, flow area, and effective length of the contaminated zone. The product of the flow area and length of the contaminated zone gives the total volume for each building. These widths, height to the water table, and volumes were calculated by Zion*Solutions* staff (Farr, 2014).

Table 6 Model Geometry for all simulations.

Structure	Width or Radius m	Height to Water Table m	Flow Area (m^2)	Contaminated Zone Length (m)	Void Space to WT m^3
Containment Buildings	20.95	4.27	140.4	44.81	6537
Auxiliary Building	80.11	11.28	903	31.5	28445
Turbine Building	40.84	5.79	571.5	45.73	26135
Crib House and Forebay	52.12	12.8	667.2	45.75	30524
Waste Water Treatment Facility	14.63	0.61	8.919	16.09	144
Spent Fuel Pool and Transfer Canals	10.06	0.91	18.64	11.17	208

3.1.3 Receptor Well Parameters for Transport Model

For the base case the flow velocity is set to zero in the DUST-MS input file. To simulate transport to a receptor well soil properties and the groundwater flow rate are required. These values are presented in Table 7. The K_d values used were identical to those in the equilibrium model. Site-specific soil Kd values for Co (1161 centimeters cubed per gram - $cm^{3/}g$) and Cs (527 cm^3/g) are much higher than used in the analysis and their use would lead to lower predicted concentrations. For conservatism, it was decided that the lowest K_d value from all sources (Sullivan, 2014) would be used. The reason for using the lowest K_d values is that the water leaving the building structures would have a high pH due to the backfill material. This could lead to changes in sorption on the soil materials as compared to the test results obtained using the local groundwater.

Table 7 Transport Parameters used to calculate peak concentrations in a receptor well located outside of the basements.

Parameter	Value	Reference
Soil Density	1.81 (g/cm3)	CRA, 2014
Soil Effective Porosity	0.29	CRA, 2014
Groundwater Darcy Velocity	41.6 m/y	CRA, 2014
Soil Kd: Co-60	223 (cm^3/g)	Sullivan, 2014
Ni-63	62 (cm^3/g)	
Sr-90	2.3 (cm^3/g)	
Cs-134	45 (cm^3/g)	
Cs-137	45 (cm^3/g)	
Eu-152	96 (cm^3/g)	
Eu-154	95 (cm^3/g)	

The modeled geometry is presented in Figure 3. The width of the Auxiliary Building is 80.1 m, which is less than the Turbine Building. The one-dimensional simulation requires that the width perpendicular to flow remain constant. Therefore, for this simulation only the portions of the Turbine Building downstream from the Auxiliary Building are modeled. The length of the Turbine Building parallel to flow is 29.3 m. Therefore, the total floor area of the Turbine Building for this simulation is 2,344 square meters (m^2). This is not the actual area of the Turbine Building modeled in the base case. The receptor well is 2 meters downstream of the Turbine Building. This assumption will have a minor impact on the final results.

The one-dimensional simulation also requires the depth to the water table to remain the same in both buildings. The actual depth to the water table is deeper in the Auxiliary Building as compared to the Turbine Building. The geometry and flow direction requires that any release from the Auxiliary Building travel through the Turbine Building. Therefore, the appropriate depth to the water table for this simulation is that of the Turbine Building, 5.79 m (19 ft.). This value was used to calculate the mixing volume. The total area available for flow (building width multiplied by the height to the water table) is 463.7 m^2.

The inventory of the Auxiliary Building is based on 1 pCi/m^2 and the total inventory is 2554 pCi. The inventory of the Turbine Building at the time of license termination will be very close to zero but is assumed to be 0.001 pCi/m^2 for a total inventory of 14.7 pCi. The differences in total area lead to the slightly less than a factor of 1,000 difference in total inventory in the two buildings.

3.1.4 Sensitivity Analysis Parameters

To quantify the impact of changes in key variables on the predicted concentrations additional calculations were performed. Characterization data indicate that the Auxiliary Building will have the majority of residual contamination. For this reason, all sensitivity analyses will be performed for that building. For sensitivity analysis all parameters were varied by 25% from their initial base case value. The range of parameters is presented in Table 8

Table 8 Parameters and their range in the sensitivity analysis.

Parameter	Base Case Value	Range
K_d	Table 6 (nuclide dependent)	± 25 % of Value in Table 5
Porosity	0.25	0.19 – 0.31
Bulk Density	1.5 g/cm^3	1.1 – 1.8 g/cm^3

In calculating potential exposures one scenario considers removing the drill spoils from a hypothetical intruder well placed in the middle of the building. These drill spoils are mixed with surface soil and the resulting dose from the contaminated soil is calculated. The K_d values selected for the base case in the backfill were selected to maximize groundwater concentrations. To examine the impact from using a higher K_d value on the soil concentrations the base case was modified to use the K_d values from the native sand. For tritium (H-3) the K_d value was raised from 0 to 1. Site-specific values for Europium K_d are not available. The 75th percentile value for K_d in soils (7222 ml/g) was used in the analysis (NRC, 2000). Table 9 lists the selected Kd values for the drill spoils sensitivity analysis.

Table 9 K_d values selected to examine the sensitivity of drill spoils predicted soil and groundwater concentrations

Nuclide	K_d (ml/g)
H-3	1
Co-60	1161
Ni-63	62
Sr-90	2.4
Cs-134	615
Cs-137	615
Eu-152	7721
Eu-154	7721

4 Results

4.1 Base Case Release Peak Groundwater Concentration Results

The conceptual model assumes that the any inventory released instantly comes to equilibrium with the fill material through the sorption process as controlled by the value of K_d. For the instant release model the maximum concentrations occur at time = 0 before any radioactive decay or transport in this model. For the diffusion controlled release, the time to the peak concentration depends on the diffusion coefficient and radionuclide half-life. Tables 9 – 14 provide the maximum concentration in each building. The tables also provide the amount of radioactivity (pCi) in solution, the amount sorbed to the solid material (pCi) and the concentration on the fill material (pCi/g) with a density of 1.5 g/cm^3.

4.1.1 Auxiliary Building

The base case for the Auxiliary Building assumes a diffusion controlled release. Uniform contamination was assumed over the first 0.5 inch of the concrete. The results of this simulation are provided in Table 10.

Table 10 Auxiliary Building Peak Groundwater Concentrations (pCi/L) per unit source of 1 pCi/m^2 and diffusion controlled release from 0.5 inch of contaminated concrete. The total inventory for each radionuclide is 6503 pCi.

Nuclide	Diffusion Coefficient (cm^2/s)	K_d (ml/g)	Time to Peak (years)	Peak Concentration pCi/L	Peak Radioactivity in Solution pCi	Peak Radioactivity Sorbed pCi	Peak Sorbed Concentration pCi/g
H-3	5.00E-07	0	0.1	9.10E-04	6503	0.0	0.00E+00
Co-60	4.10E-11	223	4	2.60E-08	0.2	249	5.80E-09
Ni-63	1.10E-09	62	37	1.90E-06	13.6	5051	1.18E-07
Sr-90	5.20E-10	2.3	21	1.96E-05	140.1	1933	4.51E-08
Cs-134	3.00E-09	45	1.5	6.89E-07	4.9	1329	3.10E-08
Cs-137	3.00E-09	45	14	2.47E-06	17.7	4766	1.11E-07
Eu-152	5.00E-11	95	10	1.07E-07	0.8	440	1.03E-08
Eu-154	5.00E-11	95	6	8.38E-08	0.6	341	7.96E-09

Examining Table 10 the impact of diffusion controlled release and sorption is clear. H-3 with no sorption and a high diffusion rate releases almost all the inventory within the first year to solution. Sr-90 with the low K_d value of 2.3 shows slightly more than 4% (140.1 pCi) of the total inventory (6503 pCi) is in solution. For all other nuclides the maximum activity in the water is less than 0.2% of the entire inventory. For Ni-63 the peak activity sorbed to the solid (5051 pCi) is slightly less than 80% of the total activity (6503 pCi). This reflects the time-dependent release from the concrete and the effects of radioactive decay. The time to peak represents the balance between the release rate, sorption, and radioactive decay. The value in the table is approximate as the concentration shows a broad peak over time. The radionuclides having a short half-life peak the earliest.

4.1.2 Containment Buildings

The two Containment Buildings are identical in geometry and therefore, the results for the unit inventory simulation apply to both buildings. In determining the potential dose, the results of this analysis will be scaled by the measured inventory in each building. The Containment Buildings will have all of the concrete inside the liner removed and residual contamination on the liner is assumed to be on the surface. For this reason, the instant release model was used and the results are presented in Table 11.

Table 11 Containment Building Peak Groundwater Concentrations (pCi/L) per unit source of 1 pCi/m^2. The total inventory for each radionuclide is 2759 pCi.

Nuclide	Half-life (years)	K_d (ml/g)	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	12.3	0	1.69E-03	2759	0	0
Co-60	5.27	223	1.26E-06	2.1	2756.9	2.81E-07
Ni-63	96	62	4.53E-06	7.4	2751.6	2.81E-07
Sr-90	29.1	2.3	1.14E-04	186.4	2572.6	2.62E-07
Cs-134	2.06	45	6.23E-06	10.2	2748.8	2.80E-07
Cs-137	30	45	6.23E-06	10.2	2748.8	2.80E-07
Eu-152	13.4	95	2.95E-06	4.8	2754.2	2.81E-07
Eu-154	8.2	95	2.95E-06	4.8	2754.2	2.81E-07

For the instant release model more than 99.5% of the material is sorbed on the backfill material for all modeled nuclides except H-3 and Sr-90. Sr-90 with the smallest non-zero K_d value of the group being modeled has slightly less than 7% of the activity in solution. Tritium (H-3), with a value of zero for K_d, has all the activity in solution.

4.1.3 Crib House/Forebay

The Crib House/Forebay is expected to contain little or no contamination based on characterization data and the contamination that may be present will be at the surface. For this reason, the instant release model was used. Table 12 provides the results of the analysis.

Table 12 Crib House Peak Groundwater Concentrations (pCi/L) per unit source of 1 pCi/m^2. The total inventory for each radionuclide is 6940 pCi.

Nuclide	Half-life (years)	K_d (ml/g)	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	12.3	0	9.08E-04	6936	0.0	1.99E-23
Co-60	5.27	223	6.78E-07	5.2	6930.8	1.51E-07
Ni-63	96	62	2.44E-06	18.6	6917.4	1.51E-07
Sr-90	29.1	2.3	6.14E-05	468.6	6467.4	1.41E-07
Cs-134	2.06	45	3.35E-06	25.6	6910.4	1.51E-07
Cs-137	30	45	3.35E-06	25.6	6910.4	1.51E-07
Eu-152	13.4	95	1.59E-06	12.1	6923.9	1.51E-07
Eu-154	8.2	95	1.59E-06	12.1	6923.9	1.51E-07

4.1.4 Fuel Building

The Spent Fuel Pool and Transfer Canals has not been fully characterized at this time. It is believed that there will be volumetric contamination in the concrete below the pool liners. For this reason diffusion controlled release is modeled assuming uniform contamination in the top 0.5 inch of concrete. The results are provided in Table 13.

Table 13 Fuel Building Peak Groundwater Concentrations (pCi/L) per unit source of 1 pCi/m^2. Release is diffusion controlled from 0.5 inch thick contaminated region. The total inventory for each radionuclide is 780 pCi.

Nuclide	Diffusion Coefficient (cm^2/s)	K_d (ml/g)	Time to Peak (years)	Peak Concentration pCi/L	Peak Radioactivity in Solution pCi	Peak Radioactivity Sorbed pCi	Peak Sorbed Concentration pCi/g
H-3	12.3	0	0.3	1.49E-02	774.8	0	0
Co-60	4.1E-11	223	3.9	4.25E-07	0.02	30	9.48E-08
Ni-63	1.1E-09	72	36	3.13E-05	1.6	605	1.94E-06
Sr-90	5.2E-10	2.3	21	3.21E-04	16.7	230	7.38E-07
Cs-134	3.0E-09	45	1.5	1.13E-05	0.6	159	5.09E-07
Cs-137	3.0E-09	45	13.3	4.07E-05	2.1	571	1.83E-06
Eu-152	5.0E-11	96	9.5	1.75E-06	0.09	52	1.68E-07
Eu-154	5.0E-11	95	6.2	1.37E-06	0.07	41	1.30E-07

The impact of diffusion controlled release on peak concentrations is slightly more pronounced than in the Auxiliary Building with a peak solution concentration for Sr-90 slightly in excess of 2 percent of the total inventory. The H-3 concentration predicted for the Fuel Building (0.015

pCi/L) is the highest predicted concentration for any of the buildings. This is due to the small amount of water available for mixing and the high diffusion release rate (over 99% of the inventory is released in the first year). The mixing height is only 0.91 m as compared to 11.28 m for the Auxiliary Building.

4.1.5 Turbine Building

The Turbine Building is expected to contain little or no contamination based on characterization data and contamination that was identified was predominantly at the surface. For this reason the instant release model is used. The results are provided in Table 14.

Table 14 Turbine Building Peak Groundwater Concentrations (pCi/L) per unit source of 1 pCi/m^2. The total inventory for each radionuclide is 14679 pCi.

Nuclide	Half-life (years)	K_d (ml/g)	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	12.3	0	2.25E-03	14679	0.0	0
Co-60	5.27	223	1.68E-06	11.0	14668.0	3.74E-07
Ni-63	96	62	6.02E-06	39.4	14639.6	3.73E-07
Sr-90	29.1	2.3	1.52E-04	991.8	13687.2	3.49E-07
Cs-134	2.06	45	8.29E-06	54.2	14624.8	3.73E-07
Cs-137	30	45	8.29E-06	54.2	14624.8	3.73E-07
Eu-152	13.4	95	3.93E-06	25.4	14653.6	3.74E-07
Eu-154	8.2	95	3.93E-06	25.7	14653.3	3.74E-07

Similar to the Crib House building, Sr-90 shows the highest solution concentration for sorbing nuclides and 6.7% of the Sr-90 is in the groundwater. Tritium (H-3) which does not sorb has the highest solution concentration.

4.1.6 Waste Water Treatment Facility

The WWTF is expected to contain little or no contamination based on characterization data and any contamination that may be present would be on the surface. For this reason the instant release model is used. The results are provided in Table 15.

The Waste Water Treatment Facility shows the highest peak concentrations per unit source term of all of the buildings with the exception of H-3. The cause for this is the very low mixing volume which is 143 m^3 and high surface area 1124 m^2. The surface area to volume ratio for this building is 7.8 m^{-1}, the largest of any building with an instant release source term. The inventory is directly proportional to surface area. Therefore, a high surface area to volume ratio will produce higher peak concentrations. The Fuel Building has a higher surface area to volume ratio but release was controlled by diffusion which limited the concentrations of everything except H-3 to lower levels than in the Waste Water Treatment Facility.

Table 15 Waste Water Treatment Facility Peak Groundwater Concentrations (pCi/L) per unit source of 1 pCi/m^2. The total inventory for each radionuclide is 1124 pCi.

Nuclide	Half-life (years)	K_d (ml/g)	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	12.3	0	3.13E-02	1124	0.0	0
Co-60	5.27	223	2.34E-05	0.8	1123.2	5.22E-06
Ni-63	96	62	8.40E-05	3.0	1121.0	5.21E-06
Sr-90	29.1	2.3	2.12E-03	75.9	1048.1	4.87E-06
Cs-134	2.06	45	1.16E-04	4.1	1119.9	5.20E-06
Cs-137	30	45	1.16E-04	4.1	1119.9	5.20E-06
Eu-152	13.4	95	5.43E-05	1.9	1122.1	5.21E-06
Eu-154	8.2	95	5.48E-05	2.0	1122.0	5.21E-06

4.2 Sensitivity Analysis

A sensitivity analysis was performed on the key parameters in the base case model for the Auxiliary Building. The key parameters in the base case model are the distribution coefficient K_d, porosity, and bulk density. Each of these was varied as defined in Table 8 for a total of six test cases. Appendix A contains the detailed results of these simulations and includes Tables identical in form to Tables 10 – 15 with the peak concentration, amount of activity in solution and sorbed to the solid, and the activity concentration on the solid (pCi/g). Additionally, there is a table providing the percent (%) change due to the variation in the parameter from the base case. The % Change was defined as:

% Change = 100*(Sensitivity Case – Base Case)/Base Case.

Thus, the % Change is positive if the sensitivity case value exceeds the base case value.

The major findings of the sensitivity analyses are:

- For all nuclides except H-3, most of the activity is sorbed onto the backfill material. Strontium with the lowest K_d still had more than 90% of the activity sorbed on the backfill.
- K_d: An increase in K_d caused a decrease in solution concentration and a slight increase in sorbed concentration. Solution concentration is approximately inversely proportional to K_d. The 25% change in K_d had a minimal impact on the amount sorbed or the backfill concentration (pCi/g). Strontium showed the largest percentage change in sorbed concentration of all the nuclides but it was less than 2.5%.
- Porosity: Changing porosity had a minor impact on the amount sorbed and solution concentration. The amount of radioactivity in solution was proportional to the porosity.

This reflects the availability of water with higher porosity having more water available for mixing and a higher total amount of activity in the water.

- Density: The solution concentration, sorbed concentration and amount in solution are inversely proportional to density. Increasing density causes a decrease in solution concentration. The change in density has a minor impact (< 2%) on the total amount of radioactivity that is sorbed.

4.2.1 Sensitivity to Release Rate

The base case model for the Auxiliary Building assumes diffusion controlled release from a 0.5 inch thick contaminated zone. For sensitivity analysis release was simulated from a 1 inch and 2 inch thick contaminated zone. In all cases, the total inventory for each nuclide remained constant at 6503 pCi. Changes in the depth of contamination can lead to changes in the total amount of mass released, the peak concentration, and the time to reach the peak concentration.

Table 16 examines the impact of contaminated zone thickness on the percentage of the total inventory released into solution over time and compares the change in total mass released to the base case 1/2 inch thick contaminated zone. H-3 has the highest diffusion coefficient and releases over 98% of the inventory in all three cases and therefore, the contaminated zone thickness only has a minor impact on the total mass released. The nuclides with a short half-life or a low diffusion coefficient in this simulation (Co-60, Sr-90, Cs-134, Eu-152, and Eu-154) show similar behavior and increasing the contaminated zone thickness by a factor of two leads to a factor of two decrease in the amount of mass released. Thus, in this region, the mass release is almost directly proportional to the contaminated zone thickness for these nuclides. The longer lived nuclides with the higher diffusion coefficients (Cs-137, and Ni-63) show similar trends but the response is much further from linear with distance than the shorter lived nuclides.

Table 16 Comparison of the percentage of the total inventory released based on the thickness of the contaminated zone. Thicknesses analyzed were 0.5 inch (base case), 1 inch and 2 inch.

Nuclide	Diffusion Coefficient	0.5 inch thick % Mass Released	1 inch thick % Mass Released	% change	2 inch thick % Mass Released	% change
H-3	5.5E-07	100.0	99.7	-0.3	98.2	-1.8
Co-60	4.1E-11	7.9	4.0	-49.8	2.0	-74.4
Ni-63	1.1E-09	92.2	74.8	-18.9	43.3	-53.0
Sr-90	5.2E-10	61.9	32.9	-46.8	16.7	-72.9
Cs-134	3.0E-09	42.4	21.4	-49.6	10.9	-74.4
Cs-137	3.0E-09	90.8	71.0	-21.8	40.9	-54.9
Eu-152	5.0E-11	13.8	6.9	-49.7	3.5	-74.4
Eu-154	5.0E-11	10.8	5.4	-49.7	2.8	-74.4

Table 17 provides the peak water concentration as a function of contaminated zone thickness and the percentage change from the base case (0.5 inch thick contaminated zone). The peak concentrations followed the same trends as the percentage of total mass released. H-3 showed only a minor decrease as most of the mass is released quickly for contaminated thickness of less than 2 inches. The other nuclides showed an almost linear response with contamination thickness as increasing the thickness by a factor of 2 leading to a decrease in peak concentration by a factor of 2.

Table 17 Comparison of the peak water concentration based on the thickness of the contaminated zone. Thicknesses analyzed were 1 inch (base case), ½ and 2 inch.

Nuclide	Diffusion Coefficient	0.5 inch thick Peak concentration (pCi/L)	1 inch thick Peak concentration (pCi/L)	% change	2 inch thick Peak concentration (pCi/L)	% change
H-3	5.5E-07	9.10E-04	9.00E-04	-1.1	8.57E-04	-5.8
Co-60	4.1E-11	2.60E-08	1.30E-08	-50.0	6.64E-09	-74.5
Ni-63	1.1E-09	1.90E-06	1.05E-06	-44.7	5.37E-07	-71.7
Sr-90	5.2E-10	1.96E-05	9.84E-06	-49.8	5.01E-06	-74.4
Cs-134	3.0E-09	6.89E-07	3.41E-07	-50.5	1.76E-07	-74.5
Cs-137	3.0E-09	2.47E-06	1.32E-06	-46.6	6.7E-07	-72.9
Eu-152	5.0E-11	1.07E-07	5.38E-08	-49.7	2.74E-08	-74.4
Eu-154	5.0E-11	8.38E-08	4.21E-08	-49.8	2.14E-08	-74.5

Table 18 provides the time to reach the peak concentration as a function of contaminated zone thickness and the percentage change from the base case (0.5 inch thick contaminated zone). The time to reach the peak concentration is a balance between the diffusion release rate and the radioactive decay rate. H-3 is very sensitive to contaminated zone thickness in the time to reach the peak concentration. This is because of the high release rate (high diffusion coefficient) of H-3. The short-lived species (Co-60, Sr-90, Cs-134, Eu-152, and Eu-154) show no sensitivity to the peak concentration time for any of the contaminated zone thicknesses tests. Ni-63 showed moderate sensitivity with the time to reach peak concentration varying between 37 and 72 years. Cs-137 showed an increase in the time to reach peak concentration of 57% in going to the 1 inch thick contaminated zone from the base case. However, it did not show a change in the time to reach the peak concentration above 1 inch contaminated zone thickness.

Table 18 Comparison of the time to reach the peak concentration in solution based on the thickness of the contaminated zone. Thicknesses analyzed were 0.5 inch (base case), 1 inch and 2 inch.

Nuclide	Diffusion Coefficient	0.5 inch thick Peak concentration time (yrs)	1inch thick Peak concentration time (yrs)	% change	2 inch thick Peak concentration time (yrs)	% change
H-3	5.5E-07	0.1	0.3	200.0	1.1	1000.0
Co-60	4.1E-11	4	4	0.0	4	0.0
Ni-63	1.1E-09	37	63	70.3	72	94.6
Sr-90	5.2E-10	21	21	0.0	21	0.0
Cs-134	3.0E-09	1.5	1.5	0.0	1.5	0.0
Cs-137	3.0E-09	14	22	57.1	22	57.1
Eu-152	5.0E-11	10	10	0.0	10	0.0
Eu-154	5.0E-11	6	6	0.0	6	0.0

4.2.2 Drill Spoils Sensitivity to K_d

As discussed in section 3.1.3 one exposure scenario includes using the drill spoils and mixes them with the native soil. To examine the change in drill spoils radionuclide concentration the K_d values in Table 9 were used. Table 19 provides the results for the new K_d values in the Auxiliary Building with all other parameters unchanged.

Table 19 Sensitivity of Drill Spoils to Distribution Coefficient (K_d)

Nuclide	Base Case K_d (ml/g)	Drill Spoils K_d (ml/g)	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	0	1	1.28E-04	914.7	5488	1.28E-07
Co-60	223	1161	4.99E-09	0.04	248	5.80E-09
Ni-63	62	62	1.90E-06	13.6	5051	1.18E-07
Sr-90	2.3	2.3	1.96E-05	140.1	1933	4.51E-08
Cs-134	45	615	5.05E-08	0.4	1332	3.11E-08
Cs-137	45	615	1.82E-07	1.3	4799	1.12E-07
Eu-152	96	7221	1.41E-09	0.0	437	1.03E-08
Eu-154	96	7221	1.10E-09	0.0	341	7.96E-09

Table 20 compares the sensitivity case to the base case for the peak concentration and peak sorbed concentration. The results for Ni-63 and Sr-90 are identical as the K_d values are the same in the two simulations. For the other nuclides increasing the K_d value led to lower predicted

Table 20 Comparison of Base Case and Drill Spoils case

Nuclide	Base Case K_d (ml/g)	Drill Spoils K_d (ml/g)	Base Case: Peak Concentration pCi/L	Drill Spoils: Peak Concentration pCi/L	Base Case: Sorbed Concentration pCi/g	Drill Spoils: Sorbed Concentration pCi/g
H-3	0	1	9.10E-04	1.30E-04	0.00E+00	1.30E-07
Co-60	223	1161	2.60E-08	4.99E-09	5.80E-09	5.80E-09
Ni-63	62	62	1.90E-06	1.90E-06	1.18E-07	1.18E-07
Sr-90	2.3	2.3	1.96E-05	1.96E-05	4.51E-08	4.51E-08
Cs-134	45	615	6.89E-07	5.05E-08	3.10E-08	3.11E-08
Cs-137	45	615	2.47E-06	1.82E-07	1.11E-07	1.11E-07
Eu-152	96	7221	1.07E-07	1.41E-09	1.03E-08	1.03E-08
Eu-154	96	7221	8.38E-08	1.11E-09	7.96E-09	8.02E-09

peak groundwater concentrations. This is most apparent for H-3 where the base case K_d value is 0 ml/g. The interesting point about this table is that even with a factor of ten increase in K_d (for example, Cs and Eu) the sorbed concentration increased only slightly (< 2%). This is a reflection of the fact that for K_d values greater than 10 more than 99% of the mass released is sorbed and therefore increasing K_d further has only a minor impact on the sorbed concentration.

4.3 Outside Receptor Well Concentration in Transport Model

The time evolution of concentration at a receptor well located two meters outside the Turbine Building was simulated using the backfill material K_d values in Table 4, the soil K_d and groundwater parameters in Table 7, and the geometry in Figure 3. The initial contamination level in the Auxiliary Building (1 pCi/m^2) was conservatively assumed to be 1000 times greater than in the Turbine Building (0.001 pCi/m^2). This assumption led to a total inventory of 6503 pCi in the Auxiliary Building and 14.7 pCi in the Turbine Building. Consistent with the Base Case, diffusion controlled release is assumed for the Auxiliary Building and Instant Release is assumed for the Turbine Building.

Table 18 provides the peak concentration in the Auxiliary Building, Turbine Building, Edge of the Turbine Building, and the Receptor Well. To quantitatively define the reduction in concentration from the Auxiliary Building to the Receptor Well the ratio of peak concentration at the well to the peak concentration in the Auxiliary Building is provided. The time to reach the peak at the Receptor Well is also provided. Recalling that the initial inventory in the Turbine Building was 450 times lower than in the Auxiliary Building, it is clear that Co-60 and Cs-134 did not move from the Auxiliary Building to the receptor well in any appreciable quantities. For the shorter lived nuclides (Co-60, Cs-134, Eu-152, and Eu-154) the combination of radioactive decay and sorption reduced the concentration by around a factor of ten in traveling two meters from the edge of the Turbine Building to the Receptor Well. H-3, the most mobile nuclide reached a maximum at the well after 1.5 years and showed a peak concentration ratio of 0.8 thus the transport through the Turbine Building did little to diminish the concentration of H-3. Sr-90, which exhibits some sorption but has a longer half-life than H-3, had a peak concentration ratio of 0.78 after 23 years, slightly less than that for the more mobile H-3. All other nuclides had a peak concentration ratio of less than 2%.

Table 21 Comparison of Peak Concentrations in the modeled region.

	Aux Bldg. (pCi/L)	Turbine Bldg. (pCi/L)	Edge of Turbine Bldg. (pCi/L)	Receptor Well (pCi/L)	Ratio Well to Auxiliary Building	Time to peak (years)
H-3	1.48E-03	1.48E-03	1.21E-03	1.19E-03	0.80	1.5
Co-60	2.5E-08	2.1E-09	2.1E-09	2.7E-11	0.001	15
Ni-63	2.02E-06	6.38E-07	5.23E-08	3.5E-08	0.017	>300
Sr-90	1.10E-05	1.18E-05	8.81E-06	8.60E-06	0.78	23
Cs-134	6.74E-07	1.02E-08	1.00E-08	3.93E-10	0.001	4.5
Cs-137	2.56E-06	1.80E-07	1.01E-08	4.61E-09	0.002	21
Eu-152	1.04E-07	4.8E-09	4.76E-09	6.94E-10	0.007	18
Eu-154	8.20E-08	4.85E-09	4.81E-09	4.30E-10	0.005	13

4.4 Discussion

The simulation of a well located in the middle of the contaminated zone is intended to provide a reasonable upper bound on peak contaminant concentrations. The following qualitative arguments support this assertion.

- The Reasonably Foreseeable Scenario, defined in NUREG 1757 as a land use scenario that is likely within the next 100 years, would not include an onsite water well which is prohibited by local municipal code.
- If the local laws were ignored, it is unlikely that anyone would drill through the backfill (concrete construction debris) to install a well.
- If a well was installed, the water will be non-potable due to the high pH (>10) that will occur from leaching of the concrete construction debris.

5 Validation

The instant release model reduces to a simple mixing bath model where the entire inventory is at equilibrium with the backfill material. The concentration for this model can be calculated as:

$$C = M/[V * (\theta + \rho Kd)]$$

Where C= concentration in solution (pCi/L)
M = inventory (pL)
V = volume (L) (2.65E7 L in Turbine Building).
Θ = effective porosity (0.25)
ρ = bulk density (g/cm^3) (1.5 g/cm^3)
K_d = distribution coefficient (cm^3/g)

A comparison was made between the DUST-MS output and the analytical solution in the equation above for the Turbine Building as an example of an instant release basement. The results showed an excellent match between the two predictions, Table 19.

Table 22 Comparison between Analytical Solution and DUST-MS results for the Turbine Building.

Nuclide	K_d	C (pCi/L)	DUST-MS C(pCi/L)
H-3	0	2.21E-03	2.21E-03
Co-60	223	1.65E-06	1.65E-06
Ni-63	62	5.94E-06	5.94E-06
Sr-90	2.3	1.50E-04	1.50E-04
Cs-134	45	8.17E-06	8.17E-06
Cs-137	45	8.17E-06	8.17E-06
Eu-152	95	3.88E-06	3.88E-06
Eu-154	95	3.88E-06	3.88E-06

Similar calculations were performed for all buildings and showed a good match between the two models.

6 Conclusions

A model for predicting peak groundwater concentrations at the ZSRP Site after decommissioning has been developed. The model uses the DUST-MS simulation model which calculates the release and transport of radioactive contamination in a groundwater system. The analysis is based on a unit source term of 1 pCi/m^2 on the entire wall and floor surface area of each of the seven buildings that will have a residual below ground, backfilled structure. Conservative assumptions based on existing data were used in the screening model for selecting parameters that impact groundwater concentration (K_d, porosity, bulk density, no flow). For example, the K_d value selected for the fill material was the lowest measured value using site-specific groundwater for any soil or fill material. The results of the model can be combined with measured data after characterization is completed to determine peak groundwater dose for all the nuclides.

A sensitivity analysis was performed for the key variables (K_d, effective porosity, bulk density) for the Auxiliary Building base case. The results of the analysis showed that the peak water concentration was inversely proportional to bulk density and K_d. The solution concentration was weakly sensitive to changes in porosity. In all cases, more than 90% of the nuclide inventory is sorbed onto the fill material.

A sensitivity analysis was performed on the release model through comparison of the diffusion change in total mass released, peak concentration, and time to reach the peak concentration for the base case, one inch contaminated zone, to results from simulations with one-half and two inch contaminated zone. For H-3, which has the highest diffusion coefficient, the mass released and peak concentration were not sensitive to the length of the contaminated zone. Over 98% of the mass was released in all three simulations. The other nuclides showed close to an inverse linear dependence on contaminated zone length with the mass release and peak concentration decreasing by close to a factor of two with an increase in length of a factor of two. The time to reach the peak concentration was independent of the length of the contaminated zones for short-lived nuclides (other than H-3) indicating that a balance between release rate and radioactive decay was achieved. For H-3 the high release rate caused the peak concentration to be reached in 0.1 years for the shortest contaminated length (1/2 inch) and 1.4 years for the two inch contaminated length simulation.

Removing the assumption of a well placed in the middle of the fill material and placing the Receptor Well two meters outside the Turbine Building, which is the closest soil (e.g. non-building) location to the Auxiliary Building where the highest residual contamination will remain, led to a three to four order of magnitude reduction in peak concentration for short-lived nuclides (Co-60; Cs-134, Eu-152, and Eu-154), a two order of magnitude reduction for Cs-137, and a factor of fifty reduction for Ni-63. H-3 showed a 20% reduction in peak dose due radioactive decay and transport to the well. Sr-90, which has high mobility and longer half-life than H-3, showed a 22% reduction in peak concentration at the Receptor Well as compared to in the Auxiliary Building.

7 References

Atkinson, A., Nelson, K., and Valentine, T.M., "Leach test characterization of cement-based nuclear waste forms," Nuclear and Chemical Waste Management, Vol. 6 (1986), 241 – 253.

Conestoga-Rovers & Associates, 2014, "Evaluation of Hydrological Parameters in Support of Dose Modeling for the Zion Restoration Project," Conestoga-Rovers & Associates, Chicago, IL, January 14, 2014, Reference No.054638, Revision 4, Report No. 3.

Farr, H.C., "Re: New Volumes" e-mail 9/24/14 to T. Sullivan

Jakob, A., F.-A. Sarott and P. Spieler, "Diffusion and sorption on hardened cement pastes - experiments and modeling results", Paul Scherer Institute. PSI-Bericht Nr. 99-05 ISSN 1019-0643, August 1999.

Milian, L., T. Sullivan (2014). *Sorption (K_d) measurements on Cinder Block and Grout in Support of Dose Assessments for Zion Nuclear Station Decommissioning,* Brookhaven National Laboratory Report to Zion*Solutions*, April 2014.

Muurinnen, A, J. Rantanen, R. Ovaskainen and O.J. Heinonen, "Diffusion Measurements in Concrete and Compacted Bentonite," Proceedings of the Materials Research Meeting, 1982.

Serne, R. J., R.O. Lokken, and L.J. Criscenti. "Characterization of Grouted LLW to Support Performance Assessment." Waste Management 12: 271-287, 1992.

Serne, J., "Selected Diffusion Coefficients for Radionuclides in Cement", personal communication.

Sullivan, T.M., "**DUST** - Disposal Unit Source Term: Data Input Guide." NUREG/CR-6041, BNL-NUREG-52375, 1993.

Sullivan, T.M., C.R. Kempf, C.J. Suen, and S.F. Mughabghab, "Low-Level Radioactive Waste Source Term Model Development and Testing," NUREG/CR-5204, BNL-NUREG-52 160, Brookhaven National Laboratory, 1988.

Sullivan, T.M., "**DUSTMS_D** - Disposal Unit Source Term – Multiple Species – Distributed Failure Data Input Guide. Rev 1.," BNL-75554-2006, Brookhaven National Laboratory, Upton, NY, 11973, January, 2006.

Sullivan, T.M., "Recommended Values for the Distribution Coefficient (K_d) to be Used in Dose Assessments for Decommissioning the Zion Nuclear Power Plant, Revision 1", BNL-Letter Report, September 24, 2014.

Szanto, Zs, Svingor, M. Molnir, L. Palcsu, I. Futo, Z. Szucs. "Diffusion of ^{3}H, ^{99}Tc, ^{125}I, ^{36}Cl, and ^{85}Sr in granite, concrete and bentonite," Journal of Radioanalytical and

Nuclear Chemistry, Vol. 252, No. 1 (2002) 133-138.

U.S. Nuclear Regulatory Commission, (NRC, 2000). *Development of Probabilistic RESRAD 6.0 and RESRADBUILD 3.0 Computer Codes,* NUREG/CR-6697, U.S. Nuclear Regulatory Commission, December 2000.

Yim, S.P, T.M. Sullivan, and L. Milian, ”Sorption (K_d) measurements in Support of Dose Assessments for Zion Nuclear Station Decommissioning,” Brookhaven National Laboratory Report to Zion*Solutions*, December 12, 2012.

Appendix A: Sensitivity Analysis Results

A.1: Base Case

The base case for the Auxiliary Building is diffusion-controlled release from the concrete floors. The initial inventory for each nuclide was 6503 pCi. There is a major difference between non-sorbing nuclides (H-3) and sorbing nuclides. The non-sorbing nuclide showed approximately 96% of the inventory in solution. With the other 4% decayed prior to release from the floors and wall. The sorbing nuclides had less than 1.2% in solution with most of the released mass sorbed. Examining the Peak Radioactivity Sorbed shows that less than ½ of the total inventory was on the backfill at any time.

Nuclide	Diffusion Coefficient (cm^2/s)	K_d (ml/g)	Time to Peak (years)	Peak Concentration pCi/L	Peak Radioactivity in Solution pCi	Peak Radioactivity Sorbed pCi	Peak Sorbed Concentration pCi/g
H-3	5.00E-07	0	1.5	8.70E-04	6267	0.0	0.00E+00
Co-60	4.10E-11	223	3.8	1.30E-08	0.09	125.3	2.90E-09
Ni-63	1.10E-09	62	72	1.05E-06	7.56	2813.7	6.51E-08
Sr-90	5.20E-10	2.3	22	9.84E-06	70.88	978.2	2.26E-08
Cs-134	3.00E-09	45	1.5	3.41E-07	2.46	663.2	1.53E-08
Cs-137	3.00E-09	45	22	1.32E-06	9.51	2567.3	5.94E-08
Eu-152	5.00E-11	96	9.5	5.38E-08	0.39	223.2	5.16E-09
Eu-154	5.00E-11	95	6	4.21E-08	0.30	172.9	4.00E-09

A.2: High K_d

K_d values are in the table below. They were increased by 25% from the base case value. Increasing the K_d value increases the amount of sorption and reduces the solution concentration. For non-sorbing nuclides there is no impact for changes in K_d.

A negative number means that the base case value is greater than the sensitivity case value.

Nuclide	Diffusion Coefficient (cm^2/s)	K_d (ml/g)	Peak Concentration pCi/L	Peak Radioactivity in Solution pCi	Peak Radioactivity Sorbed pCi	Peak Sorbed Concentration pCi/g
H-3	5.50E-07	0	8.70E-04	6267		
Co-60	4.10E-11	278.8	1.04E-08	0.075	125.3	2.90E-09
Ni-63	1.10E-09	77.5	8.38E-07	6.037	2807.0	6.49E-08
Sr-90	5.20E-10	2.88	7.97E-06	57.412	992.1	2.30E-08
Cs-134	3.00E-09	56.3	2.74E-07	1.974	666.7	1.54E-08
Cs-137	3.00E-09	56.3	1.06E-06	7.636	2579.3	5.97E-08
Eu-152	5.00E-11	120	4.30E-08	0.310	223.0	5.16E-09
Eu-154	5.00E-11	118.8	3.37E-08	0.243	173.0	4.00E-09

% Change from the Base case = 100*(Sensitivity Case- Base Case)/Base Case

Nuclide	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	0.0	0.0		
Co-60	-20.0	-20.0	0.0	0.0
Ni-63	-20.2	-20.2	-0.2	-0.2
Sr-90	-19.0	-19.0	1.4	1.4
Cs-134	-19.6	-19.6	0.5	0.5
Cs-137	-19.7	-19.7	0.5	0.5
Eu-152	-20.1	-20.1	-0.1	-0.1
Eu-154	-20.0	-20.0	0.1	0.1

A.3: Low K_d

K_d values are shown in the table below and were reduced by 25% from the base case values. Reducing K_d increases the amount in solution for sorbing nuclides but does not impact the total amount sorbed. For non-sorbing nuclides the change in K_d has no impact.

Nuclide	Diffusion Coefficient (cm^2/s)	K_d (ml/g)	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	5.50E-07	0	8.70E-04	6267		
Co-60	4.10E-11	167	1.74E-08	0.13	126	2.91E-09
Ni-63	1.10E-09	47	1.39E-06	10.01	2794	6.46E-08
Sr-90	5.20E-10	1.73	1.28E-05	92.20	954	2.21E-08
Cs-134	3.00E-09	34	4.51E-07	3.25	658	1.52E-08
Cs-137	3.00E-09	34	1.74E-06	12.53	2538	5.87E-08
Eu-152	5.00E-11	72	7.17E-08	0.52	223	5.16E-09
Eu-154	5.00E-11	72	5.61E-08	0.40	175	4.04E-09

% Change from the Base case = 100*(Sensitivity Case- Base Case)/Base Case

Nuclide	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	0.0	0.0		
Co-60	33.8	33.8	0.4	0.4
Ni-63	32.4	32.4	-0.7	-0.7
Sr-90	30.1	30.1	-2.4	-2.4
Cs-134	32.3	32.3	-0.8	-0.8
Cs-137	31.8	31.8	-1.1	-1.1
Eu-152	33.3	33.3	0.0	0.0
Eu-154	33.3	33.3	1.0	1.0

A.4: High Porosity

The porosity was increased to 0.31 from the base case value of 0.25. Increasing porosity did not impact the solution concentration but did increase the amount of radioactivity in solution due to the greater amount of water for sorbing nuclides. For non-sorbing nuclides increasing porosity decreased the solution concentration but did not impact the total amount in solution.

Nuclide	Diffusion Coefficient (cm^2/s)	K_d (ml/g)	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	5.50E-07	0	8.70E-04	6267		
Co-60	4.10E-11	223	1.30E-08	0.12	125	2.90E-09
Ni-63	1.10E-09	62	1.05E-06	9.38	2814	6.51E-08
Sr-90	5.20E-10	2.3	9.68E-06	86.47	962	2.23E-08
Cs-134	3.00E-09	45	3.41E-07	3.05	663	1.53E-08
Cs-137	3.00E-09	45	1.32E-06	11.79	2567	5.94E-08
Eu-152	5.00E-11	96	5.38E-08	0.48	223	5.16E-09
Eu-154	5.00E-11	95	4.21E-08	0.38	173	4.00E-09

% Change from the Base case =
100*(Sensitivity Case- Base Case)/Base Case

Nuclide	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	-19.3	0.1		
Co-60	0.0	24.0	0.0	0.0
Ni-63	0.0	24.0	0.0	0.0
Sr-90	-1.6	22.0	-1.6	-1.6
Cs-134	0.0	24.0	0.0	0.0
Cs-137	0.0	24.0	0.0	0.0
Eu-152	0.0	24.0	0.0	0.0
Eu-154	0.0	24.0	0.0	0.0

A.5: Low Porosity

The porosity was decreased to 0.19 from the base case value of 0.25. For sorbing nuclides decreasing the porosity did not impact the solution concentration but it did reduce the total amount of radioactivity in the water. For non-sorbing nuclides decreasing the porosity increased the solution concentration but did not impact the amount in solution.

Nuclide	Diffusion Coefficient (cm^2/s)	K_d (ml/g)	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	5.50E-07	0	1.14E-03	6241		
Co-60	4.10E-11	223	1.30E-08	0.07	125	2.90E-09
Ni-63	1.10E-09	62	1.05E-06	5.7	2814	6.51E-08
Sr-90	5.20E-10	2.3	9.68E-06	53.0	962	2.23E-08
Cs-134	3.00E-09	45	3.41E-07	1.9	663	1.53E-08
Cs-137	3.00E-09	45	1.32E-06	7.2	2567	5.94E-08
Eu-152	5.00E-11	95	5.38E-08	0.3	221	5.11E-09
Eu-154	5.00E-11	96	4.21E-08	0.23	175	4.04E-09

% Change from the Base case =
100*(Sensitivity Case- Base Case)/Base Case

Nuclide	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	31.0	-0.4		
Co-60	0.0	-24.0	0.0	0.0
Ni-63	0.0	-24.0	0.0	0.0
Sr-90	-1.6	-25.2	-1.6	-1.6
Cs-134	0.0	-24.0	0.0	0.0
Cs-137	0.0	-24.0	0.0	0.0
Eu-152	0.0	-24.0	-1.0	-1.0
Eu-154	0.0	-24.0	1.1	1.1

A.6: High Bacfkill Density

The backfill density was increased to 1.8 g/cm^3 from the base case value of 1.5 g/cm^3. Increasing the density caused both the solution concentration and sorbed concentration to decrease for sorbing nuclides. This is because the extra mass provided more sorption to reduce solution concentrations and more mass to sorb onto and therefore lower sorbed concentrations. The density did not impact non-sorbing nuclides.

Nuclide	Diffusion Coefficient (cm^2/s)	K_d (ml/g)	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	5.50E-07	0	8.70E-04	6267		
Co-60	4.10E-11	223	1.09E-08	0.08	126.1	2.43E-09
Ni-63	1.10E-09	62	8.79E-07	6.33	2827	5.45E-08
Sr-90	5.20E-10	2.3	8.29E-06	59.72	989	1.91E-08
Cs-134	3.00E-09	45	2.84E-07	2.05	663	1.28E-08
Cs-137	3.00E-09	45	1.10E-06	7.92	2567	4.95E-08
Eu-152	5.00E-11	96	4.44E-08	0.32	221	4.26E-09
Eu-154	5.00E-11	95	3.51E-08	0.25	173	3.33E-09

% Change from the Base case = 100*(Sensitivity Case- Base Case)/Base Case

Nuclide	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	0.0	0.0		
Co-60	-16.2	-16.2	0.6	-16.2
Ni-63	-16.3	-16.3	0.5	-16.3
Sr-90	-15.8	-15.8	1.1	-15.8
Cs-134	-16.7	-16.7	-0.1	-16.7
Cs-137	-16.7	-16.7	0.0	-16.7
Eu-152	-17.5	-17.5	-1.0	-17.5
Eu-154	-16.6	-16.6	0.0	-16.6

A.7: Low Density

The density was decreased to 1.1 g/cm^3 from the base case value of 1.5 g/cm^3. Reducing the density caused an increase in both the solution concentration and the sorbed concentration. The increase was inversely proportional to the density. The change in density did not impact non-sorbing nuclides.

Nuclide	Diffusion Coefficient (cm^2/s)	K_d (ml/g)	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	5.50E-07	0	8.70E-04	6267	0.0	0.00E+00
Co-60	4.10E-11	223	1.78E-08	0.13	125.8	3.97E-09
Ni-63	1.10E-09	62	1.44E-06	10.37	2829.8	8.93E-08
Sr-90	5.20E-10	2.3	1.31E-05	94.37	955.0	3.01E-08
Cs-134	3.00E-09	45	4.71E-07	3.39	671.8	2.12E-08
Cs-137	3.00E-09	45	1.79E-06	12.89	2553.1	8.06E-08
Eu-152	5.00E-11	96	7.25E-08	0.52	220.6	6.96E-09
Eu-154	5.00E-11	95	5.74E-08	0.41	172.8	5.45E-09

% Change from the Base case =
100*(Sensitivity Case- Base Case)/Base Case

Nuclide	Peak Concentration pCi/L	Radioactivity in Solution pCi	Radioactivity Sorbed pCi	Sorbed Concentration pCi/g
H-3	0.0	0.0		
Co-60	36.9	36.9	0.4	36.9
Ni-63	37.1	37.1	0.6	37.1
Sr-90	33.1	33.1	-2.4	33.1
Cs-134	38.1	38.1	1.3	38.1
Cs-137	35.6	35.6	-0.6	35.6
Eu-152	34.8	34.8	-1.2	34.8
Eu-154	36.3	36.3	0.0	36.3