Zion*Solutions*, LLC.
Technical Support Document

TSD 14-009

Brookhaven National Laboratory: Evaluation of Maximum Radionuclide Groundwater Concentrations for Basement Fill Model

Revision 3

Originator: Terry Sullivan Date: 2/24/2017
Brookhaven National Laboratory

Reviewer: _____ Date: 2/24/2017
David Fauver

Approval: _____ Date: 2/24/2017
Robert F. Yetter

BNL-107250-2017-IR-R3

Evaluation of Maximum Radionuclide Groundwater Concentrations for Basement Fill Model

Zion Station Restoration Project

Terry Sullivan
Brookhaven National Laboratory

Revision 3
February 21, 2017

Revision Log

Section	Page #	Rev. #	Date	Reason(s) for Revision
Title page		2	5/20/16	Revised Revision number
Title page		2	5/20/16	Revised effective date
Title page	17, Table 10	2	5/20/16	Changed the total pCi released for H-3 in Table 10 from 6503 pCi to 6467 pCi. The value 6503 pCi is the total inventory in the analysis and this can not be the total released. This simulation has diffusion controlled release from the concrete and some H-3 decays in the concrete prior to release.
		2	5/20/16	Added Attachment B: Transport to Turbine Building
Title page		3	2/21/17	Revised Revision number
Title page		3	2/21/17	Revised effective date
Introduction		3	2/21/17	Added text to explain the attachments.
		3	2/21/17	Added Attachment C: Demonstrtion of Similar Concentrations when buildings are contaminated to DCGL values.
		3	2/21/17	Added Attachment D Transport to Crib House
		3	2/21/17	Added Attachment E: Steam Tunnel Release
		3	2/21/17	Added Attachment F: Sump Release
			2/21/17	Added Attachment G: Compariosn of Instant and Diffusion release models in the Auxiliary Building.
		3	2/21/17	Added Attachment H. Discharge Tunnel Mixing Bath.
		3	2/21/17	Added Attachment I. Comparison of peak concentrations in transport simulations to mixing bath (no flow) simulations.

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.

After revision 1 was completed the Nuclear Regulatory Commission (NRC) requested additional information to complete their review. In particular, they asked to address transport from one building to another as they are connected by underground piping that will be left in place. Attachment B was produced to examine flow from the Containment Building to the Auxiliary Building through the Turbine Building to a well 2 meters outside of the Turbine Building.

Subsequent questions by the NRC were concerned with hot spots in penetrations that connected buildings and how they might impact peak concentrations, transport to other buildings such as the Crib House, and migration through pipes into sumps in the buildings.

A revised approach to setting inventory limits was developed to address this. All of the inventory in the penetration piping was included in the downstream building to which it was connected. The penetration piping was limited to the Derived Concentration Guidance Level (DCGL) of the downstream building with the inventory of the penetrations included. An important aspect the approach is that hot spots are no longer an issue. If properly performed, the DCGL value in each

building will lead to the same groundwater concentration in each Building. This is demonstrated in Attachment C for the Containment and Auxiliary Buildings.

Attachment D examines transport to the Crib House/Forebay buildings and is an extension of Attachment B. In this case, it is not possible to place a well downstream of the Crib House/Forebay. The calculation shows that the peak concentration occurs in the Auxiliary and Containment Buildings which were assumed to have higher inventory loadings based on characterization data. It is important to note that downstream concentrations never exceed the concentrations in the higher contaminated regions.

Current plans are to provide an engineered opening just above the water table in the steam tunnels that connect the two Containment Buildings to the Turbine Building. Attachment E examines potential releases to a well 2 meters outside of one of these openings.

The NRC raised an issue of transport of contaminants in piping to sumps in Buildings. These sumps will be backfilled with several feet of grout. Attachment F examines diffusion release through the grout that will fill the ends of the pipes that feed into the sumps.

The modelling approach to include penetrations in wall that connect the different buildings in the subsurface is to set the DCGL value in the penetration to that in the connecting building. The penetrations are expected to have surface contamination which is modeled to release instantly. In contrast, the Auxiliary Building has concrete contamination and the release is modeled as diffusion through the concrete. The diffusion process decreases the total inventory released and spreads the release out over time. To include the penetration piping inventory into the Auxiliary model requires an adjustment to account for the difference between instant and diffusion release. This adjustment multiplies the inventory in the penetration piping such that the peak concentration from the diffusion release model equals the concentration that would arise from diffusion from the Auxiliary Building component and instant release from the penetration piping component of the inventory. Appendix G provides a comparison of diffusion and instant release peak concentrations using the geometry of the Auxiliary Building.

The original mixing bath model analysis in this report did not address the Discharge Tunnels. The Discharge Tunnels run from beneath the Turbine Building and ultimately connect to the lake. The modeling approach to determine DCGL levels in the Turbine Building added the source term from the Discharge Tunnels into the Turbine and uses the DCGL levels in the Turbine Building for the Discharge Tunnel and the Turbine Building. To examine the differences in predicted concentration for a unitized source term of 1 pCi/m^2 a mixing bath model was developed for the Discharge Tunnels and is compared to the values resulting from the analysis of the Turbine Building in Appendix H.

Attachment I provides a comparison of peak concentrations in each building for each nuclide in the two transport scenarios (Attachment D and E) to the peak concentration in the building in the zero flow (mixing bath) model that is the baseline in this report. In all cases, the mixing bath model had concentrations that were equal to or greater than the value found in the same building for the transport simulation. This indicates that the mixing bath model provides an upper bound on peak concentration.

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%.

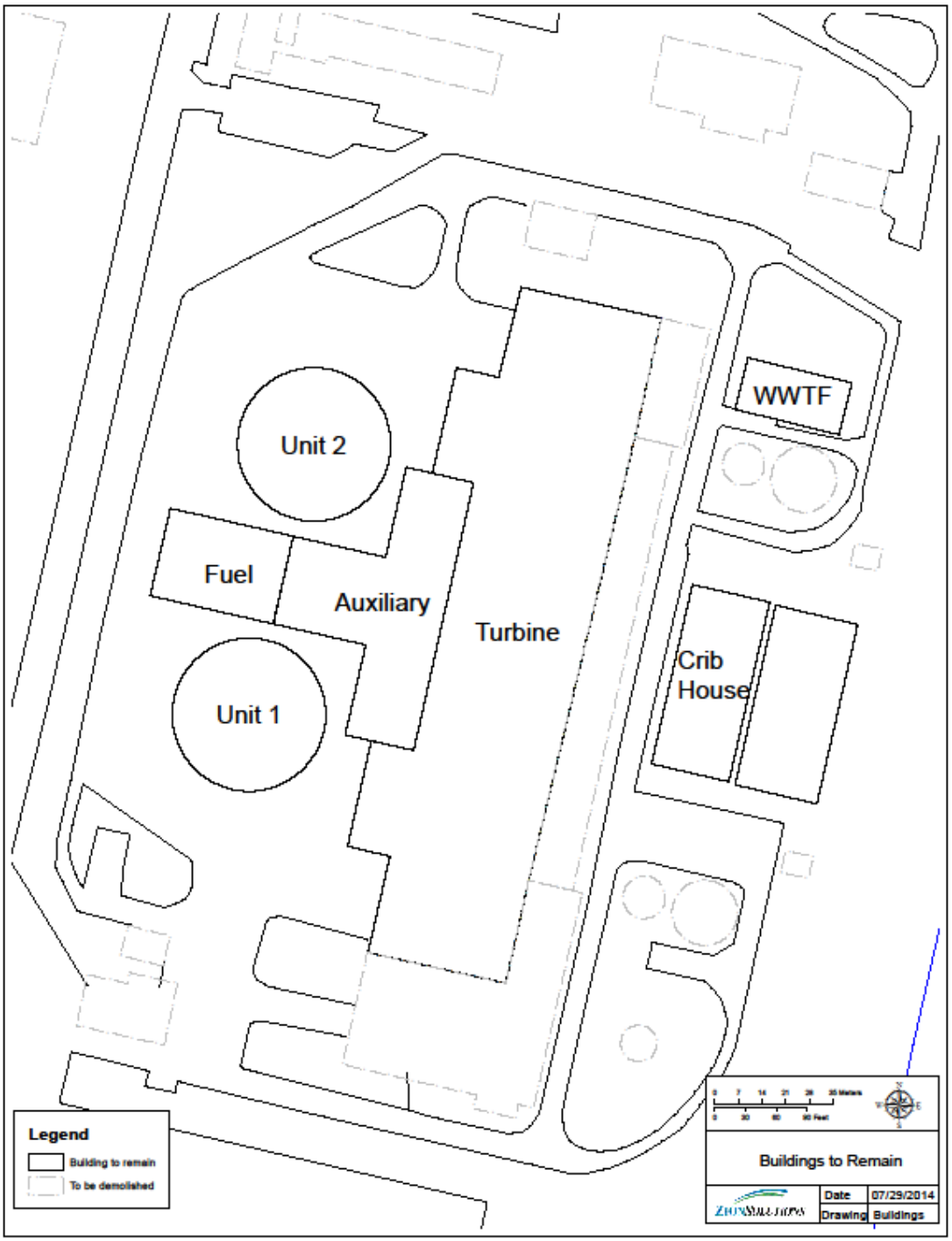

Figure 1 Zion Site building layout.

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.

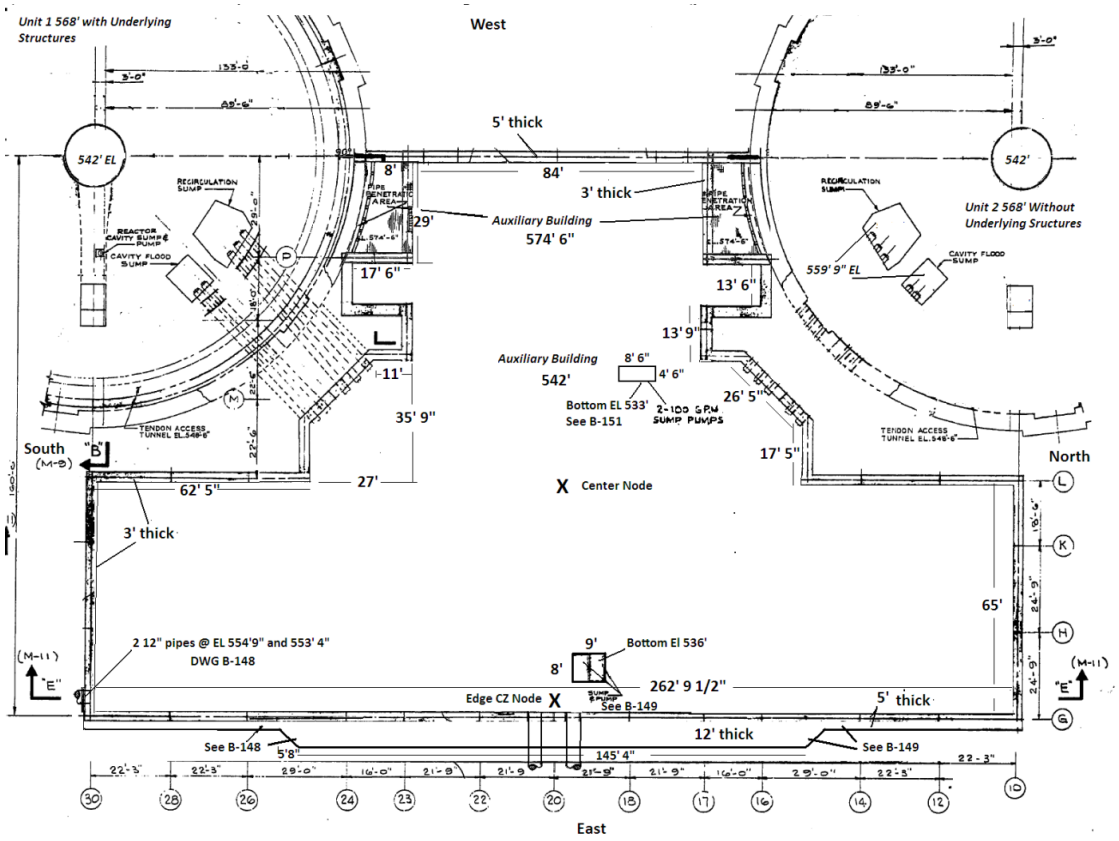

Figure 2. Geometry of the Auxiliary Building.

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.

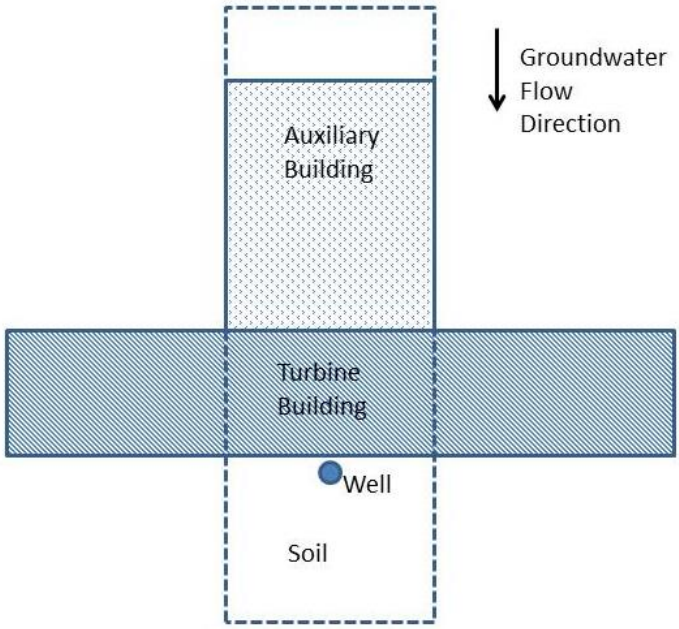

Figure 3 Schematic Representation of Flow the geometry used to assess flow to a well outside the Turbine Building.

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	95 (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 75^{th} 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	6467	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, Table 22. 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.

Attachment 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

40

Attachment B: Transport between buildings to a receptor well

Calculations to address.flow between subsurface structures

Terry Sullivan
January 24, 2016

Request for Additional Information 21

> **Basis**: In TSD 14-009, a sensitivity analysis was performed for a well receptor located outside of the basements. In this analysis, the peak concentrations in a receptor well 2 m outside the turbine building were calculated. This analysis considered source terms from the auxiliary and turbine buildings with contamination levels of 1 pCi/m^2 and 0.001 pCi/m^2 respectively. TSD 14-009 states that the well location was selected to be the closest place to put a well outside of the auxiliary building, which is the building that will have the highest levels of residual contamination. It is not clear if sources from other basements, such as the containment building basements, could affect the groundwater at this location. Also, it is not clear how the assumed concentrations will compare to the end state concentrations in those basements.

Summary Response:

Additional simulations were performed to examine contributions from the Containment to the Auxiliary building through the Turbine building to a receptor well. Three additional simulations were performed to determine if the Containment Building impacts peak concentration in the receptor well. The first simulation assumed that the Auxiliary Building contamination led to an initial concentration of 1 pCi/L in the water phase and the other two buildings were relatively clean with contamination leading to 0.001 pCi/L. Characterization data suggests that the Auxiliary Building will have the highest residual contamination. The second simulation assumed that the Containment contamination led to a peak concentration of 1 pCi/L in this region as well as in the Auxiliary Building. The third simulation assumed that all three Buildings started with an initial concentration of 1 pCi/L in the water. The results showed that due to the transport time from the reactor to receptor well contamination in the Containment Building had no impact on peak concentration in the Auxiliary Building for less mobile nuclides (Co-60, Ni-63, Cs-134, Cs-137, Eu-152, and Eu-154). The simulation that maintained a uniform concentration of 1 pCi/L in the Turbine Building had the highest concentrations at the receptor well. This is expected due to the proximity of the Turbine Building to the receptor point. The key point of the analysis is that the Turbine Building concentrations drive the peak concentration in the receptor well for short-lived less mobile nuclides (Co-60, Cs-134, Eu-152, and Eu-154) even if the inventory is three orders of magnitude lower than in the Auxiliary Building. For two nuclides (Ni-63 and Cs-137) their longer half-life allows nuclides released from the Auxiliary Building to reach the receptor well, but at levels that are 1 to 10% of their value in the Auxiliary Building. For Ni-63 and Cs-137 nuclides released from the Containment Building do not impact peak receptor well concentrations in these simulations. For mobile nuclides, e.g. H-3 and Sr-90, the impact of the Auxiliary Building can be seen in the peak concentration. Releases from the Containment Buildings at the same level as the Auxiliary Building led to a slight increase (<20%) in the peak receptor well concentration for H-3 and Sr-90.

B 1. Introduction

Calculations were performed in TSD-14-009 to demonstrate the conservatism of the assumptions regarding placing a well in the middle of the contaminated zone within each building. With current plans to backfill with cement construction demolition debris (CCDD) it is likely that the pH will make any water in these buildings non-potable. Therefore, an analysis was performed at the nearest downstream location a drinking water well could be placed. At the Zion plant this is 2 m downstream from the Turbine Building. Other buildings are connected to the Turbine Building with underground piping that could lead to contamination from other buildings entering the Turbine Building and ultimately reaching the well. These buildings include the Auxiliary Building which is directly connected to the Turbine Building through piping and both reactor containment buildings which have pipes that connect to the Auxiliary Building.

Characterization data suggest that the majority of the residual contamination will be in the Auxiliary Building. The Turbine Building should have minor, if any, contamination. The Containment Buildings will have all activated components removed and residual contamination is also expected to be much lower than in the Auxiliary Building. For these reasons the initial simulation included only the Auxiliary and Turbine Buildings and the contamination level in the Auxiliary Building l, a factor of 1000 greater than in the Turbine Building.

For completeness, the potential for contamination arising from the Containment Buildings and being transported through the Auxiliary Building and through the Turbine Building to the receptor well is addressed in this report.

The conceptual model and parameter assumptions, taken from TSD 14-009 are repeated here for completeness. The only change from the calculation in TSD 14-009 is the addition of the containment buildings and changing the source term from 1 pCi/m^2 of surface area in the building to an amount to reach 1 pCi/L in the water phase for each nuclide. The changes resulting from this will be discussed as part of the model development.

B 2. Conceptual Models of Release

B 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. Currently clean concrete construction debris (CCDD) is the likely fill material.

There are seven buildings (Figure 1) that will have residual structures beginning three feet below grade. For the purposes of this analysis consideration is given to the Containment Buildings, Auxiliary Building and the Turbine Building. 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.

B 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). 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. 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%.

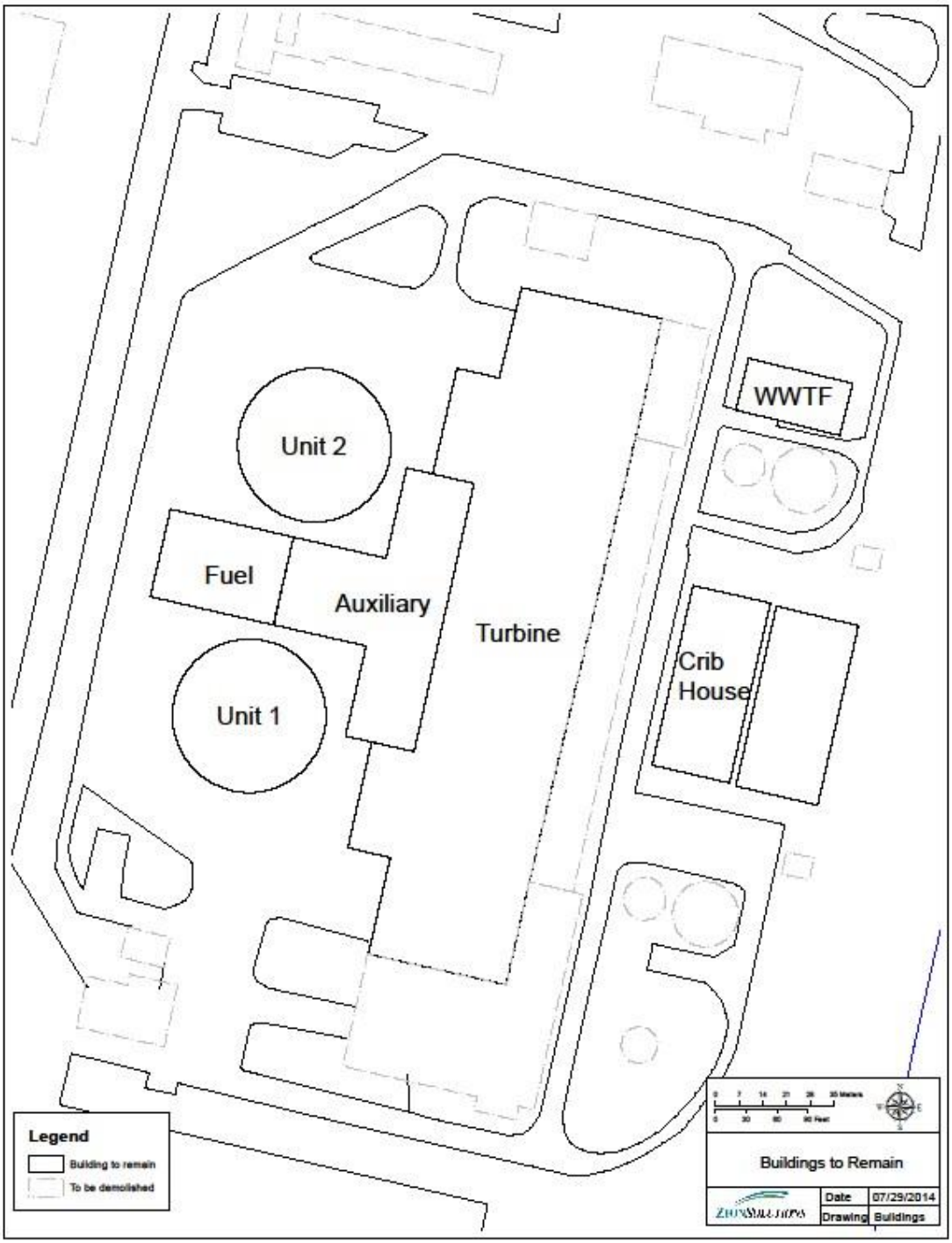

Figure 1 Zion Site building layout.

Table B-1 Mixing volumes for Buildings in the Transport Simulation

Building	Volume* (m^3)
Unit 1 Containment	6.54E+03
Unit 2 Containment	6.54E+03
Auxiliary	2.84E+04
Turbine	2.61E+04

* (From Farr, 2014)

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.

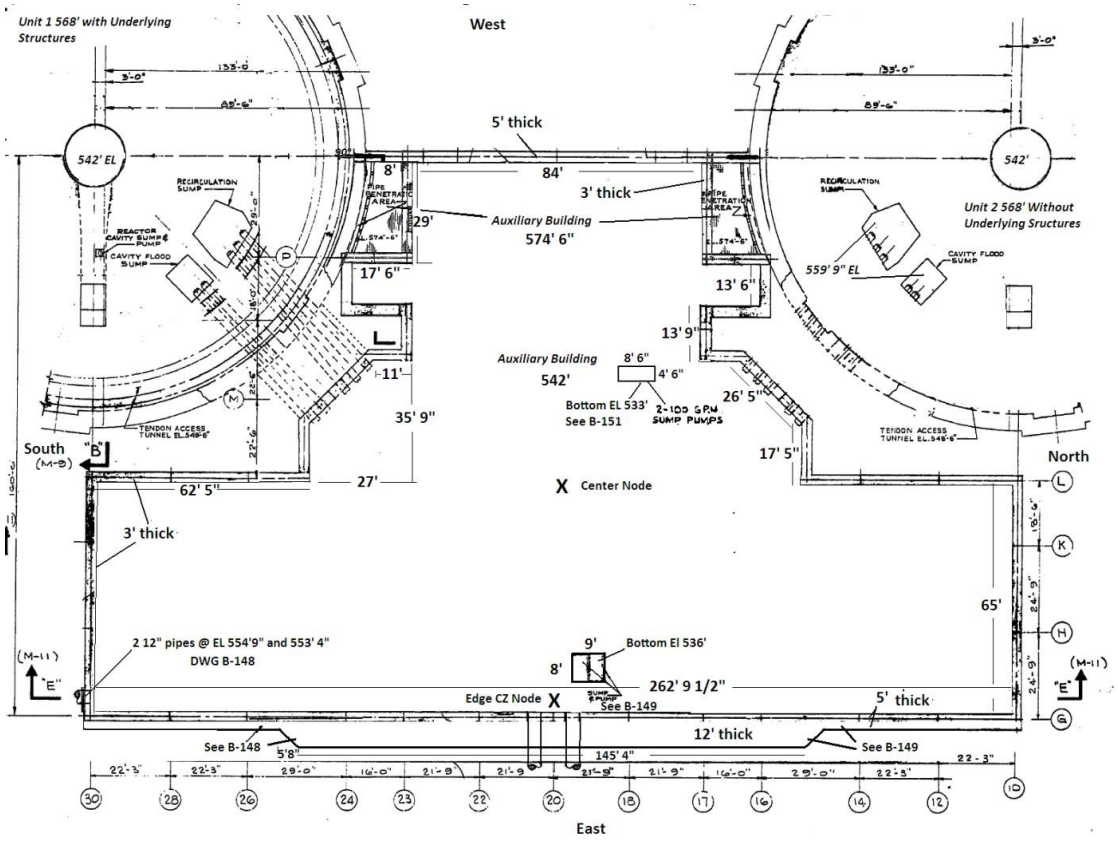

Figure 2. Geometry of the Auxiliary Building.

Table B-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
Turbine Building, Main Steam, Diesel Gen Oil Storage	560	5.79	14679	1.468E-08

Table B-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

B 2.3 Release Models

B 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. For the conceptual model using a receptor well downstream from the Turbine Building all buildings were modeled assuming an instant release of contaminants. This differs from the base case in which the Containment and Auxiliary Buildings were modeled using diffusion controlled release to represent diffusion out of the contaminated concrete. The instant release model is the most conservative approach as the entire inventory is available immediately for transport.

B 2.3.2 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 4 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).

Simulation of flow from the Containment Buildings to the Auxiliary Building is not possible within the constraints of a one dimensional model without assumptions. In this case, the Containment Building is placed directly behind the Auxiliary Building. The Auxiliary Building is modeled as being 31.5 m in the direction of flow (Table 4). The Containment Geometry in Table 4 will be modified to account for the differences in Flow Area between the Containment and Modeled Geometry. This is accomplished by modeling the flow length as 11 meters. This gives the proper volume for the Containment when adjusted down to account for the difference in flow height between the Turbine Building (5.79 m) and the Containment Building (4.27 m), shown in Table 4.

Table B- 4 Building Geometry.

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

B 2.4 Receptor Well Outside the Turbine Building

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 and this is also modeled by adding a third contaminated zone to represent the Containment Building.

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 Containment Buildings and the Auxiliary Building foundations rest 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 3. Therefore, the modeled domain contains the three contaminated regions representing the Containment Building, Auxiliary Building and the section of the Turbine Building that aligns with the Auxiliary Building and groundwater flow direction, Figure 3. 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.

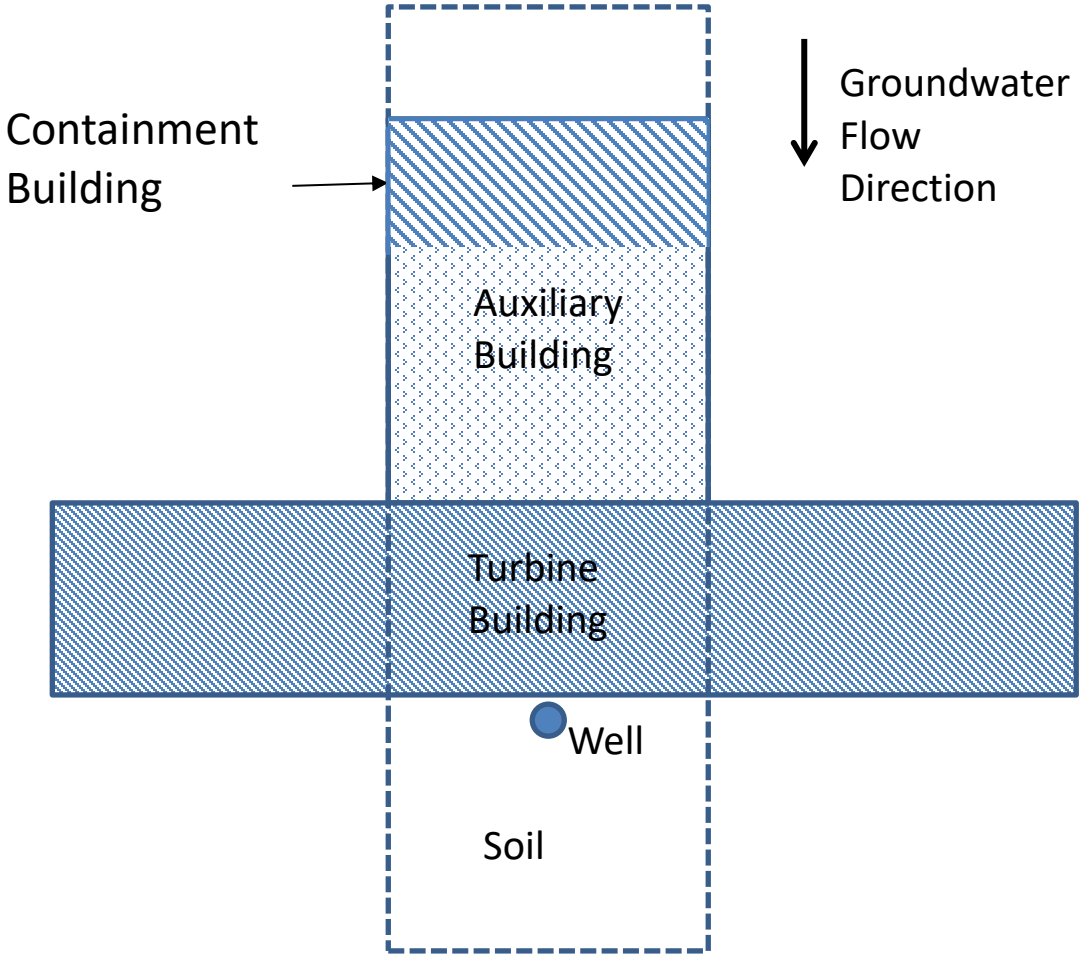

Figure 3 Schematic Representation of Flow the geometry used to assess flow to a well outside the Turbine Building.

B 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.

B 3.1 Parameters

Initial conditions assumed that the groundwater concentration of each contaminant was zero everywhere. The source term is modeled such that with the instant release of contaminants, the initial concentration is fixed at 1 pCi/L in the water in the Auxiliary Building for all three modeling scenarios.
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.

Table B-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

B 3.1.2 Receptor Well Parameters for Transport Model

To simulate transport to a receptor well soil properties and the groundwater flow rate are required. These values are presented in Table 6. 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.

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.

Table B-6 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	95 (cm^3/g)	
	Eu-154	95 (cm^3/g)	

The one-dimensional simulation also requires the depth to the water table to remain the same in all 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.
In the simulation, the Auxiliary Building is modeled with 100 cells giving a total length parallel to flow of 31.5 m. The Turbine Building is modeled with 92 cells for a total length of 29 m. The Containment Buildings are modeled with 35 cells for a total length of 11 m. The adjustment in length for the Containment Building is needed to match the flow area used in the simulation with the total Containment volume.

Three modeling scenarios considered are:
a) High contamination in the Auxiliary Building (1 pCi/L for all contaminants) and lower levels of contamination in the Containment and Turbine Buildings (0.001 pCi/L). This is the expected case.
b) High contamination in the Auxiliary Building and Containment Buildings (1 pCi/L) and a lower level in the Turbine Building (0.001 pCi/L). This would occur only if decontaminatin efforts in the Containment Building do not come close to meeting their expected goals.
c) High contamination level in all 3 Buildings (1 pCi/L). This scenario will not occur based on characterization data, but is included to provide perspective on the modeling results.

The use of a fixed groundwater contamination level provides a useful modeling approach to understand the impact of the distribution of contamination on the peak concentrations at the receptor well. was modeled with the assumption of uniform contamination across the floor of the entire building.
The inventory of the Turbine and Containment Buildings at the time of license termination will be very close to zero but is assumed to provide a water concentration of 0.001 pCi/L for the low contamination cases. The total inventory for each situation is provided in Table 7. Note that the value for the Containment Buildings includes the contributions from both buildings.

Table B-7 Inventory (pCi/L) used for each simulation.

	Containment Contribution (pCi/L)	Auxiliary (pCi/L)	Turbine (pCi/L)
Base Case	0.001	1	0.001
High Containment	1	1	0.001
All Buildings at 1 pCi/L	1	1	1

Using this modeled geometry, the initial inventory in each cell for the base case is provided in Table 8. The reference to the computational cell begins with the first cell in the Containment Building. This is not the first cell in the simulation as computational cells that are not contaminated are modeled upstream. Table 8 provides the inventory per cell in the three simulated regions (Auxiliary plus Containment Contributions, Auxiliary, and Turbine Buildings) and the total inventory in the simulation. Table 9 presents similar information for the case with high contamination levels in the Containment and Table 10 presents this information for the case when all areas have high contamination.

Table B-8. Inventory in each computational cell for the Base Case contamination levels

	Containment Building Inventory (Ci)	Auxiliary Building Inventory (Ci)	Turbine Building Inventory (Ci)	Total Inventory (Ci)
H-3	1.28E-09	3.65E-06	3.36E-09	3.65E-06
Co-60	1.72E-06	4.89E-03	4.50E-06	4.90E-03
Ni-63	4.78E-07	1.36E-03	1.25E-06	1.36E-03
Sr-90	1.90E-08	5.41E-05	4.97E-08	5.41E-05
Cs-134	3.47E-07	9.88E-04	9.10E-07	9.89E-04
Cs-137	3.47E-07	9.88E-04	9.10E-07	9.89E-04
Eu-152	7.32E-07	2.09E-03	1.92E-06	2.09E-03
Eu-154	7.32E-07	2.09E-03	1.92E-06	2.09E-03

Table B-9. Inventory in each computational cell for the High Containment contamination case

	Cells 1 – 35 Inventory (Ci)	Cells 36-135 Inventory (Ci)	Cells 136-228 Inventory (Ci)	Total Inventory (Ci)
H-3	1.28E-06	3.65E-06	3.36E-09	4.93E-06
Co-60	1.72E-03	4.89E-03	4.50E-06	6.61E-03
Ni-63	4.78E-04	1.36E-03	1.25E-06	1.84E-03
Sr-90	1.90E-05	5.41E-05	4.97E-08	7.31E-05
Cs-134	3.47E-04	9.88E-04	9.10E-07	1.34E-03
s-137	3.47E-04	9.88E-04	9.10E-07	1.34E-03
Eu-152	7.32E-04	2.09E-03	1.92E-06	2.82E-03
Eu-154	1.28E-06	3.65E-06	3.36E-09	4.93E-06

Table B-10. Inventory in each computational cell for all buildings having 1 pCi/L contamination case

	Cells 1 – 35 Inventory (Ci)	Cells 36-135 Inventory (Ci)	Cells 136-228 Inventory (Ci)	Total Inventory (Ci)
H-3	1.28E-06	3.65E-06	3.36E-06	8.29E-06
Co-60	1.72E-03	4.89E-03	4.50E-03	1.11E-02
Ni-63	4.78E-04	1.36E-03	1.25E-03	3.10E-03
Sr-90	1.90E-05	5.41E-05	4.97E-05	1.23E-04
Cs-134	3.47E-04	9.88E-04	9.10E-04	2.24E-03
Cs-137	3.47E-04	9.88E-04	9.10E-04	2.24E-03
Eu-152	7.32E-04	2.09E-03	1.92E-03	4.74E-03
Eu-154	7.32E-04	2.09E-03	1.92E-03	4.74E-03

B 4.0 Results

B 4.1 Base Case

The base case consists of contamination is 1 pCi/L in the Auxiliary Building and 0.001 pCi/L in the Containment Building and the Turbine Building. Table 11 provides the predicted peak concentrations in the center of the Auxiliary and Turbine Buildings, the downstream edge of the Turbine Building and at the Receptor Well located 2 meters outside the Turbine Building. The ratio of the concentration at the well to that in the Auxiliary Building is a measure of the impacts of transport on concentration and reflects the conservatism in assuming a well is located in the center of the Auxiliary Building.

Table B-11 Predicted peak concentrations when the Auxiliary building is contaminated to 1 pCi/L

Nuclide	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	0.71	0.60	0.59	0.59	1
Co-60	1	9.99E-04	1.0E-03	7.4E-06	7.4E-06	16.8
Ni-63	1	0.26	0.11	9.28E-02	0.09	286
Sr-90	1	0.75	0.62	6.0E-01	0.60	13.6
Cs-134	1	9.96E-04	9.80E-04	2.62E-05	0.00	5
Cs-137	1	7.08E-02	1.10E-02	7.88E-03	0.01	167
Eu-152	1	1.00E-03	9.92E-04	1.20E-04	0.00	21.2
Eu-154	1	1.00E-03	9.92E-04	6.82E-05	0.00	15.3

A few key points can be determined from Table 11:

- For mobile nuclides (H-3 and Sr-90) the dilution from the Auxiliary Building to the well is less than a factor of 2. This reflects the fast transport time.
- For short lived and less mobile nuclides the time to peak is less than 20 years. This suggests that contamination from the Auxiliary Building the major contribution to the receptor well and that Auxiliary Building contributions to the well are minor. For these nuclides, the peak concentrations are controlled by the contamination in the Turbine Building. This is also confirmed by the peak concentrations in the Turbine Building not rising above 0.001 pCi/L, the initial condition in this building.
- For Ni-63 and Cs-137 they do not reach their peak concentration for several hundred years suggesting the higher contamination in the Auxiliary Building reaches the well. However, the travel time allows for substantial decay and only peak concentrations in the well are 1% for Cs-137 and 10% for Ni-63 of their value in the Auxiliary Building.

B 4.2 Equal Contamination Levels in the Containment and Auxiliary Building Scenario.

In this scenario the contamination level in the Containment Building was set to match the value in the Auxiliary Building at 1 pCi/L. In this simplified scenario the contamination from the containment building is added immediately behind the Auxiliary Building as described in Section 3.1.2. Peak concentrations, the ratio of the peak value in the receptor well to that in the center of the Auxiliary Building and the time to reach the peak in the well are provided in Table 12.

Table B- 12 Predicted peak concentrations when Containment and Auxiliary Building are contaminated to 1 pCi/L.

Nuclide	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	0.81	0.71	0.70	0.70	1.1
Co-60	1	1.0E-03	1.0E-03	7.4E-06	7.4E-06	16.4
Ni-63	1	0.26	0.11	9.30E-02	0.09	285
Sr-90	1	0.77	0.66	0.64	0.64	15.3
Cs-134	1	9.96E-04	9.80E-04	2.62E-05	0.00	5
Cs-137	1	7.08E-02	1.10E-02	7.88E-03	0.01	166
Eu-152	1	1.00E-03	9.92E-04	1.20E-04	0.00	20
Eu-154	1	1.00E-03	9.92E-04	6.82E-05	0.00	15.2

The results from this scenario closely match the Base Case showing only a minor influence of the increased inventory simulated for the Containment Buildings. For the less mobile nuclides the increased inventory in the Containment Buildings had only a minor impact (<1%) on peak concentrations. For the mobile nuclides (H-3, Ni-63, and Sr-90) the peak well concentrations increase slightly (<20%) due to the contributions from the Containment Building.

B 4.3 Equal Contamination Levels in the Containment, Auxiliary and Turbine Buildings Scenario

In this scenario the contributions from all buildings led to an initial concentration of 1 pCi/L. As discussed previously the contributions from the Containment Building are added to a small region of the Auxiliary Building leading to an initial concentration of 3 pCi/L in this region. This increased the inventory in the Turbine Building by a factor of 1000. Peak concentrations and the time to reach the peak in the well are provided in Table 13.

The key difference in this analysis is that the less mobile nuclides had peak concentrations increase by a factor of close to 1000 at the receptor well. The peak concentrations of the mobile nuclides increased by a factor of less than 3. This reflects the importance of decay during transport of the less mobile nuclides.

Table B-13 Predicted peak concentrations when all buildings were contaminated to 1 pCi/m^2

Nuclide	Aux Bldg. (pCi/L)	Turbine Bldg. (pCi/L)	Edge of Turbine Bldg. (pCi/L)	Receptor Well (pCi/L)	Time to peak (years)
H-3	1	1	0.97	0.95	0.5
Co-60	1	1	1.00	0.01	17
Ni-63	1	1	0.99	0.58	50.1
Sr-90	1	1	0.90	0.87	4.5
Cs-134	1	1	0.98	0.03	5.1
Cs-137	1	1	0.98	0.39	23
Eu-152	1	1	0.99	0.12	19.2
Eu-154	1	1	0.99	0.07	15.2

B 4.4 Comparison of Peak Concentrations

Table 14 provides a direct comparison of the receptor well concentrations for the three cases. For nuclides (Co-60, Cs-134, Eu-152, and Eu-154) that increase in concentration by a factor of 1000 when the Turbine Building is simulated to be contaminated at 1 pCi/m^2 this suggests that their peak well concentration is controlled by the contamination in the Turbine Building. For mobile nuclides (H-3 and Sr-90) all areas of contamination contribute. For Ni-63 and Cs-137 for a similar level of contamination the Turbine Building has a greater impact on concentration but contributions from the Auxiliary Building and the Containment Building do reach the receptor well.

Table B- 14 Predicted peak concentrations at the receptor well for all three cases.

Nuclide	Base Case (pCi/L)	Auxiliary and Containment at 1 pCi/L (pCi/L)	All Contaminated to 1 pCi/L (pCi/L)
H-3	0.59	0.70	0.95
Co-60	7.4E-06	7.4E-06	0.01
Ni-63	9.28E-02	9.30E-02	0.58
Sr-90	6.0E-01	0.64	0.87
Cs-134	2.62E-05	2.62E-05	0.03
Cs-137	7.88E-03	7.88E-03	0.39
Eu-152	1.20E-04	1.20E-04	0.12
Eu-154	6.82E-05	6.82E-05	0.07

B 5.0 Conclusions

Three scenarios have been simulated to examine the potential of contaminated water from the Containment Buildings could travel through piping to the Auxiliary Building through another set of pipes to the Turbine Building and to a receptor well 2 m downgradient of the Turbine Building. The Base Case assumed contamination of 1 pCi/L in the Auxiliary Building and 0.001 pCi/L in the Containment and Turbine Buildings. This contaminant distribution was selected based on characterization data and current understanding of the expected end state. A second scenario increased the concentration in the Containment Building to 1 pCi/L to provide an upper bound on the impacts of the Containment Building on peak concentration at the receptor well. The third scenario increased the concentration in the Turbine Building to 1 pCi/L.
The key findings are:

- For mobile nuclides (H-3 and Sr-90) it does not make much difference where the contamination is located as it will reach the receptor well. Adding the contribution from the Containment at the same level as in the Auxiliary Building led only to a slight rise in concentration at the well. It did not change the maximum concentration in the Auxiliary Building.
- For Ni-63 and Cs-137 due to their long half-lifes and somewhat low distribution coefficients (62 for Ni-63 and 45 for Cs-137 in this simulation) contamination in the Containment Building and Auxiliary Building will reach the receptor well, but at peak concentrations less than 10% (Ni-63) and 1% (Cs-137) of their value in these buildings.
- For less mobile nuclides with short half-lifes (Co-60, Cs-134, Eu-152 and Eu-154) very little contamination (<0.1%) in the Auxiliary Building will reach the receptor well.

The analyses performed in these simulations are based on fixed groundwater contamination levels (pCi/L) which leads to a fixed inventory. These modeled inventories should be checked against the actual inventories when characterization is complete.

B 6.0 References

Conestoga-Rovers & Associates, “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.
Milian, L., T. Sullivan. *Sorption (K_d) measurements on Cinder Block and Grout in Support of Dose Assessments for Zion Nuclear Station Decommissioning,* Brookhaven National Laboratory Report to ZionSolutions, June 24, 2014, BNL-1055441-2014-IR.

Sullivan, T.M., "**DUST** - Disposal Unit Source Term: Data Input Guide." NUREG/CR-6041, BNL-NUREG-52375, 1993.
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”, BNL 105542-2014, June, 9, 2014.

Sullivan, T.M., "Evaluation of Maximum Radionuclide Groundwater Concentrations for Radionuclides of Concern Zion Station Restoration Project", Brookhaven National Laboratory, Draft Letter Report, December 3, 2014.

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 ZionSolutions, December 12, 2012, BNL-105981-2012-IR.

Attachment C to TSD 14-009: Comparison of Peak Unitized Concentrations in the Containment and Auxiliary Buildings

Terry Sullivan
January 13, 2017

C 1. Introduction

The DCGL values (pCi/m^2) calculated for each building (Reference Zion*Solutions* TSD 14-010, Rev. 4, "RESRAD Dose Modeling for Basement Fill Model, Soil DCGL, and Calculation of Basement Fill Model Dose Factors") are a function of the inventory released and the volume of water available for dilution. The DCGLs represent the contamination per unit surface area that lead to a predicted dose of 25 mrem/yr. Every building has a unique DCGL for each radionuclide of concern (ROC). The DCGL is directly proportional to the groundwater concentration. The dose from all pathways is linearly proportional to the groundwater concentration as the amount sorbed on to the backfill for excavation and drilling scenarios is proportional to the groundwater concentration. Therefore, the peak concentrations in each building should be the same. This analysis confirms that the water concentrations in the Containment and Auxiliary Basement are the same when the source term for each is at the DCGL.

C 2. Analysis

The Containment Building and the Auxiliary Building were used to demonstrate that the peak water concentrations in each building were the same when contamination is present at the DCGL. The DCGL calculation required a revised Surface Area (proportional to inventory) in these buildings as listed below:

Table C-1 Revised Surface Area for DCGL Calculation

Basement	Structure Areas Included in DCGL Calculation	Total Surface Area (m^2)
Containment	Containment + SFP/Transfer Canal + Fuel Transfer Tube	3486
Auxiliary	Auxiliary + SFP/Transfer Canal	7226

For consistency with previous calculations, the Auxiliary Building is set to a unitized contamination level of 1 pCi/m^2 for each nuclide. Assuming contamination in the Auxiliary building at 1 pCi/m^2, the contamination level in Containment must be set equal to 1 pCi/m2 multiplied by the ratio of the Containment DCGL to Auxiliary Basement DCGL (Containment DCGL/Auxiliary DCGL). The fourth column in the table presents this ratio and is the inventory per unit area (pCi/m^2) used in the simulation.

Table C-2 Containment and Auxiliary Building DCGL values

Nuclide	Containment DCGL (pCi/m^2)	Auxiliary DCGL (pCi/m^2)	DCGL Ratio - Containment simulated inventory (pCi/m^2)
H-3	2.64E+08	5.58E+08	4.73E-01
Co-60	6.27E+08	3.46E+10	1.81E-02
Ni-63	4.46E+09	1.21E+10	3.69E-01
Sr-90	1.59E+06	1.05E+07	1.51E-01
Cs-134	3.63E+07	3.73E+08	9.72E-02
Cs-137	4.57E+07	1.31E+08	3.49E-01
Eu-152	1.85E+09	5.81E+10	3.19E-02
Eu-154	1.28E+09	5.11E+10	2.50E-02

Similar to the initial calculations in the main body of the report, the Auxiliary Building is modeled using diffusion release from a 0.5 inch contaminated zone. This causes the time to reach peak concentration to be non-zero and depends on the diffusion coefficient and half-life of the nuclide. The Containment Building is modeled using instant release of all contamination.

C 3. Results

The mixing bath model was run with a surface contamination level of 1 pCi/m^2 in the Auxiliary Building for each ROC. The contamination level in the Containment Building is given by the values in Table C-2 determined by the ratio of the DCGLs. The surface area for each building is provided in Table C-1. The volume of each building is unchanged from the value used in the original calculations (6537 m^3 in Containment and 28445 m^3 in the Auxiliary Building). Both buildings are assumed to be backfilled with the same material at a density of 1.5 g/cm^3 and a porosity of 0.25. Sorption as represented by Kd values remains unchanged from the original calculation.

Table C-3 presents the peak concentrations in the Containment and Auxiliary Buildings and the percentage difference in the results. The results are in excellent agreement showing that the peak concentration on each building is the same when the unitized inventory in each building is scaled to the DCGL values. The slight differences are due to numerical error introduce when performing the diffusion controlled release model in the Auxiliary Building.

Table C-3 Comparison of Peak Concentration in the Containment Building and the Auxiliary Building.

Peak Concentration (pCi/L)

	Auxiliary	Containment	% Diff
H-3	1.02E-03	1.01E-03	0.98
Co-60	2.88E-08	2.89E-08	-0.35
Ni-63	2.11E-06	2.11E-06	0.00
Sr-90	2.17E-05	2.18E-05	-0.46
Cs-134	7.65E-07	7.63E-07	0.26
Cs-137	2.70E-06	2.74E-06	-1.48
Eu-152	1.19E-07	1.19E-07	0.00
Eu-154	9.30E-08	9.32E-08	-0.22

C 4. Conclusion

The results of the mixing bath model calculation demonstrate that the peak concentration will be the same in each building of the building has an inventory at the DCGL value or if the inventories are scaled relative to the DCGL value. This become important when examining flow between buildings due to the pipes and tunnels that connect them. It shows that the peak concentration cannot exceed the concentration that would arise if the inventory was based on contamination to the DCGL level. Although the DCGL value changes for each building, the groundwater concentration in the building, assuming contamination at the DCGL concentration, is the same in each building. This means that even with flow the groundwater concentration can never exceed the limiting value. This was demonstrated in Attachment B. A similar calculation to that in Attachment B that included flow down to the Crib House is provided in Attachment 4.

Attachment D to TSD 14-009. Flow between buildings Containment to Crib House

D 1.0 Introduction

Attachment B presents a calculation of flow from the Containment Building, through the Auxiliary Building through the Turbine Building to a well 2 meters outside of the Turbine Building. This analysis includes the SFP/Transfer Canal and Crib House/Forebay to the flow calculation to confirm that including the additional building in the transport calculation will not lead to concentrations in a downstream basement that exceeds the upstream concentrations.

The source terms in Attachment B were unit concentration (1 pCi/L) of groundwater for the Containment and Auxiliary Building and a fraction of the unit concentrations, e.g. 1E-03 pCi/L for the Turbine source term, to reflect actual Turbine Building characterization data. Unit water concentration source terms were used because by definition, the water concentration must be equal in each Building when the source term is at the maximum allowable in each Building. Because the Appendix B simulation was intended to assess relative concentrations in the outside well when the source term was at the maximum allowable in the Containment and Auxiliary Building, a unit water concentration in each Building represents that condition. Based on characterization data, the Turbine Building was estimated to contain activity at levels that were 1000 times lower than the maximum allowable. Therefore, on a relative basis, if the water concentration is 1 pCi/L at the maximum activity, the water concentration would be 1E-03 pCi/L if the activity is 1000 times lower than the maximum.

This simulation uses essentially the same relative source term as in Appendix B in that the water concentrations in the Auxiliary and Containment are equal at t=0. However, the recently calculated DCGLs (in units of pCi/m^2) were used as the basis for the unitized source term to provide a more direct link between the simulation source term and the DCGL values. As in Appendix B, when the maximum activity is applied as the source term, i.e., the DCGL, the same water concentrations will result. This is demonstrated to be the case in Attachment C. The concentrations are different between Attachment B and Attachment C, which is to be expected, but the critical aspect of comparison is that the relative water concentrations applied in the two simulations are the same, which they are. In both cases, the water concentrations in Containment and the Auxiliary Building are equal.

It is important to recognize that the movement of contamination through the hydraulically connected basements at Zion will always lead to a decrease in the peak concentration in each building. For example, assume that the mixing bath model shows a peak water concentration of 1 (in any units) and that this concentration will lead to a dose of 25 mrem/yr in the Auxiliary Building. As shown in Attachment C, if all buildings were contaminated at their respective Derived Concentration Guidance Level Values (DCGL) concentrations, the peak water concentration in all other buildings would also equal 1. As the water flows from the Auxiliary Building into and through other buildings, water concentrations cannot increase but will decrease with radiological decay.

A more realistic example is water flowing from the Auxiliary Building with surface contamination at the maximum level (i.e., the DCGL) and therefore contains the maximum water concentration (assumed to be 1 in this discussion). Assume that the water flows into and through the Turbine Building which contains contamination at levels at least 1000 times lower than the Auxiliary Building DCGL based on Draft FSS results (Yetter, 2016). The initial concentrations in the Turbine Building mixing bath model would be far below 1. The Turbine water concentration will increase as higher concentration water flows from the Auxiliary Basement and through the Turbine basement, but it will never be greater than 1 and will further decrease due to radioactive decay during the flow period. The dose is directly proportional to the groundwater concentration, therefore the dose from the Turbine Building will always be less than the 25 mrem/yr.

This calculation adds two additional buildings to the flow calculation in Appendix B, the Spent Fuel Building and the Crib House/Forebay. The Spent Fuel Building is being addressed by placing the inventory of this building into the Containment Building to calculate DCGLs for radionuclides of concern (ROC) and therefore, because this calculation is based on DCGLs (unitized), the Fuel Building inventory is included (unitized). There is a potentially large connection from the Turbine Building to the Crib House/Forebay through the Circulating Water Intake pipe but this pipe has been grouted. This calculation assumes that the hydraulic connection between the Turbine Building and Crib House through the Circulating Water Intake Pipe still exists. Although there was piping from the primary structures to the WWTF during operation., these pipes will be removed during decommissioning resulting in no hydraulic connection.

The calculation in this Attachment includes the buildings simulated for and discussed in Attachment B plus the Crib House/Forebay area and activity in the Spent Fuel Building by inclusion with Containment during the DCGL calculation. The other change from Attachment B is that the surface areas of each building have been revised to account for the Spent Fuel source term and the Circulating Water Intake Pipe and Discharge Tunnel during the DCGL calculations. This increase in surface area increases leads to lower DCGL values.

D 2.0 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. To maintain consistency between all calculations DUST-MS was used for all simulations. The nuclides of concern and model parameters (distribution coefficient, bulk density, porosity, etc.) are provided in the main body of the report.

D 2.1 Release Models

For the instant release model the key parameters are the distribution coefficient (K_d), porosity and bulk density of the fill material. For the conceptual model the Containment, Turbine and Crib House/Forebay Buildings were modeled using instant release. For consistency with the base case, the Auxiliary Building was modeled with diffusion controlled release to simulate the release out of the contaminated concrete. The instant release model is the most conservative approach as the entire inventory is available immediately for transport. All of the other parameters (diffusion coefficient for the Auxiliary release) are identical to those presented in the main body of the report. In particular, the water flow velocity is assumed to be that of regional groundwater flow. This will greatly overpredict transport rates and lead to higher concentrations downstream of the Auxiliary Building than if it was assumed the walls in these buildings were intact and flow was governed by the infiltration rate at the surface.

D 2.2 Model Geometry

The major change between Attachment 2 and this Attachment is the inclusion of the Crib House/Forebay in the model and including the Spent Fuel Building with Containment source term. Additionally, in Attachment B, the source was set to provide a concentration of 1 pCi/L in the Auxiliary Building. In this Attachment the source term is based on the contaminated surface area modeled for each building and normalized DCGL values.

DUST-MS is a one-dimensional model. The conceptual model contains a contaminated floor in the direction of flow. The one-dimensional geometry in the DUST-MS model requires a constant flow area to calculate the correct water concentrations above the floor. The volume being modeled is the flow area multiplied by the length of the simulation. 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 1 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 (Farr, 2014. For the Crib House/Forebay the dimensions are listed separately, but in the analysis, they are combined. 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) and are provided in Table 1.

Table D-1 Actual Building Geometry.

Structure	Width or Radius m	Height to Water Table m	Flow Area (m^2)	Contaminated Zone Length (m)	Void Space to the Water Table (m^3)
Containment Buildings	20.95	4.27	140.4	44.8	6537
Auxiliary Building	80.11	11.28	903	31.5	28445
Turbine Building	145.1	5.79	571.5	29	26135
Crib House /Forebay	52.1/52.1	12.8/11.6	667.6/604	24.4/20	16265/12061

When modeling these buildings individually, the values in Table 1 are used in the simulation. When modeling transport between these buildings assumptions are necessary to simulate the movement of contaminants and maintain a constant flow area. In this case, the height to the water table and width of each building is modeled as having the same value. This will change the volume simulated in each building. In the conceptual model, the source term is the product of the surface contamination level multiplied by the surface area of the building. Thus, the inventory is directly proportional to surface area. The groundwater concentration is proportional to the inventory divided by the volume of water available for mixing. This implies that the concentration is proportional to the Surface Area of the building divided by the volume of water available for mixing. The volume of water is the total void space to the water table multiplied by the porosity to account for the backfill. In all calculations, the porosity is held constant at a low value of 0.25 to minimize the water available for mixing and maximize groundwater concentrations. The key point of this discussion is that the groundwater concentration is directly proportional to the surface area to volume ratio and therefore, if the actual volume is adjusted, the surface area must be adjusted similarly to maintain the same groundwater concentration. The adjustments to volume and surface area are described below.

The modeled geometry is presented in Figure 1. 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 volume of the Turbine Building in this simulation is reduced as compared to the actual volume.

The one-dimensional simulation also requires the depth to the water table to remain the same in all 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.). Reducing the height of the water table in the Auxiliary Buildings reduces the simulated volume of the Auxiliary

Building by the ratio of the water table levels (5.79 m/11.28 m). The total area available for flow (building width multiplied by the height to the water table) is 463.7 m^2.

The width of the Crib House/Forebay is 52.1 m, which is less than the modeled width based on the Auxiliary Building. Maintaining a width of 80.1 m increases the volume of the Crib House/Forebay. However, the depth to the water table ranges from 11.6 m in the Forebay to 12.8 m in the Crib House, and using the depth of 5.79 m reduces the modeled volume of the Crib House/Forebay. The length of the Crib House/Forebay is 44.4 m. Using this length and the modeled geometry of 463.7 m^2 flow area and a length of 44.4 m gives a total volume of 20588 m^3, which is less than the actual volume of the Crib House/Forebay.

The Containment Building is placed directly behind the Auxiliary Building. The Auxiliary Building is modeled as being 31.5 m in the direction of flow (Table 4). The Containment Geometry in Table 4 will be modified to account for the differences in Flow Area between the Containment and Modeled Geometry. The modeled geometry is presented in Table 2.

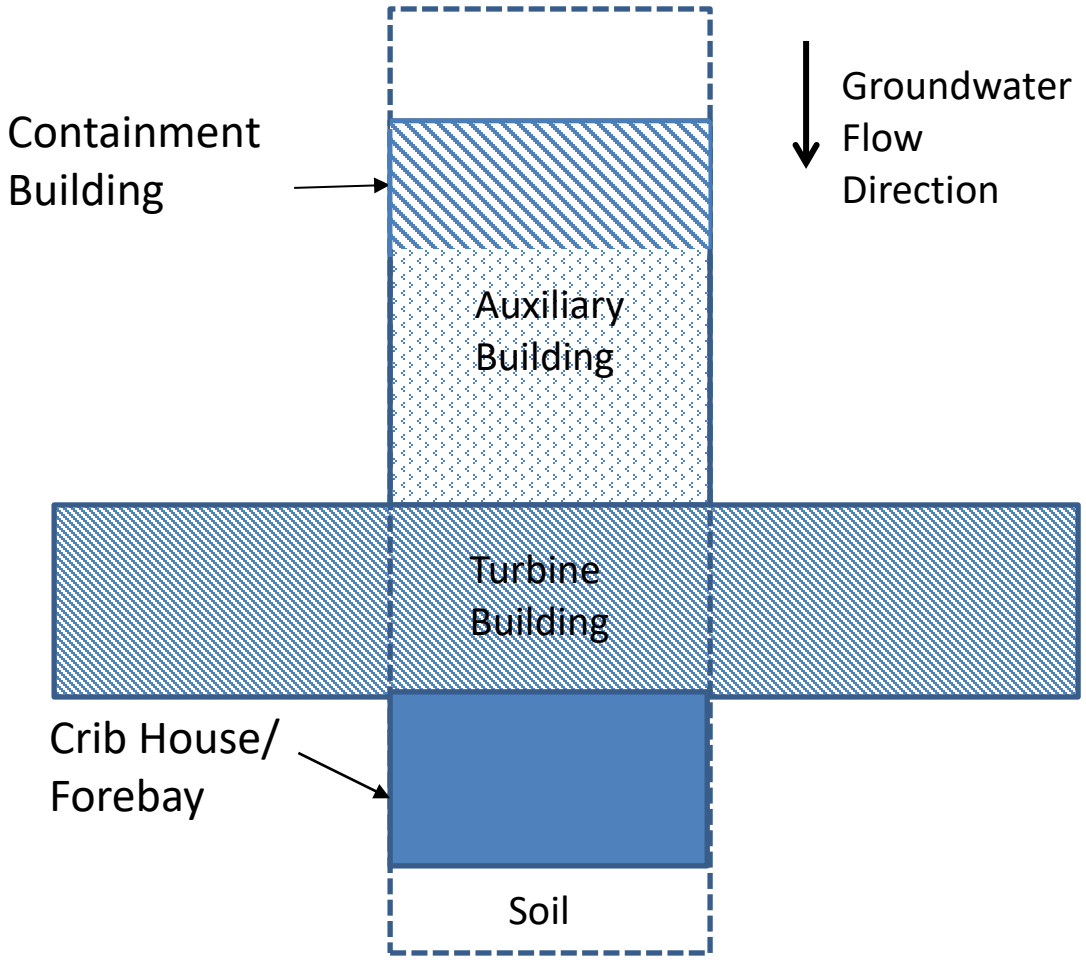

Figure D-4 Schematic of Flow through multiple buildings at the Zion Site.

Table D-2 Modeled Geometry

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	80.11	5.79	464	14.1	6537
Auxiliary Building	80.11	5.79	464	31.5	14616
Turbine Building	80.11	5.79	464	29	13456
Crib House/Forebay	80.11	5.79	464	44.4	20588

To maintain the same predicted groundwater concentration as the original geometry, the modeled inventory is scaled to match the original surface area to volume ratio in the buildings with altered model volumes. Table 3 presents the total surface area, modeled volume and the actual surface area to volume ratio. For the buildings that have reduced volumes it is assumed that the Surface Area to Volume ratio is constant throughout the building and the total inventory in the building is scaled accordingly.

Table D-3 Revised surface areas, modeled volume, and surface area to volume ratio.

Basement	Structure Areas Included in DCGL Calculation	Total Surface Area (m^2)	Modeled Volume (m^3)	Original Surface Area/Volume
Containment	Containment + SFP/Transfer Canal + Fuel Transfer Tube	3486	6537	0.53
Auxiliary	Auxiliary + SFP/Transfer Canal	7226	14616	0.25
Turbine	Turbine + Circulating Water Discharge Tunnels (2)	25191	13456	0.96
Crib House/Forebay	Revised Crib House/Forebay + Circulating Water Intake Structures (2)	18681	20588	0.66

D 2.3 Modeled Inventory

The modeled inventory assumes that the Auxiliary Building is at 1 pCi/m^2 and based on characterization data that shows concentrations to be at least 1000 times lower in the Turbine and Crib House/Forebay buildings, these two buildings are set to 0.001 pCi/m^2. Identical to Attachment C, the Containment building area contamination level is scaled to the ratio of the DCGL in the Auxiliary Building to the DCGL in containment building. Table 4 shows the contamination for each nuclide in each building per unit area and the total inventory. The total inventory is the inventory per unit area multiplied by the surface area to volume ration in Table 3.

Table D-4 Inventory per unit area (pCi/m^2) and for the entire building (pCi)

Nuclide	Auxiliary	Containment	Turbine	Crib House/ Forebay	Auxiliary	Contain-ment	Turbine	Crib House/ Forebay
	pCi/m2	pCi/m2	pCi/m2	pCi/m2	pCi	pCi	pCi	pCi
H-3	1.00E+00	4.73E-01	1.00E-03	1.00E-03	3713	1649	13.0	13.6
Co-60	1.00E+00	1.81E-02	1.00E-03	1.00E-03	3713	63	13.0	13.6
Ni-63	1.00E+00	3.69E-01	1.00E-03	1.00E-03	3713	1284	13.0	13.6
Sr-90	1.00E+00	1.51E-01	1.00E-03	1.00E-03	3713	526	13.0	13.6
Cs-134	1.00E+00	9.72E-02	1.00E-03	1.00E-03	3713	339	13.0	13.6
Cs-137	1.00E+00	3.49E-01	1.00E-03	1.00E-03	3713	1214	13.0	13.6
Eu-152	1.00E+00	3.19E-02	1.00E-03	1.00E-03	3713	111	13.0	13.6
Eu-154	1.00E+00	2.50E-02	1.00E-03	1.00E-03	3713	87	13.0	13.6

D 3.0 Results

Using the inventory values in Table 4 and the geometry values in Table 2 and simulating instant release in Containment, Turbine and Crib House/Forebay buildings and diffusion release in the Auxiliary Building, the peak concentration at the centerline of each Building was calculated, Table 5. As shown when the inventory is adjusted to the ratio of the DCGL values, the peak concentration in the Auxiliary and Containment Buildings match. The ratio of the peak concentration in each building to that in the Contain Building is also presented in Table 5. The peak concentrations in the Turbine and Crib House/Forebay buildings are lower for all nuclides and substantially lower for the short-half life nuclides with Kd values greater than 10. It is important to note that the peak concentrations in the Turbine and Crib House/Forebay buildings are always less than in the Auxiliary Building.

The time of the peak concentration was also recorded, Table 6. For the Containment Building with instant release, the time of peak concentration is always at the start of the calculation. The use of diffusion release causes a delay in the time to peak concentration in the Auxiliary Building. The time to peak concentration in the Turbine and Crib House/Forebay buildings is a function of the transport from the higher concentration Auxiliary Building to these buildings and depends on the water flow rate and sorption properties of the backfill. If the peak value occurs at T= 0 years in these buildings it indicates that the effects of sorption and radioactive decay are large enough to prevent substantial amounts of radioactivity released in the Auxiliary Building from reaching the center line of the adjacent Buildings. This occurs for the short-lived sorbing nuclides (Co-60, Cs-134, Eu-152, and Eu-154).

For the more mobile nuclides (H-3 and Sr-90) the peak centerline concentration in the Turbine Building reaches 84% of the peak in the Containment Building for H-3 and this occurs at 0.7 years (0.6 years past the peak time in the Auxiliary Building). For the Crib House/Forebay building, the peak H-3 concentration 63% of the peak in the Containment Building, reflecting decay during

transport. The peak concentrations in the Turbine Building and Crib House/Forebay increase from the initial concentration which were 1000 times less than the Auxiliary Building. However, the peak concentrations, of all radionuclides, in the Turbine Building and Crib House/Forebay never exceed those in the Containment or Auxiliary Buildings.

Table D-5 Peak Concentration for each ROC in each Building

	Peak Concentration (pCi/L)						
	Containment	Auxiliary	Aux/Cont Ratio	Turbine	Turbine /Cont Ratio	Crib House	Crib House/Cont Ratio
H-3	1.01E-03	1.01E-03	1.00	8.52E-04	0.84	6.35E-04	0.63
Co-60	2.90E-08	2.88E-08	0.99	2.88E-09	0.10	1.96E-09	0.07
Ni-63	2.12E-06	2.11E-06	1.00	7.21E-07	0.34	1.88E-08	0.009
Sr-90	2.19E-05	2.19E-05	1.00	1.92E-05	0.88	1.26E-05	0.58
Cs-134	7.67E-07	7.64E-07	1.00	1.42E-08	0.02	9.68E-09	0.013
Cs-137	2.76E-06	2.76E-06	1.00	3.08E-07	0.11	9.71E-09	0.004
Eu-152	1.20E-07	1.19E-07	0.99	6.75E-09	0.06	4.61E-09	0.04
Eu-154	9.37E-08	9.30E-08	0.99	6.75E-09	0.07	4.61E-09	0.05

Table D-6 Time to peak concentration in each building for each ROC.

	Time to peak (years)			
Nuclide	Containment	Auxiliary	Turbine	Crib House
H-3	0	0.1	0.7	1.6
Co-60	0	3.9	0	0
Ni-63	0	16	150	> 300
Sr-90	0	6.2	14	26
Cs-134	0	1.4	0	0
Cs-137	0	13.3	79	0
Eu-152	0	10.0	0	0
Eu-154	0	6.1	0	0

D 4.0 Conclusion

A simulation of transport from the Containment through the Auxiliary and Turbine Buildings to the Crib House/Forebay Building was performed using bounding flow rate assumptions. The simulation conservatively assumed that the water flow rate was the regional groundwater velocity (i.e. the concrete walls of the building did not impede flow) and that each building was directly

downstream of the other buildings. The expected flow rate would be due only to infiltration into the intact subsurface structures which is more than ten times less than the regional groundwater flow over this area.

The source term for the Auxiliary Building was set to a unitized surface area contamination level of 1 pCi/m^2. The source term for the Containment Building was set to the ratio of the Containment DCGL to the Auxiliary Buildings DCGL multiplied by the contamination level in the Auxiliary Building (1 pCi/m^2). This ensures that the source terms represent normalized activity at the DCGL levels in both buildings and that the peak water concentrations in the Auxiliary and Containment Buildings equal as is required to be the case when both buildings are at DCGL levels. Based on characterization data the contamination level in the Turbine and Crib House/Foreby buildings is 1000 times less than the Auxiliary DCGL and therefore the source term for these two buildings were set to 0.001 pCi/m^2.

The results showed that the peak concentration in the Turbine Building, for all radionuclides, never exceeds 88% of the Auxiliary Building (and Containment Building) peak concentrations. For Cs-137, the primary dose contributing radionuclide, the peak Turbine Building concentration is 11% of the Auxiliary Building (and Containment) Concentration. The peak concentration, for all radionuclides, in the Crib House/Forebay never exceeds 63% of the Auxiliary Building (and Containment) concentration. For Cs-137, the peak Crib House/Forebay concentration is 0.4% of the Auxiliary Building (and Containment) Concentration.

The peak concentrations in the downstream Turbine Building and Crib House/Forebay never exceed the upstream peak concentrations in the Auxiliary Building or Containment. The dose is assigned based on the maximum peak concentration in any of the buildings because that is where the water well is assumed to be installed. Therefore, the dose can never exceed the dose assigned to the Building with the highest levels of residual radioactivity at license termination which is expected to be the Auxiliary Building. If all buildings contain concentrations at the DCGL levels then all of the buildings will have equal water concentrations. and flow is immaterial.

D 5.0 References

Conestoga-Rovers & Associates, “Simulation of the Post-Demolition Saturation of Foundation Fill Using a Foundation Water Flow Model,” Zion Restoration Project, Conestoga-Rovers & Associates, Chicago, IL, December, 2014, Reference No.054638, Report No. 4.

Bob Yetter email to Terry Sullivan, “FSS ISOCS Results”, February 9, 2017

Attachment E to TSD 14-009. Release through engineered opening in the Steam Tunnel

E-1.0 Introduction

The plan at the Zion site is to remove all surface structures to three feet below grade at an elevation of 588 feet Mean Sea Level (MSL) and leave the underground structures in place and backfill. These buildings have thick concrete walls that are expected to remain intact for a long period of time. The main buildings are the two Containment Buildings, the Auxiliary, Turbine, and Crib House/Forebay buildings. They are all hydraulically connected through penetrations and piping that will remain. As long as the concrete walls remain effective barriers to flow, precipitation on the ground surface will migrate downward and will eventually fill the pore space of the backfill in these buildings. This could eventually lead to overflowing of the buildings. A study was performed by Conestoga Rovers & Associates (CRA, 2014) that calculated a period of 26 years to fill the Containment, Auxiliary, and Turbine buildings.

To avoid releasing contamination close to the ground surface it has been decided to construct openings in the steam tunnel that connect the Containment and Turbine Buildings. There is also a connection between the Auxiliary Building and the Steam Tunnel. The opening will be just above the nominal water table elevation of 579 feet MSL. It is expected that after the precipitation fills the subsurface structures the water will flow out of these openings. Flow to this opening could occur from either the Containment Building or Turbine Building.

The simulation in this attachment examines the flow out of the opening to a well 2 meters downstream of the opening. The contamination level in the Containment Building is expected to be higher than in the Turbine Building. For this reason, the simulations assume that all of the flow comes from the Containment Building. The Auxiliary Building is simulated as having the same concentration as the Containment Building using the unitized DCGL values as discussed in Attachment D. The distance from the Auxiliary Building to the opening is further than from the Containment Building. For this reason, the Containment Building was used as the source of high contamination. If flow was simulated from both directions, the resulting concentration at the well would be lower because of the longer transport distance from the Auxiliary Building and the lower flow velocity from the Containment Building.

The predicted flow due to infiltration (CRA, 2014) is greater than the annual well pumping volume and for conservatism it is assumed that all of the pumping volume water comes from the water that flows out of the steam tunnel penetration. This is conservative because the well water from the steam tunnel is entering the top of the water table. As it enters the water table, it would mix with clean water. Additionally, if the well was adjacent to the penetrations in the steam tunnel, the well screen depth of 3 meters would take clean water at the bottom of the well screen. If the well was located far enough away from the penetrations to have uniform mixing, sorption and radioactive decay would decrease the concentrations in the groundwater.

E-2.0 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. To maintain consistency between all calculations DUST-MS was used for all simulations. The nuclides of concern and model parameters (distribution coefficient, bulk density, porosity, etc) are provided in the main body of the report.

E-2.1 Release Models

For the instant release model the key parameters are the distribution coefficient (K_d), porosity and bulk density of the fill material. For the conceptual model the Containment, Turbine and Crib House/Forebay Buildings were modeled using instant release. For consistency with the base case, the Auxiliary Building was modeled with diffusion controlled release to simulate the release out of the contaminated concrete. The instant release model is the most conservative approach as the entire inventory is available immediately for transport. All of the other parameters (diffusion coefficient for the Auxiliary release) are identical to those presented in the main body of the report.

E-2.2 Water Flow

The CRA report (CRA 2014) calculates that on average 1186 gallons of water per day will enter the main complex (two Containment buildings, Auxiliary, Turbine Buildings, and steam tunnels). This is 4.5 m^3 of water per day or 1642 m^3 per year.

E-2.3 Model Geometry

Figure 1 shows the schematic of the connection between the steam tunnel and the Containment Building. The bottom elevation of the tunnel is at 570 feet MSL. The top of the tunnel will be at 588 feet MSL, three feet below grade. In the section with a wall opening, the top of the tunnel will be at 580 feet elevation. Therefore, the model uses a height of 10 feet for the steam tunnel. To model this using the one-dimensional DUST-MS model requires that the area perpendicular to flow remains constant. For this reason, the minimum tunnel width of 12 feet (Section C and D in Figure 1) is used. This gives an area perpendicular to flow of 120 ft^2 (11.15 m^2). The length from the Containment Building to the opening in the steam tunnel is the sum of the distances in Figure 1 and is 68.83 ft (21 m). Table 1 summarizes the main geometry parameters of the model. A well is assumed to be placed 2 meters downstream from the opening. The simplified 1-D model is found in Figure 2.

The Containment Building is modeled using the diameter of the building 44.8 meters and the flow area of 11.15 m^2. This choice of flow area is necessary in the one-dimensional model and gives a modeled volume less than the physical volume. This reduces the modeled volume to 499.5 m^3 as compared to the actual volume beneath the water table of 6537 m^3. It should be noted that the elevation of the engineered opening is at 580 foot elevation, MSL, which is one foot above the groundwater elevation at the site. Therefore, the volume available in the containment would actually be higher when modeling release through the engineered opening. For conservatism, this is neglected in the following calculations.

As discussed in Attachment D, the groundwater concentration is proportional to the total inventory released and the volume of water available to receive the inventory. The inventory is the concentration per unit area (pCi/m^2) multiplied by the total surface area found in the building. Therefore, to obtain the same groundwater concentration as the original geometry, the modeled geometry must have the same Surface Area to Volume ratio as the original geometry, (0.53). Using the modeled volume, the modeled surface area is 264.7 m^2. Similarly, the Steam Tunnel inventory, which is included with the Turbine inventory in calculating DCGL values will be determined using the actual Surface Area to Volume ratio of the Turbine Building (0.96). The modeled surface areas are provided in Table 1. This will provide the same groundwater concentration per unit surface contamination level as in the other transport calculations in Attachments D.

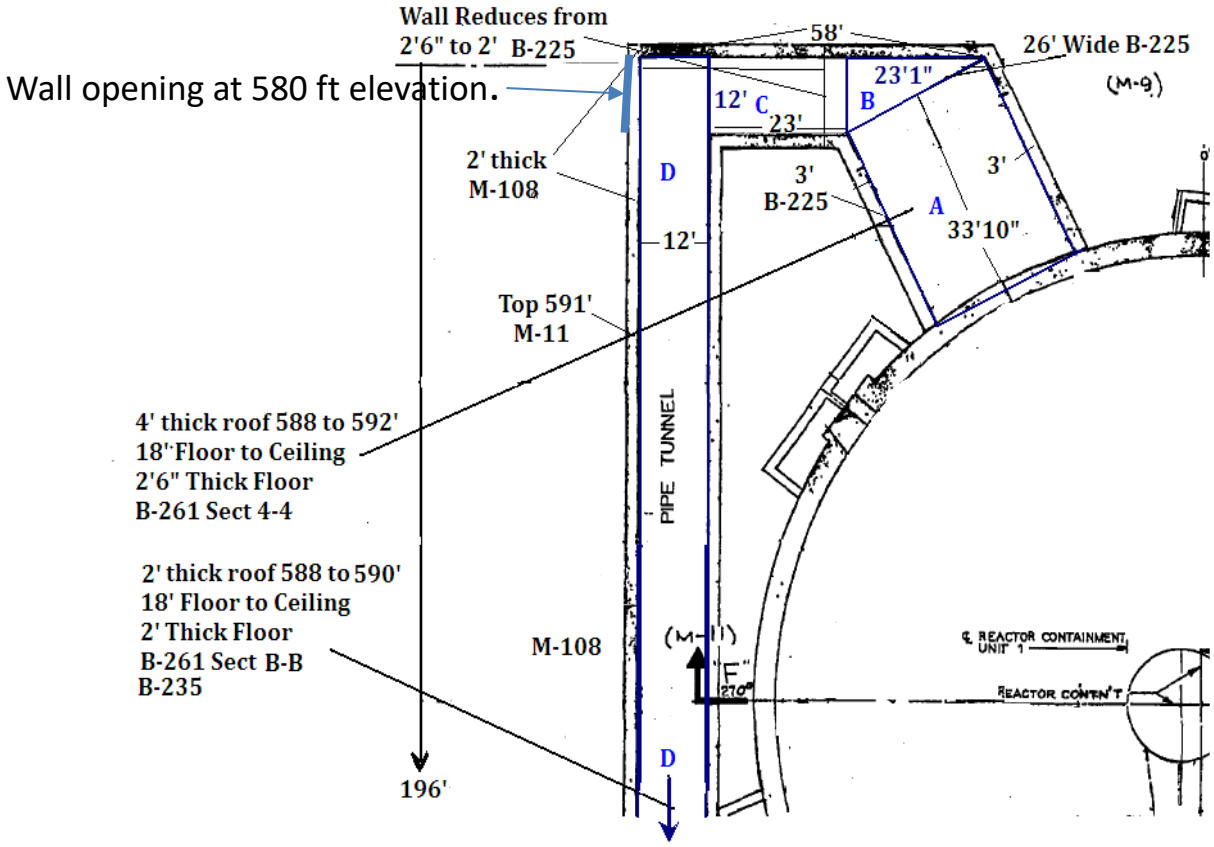

Figure 5 Containment and Steam Tunnel with location of opening for drainage.

Table E-1 Modeled Geometry

Structure	Length(m)	Height To opening (m)	Flow Area (m^2)	Contaminated Zone Length (m)	Void Space to WT m^3	Surface Area to Volume ratio (1/m)	Modeled Surface Area (m^2)
Containment Building	3.66	3.05	11.15	44.8	499.5	0.53	264.7
Steam Tunnel	3.66	3.05	11.15	21	234.1	0.76	177.9

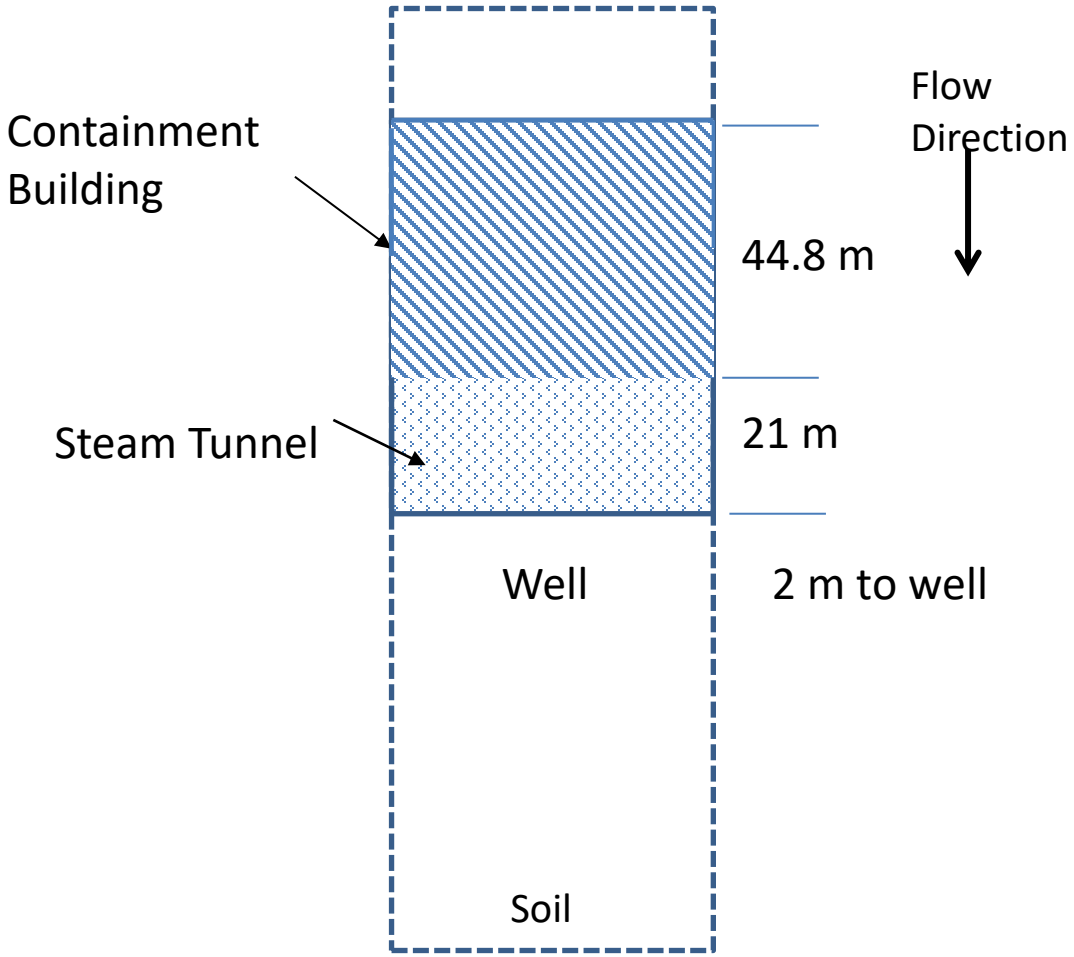

Figure 6 Schematic diagram of one-dimensional conceptual model.

E 2.4 Modeled Inventory

For consistency with Attachment D, the modeled inventory assumes that the Auxiliary Building is at 1 pCi/m^2. Based on characterization data that shows concentrations to be at least 1000 times lower in the Steam Tunnels, the modeled inventory assumes that the Steam Tunnel is at 0.001 pCi/m^2. Identical to Attachment C, the Containment building area contamination level is scaled to the ratio of the DCGL in the Containment to the DCGL in the Auxiliary Building. Table 2 shows the contamination for each nuclide in the two structures per unit area and the total inventory. The total inventory is the inventory per unit area multiplied by the surface area, Table 1.

Table E-2 Inventory per unit area (pCi/m2) and for the entire building (pCi)

Nuclide	Containment pCi/m^2	Steam Tunnel pCi/m^2	Containment pCi	Steam Tunnel pCi
H-3	4.73E-01	1.00E-03	1.25E+02	2.26E-01
Co-60	1.81E-02	1.00E-03	4.80E+00	2.26E-01
Ni-63	3.69E-01	1.00E-03	9.76E+01	2.26E-01
Sr-90	1.51E-01	1.00E-03	4.00E+01	2.26E-01
Cs-134	9.72E-02	1.00E-03	2.57E+01	2.26E-01
Cs-137	3.49E-01	1.00E-03	9.23E+01	2.26E-01
Eu-152	3.19E-02	1.00E-03	8.44E+00	2.26E-01
Eu-154	2.50E-02	1.00E-03	6.61E+00	2.26E-01

Note that the inventory per unit surface area for the containment is identical to the values used in Attachment C and D. The total inventory is different due to the different volume modeled in these attachments. However, the Surface Area to Volume scaling insures that the initial concentration in Containment is the same as in Attachments C and D.

E 3.0 Results

Transport parameters such as the distribution coefficient, porosity, and bulk density are identical to those used in the main body of the report. Using the inventory values in Table 2 and the geometry values in Table 1 and simulating instant release in Containment Steam Tunnels the peak concentration at the centerline of the Containment and Steam Tunnels was calculated. The concentration at a hypothetical well located 2 m from the opening in the Steam Tunnel was calculated. The ratio of the peak concentration in the well to that in the Containment Building is also supplied. This give a measure of the amount of decay that occurs before reaching the well. These values are provided in Table 3. The time of the peak concentration was also recorded, Table 4.

Examining Table 3 and comparing the peak concentration values in the Containment Building to those in the simulation in Attachment D show they are similar to within the third significant figure. The difference is numerical error due to discretization of the modeled domain. This indicates that the source was selected properly and the results can be compared to those in Attachment D. Due

to the high flow rate and short distance (23 meters to the well from the edge of the Containment Building) several of the contaminants (H-3, Ni-63, Sr-90, and Cs-137) reach the well with at least 60% of the value of the peak concentration in the containment building. For the shorter half-life, less mobile nuclides the amount of decay is more substantial and lower amounts reach the well.

Table E-3 Peak Concentration in each building and the Well located adjacent to the engineered opening in the Steam Tunnel with source term equal to the Groundwater DCGL

	Peak Concentration (pCi/L)			
	Containment	Steam Tunnel	Well	Ratio Well to Containment
H-3	1.00E-03	8.52E-04	7.63E-04	0.76
Co-60	2.87E-08	2.88E-09	7.29E-10	0.025
Ni-63	2.10E-06	1.82E-06	1.67E-06	0.80
Sr-90	2.17E-05	2.09E-05	2.01E-05	0.93
Cs-134	7.61E-07	6.63E-08	1.03E-08	0.014
Cs-137	2.73E-06	2.03E-06	1.70E-06	0.62
Eu-152	1.18E-07	4.26E-08	2.09E-08	0.18
Eu-154	9.27E-08	2.06E-08	6.97E-09	0.075

Table E-4 Time to peak concentration in each building for each ROC.

	Time to peak (years)		
Nuclide	Containment	Steam Tunnel	Well
H-3	0	0.05	0.08
Co-60	0	0	5.6
Ni-63	0	17.5	26.5
Sr-90	0	0.8	2.1
Cs-134	0	5.2	9.7
Cs-137	0	10.8	17.6
Eu-152	0	15.5	27.5
Eu-154	0	13.2	23.9

Using the Surface Area to Volume scaling gives the same initial concentration in the Steam Tunnel as in the Turbine Building. This is not obvious because the reported values in Attachment D for the Turbine and Attachment E for the Steam Tunnels are influenced by the higher contamination levels upstream that reach these areas. Attachment D simulates the transport of contaminants through the Containment and Auxiliary before reaching the Turbine Building. The higher contamination level and therefore higher groundwater concentrations from these upstream buildings lead to peak concentrations occurring later in time and at higher values than immediately after the instant release. Comparison of the initial concentration in the Turbine Building after the instant release and prior to being impacted by upstream buildings and in the Steam Tunnels shows

do match, Table 5. Note that the initial concentrations in Table 5 are much lower than the peak concentrations for everything except Co-60 for which they are the same. The high Kd value and short-half-life prevent appreciable quantities of Co-60 from reaching the middle of the steam tunnel.

Table E-5 Comparison of Time = 0 Concentrations in the Steam Tunnel (Attachment E) and Turbine (Attachment D).

	Steam Tunnel (pCi/L)	Turbine (pCi/L)
H-3	3.04E-06	3.04E-06
Co-60	2.27E-09	2.27E-09
Ni-63	8.15E-09	8.15E-09
Sr-90	2.05E-07	2.05E-07
Cs-134	1.12E-08	1.12E-08
Cs-137	1.12E-08	1.12E-08
Eu-152	5.32E-09	5.32E-09
Eu-154	5.32E-09	5.32E-09

A non-intuitive result of this calculation is that the ratio of the peak tritium concentration in the well to that in Containment is lower than that for Sr-90 or Ni-63, Table 3. The cause for this is the interplay between dispersion and sorption. For H-3, no sorption, the inventory is released instantly and as water flows through porous media, mechanical dispersion occurs due to the differing flow paths that are present in the porous media (e.g. backfill). The high velocity in this simulation causes the H-3 to spread out in the direction of flow reducing the peak concentration. For sorbing nuclides (e.g. Sr-90 and Ni-63) the effects of dispersion are greatly reduced. The majority of contamination (>90%) is sorbed on the backfill for these contaminants. As water moves through the contaminated region in the Containment building the residual contamination on the backfill is released to maintain the equilibrium sorption ratio. Thus, there is a long gradual release that keeps the source region concentration near it's original value and minimizes the effects of dispersion. It is important to recognize that although the peak concentration of H-3 is lowered by dispersion, due to the high flow rate, ~100% of the tritium inventory passes the well location. Similarly, ~ 97.5% of the Sr-90 inventory reaches the well due to the low Kd value (2.3 cm^3/g) In contrast, Ni-63 has a higher Kd (62 cm^3/g) and only 50% of the inventory reached the well location in thirty years.

The time to reach the well is relatively quick with the peak tritium concentration occurring 0.08 years after the release. In general, there will be a few years delay before flow goes out of the openings in the steam tunnel as it will take time to fill the pore space in the remaining backfilled structures. This will allow time for decay that is important for the shorter-lived nuclides.

Even if the Turbine Building and therefore Steam Tunnel was contaminated to it's DCGL, the contamination reaching the well would be slightly lower than the DCGL level. This is demonstrated to changing the inventory to the DCGL level in the Steam Tunnel in the calculation.

Scaling the inventory to a value of 1 pCi/m^2 in the Auxiliary Building for consistency with the other calculations in Attachments C and D requires taking the DCGL ratio in the Turbine Building to that in the Auxiliary Building. This calculation was performed and the resulting contamination levels are in Table 6. The total inventory in the modeled region of the Containment Building and the Steam Tunnel are also provided. The Containment Inventory is identical to the values found in Table 2.

Table E-6 Inventory when the modeled region of the Containment Building and the Steam Tunnel are at their DCGL levels.

Nuclide	Containment pCi/m^2	Steam Tunnel pCi/m^2	Containment pCi	Steam Tunnel pCi
H-3	4.74E-01	2.62E-01	1.26E+02	5.91E+01
Co-60	1.82E-02	1.00E-02	4.81E+00	2.26E+00
Ni-63	3.69E-01	2.04E-01	9.78E+01	4.61E+01
Sr-90	1.51E-01	8.35E-02	4.01E+01	1.88E+01
Cs-134	9.74E-02	5.38E-02	2.58E+01	1.21E+01
Cs-137	3.49E-01	1.93E-01	9.24E+01	4.35E+01
Eu-152	3.19E-02	1.76E-02	8.45E+00	3.98E+00
Eu-154	2.50E-02	1.38E-02	6.62E+00	3.12E+00

Using the inventory values in Table 6 the peak concentrations in the Containment Building, Steam Tunnel, and well located 2 meters outside the steam tunnel were calculated. The time to reach the peak concentration at the well was also calculated. These values and the ratio of the peak concentration in the well to that in the Steam Tummel are provided in Table 7. The analysis shows that the peak concentration at the well will be above 80% of the values in the Steam Tunnel for H-3, Sr-90, Ni-63, and Cs-137. The other 4 contaminants (Co-60, Cs-134, Eu-152, and Eu-154) all show some radioactive decay prior to reaching the well and a peak concentration of 25 to 61% of the peak in the steam tunnel.

Table E-7 Peak Concentrations in the case where both Containment and the Steam Tunnel are contaminated to their DCGL values.

	Peak Concentration (pCi/L)				
	Containment	Steam Tunnel	Well	Ratio Well to Steam Tunnel	Time to peak (years)
H-3	1.00E-03	1.00E-03	9.79E-04	0.98	0.05
Co-60	2.88E-08	2.89E-08	7.31E-09	0.25	5.45
Ni-63	2.10E-06	2.10E-06	1.93E-06	0.92	9.90
Sr-90	2.17E-05	2.17E-05	2.13E-05	0.98	0.65
Cs-134	7.62E-07	7.65E-07	2.91E-07	0.38	1.60
Cs-137	2.73E-06	2.73E-06	2.30E-06	0.84	5.80
Eu-152	1.19E-07	1.19E-07	7.26E-08	0.61	6.10
Eu-154	9.29E-08	9.31E-08	4.77E-08	0.51	4.65

E 4.0 Conclusion

A simulation of transport from the Containment through the Steam Tunnels to an engineered opening for drainage was performed. A location 2 m outside of the opening was selected as the nearest point a well could be placed. The inventory of the Containment Building was set to the ratio of the DCGL levels in the Containment and Auxiliary Buildings multiplied by the contamination level in the Auxiliary Building, identical to the approach in Attachment C and D. The contamination levels in the Steam Tunnels are assumed to be the same as the Turbine Building. The Steam Tunnel contamination level was therefore set to 0.001 pCi/m^2. The simulation conservatively assumed that all of the recharge water from all of the area of the buildings in the main Complex (Containment, Auxiliary Buildings, Turbine Building and other structures) flows out of one opening in the Steam Tunnels. The results showed that for mobile nuclides (H-3 and Sr-90) and for the long-lived nuclides (Ni-63 and Cs-137) the peak concentration could be within 60% of the Contaminant Building peak concentration at the well. The other nuclides (Co-60, Cs-134, Eu-152, and Eu-154) had peak well concentrations between 1 and 18% of their value in the Containment Building.

Attachment F TSD 14-009. Release From Grouted Sumps

F.1 Conceptual Model

The current plan at Zion includes leaving embedded piping in place and grouting these pipes. There is a concern over radioactive contamination traveling through embedded piping into the sumps in which they terminate. All embedded piping has been grouted and it is expected that the first several feet of the grout is not contaminated. To examine the potential release from this pathway into the sump, the maximum release from this condition will be simulated by placing the entire inventory of the pipe at the edge of the clean grout (initially uncontaminated zone) and simulating the diffusion up to the edge of the clean zone. Simulations were performed with 4 feet (122 cm) of uncontaminated grout, the expected condition, 2 feet (61 cm), and 1 foot (30.5 cm) of clean grout. The output from the model is the peak concentration, peak flux, and total mass release out of the concrete.

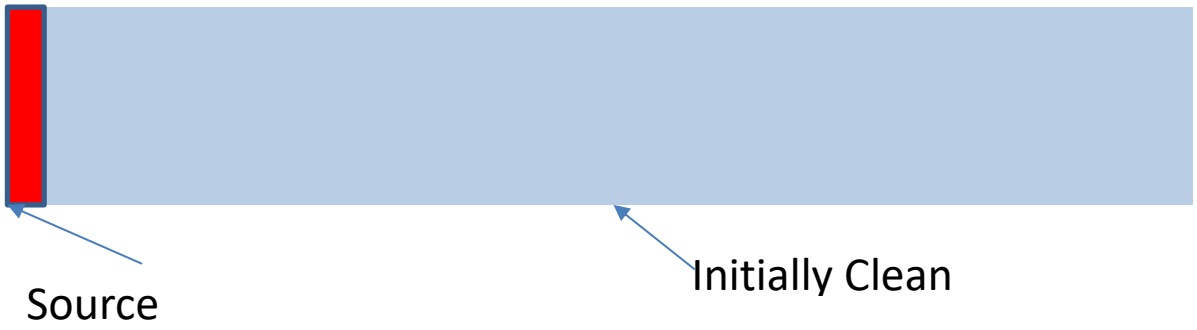

Figure F- 1 Conceptual model for diffusion release through uncontaminated grout.

The eight Radionuclides of Concern (ROC) (H-3, Co-60, Ni-63, Sr-90, Cs-134, Cs-137, Eu-152 and Eu-154) were modeled.

F.2 Input Parameters

Source Term: An "instantaneous" release of inventory will be placed in the first computational cell in the DUST-MS model. This computational cell is a boundary cell that will specify zero mass flux out of the boundary. This will force all the contamination to migrate up through the initially clean grout. The DUST-MS model for a source is controlled through an input table of release rate ($Ci/cm^3/s$) versus time. To simulate an "instantaneous" release, a release rate is specified for the first 0.1 years and zero release after that time. This 0.1-year time for release is close enough to instantaneous to not significantly impact the results due to the long transport time to reach (years) the edge of the clean grout by diffusion.

Pipe Diameter: The pipe diameter influences the peak concentration in the simulation. For a fixed release rate, the smaller the diameter the larger the peak concentration. However, it does not impact the total mass released as this is a function of area over which release occurs. A pipe diameter of 4 inches (10.16 cm) was selected for this simulation. This has an area of 81.1 cm^2.

Mass release: All nuclides were simulated to release a total of 1 mCi of inventory.

Diffusion Coefficients: The diffusion coefficients were identical to the values used to simulate release from contaminated floors via diffusion. These values were discussed in section 2.3.2, Table 4 and are provided here as Table C-1. These values are measured values and include the effects of tortuosity and sorption. For this reason, sorption is not simulated in the model and all partition coefficient values are set to 0.

Table F- 1 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

This simulation is a diffusion problem and water flow is set to zero.

The moisture content was set to 0.25 in the grout. The peak concentrations are inversely proportional to moisture content. Thus, if the moisture content was 0.125, the predicted peak concentration would double. However, the total mass released is not impacted by moisture content. For example, if the actual moisture content was 0.125 the peak concentration would double, but there would be 50% less water to migrate through. Thus, the total amount released would remain the same.

The bulk density of the grout does not impact the results because of the absence of sorption. A nominal value of 2.25 g/cm^3 was used.

F.3 Results

The simulation was conducted for 900 years. Tables C-2 through C-4 present the peak concentration, peak flux, peak mass release rate (peak flux multiplied by the modeled area - 81.1 cm^2 for a 4-inch pipe), the total mass release rate and the fractional release (total mass release divided by initial inventory) for four feet, two feet, and one feet of clean grout at the edge of the sump. Although non-zero numbers are calculated for this simulation, if the value was less than 1E-30 it is reported as <1E-30 in the tables that follow.

Table F- 2 Release from 4 feet of clean grout.

	Peak Concentration (pCi/L)	Peak Flux $Ci/cm^2/yr$	Peak Mass Release Rate mCi/yr	Total Mass Release Ci	Fractional release
H-3	1.58E-01	4.29E-10	3.48E-05	1.77E-06	1.77E-03
Co-60	<1E-30	<1E-30		<1E-30	<1E-30
Ni-63	<1E-30	<1E-30		<1E-30	<1E-30
Sr-90	<1E-30	<1E-30		<1E-30	<1E-30
Cs-134	<1E-30	<1E-30		<1E-30	<1E-30
Cs-137	4.14E-25	7.65E-36	6.20E-31	2.31E-32	2.31E-29
Eu-152	<1E-30	<1E-30		<1E-30	<1E-30
Eu-154	<1E-30	<1E-30		<1E-30	<1E-30

Table F- 3 Release from 2 feet of clean grout.

	Peak Concentration (pCi/L)	Peak Flux $Ci/cm^2/yr$	Peak Mass Release Rate mCi/yr	Total Mass Release Ci	Fractional release
H-3	4.38E+00	2.38E-08	1.93E-03	5.93E-05	5.93E-02
Co-60	<1E-30	<1E-30		<1E-30	<1E-30
Ni-63	1.01E-12	1.23E-23	9.98E-19	3.68E-20	3.68E-17
Sr-90	<1E-30	<1E-30		<1E-30	<1E-30
Cs-134	<1E-30	<1E-30		<1E-30	<1E-30
Cs-137	1.71E-10	5.46E-21	4.43E-16	1.24E-16	1.24E-13
Eu-152	<1E-30	<1E-30		<1E-30	<1E-30
Eu-154	<1E-30	<1E-30		<1E-30	<1E-30

Table F- 4 Release from 1 foot of clean grout.

	Peak Concentration (pCi/L)	Peak Flux Ci/cm^2/yr	Peak Mass Release Rate mCi/yr	Total Mass Release Ci	Fractional release
H-3	2.39E+01	2.60E-07	2.11E-02	3.35E-04	0.34
Co-60	<1E-30	<1E-30		<1E-30	<1E-30
Ni-63	1.86E-03	4.06E-14	3.29E-09	8.28E-10	8.28E-07
Sr-90	5.22E-13	5.62E-24	4.56E-19	1.19E-19	1.19E-16
Cs-134	8.88E-22	5.88E-32	4.77E-27	1.38E-28	1.38E-25
Cs-137	5.19E-04	3.10E-14	2.51E-09	4.98E-10	4.98E-07
Eu-152	<1E-30	<1E-30		<1E-30	<1E-30
Eu-154	<1E-30	<1E-30		<1E-30	<1E-30

Reviewing the fractional release values in Tables C-2 to C-4 the grout tremendously reduces the release of contaminants. H-3 is the only nuclide that shows appreciable release, with a fractional release of 0.34 at 1 foot, 0.06 at two feet and 0.002 at four feet of grout. Co-60, Eu-152, and Eu-154 show extremely small fractional release (< 1E-30) even at one foot of grout. Sr-90 also show a low fractional release of 1.2E-16 with one foot of grout and less than 1E-30 at two and four feet of grout. The fractional release rate of both Cs-137 and Ni-63 is below 1E-6 at 1 foot and decreases by another six orders of magnitude for two feet of grout.

Even though 34% of the H-3 inventory is released when there is one foot of clean grout, the diffusion process spreads the release out over time as the highest annual release rate is only 2.1% of the inventory, Table C-4. The simulation shows it requires more than 27 years to release 30% of the inventory from the grout. The cumulative mass release of H-3 is shown in Figure C-2. The peak concentration for H-3 being released from the grout is 23.9 pCi/L for a one mCi source.

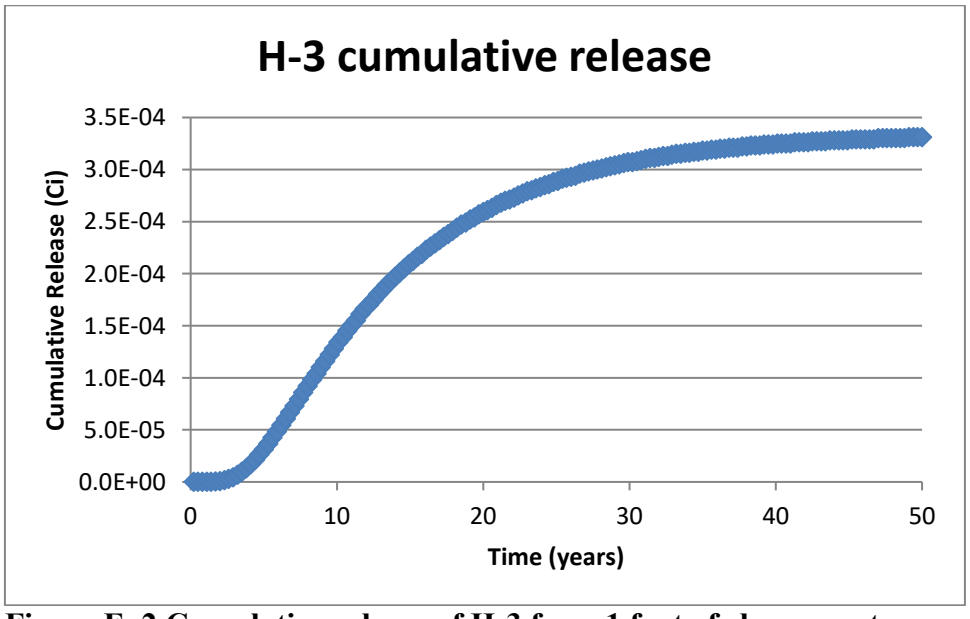

Figure F- 2 Cumulative release of H-3 from 1 foot of clean grout.

F.4. Conclusion

Grouting of the last several feet of pipe before entering the sump is an effective means of reducing release from these pipes into the sump drain area. Although it is possible to adjust the input parameters slightly (source duration, pipe diameter, and moisture content) to give a slightly different peak concentration, the cumulative mass release is not sensitive to these parameters and would change slightly (source duration) to not at all (pipe diameter and moisture content). Adjusting these parameters will not change the conclusion that one foot of clean grout controls the release of the inventory of all ROCs to less than one part in a million, except for tritium. The cumulative mass of tritium released is predicted to be 34% of the inventory. However, diffusion spreads the release over time and the peak concentration being released is less than 25 pCi/L. If the expected condition of four feet of clean grout is achieved, the release of tritium will be controlled to less than 0.2% of the initial inventory.

Attachment G to TSD 14-009 Comparison of Diffusion and Instant Release Models for the Auxiliary Building

Introduction

The approach used to include the potential contamination of underground pipes and penetrations that connect two buildings, for example the Containment and Auxiliary buildings, is to include the inventory of these pipes in both buildings when setting Derived Concentration Guideline Levels (DCGLs). An issue arises when adding inventory to the Auxiliary Building. The Auxiliary Building is assumed to have contamination distributed through the first half-inch of concrete. This inventory is released through diffusion. The connecting pipes are expected to have very little contamination and the contamination they do have is likely on the surface. They will be modeled by an instant release. The diffusion release model predicts that some of the contamination will decay in the concrete prior to release. Therefore, an adjustment is necessary for the inventory that is released from the connecting pipes and included in the Auxiliary Building inventory. The required adjustment is the scaling of the inventory in the pipe such that the peak predicted concentration in the Auxiliary Building accounts for this instant release that occurs over the limited area of the pipe. This is conservative as the peak concentration arising from diffusion release will occur at a different time than the instant release.

Modeling Approach

The peak concentration in the Auxiliary Building was calculated using the Diffusion Release model and the Instant Release Model. All other parameters were the same as in Attachment D. The results of the calculation are provided in Table 1. The concentrations for the Diffusion Release are identical to those presented in Attachment D. The ratio of Instant to Diffusion controlled release provides a scaling factor for releases from the pipe and penetrations that connect to the Auxiliary Building.

Table G- 1 Peak Groundwater Concentration for Diffusion and Instant Release Models.

Nuclide	Diffusion pCi/L	Instant pCi/L	Ratio
H-3	1.01E-03	1.02E-03	1.01
Co-60	2.89E-08	7.58E-07	26.23
Ni-63	2.11E-06	2.73E-06	1.29
Sr-90	2.18E-05	6.87E-05	3.15
Cs-134	7.63E-07	3.75E-06	4.91
Cs-137	2.74E-06	3.75E-06	1.37
Eu-152	1.19E-07	1.78E-06	14.96
Eu-154	9.32E-08	1.78E-06	19.10

Attachment H to TSD 14-009 Check Calculation Discharge Tunnel Mixing Bath Model

H 1) Introduction

The original mixing bath model analysis in the main body of this report did not address the Discharge Tunnels. The Discharge Tunnels run from beneath the Turbine Building and ultimately connect to the lake. The Discharge Tunnels are isolated from the Turbine Building by one inch thick steel plate caps welded onto the openings of the remaining portions of the pipe that open to the tunnels that are designed to be water tight. The east ends of the Discharge Tunnels are isolated from the lake by a butterfly valve as well as an additional barrier that will be installed immediately to the west of the butterfly valve. Well placement directly within the Discharge Tunnels is considered implausible for two reasons. First, the drill would need to penetrate either the one inch thick steel plate or the 2.5 feet thick, reinforced concrete of the Discharge Tunnel ceiling. Second, without hydraulic connection to the Turbine Building, the Discharge Tunnels would not have the flow to support a water well if the drill were to penetrate the highly resistant barriers. The modeling approach to determine DCGL levels in the Turbine Building was to add the source term from the Discharge Tunnels into the Turbine Building as a part of the DCGL calculations. This section provides a check calculation to apply if necessary to provide a bounding calculation for a water well drilled into the Discharge Tunnel in the as-left, isolated configuration.

H 2) Modeling Approach and Results

To examine the differences in predicted concentration for a unitized source term of 1 pCi/m^2 a mixing bath model was developed for the Discharge Tunnels and is compared to the values resulting from the analysis of the Turbine Building. The peak concentration in the Discharge Tunnel was calculated with the same parameters as in the main body of the report, (Table 5 Kd values, porosity =0.25, fill bulk density = 1.5 g/cm^3). The geometry of the discharge tunnels is complex with several changes of elevation. The total volume of the discharge tunnel is 2992 m^3 (Table 60 TSD 14-014). The tunnel was modeled as being 95.9 m long with a cross sectional area 31.2 m^2. The surface area of the discharge tunnels is 2434 m^2 (Table 60, TSD 14-014). Simulating a unitized concentration of 1 pCi/m^2 provides a total inventory of 2434 pCi. The groundwater concentration results of the calculation are provided in Table 1 and compared to the Turbine Building. The ratio of the peak concentration in the Turbine Building to the Discharge Tunnel is also provided in Table 1. As expected, it is constant and equal to the surface area to volume ratio between these two buildings (Turbine Building SA/V = 0.96; Discharge Tunnels SA/V = 0.81).

Table H- 1 Peak Groundwater Concentration for the Discharge Tunnel and Turbine Building.

Nuclide	Kd cm^3/g	Discharge Tunnel pCi/L	Turbine pCi/L	Ratio Discharge to Turbine
H-3	0	3.25E-03	3.86E-03	0.84
Co-60	223	2.43E-06	2.88E-06	0.84
Ni-63	62	8.72E-06	1.03E-05	0.84
Sr-90	2.3	2.20E-04	2.61E-04	0.84
Cs-134	45	1.20E-05	1.42E-05	0.84
Cs-137	45	1.20E-05	1.42E-05	0.84
Eu-152	95	5.70E-06	6.74E-06	0.85
Eu-154	95	5.70E-06	6.74E-06	0.85

Discharge Tunnel in As-Left Configuration

The Discharge Tunnels are to be left in place without backfill. But the as-left condition includes six inches of sediment/sludge on the bottom of the tunnel. In this case, the amount of material for sorption is reduced from that used in the base case calculation above by the ratio of the thickness of the sludge to the total thickness (0.5 ft/16 ft). This lowers the effective density from the base case value of 1.5 g/cm^3 to 0.047 g/cm^3. This also changes the water content from the base case value of 0.25 to 0.968. A simulation was conducted with all other parameters unchanged. The peak concentrations are in Table H-3.

Table H-3 Peak concentrations in the groundwater for the discharge tunnel in the as-left condition.

	Half-life (years)	Kd (cm^3/g)	Peak Concentrations As-Left Discharge Tunnel Water (pCi/L)
H-3	12.4	0	8.40E-04
Co-60	5.27	223	2.88E-06
Ni-63	96	62	2.10E-04
Sr-90	29.1	2.3	7.56E-04
Cs-134	2.06	45	2.64E-04
Cs-137	30	45	2.64E-04
Eu-152	13.3	95	1.50E-04
Eu-154	8.8	95	1.50E-04

To interpret the significance of the differences in the context of this bounding check calculation, Table H-4 provides a comparison to the base case Turbine concentrations. As expected, the water concentrations are higher for all radionuclides other than H-3 when there is less material present for sorption.

As a point of reference, the ratios in table H-4 can be interpreted by using them to calculate a "Bounding Concentration" (pCi/m^2) for Discharge Tunnels in the isolated, stand-alone configuration, that would result in 25 mrem/yr dose; a) if a well could be drilled into the structure; and b) if there were a sufficient water source in the discharge tunnels alone to support a water well (which are both considered implausible assumptions). This value is calculated using the Turbine $DCGL_B$ values from TSD 14-010 (which are conservative for application to the Turbine Building alone due to are adjustments in the $DCGL_B$ calculation). The Discharge Tunnel Bounding Concentrations are calculated by dividing the Turbine $DCGL_B$ values by the water concentration ratios in column 3 of Table H-4

The bounding calculations can be interpreted by comparison to the FSS results for the Discharge Tunnels provided in Yetter, 2017. As seen in Table H-5, the mean FSS results are a small fraction of the Bounding Concentrations in Table H-4.

Table H- 4 Comparison of water concentrations for discharge tunnel as-left configuration to turbine base case and calculation of Bounding Concentration.

	As-Left Discharge Tunnel (pCi/L)	Base Case Turbine Water (pCi/L)	Water Ratios	Turbine-specific $DCGL_B$ pCi/m^2	Isolated Discharge Tunnel Bounding Concentrations pCi/m^2
H-3	8.40E-04	3.86E-03	0.22	1.36E+08	6.26E+08
Co-60	7.12E-05	2.88E-06	24.72	7.43E+07	3.01E+06
Ni-63	2.10E-04	1.03E-05	20.39	2.31E+09	1.13E+08
Sr-90	7.56E-04	2.61E-04	2.90	8.18E+05	2.82E+05
Cs-134	2.64E-04	1.42E-05	18.59	1.68E+07	9.06E+05
Cs-137	2.64E-04	1.42E-05	18.59	2.23E+07	1.20E+06
Eu-152	1.50E-04	6.74E-06	22.26	1.71E+08	7.68E+06
Eu-154	1.50E-04	6.74E-06	22.26	1.51E+08	6.78E+06

Table H-5 Interpretation of Isolated Discharge Tunnel Bounding Concentrations

	Discharge Tunnel Mean FSS Results pCi/m^2	Fraction of Bounding Concentration
Co-60	1.65E+05	5.48E-02
Ni-63	1.41E+04	2.06E-02
Sr-90	2.98E+01	3.74E-02
Cs-134	1.87E+04	1.25E-04
Cs-137	4.48E+04	1.06E-04

References

TSD 14-014 "End State Surface Areas, Volumes, and Source Terms of Ancillary Buildings, Revision 1", June 24, 2015.

Bob Yetter email to Terry Sullivan, "FSS ISOCS Results", February 9, 2017.

Attachment I to TSD 14-009. Flow between buildings Containment to Crib House using total DCGL scaling.

I.1 Introduction

The mixing bath model based on no flow between buildings was developed to provide an upper bound on peak concentrations in each building. An alternate scenario would allow transport of radionuclides between buildings which is expected to be minimal given that the only source of water to the basements is rainwater infiltration. Attachment D addressed transport from the Containment Building to the Crib House and Attachment E addressed transport from the Containment Building to the Steam Tunnel and then a well at the location of engineered openings in the steam tunnel.

The DUST-MS model is a groundwater concentration model. For that reason, the objective of these two calculations was to demonstrate that the transport of contaminants between buildings would not lead to groundwater concentrations that would exceed the groundwater DCGL. The simulation used a normalized to 1 pCi/m^2 to represent the DCGL for each nuclide in the Auxiliary Building. Existing Final Status Survey (FSS) data has shown that the contamination level in the already backfilled Turbine Building and Crib House are less than 0.1% of the Auxiliary Building DCGL for all radionuclides. Therefore, the contamination level in these buildings was set to 0.001 pCi/m^2. The contamination in the Containment Building is expected to be low after removal of shielding and other equipment in the building but the activity in the Containment is also assumed to be at the DCGL (normalized as described in Attachment D and E).

However, the groundwater pathway is not the entire contributor to dose. There is drilling spoils pathway that contributes to the total dose used to assess Basement Derived Concentration Guideline Levels ($DCGL_B$). A calculation was performed for each of the four major basement structures (Containment, Auxiliary, Turbine, and Crib House/Forebay) to calculate the maximum concentration that would occur for the no flow mixing bath model with an inventory of the $DCGL_B$ value uniformly distributed on the walls and floors. These values are compared to the two transport cases simulated in Attachment D and E with the inventory scaled to $DCGL_B$ values as opposed to groundwater DCGL values used in Attachments D and E.

I-2 Modeling Overview

The calculations in this Attachment are identical to those in Attachment D and E with the exception that the total $DCGL_B$ value that includes both exposure scenarios (groundwater and drilling spoils) is used to scale the concentrations. As in Attachments D and E, for the transport simulations, like those found in Attachment D, the Auxiliary Building is scaled to 1 pCi/m^2 for each radionuclide and 0.001 pCi/m^2 for the Turbine and Crib House Buildings. For the Containment Building the inventory is scaled to the Auxiliary Building using the ratio of the total $DCGL_B$ for the Containment and Auxiliary Buildings.

For the no flow mixing bath scenario, the same geometry used for the transport simulations in Attachment D is used. The inventory is set to 1 pCi/m^2 when simulating the Auxiliary Building. The simulated inventory in other buildings are scaled by the ratio of the Building $DCGL_B$ to the

Auxiliary Building $DCGL_B$ value. When simulating a single building, the inventory is non-zero for one building and zero for all other buildings. For example, when examining the peak concentration in the Turbine Building when the surface contamination is at the $DCGL_B$ value in the Turbine Building uses zero inventory in the Containment, Auxiliary Building, and Crib House/Forebay Building for the simulation.

The geometry of the model representing transport from the Containment to the Crib House is specified in Tables D-1 through D-3. The geometry of the model representing transport from the Containment to the Steam Tunnel is specified in Table E.1. All transport parameters are similar to the base case and found in the main report. As before, the Auxiliary Building is modeled using diffusion release from contaminated concrete. Contamination in other buildings are simulated using an instant release of all contamination. The difference between Attachments D and E, and Attachment I is the inventory in the Containment.

I-2.1 Individual Building Inventory

As a basis for comparison the individual building inventory per unit area normalized to 1 pCi/m^2 in the Auxiliary Building is presented in Table I-1. These values are individually used to assess the peak concentration that would occur in each building at the normalized $DCGL_B$ inventory per unit area.

Table I-I Inventory per unit area (pCi/m^2) and for the entire building (pCi) with surface contamination at the $DCGL_B$ level normalized to 1 pCi/m^2 in the Auxiliary Building..

Nuclide	Auxiliary	Containment	Turbine	Crib House/ Forebay	Auxiliary	Contain-ment	Turbine	Crib House/ Forebay
	pCi/m^2	pCi/m^2	pCi/m^2	pCi/m^2	pCi	pCi	pCi	pCi
H-3	1.00E+00	4.49E-01	2.43E-01	3.56E-01	7226	1564	6124	6647
Co-60	1.00E+00	5.17E-01	2.31E-01	1.78E-01	7226	1802	5830	3319
Ni-63	1.00E+00	3.50E-01	1.90E-01	2.76E-01	7226	1218	4780	5164
Sr-90	1.00E+00	1.43E-01	7.75E-02	1.13E-01	7226	499	1953	2117
Cs-134	1.00E+00	1.42E-01	7.54E-02	9.86E-02	7226	496	1899	1842
Cs-137	1.00E+00	3.55E-01	1.90E-01	2.60E-01	7226	1235	4776	4858
Eu-152	1.00E+00	5.66E-01	2.50E-01	1.86E-01	7226	1972	6296	3483
Eu-154	1.00E+00	5.47E-01	2.45E-01	1.88E-01	7226	1903	6159	3511

I-2.2 Transport to the Crib House Modeled Inventory

As stated in the introduction, the Auxiliary Building inventory is normalized to 1 pCi/m^2 for each nuclide. Similarly, the Turbine and Crib House are normalized to 0.001 pCi/m^2 based on FSS data (Yetter, 2017). The Containment building is normalized to the $DCGL_B$ ratio of the Containment

Building to the Auxiliary Building for all pathways (TSD 14-010). The resulting inventory per unit area and total inventory are provided in Table I-2.

Table I-2 Inventory per unit area (pCi/m^2) and for the entire building (pCi)

Nuclide	Auxiliary	Containment	Turbine	Crib House/ Forebay	Auxiliary	Contain-ment	Turbine	Crib House/ Forebay
	pCi/m^2	pCi/m^2	pCi/m^2	pCi/m^2	pCi	pCi	pCi	pCi
H-3	1.00E+00	4.49E-01	1.00E-03	1.00E-03	3713	1564	13.0	13.6
Co-60	1.00E+00	5.17E-01	1.00E-03	1.00E-03	3713	1218	13.0	13.6
Ni-63	1.00E+00	3.50E-01	1.00E-03	1.00E-03	3713	499	13.0	13.6
Sr-90	1.00E+00	1.43E-01	1.00E-03	1.00E-03	3713	1802	13.0	13.6
Cs-134	1.00E+00	1.42E-01	1.00E-03	1.00E-03	3713	496	13.0	13.6
Cs-137	1.00E+00	3.55E-01	1.00E-03	1.00E-03	3713	1235	13.0	13.6
Eu-152	1.00E+00	5.66E-01	1.00E-03	1.00E-03	3713	1972	13.0	13.6
Eu-154	1.00E+00	5.47E-01	1.00E-03	1.00E-03	3713	1903	13.0	13.6

I-2-3 Modeled Inventory for the Transport through the Steam Tunnel to a Well Scenario

Attachment E modeled the movement of contaminants from the Containment Building through the steam tunnel to engineered openings and a well 2 meter outside of the opening. This simulation is meant to represent the case when the walls of the subsurface structure remain intact. Rainwater that infiltrates into the subsurface eventually fills these subsurface structures and any additional water due to infiltration will flow out of the engineered opening that is lower than the walls of the other subsurface structures and one foot above the water table. Attachment E scaled the Auxiliary Building to 1 pCi/m^2 for each nuclide and the Containment Building to the ratio of the groundwater DCGL value in Containment to that in the Auxiliary Building. This Attachment scales the Containment Building to the ratio of the $DCGL_B$ for Containment to that in the Auxiliary Building. The inventory used in this simulation is in Table I-3. Note that the Steam Tunnel is treated as part of the Turbine Building when generating $DCGL_B$ values. Therefore, it is modeled with the same contamination level (0.001 pCi/m^2) as the Turbine Building.

The total simulated inventory is also presented in Table I-3. Although the surface area contamination level (pCi/m^2) for Containment is identical in Tables I-2 and I-3, the total inventory of Containment is not the same. This is a result of the different geometries modeled. In the Steam Tunnel Model the area for flow is limited to the physical area of the Steam Tunnel. The one-dimensional nature of the model requires that the Containment is modeled with the same flow area. This reduces the modeled volume of Containment. To ensure that the proper groundwater concentration is obtained, the inventory modeled is reduced to make the Surface Area to Volume ratio in the simulation the same.

Table I-3. Inventory per unit area and total inventory using the total $DCGL_B$ for simulating flow through engineered openings in the Steam Tunnel.

Nuclide	Containment pCi/m²	Steam Tunnel pCi/m²	Containment pCi	Steam Tunnel pCi
H-3	4.49E-01	1.00E-03	1.19E+02	2.26E-01
Co-60	5.17E-01	1.00E-03	1.37E+02	2.26E-01
Ni-63	3.50E-01	1.00E-03	9.26E+01	2.26E-01
Sr-90	1.43E-01	1.00E-03	3.80E+01	2.26E-01
Cs-134	1.42E-01	1.00E-03	3.77E+01	2.26E-01
Cs-137	3.55E-01	1.00E-03	9.39E+01	2.26E-01
Eu-152	5.66E-01	1.00E-03	1.50E+02	2.26E-01
Eu-154	5.47E-01	1.00E-03	1.45E+02	2.26E-01

I.3.0 Results

I-3.1 Peak Concentrations for the no flow mixing bath scenario

Using the inventory values in Table I-1 each building was individually simulated without flow to represent the mixing bath model. The peak concentrations are provided in Table I-4. As shown when the inventory is adjusted to the ratio of the building DCGL values, the peak concentration in the Auxiliary and Containment Buildings no longer match as they did in Attachment D. The difference occurs due to the different ratios of Groundwater DCGL and total $DCGL_B$ in these buildings. The building $DCGL_B$ accounts for both scenarios and insignificant contributors.

Table I-4 Peak Concentrations for Individual Buildings with an inventory at the $DCGL_B$ level scaled to 1 pCi/m² in the Auxiliary Building for the mixing bath model.

Nuclide	Containment pCi/L	Auxiliary pCi/L	Turbine pCi/L	Crib House pCi/L
H-3	9.62E-04	1.01E-03	9.35E-04	9.35E-04
Co-60	8.28E-07	2.88E-08	6.98E-08	3.49E-08
Ni-63	2.01E-06	2.11E-06	1.96E-06	1.95E-06
Sr-90	2.08E-05	2.17E-05	2.01E-05	2.01E-05
Cs-134	1.12E-06	7.64E-07	1.07E-06	9.54E-07
Cs-137	2.81E-06	2.75E-06	2.69E-06	2.52E-06
Eu-152	2.13E-06	1.19E-07	1.68E-06	8.59E-07
Eu-154	2.05E-06	9.30E-08	1.65E-06	8.65E-07

I 3.2 Peak Concentrations for the Transport to the Crib House scenario

Using the inventory values in Table I-2, the geometry values in Table D-2, and simulating instant release in Containment, Turbine and Crib House/Forebay buildings and diffusion release in the Auxiliary Building, the peak concentration at the centerline of each Building was calculated, Table I-6. The peak concentrations in the Turbine and Crib House/Forebay buildings are lower for all nuclides and substantially lower for the short-half life nuclides with Kd values greater than 10.

Table I-5 Peak Concentration for each ROC in each Building for transport through the buildings

	Containment (pCi/L)	Auxiliary (pCi/L)	Turbine (pCi/L)	Crib House (pCi/L)
H-3	9.62E-04	9.83E-04	8.45E-04	6.28E-04
Co-60	8.28E-07	2.88E-08	2.88E-09	1.96E-09
Ni-63	2.01E-06	2.12E-06	7.21E-07	1.88E-08
Sr-90	2.08E-05	2.14E-05	1.89E-05	1.25E-05
Cs-134	1.12E-06	7.64E-07	1.42E-08	9.68E-09
Cs-137	2.81E-06	2.76E-06	3.09E-07	9.71E-09
Eu-152	2.13E-06	1.19E-07	6.75E-09	4.61E-09
Eu-154	2.05E-06	9.30E-08	6.75E-09	4.61E-09

Table I-6 provides the ratio of the peak concentration for the case of transport, Table I-5, to that of the individual building mixing bath model, Table I-4. A value greater than 1 suggests that if the model conditions hold, transport could lead to an increase in concentration in a building that exceeds the DCGL level for that building. All ratios are equal to 1 or less indicating that the mixing bath model provides an upper bound on peak concentration.

Table I-6 Ratio of peak centerline concentration in the transport case for all buildings as compared to the mixing bath model for each building.

Nuclide	Containment	Aux	Turbine	Crib House
H-3	1.00	0.97	0.90	0.67
Co-60	1.00	1.00	0.04	0.06
Ni-63	1.00	1.00	0.37	0.010
Sr-90	1.00	0.99	0.94	0.62
Cs-134	1.00	1.00	0.013	0.010
Cs-137	1.00	1.00	0.11	0.004
Eu-152	1.00	1.00	0.004	0.005
Eu-154	1.00	1.00	0.004	0.005

1.3.1 Transport through engineered opening in the Steam Tunnel Wall to a Well

Using the inventory values in Table I-2, the geometry values in Table E-1, and simulating instant release in Containment and Steam Tunnels, the peak concentration at the centerline of the Containment Building, Steam Tunnel, and at the Well was calculated and are provided in Table I-7. The well concentration in Table I-7 is always less than the Containment concentration.

Table I-7 Peak centerline concentrations in Containment, Steam Tunnel and the Well

Nuclide	Containment (pCi/L)	Steam Tunnel (pCi/L)	Well (pCi/L)
H-3	9.62E-04	8.10E-04	7.24E-04
Co-60	8.19E-07	1.86E-08	7.90E-10
Ni-63	1.99E-06	1.72E-06	1.58E-06
Sr-90	2.05E-05	1.98E-05	1.90E-05
Cs-134	1.11E-06	9.29E-08	1.43E-08
Cs-137	2.77E-06	2.05E-06	1.71E-06
Eu-152	2.10E-06	7.40E-07	3.60E-07
Eu-154	2.03E-06	4.30E-07	1.43E-07

Table I-8 provides the ratio of predicted concentrations at the centerline of the Containment Building and the Steam Tunnel for the transport case with scaling the Containment inventory loading (pCi/m^2) to the building $DCGL_B$ (Table I-7) and the no flow mixing bath model (Table I-4). For all nuclides, the ratio is less than one indicating that the no flow mixing bath model provides an upper bound for peak concentration (and dose). The ratio of the Well to Containment Concentration is also included in Table I-8, however there is no mixing bath model equivalent. It is evident form Table I-7 that the groundwater dose from the Well will be less than Containment because the water concentrations are less. The potential for drilling spoils dose from the Well must also be addressed. The depth of the contaminated zone at the well can at most be the combination of the height of the opening above the water table (1 foot) and the height of the well screen (3 m). The depth of contamination in the Containment Building is 4.27 m, which is greater than the depth of contamination near the Well. Therefore, the drilling spoils dose from the Well must also be less than the drilling spoils dose for the Containment Building. The ratio of the Well to Containment can then be used to demonstrate that the no flow mixing bath model for Containment provides an upper bound for peak concentration (and dose) in the Well.

Table I-8 **Ratio of peak concentrations in Containment, Steam Tunnel and Well as compared to the mixing bath model for each building with source term at $DCGL_B$**

Nuclide	Containment	Steam Tunnel	Well[1]
H-3	1.0	0.87	0.75
Co-60	1.0	0.27	0.001
Ni-63	1.0	0.88	0.79
Sr-90	1.0	0.99	0.93
Cs-134	1.0	0.09	0.01
Cs-137	1.0	0.76	0.62
Eu-152	1.0	0.44	0.17
Eu-154	1.0	0.26	0.07

[1] Ratio of Well to Containment Concentration

I 4.0 Conclusion

The peak groundwater concentrations were calculated for the Containment, Auxiliary, Turbine, and Crib House/Forebay structures for the no flow mixing bath model using the $DCGL_B$ values for surface area contamination normalized a value of 1 pCi/m^2 in the Auxiliary Building. The resulting peak concentration values of the no flow mixing bath model represent the groundwater concentration that would lead to a dose of 25 mrem/yr from the groundwater and drilling spoils pathways for that nuclide. To address the impacts of transport from one building to another, two simulations were performed. The first examined transport from the Containment through the Auxiliary and Turbine Buildings to the Crib House/Forebay Building using bounding flow rate assumptions similar to Attachment D but inventory based on $DCGL_B$ values as opposed to groundwater DCGL values used in Attachment D. Another simulation was performed of transport from the Containment Building through the Steam Tunnel and out to a well 2 meters from the engineered opening in the Steam Tunnel, similar to Attachment E with the exeoption of using $DCGL_B$ values. This represents the case when the subsurface walls are intact and flow occurs out the engineered opening in the Steam Tunnel above the water table.

Assuming maximum concentrations in the Containment and the Auxiliary buildings, i.e., at the $DCGL_B$ concentration, the ratio of the peak concentration in the transport simulation to the no-flow mixing bath model is always lower for the downstream buildings, (Steam Tunnel, Turbine, Crib House and the outside well) than for the Containment or Auxiliary building. The ratios of the peak transport simulation concentration to the peak no-flow mixing bath concentration represent peak dose because the denominator (Table I-5 peak no-flow mixing bath peak) represents the concentration corresponding to 25 mrem/yr including both the groundwater and drilling spoils scenario. As seen in Table I-6 and I-8, the peak ratios for the 'downstream' buildings in the flow simulation are always lower than Containment and Auxiliary building ratios. Therefore, the dose calculated for the basement with the maximum concentration of residual radioactivity using the no-flow mixing bath model will be the maximum dose.

References

Bob Yetter email to Terry Sullivan, "FSS ISOCS Results", February 9, 2017.

Fauver, 2017, "RESRAD Dose Modeling for Basement Fill Model, Soil DCGL and Calculation of Basement Fill Model Dose Factors, Revision 4" TSD 14-010-R4, 2017.