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BASEMENT FILL MODEL EVALUATION OF MAXIMUM RADIONUCLIDE CONCENTRATIONS FOR INITIAL SUITE OF RADIONUCLIDES

Zion Station Restoration Project

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Revision 1, May 2016

Informal Report

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Revision Log

Section	Page #	Rev. #	Date	Reason(s) for Revision
Title page		1	5/20/16	Revised Revision number
Title page		1	5/20/16	Revised effective date
3.2.1	12, Table 7	1	5/20/16	Changed the total pCi released for H-3 in Table 7 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.
3.2.3	14, Table 9	1	5/20/16	Fixed typo error for Ni-63 peak concentration value. The correct value, 2.44E-6 pCi/L now matches the value for Ni-59.

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1 Introduction

Zion*Solutions* is in the process of decommissioning the Zion Nuclear Power Plant. The current decommissioning plan involves removing all above grade structures to a depth of 3 feet below grade. The remaining underground structures will be backfilled with clean material. The final selection of fill material has not been made. The remaining backfilled structures will be the two reactor Containment Buildings, a Spent Fuel Building, an Auxiliary Building, a Turbine Building, a Crib House/Forebay Building and a Waste Water Treatment Facility (WWTF).

Remaining structures will contain low amounts of residual radioactive material. The bulk of the source term will be contained in the concrete floors which are twenty to thirty feet below grade in most buildings. Current interior demolition plans are to remove all concrete in the Unit 1 and Unit 2 Reactor Buildings inside the steel liner. 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. Thus the end state source term will be well below grade and below the water table eliminating conventional pathways such as direct radiation and inhalation rendering groundwater related pathways the most significant potential sources of future exposure. Note that the Spent Fuel Pool concrete under the liner is also expected to be significantly contaminated but characterization has not yet been performed due to inaccessibility. The floor of the spent fuel pool is 15 feet below grade.

In order to terminate the Part 50 license, the Zion Solutions Restoration Project (ZSRP) must demonstrate that the dose from remaining residual radioactivity does not cause a hypothetical individual to receive a dose in excess of 25 mrem/y^{-1} as specified in 10 CFR 20 Subpart E. To demonstrate compliance, the modeling of the fate and transport of radioactive material to a receptor is required. For the backfilled basements, this modeling the release of radioactivity from the concrete and mixing with the water contained in the fill material. This report determines the maximum groundwater concentrations in the basement fill for an initial suite of 26 radionuclides designated by the ZSRP as having a potential of being present on the Zion Nuclear Power Station (ZNPS). The results of this analysis are used by ZSRP to determine the relative dose contribution from all 26 radionuclides to identify the “insignificant dose contributors” and select the radionuclides of concern (ROCs) to be included in more detailed calculations.

This report uses the same methods described in detail in the DUST Report (Sullivan, 2014) with the addition of parameters required for the additional radionuclides evaluated in this report. The applicable parts of the conceptual model and input parameter descriptions from the DUST Report are repeated for completeness in order to allow this report to be a stand alone document.

Calculation of the release of radioactive material from the remaining building basements requires site-specific information on the hydrogeologic transport properties (effective porosity, bulk density, hydraulic) and chemical transport properties (sorption). Conestoga-Rovers & Associates (CRA) has collected a substantial amount of site-specific hydrogeologic data (CRA, 2014).

However, this screening calculation estimates only the water concentration in the basement fill. No transport away from the basements is assumed, which would result in lower concentrations.

Brookhaven National Laboratory has determined site-specific sorption data for five nuclides and four soil types, two concrete construction demolition debris, two cinder block materials, and one grout material that are under consideration for the fill (Yim, 2012, Milian, 2014). In addition, sand from the local region could be used as part or all of the backfill. The composition of the fill material has not been finalized.

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 radionuclide concentrations in the fill material and the water in the interstitial spaces of the fill.
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 by ZSRP in selecting ROCs for detailed dose assessment calculations.

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), 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.

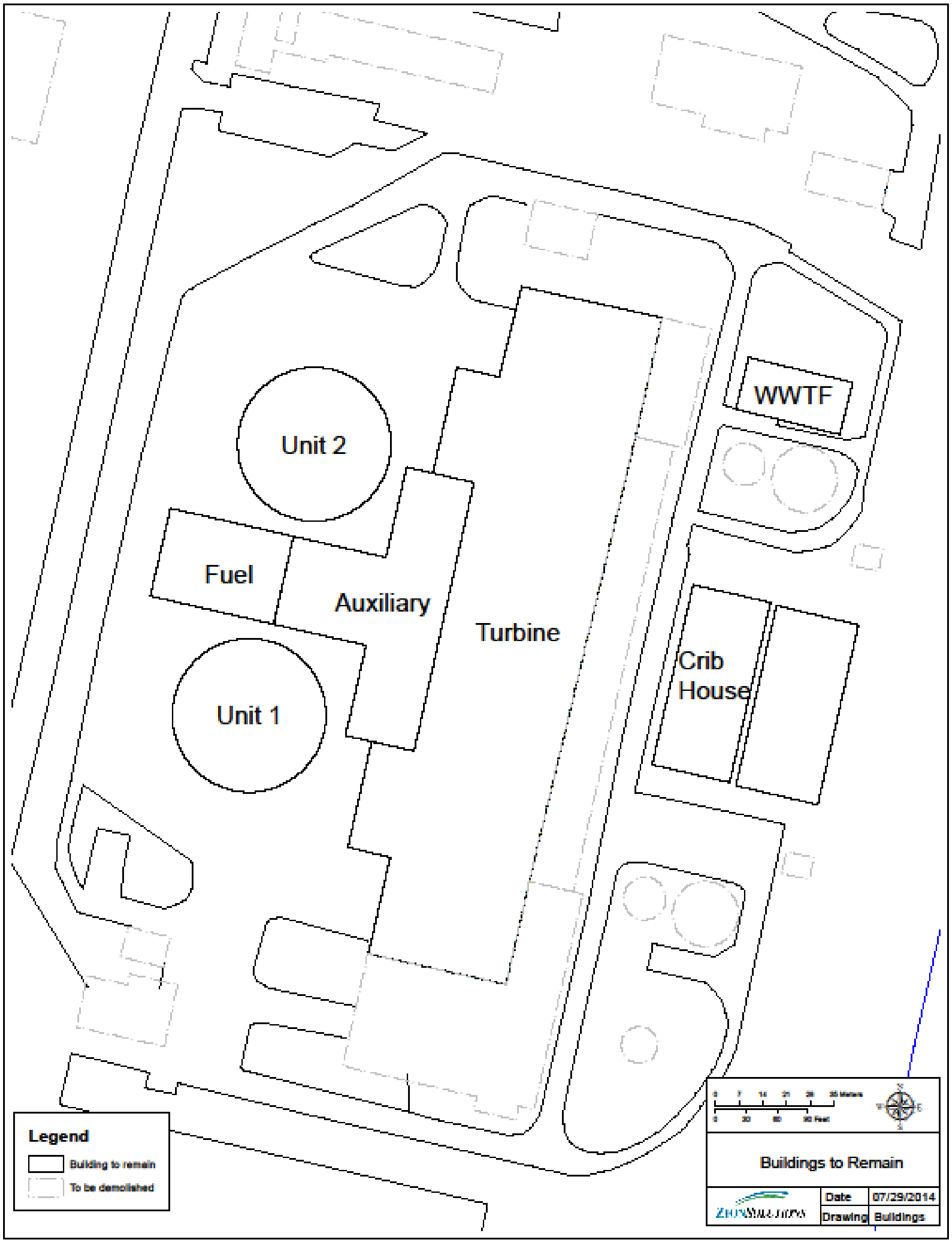

Figure 1 Zion Site layout for buildings that will have a residual underground structure.

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.

The conceptual model for release is important for determining the amount of material in the water and fill. 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.

A second important parameter is the volume of water available to mix with released radionuclides. 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%.

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 modeled as an Instant Release.

Table 1 Mixing volume and release rate assumption

Building	Volume* (m^3)	Release Rate Assumption
Unit 1 Containment	6.537E+03	Instant Release (loose surface contamination)
Unit 2 Containment	6.537E+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)

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 the Instant Release model is used.

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.

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 width was adjusted to match the total volume in Table 1. 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.

The inventory for each building was based on a uniform contamination level for each nuclide 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. Table 2 contains the building surface areas for the calculations of inventory. From Table 2 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.

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

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 potential radionuclides (Table 3). All nuclides in Table 3 were used in the simulation of maximum groundwater concentration.

Table 3. Potential Radionuclides of Concern at the Zion Power Plant

Radionuclides
H-3
C-14
Fe-55
Ni-59
Co-60
Ni-63
Sr-90
Nb-94
Tc-99
Ag-108m
Sb-125
Cs-134
Cs-137
Pm-147
Eu-152
Eu-154
Eu-155
Np-237
Pu-238
Pu-239/240
Pu-241
Am-241
Am-243
Cm-243/244

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

Serne (Serne, 2001) provided a table of best estimates for diffusion coefficients for forty-four elements relevant for nuclear waste disposal including most of the elements listed in Table 3. In general, the recommended diffusion coefficient for most elements was less than the value recommended for Cs. Cs is known to be relatively mobile in cement systems and in fact, with the exception of H-3, has the highest recommended value for diffusion coefficient in Table 4. The elements with diffusion coefficients greater than Cs included elements that tend to form anionic species (such as Tc and I). Based on this information, all elements in Table 3 without a specific diffusion coefficient in Table 4 are set to the value used for Cs with the exception of Tc, which is assigned the same diffusion coefficient as H-3. This should provide a conservative upper bound on release of these species from the Auxiliary and Fuel Buildings. For the other buildings the release is instantaneous and the diffusion coefficient does not impact release.

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.

3 Analysis

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. Table 5 also provides the diffusion coefficient used to simulate release from volumetrically contaminated concrete in the Auxiliary Building and the Fuel Building.

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) and diffusion coefficients

Radionuclide	Half Life (years)	Basement Fill K_d to Be Used ml/g	Recommended Diffusion Coefficient (cm^2/s)
H-3	12.4	0	5.5E-7
C-14	5730	1.2	3.0E-9
Fe-55	2.7	2857	3.0E-9
Ni-59	75000	62	1.1E-9
Co-60	5.27	223	4.1E-11
Ni-63	96	62	1.1E-9
Sr-90	29.1	2.3	5.2E-10
Nb-94	20300	45	3.0E-9
Tc-99	213000	0	5.5E-7
Ag-108m	127	27	3.0E-9
Sb-125	2.77	17	3.0E-9
Cs-134	2.06	45	3.0E-9
Cs-137	30	45	3.0E-9
Pm-147	2.62	95	3.0E-9
Eu-152	13.3	96	5.0E-11
Eu-154	8.8	96	5.0E-11
Eu-155	4.96	96	5.0E-11
Np-237	2140000	1	3.0E-9
Pu-238	87.7	174	3.0E-9
Pu-239	24100	174	3.0E-9
Pu-240	65400	174	3.0E-9
Pu-241	14.4	174	3.0E-9
Am-241	432	177	3.0E-9
Am-243	7380	177	3.0E-9
Cm-243	28.5	891	3.0E-9
Cm-244	18.1	891	3.0E-9

3.1.1 Diffusion Controlled Release Model

For the diffusion release model the selected diffusion coefficients were presented in Table 5. 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

3.2 Peak Groundwater Concentration Results

3.2.1 Auxiliary Building

The conceptual model assumes that the inventory is released through diffusion through a one-half inch uniformly contaminated zone on the floor. The nuclides released into the water instantly reach equilibrium with the fill material through the sorption process as controlled by the value of K_d. The results of this model are presented in Table 7. In addition to the maximum groundwater concentration, the table provides the amount of radioactivity (pCi) in solution, the amount sorbed to the solid material (pCi) and the peak concentration sorbed on the soil (pCi/g). The total inventory was 6.503E-09 Ci (6503 pCi) for each nuclide modeled. The volume between the floor and the water table is 28445 m^3, Table 1.

Examining Table 7 the impact of sorption is clear. For example, consider Sr-90 with a Kd of 2.3 ml/g, the solution concentration is less than 3% of the value for Kd = 0. Table 7 shows the impact of diffusion and decay on release as the sum of the radioactivity in the water and soil is often substantially less than the total inventory of 6503 pCi.

Table 7 Peak Groundwater Concentrations (pCi/L) per unit source of 1 pCi/m^2 in the Auxiliary Building

Nuclide	Half-life (years)	K_d (ml/g)	Diffusion Coefficient (cm^2/s)	Peak Concentration pCi/L	Peak Radioactivity in Solution pCi	Peak Radioactivity Sorbed pCi	Peak Sorbed Concentration pCi/g
H-3	12.4	0	5.5E-07	9.10E-04	6467	0	0
C-14	5730	1.2	3.0E-09	1.11E-04	793	5710	1.33E-07
Fe-55	2.7	2857	3.0E-09	1.24E-08	0.09	1519	3.54E-08
Ni-59	75000	62	1.1E-09	2.45E-06	17.4	6512	1.52E-07
Co-60	5.27	223	4.1E-11	2.6E-08	0.2	249	5.80E-09
Ni-63	96	62	1.1E-09	1.90E-06	13.6	5051	1.18E-07
Sr-90	29.1	2.3	5.2E-10	1.96E-05	140.1	1933	4.51E-08
Nb-94	20300	45	3.0E-09	3.38E-06	24	6521	1.52E-07
Tc-99	213000	0	5.5E-07	9.15E-04	6503	0	0
Ag-108m	127	27	3.0E-09	5.23E-06	37	6054	1.41E-07
Sb-125	2.77	17	3.0E-09	2.08E-06	15	1516	3.54E-08
Cs-134	2.06	45	3.0E-09	6.89E-07	5	1329	3.10E-08
Cs-137	30	45	3.0E-09	2.47E-06	17.7	4766	1.11E-07
Pm-147	2.62	95	3.0E-09	3.68E-07	3	1499	3.50E-08
Eu-152	13.3	95	5.0E-11	1.07E-07	0.8	440	1.02E-08
Eu-154	8.8	95	5.0E-11	8.38E-08	0.6	341	7.96E-09
Eu-155	4.96	95	5.0E-11	6.39E-08	0	260	6.07E-09
Np-237	2140000	1	3.0E-09	1.31E-04	936	5616	1.31E-07
Pu-238	87.7	174	3.0E-09	7.84E-07	6	5848	1.36E-07
Pu-239	24100	174	3.0E-09	8.75E-07	6	6527	1.52E-07
Pu-240	65400	174	3.0E-09	8.74E-07	6	6519	1.52E-07
Pu-241	14.4	174	3.0E-09	4.78E-07	3	3566	8.32E-08
Am-241	432	177	3.0E-09	8.42E-07	6	6389	1.49E-07
Am-243	7380	177	3.0E-09	8.60E-07	6	6526	1.52E-07
Cm-243	28.5	889	3.0E-09	1.24E-07	1	4736	1.10E-07
Cm-244	18.11	889	3.0E-09	1.04E-07	1	3973	9.27E-08

3.2.2 Containment Building

The conceptual model assumes that the inventory is released instantly to the water and instantly reaches equilibrium with the fill material through the sorption process as controlled by the value of K_d. The results of this model are presented in Table 8. In addition to the maximum groundwater concentration, the table provides the amount of radioactivity (pCi) in solution, the amount sorbed to the solid material (pCi) and the peak concentration sorbed on the soil (pCi/g). The total inventory for each modeled nuclide in the Containment Building assuming a contamination level of 1 pCi/m^2 on the floors and walls is 2759 pCi. The volume between the floor and the water table is 6537 m^3, Table 1.

Table 8 Peak Groundwater and Soil Concentrations per unit source of 1 pCi/m^2 in the Containment Building

	Half-life (years)	Kd	Peak Concentration (pCi/L)	Activity in Solution	Activity Sorbed	Peak Concentration (pCi/g)
H-3	12.4	0	1.69E-03	2759	0	0
C-14	5730	1.2	2.06E-04	336.6	2422.4	2.47E-07
Fe-55	2.7	2857	9.82E-08	0.2	2758.8	2.81E-07
Ni-59	75000	62	4.53E-06	7.4	2751.6	2.81E-07
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
Nb-94	20300	45	6.23E-06	10.2	2748.8	2.8E-07
Tc-99	213000	0	1.69E-03	2759	0	0
Ag-108m	127	27	1.04E-05	16.9	2742.1	2.8E-07
Sb-125	2.77	17	1.64E-05	26.7	2732.3	2.79E-07
Cs-134	2.06	45	6.23E-06	10.2	2748.8	2.8E-07
Cs-137	30	45	6.23E-06	10.2	2748.8	2.8E-07
Pm-147	2.62	95	2.95E-06	4.8	2754.2	2.81E-07
Eu-152	13.3	95	2.95E-06	4.8	2754.2	2.81E-07
Eu-154	8.8	95	2.95E-06	4.8	2754.2	2.81E-07
Eu-155	4.96	95	2.95E-06	4.8	2754.2	2.81E-07
Np-237	2140000	1	2.41E-04	394.0	2365.0	2.41E-07
Pu-238	87.7	174	1.62E-06	2.6	2756.4	2.81E-07
Pu-239	24100	174	1.62E-06	2.6	2756.4	2.81E-07
Pu-240	65400	174	1.62E-06	2.6	2756.4	2.81E-07
Pu-241	14.4	174	1.62E-06	2.6	2756.4	2.81E-07
Am-241	432	177	1.59E-06	2.6	2756.4	2.81E-07
Am-243	7380	177	1.59E-06	2.6	2756.4	2.81E-07
Cm-243	28.5	891	3.15E-07	0.5	2758.5	2.81E-07
Cm-244	18.11	891	3.15E-07	0.5	2758.5	2.81E-07

The peak ground water concentrations in the Containment Building are predicted to be greater than in the Auxiliary Building for a unit contamination level of 1 pCi/m^2. This reflects the lower amount of water available for mixing and the higher release rate in the Containment Building. With the exception of the five nuclides (H-3, C-14, Sr-90, Tc-99, and Np-237) that had a distribution coefficient value of less than 3, more than 99% of the activity released was sorbed on the solid backfill. This is the cause for the soil concentrations being similar for most nuclides.

3.2.3 Crib House/Forebay Building

The conceptual model assumes that the inventory is released instantly similar to the Containment Building. The results of this model are presented in Table 9. In addition to the maximum groundwater concentration, the table provides the amount of radioactivity (pCi) in solution, the amount sorbed to the solid material (pCi) and the peak concentration sorbed on the soil (pCi/g). The total inventory for each modeled nuclide in the Containment Building assuming a contamination level of 1 pCi/m^2 on the floors and walls is 6940 pCi. The volume between the floor and the water table is 30524 m^3, Table 1.

Table 9 Peak Groundwater and Soil Concentrations per unit source of 1 pCi/m^2 in the Crib House/Forebay Building

	Half-life (years)	Kd	Peak Concentration (pCi/L)	Activity in Solution	Activity Sorbed	Peak Concentration (pCi/g)
H-3	12.4	0	9.08E-04	6936	0	0
C-14	5730	1.2	1.11E-04	845.8	6094.2	1.33E-07
Fe-55	2.7	2857	5.29E-08	0.4	6939.6	1.52E-07
Ni-59	75000	62	2.44E-06	18.6	6921.4	1.51E-07
Co-60	5.27	223	6.78E-07	5.2	6934.8	1.51E-07
Ni-63	96	62	2.44E-06	14.6	6925.4	1.51E-07
Sr-90	29.1	2.3	6.14E-05	468.5	6471.5	1.41E-07
Nb-94	20300	45	3.35E-06	25.6	6914.4	1.51E-07
Tc-99	213000	0	9.09E-04	6936	0	0
Ag-108m	127	27	5.58E-06	42.5	6897.5	1.51E-07
Sb-125	2.77	17	8.80E-06	67.2	6872.8	1.5E-07
Cs-134	2.06	45	3.35E-06	25.6	6914.4	1.51E-07
Cs-137	30	45	3.35E-06	25.6	6914.4	1.51E-07
Pm-147	2.62	95	1.59E-06	12.1	6927.9	1.51E-07
Eu-152	13.3	95	1.59E-06	12.1	6927.9	1.51E-07
Eu-154	8.8	95	1.59E-06	12.1	6927.9	1.51E-07
Eu-155	4.96	95	1.59E-06	12.1	6927.9	1.51E-07
Np-237	2140000	1	1.30E-04	990.8	5949.2	1.3E-07
Pu-238	87.7	174	8.70E-07	6.6	6933.4	1.51E-07
Pu-239	24100	174	8.70E-07	6.6	6933.4	1.51E-07
Pu-240	65400	174	8.70E-07	6.6	6933.4	1.51E-07
Pu-241	14.4	174	8.70E-07	6.6	6933.4	1.51E-07
Am-241	432	177	8.55E-07	6.5	6933.5	1.51E-07
Am-243	7380	177	8.55E-07	6.5	6933.5	1.51E-07
Cm-243	28.5	891	1.70E-07	1.3	6938.7	1.52E-07
Cm-244	18.11	891	1.70E-07	1.3	6938.7	1.52E-07

3.2.4 Turbine Building

The conceptual model assumes that the inventory is released instantly similar to the Containment Building. The results of this model are presented in Table 10. The table provides the maximum groundwater concentration (pCi/L), the table provides the amount of radioactivity (pCi) in solution, the amount sorbed to the solid material (pCi) and the peak concentration sorbed on the soil (pCi/g). The total inventory for each modeled nuclide in the Containment Building assuming a contamination level of 1 pCi/m^2 on the floors and walls is 14679 pCi. The volume between the floor and the water table is 26135 m^3, Table 1.

Table 10 Peak Groundwater and Soil Concentrations per unit source of 1 pCi/m^2 in the Turbine Building.

	Half-life (years)	Kd	Peak Concentration (pCi/L)	Activity in Solution	Activity Sorbed	Peak Concentration (pCi/g)
H-3	12.4	0	2.25E-03	14679	0	0
C-14	5730	1.2	2.74E-04	1790.2	12888.8	3.29E-07
Fe-55	2.7	2857	1.31E-07	0.9	14678.1	3.74E-07
Ni-59	75000	62	6.02E-06	39.4	14639.6	3.73E-07
Co-60	5.27	223	1.68E-06	11	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
Nb-94	20300	45	8.29E-06	54.2	14624.8	3.73E-07
Tc-99	213000	0	2.25E-03	14679	0	0
Ag-108m	127	27	1.38E-05	90.1	14588.9	3.72E-07
Sb-125	2.77	17	2.18E-05	142.2	14536.8	3.71E-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
Pm-147	2.62	95	3.93E-06	25.7	14653.3	3.74E-07
Eu-152	13.3	95	3.93E-06	25.7	14653.3	3.74E-07
Eu-154	8.8	95	3.93E-06	25.7	14653.3	3.74E-07
Eu-155	4.96	95	3.93E-06	25.7	14653.3	3.74E-07
Np-237	2140000	1	3.21E-04	2097.1	12581.9	3.21E-07
Pu-238	87.7	174	2.15E-06	14.0	14665.0	3.74E-07
Pu-239	24100	174	2.15E-06	14.0	14665.0	3.74E-07
Pu-240	65400	174	2.15E-06	14.0	14665.0	3.74E-07
Pu-241	14.4	174	2.15E-06	14.0	14665.0	3.74E-07
Am-241	432	177	2.11E-06	13.8	14665.2	3.74E-07
Am-243	7380	177	2.11E-06	13.8	14665.2	3.74E-07
Cm-243	28.5	891	4.19E-07	2.7	14676.3	3.74E-07
Cm-244	18.11	891	4.19E-07	2.7	14676.3	3.74E-07

3.2.5 Spent Fuel Building

The conceptual model assumes that the inventory is released through diffusion through a one-half inch uniformly contaminated zone on the floor, similar to the Auxiliary Building. The results of this model are presented in Table 11. The table provides the maximum groundwater concentration (pCi/L), the table provides the amount of radioactivity (pCi) in solution, the amount sorbed to the solid material (pCi) and the peak concentration sorbed on the soil (pCi/g). The total inventory was 7.80E-10 Ci (780 pCi) for each nuclide modeled. The volume between the floor and the water table is 208 m^3, Table 1.

Table 11 Peak Groundwater Concentrations (pCi/L) per unit source of 1 pCi/m^2 in the Spent Fuel Building

Nuclide	Half-life (years)	K_d (ml/g)	Diffusion Coefficient (cm^2/s)	Peak Concentration pCi/L	Peak Radioactivity in Solution pCi	Peak Radioactivity Sorbed pCi	Peak Sorbed Concentration pCi/g
H-3	12.4	0	5.5E-07	1.49E-02	774.8	0	0
C-14	5730	1.2	3.0E-09	1.83E-03	95.2	685.2	2.2E-06
Fe-55	2.7	2857	3.0E-09	2.04E-07	0.011	181.8	5.83E-07
Ni-59	75000	62	1.1E-09	4.02E-05	2.1	777.6	2.49E-06
Co-60	5.27	223	4.1E-11	4.25E-07	0.022	30	9.48E-08
Ni-63	96	62	1.1E-09	3.13E-05	1.6	605	1.94E-06
Sr-90	29.1	2.3	5.2E-10	3.21E-04	16.7	230.3	7.38E-07
Nb-94	20300	45	3.0E-09	5.53E-05	2.9	776.4	2.49E-06
Tc-99	213000	0	5.5E-07	1.50E-02	780	0.0	0
Ag-108m	127	27	3.0E-09	8.56E-05	4.5	721.1	2.31E-06
Sb-125	2.77	17	3.0E-09	3.41E-05	1.8	180.9	5.8E-07
Cs-134	2.06	45	3.0E-09	1.13E-05	0.6	158.7	5.09E-07
Cs-137	30	45	3.0E-09	4.07E-05	2.1	571.4	1.83E-06
Pm-147	2.62	95	3.0E-09	6.03E-06	0.3	178.7	5.73E-07
Eu-152	13.3	95	5.0E-11	1.75E-06	0.09	51.9	1.66E-07
Eu-154	8.8	95	5.0E-11	1.37E-06	0.07	40.6	1.3E-07
Eu-155	4.96	95	5.0E-11	1.04E-06	0.05	30.8	9.88E-08
Np-237	2140000	1	3.0E-09	2.14E-03	111.3	667.7	2.14E-06
Pu-238	87.7	174	3.0E-09	1.28E-05	0.67	694.9	2.23E-06
Pu-239	24100	174	3.0E-09	1.43E-05	0.74	776.3	2.49E-06
Pu-240	65400	174	3.0E-09	1.43E-05	0.74	776.3	2.49E-06
Pu-241	14.4	174	3.0E-09	7.83E-06	0.41	425.1	1.36E-06
Am-241	432	177	3.0E-09	1.38E-05	0.72	762.1	2.44E-06
Am-243	7380	177	3.0E-09	1.41E-05	0.73	778.7	2.5E-06
Cm-243	28.5	889	3.0E-09	2.03E-06	0.11	564.3	1.81E-06
Cm-244	18.11	889	3.0E-09	1.70E-06	0.09	472.6	1.51E-06

3.2.6 Waste Water Treatment Facility

The conceptual model assumes that the inventory is released instantly to the water, similar to the Containment Building. The results of this model are presented in Table 12. The table provides the maximum groundwater concentration (pCi/L), the table provides the amount of radioactivity (pCi) in solution, the amount sorbed to the solid material (pCi) and the peak concentration sorbed on the soil (pCi/g). The total inventory for each modeled nuclide in the WWTF assuming a contamination level of 1 pCi/m^2 on the floors and walls is 1124 pCi. The volume between the floor and the water table is 144 m^3, Table 1.

Table 12 Peak Groundwater and Soil Concentrations per unit source of 1 pCi/m^2 in the Crib House/Forebay Building

	Half-life (years)	Kd	Peak Concentration (pCi/L	Activity in Solution	Activity Sorbed	Peak Concentration (pCi/g)
H-3	12.4	0	3.13E-02	1126	0	0
C-14	5730	1.2	3.82E-03	137.5	990.3	4.58E-06
Fe-55	2.7	2857	1.82E-06	0.1	1124.9	5.21E-06
Ni-59	75000	62	8.40E-05	3.0	1124.8	5.21E-06
Co-60	5.27	223	2.34E-05	0.8	1125.5	5.21E-06
Ni-63	96	62	8.40E-05	3.0	1124.8	5.21E-06
Sr-90	29.1	2.3	2.12E-03	76.2	1051.4	4.87E-06
Nb-94	20300	45	1.16E-04	4.2	1123.7	5.2E-06
Tc-99	213000	0	3.13E-02	1128	0	0
Ag-108m	127	27	1.92E-04	6.9	1120.9	5.19E-06
Sb-125	2.77	17	3.03E-04	10.9	1114.1	5.16E-06
Cs-134	2.06	45	1.16E-04	4.2	1123.4	5.2E-06
Cs-137	30	45	1.16E-04	4.2	1123.4	5.2E-06
Pm-147	2.62	95	5.48E-05	2.0	1125.3	5.21E-06
Eu-152	13.3	95	5.48E-05	2.0	1125.3	5.21E-06
Eu-154	8.8	95	5.48E-05	2.0	1125.3	5.21E-06
Eu-155	4.96	95	5.48E-05	2.0	1125.3	5.21E-06
Np-237	2140000	1	4.48E-03	161.1	966.7	4.48E-06
Pu-238	87.7	174	3.00E-05	1.1	1126.8	5.22E-06
Pu-239	24100	174	3.00E-05	1.1	1126.8	5.22E-06
Pu-240	65400	174	3.00E-05	1.1	1126.8	5.22E-06
Pu-241	14.4	174	3.00E-05	1.1	1126.8	5.22E-06
Am-241	432	177	2.95E-05	1.1	1126.8	5.22E-06
Am-243	7380	177	2.95E-05	1.1	1126.8	5.22E-06
Cm-243	28.5	891	5.85E-06	0.2	1124.9	5.21E-06
Cm-244	18.11	891	5.85E-06	0.2	1124.9	5.21E-06

The WWTF has the highest predicted concentrations of any building. This is because of the limited volume of water available for mixing. The ratio of the modeled inventory in the WWTF to the volume is 7.8 pCi/m^3. This ratio in other buildings ranges from 3.75 pCi/m^3 in the Spent Fuel Building down to 0.23 pCi/m^3 in both the Auxiliary Building and the Crib House/Forebay Building. The predicted concentrations are consistent with the ratio of the modeled inventory to volume. The actual concentrations will be determined based on characterization data.

4.0 Conclusions

A screening model for predicting peak groundwater and soil radionuclide concentrations at the Zion Nuclear Power Station after decommissioning has been developed. Values for each of the six underground structures that will remain after decommissioning are provided. Two structures, the Auxiliary Building and the Spent Fuel Building used a diffusion controlled release from the concrete as the conceptual model. The other four buildings, Containment, Crib House/Forebay, Turbine and the WWTF, assumed instant release of the entire inventory into the water/backfill region between the building floor and the water table. The choice of release model was based on existing data and process knowledge. The approach uses the DUST-MS simulation model to calculate the release and peak concentrations. The analysis is based on a unit source term of 1 pCi/m^2 on the entire floor and walls of for each building. Conservative assumptions based on existing data were used in the screening model for selecting parameters that impact release (Diffusion coefficient) and groundwater concentration (K_d, porosity, bulk density). The results can be combined with characterization data to determine peak groundwater dose for all the nuclides and screen out those that are not significant contributors to dose.

5.0 References

Conestoga-Rovers & Associates, “Evaluation of Hydrological 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.

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.   *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.

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, 2001]
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., "**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.

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.

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.