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Estimates for Release of Radionuclides from Potentially Contaminated Concrete at the Haddam Neck Nuclear Plant

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1) Introduction

Decommissioning of the Haddam Neck Nuclear Power Plant operated by Connecticut Yankee is in progress. Figure 1 shows a schematic of the Containment Building and Spent Fuel Pool (SFP) Building**.**

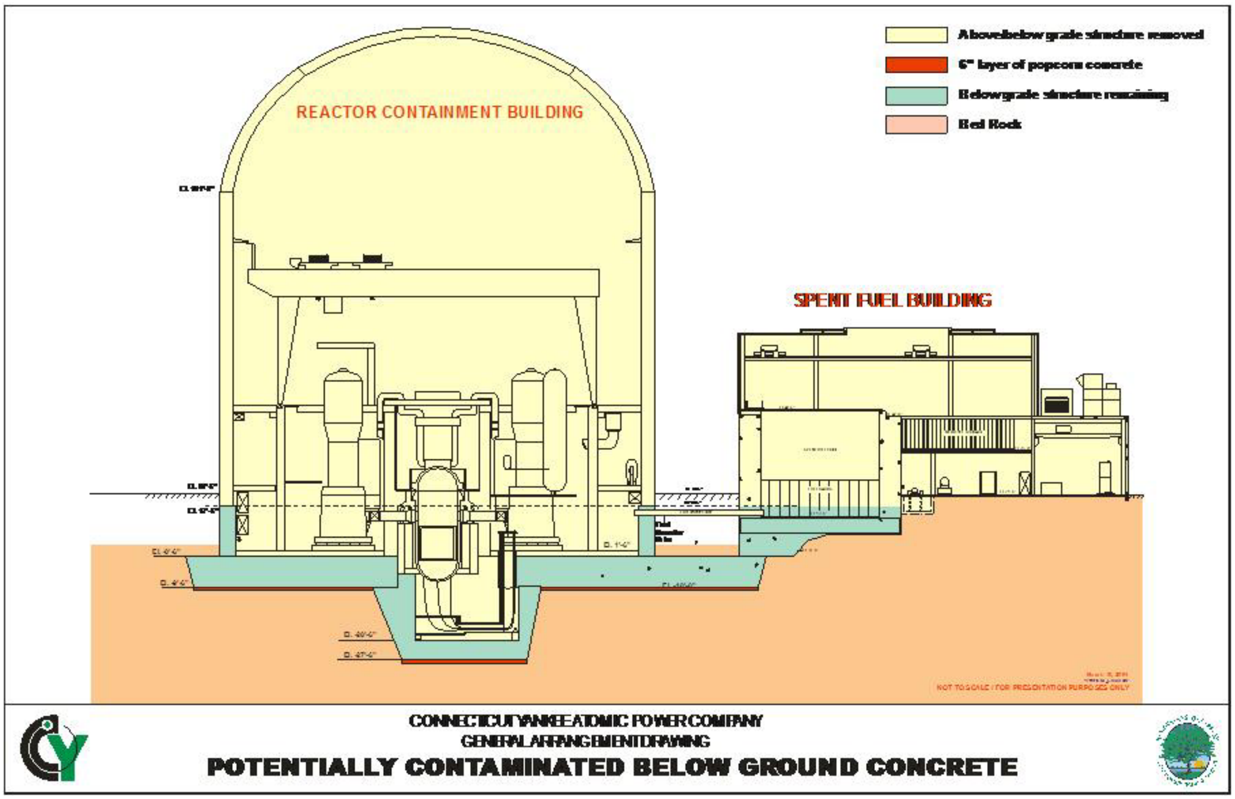

Figure 1 Schematic diagram of potentially contaminated subsurface concrete.

Consideration is being given to leaving some subsurface concrete from the Containment, Spent Fuel and certain other buildings in place following NRC license termination. Characterization data of most of these structures show small amounts of residual contamination. The In-Core Sump area of the Containment Building has shown elevated levels of tritium, Co-60, Fe-55, and Eu-152 and lesser quantities of other radionuclides due to neutron activation of the concrete in this area. This analysis is provided to determine levels of residual contamination that will not cause releases to the groundwater in excess of the acceptable dose limits.

1.1 Objective:

Calculate a conservative relationship between the radionuclide concentration of subsurface concrete and the maximum groundwater concentration (pCi/L) for the concrete that may remain following license termination at Connecticut Yankee.

1.2 Approach:

a) Determine the inventory. Based on dimensions of subsurface structures, estimates of the total inventory will be obtained for a unit concrete concentration of 1 pCi/g. The use of a unit inventory will allow scaling to actual values once characterization is complete. The calculation will allow determination of a maximum groundwater concentration in the basement once decontamination is completed and the basement backfilled.
b) Select key radionuclides. Characterization data have shown that H-3 is the most widespread concrete contaminant with by far the highest levels in the In-core Sump area. The In-core sump area concrete also has elevated levels of Fe-55, Eu-152 and to a lesser extent Co-60. Low volumetric concentrations of Sr-90 and Cs-137 have also been measured in the concrete in certain other subsurface structures. A template will be developed that will allow Connecticut Yankee to assess any radionuclide that is detected at concentrations greater than the Minimum Detectable Activity.
c) Calculate release rates from the concrete. Assume diffusion-controlled release from the concrete using literature values for diffusion coefficients, and dimensions of subsurface structures. The in-core sump will be rendered inaccessible using flowable fill. Diffusion of radionuclides through this material into the containment basement will be included in the determination of groundwater concentrations.
d) Calculate maximum groundwater concentration based on the maximum water concentration and the volume of water in the remaining containment structure. This approach uses the diffusion-controlled release model to calculate the maximum activity available to the water. For diffusion-controlled release, the maximum release rate occurs in the first year and continually decreases. The released activity undergoes radioactive decay. The maximum activity is calculated from a balance of release rates due to diffusion and the radioactive decay rate. At some point, the release rate equals the radioactive decay rate. At this time, the maximum activity available to the water is calculated. This approach does not account for transport away from the concrete structures due to the flow of groundwater through and around basements that would lead to lower available activity concentration in groundwater. Using the maximum activity available to the water, sorption effects on the backfill soil are considered and the concentration in the water is calculated based on the volume of water in the remaining containment structure.

In the above approach, consideration was given to using the annual well pumping volume (885 m^3/yr) as opposed to the volume of water in the containment structure (1,370 m^3) as the mixing volume. As it is assumed that the contamination is instantly distributed uniformly throughout the entire containment volume, the containment volume is the appropriate choice for calculating groundwater concentration.

2) Inventory

The inventory determined below uses the volume of concrete and the assumption that the concrete is uniformly contaminated at a concentration of 1 pCi/g. Calibration to measured inventories will be achieved by multiplying the values calculated at 1 pCi/g by the average measured or estimated concentration in each subsurface structure once sufficient characterization data are available to ensure that the data is representative or conservative.

Values for concrete volume for the Containment and SFP building are in Appendix A. The In-core sump has a complicated geometry and it is expected that the side walls of the sump will have different activation levels than the floor level. For this reason, the In-core sump is represented by 3 regions. A circular top region which is 11.5 ft high having a radius of 8 ft and thickness of 2 ft. The bottom region has identical geometry. The floor is represented as a circular disk with a 10 ft radius and 2 ft thickness. Analysis was performed for the exterior wall, mat, the in-core sump beneath the reactor in the containment building and for the walls and floor of the Spent Fuel Pool. Contaminant levels in the upper-half walls of the In-core sump(below the ledge that supported the Neutron Shield Tank) are expected to be much higher than in the lower half walls due to the greater neutron flux experienced in this region. Therefore, the cylindrical portion of the in-core sump was divided into two regions for the analysis.

An example of the inventory calculation for each structure follows:
Volume of containment wall – 35,496 ft^3
Concrete density – 150 lbs/ft^3
Concrete Mass – 5.3×10^6 lbs = 2.4×10^9 gms.
Inventory at 1 pCi/g - 2.4×10^{-3} Ci.

Table 1 Activity Inventory per unit contaminant loading of 1 pCi/g in the concrete.

Region	Volume (ft^3)	Curies at Unit Activity Concentration (Ci)/(pCi/g)
Containment Building		
Containment Wall	35,496	2.4×10^{-3}
Containment Mat	164,511	1.12×10^{-2}
In-core sump top section	1,301	8.9×10^{-5}
In-core sump bottom section	1,301	8.9×10^{-5}
In-core sump floor	628	4.28×10^{-5}
Total	229,580	1.51×10^{-2}
Spent Fuel Pool		
Wall Volume	8,173	5.6×10^{-4}
Floor Volume	14,128	1.0×10^{-3}
Total	22,301	1.56×10^{-3}

*Based on uniform contamination at 1 pCi/g.

3) Key Radionuclides

Preliminary volumetric concrete samples have found H-3, Co-60, and Eu-152 in the walls and floors of certain basements at Connecticut Yankee. Fe-55, Sr-90 and Cs-137 have been also found in less significant quantities. If characterization data finds appreciable levels of other radionuclides calculations can be performed as needed.

4) Release Rate from the Concrete

Rather than assuming that the entire radioactivity inventory is instantly transferred from the concrete into the groundwater (as was assumed in the revision 1a of the CY LTP), a more realistic diffusion-based transfer is assumed. A large body of experimental data suggests that diffusion controls the release out of concrete (Serne, 1992, Sullivan, 1993). Therefore, a more realistic conceptual model for estimating release from the contaminated concrete assumes that diffusion is the rate-controlling mechanism.

The diffusion process in concretes is slow and even for the radionuclides with the highest diffusion coefficient, such as H-3; transport is limited to 15 to 20 cm/yr. The subsurface structures are several meters in thickness and thus, release can be modelled assuming that the solid can be modelled as a semi-infinite media. This approximation assumes that depletion effects due to the finite-size of the contaminated region are not important. However, geometry effects are incorporated through calculating the surface area to volume ratio of the contaminated region. The assumption that depletion is not important is accurate until a fractional release rate of 20%. After this point, the semi-infinite media approximation overpredicts releases.

Diffusion-controlled release from a semi-infinite media of a non-radioactive substance can be described using the equation (Sullivan, 1988)

$$CFR = 2 \times f \times (SA/V) \times (Dt/\pi)^{0.5} \qquad (1)$$

Where CFR = cumulative fractional release of the material.
f = conversion factor = 0.01 m/cm
SA = surface area (m^2)
V = volume of concrete (m^3)
D = diffusion coefficient (cm^2/s), and
t = time (s)

The semi-infinite media approximation is valid for a cumulative fractional release of up to 0.2 (or 20% of the entire inventory) (Sullivan, 1988). At CFR values above 20%, the semi-infinite media predicts higher CFR than finite geometry models due to depletion effects. Therefore, the use of Eqn. 1 is conservative. The use of Eqn (1) to estimate CFR is appropriate as demonstrated in the results presented in Tables 5 thru 10. Only one area and one isotope (tritium) exceed a cumulative fractional release of 20 % in one year making this assumption valid. Analytical models that account for finite geometry and depletion effects are available for estimating higher fractional release values (Sullivan, 1988). The influence of radioactive decay is considered in a separate step of this analysis.

For the Connecticut Yankee system, two geometries are considered in the modelling. Plane geometry is applied to walls, including the containment building wall, and floors. Cylindrical geometry is applied to the containment mat. For planar geometry, the surface area, SA_p, is:

$$SA_p = 2 \times (H \times W + W \times D + D \times H) \qquad (2)$$

Where H = wall height (m)
W = wall width (m)
D = wall depth (m)

The volume, V_p, is:

$$V_p = H \times W \times D \qquad (3)$$

For cylindrical geometry, the surface area, Sa_c, is:

$$Sa_c = 2 \times \pi \times R \times (R + H) \qquad (4)$$

Where R = radius (m)
and H = height (m)

The volume, V_c, is:

$$V_c = \pi \times R^2 \times H. \qquad (5)$$

Equations 1 – 5 can be used to estimate the cumulative fractional release from the subsurface structures at the Haddam Neck Plant (HNP). This approach calculates the release out of all surfaces of the structures.

With diffusion controlled processes and a homogeneous distribution of radionuclides in the concrete, the maximum yearly release occurs in the first year.

4.1) Experimental Diffusion Coefficients

There are a number of studies in the literature pertaining to diffusion of radionuclides through concrete. The radionuclides that have been identified as volumetric contaminants in concrete at Haddam Neck include: H-3, Fe-55. Co-60, Sr-90, Cs-137 and Eu-152. Table 2 summarizes the literature values and provide the value selected for use during the analysis. For the purposes of analysis, the largest diffusion coefficient values found in the literature were used as they provide an upper bound on diffusive releases. Studies conducted for the diffusion in concrete of Cs-137 and Co-60 from the Haddam Neck Plant (Mattigood, 2002) had measured values that are two to five orders of magnitude lower than in other studies. Release rates are proportional to the square root of the diffusion coefficient (Eqn. 1). This suggests that the release rate values used in these simulations for Cs-137 and Co-60 will be at least one order of magnitude higher than the values expected based on the measured diffusion coefficients. In addition, it is likely that the predicted release rates of other radionuclides (H-3, Fe-55. Eu-152, and Sr-90) may be much higher than the actual values as conservative values were selected for the diffusion coefficients. Some of the conservatism currently in the calculation can be removed (i.e., lower diffusion coefficient) if site-specific data are used.

Table 2 Concrete diffusion coefficients selected from the literature for evaluation of release

Radionuclide	Literature Diffusion Coefficient Values (cm^2/s)	References	Selected Diffusion Coefficient (cm^2/s)
H-3	6.0×10^{-9} - 5.5×10^{-7}	Matsuzuro, 1976; Serne, 2001; Szanto, 2002	5.5×10^{-7}
Fe-55	5.0×10^{-11}	Serne, 1992	5.0×10^{-11}
Co-60	4.0×10^{-15} - 4.0×10^{-11}	Mattigood, 2002; Muurinen, 1983	4.0×10^{-11}
Sr-90	1.0×10^{-11} - 5.2×10^{-10}	Sullivan, 1988	5.2×10^{-10}
Cs-137	2.7×10^{-15} - 3.0×10^{-9}	Mattigood, 2002; Atkinson, 1986	3.0×10^{-09}
Eu-152	1.0×10^{-11}	Serne, 1992	1.0×10^{-11}

4.2) Calculated Diffusion-controlled Release Rates

Facility dimensions were used with equations 1 – 5 and the selected diffusion coefficients to estimate diffusion-controlled release from the subsurface concrete structures uniformly contaminated to 1 pCi/g. Table 3 summarizes the geometry used to model release and surface area to volume ratio for subsurface structures modeled in plane geometry. The containment wall is modeled in planar geometry although it is cylindrical. This assumption is valid because of the long length of the structure relative to the thickness of the walls. Thus, curvature effects are minimal. The concrete around the pressure vessel is also modeled using planar geometry. The length used in the calculations was selected to conserve the volume of the cylindrical surfaces (e.g. containment wall and around the pressure vessel). It is approximately equivalent to a radius that is the average of the inner and outer radii of the wall.

The containment mat underlies the entire containment building and is essentially cylindrical in shape. However, internal sections of the cylinder are missing for the in core sump. As a first approximation, it is assumed that the mat is continuous and 9.5 feet thick with a radius of the mat is 75.5 ft. Using these values, the calculated volume is 170,125 ft^3. Engineering drawings indicate the containment mat has a volume of 164,511 ft^3. For the purposes of estimating release, the larger volume will be used. This will increase the total inventory available for release by the ratio of the estimated to actual volume. The floor of the In-core sump was also modeled in cylindrical geometry. Table 4 presents the values used in the analysis.

Part of the walls of the containment building (approximately the top nine feet) and the walls of the SFP will be above the water table. For conservatism, it is assumed that release from these walls occurs at the same rate as if they were below the water table and that the released contaminants are immediately in the saturated zone. In practice, it would take additional time for these releases to reach the water table. The amount of time would depend upon flow and sorption characteristics of the unsaturated zone but could be tens of years for radionuclides that have a high degree of sorption.

Table 3 Geometrical factors for plane geometry subsurface structures.

Location	Height (ft)	Width (ft)	Length (ft)	Volume (ft^3)	SA_p/V_p (1/ft)
SPENT FUEL POOL					
North Wall	8	6	49	2,352	0.62
South Wall	8	6	49	2,352	0.62
East Wall	8	6	36	1,728	0.64
West Wall	8	6	36	1,728	0.64
Floor	6	48	49	14,112	0.42
Additional Floor*	6	3.5	3.5	74	1.48
CONTAINMENT BUILDING					
Wall**	18	4.5	438.2	35,496	0.56
PLENUM AROUND REACTOR					
In-core sump top section	11.5	2	56.5	1,301	1.21
In-core sump bottom section	11.5	2	56.5	1,301	1.21

* Actual dimensions of the pool floor are complicated by irregular geometry. Total volume of pool floor was estimated to be 14,184 ft^3 (Appendix A). The additional floor adds 74 ft^3 of volume to make the total volume match the estimated volume. Due to its high SA/V ratio, it will release a higher fraction of radionuclides than other subsurface components.
** Containment building is cylindrical, but the wall is modeled as a plane. This is a good approximation due to the long length relative to height or width.

Table 4 Geometrical factors for cylindrical geometry subsurface structures.

Location	Height (ft)	Radius (ft)	Volume (ft^3)	SA_c/V_c (1/ft)
Containment Mat	9.5	75.5	170,125	0.24
In-core sump floor	2	10	628	1.2

The total curies released, M_t, over the first year is obtained from the following expression.

$$M_t = CFR_s \text{ (1 year)} * I_s \tag{6}$$

Where CFR_s(1year) = cumulative fractional release from a subsurface facility in 1 year.
I_s = total inventory of the subsurface facility (Curies).

The In-core sump will be filled with a flow able grout from elevations -20'6" to 0' 6'. The top of the fill will be 2.5 feet above the expected highest elevation of activated concrete. Therefore, even after release from the activated concrete walls, the radionuclides will need to diffuse through a minimum of 2.5 feet of grout fill. To evaluate the maximum release from the In-core sump region calculations were performed for 2 feet of contaminated concrete of 1 pCi/g covered by 2.5 feet of uncontaminated grout fill. The diffusion coefficient in the grout and concrete were selected to be identical (Table 2). as these materials should behave similarly and the values in Table 2 were the highest found in the literature for cement. The calculations show that 2.5 feet of clean grout was an effective barrier to release. The tritium release in the first year from the In-core top section was reduced from 1.57×10^{-5} Ci to 3.74×10^{-8} Ci. The release of other radionuclides decreased by more than seven orders of magnitude due to their lower diffusion coefficients and short half-lifes. The maximum release rate of any radionuclide, other than H-3, is less than 1 pCi/yr. Therefore, these projected releases are inconsequential as compared to projected releases from other components of the system.

Tables 5 – 10 summarize the predicted release rates for the Spent Fuel Pool and Containment Building over the first year for H-3, Fe-55, Co-60, Sr-90, Cs-137, and Eu-152. In these tables, the predicted release rate from the In-core sump is less than 1 pCi/yr for all radionuclides other than H-3. For conservatism, the release rate from each section of the In-core sump has been set to 1×10^{-12} Ci/yr (1 pCi/yr) for all radionuclides other than H-3. H-3 release rates were set to the values calculated for having 2.5 feet of clean grout backfill. These tables contain:

- the inventory in each subsurface structure based on a uniform contamination level of 1 pCi/g,
- % of total inventory for the structure. Defined as 100 times the ratio of the inventory of the structure to the inventory of all structures.
- the maximum yearly fractional release from the facility (CFR (1 year)). These values were derived from Equation 1, the geometry factors in Tables 3 and 4, and diffusion coefficients in Table 2.
- the total radioactivity released (Ci)
- % of total released from the structure. Defined as 100 times the ratio of the curies released from the facility to the curies released from all facilities.

Tables 5 - 10 show that the majority of the concrete mass is in the containment mat and this facility has the highest predicted release accounting for over 56% of the total assuming that all facilities have an initial assumed concentration of 1 pCi/g. The containment building is predicted to release 84% of the total radioactivity released. The grout backfill effectively reduces release from the In-core sump to far less than 1% of the total release from all contaminated walls. For a uniform initial concentration in the subsurface concrete, diffusion limits release to less than 5% of the H-3 inventory, less than 0.05% of the Fe-55 inventory, less than 0.04% of the Co-60 inventory, less than 0.15% of the Sr-90 inventory, less than 0.35% of the Cs-137 inventory and less than 0.02% of the Eu-152 inventory. Although the above percentages are subject to change as the actual characterization data is applied to the unitized values used here, the above gives a sense of the relative contributions to future groundwater activity from the different plant areas.

Characterization data are still being collected. A spatial distribution of contaminant levels is expected throughout the core samples. Modeling studies (Sullivan, 2004) indicate that with the high diffusion coefficient of H-3, tritium residing within the first eight inches (20 cm) of the surface can contribute to the peak release rate rate (first year) from the concrete. For Sr-90, due to its lower diffusion coefficient, radionuclides within the first inch of the surface contribute to peak release. Therefore, the H-3 concentrations will be assessed using an average concentration in the first eight inches from each surface and all other radionuclides will be assessed using an average depth of 1 inch. Although the peak release for a diffusion-controlled release is only influenced by the concentrations within the first few inches of the surface, the inventory of the entire mass of concrete is used in calculating the release in the model for consistency with the geometry used.

Table 5 Tritium maximum yearly curies release for subsurface structures

Radionuclide	Facility	Inventory (Ci)	% of Inventory	Maximum Yearly Release CFR(1)	Released (Ci)	% of total released
H-3	Containment mat	1.16E-02	73.6	3.49E-02	4.04E-04	56.6
H-3	Containment wall	2.42E-03	15.3	8.23E-02	1.99E-04	27.8
H-3	SFP North	1.60E-04	1.0	9.16E-02	1.47E-05	2.1
H-3	SFP South	1.60E-04	1.0	9.16E-02	1.47E-05	2.1
H-3	SFP East	1.18E-04	0.7	9.38E-02	1.11E-05	1.5
H-3	SFP West	1.18E-04	0.7	9.38E-02	1.11E-05	1.5
H-3	SFP Floor	9.62E-04	6.1	6.11E-02	5.87E-05	8.2
H-3	SFP additional	4.81E-06	0.03	2.17E-01	1.04E-06	0.1
H-3	In-core sump top	8.86E-05	0.6	4.22E-04	3.74E-08	0.0
H-3	In-core sump bottom	8.86E-05	0.6	4.22E-04	3.74E-08	0.0
H-3	In-core sump floor	4.28E-05	0.3	3.58E-04	1.53E-08	0.0
TOTAL		1.58E-02			7.15E-04	

Table 6 Iron maximum yearly curies release for subsurface structures

Radionuclide	Facility	Inventory (Ci)	% of Inventory	Maximum Yearly Release CFR(1)	Released (Ci)	% of total released
Fe-55	Containment mat	1.16E-02	73.6	3.49E-04	4.04E-06	56.6
Fe-55	Containment wall	2.42E-03	15.3	8.23E-04	1.99E-06	27.8
Fe-55	SFP North	1.60E-04	1.0	9.16E-04	1.47E-07	2.1
Fe-55	SFP South	1.60E-04	1.0	9.16E-04	1.47E-07	2.1
Fe-55	SFP East	1.18E-04	0.7	9.38E-04	1.11E-07	1.5
Fe-55	SFP West	1.18E-04	0.7	9.38E-04	1.11E-07	1.5
Fe-55	SFP Floor	9.62E-04	6.1	6.11E-04	5.87E-07	8.2
Fe-55	SFP additional	4.81E-06	0.03	2.17E-03	1.04E-08	0.1
Fe-55	In-core sump top	8.86E-05	0.6	1.13E-08	1.00E-12	0.0
Fe-55	In-core sump bottom	8.86E-05	0.6	1.13E-08	1.00E-12	0.0
Fe-55	In-core sump floor	4.28E-05	0.3	2.34E-08	1.00E-12	0.0
Total		1.58E-02			7.15E-06	

Table 7 Cobalt maximum yearly curies release for subsurface structures

Radionuclide	Facility	Inventory (Ci)	% of Inventory	Maximum Yearly Release CFR(1)	Released (Ci)	% of total released
Co-60	Containment mat	1.16E-02	73.6	3.12E-04	3.62E-06	56.6
Co-60	Containment wall	2.42E-03	15.3	7.36E-04	1.78E-06	27.8
Co-60	SFP North	1.60E-04	1.0	8.20E-04	1.32E-07	2.1
Co-60	SFP South	1.60E-04	1.0	8.20E-04	1.32E-07	2.1
Co-60	SFP East	1.18E-04	0.7	8.39E-04	9.89E-08	1.5
Co-60	SFP West	1.18E-04	0.7	8.39E-04	9.89E-08	1.5
Co-60	SFP Floor	9.62E-04	6.1	5.46E-04	5.25E-07	8.2
Co-60	SFP additional	4.81E-06	0.03	1.94E-03	9.32E-09	0.1
Co-60	In-core sump top	8.86E-05	0.6	1.13E-08	1.00E-12	0.0
Co-60	In-core sump bottom	8.86E-05	0.6	1.13E-08	1.00E-12	0.0
Co-60	In-core sump floor	4.28E-05	0.3	2.34E-08	1.00E-12	0.0
Total		1.58E-02			6.39E-06	

Table 8 Strontium maximum yearly curies release for subsurface structures

Radionuclide	Facility	Inventory (Ci)	% of Inventory	Maximum Yearly Release CFR(1)	Released (Ci)	% of total released
Sr-90	Containment mat	1.16E-02	73.6	1.12E-03	1.30E-05	56.6
Sr-90	Containment wall	2.42E-03	15.3	2.65E-03	6.42E-06	27.8
Sr-90	SFP North	1.60E-04	1.0	2.96E-03	4.74E-07	2.1
Sr-90	SFP South	1.60E-04	1.0	2.96E-03	4.74E-07	2.1
Sr-90	SFP East	1.18E-04	0.7	3.03E-03	3.57E-07	1.5
Sr-90	SFP West	1.18E-04	0.7	3.03E-03	3.57E-07	1.5
Sr-90	SFP Floor	9.62E-04	6.1	1.97E-03	1.89E-06	8.2
Sr-90	SFP additional	4.81E-06	0.0	6.99E-03	3.36E-08	0.1
Sr-90	In-core sump top	8.86E-05	0.6	5.73E-03	1.00E-12	0.0
Sr-90	In-core sump bottom	8.86E-05	0.6	5.73E-03	1.00E-12	0.0
Sr-90	In-core sump floor	4.28E-05	0.3	4.86E-03	1.00E-12	0.0
Total		1.58E-02			2.30E-05	

Table 9 Cesium maximum yearly curies release for subsurface structures

Radionuclide	Facility	Inventory (Ci)	% of Inventory	Maximum Yearly Release CFR(1)	Released (Ci)	% of total released
Cs-137	Containment mat	1.16E-02	73.6	2.70E-03	3.13E-05	56.6
Cs-137	Containment wall	2.42E-03	15.3	6.37E-03	1.54E-05	27.8
Cs-137	SFP North	1.60E-04	1.0	7.10E-03	1.14E-06	2.1
Cs-137	SFP South	1.60E-04	1.0	7.10E-03	1.14E-06	2.1
Cs-137	SFP East	1.18E-04	0.7	7.27E-03	8.56E-07	1.5
Cs-137	SFP West	1.18E-04	0.7	7.27E-03	8.56E-07	1.5
Cs-137	SFP Floor	9.62E-04	6.1	4.73E-03	4.55E-06	8.2
Cs-137	SFP additional	4.81E-06	0.0	1.68E-02	8.07E-08	0.1
Cs-137	In-core sump top	8.86E-05	0.6	1.13E-08	1.00E-12	0.0
Cs-137	In-core sump bottom	8.86E-05	0.6	1.13E-08	1.00E-12	0.0
Cs-137	In-core sump floor	4.28E-05	0.3	2.34E-08	1.00E-12	0.0
Total		1.58E-02			5.54E-05	

Table 10 Europium maximum yearly curies release for subsurface structures

Radionuclide	Facility	Inventory (Ci)	% of Inventory	Maximum Yearly Release CFR(1)	Released (Ci)	% of total released
Eu-152	Containment mat	1.16E-02	73.6	1.56E-04	1.81E-06	56.6
Eu-152	Containment wall	2.42E-03	15.3	3.68E-04	8.90E-07	27.8
Eu-152	SFP North	1.60E-04	1.0	4.10E-04	6.58E-08	2.1
Eu-152	SFP South	1.60E-04	1.0	4.10E-04	6.58E-08	2.1
Eu-152	SFP East	1.18E-04	0.7	4.20E-04	4.94E-08	1.5
Eu-152	SFP West	1.18E-04	0.7	4.20E-04	4.94E-08	1.5
Eu-152	SFP Floor	9.62E-04	6.1	2.73E-04	2.63E-07	8.2
Eu-152	SFP additional	4.81E-06	0.03	9.69E-04	4.66E-09	0.1
Eu-152	In-core sump (top)	8.86E-05	0.6	1.13E-08	1.00E-12	0.0
Eu-152	In-core-sump (bottom)	8.86E-05	0.6	1.13E-08	1.00E-12	0.0
Eu-152	In-core sump floor	4.28E-05	0.3	2.34E-08	1.00E-12	0.0
Total		1.58E-02			3.20E-06	

5) Maximum Water Concentration

The predicted maximum water concentration is a function of the release rate, the amount sorbed on to the solid phase (backfill material) and the volume of water into which release occurs. The conceptual model used to calculate the water concentration assumes that all releases from the subsurface structures are released directly into the backfilled region of the containment building. The maximum annual release rates calculated in Section 4 are used as the starting point for calculating water concentrations. These radionuclides can sorb on the backfill and thereby are temporarily removed from the water column. This process is discussed in section 5.1. In addition, although Section 4 provides the maximum annual release, the release will continue at a lower rate in subsequent years To account for this, an activity balance is performed that balances release rates from the subsurface structures with radioactive decay of material previously released. When these two rates are equal, the maximum activity in the water can be calculated. This process is discussed in Section 5.2. Using the maximum activity in the water, the maximum water concentration can be obtained, as provided in Section 5.3.

The above approach is conservative because it assumes that all releases are collected in a single well at the time of maximum concentration. The main factors in the assumption that are conservative is the mixing of releases from the SFP and the Containment building and allowing for a single well to collect all releases from both the internal and external surfaces of the subsurface structures.

5.1) Sorption

After release from the concrete, the radionuclides will be sorbed onto the surrounding backfill or soil. Sorption will reduce the amount of activity available to be removed through a well. Therefore, the final factor needed to calculate the concentration in water involves sorption onto the surrounding porous media. Sorption will reduce the solution concentration as follows:

$$M_t = RM_w \qquad (7)$$

Where M_t = total curies released,
R = retardation coefficient, and
M_w = curies in the water

The desired parameter is the curies in the water, which can be found from the expression

$$M_w = M_t/R \qquad (8)$$

The retardation coefficient represents the effects of sorption and is expressed as follows:

$$R = 1 + \rho K_d/\eta \qquad (9)$$

Where ρ= bulk density = 1.56 g/cm^3 (LTP, Table F-1)
K_d = distribution coefficient (cm^3/g),

η = effective porosity = 0.35 (dimensionless), (LTP, Table F-1)

Distribution coefficients were taken from site-specific analysis of the proposed backfill for the containment structure at Connecticut Yankee (Fuhrmann, 2004). Distribution coefficients were obtained for Fe-55, Co-60, Sr-90, and Cs-137 on two soil types designated A and B and a mix of these two soils. Type B was selected for backfill inside of basements and the mix of the two soils will be used to backfill outside of buildings and inside building footings. Table 11 presents the measured Kd values for Soil Type B and the mix of Soils A and B. The large difference in Kd values for some radionuclides in these two soils is believed to be due to the difference in the pH values. For conservatism, the lowest measured average site-specific Kd value will be used in the analysis. This value is designated by an asterisk in Table 11.

Table 11 Average measured distribution coefficient (K_d) (Fuhrmann, 2004).

Radionuclide	Soil B: K_d (cm^3/g)	Mix of Soils A and B: K_d (cm^3/g)
Fe-55	1200*	1200
Co-60	220	22*
Sr-90	10*	44
Cs-137	149	45*

* Value used in analysis for maximum water concentration.

Using the selected values of Kd from table 11 and values of Kd for H-3 and Eu-152 from table F-1 of the CY LTP, the water concentration of each radionuclide can be determined. Table 12 presents the selected Kd value, calculated retardation coefficient, and fraction of the radionuclide that remains in the water phase

Table 12 Ratio of mass in the water to total mass as a function of distribution coefficient (K_d).

Radionuclide	K_d	R	M_w/M_t
H-3	0.06	1.26	0.79
Fe-55	1200	5350	1.90×10^{-4}
Co-60	22	99	1.01×10^{-2}
Sr-90	10	45.6	2.19×10^{-2}
Cs-137	45	202	4.96×10^{-3}
Eu-152	825	3678	2.72×10^{-4}

Using the ratios in table 12, an estimate of the maximum water concentration can be obtained from the total curies released from Tables 5 to 10.

5.2) Maximum Water Inventory

Section 4 presented the calculation of the maximum annual release rate due to diffusion out of the subsurface structures. However, the maximum water concentration depends upon the release rate, the transport rate away from the facility, and radioactive decay. Species that undergo significant sorption will accumulate on the solid phase of the subsurface media and will migrate less than those with less sorption over the course of a year. At the Connecticut Yankee site, the flow parameters used in the soil DCGL calculations presented in the License Termination Plan (LTP) are presented in Table 13. The parameters needed to estimate transport velocity include saturated hydraulic conductivity (Ksat), hydraulic gradient, and effective porosity. The water flow velocity is the product of Ksat and the hydraulic gradient divided by the effective porosity and has a value of 50 m/yr. For Sr-90 the retardation factor, which is the ratio of water flow to contaminant flow, is 45.6 and therefore, Sr-90 is anticipated to move slightly more than 1 m/yr (water flow velocity divided by retardation coefficient). The low velocity compared to the length of the facilities suggests that concentrations may increase over time for Sr-90 as release continues. Since the movement of Sr-90 is slow compared to the distance of the containment mat, this simplistic model suggests that concentrations near the walls and floor will build-up in time due to the continued release from the contaminated walls. This indicates that the release from the contaminated walls is occurring at a faster rate than the transport away from the walls and radioactive decay. As the diffusion process releases activity from

Table 13 Saturated zone flow parameters

	Value	Units
Ksat	1030	m/yr
Effective porosity	0.35	
Hydraulic gradient	0.017	m/m

the walls and floors, the activity concentration in the groundwater increases. The activity also decreases due to radioactive decay. The intent of these calculations is to calculate the maximum activity that could be in the water using an activity balance that includes release and radioactive decay. Dilution due to flow out of the containment area due to natural processes or pumping is not considered. In this activity balance, the maximum activity present in the water/soil system is calculated as a function of release rates and decay from the following expression.

$M_s(t)$ = Activity released to solution adjusted for radioactive decay

In this activity balance, the maximum activity present in the water/soil system is calculated as a function of release rates and decay, $M_s(t)$. This calculation uses an approximation since the exact analytical expression for $M_s(t)$ represents an integral equation that requires numerical evaluation. This approximation is obtained by performing an activity balance over a time interval. The balance is determined from the activity in solution at the beginning of the time interval (t(i-1)) corrected for radioactive decay to the end of the time interval (t(i)) and the addition of activity through leaching as expressed in Equation 10.

$$M_s(t) = M_s(t(i-1))e^{-\lambda(t(i)-t(i-1))} + I(0)*(CFR(t(i)) - CFR(t(i-1))\ e^{-\lambda(t(i)+t(i-1))/2} \qquad (10)$$

Where λ = decay constant,
I(0) = the initial activity in curies of the contaminated zone at time 0,
CFR(i) = cumulative fractional release at time t_i uncorrected for decay (Eqn. 1).

The first term is the mass in solution at the beginning of the time step (Ms(t(i-1)) reduced by radioactive decay over the time interval t(i) – t(i-1). The second term is the increase in activity through diffusion-controlled release over that time interval corrected for radioactive decay to the middle of the time interval, ((t(i) + t(i-1))/2). Using the middle of the time interval is the approximation that limits Eqn (10) from being an exact analytical solution. At time= 0, the cumulative fractional release is 0 and the activity in the groundwater is 0. Therefore, after the first time interval, the activity in solution is the cumulative fractional release over that time period adjusted for decay. Over the next time increment (from time (2) to time (1)) this activity in solution at time (1) decreases due to radioactive decay and increases due to diffusion-controlled release. At some point in time, depending on decay constant and CFR values, the total concentration of radioactivity in solution has a maximum value. This occurs when the decay rate is balanced by the release rate.

Figure 2 presents the results of this calculation for Sr-90 from the East Wall of the Spent Fuel Pool. The East wall was selected for illustration as this has the highest initial release rate because it has the highest surface area to volume ratio of any subsurface facility with the exception of the In-Core Sump. Releases from the In-Core Sump are limited by the flowable fill. Therefore, the East Wall has the highest fraction of radioactivity in solution of any of the subsurface structures. Figure 2 compares the cumulative fractional release to the curies available in solution as calculated by Equation 10. The cumulative release always increases in time. After 1 year, the fractional release is approximately 3 x 10^{-3} and the activity in solution is essentially the same as the activity released as there is no time for decay. As time progresses, these two curves diverge. This is because the activity released and in solution undergoes decay (first term in equation 10). Whereas the cumulative fractional release does not account for decay once the material is released. The fraction of the activity in solution is important for assessing maximum concentrations in the water. For Sr-90, this value increases to slightly less than 9×10^{-3} with the peak occurring around 20 years. Therefore, the buildup factor, which is the ratio of the peak concentration in solution divided by the concentration after the first years release (maximum release rate) is less than a factor of 3. The peak buildup factor for Sr-90 is 2.84 and occurs after 21 years. Buildup factors calculated for each radionuclide are presented in Table 14. The activity released in the first year is multiplied by the buildup factors to calculate the maximum water concentrations for each radionuclide.

Table 14 Maximum build-up factors for water concentrations.

Radionuclide	Buildup Factor
H-3	1.91
Fe-55	1.62
Co-60	1.65
Sr-90	2.84
Cs-137	2.86
Eu-152	1.93

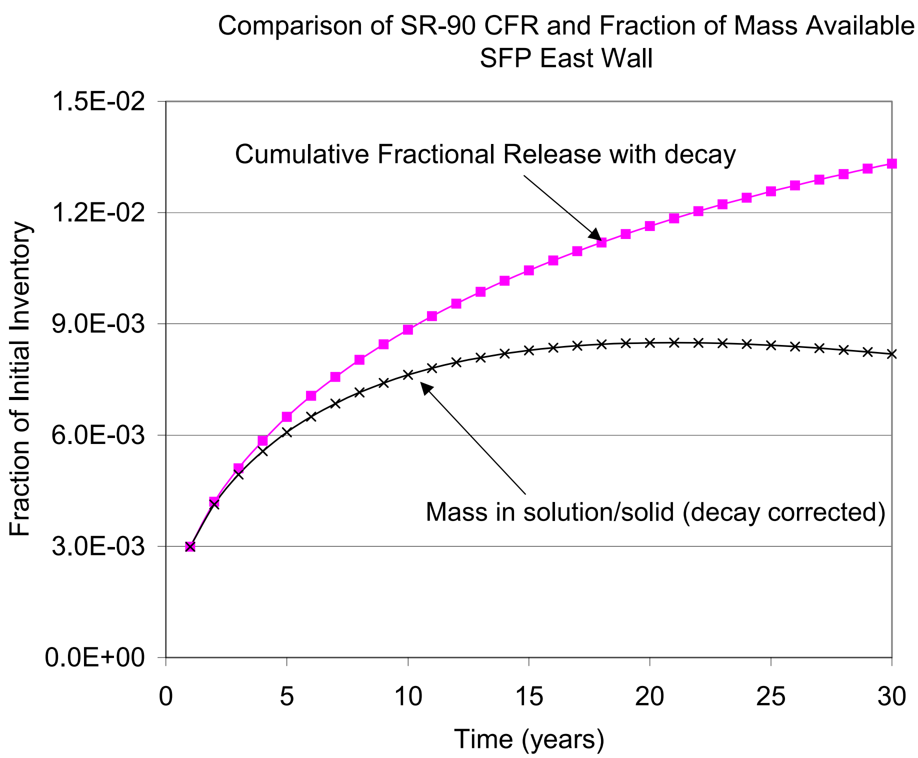

Figure 2 Comparison of cumulative fractional release and actual curies in solution for Sr-90 released from the east wall of the spent fuel pool.

5.3) Maximum Water Concentration

The maximum water concentration is calculated from the following equation.

$$C_w = I * CFR(1) \times B/(R \times W_p) \qquad (11)$$

Where I = inventory (Ci) in the subsurface facility,
CFR(1) = fraction of inventory released in the first year (1/y)
B = build-up factor that could occur if no transport away from the source were permitted,
R = retardation factor
W_p = dilution volume, (1.37×10^6 l/y)

The dilution volume was calculated as the volume of water in the remains of the containment structure. It is planned to drill holes through the containment wall to allow flow into the center section of the structure. Based on data, the expected groundwater elevation will be 8.5 feet, which is eight feet above the base of the containment mat. The volume of water above the containment mat assuming 35% porosity is 1.37×10^6 liters. This is approximately 55% more water than the annual pumping volume used in calculating the soil DCGLs. Therefore, a dilution volume of 1.37×10^6 liters is used for calculating the water concentration. The activity release from the Spent Fuel Pool structures and the Containment Building structures are summed to calculate the maximum water concentration. This implies that one well collects all of the releases from these two buildings. While it is not clear that this is physically possible, particularly for radionuclides other than H-3, which move at rates of less than 1 m/y, it is conservative. In addition, in modelling releases, it was assumed that release occurred over both the interior and exterior of the walls. Both contributions are assumed to enter the central area of the subsurface containment structure for collection in a well. Considering only the internal releases would decrease the concentrations by approximately a factor of two. For the containment sump area, releases external to the central area were not considered. Detailed flow and transport calculations could be performed if a more precise estimate of mixing between releases from the Spent Fuel Pool and Contaminant Building is desired to remove some of the conservatism present in this calculation.

Tables 15 - 20 present the inventory, maximum cumulative fractional release rate, maximum inventory release in one year, and the predicted maximum water concentration that would occur from each subsurface facility and sums the total from all facilities for all radionuclides. The results in Tables 15 - 20 indicate that the containment mat and wall are the two major sources for release for the assumed concrete contamination of 1 pCi/g. This is due to their larger surface area as compared to the SFP and In-core sump. The total predicted maximum water concentrations are based on a uniform concrete contamination of 1 pCi/g and results in groundwater concentrations that are individually less than the drinking water standards (EPA MCLs). The predicted water concentrations are directly proportional to the assumed initial concentration in the concrete.

Table 15 Predicted maximum tritium water concentrations from subsurface facilities uniformly contaminated to 1 pCi/g.

Facility	Inventory (Ci)	Maximum Annual Fractional Release Rate	Maximum Yearly Release (Ci)	Maximum Water Concentration (pCi/L)
	Tritium Releases, R=1.27, B = 1.91			
Containment mat	1.16E-02	3.49E-02	4.04E-04	445
Containment wall	2.42E-03	8.23E-02	1.99E-04	219
SFP North	1.60E-04	9.16E-02	1.47E-05	16
SFP South	1.60E-04	9.16E-02	1.47E-05	16
SFP East	1.18E-04	9.38E-02	1.11E-05	12
SFP West	1.18E-04	9.38E-02	1.11E-05	12
SFP Floor	9.62E-04	6.11E-02	5.87E-05	65
SFP additional	4.81E-06	2.17E-01	1.04E-06	1.1
In-core sump top	8.86E-05	4.22E-04	3.74E-08	4.10E-02
In-core sump bottom	8.86E-05	4.22E-04	3.74E-08	4.10E-02
In-core sump floor	4.28E-05	3.58E-04	1.53E-08	1.68E-02
Total	1.58E-02		7.15E-04	786

Table 16 Predicted maximum Fe-55 water concentrations from subsurface facilities uniformly contaminated to 1 pCi/g.

Facility	Inventory (Ci)	Maximum Annual Fractional Release Rate	Maximum Yearly Release (Ci)	Maximum Water Concentration (pCi/L)
	Fe-55 Releases, R=5350 , B = 1.62			
Containment mat	1.16E-02	3.49E-04	4.04E-06	8.94E-04
Containment wall	2.42E-03	8.23E-04	1.99E-06	4.40E-04
SFP North	1.60E-04	9.16E-04	1.47E-07	3.25E-05
SFP South	1.60E-04	9.16E-04	1.47E-07	3.25E-05
SFP East	1.18E-04	9.38E-04	1.11E-07	2.44E-05
SFP West	1.18E-04	9.38E-04	1.11E-07	2.44E-05
SFP Floor	9.62E-04	6.11E-04	5.87E-07	1.30E-04
SFP additional	4.81E-06	2.17E-03	1.04E-08	2.30E-06
In-core sump top	8.86E-05	1.13E-08	1.00E-12	2.21E-10
In-core sump bottom	8.86E-05	1.13E-08	1.00E-12	2.21E-10
In-core sump floor	4.28E-05	2.34E-08	1.00E-12	2.21E-10
Total	1.58E-02		7.15E-06	1.58E-03

Table 17 Predicted maximum cobalt water concentrations from subsurface facilities uniformly contaminated to 1 pCi/g.

Facility	Inventory (Ci)	Maximum Annual Fractional Release Rate	Maximum Yearly Release (Ci)	Maximum Water Concentration (pCi/L)
Cobalt Releases, R=99, B = 1.65				
Containment mat	1.16E-02	3.12E-04	3.62E-06	4.40E-02
Containment wall	2.42E-03	7.36E-04	1.78E-06	2.16E-02
SFP North	1.60E-04	8.20E-04	1.32E-07	1.60E-03
SFP South	1.60E-04	8.20E-04	1.32E-07	1.60E-03
SFP East	1.18E-04	8.39E-04	9.89E-08	1.20E-03
SFP West	1.18E-04	8.39E-04	9.89E-08	1.20E-03
SFP Floor	9.62E-04	5.46E-04	5.25E-07	6.39E-03
SFP additional	4.81E-06	1.94E-03	9.32E-09	1.13E-04
In-core sump top	8.86E-05	1.13E-08	1.00E-12	1.22E-08
In-core sump bottom	8.86E-05	1.13E-08	1.00E-12	1.22E-08
In-core sump floor	4.28E-05	2.34E-08	1.00E-12	1.22E-08
Total	1.58E-02		6.39E-06	7.77E-02

Table 18 Predicted maximum strontium water concentrations from subsurface facilities uniformly contaminated to 1 pCi/g.

Facility	Inventory (Ci)	Maximum Annual Fractional Release Rate	Maximum Yearly Release (Ci)	Maximum Water Concentration (pCi/L)
Strontium Releases, R=45.6, B = 2.84				
Containment mat	1.16E-02	1.12E-03	1.30E-05	5.93E-01
Containment wall	2.42E-03	2.65E-03	6.42E-06	2.92E-01
SFP North	1.60E-04	2.96E-03	4.74E-07	2.16E-02
SFP South	1.60E-04	2.96E-03	4.74E-07	2.16E-02
SFP East	1.18E-04	3.03E-03	3.57E-07	1.62E-02
SFP West	1.18E-04	3.03E-03	3.57E-07	1.62E-02
SFP Floor	9.62E-04	1.97E-03	1.89E-06	8.62E-02
SFP additional	4.81E-06	6.99E-03	3.36E-08	1.53E-03
In-core sump top	8.86E-05	5.73E-03	1.00E-12	4.55E-08
In-core sump bottom	8.86E-05	5.73E-03	1.00E-12	4.55E-08
In-core sump floor	4.28E-05	4.86E-03	1.00E-12	4.55E-08
Total	1.58E-02		2.30E-05	1.05

Table 19 Predicted maximum cesium water concentrations from subsurface facilities uniformly contaminated to 1 pCi/g.

Facility	Inventory (Ci)	Maximum Annual Fractional Release Rate	Maximum Yearly Release (Ci)	Maximum Water Concentration (pCi/L)
	Cesium Releases, R=202, B = 2.86			
Containment mat	1.16E-02	2.70E-03	3.13E-05	3.24E-01
Containment wall	2.42E-03	6.37E-03	1.54E-05	1.60E-01
SFP North	1.60E-04	7.10E-03	1.14E-06	1.18E-02
SFP South	1.60E-04	7.10E-03	1.14E-06	1.18E-02
SFP East	1.18E-04	7.27E-03	8.56E-07	8.87E-03
SFP West	1.18E-04	7.27E-03	8.56E-07	8.87E-03
SFP Floor	9.62E-04	4.73E-03	4.55E-06	4.71E-02
SFP additional	4.81E-06	1.68E-02	8.07E-08	8.36E-04
In-core sump top	8.86E-05	1.13E-08	1.00E-12	1.04E-08
In-core sump bottom	8.86E-05	1.13E-08	1.00E-12	1.04E-08
In-core sump floor	4.28E-05	2.34E-08	1.00E-12	1.04E-08
Total	`1.58E-02		5.54E-05	0.57

Table 20 Predicted maximum europium water concentrations from subsurface facilities uniformly contaminated to 1 pCi/g.

Facility	Inventory (Ci)	Maximum Annual Fractional Release Rate	Maximum Yearly Release (Ci)	Maximum Water Concentration (pCi/L)
	Europium, R=3678, B = 1.93			
Containment mat	1.16E-02	3.49E-04	4.04E-06	6.94E-04
Containment wall	2.42E-03	8.23E-04	1.99E-06	3.41E-04
SFP North	1.60E-04	9.16E-04	1.47E-07	2.52E-05
SFP South	1.60E-04	9.16E-04	1.47E-07	2.52E-05
SFP East	1.18E-04	9.38E-04	1.11E-07	1.90E-05
SFP West	1.18E-04	9.38E-04	1.11E-07	1.90E-05
SFP Floor	9.62E-04	6.11E-04	5.87E-07	1.01E-04
SFP additional	4.81E-06	2.17E-03	1.04E-08	1.79E-06
In-core sump top	8.86E-05	1.13E-08	1.00E-12	3.84E-10
In-core sump bottom	8.86E-05	1.13E-08	1.00E-12	3.84E-10
In-core sump floor	4.28E-05	2.34E-08	1.00E-12	3.84E-10
Total	1.58E-02		7.15E-06	1.23E-03

Table 21 summarizes the maximum water concentration from all subsurface structures. All are below the drinking water standards.

Table 21 Maximum water concentration from all sources for each radionuclide for concrete concentration of 1 pCi/g.

Radionuclide	Maximum Water Concentration (pCi/L)
H-3	786
Fe-55	$1.58\ 10^{-2}$
Co-60	$7.77\ 10^{-3}$
Sr-90	1.05
Cs-137	0.57
Eu-152	$1.23\ 10^{-3}$

6) Discussion

The approach used to calculate maximum water concentration is conceptually similar to assuming that there is a 'bathtub' filled with groundwater to the average water elevation of 8.5 ' along with the subsurface soil. The maximum concentration in this 'bathtub' is calculated based on concrete release characteristics, radioactive decay, and soil sorption characteristics. For all structures other than the In-core sump, release is conservatively assumed to occur from both sides of the subsurface structures into the' bathtub'. Release from the In-core sump is assumed to travel through the flowable grout used to fill the sump. Releases to the outside of the containment structure from the In-core sump are assumed to be insignificant due to the large diffusion distance from the ICI sump walls. In addition, all releases from the SFP are assumed to enter the subsurface containment bathtub instantly. These conservative assumptions remove the need for detailed water flow calculations and provide at least a factor of 2 in conservatism.

The calculations performed assume a uniform distribution within the concrete. Due to the thickness of the concrete, it is likely that concentrations in the middle of the concrete will be much lower than on the outside. This, however, will not have a major impact on the predicted peak water concentrations. Numerical studies show that for Sr-90, with a diffusion coefficient of 5.2×10^{-10} cm^2/s, only the first 2.5 cm will contribute to groundwater contamination in the first year. For H-3, with a much larger diffusion coefficient than Sr-90, 5.5×10^{-7} cm^2/s, contamination within the first 15 – 20 cm will contribute to groundwater contamination in the first year.

6.1) Impacts of Rebar on Predicted Release

It has been assumed that the concrete has been uniformly contaminated to 1 pCi/g. However, the concrete will also contain rebar. In the In-core sump region, the rebar will be activated and will have a different activity profile than the concrete. It is likely that radioactive iron and cobalt concentrations in the rebar will be higher than in the concrete and therefore, the potential for increased release needs to be examined.

For metals, release is generally controlled by corrosion processes. Therefore, release of radioactivity from the activated metals will require corrosive agents (Cl, etc.) to diffuse into the concrete and reach the rebar. Once released from the rebar, the released radioactive contamination will have to diffuse out of the concrete to enter the groundwater. There will be several inches (2.5 – 3) of concrete between the rebar and concrete surface. Therefore, even if the concentrations in the vicinity of the rebar are much greater than in the concrete, their ultimate release will be greatly diminished due to the need to diffuse through cement. Calculations of diffusion through concrete show that for the diffusion coefficients used in this analysis, Table 2, iron and cobalt release rates will be diminished by two to three orders of magnitude over three inches. In addition, at the In-core sump region, once the contaminants reach the surface, they will have to diffuse through at least 2.5 feet of concrete from the flowable fill that will be used. For these reasons, releases from activated rebar are not anticipated to provide much of a contribution to the total mass released.

To account for activated rebar, the most conservative approach would be to assume that the contamination level used to model releases is the largest of the measured contamination levels in the concrete and rebar. This approach will avoid the need for detailed modelling of rebar corrosion, radionuclide release, and subsequent transport of radionuclides to the surface of the concrete. This approach will be very conservative in the case of the rebar concentration being greater than the cement concentration of radioactivity as it neglects diffusion through the concrete that covers the rebar.

7) Template for other radionuclides or conditions

Tables 15 – 20 show the calculated maximum water concentration based on a uniform concentration of 1 pCi/g. These tables can be used directly to estimate the release of H-3, Co-60, Sr-90, Cs-137, and Eu-152. As additional characterization data are collected, the average measured concentration values for the subsurface structures can be multiplied by the maximum water concentrations in Tables 15 – 20 to obtain an estimate of the future groundwater concentration. For all of the above radionuclides, except, H-3, diffusion through the concrete is a slow process and the average concentration within the first inch (2.5 cm) will be used to estimate releases. For H-3, the average concentration over the first eight inches (20 cm) can be used.

If characterization data determine that radionuclides other than those found in Tables 15 – 20 are present, upper bounds for release of these nuclides can be estimated by choosing the radionuclide in Tables 15 – 20 that has a higher diffusion coefficient and lower retardation (distribution) coefficient than the radionuclide of concern. As a minimum, all radionuclides identified as having detectable quantities in CY concrete will be examined for their impact on groundwater. As an example, the minimum site-specific Kd for Am-241 was measured as 200 cm^3/g in the mix of soils A and B (Fuhrmann, 2004). Reviewing literature data of diffusion coefficients in cement for Am-241 a value of 5×10^{-13} cm^2/s (Serne, 1992) was suggested. Therefore, using either Cs-137 (Kd = 45, $D = 3 \times 10^{-9}$ cm^2/s) or Co-60 (Kd = 22, $D = 4 \times 10^{-11}$ cm^2/s) as a surrogate to determine an upper bound would be appropriate. In this case, Co-60 has a lower release rate than Cs-137 and use of the Co-60 release rate as an upper bound for Am-241 would be appropriate. When site-specific Kd values are not available, the estimated Kd value should be consistent with the value used in Table F-1 of the CY License Termination Plan (LTP) or the soil DCGL calculations. Literature values of diffusion coefficients are presented in Table 22. It is likely that Sr-90 will bound most of the other radionuclides (except Tc-99 and C-14) under consideration because it has a relatively high diffusion coefficient in cement (5 x 10^{-10} cm^2/s) and a relatively low distribution coefficient (10 cm^3/g).

For Tc-99 and C-14, site-specific Kd values are not available for these long-lived radionuclides. However, the value selected for C-14 in Table F-1 is 11 cm^3/g. In cement chemical environments, carbon often forms carbonates and is not readily transported through the cement. A few experiments have estimated diffusion coefficients based on leaching data. The estimated diffusion coefficient values range from 7 x 10^{-15} to 1 x 10^{-12} cm^2/s (Habeyab, 1985, Serne, 2001). Therefore, Sr-90 can be used as an upper bound for C-14. For Tc, the Kd value in Table F-1 is 0.51, therefore, H-3 could be used as an upper bound for Tc. The diffusion coefficient for Tc in cement will be lower than that of H-3 (Table 22). If use of a surrogate as an upper bound is not satisfactory for a particular radionuclide, the approach used to generate Tables 15 – 20 could be used to generate future groundwater concentrations using the values for cement diffusion coefficients (Table 22) and Kd values presented in Table F-1 of the CY License Termination Plan.

Table 22 Cement diffusion coefficients (adapted from Serne, 2001).

Radiounuclide	Diffusion Coefficient (cm^2/s)
Ac	5.00E-11
Ag	5.00E-11
Am	5.00E-13
C-14 as carbonate	1.00E-12
Cm	5.00E-11
Co	5.00E-11
Cs	5.00E-10
Eu	5.00E-11
Fe	5.00E-11
H-3	5.00E-08
Mn	5.00E-11
Nb	5.00E-11
Ni	5.00E-10
Np(V)	5.00E-10
Pa	5.00E-08
Pb	1.00E-11
Pu	5.00E-11
Ra	5.00E-11
Sr	5.00E-11
Tc	1.00E-08
Th	1.00E-12
U	1.00E-12

8) Conclusions

Subsurface structures that are currently part of the Containment and Spent Fuel Pool buildings may be left in place at the Haddam Neck Plant. This analysis has determined the relationship between volumetric contamination within these structures and the maximum future groundwater concentration. Estimates of the maximum water concentration of H-3, Fe-55, Co-60, Sr-90, Cs-137, and Eu-152 that could occur due to releases from these subsurface structures have been obtained for a unit concentration of the radionuclides in the concrete structure. Release from the concrete is controlled by diffusion. Maximum groundwater concentrations are calculated as a function of release rate, radioactive decay, and sorption with the assumption that releases from all structures are well mixed in the volume of water that will reside above the containment mat (1.37×10^6 liters). The maximum estimated concentrations for each radionuclide expected at CY are in Table 21 based on a uniform concrete concentration of 1 pCi/g.

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Appendix A: Subsurface building dimensions

A.1 Containment Building:

1 Damp proofing was applied to exterior concrete surfaces to el. 21' 6" ±

2 Below the concrete mat popcorn concrete, leveling or make bedrock concrete was poured to el. -10' 0" and is bounded by bedrock and the popcorn concrete.

3 Demolition of the Containment Building will include the dome and walls to 4 feet below grade elevation of 21' 6" to approximate elevation 17' 6".

4 The bottom surface of the popcorn concrete is exposed to bedrock and was not calculated due to its porosity.

5 The mat concrete is poured on the popcorn concrete. It's surface area exposed to the popcorn concrete was calculated but not included in the total surface area exposed.

6 Water seals were prepared for each concrete pour. A water seal consisted of a labyrinth with polyvinylchloride sheeting inserts.

Containment Building:

1. The interior radius of the containment is 67' 6" R.
2. The wall thickness is 4' 6".
3. The exterior radius is 67' 6" + 4' 6" = 72' R
4. Volume of the wall is Height x (exterior area of the circle – interior area of the circle).
5. Volume of wall in cu ft = (el. -0 6" to el. 17' 6") x ((Πx $(72)^2$) – (Πx $(67.5)^2$))
6. Volume of wall in cu ft = (18) x (16,286 – 14,314)
7. Volume of wall in cu ft = (18) x (1,972)
8. Volume of Wall in cu ft = 35,496 cu ft

9. = Exterior of wall exposed to earth in sq ft = 2*Π*R*H = 2*3.14159*72*18 = 8143 ft^2

10. The containment mat has a radius 3' 6" larger than the exterior wall of the containment or = 72' 0" + 3' 6" = 75' 6" R
11. The containment mat is 9' 6" thick, less the volume of the Incore Sump volume (keyhole) and the Containment Sump and RHR Recycle volumes.
12. From concrete prints, 16103-50082 sh1, FC-34A, the volume of concrete poured in the mat is 6,093 cu yd or 6,093 cu yd x 27 cu ft/cu yd = 164,511 cu ft.

13. The surface area of the containment mat exposed to earth in sq ft = Height x circumference

14. The surface area of the containment mat exposed to earth in sq ft = 9.5 ft x 2 x Π x 75.5 ft
15. The surface area of the containment mat exposed to earth in sq ft = 4,507 sq ft
16. The surface area of the containment mat exposed to popcorn concrete in sq ft = surface area of the circle
17. The surface area of the containment mat exposed to popcorn concrete in sq ft = Πx $(R)^2$
18. The surface area of the containment mat exposed to popcorn concrete in sq ft = Πx $(75.5)^2$
19. The surface area of the containment mat exposed to popcorn concrete in sq ft = Πx 5700
20. The surface area of the containment mat exposed to popcorn concrete in sq ft = 17,908 sq ft

21. There is .5 ft of popcorn concrete under the containment mat.
22. The volume of popcorn concrete in cu ft = Height x (area of the circle).
23. The volume of popcorn concrete in cu ft = .5 ft x $\prod (75.5)^2$
24. The volume of popcorn concrete in cu ft = 8,954 cu ft

25. The surface area of the popcorn concrete exposed to earth in sq ft = The surface area created by the increased circumference + the surface area of the height of the popcorn concrete.
26. The surface area of the popcorn concrete exposed to earth in sq ft = (Popcorn concrete surface area - Containment exterior surface area) + (height of popcorn concrete x circumference of the popcorn concrete).
27. The surface area of the popcorn concrete exposed to earth in sq ft = $(\prod (75.5)^2) - (\prod (72)^2) + (.5 \text{ x } 2 \text{ x} \prod \text{ x } 75.5)$
28. The surface area of the popcorn concrete exposed to earth in sq ft = (17,908 – 16,286) + (237)
29. The surface area of the popcorn concrete exposed to earth in sq ft = 1,622 + 237
30. The surface area of the popcorn concrete exposed to earth in sq ft = 1,859 sq ft

Refer to attached sketches

Containment Building		
Item	**Description**	**Dimension**
8	Volume of Containment concrete Wall in cu ft	35,496 cu ft
13	Volume of Containment concrete mat in cu ft	164,511 cu ft
25	Volume of Popcorn concrete in cu ft	8,954 cu ft
	Total volume of Primary Containment in cu ft	208,961 cu ft
10	Surface area of Containment concrete walls exposed to earth in sq ft	8143 sq ft
16	Surface area of Containment concrete mat exposed to earth in sq ft	4,507 sq ft
31	Surface area of Containment Popcorn concrete exposed to earth in sq ft	1,859 sq ft
	Total Primary Containment concrete exposed to earth in sq ft	14509 sq ft
	Grade elevation	21ft 6in
1	Inside Radius	67ft 6in R
2	Wall Thickness	4ft 6in
3	Grade elevation	21ft 6in
4	Top of Steel liner elevation	-0ft 6in
5	Bottom of concrete elevation	-9ft 6in
6	Popcorn concrete thickness	6in
7	Bedrock elevation	-10ft 0in
8	Mat concrete volume 6,093 cu yd from construction print 16103-50082 sh1, FC-34A. This is the amount of poured concrete. Takes into account the Incore Sump area (keyhole) and the containment sump and RHR Recycle sump areas.	164,511 cu ft
9	Interior Wall Volume in cu ft	257,650 cu ft
10	Exterior Wall Volume in cu ft	293,148 cu ft
11	Wall volume from elevation -0ft 6in to elevation 17ft 6 in, a total of 18ft	35,498 cu ft
12	Bedrock elevation at Bottom of Incore Sump Area	-27ft 0 in
13	East West Centerline of Incore Sump dimension 15ft radius	
14	North South Centerline of Incore Sump dimension 15ft 10in	
15	Mat radius 67ft 6in + 4ft 6in + 3ft 6in = 75ft 6in	75ft 6in R
16	Total concrete volume below elevation 17ft 6in = Containment wall volume + Concrete mat volume + Popcorn concrete volume. Total concrete volume = 35,496 + 164,511 + 8,954	208,961 cu ft
17	Weight of concrete	150 # / cu ft

A.2 Spent Fuel Building:

1 Damp proofing was applied to exterior concrete surfaces to el. 20’ 6” ±.

2 The Spent Fuel Building will be demolished to 4 feet below grade level.

3 The remaining portion of the Spent Fuel Building below grade level is the spent fuel pool.

Spent Fuel Building:

1. The walls and floor of the spent fuel building are 6 ft thick.
2. The bottom of the spent fuel pool is elevation 13’ 5 ¾” called 13’ 6”.
3. The bottom of the cask laydown area in the pool area on the southeast side of the pool is at elevation 10’ 113/4” called 11’ 0”.
4. The interior dimensions of the cask laydown area are N/S 10’ 0”, E/W 9’ 0”.
5. The bottom of the cask laydown are concrete is at elevation 5’ 0”.
6. The bottom of the cask laydown area concrete in the N/S direction is 22’ 0” and in the E/W direction is 21’ 0”.
7. The bottom of the pool concrete is at elevation 7’ 6”
8. The walls of the spent fuel pool are poured in water seals of the pool floor concrete.
9. The northwest side of the spent fuel pool reinforced concrete floor foundation sits on concrete fill that is approximately 15’ 0” wide in the E/W direction and approximately 24’ 0” long in the N/S direction between elevations -0’ 6” to 7’ 6” and sits partially on the containment mat at the -0’ 6” elevation.
10. If the spent fuel building is demolished to 4 feet below grade to elevation 17’ 6” the walls of the spent fuel pool that remain go to elevation 13’ 6” where they are poured into the reinforced concrete floor of the spent fuel pool. The walls of the spent fuel pool will be 4 feet high.

11. The volume of the north south walls in cu ft = Length x Height x Thickness x 2 walls
12. The volume of the north south walls in cu ft = 49 x 4 x 6 x 2
13. The volume of the north south walls in cu ft = 2,352 cu ft

14. The volume of the east west walls in cu ft = Length x Height x Thickness x 2 walls
15. The volume of the east west walls in cu ft = 36 x 4 x 6 x 2
16. The volume of the east west walls in cu ft = 1,728 cu ft

17. The surface area of the north south walls exposed to earth in sq ft = Length x Height x 2
18. The surface area of the north south walls exposed to earth in sq ft = 49 x 4 x 2
19. The surface area of the north south walls exposed to earth in sq ft = 392 sq ft

20. The surface area of the east west walls exposed to in sq ft = Length x Height x 2
21. The surface area of the east west walls exposed to earth in sq ft = 48 x 4 x 2
22. The surface area of the east west walls exposed to earth in sq ft = 384 sq ft

23. The pool floor volume of reinforced concrete in cu ft = (Volume of Cask area) + volume of non cask area + volume of remainder of pool floor
24. The pool floor cask area reinforced concrete volume in cu ft = (22 x 23.5 x 8) – (9 x 10 x 2.5) – (2.5 x 2.5 x .5 x 22) = 4136 -225 – 69 = 3,842 cu ft
25. The pool floor non cask area reinforced concrete volume in cu ft = (27 x 23.5 x 6) = 3,807 cu ft
26. The pool floor remaining reinforced concrete volume in cu ft = (24.5 x 49 x 6) = 7,203 cu ft
27. The pool floor volume of reinforced concrete in cu ft = 3,842 + 3,807 + 7,203 = 14,852 cu ft

28. The pool floor surface area exposed to earth in sq ft = (Cask laydown surface area) + (non cask laydown surface area) + (Remainder of floor side surface area)
29. The pool floor cask E/W laydown surface area exposed to earth in sq ft = (23.5 x 8) – (2.5 x 9) - (.5 x 2.5 x 2.5) = 188 -23-3 = 162 sq ft
30. The pool floor cask N/S laydown surface area exposed to earth in sq ft = (27 x 6) + (24.5 x 6) = 162 + 147 = 309 sq ft
31. The pool floor non cask laydown surface area exposed to earth in sq ft = (24.5 x 8) – (2.5 x 10) – (.5 x 2.5 x 2.5) = 196 – 25 – 3 = 168 sq ft
32. The pool floor Remainder N/S E/W surface area exposed to earth in sq ft = (24.5 x 6) + (24.5 x 6) + (48 x 6) = 147 + 147 + 288 = 582 sq ft
33. The pool floor surface area exposed to earth in sq ft = 162 + 309 + 168 + 582 = 1,221 sq ft

34. The pool floor fill concrete on the north west side sits on the containment mat. The volume of the fill concrete = (21.25 x 8 x 15) – (.5 x 4 x 8 x 21.25) = 2,550 – 340 = 2,210 cu ft

Spent Fuel Pool Building		
Item	**Description**	**Dimension**
13	The volume of the north south walls in cu ft = 2,352 cu ft	2,352 cu ft
16	The volume of the east west walls in cu ft = 1,728 cu ft	1,728 cu ft
27	The pool floor volume of reinforced concrete in cu ft = 3,842 + 3,807 + 7,203 = 14,852 cu ft	14,852 cu ft
	Total volume of fuel pool reinforced concrete	18,932 cu ft
19	The surface area of the north south walls exposed to earth in sq ft = 392 sq ft	392 sq ft
22	The surface area of the east west walls exposed to earth in sq ft = 384 sq ft	384 sq ft
29	The pool floor cask E/W laydown surface area exposed to earth in sq ft = (23.5 x 8) – (2.5 x 9) - (.5 x 2.5 x 2.5) = 188 - 23-3 = 162 sq ft	162 sq ft
30	The pool floor cask N/S laydown surface area exposed to earth in sq ft = (27 x 6) + (24.5 x 6) = 162 + 147 = 309 sq ft	309 sq ft
31	The pool floor non cask laydown surface area exposed to earth in sq ft = (24.5 x 8) – (2.5 x 10) – (.5 x 2.5 x 2.5) = 196 – 25 – 3 = 168 sq ft	168 sq ft
32	The pool floor Remainder N/S E/W surface area exposed to earth in sq ft = (24.5 x 6) + (24.5 x 6) + (48 x 6) = 147 + 147 + 288 = 582 sq ft	582 sq ft
33	The pool floor surface area exposed to earth in sq ft = 162 + 309 + 168 + 582 = 1,221 sq ft	1,221 sq ft
	Total pool surface area exposed to earth = 392 + 384 + 1,221 = 1,997 sq ft	1,997 sq ft
1	Grade elevation	21ft 6in
2	Bottom of inside pool elevation	13ft 6in
3	Bottom of pool inside cask laydown area elevation	11ft 0in
4	Wall Thickness	6ft
5	Floor thickness	6ft
6	N/S Length	49ft 0in
7	E/W Length	48ft 0in