Docket No. 50-213
CY-04-131

Attachment 2

Haddam Neck Plant
License Amendment Request, Use of a Basement Fill Model (Revising the Buried Debris Dose Model), and a Revision to Surface Contamination Release Limits for Various Piping Sizes
Technical Analysis and Regulatory Analysis Including Significant Hazards Consideration Discussion

Estimates for Release of Radionuclides from Potentially Contaminated Concrete at the Haddam Neck Plant
Prepared By Brookhaven National Laboratory

December 2004

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November 15, 2004
Revision 0

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Estimates for Release of Radionuclides from Potentially Contaminated Concrete
At the Haddam Neck Nuclear Power Station

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.

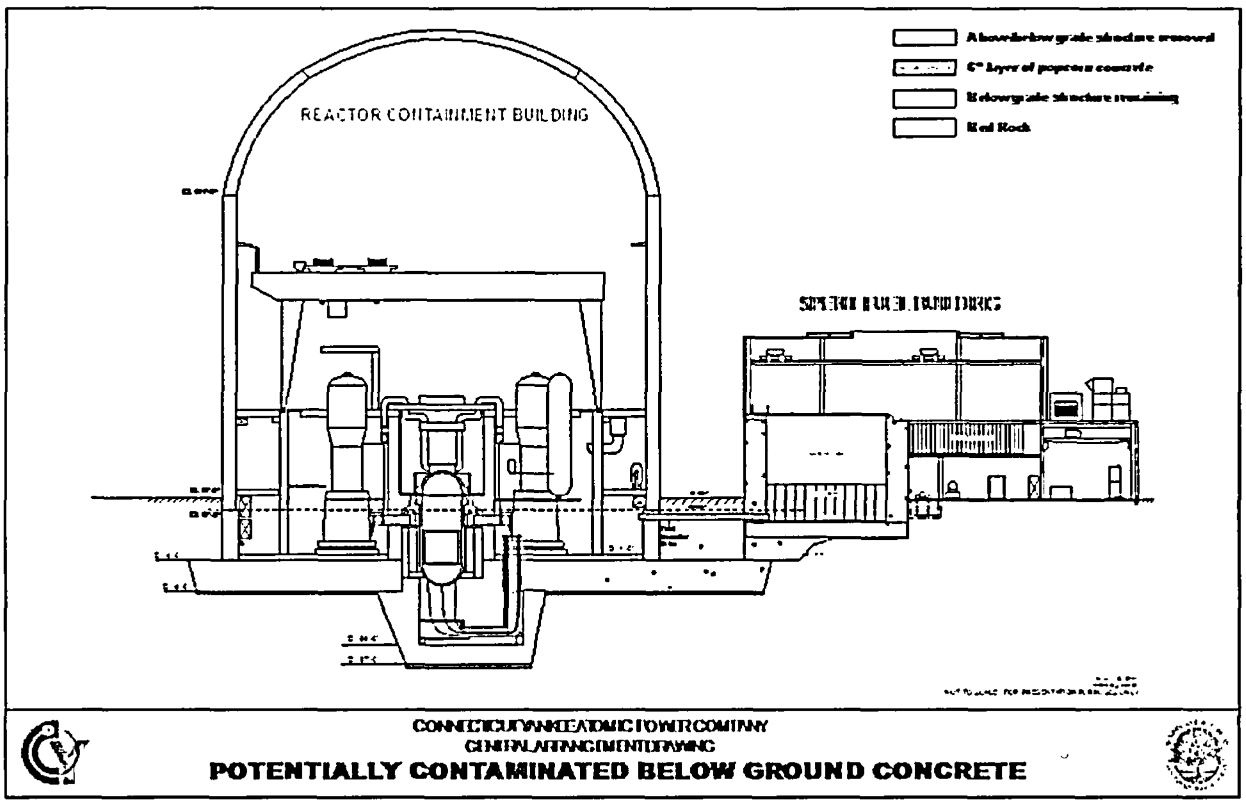

Figure 1 Schematic diagram of potentially contaminated subsurface concrete.

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)

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

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.

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

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:

$$M_t = RM_w \tag{7}$$

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

$$M_w = M_t/R \tag{8}$$

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

$$R = 1 + \rho K_d/\eta \tag{9}$$

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

Table 13 Saturated zone flow parameters

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

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

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 3×10^{-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 x 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

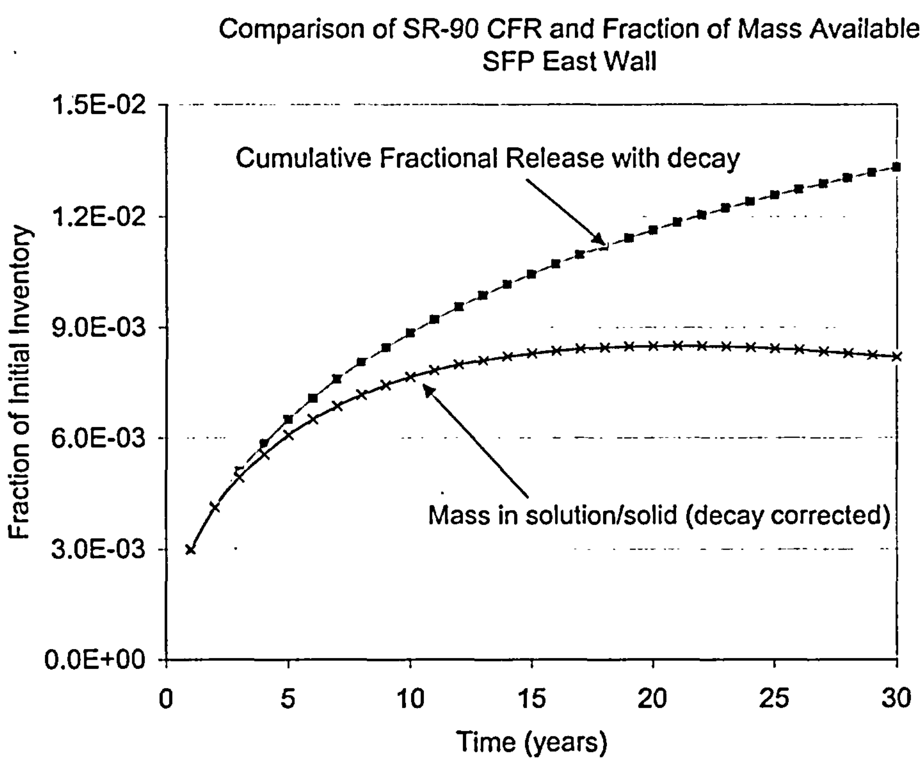

Figure 2 Comparison of cumulative fractional release and actual curies in solution for Sr-90 released from the east wall of the spent fuel pool.

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 = 3×10^{-9} cm^2/s) or Co-60 (Kd = 22, D = 4×10^{-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 (5×10^{-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 7×10^{-15} to 1×10^{-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 x 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 x 2 x $\prod$ 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

Docket No. 50-213
CY-04-131

Attachment 3

Haddam Neck Plant
License Amendment Request, Use of a Basement Fill Model (Revising the Buried Debris Dose Model), and a Revision to Surface Contamination Release Limits for Various Piping Sizes
Technical Analysis and Regulatory Analysis Including Significant Hazards Consideration Discussion

Kd Values of Backfill Material For Connecticut Yankee
Prepared By Brookhaven National Laboratory

December 2004

K_d Values of Backfill Material for Connecticut Yankee

Final Report

November 9, 2004

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K_d Values of Backfill Material for Connecticut Yankee

Methods

Two soil samples were received from Connecticut Yankee for use in K_d experiments. These are prospective backfill materials that are gravelly sands, labeled Butler 081204-A1, and Butler 081304-B1. They are referred to below as Soil A and Soil B. About 0.5 kg of each damp soil was oven dried at 60°C. They were sieved through a number 12 (1.7 mm) mesh, and each fraction was weighed. The fines were used for the K_d experiments. The calculated K_d was adjusted by multiplying by the fraction of fines in the material. This adjustment assumes a K_d of zero for the coarse fraction.

Groundwater from the site was filtered through a 0.7 micrometer filter. Aliquots of 55 g of water were weighed into sample cups, radioactive tracer (after dilution and pH adjustment) was added to each cup (typically 0.5 mL). An aliquot of each was withdrawn into a plastic syringe, filtered through a 0.2 micron Surfactant Free Cellulose Acetate (SFCA) syringe filter. For gamma emitters, 5 mL were pipetted into plastic sample vials for radionuclide analysis on a HPGe gamma-ray detector connected to a Canberra spectroscopy system. For Fe-55, with no gamma ray, 1 mL of the filtered sample was pipetted into 18 mL of Ultima-Gold liquid scintillation cocktail. These were the References used to determine the original concentration of each solution. Two grams of the dried and sieved soil were then added to each experiment. The "Mix" samples were 1g of each soil, combined. All initial K_d experiments were conducted in triplicate.

After reacting, pH was measured and samples were withdrawn and filtered in the same way as the references. Gamma-emitters were counted on an HPGe gamma-ray detector, always in the same geometry. Count times varied from 1000 to 30000 seconds. For Fe-55 Liquid Scintillation Counting was done on a Wallac Model 1414 instrument. For short half-life isotopes, the sample and the reference were counted sequentially. The difference in count rates between the reference and the sample was determined for each specimen. From this the K_d (concentration on solid / concentration in liquid) (in mL/g) was calculated based on the quantity of activity removed from the solution. This value was

corrected for gravel content by multiplying by the decimal fraction of fines present in the material.

In order to better define the K_d at specific pH values, several sets of experiments were run in which the pH of the K_d experiments were manipulated with either NaOH (1M or 0.1 M) or HCl (0.15M). After addition of the reagent, the samples were allowed to sit overnight for the pH to reach steady-state, and were then sampled as described above.

Results

Gravel and fine fractions were separated at 1.7 mm. Results are shown in table 1, along with the pH of the groundwater – soil mixture. The size fractions may differ somewhat from the actual content of bulk material since they were taken from about 0.5 kg samples.

Table 1. Gravel, Fines and pH

	% fines	% gravel	pH
Soil A	61.9	38.1	3.6
Soil B	80.9	19.1	5.5
Mix	71.4	28.6	4.5

The best value of K_d for each element tested is given in Table 2. It became apparent from initial experiments that Soil B generally has the most sorption ability. Soil A has exceptionally low values which appear to be related to its' very acidic nature. K_d of these materials varies with pH, as one would expect. Consequently, subsequent experiments focused on refining the K_d value for Soil B at the pH of the ground water in contact with that soil. For Soil B the best K_d values are based on pH of 5.5 ± 0.3, while those of the Mix are for pH of 4.5 ± 0.3. pH of Soil A in the different sorption experiments varied from 3.9 to 2.6 but because the natural pH of this material is so acidic, and the K_ds are so low, few refinements of these values were conducted. The exceptions are for Co and Sr as discussed below. In Table 2, when experimental pH values were sufficiently close to the desired value (see Table 1) the average K_d and one standard deviation distribution are

given. If the best value was obtained from interpolation of pH adjusted experimental data, then just the value is reported.

Table 2. K_d values averages of triplicate experiments (± one standard deviation) adjusted for gravel content

	Co-57	Cs-137	Am-241	Fe-55	Sr-85
Soil A	2 ± 0.08	23 ± 0.7	18 ± 0.2	160 ± 3.8	15 ± 0.7
Soil B	220 ± 31	149 ± 9	1550	1200	10
Mix	22	45 ± 4.4	200	1200	44

Cobalt. Figure 1 is a plot of Co K_d as a function of pH for all soil types. At pH values below 5 (Soil A) there was little removal of Co from solution, but above pH 5 (Soil B) Co was effectively removed from solution. This can be attributed either to adsorption or precipitation, perhaps with an iron mineral. The mixture does fall between the two, but not half-way between. This is because sorption is a non-linear function of pH, which is typically a major factor controlling sorption. An experiment was conducted on Co samples of Soil A in which NaOH was added to the soil – water mixture to adjust the pH. This material is strongly buffering, but as the pH was increased, so too was the K_d for Co. The adjusted K_d follows the curve for the unadjusted samples of the Mix and Soil B, indicating that the materials are similar and that pH is a major control on quantity of Co in solution. Figure 2 shows the Co K_d values for each of the triplicate samples, unadjusted for pH. The best value for Co sorbed on Soil B is at least 220 mL/g .

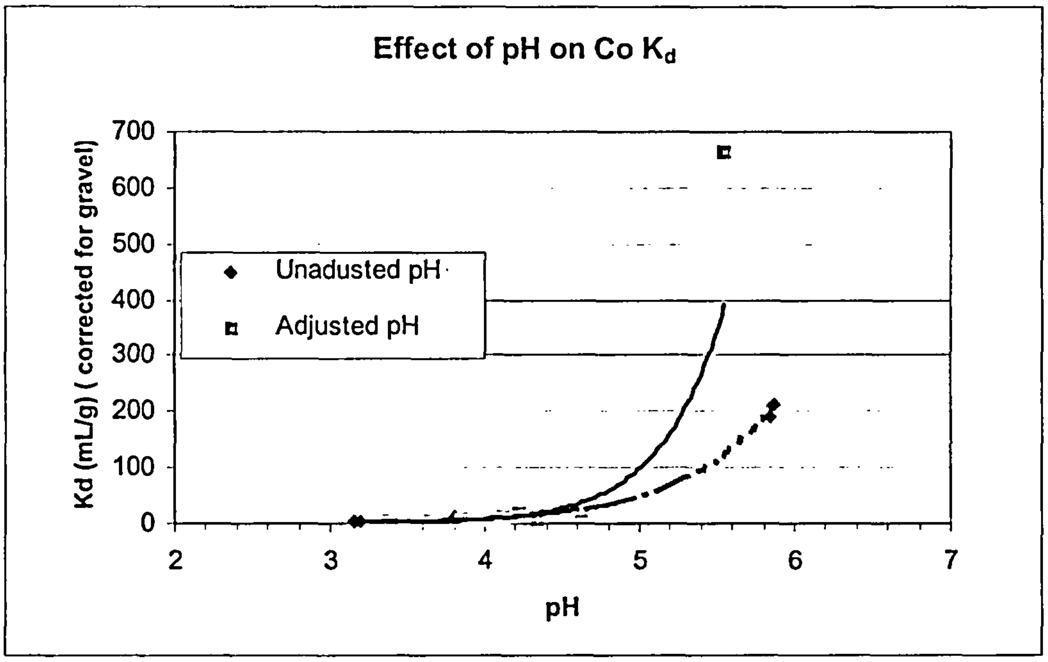

Figure 1. Effect of pH on Co sorption on backfill materials.

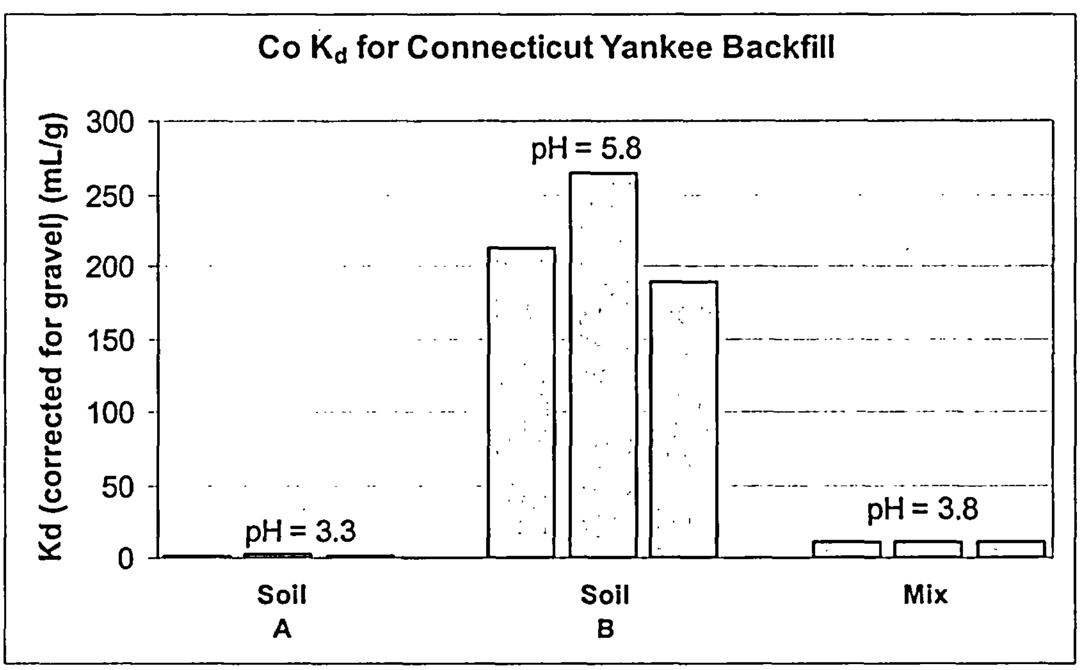

Figure 2. K_d values of Co on backfill materials, unadjusted for pH.

Cesium. Values of K_d for Cs-137 on the backfill materials are shown in Figure 3. The best value for Cs sorbed onto Soil B is 150 mL/g.

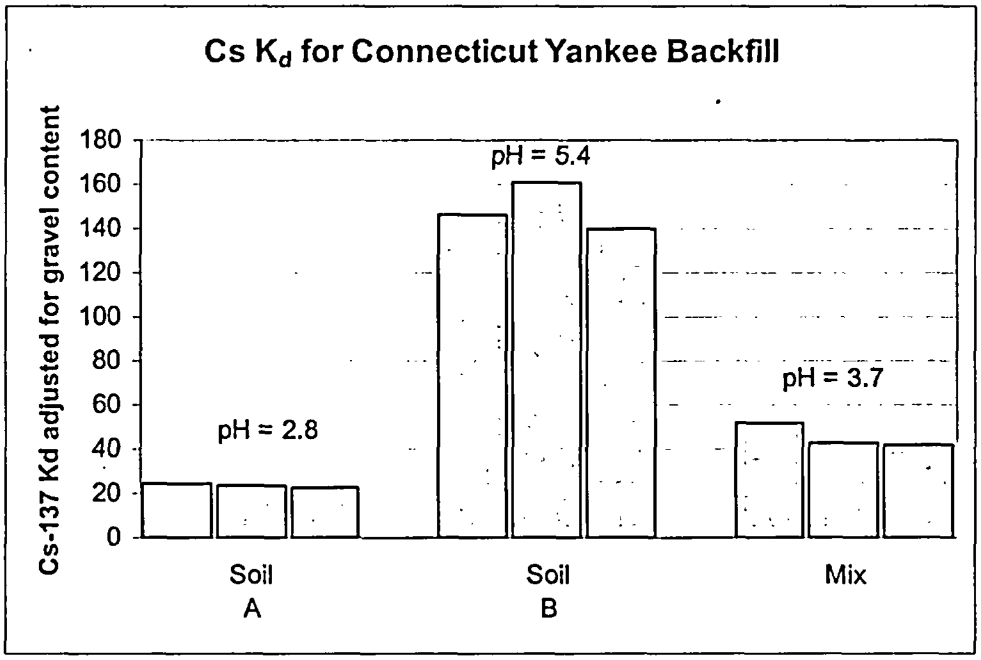

Figure 3. K_d of Cs on backfill materials.

Iron. The sorption (or precipitation) of Fe on the backfill materials is shown, in Figure 4, as a function of pH. Above pH 4.5, Fe-55 is completely removed from solution. The K_d values around 1200 represent a detection limit. This group of samples contains both Soil B and the Mix, so the best value K_d for Fe-55 in both materials is at least 1200 ml/g.

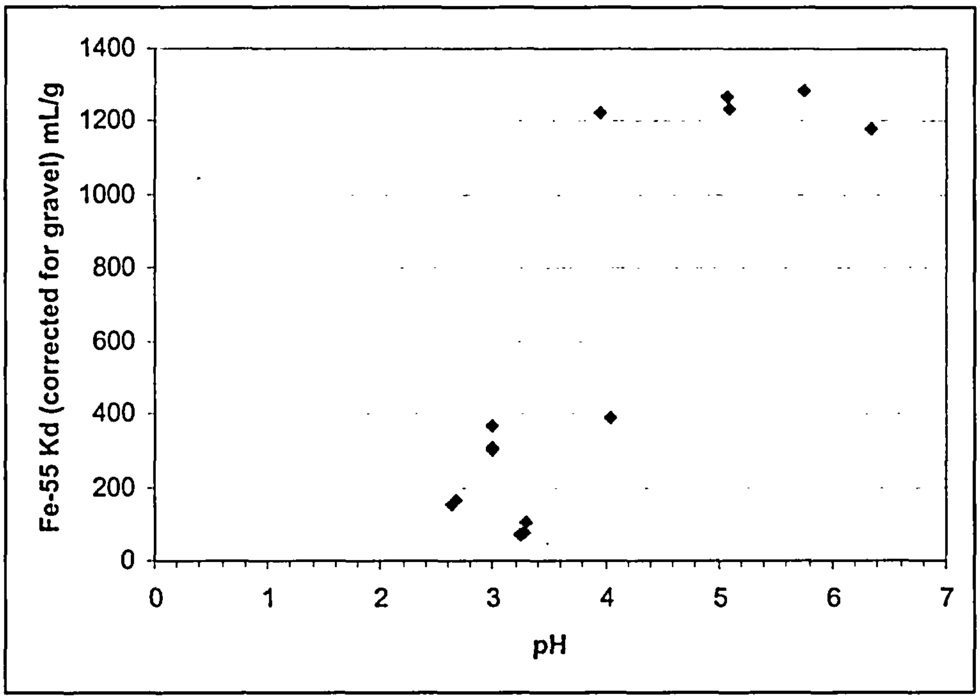

Figure 4. Fe-55 K_d values are strongly controlled by pH. Around pH 4.5 and above Fe-55 is completely removed from solution.

Americium. K_d values for Am-241 are given in Figure 5. Soil B had the highest, averaging 335 mL/g. With a minor addition of NaOH (after the original neutralization of the tracer) the pH changed very little, from 5.3 to 5.5, however the K_d increased to 1550 mL/g. The best value for Am on Soil B is 1550 mL/g. However, for the Mix soil, with its' lower pH, the best value is about 200 mL/g at pH = 4.5. This is interpolated from Figure 5. Figure 6 shows the values for Am in the samples with unadjusted pH.

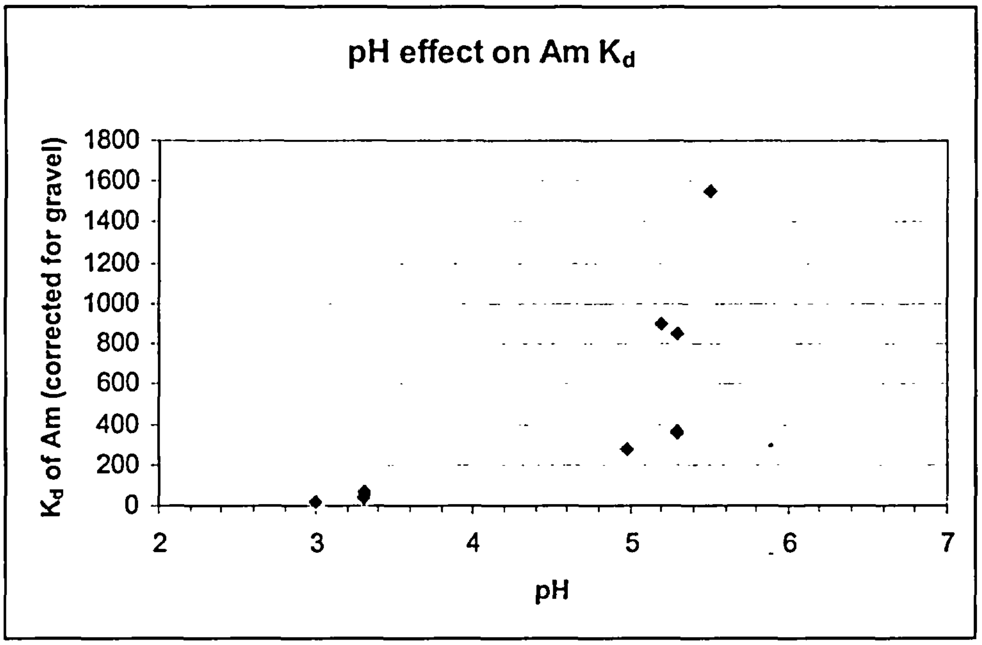

Figure 5. Effect of pH on Am K_d.

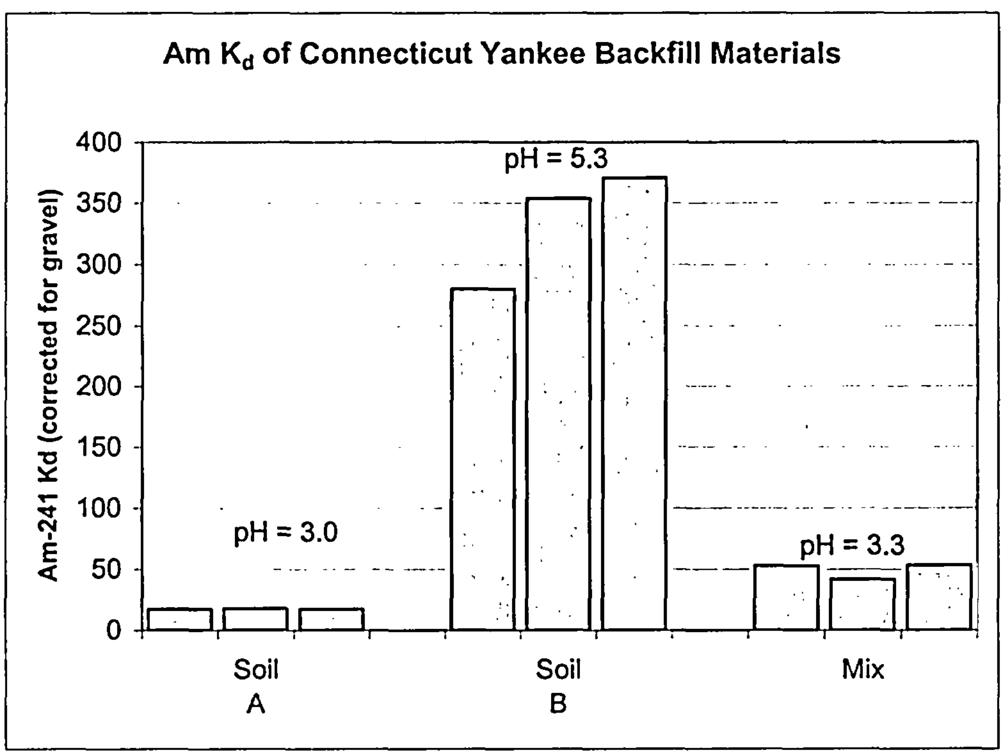

Figure 6. K_d values for Am on backfill soils, unadjusted pH.

Strontium. K_d values for Sr-85 are shown in Figure 7 as measured in samples with uncorrected pH. These pH values are slightly higher than desired because the tracer, as received from the supplier was alkaline instead of being in 0.5 M HCl. Of interest here is that the Mix soil has a higher K_d than does either of its components. This anomalous behavior could only occur if the K_d of Soil A increased dramatically with increasing pH. Because of the relatively low K_d value for Sr, and the low drinking water limits for Sr-90, it was decided to assess this behavior in detail in order to provide the best K_d value for Sr. Consequently, a series of pH adjustments were made to track the evolution of K_d with pH.

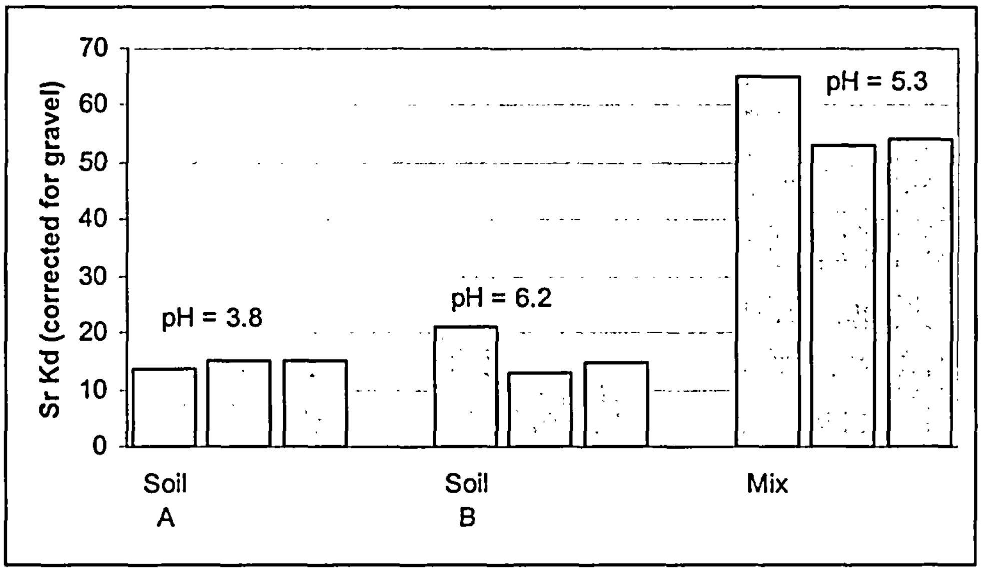

Figure 7. K_d values for Sr on Connecticut Yankee backfill material, uncorrected for pH.

The uptake of each soil was determined over a range of pH, as shown in Figure 8. Soil B is relatively unresponsive to pH, with respect to Sr sorption. It ranges from a low with K_d = 5.4 at pH = 4.5 to a high at pH = 6.1 of 21. The best value, at pH = 5.5, is 10 mL/g. Soil A, while not applicable as backfill, because of its' low sorption coefficients for other elements, was included in these experiments to assess the behavior of the Mix soil. The K_d of Soil A increased significantly as pH was increased, from a low of 13.8 at pH = 3.7 to a high of 55 at pH = 5.4. The Mix soil has relatively high K_ds for Sr, with little

change over the short range of pH tested. The best value for the Mix at pH = 4.5 is 44 mL/g.

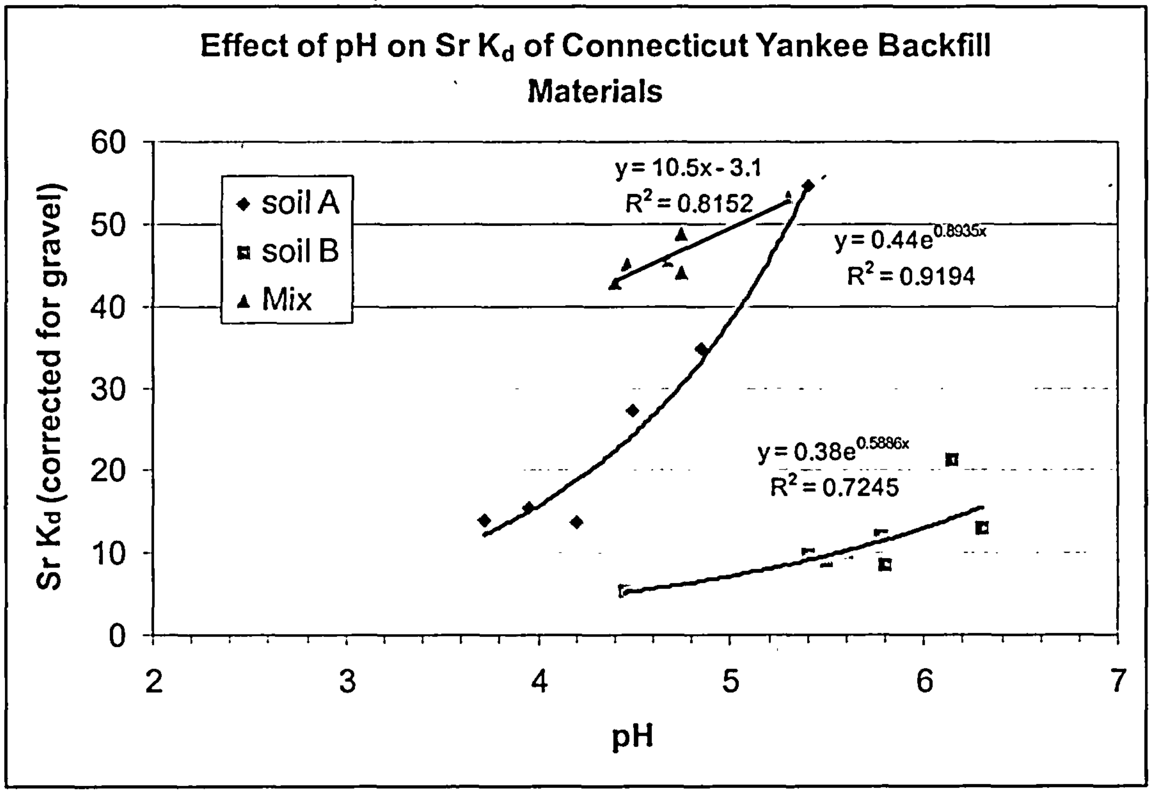

Figure 8. Effect of pH on Sr K_d of Connecticut Yankee backfill material.

Docket No. 50-213
CY-04-131

Attachment 4

Haddam Neck Plant
License Amendment Request, Use of a Basement Fill Model (Revising the Buried Debris Dose Model), and a Revision to Surface Contamination Release Limits for Various Piping Sizes
Technical Analysis and Regulatory Analysis Including Significant Hazards Consideration Discussion
Marked - up Pages of the HNP LTP

December 2004

List of the Marked-up (Affected) LTP Pages

Page No. (s)	Revision Number
1-6 through 1-12	Revision 2
2-2 and 2-3	Revision 2
5-2	Revision 2
5-7	Revision 2
5-13 through 5-70	Revision 2
6-1 through 6-22	Revision 2
Figure 6-6	Revision 2

1.3.5 Final Status Survey Plan

The primary objectives of the final status survey are to:

- select/verify survey unit classification,
- demonstrate that the level of residual radioactivity for each survey unit is below the release criterion, and
- demonstrate that the potential dose from small areas of elevated activity is below the release criterion for each survey unit.

The purpose of the Final Status Survey Plan is to describe the methods to be used in planning, designing, conducting, and evaluating final status surveys at the HNP site to demonstrate that the site meets the NRC's radiological criteria for unrestricted use. Chapter 5 of the LTP describes the Final Status Survey plan, which is consistent with the guidelines of MARSSIM. The HNP survey plan allows for the use of advanced technologies as long as the survey quality is equal to or better than traditional methods described in MARSSIM. Since MARSSIM is not readily applicable to complex nonstructural components within buildings, the current "no detectable" criteria will be applied to nonstructural components and systems at time of FSS (with the exception of those items discussed in Section 5.4.7.5). The plan also describes methods and techniques used to implement isolation controls to prevent contaminating remediated areas (as discussed in additional detail in Section 5.4.6). The HNP Final Status Survey Plan incorporates measures to ensure that final survey activities are planned and communicated to regulatory agencies to allow the scheduling of inspection activities by these agencies if so desired.

1.3.6 Compliance with the Radiological Criteria for License Termination

Chapter 6 together with Chapter 5, Final Status Survey Plan, describes the process to demonstrate compliance with the radiological criteria of 10CFR20.1402 (Reference 1-16) for unrestricted use for the HNP site. CYAPCO has selected the RESRAD computer code (Version 5.91) to model dose from soils, ~~concrete debris, concrete basements that may remain~~ and ground water, and its counterpart, RESRAD-BUILD (Version 2.37), to model dose from structures.

For building basements to remain after unrestricted release of the site, the Basement Fill Model is used to calculate the future groundwater dose. The future groundwater dose is that which results from the leaching of radionuclides from buried concrete, the containment liner and embedded piping that is contained in basements to remain. This model is discussed in detail in Section 6.8.2. The characterization sampling to be performed to supply the input to the calculation of future groundwater dose using the Basement Fill Model is discussed in Chapter 5.

For building footings, an alternate criteria to the Concrete Debris DCGLs will be applied as part of the Basement Fill Model. For footings that are to remain and are volumetrically contaminated, the radioactivity inventory in the footing will be assessed and the total quantity will be conservatively included with the other sources to the containment basement in calculating future groundwater dose. This bounds the dose calculation as the calculation of the future groundwater concentration in containment includes the major radioactivity sources contained in subsurface structures to remain after license termination. Basements other than the containment and the fuel pit will be analyzed independently using the Basement Fill Model as they are not expected to contain significant levels of radioactivity and occur later in the decommissioning.

Two primary scenarios have been selected as input to the RESRAD codes for calculating the radionuclide-specific Derived Concentration Guideline Levels (DCGLs). DCGLs are the concentration and surface radioactivity limits that will be the basis for performing the final status survey. These scenarios are the resident farmer scenario for site soils~~, concrete debris, concrete foundations/basements~~ and ground water and the building occupancy scenario for site buildings. ~~Since concrete buildings may be demolished after acceptance of the final status surveys, the future potential use of concrete debris has been evaluated to ensure that the reuse is adequately bounded by doses calculated in the LTP. This evaluation considered the use of concrete debris as backfill on site. This evaluation uses the resident farmer scenario to calculate impacts from the concrete including the conservative assumption that future drinking water originates in a well located in the buried debris. The results of this additional scenario have been analyzed to ensure the most limiting radionuclide-specific DCGLs are used to calculate operational DCGLs for building surface surveys. C~~Current decommissioning plans do not include the placement of concrete debris in facility basements, the Concrete Debris Scenario, approved as part of the LTP approved in November 2002 is no longer applicable. If the decommissioning plans change, ~~the methodology outlined above will be conservatively applied for the remaining basement structures. T~~the option to use concrete debris as backfill (and the associated Concrete Debris DCGLs) is retained.

1.3.7 Update of Site-Specific Decommissioning Costs

In accordance with 10CFR50.82 (a)(9)(ii)(F), Chapter 7 provides an updated, site-specific estimate of the remaining decommissioning costs. It also includes a comparison of these estimated costs with the present funds set aside for decommissioning and a description of the means to ensure that there will be sufficient funds for completing decommissioning.

1.3.8 Supplement to the Environmental Report

In accordance with 10CFR50.82 (a)(9)(ii)(G), Chapter 8 demonstrates that decommissioning activities will be accomplished with no significant adverse environmental impacts. Decommissioning and license termination activities remain bounded by the site-specific decommissioning activities described in:

- the PSDAR,
- the previously issued environmental assessment,
- the environmental impact statement,
- NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities (FGEIS)" (Reference 1-17), and
- NUREG-1496, "Generic Environmental Impact Statement in Support Rulemaking for Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities." (Reference 1-18).

The HNP PSDAR was submitted to the NRC in accordance with 10CFR50.82 (a)(4)(i). In the PSDAR, CYAPCO performed an environmental review to evaluate actual or potential environmental impacts associated with proposed decommissioning activities. This evaluation used NUREG-0586 and two previous site-specific environmental assessments as its basis. One site-specific assessment was performed from the conversion of the provisional operating license to a full-term operating license, and another was performed more recently from the recapture of the construction period time duration. The environmental review concluded that the impacts due to HNP decommissioning are bounded by the previously issued environmental impact statements.

As discussed in Chapter 6, the DCGLs for site buildings are calculated using the building occupancy scenario as the primary modeling scenario. ~~Adding to the conservatism, an additional modeling scenario has been considered as discussed in Section 6.8.2 (i.e., resident farmer for concrete debris).~~ Buildings to remain after release from the NRC license~~foundations/basements~~, which are decontaminated at or below

the DCGLs, could be allowed to remain standing after the final status survey. ~~These buildings and building foundations/basements could then be demolished and the debris dispositioned in a number of different manners.~~ Consideration of the building occupancy scenario ~~(as well as other scenarios)~~ in determining the DCGL is compatible with the information in SECY 00-41 (Reference 1-19). SECY 00-41 concluded that the building occupancy and resident farmer scenarios, as well as assumptions used in the FGEIS to estimate public dose, are sufficiently conservative to bound such a condition. Chapter 8 also provides a summary description of the process CYAPCO will use to ensure that the non-radiological aspects of decommissioning meet state and federal requirements for release of the site.

1.4 Decommissioning Approach

1.4.1 Overview

This section provides an overview of CYAPCO's approach to decommissioning the HNP site. References to the section in the LTP, where details concerning the particular step or stage of the decommissioning process are described, are given in parentheses.

Upon the decision to permanently cease power operations at the HNP site, CYAPCO began site characterization activities (Chapter 2). This characterization effort, which was performed to the guidelines of MARSSIM, included a Historical Site Assessment (HSA); a review of historical survey documentation; and measurements, samples, and analyses to further define the present radiological conditions of the site. The effort also addressed the status of the site relative to hazardous and state regulated non-radioactive materials.

The initial site characterization, together with geologic and hydrogeologic investigations of the site, provides the basis for the conceptualization of the site and the selection of the appropriate scenarios, models, and critical groups to address the possible future uses of the site. Conceptualization (creating the overall model for the site), which considers future use, characterization, geologic and hydrogeologic data, is also important in selecting the dose modeling code to be used to calculate the DCGLs and in the development of the Basement Fill Model for calculating "future groundwater" dose. These DCGLs correspond to a dose to the average member of the selected critical group that does not exceed 25 mrem/yr TEDE (Chapter 6).

Concurrent with site characterization and the conceptualization of the site, decommissioning activities are taking place. Activities performed during this period include the removal of contaminated components from the site for final disposition and demolition of some site buildings (Chapter 3).

Remediation of some site structures and soils will be performed, based upon the input of the initial site characterization and the DCGLs determined by dose modeling. In addition, remediation of groundwater may also be necessary to meet the dose criteria. Title 10 of the CFR, Section 20.1402 has dual criteria, namely 25 mrem/yr TEDE and ALARA. Accordingly additional remediation activities are evaluated to determine the cost/benefit of remediation beyond that which is necessary to meet the DCGLs along with the future groundwater dose calculated by the Basement Fill Model for the remaining portions of the SSCs. If the additional remediation activities are determined to be appropriate, they will be performed (Chapter 4). Once survey areas have been remediated to the required level, controls will be put into place to prevent re-contamination of the surveyed areas. (Section 5.4.6)

The Final Status Survey Plan (Chapter 5) describes the methodology by which land areas and buildings will be verified to be at or below the DCGLs (after accounting for the future groundwater dose), and thus

meet the site release criteria for unrestricted use. Once final status surveys are performed for a specific land area or building, the data collected will be documented in a release record. Periodically, several release records will be compiled into a FSS Report and made available to the NRC as evidence of completion of activities and acceptability of the area for unrestricted release. CYAPCO plans to communicate the schedules for these final status surveys to the NRC so that independent confirmatory surveys can be scheduled and performed, as necessary.

CYAPCO may pursue backfill~~demolition~~ activities once the survey results for a survey area or group of survey areas are completed. For facility SSCs remaining onsite, the final status survey results will be compiled in a series of reports by survey area(s) and will be made available for NRC review and inspection. CYAPCO plans to demolish most structures to 4 feet below grade and selected basements and to dispose of the wastes generated at an LLW waste or other appropriate facility. Final status surveys will be performed to document the radiological condition of all remaining footings~~foundations~~/basements and soil. The dose modeling approach, described in Chapter 6, evaluates potential exposures resulting from any remaining concrete structures~~, debris~~, footings~~undations~~/basements to ensure that the doses are bounded by the conservative DCGLs specified in the plan. CYAPCO does not intend to use on-site burial, disposal or incineration of any low-level radioactive waste. Materials remaining onsite will meet the appropriate DCGLs, after accounting for "future groundwater" dose, for unrestricted release, and thus are not low-level radioactive waste. (after accounting for future groundwater dose)

CYAPCO may also choose to remove specific land areas (and any associated buildings) from the 10CFR50 license after they have been surveyed and the results documented and provided to the NRC for its review and concurrence. A more detailed discussion of the phased release approach is provided in the following subsection. Upon completion of remediation and/or demolition, final status surveys, and confirmation that land areas (and any associated buildings) on the HNP site meet the site release criteria, CYAPCO will have completed the decommissioning process.

1.4.2 Phased Release Approach

CYAPCO may choose to remove specific areas from the license in a phased manner. The approach for phased release and removal from the license, after approval of the License Termination Plan, is as follows:

1. Following completion of decommissioning activities and final status survey of a survey unit, CYAPCO will compile a final status survey report to address the area or building, where decommissioning and remediation tasks are complete and the criteria of 10CFR20.1402 have been met. The results of these surveys will be documented in a report, which is provided to the NRC for its review. A report will contain a compilation of release records of the areas surveyed. A release record documents the as-left radiological condition of a survey area or survey unit.

2. Prior to a request to release a survey area from the license, the licensee will perform a Capture Zone Analysis and will assure that the ground water dose contribution is included for all applicable survey areas per the process described in Section 5.4.7.1 of the LTP.

3. CYAPCO will review and assess the impacts on the following documents in preparation of removing a land area (and any associated buildings) from the license:

    - Updated Final Safety Analysis Report and Technical Specifications;
    - Environmental Monitoring Program;
    - Offsite Dose Calculation Manual;
    - Defueled Emergency Plan;

- Security Plan;
- Post Shutdown Decommissioning Activities Report;
- License Termination Plan;
- Ground Water Monitoring Program;
- 10CFR100 Siting Criteria; and
- Environmental Report.

The reviews will include an assessment to ensure that the land area(s), and any associated building(s), to be released will have no adverse impact on the ability of the site in aggregate to meet the Part 20, Subpart E, criteria for unrestricted release. The reviews will also include the impacts on the discharge of effluents and the limits of 10CFR 20, as they pertain to the public.

4. A letter of intent to remove a portion of the property from the Part 50 license will be sent to the NRC, at least sixty (60) days before the anticipated date for release of the subject survey area(s). This letter will contain a summary of the assessments performed, as described above, and, for areas designated as "impacted," will include the FSS report for the subject survey units(s) or area(s).

5. Once the land area(s), and any associated building(s), have been verified ready for release, no additional surveys or decontamination of the subject building or area will be required (beyond those outlined in Section 5.4.6 intended for isolation and controls) unless administrative controls to prevent re-contamination are known or suspected to have been compromised. Following completion of the final status survey and submittal of the associated report, the NRC will review the report and conduct the applicable NRC confirmatory inspections.

6. Once the area(s), and any associated building(s), have been released from the license, remaining material can be dispositioned in accordance with state and federal requirements.

7. Upon completion of the HNP Decommissioning Project, a final report will be prepared, summarizing the release of areas of the HNP site from the 10CFR50 license.

1.5 License Termination Plan Change Process

CYAPCO submitted the License Termination Plan to the NRC as a supplement to the Updated Final Safety Analysis Report (Reference 1-20). The NRC subsequently approved the License Termination Plan via License Amendment No. 197 (Reference 1-21). License Amendment 197 also adds a license condition, which provides the criteria against which changes to the License Termination Plan are evaluated to determine if prior NRC approval is required in addition to the criteria specified in 10 CFR 50.59 and 10 CFR 50.82(a)(6) and (a)(7). A change to the LTP requires NRC approval prior to being implemented, if the change:

(a) Increases the radionuclide-specific derived concentration guideline levels (as discussed in Section of the LTP) or area factors (as discussed in Section 5.4.7.4 of the LTP);

(b) Increases the probability of making a Type I decision error above the level stated in the LTP (discussed in Section 5.5.1.1 of the LTP);

(c) Increases the investigation level thresholds for a given survey unit classification (as given in Table 5-8 of the LTP);

(d) Changes the classification of a survey unit from a more restrictive classification to a less restrictive classification (e.g. Class 1 to Class 2, or Class A to Class B). Definitions for the different classifications for structures and surface soils are provided in Section 2.3.3.2 of the LTP, and definitions for the different classifications for subsurface soils are provided in Section 2.3.3.1.5 of the LTP;

(e) Reduces the coverage requirements for scan measurements (Table 5-9 of the LTP); or

(f) Involves reliance upon statistical tests other than the WRS or Sign Test (as discussed in Section 5.8 of the LTP) for data evaluation.

1.6 References

1-1 Code of Federal Regulations, Title 10, Part 50.82, "Termination of License."

1-2 Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," January 1999.

1-3 Draft Regulatory Guide-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination," August 1998.

1-4 NUREG-1727, "NMSS Decommissioning Standard Review Plan," dated September 2000.

1-5 Haddam Neck Facility Operating License (DPR-61) issued December 27, 1974, as amended December 14, 1999.

1-6 Haddam Neck Updated Final Safety Analysis Report, dated August 8, 2000.

1-7 Letter B16066 from CYAPCO to the USNRC, "Haddam Neck Plant Certifications of Permanent Cessation of Power Operation and that Fuel Has Been Permanently Removed from the Reactor," dated December 5, 1996.

1-8 Letter CY-97-075 from CYAPCO to the USNRC, "Haddam Neck Plant Post Shutdown Decommissioning Activities Report," dated August 22, 1997.

1-9 USNRC Memorandum from Fairtile to Weiss dated January 28, 1998, regarding CYAPCO Post-Shutdown Decommissioning Activities Report.

1-10 Letter CY-98-005 from CYAPCO to the USNRC, "Decommissioning Updated Final Safety Analysis Report," dated January 26, 1998.

1-11 USNRC Safety Evaluation, related to Amendment No. 193 to Facility Operating License No. DPR-61, Connecticut Yankee Atomic Power Company, Connecticut Yankee Atomic Power Station, Docket 50-213, dated June 30, 1998.

1-12 USNRC Safety Evaluation, related to Amendment No. 195 to Facility Operating License No. DPR-61, Connecticut Yankee Atomic Power Company, Connecticut Yankee Atomic Power Station, Docket 50-213, dated October 19, 1999.

1-13 NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual)," dated December 1997.

1-14 "Connecticut Yankee Haddam Neck Plant Characterization Report," dated January 6, 2000.

1-15 "Haddam Neck Plant Historical Site Assessment Supplement," dated August 14, 2001.

1-16 Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use."

1-17 NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated August 1988, as supplemented on November 2002.

1-18 NUREG-1496, "Generic Environmental Impact Statement in Support Rulemaking for Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities," dated July 1997.

1-19 SECY 00-41, "Use of Rubblized Concrete Dismantlement to Address 10CFR Part 20, Subpart E, Radiological Criteria for License Termination," February 14, 2000.

1-20 Connecticut Yankee Atomic Power Company (CYAPCO) letter to USNRC dated July 7, 2000, and supplemental letters dated June 14, July 31, August 15, August 22, September 6, and September 7, 2001, and August 20 and October 10, 2002.

1-21 J. Donohew (USNRC) to K. J. Heider (CYAPCO), "Haddam Neck Plant – Issuance of Amendment RE – Approval of License Termination Plan (LTP) TAC No. MA9791," dated November 25, 2002.

2 SITE CHARACTERIZATION

2.1 Introduction

Initial site characterization of the Haddam Neck Plant (HNP) began following the permanent cessation of operations in the fall of 1997, and was completed in the fall of 1999. This initial characterization effort included a historical site assessment (HSA) – a review of historical survey documentation and measurements, samples, and analyses to further define the present radiological conditions of the site. The effort also addressed the status of the site relative to hazardous and state regulated non-radioactive materials. The initial characterization was performed to the guidelines of NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" (Reference 2-1). The HSA consisted of a review and compilation of site historical records, e.g., 10CFR50.75(g) records, radiological incident files, operational survey records, and annual environmental reports to the NRC. Personnel interviews were conducted with present and former plant employees and selected contractors to determine operational events that caused contamination in areas or systems not designed to contain radioactive or hazardous materials. Documentation from operational surveys, available through site document control facilities, was reviewed for historical information regarding radiological conditions throughout the site. The operational Radiation Protection Program provides continuing input regarding site radiological conditions. Measurements and samples beyond the scope of the operational survey program have been conducted in areas recognized as needing additional information in order to assess the type, magnitude, and extent of contamination.

The site characterization program used the same QA practices as employed by the operational radiation safety program. These practices included the use of approved procedures for the calibration, testing and use of both laboratory and portable equipment. Trained and qualified personnel collected data. Samples were controlled by administrative procedures to ensure that sample integrity is maintained. When offsite laboratories were used, they were required to perform daily instrumentation checks. Other quality control measures for offsite laboratories included periodic method blanks, replicate (duplicate) samples and participation in an inter-laboratory comparison program (e.g., cross checks). Performance of these checks, by offsite laboratories, was verified periodically by QA auditors.

The objectives of a characterization program are to collect data adequate to:

1. Divide the HNP site into manageable sections or areas for survey and classification purposes;

2. Identify the potential and known sources of radioactive contamination in systems, on structures, in surface or subsurface soils, and in groundwater;

3. Determine the initial classification of each survey area or unit;

4. Develop the initial radiological and hazardous material information in support of facility dismantlement and remediation planning and radioactive waste disposal activities;

5. Develop the radiological information in support of Final Status Survey design including minimum instrument performance standards and Quality Assurance requirements; and

6. Identify any unique radiological or hazardous material health and safety issues associated with decommissioning.

Characterization efforts at the HNP decommissioning project are an iterative process spanning all aspects of the remediation activities. The information developed during the initial HNP characterization program represents a radiological and hazardous material assessment based on the knowledge and data available at the end of 1999. This information was sufficient to satisfy the objectives listed above. Additional measurements and samples obtained during the remediation process will continue to be assessed to ensure adequacy of area classification and effectiveness of the Final Status Survey to show compliance with the established Derived Concentration Guideline Levels (DCGLs), in accordance with the guidelines of MARSSIM and to provide adequate data to allow confident calculation of the future groundwater dose from basements to remain on -site after release from the license.

The LTP provides the detailed information related to the decommissioning approach, dismantlement and bulk disposal, which will be used by CYAPCO to complete the decommissioning of the HNP. CYAPCO has replaced the future groundwater dose calculation method for basements~~not changed the dose modeling information from the~~ contained in LTP Revision 1a with the Basement Fill Model discussed in the LTP Revision 3. The decommissioning approach provided in Revision 1a of the LTP may be elected in the future for selected areas of the site. In the event that this approach is selected, the area classification approach in Revision 1a of the LTP will be implemented. Appendix H contains historical information from Table 2-10.

2.2 Historical Site Assessment

2.2.1 Introduction

The HSA for the HNP commenced in 1997, under the direction of the CYAPCO Radiation Protection Department staff. The process for conducting the HSA was established in accordance with MARSSIM guidelines. The HSA focused on historical events and routine operational processes that resulted in contamination of the plant systems, onsite buildings, exterior grounds and subsurface areas within the Radiologically Controlled Area (RCA); and grounds and subsurface areas outside of the RCA, but within the owner controlled area. The HSA, as part of the initial characterization program, was conducted to support the objectives detailed in Section 2.1.

In 1999, the HSA process became the task of Bechtel Power Corporation, as the Decommissioning Operations Contractor (DOC) at the time. The HSA was completed in the fall of 1999. The initial characterization report was issued in January of 2000 (Reference 2-2), and a Historical Site Assessment Supplement (Reference 2-3) was issued in August of 2001. Section 2 of the License Termination Plan provides a summary of findings from the HSA and the information that is the basis for area classifications, input into the development of DCGLs, development of remediation plans, and design of the Final Status Survey. The scope of the HSA included potential contamination from radioactive materials, hazardous materials, and state-regulated materials. Ongoing characterization activities are being conducted as part of the CYAPCO's self-managing of the HNP decommissioning. The LTP includes a summary of the information contained in References 2-1 and 2-2. Additional characterization information and confirmation will continue throughout the decommissioning as part of the FSS process. The LTP will generally not be updated to include this additional characterization.

2.2.2 Methodology

The HSA was designed to evaluate input from two separate sources — plant records and personnel interviews. The review of plant records consisted of routine radioactive effluent release reports, non-routine reports submitted to the NRC under provisions of the technical specifications, 10CFR20, or 10CFR50; plant incident reports or condition reports; and findings documented in accordance with other

assessment processes such as the Quality Assurance Program (QAP) and oversight activities. The information obtained through this process forms the input data for the records that are maintained on site to satisfy the requirements of 10CFR50.75(g)(1). The objective of the document reviews was to identify events that caused the contamination of systems, buildings, external surfaces, subsurface areas, or waterways, via atmospheric releases, liquid releases, or release of solid radioactive material. For each event, available supporting documentation regarding event description, facility and system design, radiological surveys and analysis, remediation efforts, and post remediation surveys was collected and reviewed. The CYAPCO nuclear records management system was the primary source of plant record information gathered during the HSA process.

To facilitate correlation of the impact of an event to physical locations on the plant site and to provide a means to correlate subsequent survey data, the owner-controlled area has been divided into areas with numeric designations. Figures 2-1 through 2-9 provide the area identification numbers for buildings and grounds within the owner controlled area. The area designations form the basis for survey units presented in Table 2-10.

In addition to the review of plant records, interviews with individuals involved in nuclear operations at HNP were conducted. Personnel interviewed included selected present and former employees and contractors involved in operations, maintenance, and radiation protection activities at the site. Information regarding unplanned releases or other events that could have resulted in site contamination was obtained through site staff and all-hands meetings, the daily plant newsletter, and Northeast Utilities system-wide publications. The effort was designed to ensure that historical events were identified that had an impact on the radiological or hazardous material status of the site. Information gathered from the interviews was reviewed and included as appropriate in the 10CFR50.75(g) database. During the HSA, CY did not identify any time gaps in information for operational history.

In addition, CY has reviewed construction activities that resulted in redistribution of materials (soils). Generally, materials that were removed from the Industrial Area or Radiological Controlled Area were placed on the Southwest Site Storage Area (Survey Area 9520), Central Peninsula Area (Survey Area 9530), Southeast Protected Area Grounds (Survey Area 9308) or the South East Landfill Area (Survey Area 9535). All these areas have received characterization surveys. Materials from construction activities that took place outside of the Industrial Area or Radiological Controlled Area generally have been released to offsite locations. These locations have been identified and surveyed under the Offsite Material Recovery Program (Section 2.2.4.2.4 contains further details).

2.2.3 Instrumentation Selection, Use, and Minimum Detectable Concentration

Radiological surveys performed in support of the initial site characterization were conducted by qualified Radiation Protection personnel. Surveys were performed by the station staff using instrumentation calibrated and maintained in accordance with station procedures utilizing National Institute of Standards and Technology (NIST) traceable calibration sources. The program consists of both periodic calibrations and response checks, when instruments are in use. The selection of instrumentation was based on the objectives of the surveys, expected radionuclide mix, and the ambient background in the area. Table 2-1 identifies instrumentation typically used at the Haddam Neck Plant. Where appropriate, typical efficiencies and Minimum Detectable Activities (MDA) at HNP have been included.

Site characterization activities included land area surveys: within the protected area, of the landfill area, of selected portions of the Primary Auxiliary Building (PAB), of the containment and waste storage building, and surface contamination surveys of the paved areas inside the security fence but outside the Radiologically Controlled Area, and the Turbine Building floors at grade elevation and on the operating floor. The details of these surveys, including the instrumentation used and the minimum detectable concentrations (MDC) are included in the survey reports. Specific reports for these surveys are identified in the references section.

5 FINAL STATUS SURVEY PLAN

5.1 Introduction

The purpose of the Final Status Survey Plan is to describe the methods to be used in planning, designing, conducting, and evaluating final status surveys at the HNP site. These surveys serve as key elements to demonstrate that the dose from residual radioactivity is less than the maximum annual dose criterion for license termination for unrestricted use specified in 10CFR20.1402 (Reference 5-1). The additional requirement of 10CFR20.1402 that all residual radioactivity at the site be reduced to levels that are as low as reasonable achievable (ALARA) is addressed in Chapter 4. The Final Status Survey Plan was developed using the guidance of Draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination" (Reference 5-2); NUREG-1575, "The Multi-Agency Radiological Site Survey and Investigation Manual (MARSSIM)" (Reference 5-3); and Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" (Reference 5-4). In September of 2000, the NRC incorporated much of the guidance of DG-4006 into various sections of NUREG-1727; "NMSS Decommissioning standard Review Plan" dated September 2000 (Reference 5-5). References to the corresponding sections of NUREG-1727 (in which the guidance of DG-4006 has been incorporated) have been given in specific sections of this LTP, as appropriate.

The final status survey process described in this plan adheres to the guidance of MARSSIM for the design of final status surveys. However, advanced survey technologies may be used to conduct radiological surveys that can effectively scan 100% of the surface and record the results. This survey plan allows for the use of these advanced technologies, where survey quality and efficiency can be increased, as long as certain criteria are met. These criteria ensure that the survey results are at least equivalent to those that would have been obtained using the non-parametric sampling methods of MARSSIM in terms of their statistical confidence. In cases where advanced survey technologies are to be used, a technical support document will be developed to describe the technology to be used and to demonstrate how the technology meets the objectives of the survey. These technical support documents will be referenced, as appropriate, in Final Status Survey Reports.

5.2 Scope

The final status survey plan encompasses the radiological assessment of all affected structures, systems and land areas for the purpose of quantifying the concentration of any residual activity that exists following all decontamination activities. Concentration limits will be established to represent the maximum annual dose rate criterion for unrestricted release specified in 10CFR20.1402.

5.3 Summary of the Final Status Survey Process

The final status survey provides data to demonstrate that all radiological parameters satisfy the established guideline values and conditions. The primary objectives of the final status survey are to:

- select/verify survey unit classification,
- demonstrate that the potential dose from residual radioactivity is below the release criterion for each survey unit, and

- demonstrate that the potential dose from small areas of elevated activity is below the release criterion for each survey unit.

The final status survey process consists of four principal elements:

- planning,
- design,
- implementation, and
- assessment.

The Data Quality Objective (DQO) and Data Quality Assessment (DQA) processes are applied to these four principal elements. DQOs allow for systematic planning and are specifically designed to address problems that require a decision to be made and provide alternate actions (as is the case in FSS). The Data Quality Assessment (DQA) process is an evaluation method used during the assessment phase of FSS to ensure the validity of survey results and demonstrate achievement of the sampling plan objectives (e.g., to demonstrate compliance with the release criteria in a survey unit).

Survey planning includes review of the Historical Site Assessment (HSA) and other pertinent characterization information to establish the radionuclides of concern and survey unit classifications. Survey units are fundamental elements for which final status surveys are designed and executed. The classification of a survey unit determines how large it can be in terms of surface area. If any of the radionuclides of concern are present in background, the planning effort may include establishing appropriate reference areas to be used to establish baseline concentrations for these radionuclides and their variability. Reference materials are specified for establishing background instrument responses for cases where gross activity measurements are to be made. A reference coordinate system is used for documenting locations where measurements were made and to allow replication of survey efforts if necessary.

Before the survey process can proceed to the design phase, concentration levels that represent the maximum annual dose criterion of 10CFR20.1402 must be established. These concentrations are established for either surface contamination or volumetric contamination. They are used in the survey design process to establish the minimum sensitivities required for the available survey instruments and techniques, and in some cases, the spacing of fixed measurements or samples to be made within a survey unit. Surface or volumetric concentrations that correspond to the maximum annual dose criterion are referred to as Derived Concentration Guideline Levels, or DCGLs. Volumetric sample results will in Footings some cases be used to calculate the "future groundwater" dose in building basements using the Basement Fill Model rather than the application of DCGLs. The future groundwater dose is that which results from the leaching of radionuclides from buried concrete, the containment liner and embedded piping that is contained in basements to remain. A DCGL established for the average residual radioactivity in a survey unit is called a $DCGL_W$. Values of the $DCGL_W$ may then be increased through the use of area factors to obtain a DCGL that represents the same dose to an individual for residual radioactivity over a smaller area within a survey unit. The scaled value is called the $DCGL_{EMC}$, where EMC stands for elevated measurement comparison.

After the $DCGL_W$ is established, a survey design is developed that selects the appropriate survey instruments and techniques to provide adequate coverage of the unit through a combination of scans, fixed measurements, and sampling. This process ensures that data of sufficient quantity and quality are obtained to make decisions regarding the suitability of the survey design assumptions and whether the unit meets the release criterion. Approved site procedures will direct this process to ensure consistent implementation and adherence to applicable requirements.

Survey implementation is the process of carrying out the survey plan (package) for a given survey unit. This consists of scan measurements, fixed measurements, and collection and analysis of samples. Data will be stored using a data management system.

- **Optimize the Design for Obtaining Data**

  The first six steps are the DQOs that develop the performance goals of the survey. This final step in the DQO process leads to the development of an adequate survey design.

5.4.2 Classification of Survey Areas and Units

The adequacy of the final status survey process rests upon partitioning the site into properly classified survey units of appropriate physical area. Chapter 2 of this document discusses in detail the HSA for the HNP site and the classifications assigned to all of the site structures and grounds. Characterization is an ongoing effort throughout the decommissioning process, and survey unit classifications may be modified on the basis of new characterization information or impacts from decommissioning activities. The process described in Section 1.5 will be used to evaluate these changes. Survey areas have been determined as described in Section 2.3.3.2. The current approach is generally to remove the above-grade portions of site buildings and structures. Originally, the above-grade portions had been identified as survey areas and had been given MARSSIM classifications in Table 2-10. As final status survey activities are no longer planned for these areas, their survey area designations have been subsequently removed from Table 2-10. However, Appendix H contains the historical information from Table 2-10. If it becomes necessary to final status survey these areas, the survey area designation and initial classification listed in Appendix H will be used. <u>For the subsurface areas that will be evaluated using the basement fill model, area classifications and survey unit criteria do not apply.</u>

5.4.3 Survey Units

A survey area may consist of one or more survey units. A survey unit is a physical area consisting of structures or land areas of a specified size and shape which will be subject to a final status survey. Compliance with the applicable criteria will be demonstrated for each survey unit.

Survey units are limited in size based on classification, exposure pathway modeling assumptions, and site-specific conditions. The surface area limits, used in establishing the initial set of survey units for the HNP Final Status Survey Plan, are provided in Table 5-1 for structures and land areas. The area limits for structures refer to floor area, and not the total surface area, which would include the walls and ceiling. This is consistent with the guidance of DG-4006 (as incorporated in Section 2 of Appendix E to NUREG-1727) and MARSSIM. The floor area limits given in Table 5-1 were also used to establish survey unit sizes for structures such as roofs or exterior walls of buildings. The limits given in Table 5-1 will also be used should the need arise to establish any new survey units beyond the initial set given in this plan.

As indicated in Table 2-10 and 2-11A, B, and C, and Figures 2-1 through 2-9, areas of HNP that are classified as impacted have been divided into survey units to facilitate survey design. Each survey unit has been assigned an initial classification based on the site characterization process and the historical site assessment.

Table 5-1
HNP Survey Unit Surface Area Limits

Survey Unit Classification	Surface Area Limit
Class 1:	
Structures (floor area)	$\leq 100\ m^2$
Land areas	$\leq 2{,}000\ m^2$
Class 2:	
Structures (floor area)	$100\ m^2 \leq$ area $< 1{,}000\ m^2$
Land areas	$2{,}000\ m^2 \leq$ area $< 10{,}000\ m^2$
Class 3:	
Structures (floor area)	no limit
Land areas	no limit

A survey unit can have only one classification. Thus, situations may arise where it is necessary to create new survey units by subdividing areas within an existing unit. For example, residual radioactivity may be found within a Class 3 survey unit, or residual radioactivity in excess of the $DCGL_W$ may be found in a Class 2 unit. In such cases, it may be appropriate to define a new survey unit within the original unit that has a lower (more restrictive) classification. Alternately, the classification of the entire unit can be made more restrictive.

Likewise, survey units may need to be added or sub-divided to account for attached fixtures (such as cable trays, piping, and pipe hangers) that remain in an area after decommissioning activities are completed. The decision to define a new survey unit to account for attached fixtures can be made out of necessity, for compliance with the area limits from Table 5-1, or out of convenience to allow for a consistent survey approach within a given unit. If situations arise where it is neither necessary nor convenient to define additional survey units for attached fixtures, the fixtures will be considered to be part of the unit they are attached to. Attached fixtures and their impact on survey unit definitions cannot be addressed *a priori*, since major decommissioning activities are still ongoing.

5.4.4 Reference Coordinate Systems

The reference coordinate system depicted in Figures 5-1 and 5-2 will be used to provide a general reference for locations within a survey unit. This coordinate system will not be used to explicitly specify locations for fixed measurements or samples, but instead will serve as a convenience for documenting survey efforts and other information pertaining to a given survey unit. The coordinate system could also provide a means to specify general locations for measurements or samples performed for quality control or verification purposes.

At a minimum, each survey unit will have a benchmark defined that will serve as an origin for documenting survey efforts and results. This benchmark (origin) will be provided on the map or plot included in the final status survey package. Any coordinate systems used for surveys will typically take the form of a grid of intersecting, perpendicular lines; but other patterns (e.g., triangular and polar) may be used as convenient. Physical gridding of a survey unit will only be done in cases where it is beneficial and cost effective to do so. When physical gridding is used, benchmark locations will be designated by either marking a spot with surveyor's paint (or equivalent) for indoor areas or setting an iron pin (or

equivalent) for outdoor areas. If needed, grid lines or measurement locations will be marked (e.g., with chalk lines, paint, surveyor's flags), as appropriate. Global positioning systems will also be used as practical.

5.4.5 Reference Areas and Materials

The DQO process will be used during the planning phase in the preparation of a final status survey plan to determine whether media specific backgrounds, ambient area background or no background will be applied to a survey area or unit. The approach used for a specific survey unit will be based on the survey unit classification and the applicable DCGLs.

If applied, media specific backgrounds will be determined via measurements made in one or more reference areas and on various materials selected to represent the baseline radiological conditions for the site. The determination of media specific background will be controlled with a documented survey plan, which will include the DQO process. These data will be evaluated in a technical support document and available for inspection by the NRC. This process will ensure that the collected data meet the needs of the final status survey. The collected data may be used as the reference area data set when using the Wilcoxon Rank Sum test, or, for survey units with multiple materials, background data may be subtracted from survey unit measurements (using paired observations) if the Sign Test is applied.

Table 5-2 gives typical ranges for backgrounds expected to be encountered at the HNP during final survey activities. Ranges are given for several detector types (gross counters) and encompass the variability expected for different materials. The data in Table 5-2 are derived from both NUREG-1507 (Reference 5-6) and from experience at the HNP.

Table 5-2
Typical Media Specific Backgrounds

Instrument	Nominal Background Range
gas proportional counter (100 cm^2) α-only mode β-only mode α+β mode	 1 cpm - 20 cpm for ceramic tile; 1 cpm - 10 cpm for other materials 300 cpm - 1,250 cpm for all materials 280 cpm - 1,250 cpm for all materials
pancake GM probe (20 cm^2)	40 cpm - 125 cpm for all materials
ZnS (100 cm^2)	1 cpm - 10 cpm for ceramic tile; 1 cpm - 5 cpm for other materials
plastic scintillator (100 cm^2)	500 cpm - 1,500 cpm for all materials
NaI 1 inch by 1 inch 1.25 inch by 1.5 inch 2 inch by 2 inch	 2,000 cpm - 4,000 cpm for soil 3,000 - 6,000 cpm for soil 8,000 cpm - 16,000 cpm for soil

Depending on the values of the applicable DCGLs, an alternative method to using material specific backgrounds may be used during final status surveys. This alternative method will involve the determination of the ambient area background in the survey unit and will only be applicable to beta-gamma detecting instruments. This determination will be made prior to performing a final status survey

at a location within a survey area that is of sufficient distance (or attenuation) from the surfaces to eliminate beta particles originating from the surfaces from reaching the detector. At such a location, the ambient background radiation will be due only to ambient gamma radiation and will be a background component of all surface measurements. The average background determined at this location can be used as a conservative estimate since it is expected to be less than the material specific background for the material in the room because it does not fully account for the naturally occurring radioactivity in the materials. Using this lower ambient background will result in conservative calculated residual radioactivity levels. If the average background reading exceeds a predetermined value, the survey would be terminated and an investigation performed to determine and eliminate the reason for the elevated reading. Each of the survey unit readings would subtract this average background value and the Sign Test applied.

Whether or not they are radionuclide specific, all background measurements should account for both spatial variability over the area being assessed and the precision of the instrument or method being used to make the measurements. Thus, the same materials or areas may require more than one background assessment to provide the requisite background information for the various survey instruments or methods expected to be used for final status surveys. The result of these background assessments provides the basis for determining the mean and its associated standard deviation.

5.4.6 Area Preparation: Isolation and Control

Before final status survey activities can begin in an area, a transition must occur where planned decommissioning activities are completed and the area is subsequently assessed to scope the required isolation and control measures. This includes establishing if the area is ready for final survey activities and identifying any work practice issues that must be addressed in survey planning and design. Determination of readiness for final status survey will be based on characterization and/or remediation surveys indicating that the residual radioactive material is likely to comply with the final status survey criteria.

During and following this assessment, the remediation of an area for the purposes of removal of residual radioactivity, a Remediation Action Survey (RAS) will be performed. This RAS will include scanning and sampling of the areas, as necessary, using appropriate instrumentation to ensure the intended remediation has been accomplished. If no further remediation is required, a turnover survey may be performed as necessary. Data from the RAS may be used for the turnover survey and as input to the design and DQOs for the surface and subsurface FSS.

For areas where an excavation has resulted from the removal of materials not contaminated with residual radioactivity, and contamination has not been detected or suspected, a graded approach will be used to assess the excavation. Limited biased sampling (and scanning of suspect areas) will be conducted in the excavation. If contaminated material is detected, additional sampling will be conducted to determine the extent and magnitude of the contamination. Remediation and a Remedial Action Survey will be conducted as necessary. The characterization information from this assessment may be used for any turnover survey needed and as input to the design and DQOs for the surface and subsurface FSS.

5.4.6.1 Structures

The structures that remain (including below-ground portions of structures below elevation 17ft-6in) will be decontaminated and prepared for final status survey. Following a readiness assessment for final status survey, isolation and control measures will be implemented to prevent the introduction of plant-related contamination into soils or structures in the area, prior to, during or after final survey activities. These

control measures will include posting (e.g., with a placard or sign) areas that have been turned over for final status survey. Isolation and control measures are implemented for areas such as an entire structure. If additional remediation is required in an area following the implementation of isolation and control measures, local contamination control measures such as tents, HEPA filters, or vacuums will be employed as appropriate.

Prior to transitioning an area from decommissioning activities to isolation and control, a walkdown may be performed to identify access requirements and to specify the required isolation and control measures. The physical condition of the area will also be assessed, with any conditions that could interfere with final survey activities identified and addressed. If any support equipment needed for final survey activities, such as ladders or scaffolding, are in place, it will be evaluated to ensure that it does not pose the potential for introducing radioactive material into the area. Industrial safety and work practice issues, such as access to elevated areas requiring fall protection, will also be identified during the pre-survey evaluation. Operational health physics or decontamination support data, if available, will be reviewed to identify any potential areas where additional decontamination may be required prior to commencing final survey activities. In some instances, turnover surveys may be performed to verify that an area is ready for final survey.

The following criteria must be met for an area to be deemed ready for isolation and control:

- all planned decommissioning activities in the area are complete, including removal, as necessary, of items (e.g., equipment mounts, wall hangers, and exposed studs) that could interfere with final survey activities;
- all planned decommissioning activities in areas either adjacent to the area to be isolated or that could otherwise affect it are controlled using isolation and control techniques, are complete or are deemed not to have any reasonable potential to spread plant-related radioactive material to the area;
- all tools and equipment not needed for final survey activities are removed;
- any equipment to be used for final survey activities is evaluated to ensure it does not pose the potential for introducing plant-related radioactive material into the area; and
- where practical, transit paths to or through the area, except those required to support final survey activities, are eliminated or re-routed.

Once the area meets the isolation and control criteria, isolation and control will be achieved through:

- a combination of personnel training, physical barriers and postings, and site notices as appropriate, to prevent unauthorized access to an isolated area;
- implementation of provisions to prevent the introduction of plant-related radioactive material by persons authorized to enter the area; and
- measures to prevent the introduction of plant-related radioactive material through the air or through other paths, such as systems or piping.

Measures to prevent against the introduction of plant-related radioactive material by persons entering an isolated area may include personnel frisking stations at the entry point, the use of "sticky pads", or other such routine methods. Isolation from airborne material may include sealing off openings, including doors and ventilation ducts. Although not likely to be encountered, if a potential for waterborne material is deemed to exist (e.g., floor drains or penetrations left by decommissioning activities), similar measures will be taken to ensure such sources are sealed off from the isolated area. In addition to these physical controls, access points to buildings will be posted with signs providing contact information for approval to conduct decommissioning and demolition activities in the area. An administrative process will be used to evaluate, approve (or deny), and document all plant related activities conducted in these areas during and following final status surveys.

Following the final status survey, and any regulatory confirmation, the excavations associated with the structures will be backfilled with bulk fill material. Any isolation and control measures needed at the restored surface will be implemented to protect the area from contamination.

5.4.6.2 Open Land Areas

For open land areas, access roads and trails will be posted (as well as informational notices) with signs providing contact information for approval to conduct plant-related activities in the area. An administrative process will be used to evaluate, approve (or deny), and document all plant related activities conducted in these open land areas during and following final status surveys. Land areas will be inspected quarterly and any material that has been deposited since the last inspection will be investigated.

5.4.6.3 Excavation Land Areas Resulting from Radiological Remediation

These are land areas where there has been excavation for the purpose of radiological remediation of the soil. These areas will be posted with signs providing contact information for approval to conduct decommissioning and demolition activities in the area. An administrative process will be used to evaluate, approve (or deny), and document all plant related activities conducted in these excavations during and following final status surveys.

5.4.6.4 Bedrock

There are areas of the site where bedrock will be exposed as a result of building demolition and soil remediation. These areas include, but are not limited to, the Tank Farm area, the Spent Resin Facility and Ion Exchange Facility, and the RHR pit area of the Primary Auxiliary Building. Isolation and control of bedrock areas will be the same as for open land areas with added controls for deep excavation personnel safety requirements.

5.4.6.5 Excavations Resulting from the Removal of Piping Conduit

Areas that are excavated for the purposes of removing piping, conduit or other subsurface construction will be controlled to ensure personnel safety and to reduce the potential for plant-related activities to contaminate the area. These areas will be posted with signs providing contact information for approval to conduct decommissioning and demolition activities in the area. An administrative process will be used to evaluate and approve (or deny), and document all plant-related activities conducted in these excavations. Any isolation and control measures needed at the restored surface will be implemented to protect the area from contamination.

5.4.7 Selection of DCGLs

Residual levels of radioactive material that correspond to allowable radiation dose standards are calculated by analysis of various pathways (direct radiation, inhalation, ingestion, etc.), media (concrete, soils, and groundwater) and scenarios through which exposures could occur. These derived levels, known as ~~derived~~ Derived ~~concentration~~ Concentration ~~guideline~~ Guideline ~~levels~~ Levels (DCGLs), are presented in terms of surface or mass activity concentrations. DCGLs usually refer to average levels of radiation or radioactivity above appropriate background levels. DCGLs applicable to building or other structural surfaces are expressed in units of activity per surface area (dpm/100 cm^2). When applied to soil, sediments or structural materials where the radionuclides are distributed throughout, DCGLs are expressed in units of activity per unit of mass (pCi/g).

Chapter 6 of this plan describes in detail the modeling performed to develop the radionuclide-specific DCGLs for soil, groundwater and~~, concrete debris,~~ building surfaces. ~~, building foundations/basements and activated concrete~~. DCGLs are not needed for building footings (volumetrically contaminated), basements and activated concrete sources. Dose from these locations will be calculated using the Basement Fill Model. ~~These~~ For situations where gross activity measurement methods are used to demonstrate compliance with the license termination criteria, the radionuclide specific DCGLs will be used to establish gross activity DCGLs. These gross activity ~~values will be used to establish DCGLs for survey units in cases where measurements are made that are not radionuclide specific or when hard-to-detect radionuclides are present that necessitate the need for a surrogate radionuclide. In such cases, appropriate~~ DCGLs will be established based on a representative radionuclide mix established for each survey unit. In cases where measurable activity still exists, it is expected that the radionuclide mix will be established based on gamma-ray spectroscopy and alpha spectroscopy (where conditions warrant) or equivalent analyses on representative samples, with scaling factors used to establish the activity contribution for any hard-to-detect radionuclides that might be present. Scaling factors will be selected from available composite waste stream analyses or similar assays. Such analyses are performed periodically and documented in support of waste characterization needs.

For cases of survey units for which there is no measurable activity distinguishable from background, a representative radionuclide mix will be selected based upon historical characterization information for the survey unit of interest or for units with similar history and physical characteristics (e.g., information from adjacent areas).

~~Chapter 6 of this plan establishes the basis for the DCGLs in soil, and sediments, to be the resident farmer scenario, and the DCGLs for structures to be the building occupancy scenario. For structures, an additional scenario was evaluated for buried concrete debris and foundations/basements. This is described in Section 6.8.~~

To show compliance with 25 mrem/yr and ALARA, the unity rule will be applied in those areas in which the dose can be a result of soil, existing groundwater and future groundwater residual radioactivity. ~~both surface radioactivity bounded by the resident farmer—concrete debris scenario and resident farmer—soil scenario.~~ Use of the unity rule, as discussed in Section 5.4.7.1, will result in the development of operational DCGLs on a radionuclide-specific basis.

5.4.7.1 Operational DCGLs

The DCGLs are developed in Chapter 6 for exposures due to three potential media. These exposures include that from residual radioactivity in soil, existing groundwater (GW) radioactivity, and additional future groundwater radioactivity from the building basements and footings. ~~structures and the potential~~

~~burial of subsurface concrete structures and possibly concrete debris containing residual radioactivity.~~ The areas of the site where these exposures could occur concurrently are where subsurface structures and concrete are buried and existing groundwater contamination may be present. This area represents approximately 15,600 m^2 and includes the industrial area of the site. For this area, the total dose from these sources, H_{Total} can be expressed as:

$$H_{Total} = H_{Soil} + H_{Existing\,GW} + H_{FutureGW} \qquad \text{(Equation 5-1)}$$

For soil and existing groundwater~~these individual media~~, the dose from the residual radioactivity from radionuclide i is:

$$H^i = 25 * \frac{C^i}{DCGL^i} \qquad \text{(Equation 5-2)}$$

For the future groundwater pathway, the dose is determined from the basement fill model.

Since the limit for the total annual dose is 25 mrem from all media (and all pathways), a reduction to the soil and existing groundwater DCGLs in Chapter 6 is needed, since these are based on an annual dose of 25 mrem from each media. The DCGLs in Chapter 6 are therefore considered "Base-Case (Base)" values. The reduced DCGLs, or "Operational DCGLs" ($DCGL_{OP}$), can be related to the base case DCGLs using the principal relationship from:

$$H^i = 25 * \frac{DCGL^i_{OP}}{DCGL^i_{Base}} \qquad \text{(Equation 5-3)}$$

In the case of existing groundwater, the contamination concentration to be used for calculating dose is the highest measured at any point within the survey area or within the plume area boundary distance (largest capture zone radius as determined by the capture zone analysis described below) from the subject survey area at the time of notification of the NRC of intent to release the subject survey area from the license.

The following considerations may be included in determining if the results trend is sufficient to utilize the groundwater well sample results in the dose calculation for an affected survey unit:

- Fate and transport simulations will identify the projected area of highest groundwater concentration on site.
- The locations of existing wells will be examined in relation to the simulation results and additional wells constructed to ensure adequate monitoring of the area(s) of anticipated highest groundwater radionuclide Substances Of Concern (SOC) concentrations.
- Monitoring wells from which the sample results are to be used for the dose calculation for a survey unit will have been sampled quarterly for at least 18 months including two springtime high water table periods. In the case of areas where remediation (e.g., removal of contaminated soil below the average water table) has been conducted using groundwater depression, the 18 month monitoring period will begin. when use of the groundwater depression systems has ended. Prior

to turning off the depression system, remediation will have been completed and excavation backfilled.

- Monitoring well results show groundwater contaminant concentrations to be below closure criteria as discussed in this section, and exhibit steady or decreasing trends.

The 18 month monitoring period is sufficient for the following reasons:

- Historical releases at HNP and subsequent migration of groundwater contaminants appear to have resulted in dispersion of SOCs in groundwater. Actions completed to date have removed primary contaminant sources (e.g., contaminated process solutions) and processes (e.g., bulk waste water processing with leaking tanks) that historically contributed to observed groundwater contamination. As a result, only secondary contaminant sources (which could include residual subsurface soil contamination, grossly-contaminated groundwater and contaminated subsurface structures) remain at the site. The highest concentrations generally remain near historical source areas in wells that are completed within the unconsolidated soil formation that is slated for remediation.

- For all areas where groundwater contamination has been detected, this duration (when two springtime periods are included) ensures that the effect of the high water table season is included twice. Seasonal high water table levels impacting contaminated soils above the average water table level is one of the factors that can cause a seasonal increase in groundwater radionuclide concentrations.

- For areas where remediation has been conducted below the normal water table for the purpose of removing media suspected of contributing to groundwater contamination, the 18 month period (after the area has been backfilled and returned to normal groundwater levels) is expected to provide sufficient time for groundwater to leach through the remediated and backfilled area and for sampling of nearby monitoring wells to ensure the effectiveness of the remediation. As stated above, this will be confirmed by ensuring that the groundwater activity levels are steady or decreasing during this 18 month monitoring period.

~~In the case of potential future groundwater contamination from buried site building subsurface foundations/basements or concrete debris containing residual radioactivity, the dose to the resident farmer is limited to the water dependent pathways from the buried concrete debris scenario as described in Chapter 6. Therefore for the concrete debris case, the dose component for each radionuclide i is modified by the fraction of the total dose delivered from the water dependent pathway, f_w, as calculated using the information in Appendix G, Table G-4. This is provided~~ The future groundwater dose component of equation 5-1 can be further stated as follows:

$$H^{i}_{FutureGW} = FutureGroundwaterDose \quad \text{(Equation 5-4)}$$

For building basements and footings to remain, the future groundwater dose will be calculated by the Basement Fill Model. The dose calculation method future groundwater is discussed in Section 6.8.

The $H_{ExistingGW}$ term, from Equation 5-1, will be applied to survey areas in which the presence of groundwater contamination has been detected and survey areas that are within the plume influence

boundary distance of detectable ground contamination. "Detected groundwater contamination" is defined as the presence of:

- Plant-related radionuclides, which are also present in background, at a concentration greater than two standard deviations over background, or

- Plant-related radionuclides, not present in background, at a concentration greater than the Minimum Detectable Concentration and greater than two times the standard deviation in the net concentration.

Table 5-3 provides the survey areas to which the $H_{ExistingGW}$ term would currently be applied. Table 5-3 is based upon the groundwater plume (locations where tritium concentration was measured/predicted as 1000 pCi/liter or greater) as depicted in the 1999 Malcolm Pirnie Groundwater Monitoring Report, including a plume influence boundary distance of 100 meters (Figure 5-3).

It is noted, however, that characterization efforts for groundwater contamination are still ongoing and the survey areas to which the $H_{ExistingGW}$ term are applied may change. Those changes will be communicated to the NRC. This change may be caused by changes in the location of the plume, detection of groundwater contamination at locations outside the plume or changes to the plume influence boundary distance. The Phase 2 Hydrogeologic Work Plan, as described in Section 2.3.3.1.6, will provide additional characterization of groundwater that will be used to better define the groundwater contamination plume.

Prior to the request to release any portion of the site from the license, CYAPCO will prepare and make available for inspection a capture zone analysis based on data collected as part of the Phase 2 Hydogeological Work Plan, to better define the plume influence boundary distance. The "capture zone" is the area surrounding a hypothetical well to be used by the resident farmer, from which existing groundwater contamination could be drawn into the resident farmer's well. The analysis used to determine this area will use the hydrogeological conditions and parameters assumed in the Resident Farmer Scenario as described in Chapter 6 of the LTP. If this capture zone analysis (using the well pumping rate assumed in the dose calculations presented in Chapter 6) determines that the maximum capture zone radius is greater than 100 meters, the NRC will be notified.

Table 5-3
Survey Areas Affected by Groundwater Contamination

Survey Area		
9102	9120	9312
9104	9126	9313
9106	9128	9502
9108	9302	9512
9110	9304	9514
9112	9306	9518
9114	9307	9520
9116	9308	9522
9118	9310	

The compliance formulation for these resident farmer exposure scenarios is re-written as:

(Equation 5-5)

$$1 \geq \sum(i)\left[\frac{DCGL^i_{OP-Soil}}{DCGL^i_{Base-Soil}} + \frac{DCGL^i_{OP-ExistingGW}}{DCGL^i_{Base-ExistingGW}} + \frac{H_{FutureGW}}{25}\right]$$

$$\cancel{1 \geq \frac{DCGL^i_{OP-Soil}}{DCGL^i_{Base-Soil}} + \frac{DCGL^i_{OP-ExistingGW}}{DCGL^i_{Base-ExistingGW}} + FutureGroundwaterL}$$

~~(Equation 5-5)~~

For simplicity Equation 5-5 may be re-written as:

$$1 \geq \sum(i)\left[f^i_{Soil} + f^i_{ExistingGW} + f^i_{FutureGW}\right] \quad \text{(Equation 5-6)}$$

where, for a given radionuclide i, f^i_{Soil} is the fraction of the total dose from soil, $f^i_{ExistingGW}$ is the fraction of the total dose from existing contamination in groundwater, and $f^i_{FuturegGW}$ is the fraction of the 25 mrem dose that is calculated by the Basement Fill Model.

$$\cancel{1 \geq f^i_{Soil} + f^i_{ExistingGW} + FutureGroundwaterDose} \quad \text{~~(Equation 5-6)~~}$$

~~where, for a given radionuclide i, f^i_{Soil} is the fraction of the total dose from soil, $f^i_{ExistingGW}$ is the fraction of the total dose from existing contamination in groundwater, the $f^i_w f^i_{ConcreteDebris}$ is the fraction of the total dose from the water dependent pathways from subsurface foundation/basement or concrete debris.~~

The use of this equation requires that only one variable be unknown. Therefore, values for Future Groundwater Dose ~~f_{Soil}~~ and $f_{ExistingGW}$ will need to ~~be known or~~ be selected in order to calculate f_{Soil} ~~the building surface operational DCGL~~. As t~~The~~e ~~final selection of the~~ building surface operational DCGL are independent of soil and groundwater dose contribution, they will be set based on an ALARA evaluation and/or an administrative dose level at or below 25 mrem/yr. ~~the lower of the DCGLs determined from Equation 5-5 or the building occupancy DCGL from Chapter 6. For areas of surface contamination, this comparison will be performed using the surface contamination building occupancy DCGLs, listed in Table 6-3, and concrete debris DCGLs in units of dpm/100 cm² given in Table 6-5. For areas that are volumetrically contaminated, this comparison will be performed using the volumetric contamination building occupancy DCGLs, as listed in Table 6-3, and concrete debris DCGLs in units of~~

~~pCi/g, as listed in Table 6-4. Like the base-case DCGLs, the operational DCGL will be different for each nuclide.~~

The following example is provided to illustrate the use of the operational DCGLs for ~~surficial contamination~~land areas that have the potential for existing and future groundwater dose, with the following assumptions:

- $f_{soil} = 0.3$
- $f_{existingGW} = 0.2$
- Therefore, ~~Therefore, f_w $f_{Concrete\ Debris}$~~Future Groundwater Dose (fraction of 25 mrem/yr) = 0.5

~~Using these values, the operational DCGLs for the buried building debris are calculated as 0.5 of the base-case concrete debris values from Chapter 6 and compared to the building occupancy DCGLs also in Chapter 6. In this final comparison, the more restrictive DCGL will be used as the operational DCGL for building surfaces.~~

The above determination will be made prior to the performance of any final status surveys of soils or building surveys in areas where existing groundwater contamination will impact the potential dose. This determination will be provided in the FSS Report or a technical support document and will be applied to the affected survey areas.

The following table provides an example of the building surface operational DCGL for Cs-137 using the fractional values from above.

Table 5-4
Operational DCGL Example for Cs-137
Using Fractional Values from Above ~~and Assuming f_w = 0.354~~

	Base Case DCGL/Dose	Operational DCGL/Dose
Soil (pCi/g)	7.91E+00	2.37E+00
Existing Groundwater (pCi/l)	4.31E+02	8.62E+01
~~Concrete Debris[1] (dpm/100cm²)~~Future Groundwater Dose (mrem/yr)	~~8.31E+06~~25	~~1.17E+07[1]~~12.5
~~Building Occupancy (dpm/100cm2)~~	~~4.30E+04~~	[redacted]
~~Selected Building Surface Operational DCGL~~	[redacted]	~~4.30E+04~~

1 $$\cancel{DCGL^{Cs-137}_{OP-ConcreteDebris}} = \frac{0.5 \times 8.31E+06}{0.354} = \cancel{1.17E+07\ dpm/100cm^2}$$

5.4.7.2 Gross Activity DCGLs

For alpha or beta surface activity measurements, field measurements will typically consist of gross activity assessments rather than radionuclide-specific techniques. Gross activity DCGLs will be established, based on the representative radionuclide mix, as follows:

$$DCGL_{GA} = \frac{1}{\sum_{1}^{n} \frac{f_i}{DCGL_i}} \qquad \text{(Equation 5-7)}$$

where:

f_i = fraction of the total activity contributed by radionuclide i
i = the number of radionuclides
$DCGL_i$ = DCGL for radionuclide i

Gross activity DCGLs can be developed for gross beta measurements, or a gross beta DCGL can be scaled so that it acts as a surrogate for gross alpha (see Section 5.4.7.3). Equation 5-7 will be applied for radionuclides that are present in a survey unit in concentrations greater than 5% of their respective DCGL. The aggregate of all radionuclides not included in the gross activity DCGL, based on the percentage of their respective DCGL, will not exceed 10%. This practice is conservative relative to the process presented in 10CFR20 in which radionuclides that contribute less than 10% to dose, provided the aggregate does not exceed 30%, and are not required to be included in the dose assessment.

5.4.7.3 Surrogate Ratio DCGLs

It is acceptable industry practice to assay a Hard-To-Detect (HTD) radionuclide by using a surrogate relationship to an Easy-To-Detect (ETD) radionuclide. A common example would be to use a beta measurement to assay an alpha emitting radionuclide. Another example would be to relate a specific radionuclide, such as cesium-137, to one or more radionuclides of similar characteristics. In such cases, to demonstrate compliance with the release criteria for the survey unit the DCGL for the surrogate radionuclide or mix of radionuclides must be scaled to account for the fact that it is being used as an indicator for an additional radionuclide or mix of radionuclides. The result is referred to as the surrogate DCGL.
The following process will be applied to assess the need to use surrogate ratios for final status surveys (FSS).

- Determine whether HTD radionuclides (e.g., TRU, Sr-90, H-3) are likely to be present in the survey unit based on process knowledge, historical data or characterization.

- When HTD radionuclides are likely to be present establish a relationship using a representative number of samples (typically six or more). The samples may come from another survey unit if the source of the contamination and expected concentrations are reasonably the same. These samples will be analyzed for ETD and HTD radionuclides using gross alpha , alpha spectroscopy, gross beta analysis or gamma spectroscopy techniques.

- Screen HTD radionuclides using the 5% and 10% rule described in Section 5.4.7.2.

Radionuclides not screened out will require a surrogate DCGL. Surrogate relationships will be determined from the samples results using one of methods described below.

- Develop a surrogate relationship for each HTD radionuclide.

$$DCGL_{surrogate} = DCGL_{ETD}\, x \frac{DCGL_{HTD}}{(f_{HTD:ETD} \times DCGL_{ETD}) + DCGL_{HTD}}$$ *(Equation 5-8)*

- Determine the average surrogate DCGL and the standard deviation from the surrogate relationships.

If the %CV (coefficient of variation) of the average surrogate DCGL is within 25% then the average surrogate DCGL will be applied to the survey area. The %CV is the percent ratio of the standard deviation to the average surrogate DCGL. If this criterion is not met, the following steps will be applied.

  - Following a more detailed spatial analysis of the radionuclide mix distribution, the unit may be subdivided into separate survey units based on the spatial distribution.
  - The lowest surrogate DCGL from the observed radionuclide mix may be applied to the entire survey unit.
  - Additional samples may be collected and analyzed to allow for a detailed analysis and documented evaluation of the radionuclide distribution resulting in the use of a specific DCGL for the survey unit.

–

- The surrogate DCGL may be computed from a simple recurrence formula as:

$$\frac{C_{ETD}}{DCGL_{Surrogate}} = \frac{C_{ETD}}{DCGL_{ETD}} + \frac{C_1}{DCGL_1} + \frac{C_2}{DCGL_2} + \ldots + \frac{C_i}{DCGL_i}$$ *Equation (5-9)*

or, for simplification

$$\frac{C_E}{D_{Surrogate}} = \frac{C_E}{D_E} + \frac{C_1}{D_1} + \frac{C_2}{D_2} + \ldots + \frac{C_i}{D_i}$$ *Equation (5-10)*

where:

D_E ≡the DCGL for the easy-to-detect radionuclide
D_1 ≡the DCGL for the first hard-to-detect radionuclide
D_2 ≡the DCGL for the second hard-to-detect radionuclide
D_i ≡the DCGL for the ith hard-to-detect radionuclide
f_1 ≡the activity ratio of the first hard-to-detect radionuclide to the easy-to-detect radionuclide
f_2 ≡the activity ratio of the second hard-to-detect radionuclide to the easy-to-detect radionuclide
f_i ≡the activity ratio of the ith hard-to-detect radionuclide to the easy-to-detect radionuclide

Consider the case of three HTD radionuclides from which a surrogate will be calculated.

$$DCGL_{Surrogate} = \frac{(D_E D_1 D_2 D_3)}{(D_1 D_2 D_3)+(f_1 D_E D_2 D_3)+(f_2 D_E D_1 D_3)+(f_3 D_E D_1 D_2)}$$ Example 5-1

A general expression for the surrogate equation based on recursive relationships is provided by Equation 5.11 for n HTD radionuclides.

$$DCGL_{Surrogate} = \frac{1}{1/D_E + \sum_{i=1}^{n} f_i / D_i}$$ Equation 5.11

5.4.7.4 Elevated Measurement Comparison (EMC) DCGLs

The DCGL established for the average residual contamination in a survey unit is $DCGL_W$. Values of the $DCGL_W$ may be scaled through the use of area factors to obtain a DCGL that represents the same dose to an individual from residual contamination over a smaller area within a survey unit. Such a value is called $DCGL_{EMC}$, where the subscript EMC stands for elevated measurement comparison. The $DCGL_{EMC}$ is defined as the product of the applicable $DCGL_W$ and a correction factor know as the area factor.

The area factor is equal to the ratio of the dose from the base-case contaminated area to the dose from a smaller contaminated area with the same radioactive source concentration. Area factors are required for both the resident farmer and the building occupancy scenarios. Area factors for both the resident farmer and building occupancy scenarios have been calculated (Reference 5-7) for the radionuclides of concern at the HNP site considering all applicable potential pathways of exposure.

For the resident farmer scenario, RESRAD (Version 5.91) was used to determine area factors. For the building occupancy scenario, RESRAD-BUILD (Version 2.37) was used to determine area factors. Area factors will not be computed for areas smaller than 1 m^2 for either the resident farmer or the building occupancy scenarios.

Table 5-5 summarizes the outputs of the RESRAD code for the radionuclides of concern for the Resident Farmer scenario. Table 5-6 summarizes the outputs of the RESRAD-BUILD code for the Building Occupancy scenario.

Table 5-5
Area Factors for the Resident Farmer Scenario

	Size of Elevated Area (m²)																			
	15600	13000	10000	7500	5000	2500	1000	750	500	250	100	75	50	25	10	8	6	4	2	1
H-3	1.00E+00	1.09E+00	1.22E+00	1.35E+00	1.51E+00	1.71E+00	1.87E+00	2.34E+00	3.21E+00	5.45E+00	1.06E+01	1.28E+01	1.69E+01	2.62E+01	5.42E+01	6.76E+01	9.00E+01	1.35E+01	2.67E+02	5.28E+02
C-14	1.00E+00	1.19E+00	1.50E+00	1.90E+00	2.58E+00	4 08E+00	6 89E+00	1.06E+01	1.92E+01	5.34E+01	2.03E+02	3.08E+02	5.52E+02	1.48E+03	5.26E+03	7.12E+03	1.05E+04	1.79E+04	4.33E+04	1.01E+05
Mn-54	1.00E+00	1.00E+00	1.01E+00	1.01E+00	1.02E+00	1.04E+00	1.06E+00	1.10E+00	1.14E+00	1.21E+00	1.32E+00	1.38E+00	1.46E+00	1.71E+00	2.28E+00	2.59E+00	3.07E+00	3.93E+00	6.19E+00	1.04E+01
Fe-55	1.00E+00	1.16E+00	1.44E+00	1.79E+00	2.36E+00	3.48E+00	4 85E+00	6.47E+00	9.71E+00	1.94E+01	4.85E+01	6.46E+01	9 68E+01	1.93E+02	4.80E+02	5.99E+02	7.95E+02	1.18E+03	2.32E+03	4.46E+03
Co-60	1.00E+00	1.02E+00	1.03E+00	1.06E+00	1.08E+00	1.11E+00	1.14E+00	1.18E+00	1.22E+00	1.30E+00	1.41E+00	1.47E+00	1.55E+00	1.82E+00	2.44E+00	2.78E+00	3.29E+00	4.22E+00	6.67E+00	1.13E+01
Ni-63	1.00E+00	1.15E+00	1.40E+00	1.71E+00	2.18E+00	3.03E+00	3.94E+00	5.25E+00	7.87E+00	1.57E+01	3.94E+01	5.25E+01	7.87E+01	1.57E+02	3.93E+02	4 92E+02	6.56E+02	9 83E+02	1.96E+03	3.92E+03
Sr-90	1.00E+00	1.05E+00	1.11E+00	1.16E+00	1.22E+00	1.29E+00	1.33E+00	1.77E+00	2.66E+00	5.31E+00	1.32E+01	1.76E+01	2.63E+01	5.22E+01	1.28E+02	1.59E+02	2.11E+02	3.14E+02	6.16E+02	1.20E+03
Nb-94	1.00E+00	1.00E+00	1.01E+00	1.01E+00	1.02E+00	1.04E+00	1.06E+00	1.07E+00	1.08E+00	1.12E+00	1.20E+00	1.25E+00	1.31E+00	1.53E+00	2.03E+00	2.31E+00	2.74E+00	3.50E+00	5.50E+00	9.24E+00
Tc-99	1.00E+00	1.02E+00	1.04E+00	1.06E+00	1.07E+00	1.09E+00	1.10E+00	1.47E+00	2.21E+00	4.42E+00	1.10E+01	1.47E+01	2.21E+01	4.41E+01	1.10E+02	1.38E+02	1.84E+02	2.76E+02	5.51E+02	1.10E+03
Ag-108m	1.00E+00	1.00E+00	1.01E+00	1.01E+00	1.02E+00	1.04E+00	1.06E+00	1.07E+00	1.08E+00	1.12E+00	1.20E+00	1.25E+00	1.31E+00	1.53E+00	2.02E+00	2.30E+00	2.72E+00	3.48E+00	5.47E+00	9.17E+00
Cs-134	1.00E+00	1.06E+00	1.14E+00	1.22E+00	1.31E+00	1.43E+00	1.53E+00	1.63E+00	1.74E+00	1.92E+00	2.14E+00	2.24E+00	2.37E+00	2.78E+00	3.71E+00	4.22E+00	5.00E+00	6 40E+00	1.01E+01	1.69E+01
Cs-137	1.00E+00	1.08E+00	1.20E+00	1.31E+00	1.46E+00	1.65E+00	1.80E+00	1.98E+00	2.21E+00	2.54E+00	2.93E+00	3.08E+00	3.28E+00	3.89E+00	5.20E+00	5.92E+00	7.01E+00	8.98E+00	1.41E+01	2.38E+01
Eu-152	1.00E+00	1.00E+00	1.01E+00	1.01E+00	1.02E+00	1.04E+00	1.06E+00	1.07E+00	1.08E+00	1.12E+00	1.20E+00	1.25E+00	1.31E+00	1.54E+00	2.04E+00	2.32E+00	2.75E+00	3.52E+00	5.54E+00	9.32E+00
Eu-154	1.00E+00	1.00E+00	1.00E+00	1.01E+00	1.02E+00	1.04E+00	1.06E+00	1.07E+00	1.08E+00	1.12E+00	1.20E+00	1.25E+00	1.32E+00	1.54E+00	2.06E+00	2.34E+00	2.77E+00	3.55E+00	5.61E+00	9.50E+00
Eu-155	1.00E+00	1.00E+00	1.00E+00	1.01E+00	1.02E+00	1.04E+00	1.06E+00	1.06E+00	1.07E+00	1.10E+00	1.17E+00	1.22E+00	1.27E+00	1.47E+00	1.91E+00	2.16E+00	2.55E+00	3.23E+00	4.96E+00	7.99E+00
Pu-238	1.00E+00	1.00E+00	1.01E+00	1.01E+00	1.01E+00	1.01E+00	1.02E+00	1.35E+00	2.02E+00	3.99E+00	9.69E+00	1.27E+01	1.85E+01	3.44E+01	7.14E+01	8.36E+01	1.01E+02	1.29E+02	1.82E+02	2.35E+02
Pu-239	1.00E+00	1.00E+00	1.01E+00	1.01E+00	1.01E+00	1.01E+00	1.02E+00	1.35E+00	2.02E+00	4 00E+00	9.70E+00	1.27E+01	1.86E+01	3.44E+01	7.15E+01	8.39E+01	1.02E+02	1.30E+02	1.83E+02	2.37E+02
Pu-241	1.00E+00	1.00E+00	1.00E+00	1.01E+00	1.01E+00	1.01E+00	1.01E+00	1.34E+00	1.99E+00	3.85E+00	8.82E+00	1.13E+01	1.57E+01	2 65E+01	4 83E+01	5 60E+01	6.71E+01	8.53E+01	1.23E+02	1.70E+02
Am-241	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.01E+00	1.01E+00	1.01E+00	1.34E+00	1.99E+00	3.84E+00	8.82E+00	1.13E+01	1.57E+01	2.65E+01	4.81E+01	5.58E+01	6 69E+01	8.51E+01	1.23E+02	1.70E+02
Cm-243	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.01E+00	1.01E+00	1.02E+00	1.25E+00	1.64E+00	2 40E+00	3.44E+00	3.79E+00	4.26E+00	5.28E+00	7.26E+00	8.26E+00	9.79E+00	1.25E+01	1.94E+01	3.14E+01

Table 5-6
Area Factors for the Building Occupancy Scenario

	Size of Elevated Area (m^2)									
	100	75	50	25	10	8	6	4	2	1
H-3	1.0	1.3	2.0	4.0	10.0	12.5	16.6	25.0	50.1	100.0
C-14	1.0	1.3	2.0	4.0	10.0	12.5	16.7	25.0	50.1	100.0
Mn-54	1.0	1.1	1.2	1.6	2.4	2.8	3.3	4.2	7.1	12.6
Fe-55	1.0	1.3	2.0	4.0	10.0	12.5	16.7	25.1	49.9	100.0
Co-60	1.0	1.1	1.2	1.6	2.5	2.8	3.3	4.3	7.1	12.7
Ni-63	1.0	1.3	2.0	4.0	10.0	12.5	16.7	25.0	50.2	100.0
Sr-90	1.0	1.3	1.9	3.3	7.0	8.4	10.6	14.8	27.0	50.5
Nb-94	1.0	1.1	1.2	1.6	2.4	2.8	3.3	4.3	7.1	12.6
Tc-99	1.0	1.3	2.0	4.0	10.0	12.5	16.7	25.1	50.2	100.0
Ag-108m	1.0	1.1	1.2	1.6	2.4	2.8	3.3	4.3	7.1	12.7
Cs-134	1.0	1.1	1.3	1.6	2.5	2.8	3.4	4.4	7.3	13.1
Cs-137	1.0	1.1	1.3	1.7	2.6	2.9	3.5	4.5	7.5	13.4
Eu-152	1.0	1.1	1.2	1.6	2.5	2.8	3.3	4.3	7.2	12.7
Eu-154	1.0	1.1	1.2	1.6	2.4	2.8	3.3	4.3	7.1	12.7
Eu-155	1.0	1.1	1.3	1.7	2.6	2.9	3.5	4.5	7.5	13.4
Pu-238	1.0	1.3	2.0	4.0	10.0	12.5	16.6	24.9	49.7	99.5
Pu-239	1.0	1.3	2.0	4.0	10.0	12.5	16.7	24.9	49.8	99.5
Pu-241	1.0	1.3	2.0	4.0	10.0	12.5	16.7	25.0	50.1	100.0
Am-241	1.0	1.3	2.0	3.9	9.7	12.1	16.0	23.8	47.2	93.7
Cm-243	1.0	1.3	1.9	3.8	9.0	11.0	14.5	21.2	40.8	79.6

5.4.7.5 Building Basement and Footings

After completion of final status survey activities of the remaining portions of structures, some subsurface concrete~~-debris~~ may remain in the form of building footings.

As these structures are solid concrete or steel structures, and will not be left in a condition to allow them to realistically be occupied, the only applicable dose pathway is ~~due to~~from groundwater contamination from the leaching of radionuclides from these structures. The dose model for the calculation of this "future groundwater" dose is called the Basement Fill Model and is discussed in Section 6.8. The sampling to be performed to determinat~~ion~~e the radioactivity inventory to be used in calculating future groundwater dose is discussed in Section 5.7.1.6.

5.4.7.6 Release Limits for Non-Structural Components and Systems

In general, non-structural components and systems will be surveyed to site unconditional release limits, i.e., no detectable radioactive (licensed) material. These surveys will be performed in accordance with health physics procedures and are consistent with the requirements of NRC Information Notice 85-92, "Surveys of Wastes Before Disposal From Nuclear Reactor Facilities," and IE Circular 81-07, "Control of Radioactively Contaminated Material." Separate limits will be applied at the time of Final Status Survey to the buried piping located in the saturated subsurface areas of the site and to embedded piping and penetrations. These limits are discussed in the following paragraphs.

For buried piping in contact with the saturated zone, an analysis has been performed to determine surface activity limits for the remaining piping that will result in no more than a 1 mrem/yr dose (Reference 5-8). This piping will be grouted with concrete (after any required remediation and surveying), as agreed to with the State of Connecticut DPUC. To simplify the analysis, the piping material is assumed to be eroded away, leaving the slug of grout with the contamination from the interior surface of the piping. Consistent with these simplified assumptions, the DCGLs calculated in Chapter 6 -of -the LTP approved in November 2002 for concrete debris are used in developing the surface contamination limits for this piping.

In order to calculate the release limits for the piping (corresponding to 1 mrem/yr), first, for each radionuclide, the DCGL representing 25 mrem/yr from all pathways for concrete debris and the fraction of dose from the water dependent pathways were used to determine the volumetric limits from water dependent pathways only (as the buried piping is well below the soil surface, thus eliminating external dose contribution, and is in contact with the groundwater). These limits are then normalized to represent a volumetric limit that would result in 1 mrem/yr. Finally, the volumetric contamination is converted to surface contamination~~, assuming a 4-inch diameter~~ for various piping diameter sizes, (bounding value for the pipe diameters in question, because the larger the diameter, and subsequently the radius, the larger the surface activity limits can be). The release limits to be applied to this piping are given in Table 5-7. The surface contamination levels for the various piping sizes when converted to the volumetric contamination based on the grouting of the piping does not change the effective volumetric concentrations being left. This is a result of using the volumetric limit that results in 1 mrem/yr dose in pCi/gm to scale the surface contamination limits for various piping sizes.

DELETE AND REPLACE WITH NEW TABLE 5-7

Table 5-7
Release Limits For Remaining Buried Piping

Radionuclide	Surface Limit Resulting in 1 mrem/yr Dose (dpm/100cm^2)
H-3	5.21E+03
C-14	7.77E+04
Mn-54	5.31E+04
Fe-55	6.17E+04
Co-60	3.21E+05
Ni-63	1.52E+05
Sr-90	1.87E+02
Nb-94	1.37E+05
Tc-99	2.44E+04
Ag-108m	1.37E+06
Cs-134	8.35E+04
Cs-137	9.66E+04
Eu-152	2.68E+05
Eu-154	1.87E+05
Eu-155	1.20E+06
Pu-238	7.50E+02
Pu-239	6.82E+02
Pu-241	1.14E+04
Am-241	3.33E+02
Cm-243	4.61E+02

Embedded pipe represents medium- to large-bore penetrations (up to 42-inch) or small-bore piping (4-inch to 12-inch) that was built into concrete walls and run through structures including walls, ceilings and floors. The length of the piping for each segment is short, approximately the length of the thickness of the structure that the pipe penetrates, and in most cases it is expected to communicate perpendicular to the surface penetrated. The total length of this type of pipe has been estimated to be less than 1000 feet, segregated into a substantial number of individual segments.

Where the gross activity beta-to-alpha ratio at the time of FSS is 15:1 or greater, the piping will be left in place, and the building surface DCGLs will be applied during FSS. The basis and rationale for applying these DCGLs to embedded pipe are provided below:

- It is unlikely that access to piping 24 inches or less in diameter could occur.
- The majority of piping and penetration lengths greater than 24 inches in diameter are either run vertically (i.e., run through floor or ceiling) or are located six feet or more above the floor elevation. Thus it is unlikely that access to these pipes and penetrations would occur.

New Table

Table 5-7

Release Limits for Buried Piping, dpm/100cm2

Radionuclide	1"	2"	2 1/2 "	3"	4"	8"	10"	12"	14"	16"	18"	20"	24"	26"
H-3	1.30E+03	2.61E+03	3.26E+03	3.91E+03	5.21E+03	1.04E+04	1.30E+04	1.56E+04	1.82E+04	2.08E+04	2.34E+04	2.61E+04	3.13E+04	8.47E+03
C-14	1.94E+04	3.89E+04	4.86E+04	5.83E+04	7.77E+04	1.55E+05	1.94E+05	2.33E+05	2.72E+05	3.11E+05	3.50E+05	3.89E+05	4.66E+05	1.26E+05
Mn-54	1.33E+04	2.66E+04	3.32E+04	3.98E+04	5.31E+04	1.06E+05	1.33E+05	1.59E+05	1.86E+05	2.12E+05	2.39E+05	2.66E+05	3.19E+05	8.63E+04
Fe-55	1.54E+04	3.09E+04	3.86E+04	4.63E+04	6.17E+04	1.23E+05	1.54E+05	1.85E+05	2.16E+05	2.47E+05	2.78E+05	3.09E+05	3.70E+05	1.00E+05
Co-60	8.03E+04	1.61E+05	2.01E+05	2.41E+05	3.21E+05	6.42E+05	8.03E+05	9.63E+05	1.12E+06	1.28E+06	1.44E+06	1.61E+06	1.93E+06	5.22E+05
Ni-63	3.80E+04	7.60E+04	9.50E+04	1.14E+05	1.52E+05	3.04E+05	3.80E+05	4.56E+05	5.32E+05	6.08E+05	6.84E+05	7.60E+05	9.12E+05	2.47E+05
Sr-90	4.68E+01	9.35E+01	1.17E+02	1.40E+02	1.87E+02	3.74E+02	4.68E+02	5.61E+02	6.55E+02	7.48E+02	8.42E+02	9.35E+02	1.12E+03	3.04E+02
Nb-94	3.43E+04	6.85E+04	8.56E+04	1.03E+05	1.37E+05	2.74E+05	3.43E+05	4.11E+05	4.80E+05	5.48E+05	6.17E+05	6.85E+05	8.22E+05	2.23E+05
Tc-99	6.10E+03	1.22E+04	1.53E+04	1.83E+04	2.44E+04	4.88E+04	6.10E+04	7.32E+04	8.54E+04	9.76E+04	1.10E+05	1.22E+05	1.46E+05	3.97E+04
Ag-108m	3.43E+05	6.85E+05	8.56E+05	1.03E+06	1.37E+06	2.74E+06	3.43E+06	4.11E+06	4.80E+06	5.48E+06	6.17E+06	6.85E+06	8.22E+06	2.23E+06
Cs-134	2.09E+04	4.18E+04	5.22E+04	6.26E+04	8.35E+04	1.67E+05	2.09E+05	2.51E+05	2.92E+05	3.34E+05	3.76E+05	4.18E+05	5.01E+05	1.36E+05
Cs-137	2.42E+04	4.83E+04	6.04E+04	7.25E+04	9.66E+04	1.93E+05	2.42E+05	2.90E+05	3.38E+05	3.86E+05	4.35E+05	4.83E+05	5.80E+05	1.57E+05
Eu-152	6.70E+04	1.34E+05	1.68E+05	2.01E+05	2.68E+05	5.36E+05	6.70E+05	8.04E+05	9.38E+05	1.07E+06	1.21E+06	1.34E+06	1.61E+06	4.36E+05
Eu-154	4.68E+04	9.35E+04	1.17E+05	1.40E+05	1.87E+05	3.74E+05	4.68E+05	5.61E+05	6.55E+05	7.48E+05	8.42E+05	9.35E+05	1.12E+06	3.04E+05
Eu-155	3.00E+05	6.00E+05	7.50E+05	9.00E+05	1.20E+06	2.40E+06	3.00E+06	3.60E+06	4.20E+06	4.80E+06	5.40E+06	6.00E+06	7.20E+06	1.95E+06
Pu-238	1.88E+02	3.75E+02	4.69E+02	5.63E+02	7.50E+02	1.50E+03	1.88E+03	2.25E+03	2.63E+03	3.00E+03	3.38E+03	3.75E+03	4.50E+03	1.22E+03
Pu-239	1.71E+02	3.41E+02	4.26E+02	5.12E+02	6.82E+02	1.36E+03	1.71E+03	2.05E+03	2.39E+03	2.73E+03	3.07E+03	3.41E+03	4.09E+03	1.11E+03
Pu-241	2.85E+03	5.70E+03	7.13E+03	8.55E+03	1.14E+04	2.28E+04	2.85E+04	3.42E+04	3.99E+04	4.56E+04	5.13E+04	5.70E+04	6.84E+04	1.85E+04
Am-241	8.33E+01	1.67E+02	2.08E+02	2.50E+02	3.33E+02	6.66E+02	8.33E+02	9.99E+02	1.17E+03	1.33E+03	1.50E+03	1.67E+03	2.00E+03	5.41E+02
Cm-243	1.15E+02	2.31E+02	2.88E+02	3.46E+02	4.61E+02	9.22E+02	1.15E+03	1.38E+03	1.61E+03	1.84E+03	2.07E+03	2.31E+03	2.77E+03	7.49E+02
Totals	1.01E+06	2.03E+06	2.53E+06	3.04E+06	4.05E+06	8.10E+06	1.01E+07	1.22E+07	1.42E+07	1.62E+07	1.82E+07	2.03E+07	2.43E+07	6.58E+06

Embedded pipe represents medium- to large-bore penetrations (up to 42-inch) or small-bore piping (4-inch to 12-inch) that was built into concrete walls and run through structures including walls, ceilings and floors. The length of the piping for each segment is short, approximately the length of the thickness of the structure that the pipe penetrates, and in most cases it is expected to communicate perpendicular to the surface penetrated. The total length of this type of pipe has been estimated to be less than 1000 feet, segregated into a substantial number of individual segments.

Where the gross activity beta-to-alpha ratio at the time of FSS is 15:1 or greater, the piping will be left in place, and the building surface DCGLs will be applied during FSS. The basis and rationale for applying these DCGLs to embedded pipe are provided below:

- It is unlikely that access to piping 24 inches or less in diameter could occur.
- The majority of piping and penetration lengths greater than 24 inches in diameter are either run vertically (i.e., run through floor or ceiling) or are located six feet or more above the floor elevation. Thus it is unlikely that access to these pipes and penetrations would occur.
- An evaluation of the doses associated with accessing the piping and penetration was performed using a conservative radionuclide mixture where the gross activity beta-to-alpha ratio is 15:1 (Reference 5-9). Based upon the information contained in HNP waste stream characterization data, this mixture is expected to bound those conditions found at the site. This mixture corresponds to a composite sample of contamination from the Waste Disposal Building, where the beta emitting radionuclides corresponding to the gross beta activity include: Mn-54, Co-60, Sr-90, Nb-94, Tc-99, Ag-108m, Cs-134, and Cs-137; and the gross alpha radionuclides include: Pu-238, Pu-239/240, Cm-233/234, and Am-241. This evaluation calculated doses for a variety of pipe diameters (12-, 24-, 36-, and 42-inch), conservatively assuming the same duration of occupancy used in the building occupancy scenario (2340 hours per year) and applied a dose due to inhalation and ingestion that is twice those calculated in the building occupancy scenario. The results of the evaluation showed that the doses calculated using these conservative assumptions were only slightly higher than those associated with the building occupancy scenario and were thus acceptable.

As the evaluation is valid for situations in which the gross beta-to-alpha ratio for an embedded pipe is 15:1 or greater (at the time of FSS), if this condition is not met at the time of FSS, the piping will be removed, grouted, or capped to prevent access.

When present in a survey unit, embedded pipe and penetrations will be evaluated using the data quality objective process during survey planning and either removed or incorporated into the survey sample design, using the building surface DCGL as the applicable release criteria (under the conditions stated above). The decision to remove these pipes will be done as part of an ALARA evaluation for the subject survey unit.

5.5 Final Status Survey Design Elements—Surface Soils, ~~and~~ Structures and Basements

Sampling and surveys required to support the Basement Fill Model calculation for future groundwater dose will be taken of the basement concrete structures to determine the volumetric concentrations for

subsurface structures within the water table. The number of samples to be taken for each basement and major subsurface feature to remain is given in Section 5.7.1.6. An assessment will be performed to determine the inventory of radioactivity in any footings that exhibit measurable radioactivity (i.e. > 2 sigma error of the analysis Minimum Detectable Activity) in concrete samples or in the surrounding soil. These inventories for footings will be included in the Basement Fill Model calculation as described in Section 6.8. The final status survey design elements and requirements for all other media and materials is further discussed below.

The general approach prescribed by MARSSIM for final status surveys requires that at least some minimum number of measurements or samples be taken within a survey unit, so that the non-parametric statistical tests used for data assessment can be applied with adequate confidence. Decisions regarding whether a given survey unit meets the applicable release criterion are made based on the results of these tests. Scanning measurements are used to check the design basis for the survey by evaluating if any small areas of elevated activity exist that would require reclassification, tighter grid spacing for the fixed measurements, or both. However, MARSSIM also recognizes that alternatives to this general approach for final status surveys exist. Specifically, MARSSIM states that if the equipment and methodology used for scanning are capable of providing data of the same quality as fixed measurements (e.g., detection limit, location of measurements, ability to record and document results), then scanning may be used in place of fixed measurements, provided that results are documented for at least the number of locations that would have been necessary had fixed measurements been used.

Final status surveys for the HNP surface soils and structures will be designed, following MARSSIM guidance, using combinations of fixed measurements, traditional scanning surveys, and other advanced survey methods, as appropriate, to evaluate survey units relative to their applicable release criteria. As MARSSIM does not directly address final status survey for subsurface soils, the principles of MARSSIM will guide the design of these surveys. Subsurface survey considerations can be found in Section 5.7.3.2.2.

Under MARSSIM, the level of survey effort required for a given survey unit is determined by the potential for contamination as indicated by its classification. Class 3 survey units receive judgmental scanning and randomly located measurements or samples. Class 2 survey units receive scanning over a portion of the survey unit based on the potential for contamination, combined with fixed measurements or sampling performed on a systematic grid. Class 1 survey units receive scanning over 100% of the survey unit combined with fixed measurements or sampling performed on a systematic grid. Depending on the sensitivity of the scanning method, the grid spacing may need to be adjusted to ensure that small areas of elevated activity are detected.

For combinations of fixed measurements and traditional scanning, MARSSIM methodology is to select a requisite number of measurement locations to satisfy the decision error rates for the non-parametric statistical test to be used for data evaluation and to account for sample losses or data anomalies. The purpose of scans is to confirm that the area was properly classified and that any small areas of elevated activity are within acceptable levels (i.e., are less than the applicable $DCGL_{EMC}$). Depending on the sensitivity of the scanning method used, the number of fixed measurement locations may need to be increased so the spacing between measurements is reduced. Details on selecting the number and location of fixed measurements are the subject of Section 5.5.1 and subsequent subsections of this plan. The coverage requirements that will be applied for scans performed in support of final status surveys for the HNP site are:

- For Class 1 survey units, 100% of the surface will be scanned;

- For Class 2 survey units, between 10% and 100% of the surface will be scanned in a combination of systematic and judgmental measurements for outdoor units and for floor and lower walls of structures; and 10% to 50% of the surface will be covered for upper walls and ceilings;

- Scanning will be done on a judgmental basis for Class 3 survey units.

Though the emphasis of the document is on conducting final status surveys through a combination of fixed measurements and scans, MARSSIM also allows for use of advanced survey technologies as long as these techniques meet the applicable requirements for data quality and quantity. "Advanced technologies" in this context refers to survey techniques where the instrument is capable of recording data as an area is surveyed and the measurement sensitivity is an acceptable fraction of the applicable $DCGL_W$ (see Section 5.7.1.3). Such methods are desirable for final status surveys since they allow survey units to be assessed with a single measurement rather than separate fixed measurements and scans.

Advanced survey techniques may be used alone or in combination with fixed measurements and scans to assess a survey unit. For Class 1 and Class 2 units, two conditions must be met for advanced technologies to be employed as the only survey technique: an acceptable fraction of the survey unit surface area must be scanned; and the minimum detectable concentration (MDC) for the measurements must be an acceptable fraction of the $DCGL_W$. For Class 1 units, 100% of the area must be covered. For Class 2 units, the coverage requirements for advanced technologies to be used alone are from 50% to 100% of the area for outdoor survey units or for floors and lower walls; and from 10% to 50% of the area for upper walls and ceilings. In cases where these coverage requirements cannot be achieved by an advanced survey technology or where the MDC is too large relative to the applicable $DCGL_W$ (see Section 5.5.1.5), the survey will be augmented with fixed measurements and traditional scans as necessary in accordance with Section 5.5.1 and subsequent subsections of this plan. Advanced technologies may be used for judgmental assessments in Class 3 areas as long as the following MDC requirements are met.

For fixed measurements, MARSSIM states that MDCs should be as far below the $DCGL_W$ as possible, with values less than 10% of the $DCGL_W$ being preferred, and up to 50% of the $DCGL_W$ being acceptable. These same criteria will be used when deciding if advanced survey techniques can be used in place of fixed measurements and traditional scans for a given survey unit. MDCs for advanced techniques will be computed using background count rates obtained using appropriate reference materials.

With respect to the survey methods and techniques discussed above, the survey design criteria that will be employed for final status surveys for the HNP site are summarized below. Note that "fixed measurements" is used interchangeably to refer to measurements or samples taken at specific locations.

- For Class 1 or Class 2 survey units, advanced survey technologies may be used exclusively only in survey units for which the above coverage requirements can be achieved and MDCs are no greater than 50% of the applicable $DCGL_W$.

- For Class 1 or Class 2 survey units for which advanced technologies would have an acceptable MDC, but the above coverage requirements cannot be achieved, advanced technologies may be used over 100% of the accessible area with a combination of fixed measurements and traditional scans used over the remainder of the area as specified in Section 5.5.1 and subsequent subsections of this plan.

For any survey units for which advanced survey techniques are impractical, fixed measurements and traditional scans will be used exclusively in accordance with this plan.

5.5.1 Selecting the Number of Fixed Measurements and Locations

The MARSSIM methodology for evaluating whether a survey unit meets its applicable release criterion using fixed measurements plus scans is based on using non-parametric statistical tests for data assessment. Specifically, the methods of MARSSIM are based on two non-parametric tests: the Wilcoxon Rank Sum (WRS) test and the Sign test, as discussed in Section 5.8 Selection of the required minimum number of data points depends on which statistical test is going to be used to evaluate the data, and thus depends on what type of measurements are to be made (gross measurement, net measurement or radionuclide specific) and if the radionuclide(s) of interest appear(s) in background.

5.5.1.1 Establishing Acceptable Decision Error Rates

One input to the process of selecting the required number of data points for a given survey, which does not depend on the statistical test applied, is the selection of the acceptable decision error rates. Decision errors refer to making false decisions by either rejecting a null hypothesis when it is true (a Type I error) or accepting a null hypothesis when it is false (a Type II error). With respect to final status surveys, the null hypothesis is that the survey unit of interest contains residual contamination in excess of the applicable release criterion. Thus, a Type I error refers to concluding that an area meets the release criteria when in fact it does not. The probability of making a Type I error is referred to as alpha (α). Likewise, a Type II error refers to concluding a unit does not meet the release criteria when it actually does. The probability of making a Type II error is denoted beta (β). Selecting values of α or β that are too low will result in an excessive number of fixed measurements being required. Likewise, selecting a β value that is too large can result in excessive costs in that survey units that meet the release criterion could be subjected to superfluous remediation efforts. Under the current regulatory models, an α value that is too large equates to greater risk to the public in that there is a greater chance of releasing a survey unit that does not meet the release criterion.

NRC draft regulatory guide DG-4006 (as incorporated in Section 7.2 of Appendix E to NUREG-1727) recommends that the α decision error rate be set to 0.05 (5%) and that "any value of β is acceptable to the NRC." Thus, decision error rates for final status surveys designed for the HNP site will be set as follows:

- the α value will always be set to 0.05 unless prior NRC approval is granted for using a less restrictive value;
- the β value is nominally set to 0.05, but may be changed if it is found that more fixed measurements than necessary are being made to demonstrate compliance with the release criterion.

5.5.1.2 Determining the Relative Shift

Another input to the process of selecting the required number of measurements that is somewhat independent of the statistical test to be employed is the determination of what is called the relative shift. The relative shift is a parameter that quantifies the concentrations to be measured in a survey unit relative to the variability in these measurements. The relative shift is a function of the $DCGL_W$, a parameter called the "Lower Bound of the Gray Region" (LBGR), and either the expected standard deviation of the

measurements to be made in the survey unit (σ_s) or the standard deviation established for the corresponding reference area (σ_r). The choice of σ_s or σ_r depends on whether the survey data are to be evaluated against a reference area(s). Reference areas are used if the WRS test is applied or, where gross measurements are to be background subtracted, the Sign test may be used. If a reference area is required, the larger of the values of σ_s or σ_r is used. The σ_s values will be selected by:

- using existing characterization or remediation support survey data or
- making preliminary measurements.

Values of σ_r will be computed using data collected from measurements in reference areas or from reference materials, as appropriate.

Given that σ_s and σ_r values should reflect a combination of the spatial variability in the concentration and the precision in the method of measurement, these values will be selected based on existing survey data only when the existing measurements were made using techniques equivalent to those to be used during the final status survey.

The LBGR represents the concentration to which the survey unit must be cleaned (decontaminated) in order to have an acceptable probability of passing the statistical test. The difference between the $DCGL_W$ and the LBGR, known as the shift, can be thought of as a measure of the resolution of the measurements that will be made in a survey unit. The shift is denoted as Δ.

The relative shift (Δ/σ) is computed as the quotient of the shift and the appropriate standard deviation values. If no reference area data are needed to evaluate the survey results, the expected standard deviation of the measurements (σ_s) is used. If a reference area is required, the larger of the values of σ_s or σ_r is used.

To compute the relative shift, the appropriate sigma value and an initial LBGR are selected. The initial value for LBGR will be based upon site specific information, if available; otherwise, per MARSSIM and DG-4006 (as incorporated in Section 7.1 of Appendix E to NUREG-1727) the initial value for the LBGR will be set to one-half of the $DCGL_W$. If the resulting relative shift is not in the range of 1.0 and 3.0, the LBGR is adjusted until it is. If the relative shift is too low, the LBGR is decreased; and if the relative shift is too high, the LBGR is increased.

5.5.1.3 Selecting the Required Number of Measurements for the WRS Test

The minimum number of fixed measurements required when the WRS is computed by the following equation:

$$N = \frac{1}{2} \times \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{3\left(P_r - 0.5\right)^2} \qquad \text{(Equation 5-12)}$$

where: N = the minimum number of measurements required for each survey area or reference area;
$Z_{1-\alpha}$ = the percentile represented by the α decision error;
$Z_{1-\beta}$ = the percentile represented by the β decision error; and
P_r = the probability that a random measurement from the survey unit exceeds a random measurement from the reference area by less than the $DCGL_W$ when the survey unit median is equal to the LBGR concentration above background.

Values of P_r, $Z_{1-\alpha}$ and $Z_{1-\beta}$ will be taken from Tables 5.1 and 5.2 of MARSSIM. P_r is a function of the relative shift, and $Z_{1-\alpha}$ and $Z_{1-\beta}$ depend on the selected values for α and β.

The value of N computed for the WRS test applies for both the survey unit and the reference area (i.e., at least N measurements should be performed in both areas). To ensure against lost or unusable data, the value of N will be increased by at least a factor of 1.2 when assigning the number of measurements to be made.

5.5.1.4 Selecting the Required Number of Measurements for the Sign Test

The minimum number of fixed measurements required when the Sign test is computed by the following equation:

$$N = \frac{(Z_{1-\alpha} + Z_{1-\beta})^2}{4(Sign\ p - 0.5)^2} \qquad \text{(Equation 5-13)}$$

where: N = the minimum number of measurements required;
$Z_{1-\alpha}$ = the percentile represented by the α decision error;
$Z_{1-\beta}$ = the percentile represented by the β decision error; and
Sign p = the probability that a random measurement from the survey unit will be less than the $DCGL_W$ when the survey unit median concentration is equal to the LBGR.

Values for Sign p will be taken from Table 5-4 of MARSSIM.

To ensure against lost or unusable data, the value of N will be increased by at least a factor of 1.2 when assigning the number of measurements to be made.

5.5.1.5 Assessing the Need for Additional Measurements in Class 1 Survey Units

Given the potential for small areas of elevated activity in Class 1 survey units, evaluations must be performed to assess the potential for missing such areas while scanning in locations not covered by fixed measurements. This evaluation, referred to as the Elevated Measurement Comparison (EMC), is performed by comparing the MDC of the scanning technique to the $DCGL_{EMC}$ for the survey unit of interest. If the scanning MDC is larger than the $DCGL_{EMC}$, additional measurements may be required beyond the minimum number computed via Equation 5-9 or 5-10. The effect of these additional measurement points is to tighten the grid spacing for the fixed measurements, thus reducing the probability of missing a small area of elevated activity to an acceptable level.

The adequacy of the scanning technique will be evaluated by calculating a scanning MDC, expressed as a fraction of the $DCGL_{EMC}$ as shown below.

As described in Section 5.4.7.4, the relationship between the $DCGL_{EMC}$ and the $DCGL_W$ using the area factor for nuclide i is:

$$DCGL^i_{EMC} = AF^i DCGL^i_W \qquad \text{(Equation 5-14)}$$

Where: AF^i is the area factor for radionuclide i.

For soil, the relationship between a scanning minimum detectable count rate (MDCR) and the minimum detectable soil concentration is:

$$MDC^i(pCi/g) = \frac{MDCR(cpm)}{E^i(cpm/pCi/g)} \qquad \text{(Equation 5-15)}$$

Where: E^i is the conversion factor (in cpm/pCi/g) for the radionuclide i (instrument efficiency for scanning)

The soil scanning MDC expressed as a fraction of the $DCGL_{EMC}$ is calculated by the following equation:

$$MDC(fDCGL_{EMC}) = MDCR\sum\frac{f^i}{E^i DCGL^i_{EMC}} = MDCR\sum\frac{f^i}{E^i AF^i DCGL^i_W} \qquad \text{(Equation 5-16)}$$

Where: f^i is the decimal fraction of the radionuclide mix comprised by radionuclide i and is based upon characterization data, as a part of the Final Status Survey.

An example calculation to determine the soil scanning MDC expressed as a fraction of the $DCGL_{EMC}$ when multiple radionuclides are present is shown as follows:

Assumptions:
Two radionuclides are present; Cs-137 and Co-60
Cs-137 fraction in mix (f) = 0.75
Co-60 fraction in mix (f) = 0.25
Cs-137 efficiency (E) = 228 cpm/pCi/g
Co-60 efficiency (E) = 882 cpm/pCi/g
Elevated area = 100 m^2
Cs-137 area factor (AF) from Table 5-5 = 2.93
Co-60 area factor (AF) from Table 5-5 = 1.41
Cs-137 $DCGL_W$ from Table 6-1 = 7.91 pCi/g
Co-60 $DCGL_W$ from Table 6-1= 3.81 pCi/g
MDCR = 2,000 cpm

$$MDC(fDCGL_{EMC}) = 2{,}000\left[\frac{0.75}{(228)(2.93)(7.91)} + \frac{0.25}{(882)(1.41)(3.81)}\right] = 0.4$$

For scanning building surfaces, the following equation from MARSSIM provides the method to calculate the MDC for beta-gamma measurements. It has been repeated here below for clarity:

$$MDC(dpm/100cm^2) = \frac{1.38\sqrt{B}}{\sqrt{p}\varepsilon_i\varepsilon_s\left(\frac{A}{100}\right)t} \qquad \text{(Equation 5-17)}$$

where t is the time the detector spends over a source of radionuclide i which can be related to the travel velocity of the probe, V(cm/min), and the minimum dimension of the detector, L (cm), as:

$$t(\min) = \frac{L(cm)}{V(cm/\min)} \qquad \text{(Equation 5-18)}$$

Equation 5-14 can be rewritten as follows:

$$MDC^i(dpm/100cm^2) = \frac{1.38\sqrt{\frac{B}{t^2}}}{\sqrt{p}\varepsilon_i^i\varepsilon_s^i\left(\frac{A}{100}\right)} = \frac{1.38\sqrt{\frac{R_b}{t}}}{\sqrt{p}\varepsilon_i^i\varepsilon_s^i\left(\frac{A}{100}\right)} = \frac{1.38\sqrt{R_b}}{\sqrt{p}\sqrt{t}\varepsilon_i^i\varepsilon_s^i\left(\frac{A}{100}\right)} \qquad \text{(Equation 5-19)}$$

Substituting equation 5-15 into 5-16 gives:

$$MDC^{i}(dpm/100cm^{2}) = \frac{1.38\sqrt{R_b}}{\sqrt{p}\,\varepsilon_i^i \varepsilon_s^i \left(\frac{A}{100}\right)\sqrt{\frac{L}{V}}} \qquad \text{(Equation 5-20)}$$

In accordance with MARSSIM, the MDCR for an analog detector with an audible signal (for d'=1.38) is:

$$MDCR(cpm) = \frac{1.38\sqrt{B}}{t} = \frac{1.38\sqrt{R_b}}{\sqrt{t}} = \frac{1.38\sqrt{R_b}}{\sqrt{\frac{L}{V}}} \qquad \text{(Equation 5-21)}$$

Using this, equation 5-20 is re-written as:

$$MDC^{i}(dpm/100cm^{2}) = \frac{MDCR}{\varepsilon_i^i \varepsilon_s^i \sqrt{p}\left(\frac{A}{100}\right)} \qquad \text{(Equation 5-22)}$$

To allow for multiple radionuclides, the scan MDC expressed as a fraction of the $DCGL_{EMC}$ is:

$$MDC(fDCGL_{EMC}) = \frac{MDCR}{\sqrt{p}\left(\frac{A}{100}\right)} \sum \frac{f^{i}}{\varepsilon_i^i \varepsilon_s^i DCGL_{EMC}^{i}} \qquad \text{(Equation 5-23)}$$

By substituting $DCGL_{EMC}^{i} = AF^{i} DCGL_{W}^{i}$ into Equation 5-23 yields the building surface scanning MDC equation expressed as a fraction of the $DCGL_{EMC}$:

$$MDC(fDCGL_{EMC}) = \frac{MDCR}{\sqrt{p}\left(\frac{A}{100}\right)} \sum \frac{f^{i}}{\varepsilon_i^i \varepsilon_s^i AF^{i} DCGL_{W}^{i}} \qquad \text{(Equation 5-24)}$$

An example calculation to determine the building surface scanning MDC expressed as a fraction of the $DCGL_{EMC}$ when multiple radionuclides are present is shown below:

Assumptions:

Two radionuclides are present; Cs-137 and Co-60
Cs-137 fraction in mix (f) = 0.75
Co-60 fraction in mix (f) = 0.25
Probe width (L) = 10.2 cm (4 inches)
Scan rate (V) = 305 cm/min (2 inches/sec)
Background count rate (R_b) = 200 cpm

$\varepsilon_i = 0.3$ for Co-60
$\varepsilon_i = 0.38$ for Cs-137
$\varepsilon_s = 0.25$ for Co-60
$\varepsilon_s = 0.5$ for Cs-137
Surveyor Efficiency, p=0.5
Probe area (A) = 100 cm^2
MDCR = 107 cpm
Elevated area = 10 m^2
Cs-137 area factor (AF) from Table 5-6 = 2.6
Co-60 area factor (AF) from Table 5-6 = 2.5
Cs-137 $DCGL_W$ from Table 6-3 = 4.30E+04 dpm/100 cm^2
Co-60 $DCGL_W$ from Table 6-3= 1.11E+04 dpm/100 cm^2

$$MDC(fDCGL_{EMC}) = \frac{107}{\sqrt{0.5}\left(\frac{100}{100}\right)}\left[\frac{0.75}{(0.38)(0.5)(2.6)(4.30E4)} + \frac{0.25}{(0.3)(0.25)(2.5)(1.11E4)}\right] = 0.02$$

As shown in these two examples, the fraction of $DCGL_{EMC}$ is less than one. Therefore no additional measurements are required.

If the value of MDC ($fDCGL_{EMC}$) is greater than one, additional measurements may need to be taken in the survey unit as determined by taking the following steps.

Determine the size of the elevated area from Table 5-5 or Table 5-6 corresponding to the highest $fDCGL_{EMC}$ which is still less than one. That area is denoted as A_{EMC}.

The number of measurements (N_{EMC}) required to detect an area of elevated concentration equal to A_{EMC} is then computed as

$$N_{EMC} = \frac{A}{A_{EMC}} \quad \text{(Equation 5-25)}$$

where A is the total area of the survey unit. N_{EMC} (computed via Equation 5-25) is then compared to N, the number of fixed measurement points computed via Equation 5-12 or 5-13. The larger of N_{EMC} or N is then used as the requisite number of fixed measurement locations and to compute the grid spacing.

5.5.1.6 Determining Measurement Locations

For Class 1 and Class 2 survey units, fixed measurements will be performed over a systematic measurement pattern consisting of a grid having either a triangular or a square pitch. The pitch (grid spacing) will be determined based on the number of measurements required and whether the desired grid is triangular or square.

Systematic grids will not be used for surveys involving fixed measurements for Class 3 units. Instead, fixed measurement locations will be selected at random throughout the survey unit area by generating pairs of random numbers between zero and one. One pair of random numbers will be generated for each fixed measurement to be made. The random number pairs, representing (x, y) coordinates, will be multiplied by the maximum length and width dimensions of the survey unit to yield the location for each fixed measurement. For odd-shaped survey units, a rectangular area encompassing the survey unit will be used to establish the maximum length and width. A new pair of random numbers will be generated if any of them give locations that are not actually within the survey unit boundaries. New pairs of numbers will also be generated in cases where a measurement cannot be made at a specific location because of an obstruction, inaccessibility, etc.

The spacing to be used in setting up the systematic grid used to establish fixed measurement locations for Class 1 and Class 2 areas will be computed as

$$L = \sqrt{\frac{A}{0.866N}} \quad \text{for a triangular grid, or} \qquad \text{(Equation 5-26)}$$

$$L = \sqrt{\frac{A}{N}} \quad \text{for a square grid} \qquad \text{(Equation 5-27)}$$

where L = grid spacing (dimension is square root of the area),
A = the total area of the survey unit, and
N = the desired number of measurements.

In the case of Class 1 units, the value used for N in Equations 5-26 and 5-27 should be the larger of that from Equations 5-12 or 5-13 (if the scan MDC is sufficient to see small areas of elevated activity) or Equation 5-25. In all cases, the value of N should include additional measurements required to ensure against losses or unusable data.

Once the grid spacing is established, a random starting point will be established for the survey pattern using the same method as described above for selecting random locations for Class 3 units. Starting from this randomly-selected location, a row of points will then be established parallel to one of the survey unit axes at intervals of L. Additional rows will then be added parallel to the first row. For a triangular grid, additional rows will be added at a spacing of 0.866L from the first row, with points on alternate rows spaced mid-way between the points from the previous row. For a square grid, points and rows will be spaced at intervals of L. Section 5.5.2.5 of MARSSIM describes the process to be used for selecting fixed measurement locations and provides examples of how to establish both a systematic grid and random measurement locations.

Software tools that accomplish the necessary grid spacing, including random starting points and triangular or square pitch, may be employed during Final Status Survey. When available, this software will be used

with suitable mapping programs to determine coordinates for a global positioning system (GPS). The use of these tools will provide a reliable process for determining, locating and mapping measurement locations in open land areas separated by large distances and will be helpful during independent verification.

5.5.2 Judgmental Assessments

For those Class 2 and Class 3 survey units for which 100% of the area is not surveyed, it is important to consider performing judgmental assessments to augment any regimented measurements made in accordance with the above guidance. Such assessments may consist of biased sampling or measurements performed in locations selected on the basis of site knowledge and professional judgment. Judgmental assessments serve to provide added assurance that residual contamination at the site has been adequately located and characterized.

In addition to any judgmental measurements deemed necessary to provide comprehensive survey coverage for a given survey unit, the survey process should include an isotopic mix evaluation in cases where measurable activity still exists. Doing so will allow an assessment of the adequacy of the $DCGL_W$ selected for the survey unit in question to be made during the subsequent data assessment phase. For gross count measurements (i.e., not radionuclide specific), radionuclide mix information will also allow for an evaluation of the suitability of the efficiencies applied in converting raw count data to activity.

The basis for judgmental assessments will be documented in the Final Status Survey Report and will receive a technical review in accordance with plant procedures.

5.5.3 Data Investigations

5.5.3.1 Investigation Levels

An important aspect of the final status survey is the selection and implementation of investigation levels. Investigation levels are levels of radioactivity used to indicate when additional investigations may be necessary. Investigation levels also serve as a quality control check to determine when a measurement process begins to deviate from expected norms. For example, a measurement that exceeds an investigation level may indicate a failing instrument or an improper measurement. However, in general, investigation levels are used to confirm that survey units have been properly classified.

When an investigation level is exceeded, the first step is to confirm that the initial measurement/sample actually exceeds the particular investigation level. Depending on the results of the investigation actions, the survey unit may subsequently require reclassification, remediation, and/or resurvey. Investigation levels are established for each class of survey unit. The investigation levels (criteria), to be employed for the HNP final status survey effort, are given in Table 5-8.

Table 5-8
Investigation Levels

Survey Unit Classification	For fixed measurements or samples, perform investigation if:	For scan measurements, perform investigation if:
Class 1	> $DCGL_{EMC}$ or > $DCGL_W$ and a statistical outlier.	> $DCGL_{EMC}$
Class 2	> $DCGL_W$	> $DCGL_W$ or > MDC_{scan} if MDC_{scan} is greater than the $DCGL_W$
Class 3	> 0.5 × $DCGL_W$	Detectable over background.

For Class 1 survey units, measurements above the $DCGL_W$ are not necessarily unexpected. However, such a result may still indicate a need for further investigation if it is significantly different than the other measurements made within the same survey unit. Thus, some additional evaluation criterion is needed to assess if results from fixed measurements or samples in a Class 1 survey unit that exceed the $DCGL_W$ warrant further attention. Measurements in Class 1 survey units that exceed the $DCGL_W$ and differ from the mean of the remaining measurements by more than three standard deviations will therefore be investigated. Measurements in Class 1 units that exceed the $DCGL_W$, but do not differ from the mean by as much may still be investigated on the basis of professional judgment, as may any measurements that differ significantly from the rest of the measurements made within a given survey unit.

In Class 2 or Class 3 areas, neither measurements above the $DCGL_W$ nor areas of elevated activity are expected. Thus, any fixed measurements or sampling results that exceed the $DCGL_W$ in these areas will be investigated. In the case of Class 3 areas, where any residual radioactivity would be unexpected, fixed measurement or sample results that are greater than 0.5 x $DCGL_W$ will be investigated. Because the survey design for Class 2 and Class 3 survey units is not driven by the elevated measurement comparison, any indication of residual radioactivity in excess of the $DCGL_W$ during the scan of a Class 2 unit will warrant further investigation. For Class 3 units, any scan measurement that shows a positive indication over background will be investigated.

In cases where an advanced survey method is used instead of fixed measurements or samples, the investigation levels given in Table 5-8 for fixed measurements or samples will be applied with the exception of the statistical outlier test for measurements in Class 1 survey units. In cases where advanced survey methods are used as a means of traditional scanning, the investigation levels for scan measurements in Table 5-8 will be used.

5.5.3.2 Investigations

Locations where initial measurements give results that exceed an applicable investigation level will be identified for confirmatory measurements. If it is confirmed that residual activity exists in excess of the investigation level, additional measurements will be made to determine the extent of the area of elevated activity and to provide reasonable assurance that other areas of elevated activity do not exist. Potential sources of the elevated activity will be postulated and evaluated against the original classification of the survey unit and its associated characterization data. The possibility of the source of the elevated activity having affected other adjacent or nearby survey units will also be evaluated. Documentation will be compiled containing the results from the investigation surveys and showing any areas where residual activity was confirmed to be in excess of the investigation level. If residual activity in excess of the

applicable investigation level is confirmed, the documentation will also address the potential source(s) of the activity and the impact this has on the original classification assigned to the survey unit. A decision will then be made regarding re-classification of the unit in whole or in part.

5.5.3.3 Remediation

"Remediation" in the context of the LTP is intended to mean activities performed to meet the criteria of 10CFR20, Subpart E. Activities to remove materials may be performed for other reasons, and thus are not considered to be "remediation." If during the time of Final Status Survey, any areas of residual activity found to be in excess of the $DCGL_{EMC}$, they will be remediated to reduce the activity to acceptable levels. Areas of residual activity may also need to be remediated to meet the ALARA criterion. Remediation actions are discussed in Chapter 4 and documented as described in Section 5.9.

5.5.3.4 Re-classification

The decision to reclassify an area, or part of an area, is made following a review of the basis for the original classification, considering the evaluation process outlined in Section 5.5.3.2 (consistent with MARSSIM). This process includes sufficient additional measurements to confirm the residual contamination, determine the nature and extent of the contamination present, provide assurance that other areas of elevated activity do not exist within the survey unit, and evaluate the impact (if any) of the affected area on nearby survey units. The results of these measurements will be evaluated, and the area, or part of the area, will be reclassified and resurveyed per Section 5.5.3.5 in a manner that is consistent with the process described in MARSSIM. Additionally, if required remediation actions are taken in the area, it will be resurveyed per Section 5.5.3.5 in a manner that is consistent with the process described in MARSSIM. Re-classification of areas from a less to a more restrictive classification may be done without prior NRC approval; however, re-classification to a less restrictive classification would require prior NRC approval.

5.5.3.5 Re-survey

If a survey unit is re-classified (in whole or in part), or if remediation is performed within a unit, then the areas affected are subject to re-survey. Any re-surveys will be designed and performed as specified in this plan based on the appropriate classification of the survey unit. That is, if a survey unit is re-classified or a new survey unit is created, the survey design will be based on the new classification.

For example, a Class 3 area that is subdivided due to the unexpected presence of radioactivity will be divided into at least two areas. One of these may remain as a Class 3 area while the other may be a Class 2 area. In order to maintain the survey design Type I and Type II decision error rates in the Class 3 area, additional measurements may be required to be performed at randomly selected locations until the required total number of measurements is met (see Section 5.5.1). The new sub-divided Class 2 survey area will then be surveyed using a new survey design. The Type I and II decision error rates used are documented in the final status survey documentation.

A Class 2 area that is subdivided due to the levels of radioactivity identified will be divided into at least two areas as well. In this case if the original survey design criteria has been satisfied, no additional action is required, otherwise the remaining Class 2 survey unit will be redesigned. The new sub-divided survey unit will be surveyed against a new survey design.

If remediation is required in only a small area of a Class 1 survey unit, any replacement measurements or samples required will be made within the remediated area at randomly selected locations following

verification that the remediation activities did not affect the remainder of the unit. Re-survey will be required in any area of a survey unit affected by subsequent remediation activities.

5.6 Survey Protocol for Non-Structural Systems and Components

The guidance provided in MARSSIM and DG-4006 for conducting final status surveys does not include guidance for conducting final status surveys for non-structural system or components. Per DG-4006, "non-structural systems and components" refers to anything not attached to or not an integral part of a building or structure. Given that the methods of the MARSSIM do not apply to non-structural systems and components, an alternative set of release criteria must be chosen to facilitate site remediation for license termination purposes.

The current site unconditional release limit of no detectable radioactive (licensed) material will be used to survey non-structural systems and components (excluding the cases discussed below). Non-structural systems and components meeting the criteria can be released, after survey. Those not meeting the release criteria will be disposed of as radioactive waste.

Buried pipe that is located within the saturated subsurface areas of the site (to remain on site) will be surveyed to the limits set forth in Table 5-7. Full-length surveys will be performed for this piping, typically using conventional methods and instrumentation. If advanced technology instrumentation, such as in-situ gamma-spectroscopy, is selected for use, a technical support document will be developed which describes the technology to be used and how the technology meets the objectives of the survey. This document will be available for NRC inspection in support of final status survey activities. Detection limits for surface activity assessments for this buried piping should be at least equivalent to the release limits given in Table 5-7 at the 95% confidence level. Detection limits will be computed using the methods described in Section 5.7.2.5 of this plan. If necessary, scaling factors may be applied to establish gross activity levels via radionuclide-specific measurements or other assessments, as appropriate.

As discussed in Section 5.4.7.5, embedded piping to remain on site will be surveyed to the building surface DCGLs, provided that the gross activity beta-to-alpha ratio associated with that piping (or penetration) at the time of FSS is greater than or equal to 15:1. If the gross activity beta-to-alpha ratio at the time of FSS is found to be less than 15:1, the associated piping or penetration will be capped, grouted, or removed.

Evaluations as to whether material should be considered as a structure or a component will be via the guidance of Section B of DG-4006 and comparisons with the dose modeling scenarios used to develop the DCGLs that govern release of grounds and structures. Examples of parts of buildings or structures that are considered in the development of DCGLs include floors, walls, ceilings, doors, windows, sinks, hoods, lighting fixtures, built-in laboratory benches, and built-in furniture. Examples of non-structural systems and components include pumps, motors, heat exchangers, and piping between components.

5.7 Survey Implementation and Data Collection

The requirements and objectives outlined in this plan and the project QA plan will be incorporated into Standard Operating Procedures (SOPs). Procedures will govern the survey design process, survey performance and data assessment (decision making). The final status survey design will be carried out in accordance with the SOPs and the QA plan, resulting in the generation of raw data. The product of the survey design process is a survey package, which addresses various elements of the survey, including, but not limited to:

- maps of the survey area showing the survey unit(s) and measurement/sample locations, as appropriate;
- applicable DCGLs
- instrumentation to be used;
- instrument calibration;
- types and quantities of measurements or samples to be made or collected;
- investigation criteria;
- QA/QC requirements (e.g., replicate measurements or samples);
- personnel training;
- applicable health and safety procedures;
- approved survey procedures; and
- applicable operating procedures.

An important element of the survey design process is establishing the DCGLs for the measurements to be made. The DCGLs will be determined as described in Section 5.4.7 based on characterization data for the survey unit(s) being considered. Isotopic mix, material backgrounds, and the variability of these will all be considered. The detection limit requirements dictated by the DCGLs affect the selection of both the instrumentation to be used for a given survey and the survey method(s) to be employed (advanced survey methods, fixed measurements, sampling; or combinations thereof).

5.7.1 Survey Methods

The survey methods to be employed in the final status surveys will consist of combinations of advanced technologies, scanning, fixed measurements, sampling, and other methods as needed to meet the survey objectives. Additional methods may be used if such become available between the time this plan is adopted and the completion of final survey activities. However, any new technologies must still meet the applicable requirements of this plan. Note that in some cases, the same instrument may be used for more than one type of survey. For instance, a sodium-iodide (NaI) detector may be used in either a scanning mode or for fixed spectroscopic measurements.

5.7.1.1 Scanning

Scanning is the process by which the operator uses portable radiation detection instruments to detect the presence of radionuclides on a specific surface (i.e., ground, wall, floor, equipment). The term scanning survey is used to describe the process of moving portable radiation detectors across a surface with the intent of locating residual radioactivity. Investigation levels for scanning surveys are determined during survey planning to identify areas of elevated activity. Scanning surveys are performed to locate radiation anomalies indicating residual gross activity that may require further investigation or action. These investigation levels may be based on the $DCGL_W$ or the $DCGL_{EMC}$.

No matter what survey approach is selected (combination of instrumentation and techniques), one of the most important elements of a survey is *a priori* scanning to confirm that the unit is properly classified and to identify any areas where residual activity levels are elevated relative to the $DCGL_W$. The purpose of scanning is to detect areas of residual activity that may not be detected by other measurement methods. Thus, scanning should always be performed prior to any fixed measurement or sample collection in a survey unit. If the scanning indicates that the unit or some area within the unit has been improperly classified, then the survey design process must be evaluated to either assess the effect of re-classification on the survey unit as a whole (if the whole unit requires re-classification) or a new design must be established for the new unit(s) (in the case of sub-division). A new survey design will require a re-evaluation of the survey strategy to decide if it can meet the requirements of the revised survey design. If not, the survey strategy must be revised based on the available instrumentation and methods.

Table 5-9 gives the area coverage requirements when scanning is used with fixed measurements.

Table 5-9
Traditional Scanning Coverage Requirements

Survey Unit Classification	Required Scanning Coverage Fraction
Class 1	100%
Class 2	Outdoor areas, floors, or lower walls of buildings: 10% to 100% Upper walls or ceilings: 10% to 50%
Class 3	Judgmental

5.7.1.2 Fixed Measurements

Fixed measurements are taken by placing the instrument at the appropriate distance above the surface, taking a discrete measurement for a pre-determined time interval, and recording the reading. Fixed measurements may be collected at random locations in a survey unit or may be collected at systematic locations and supplement scanning surveys for the identification of small areas of elevated activity. Fixed measurements may also be collected at locations identified by scanning surveys as part of an investigation to determine the source of the elevated instrument response. Professional judgment may also be used to identify locations for fixed measurements to further define the areal extent of contamination. Locations for fixed measurements specified by a given survey design will be established as discussed in Section 5.5.

5.7.1.3 Advanced Technologies

In the context of this Plan, advanced technologies refer to survey instruments or methods that create a spatially-correlated log of the measurements made as the detector is passed over an area. This logging of all of the measurements allows quantitative assessments of activity levels to be made, thus serving the same role as fixed measurements. Having all of the measurements logged allows statistical analyses to be made using a large number of samples, which provides for enhanced detection sensitivity relative to traditional scanning. The sensitivity achieved using advanced survey methods may, in some cases, be small enough relative to the $DCGL_W$ that the advanced method alone will allow a decision to be made as to whether a survey unit meets the release criterion without the need for additional fixed measurements. The fact that the instrument records every measurement made over the entire area it covers inherently addresses the issue of small areas of elevated activity. Average and maximum residual activity

concentrations can be quantified over any area desired, allowing one to assess compliance with the applicable criteria ($DCGL_W$ or $DCGL_{EMC}$) by inspection.

If advanced technology instrumentation is selected for use, a technical support document will be developed which describes the technology to be used and how the technology meets the objectives of the survey. This document will be available for NRC inspection in support of final status survey activities.

5.7.1.4 Other Advanced Survey Technologies

Other instruments and methods that may be used for final status surveys include, but are not limited to, in situ gamma spectrometry, in situ object counting systems, and systems capable of traversing ducting or piping. Like the advanced technologies discussed above, these other methods may in some cases provide sufficient area coverage so that augmenting the measurement with scanning is not necessary.

In situ gamma spectrometry is an established technique for assaying the average radionuclide concentration in large volumes of material (for example, soil and activated concrete). It has the advantage of being able to assess large areas with a single measurement. If desired, the detector's field of view can be reduced through collimation to allow assay of smaller areas.

In situ object counting refers to gamma spectrometry systems that include software capable of modeling photon transport in complex geometries for the purpose of estimating detector efficiencies. This eliminates the need for a calibration geometry representing the object to be counted. Such systems are useful for assaying complex components such as heat exchangers. "Pipe crawler" systems may be employed to survey a length of piping or ducting.

If advanced technology instrumentation is selected for use, a technical support document will be developed which describes the technology to be used and how the technology meets the objectives of the survey. This document will be available for NRC inspection in support of final status survey activities.

5.7.1.5 Samples

Sampling is the process of collecting a portion of a medium as a representation of the locally remaining medium. The collected portion of the medium is then analyzed to determine the radionuclide concentration. Examples of materials that may be sampled include soil, sediments, concrete, paint, and groundwater.

Section 5.10, "Quality Assurance and Quality Control Measures" addresses QA requirements for final status survey activities that apply to onsite and offsite laboratories employed to analyze samples as a part of the final status survey process. Performance of laboratories will be verified periodically by QA auditors. This verification will include reviews of personnel training, procedures and equipment operation.

Trained and qualified individuals will collect and control samples. All sampling activities will be performed under approved procedures. CYAPCO will utilize a chain-of-custody (COC) process to ensure sample integrity.

5.7.1.6 Contaminated Concrete Basements

The Basement Fill Model treats contaminated concrete as a volumetric source of radioactivity. It is therefore appropriate to utilize volumetric concrete sample results to determine the data to be used in the calculation. Table 5-10 shows the number of samples that have been taken to date and the number of

additional samples that will be taken to provide enough characterization to allow the confident calculation of the future groundwater dose. The samples taken will be analyzed so that the profile with the depth of the concrete can be confidently shown. Except for the in core sump, the sampling will include analysis of concrete from the inside and outside surfaces and for areas inside the wall with at least 15% of the wall/floor thickness characterized. For the In Core Sump, the total depth of the sample will be analyzed to determine the radioactivity profile for this area.

Table 5-10

Volumetric Concrete Sample Requirements

Basement Area to Remain (Below Elevation 17'6")	Concrete Samples Collected to Date (10/30/2004)	Additional Concrete Samples to be Collected	Minimum Total Number Of Samples To be Used for the Inventory Calculation
Containment Mat	8	6	14
Containment Walls	4	6	10
In Core Sump	1	8	9
Spent Fuel Pool	0	12	12
Cable Vault	7	6	13
"B" Switchgear Building	0	8	8
Discharge Tunnels/Structure	0	10	10
Intake Structure	2	6	8

The mechanism that has caused volumetric contamination in concrete is in many cases specific to certain radionuclide. Radionuclides such as H-3 and Sr-90 have been detected in concrete in contact with contaminated groundwater. Areas that have been subject to substantial neutron flux typically display H-3, Fe-55, Co-60, Eu-152 among others. To adequately assess the volumetritic contamination of concrete, a wafer from at least 20% of the locations listed in table 5-10 will be analyzed for all 20 radionuclides listed in Table 2-12. For radionuclides expected in certain areas of concrete, a sufficient number of wafers from all locations will be analyzed for the expected radionuclides to allow determination of a profile in the concrete at that location.

5.7.2 Survey Instrumentation

5.7.2.1 Survey Instrument Data Quality Objectives

The data quality objectives process includes the selection of instrumentation appropriate for the type of measurement to be performed (i.e., fixed measurement, scan or both), that are calibrated to respond to a radiation field under controlled circumstances; evaluated periodically for adequate performance to established quality standards; and sensitive enough to detect the radionuclide(s) of interest with a sufficient degree of confidence. The specific DQOs for instruments are established early in the planning phase for FSS activities, implemented by standard operating procedures and executed in the survey plan. Further discussion of the DQOs for instruments is provided below.

5.7.2.2 Instrument Selection

The selection and proper use of appropriate instruments for both fixed measurements and laboratory analyses is one of the most important factors in assuring that a survey accurately determines the radiological status of a survey unit and meets the survey objectives. The survey plan design must establish acceptable measurement techniques for scanning and direct measurements. The DQO process must include consideration as to the type of radiation, energy spectrum and spatial distribution of radioactivity as well as the characteristics of the medium to be surveyed (e.g., painted, scabbled, chemically decontaminated).

The particular capabilities of a radiation detector establish its potential for being used in conducting a specific type of survey based on the factors discussed above. Radiation survey parameters that will be needed for final survey purposes include surface activities and radionuclide concentrations in soil. To determine these parameters, both field measurements and laboratory analyses will be necessary. For certain radionuclides or radionuclide mixtures, both alpha and beta radiation may have to be measured. In addition to assessing average radiological conditions, the survey objectives must address identifying small areas of elevated activity.

Instruments must be stable and reliable under the environmental and physical conditions where they will be used, and their physical characteristics (size and weight) should be compatible with the intended application. This has been the case for typical radiation detection instrumentation used at HNP for operational surveys as well as scoping and characterization surveys.

The radiation detectors to be used for final survey activities at the Haddam Neck Plant can be divided into three general classes:

- gas-filled detectors,
- scintillation detectors, and
- solid-state detectors.

Gas-filled detectors include ionization chambers, proportional counters (both gas-flow and pressurized) and Geiger-Mueller (GM) detectors. Scintillation detectors include plastic scintillators, zinc-sulfide (ZnS) detectors and sodium-iodide (NaI) detectors. Solid-state detectors include both n-type and p-type intrinsic germanium detectors.

Finally, the DQO process must evaluate, depending on the type of radiation of interest, and on the application, the ability of instrumentation to measure levels that are less than the DCGL. In some cases instruments used for scanning may have detection limits that are greater than the $DCGL_W$. This is recognized by MARSSIM as an acceptable approach as long as the grid spacing (for Class 1 survey units) and investigation levels used are in accordance with Sections 5.5.1.5, 5.5.1.6 and 5.5.3.1, respectively, of this plan. The DQO process for instrument selection is performed in the planning phase for an FSS activity and is typically documented by a technical support document, which is referenced in the survey plan.

5.7.2.3 Calibration and Maintenance

All instrumentation used for measurements to demonstrate compliance with the radiological criterion for license termination at the Haddam Neck Plant will be calibrated and maintained under approved plant procedures and the project QA plan or vendor QA plan that satisfies the requirement of the project QA plan. Instruments will be calibrated for normal use under typical field conditions at the frequency specified by vendor instructions or by approved plant procedures (at least annually). Calibration standards will be traceable to the National Institute of Standards and Technology (NIST). If external

vendors are used for instrument calibration or maintenance, these services must be approved and conducted under the project QA plan. Calibration records will be maintained as required by plant procedures and the project QA plan.

Instruments used to measure gross beta surface activity will be calibrated using radionuclides such as Tc-99, Co-60, or Cs-137 so as to represent the beta energies for the beta-emitting radionuclides that will be encountered during final survey activities. Likewise, radionuclides such as Pu-239 or Th-230 may be used to calibrate instruments used to assess alpha surface activity so the alpha energies of the transuranic (TRU) radionuclides that may be encountered are adequately represented.

The DQO process must consider the field conditions the instrument will be used in to determine the affect and magnitude of variation from conditions established during calibration. These conditions might include source to detector geometry (including distance and solid angle), size and distribution of the source relative to the detector, and composition and condition of surface to be assessed. Most of these factors should have been determined during the instrument selection process. In some cases, instrument efficiencies may require modifications to account for surface conditions or coverings. Such modifications, if necessary, will be established using the information in Section 5 of NUREG-1507 and pertinent site characterization data. This will be performed during the planning process and documented by a technical support document and referenced in the survey plan. This technical support document will include the evaluation supporting instrument selection.

5.7.2.4 Response Checks

The DQO process determines the frequency of response checks, typically before issue and after an instrument has been used (typically at the end of the work day but in some cases this may be performed during an established break in activity, e.g., lunch). This additional check will expedite the identification of a potential problem before continued use in the field. Instrumentation will be response checked in accordance with plant procedures. If the instrument response does not fall within the established range, the instrument will be removed from use until the reason for the deviation can be resolved and acceptable response again demonstrated. If the instrument fails a post-survey source check, all data collected during that time period with the instrument will be carefully reviewed and possibly adjusted or discarded, depending on the cause of the failure. In the event that data are discarded, the affected area will be resurveyed.

5.7.2.5 MDC calculations

The DQO process evaluates the ability of the instrument to measure radioactivity at levels below the applicable DCGL. This evaluation will be performed and documented by a technical support document and referenced by the survey plan. This evaluation may also be included with the technical support document discussed in Section 5.7.2.2 above.

Instrument detection limits are typically quantified in terms of their Minimum Detectable Concentration, or MDC. The MDC is the concentration that a given instrument and measurement technique can be expected to detect 95% of the time under actual conditions of use.

Instruments and methods used for field measurements will be capable of meeting the investigation level in Table 5-8.

Before any measurements are performed, the instruments and techniques to be used must be shown to have sufficient detection capability relative to the applicable DCGLs. The detection capability of a given instrument and measurement technique is quantified by its MDC.

5.7.2.5.1 MDCs for Fixed Measurements

Per NUREG-1507, MDCs for fixed measurements are computed as

$$MDC_{fixed} = \frac{3 + 4.65\sqrt{B}}{Kt} \qquad \text{(Equation 5-28)}$$

where 3 and 4.65 = constants as described in NUREG-1507;
B = background counts during the measurement time interval (t);
t = counting time; and
K = a proportionality constant that relates the detector response to the activity level in the sample being measured.

The proportionality constant K typically encompasses the detector efficiency, self-absorption factors and probe area corrections, as required. The dimensions of the counting interval "t" are consistent with those for the MDC and the proportionality constant K. Thus, "t" would be in minutes to compute an MDC in dpm/100 cm^2.

An example calculation to determine the MDC_{fixed} for the detection of Co-60 with a 100 cm^2 gas proportional detector is shown below.

Assumptions:
Background count rate = 200 cpm
t = 1 minute
B = 200 counts in the measurement time interval (t)
$K = \varepsilon_i \varepsilon_s (A/100)$, where A = area of the detector in cm^2
$\varepsilon_i = 0.38$ cpm/dpm
$\varepsilon_s = 0.25$ (from ISO 7503-1) emissions per disintegration
A = 100 cm^2

$$MDC_{fixed} = \frac{3 + 4.65\sqrt{200}}{(0.38)(0.25)(100/100)(1)} = 724 dpm/100 cm^2$$

Actual values for ε_s will be selected from ISO 7503-1 (Reference 5-12) or NUREG-1507 or empirically determined and documented prior to performing the final status survey.

5.7.2.5.2 MDCs for Beta-Gamma Scan Surveys for Structure Surfaces

As recommended in Draft Guide-4006 (and as incorporated into Section 5.1 of Appendix E to NUREG-1727), MDCs for surface scans for structure surfaces for beta and gamma emitters will be computed via

$$MDC_{structure,scan} = \frac{1.38\sqrt{B}}{\sqrt{p}\varepsilon_i\varepsilon_s\left(\frac{A}{100}\right)t} \text{dpm/100cm}^2 \qquad \text{(Equation 5-29)}$$

where 1.38 = sensitivity index,
B = number of background counts in time interval t,
p = surveyor efficiency,
ε_i = instrument efficiency for the emitted radiation (cpm per dpm),
ε_s = source efficiency (intensity) in emissions per disintegration,
A = sensitive area of the detector (cm^2),
t = time interval of the observation while the probe passes over the source (minutes).

The value of 1.38 used for the sensitivity index corresponds to a 95% confidence level for detection of a concentration at the scanning MDC with a false positive rate of 60%. The numerator in Equation 5-29 represents the minimum detectable count rate that the observer would "see" at the performance level represented by the sensitivity index. The surveyor efficiency (p) will be taken to be 0.5, as recommended by DG-4006 (and incorporated into Section 5.1 of Appendix E to NUREG-1727). The factor of 100 corrects for probe areas that are not 100 cm^2. In the case of a scan measurement, the counting interval is the time the probe is actually over the source of radioactivity. This time depends on scan speed, the size of the source, and the fraction of the detector's sensitive area that passes over the source; with the latter depending on the direction of probe travel. The source efficiency term (ε_s) in Equation 5-29 may be adjusted to account for effects such as self-absorption, as appropriate.

An example calculation to determine the $MDC_{structure,\ scan}$ for the detection of Co-60 with a 100 cm^2 gas proportional detector follows.

Assumptions:

Probe width = 4 inches
Scan rate = 2 inches/sec
Background count rate = 200 cpm
t = 2 seconds = 0.033 minute
B = 6.7 counts in the measurement time interval (t)
p = 0.5
ε_i = 0.38 cpm/dpm
ε_s = 0.25 (from ISO 7503-1) emissions per disintegration
A = 100 cm^2

$$MDC_{structure,scan} = \frac{1.38\sqrt{6.7}}{\sqrt{0.5}(0.38)(0.25)\left(\frac{100}{100}\right)(0.033)} = 1611 \text{dpm}/100\text{cm}^2$$

Actual values for ε_s will be selected from ISO 7503-1 or NUREG-1507 or empirically determined and documented prior to performing the final status survey.

5.7.2.5.3 MDCs for Alpha Scan Surveys for Structure Surfaces

In cases where alpha scan surveys are required, MDCs must be quantified differently than those for beta-gamma surveys because the background count rate from a typical alpha survey instrument is nearly zero (1 to 3 counts per minute typically). Since the time that an area of alpha activity is under the probe varies and the background count rates of alpha survey instruments is so low, it is not practical to determine a fixed MDC for scanning. Instead, it is more useful to determine the probability of detecting an area of contamination at a predetermined DCGL for given scan rates. In general, it is expected that separate alpha and beta surface activity measurements will not be necessary at the HNP and that surrogate measurements will instead be used for alpha surface activity assessments (see Section 5.4.7.3).

For alpha survey instrumentation with a background around one to three counts per minute, a single count will give a surveyor sufficient cause to stop and investigate further. Thus, the probability of detecting given levels of alpha emitting radionuclides can be calculated by use of Poisson summation statistics. Doing so (see MARRSIM Section 6.7.2.2 and Appendix J for details), one finds that the probability of detecting an area of alpha activity of 300 dpm/100cm^2 at a scan rate of 3 cm per second (roughly 1 inch per second) is 90% if the probe dimension in the direction of the scan is 10 cm. If the probe dimension in the scan direction is halved to 5 cm, the detection probability is still 70%. Choosing appropriate values for surveyor efficiency, instrument and surface efficiencies will yield MDCs for alpha surveys for structure surfaces. If for some reason lower MDCs are desired, then scan speeds can be adjusted, within practical limits, via the methods of Section 6.7.2.2 and Appendix J of the MARSSIM.

5.7.2.5.4 MDCs for Gamma Scans of Land Areas

As recommended in DG-4006 (and Section 5.1 of Appendix E to NUREG-1727), the values given in Table 6.7 of MARSSIM may be adopted for gamma scans of land areas if NaI detectors of the dimensions considered in the table are used. If larger NaI detectors (e.g., 3 inch by 3 inch) or other detector types (e.g., plastic scintillator) are used, then the scan MDC will be computed using the methods of Section 6.7.2.1 of MARSSIM. This is the same method as was used to derive the values given in MARSSIM Table 6.7. As an alternative, a specific technical study may be performed and documented to establish efficiency to a soil standard consistent with MARSSIM guidance.

The radionuclides represented in MARSSIM Table 6.7 encompass those expected to be encountered in gamma scans for land areas at the HNP. If desired, the methods of Sections 5.4.7.2 and 5.4.7.3 of this plan may be used to establish scan MDCs based on radionuclide mix ratios. Alternatively, the most limiting value for the radionuclide mix may be used, with most limiting in this case meaning the radionuclide for which the MDC is the largest fraction of its $DCGL_W$ for soil, while still meeting the criteria of 5.5.3.1.

An example calculation to determine the $MDC_{land, scan}$ for the detection of Cs-137 with a 2"x2" NaI detector is shown below.

The Minimum Detectable Count Rate (MDCR) for a surveyor must be calculated prior to determining the scan MDC. The MDCR is dependent upon the background counts expected during time, t, at which the detector is located over the localized contamination. The MDCR for a surveyor is calculated using the following expression:

$$MDCR_{surveyor} = \frac{1.38\sqrt{b}}{\sqrt{p}\ t} \qquad \text{(Equation 5-30)}$$

where b = the background counts expected during time, t
t = the time the detector is located above the localized contamination
p = the surveyor efficiency

Assumptions:

Scan speed = 0.5 meters/sec
Localized contamination diameter = 56 cm
Background count rate = 7000 cpm
b = 130.67 counts in the measurement time interval (t)
t = 0.0187 minute
p = 0.5

$$MDCR_{surveyor} = \frac{1.38\sqrt{130.67}}{\sqrt{0.5}\ (0.019)} = 1195cpm$$

Next, the Minimum Detectable Exposure Rate (MDER) is calculated by dividing the $MDCR_{surveyor}$ by the response to exposure rate factor for Cs-137 of 900 cpm/μR/h from MARSSIM Table 6.7 as follows:

$$MDER = \frac{1195cpm}{900cpm/\mu R/h} = 1.33\mu R/h$$

The Microshield™ modeling code is used to calculate the exposure rate from the localized contamination. Assuming a localized contamination depth of 15 cm, a density of 1.6 g/cm^3, a dose point of 10 cm above the surface and an initial concentration of 5 pCi/g of Cs-137, results in a calculated exposure rate equal to 1.268 μR/h. The scan MDC is calculated by dividing the MDER by the localized contamination exposure rate conversion factor as follows:

$$Cs-137scanMDC = 5pCi/g\frac{1.33\mu R/h}{1.268\mu R/h} = 5.24pCi/g$$

The scan MDCs will be documented prior to performing the final status survey.

5.7.2.6 Typical instrumentation and MDCs

Table 5-~~10~~-11 provides nominal data for the types of field instrumentation anticipated for use in the final survey efforts for the Haddam Neck Plant. The efficiencies listed in Table 5-~~10~~-11 are the total efficiencies in counts/disintegration, and the background count-rates shown are nominal values for generic materials. This table is provided to show the relative sensitivity of some of the types of instruments that will be used during the final status surveys and allow the readers to compare the sensitivities to the DCGLs in Chapter6 of the LTP. The instrument efficiency (ε_I) and source efficiency (ε_s) will be evaluated for instruments used for final status survey measurements and documented as part of the calibration records. This evaluation will include the effects of surface to detector distances, surface coatings and the depth of contamination in material (e.g., concrete) on instrument performance. Instrument calibration sources will be chosen that are appropriate for use for the radionuclides expected to be present post remediation. Instrument readings will be converted to activity by selecting conservative efficiency factors based upon the building surface conditions (including the depth of contamination in concrete).

Table 5-~~10~~11
Available Instruments and Associated MDCs

Instrument	Application	Nominal Efficiency (Not Media Specific)	Nominal Background	Nominal MDC (fixed measurement)	Nominal Scan MDC
pancake GM probe (20 cm^2)	beta-gamma scans or fixed measurements for structure surfaces	17% (Tc-99)	50 cpm	1,050 dpm/100 cm^2 (1 minute count)	3140 dpm/100 cm^2
gas proportional counter (100 cm^2)	alpha or beta scans or fixed measurements for structure surfaces	β plateau: 16% (Tc-99); α plateau: 23% (Am-241)	350 cpm (β plateau); 15 cpm (α plateau)	560 dpm/100 cm^2 (β plateau) 90 dpm/100 cm^2 (α plateau); 1 minute counts	1770 dpm/100 cm^2 (β plateau); 400 dpm/100 cm^2 (α plateau)
plastic scintillator (100 cm^2)	beta-gamma scans or fixed measurements for structure surfaces	30% (Co-60)	600 cpm	390 dpm/100 cm^2 (1 minute count)	1230 dpm/100 cm^2
dual-phosphor scintillator (100 cm^2)	scans or fixed measurements; α and β, independently or simultaneously	20% (Co-60) 18% (Am-241)	300 cpm (β mode); 6 cpm (α mode)	420 dpm/100 cm^2 (β mode); 80 dpm/100 cm^2 (α mode)	1300 dpm/100 cm^2 (β mode); 400 dpm/100 cm^2 (α mode)
ZnS scintillator (100 cm^2)	alpha scans or fixed measurements on structure surfaces	19% (Pu-239)	2 cpm	50 dpm/100 cm^2 (1 minute count time)	400 dpm/100 cm^2
1.25-inch by 1.5-inch NaI	gamma scans for soil	Varies with energy	Varies with energy	N/A	6 pCi/g Co-60 11 pCi/g Cs-137
2-inch by 2-inch NaI	gamma scans for soil	Varies with energy	Varies with energy	N/A	1.5 pCi/g Co-60 6 pCi/g Cs-137
3-inch by 3-inch NaI	in-situ gamma spectroscopy – soil	Varies with energy and geometry	Varies with energy and geometry	0.1 pCi/g Co-60 0.2 pCi/g Cs-137 (10 minute counts)	N/A
HPGe	in-situ gamma spectroscopy – soil	Varies with energy and geometry	Varies with energy and geometry	0.05 pCi/g Co-60 0.05 pCi/g Cs-137 (10 minute counts)	N/A
position-sensitive proportional counter	scan-and-record surveys	Co-60 (β): 18% Am-241 (α): 23%	350 cpm/100 cm^2 beta 15 cpm/100 cm^2 alpha	Typical values are 1,925 dpm/100 cm^2 β and 200 dpm/100 cm^2 α	

5.7.3 Survey Considerations

The available complement of survey instrumentation and techniques will be evaluated to select an integrated approach that will effectively measure residual radioactivity for a given survey unit. The survey design must rely on both the historical site assessment and pertinent data from characterization or remediation support surveys to ensure a complete survey approach. Considerations that will be addressed in the selection of survey instrumentation and techniques include, but are not limited to:

- the types of measurements required;
- suitability for the expected physical and environmental conditions;
- MDCs for advanced survey methods, traditional scanning surveys, fixed measurements, and sampling relative to the $DCGL_W$ and the $DCGL_{EMC}$;
- radionuclide mix, including hard-to-detect and alpha-emitting radionuclides;
- expected spatial variability of any suspected residual contamination;
- accessibility of areas (may impact coverage for scanning surveys); and
- the need for any judgmental assessments to address areas believed to have a higher potential for contamination or situations such as potential sub-surface contamination where prudence would dictate some additional sampling.

5.7.3.1 Survey Considerations for Buildings, Structures and Equipment

The condition of surfaces following decontamination activities can affect the choice of survey instruments and techniques. Removing contamination that has penetrated a surface usually involves removing the surface material. As a result, the floors and walls of decontaminated facilities can be scarred or broken up and uneven. Such surfaces are more difficult to survey because it is not possible to maintain a fixed distance between the detector and the surface. In addition, scabbled or porous surfaces may attenuate radiation - particularly alpha and low-energy beta particles, and pose an increased risk of damage to detector probe faces. Surface irregularities may also cause difficulty in rolling or maneuvering detector systems on wheels.

Part of the planning for the final status survey of a particular survey unit will include an evaluation of the surfaces to be monitored. For conventional instrumentation, surface anomalies will be identified as part of this process and will be taken into account when selecting efficiencies to convert instrument readings to activity and in the calculation of the corresponding MDCs. Conservative values will be chosen based upon surface conditions. If the condition of the surface in the area changes in a more conservative direction (e.g. shorter detector to surface distance), the effect on the MDC will be assessed but may not be re-derived. If the condition of the surface changes in a non-conservative direction (e.g. different construction material which has higher natural radioactivity) the MDC will be assessed and re-derived.

Expansion joints, stress cracks, floor/wall interfaces, and penetrations into floors and walls for piping, conduit, anchor bolts, etc., are potential sites for accumulation of contamination and pathways for migration into sub-floor soil and hollow wall spaces. Roof surfaces and drainage points are also important survey locations. In some cases, it may be necessary to core, drill, or use other methods as necessary to gain access to areas for sampling.

5.7.3.1.1 Activity Beneath Surfaces

Floors, walls, and ceilings of structures may have surface irregularities such as cracks and crevices that require special consideration in the survey process. Such considerations may consist of fixed measurements, longer count times, adjustments to counting efficiencies, sampling of material, or any combinations of these approaches.

Plant areas where residual radioactive material beneath a painted surface is known or suspected to be present will also require special consideration. Sampling will be performed, as appropriate, to confirm or deny the presence of residual activity. If activity is found, the samples should be used to determine both the radionuclides that are present and the density-thickness of the paint layer(s) in order to assess the need for correction factors for counting efficiencies. Such corrections, if required, will be determined following the guidance given in Section 5 of NUREG-1507. The effect of any such corrections on instrument MDCs will be assessed to ensure that measurements can still be performed with the required sensitivity relative to the applicable DCGLs.

5.7.3.1.2 Below-Grade Building Foundations

5.7.3.1.2.1 Basements

The interior surfaces of the containment liner located below grade ~~below-grade basements (i.e., those more than four feet below ground level)~~ will be surveyed and decontaminated to meet the Building Occupancy DCGLs discussed in Chapter 6. The contamination levels used in the design of this survey will be used as the basis of the inventory of radioactivity released from the containment liner and embedded piping in the Basement Fill Model. Exterior surfaces of below-grade basements will be evaluated using the historical site assessment and other pertinent records to determine the potential for sub-surface contamination on these surfaces of below-grade basements. One method available to evaluate the exterior surfaces is the use of core bores through foundation or walls and the taking of soil samples at locations having a high potential for the accumulation and migration of radioactive contamination to sub-surface soils. These biased locations for soil and concrete assessment could include stress cracks, floor and wall interfaces, penetrations through walls and floors for piping, run-off from exterior walls, and leaks or spills in adjacent outside areas, etc. If the soil is found to be free of residual radioactivity at the biased locations, it will be assumed that the exterior surface of the foundation is also free of residual activity. Otherwise, additional sampling may be necessary to determine the extent of decontamination and remediation efforts. Another method available for evaluating the exterior surfaces of below-grade foundations is gamma well logging. Soil in biased locations next to the exterior of the buildings may be evaluated using this technique. This technique can provide for rapid isotopic analysis of soils without sampling. This approach is further described in Section 5.7.3.1.2.2.

For basements that are to remain after removal of all except ISFSI areas from the license, ~~an~~ any FSS will required be performed on the internal surfaces. The results of characterization of the external surfaces will be used in the design and DQOs of the subsurface FSS to be performed in the area.

Basement concretes that is to remain (except the containment liner as discussed above) will be characterized to determine the extent of any volumetric contamination of the concrete in accordance with Table 5-10. The results will be used in the calculation of the Basement Fill Model.

5.7.3.1.2.2 Footings

After completion of final status survey activities of the remaining portions of structures, some concrete may remain in the form of building footings.

There are several building foundations that are to remain. The current approach includes the demolition of buildings to four feet below grade. This will remove ground-level floors and portions of footings and foundation supports. Surfaces of these below-grade structures will be evaluated using the historical site assessment and other pertinent records to determine the potential for sub-surface contamination on the surfaces of the foundations. Soil samples will be taken in the vicinity of the footings/foundation. If the soil is found to be free of residual radioactivity at the biased locations, it will be assumed that the exterior surface of the foundation is also free of residual activity. If soil samples contain residual radioactivity, the exterior surfaces sampled or ~~will be further~~ assessed as follows: ~~characterized. The results of characterization of the external surfaces will be used in the design and DQOs of the subsurface FSS to be performed in the area.~~ will be

These footings will be sampled using concrete sampling (at least 3 samples per footing or group of footings) or assessed using the results of nearby soil sampling to determine their radioactivity content. If the assessment is done using soil samples, the following methodology will be used:

- The average of the soil samples result (for plant-related radionuclides detected) will be calculated for each footing (at least 3 samples per footing or group of footings).
- For Tritium:
  - Using the soil distribution coefficient data (Kds) shown in Table F-1 and equation 6-1 of the LTP, the groundwater concentration in equilibrium with the soil concentrations in the last bullet will be determined
  - Using the concrete distribution coefficients shown in Table F-4 of LTP and equation 6-1 the concrete concentrations in equilibrium with the groundwater concentrations calculated in the last bullet will be determined.
- For radionuclides other than H-3:
  - At least six pairs of concrete samples with adjacent soil samples will be collected at locations affected by plant leakage and/or groundwater contamination
  - The average and %CV (coefficient of variation) of the ratios of concrete concentrations to soil concentrations will be calculated
  - If the %CV of the data is less then 25 %, the average ratio will be used to determine concrete concentrations from adjacent soil samples
  - If the %CV is 25% or greater more samples will be taken until a satisfactory variance is calculated or the worst case ratio will be used

The results of this sampling or assessment will then be used as input to the calculation of future groundwater dose using the basement fill model as discussed in Chapter 6.

The results of sampling and/or assessment of the external surfaces will be used in the design and DQOs of the subsurface FSS to be performed in the area. ~~This approach is discussed in Chapter 6.~~ Following any required remediation approximately 4 feet of backfill soil would be placed over the footings to restore the area to grade elevation.

The need for a final status survey of the areas with below grade structures will be determined on a graded approach.

For footings and other structures to remain that have a very low or no potential for contamination such as:

- Buildings outside the RCA
- Building shown by characterization sampling to be free of residual radioactivity

The final status survey will consist of a surface and subsurface FSS of the area including the subsurface structures.

5.7.3.1.3 Sewer Systems, Plumbing and Floor Drains

Residual radioactivity in sanitary piping or floor drains will be evaluated in the same manner as for non-structural plant systems or components, discussed in Sections 5.4.7.5 and 5.6. Assessment of residual activity levels in piping or floor drains will be via sampling of sediments, fixed measurements, scanning, or a combination of these methods, as appropriate.

All non-RCA sanitary systems at the Haddam Neck Plant drain to on-site leach fields. These systems are independent of other plant systems and all surface water or storm drains. If any residual radioactivity is suspected in portions of the sanitary plumbing systems, evaluations for both the leach fields and the associated system piping may be required. Radiological assessments of piping will be made as described in Section 5.6 of this plan, i.e., by full length surveys of interior surfaces. Evaluations required for any affected leach fields will be made as described in Section 5.7.3.2.2 of this plan, for sub-surface activity. All operable RCA-located systems currently drain to the aerated drains system and are part of the normal plant effluent. Thus, there is no leach fields associated with these systems. During the plant lifetime, toilet facilities, showers and sinks, contained within the RCA, drained to the plant sanitary system and associated leach field. Any piping associated with the systems, which is proposed to remain following decommissioning will be evaluated as described above.

5.7.3.1.4 Ventilation Ducts – Interiors

Radiological assessments of ventilation systems will be made by taking measurements at appropriate access points where activity levels should be representative of those on the interior surfaces. Assessments may also be made using in-situ gamma-spectroscopy provided adequate instrument efficiencies and detection limits can be achieved. Exterior surfaces of such systems will be evaluated as part of the building or structure in cases where the system is attached to it or is otherwise an integral component.

5.7.1.3.5 Piping and Embedded Piping

The construction of the Haddam Neck Plant was such that there is not expected to be a significant amount of embedded piping to consider in the final survey effort. Most of the radiologically affected piping is in pipe trenches, and thus can be accessed and removed as necessary. Currently approximately 1000 feet of embedded piping is forecasted to remain after Final Status Survey. Any affected embedded piping remaining at the time of Final Status Survey is expected to be in wall penetrations between areas. Sections of such piping are not expected to be very long (no longer than the wall thickness) and thus should be able to be sampled or surveyed as appropriate to evaluate residual activity levels against the applicable release criteria. The Final Status Survey design of areas containing embedded piping will address this media

during the DQO process. Expected outputs of the DQO process include defining the appropriate type of data to collect; survey measurement processes and survey instrument sensitivity; potential contaminants and appropriate DCGL for the assumed exposure pathway.

5.7.3.1.6 Activated Concrete

Although concrete cores have been obtained in Containment, most have been obtained in areas not subject to the highest levels of neutron activation. Areas subject to the highest neutron activation are currently inaccessible, and, therefore, specific characterization data is not yet available in all areas. However, neutron activation data from Maine Yankee, Trojan and Yankee Nuclear Power Station indicate that H-3 and Fe-55 are present in the highest concentrations. Other radionuclides such as C-14, Co-60, Eu-152 and Ni-63 are also present. Based upon these data, the activation products Eu-152 and Eu-154 were included in the list of radionuclides expected to be present at HNP (Table 2-12).

As the decommissioning progresses and high dose rate components are removed, additional characterization of structures within Containment, including activated concrete and structural components, will take place. These characterization samples will typically be analyzed by gamma spectroscopy with some samples being analyzed for "hard-to-detect" radionuclides. Therefore, a representative sample of characterization and final status survey samples will be screened for neutron activation.

determine the radionuclide concentrations to be used in "Basement Fill Model" calculation.

In-situ gamma spectroscopy may be used to perform remediation surveys for activated concrete to ~~demonstrate that it meets the applicable volumetric DCGLs.~~ If in-situ gamma-spectroscopy is selected for use, a technical support document will be developed which describes the technology to be used and how the technology meets the objectives of the survey. This document will be available for NRC inspection in support of final status survey activities.

of ~~for~~ calculating future groundwater dose with the Basement Fill model.

Such surveys would be conducted so that 100% ~~of the affected volume was covered~~ in overlapping measurements. Embedded materials (such as rebar) will be treated as concrete for purposes ~~of assigning, DCGLs.~~ Assessments for any "hard-to-detect" radionuclides that might be present in activated concrete will be by either direct measurements (core-bores or equivalent) or by establishing surrogate concentrations ~~DCGLs~~ for these radionuclides relative to some radionuclide easily measured via gamma-spectroscopy (Co-60, for example). Surrogate ratios will be established using pertinent characterization data for the survey unit of interest. Final status surveys of these areas will also include collection and analysis of concrete and rebar samples.

Basements that may contain activated concrete and are to remain will be characterized to determine the extent of any volumetric contamination of the concrete in accordance with Table 5-10. The results will be used in the calculation of the Basement Fill Model.

5.7.3.1.7 Systems and Equipment Interiors and Exteriors

Surface activity assessments for non-structural systems and components will be made by making measurements at traps and other appropriate access points where activity levels should be representative of those on the interior surfaces. Assessments may also be made via in-situ gamma-spectroscopy, provided adequate instrument efficiencies and detection limits can be achieved. If necessary, scaling factors may be applied to establish gross activity levels via radionuclide-specific measurements or other assessments, as appropriate.

5.7.3.2 Survey Considerations for Outdoor Areas

5.7.3.2.1 Residual Radioactivity in Surface Soils

In this context, surface soil refers to outdoor areas where the soil is, for purposes of dose modeling, considered to be uniformly contaminated from the surface down to some specified depth. These areas will be surveyed through combinations of sampling, scanning, and in-situ measurements, as appropriate.

5.7.3.2.2 Residual Radioactivity in Subsurface Soils

Residual radioactivity in subsurface soils refers to residual radioactivity residing under the top 6 inches of soil or is underneath structures such as building floors/foundations. Such areas include, but are not limited to, areas under buildings, building floors/foundations, or components where leakage was known or suspected to have occurred in the past; on-site storage areas where radioactive materials have been identified; and areas containing spoils from past dredging of the discharge canal. However, the assessment of all subsurface soil contamination is not currently complete. Soil in difficult to access areas such as under buildings will be deferred until later in the decommissioning process. As a part of survey planning, borehole logs will be reviewed, when available.

The DQO process for subsurface areas will be similar to the DQO process used for other surveys at HNP (e.g., final status survey for surface soils). However, there may be differences in design input parameters as necessary to satisfy the objectives of the plan. Additional detail regarding subsurface input parameters and methodology are provided below. Surveys (i.e., scoping, characterization, remediation and final status survey) for subsurface areas will be performed under a documented survey plan developed using the DQO process. The level of effort with which the DQO process is used as a planning tool is commensurate with the type of survey and the necessity of avoiding a decision error. This is the graded approach of defining data quality requirements as described previously in the LTP. For example scoping and characterization survey plans intended to collect data might only require a survey objective and the instrumentation and analyses specifications necessary to meet that survey objective. Remediation and final status survey plans which require decisions would need additional effort during the planning phase according to the level of risk of making a decision error and the potential consequences of making that error.

The DQA process will be used to assess data and demonstrate achievement of the sampling plan objectives. The level of effort expended during the DQA process will typically be consistent with the graded approach used during the DQO process. The DQA process will include a review of the DQOs and survey plan design, review of preliminary data, use of appropriate statistical testing when applicable (see discussion below), verify the assumptions of the statistical tests and draw conclusions from the data.

Evaluation of subsurface soil at HNP during final status survey will be a combination of systematic and biased measurements. Measurements may be either in-situ gamma spectroscopy by well logging or other advanced technology, provided the MDC meets the criteria discussed in Section 5.7.2, or by sampling. If advanced technology instrumentation is selected for use, a technical support document will be developed which describes the technology to be used and how the technology meets the objectives of the survey. This document will be available for NRC inspection in support of final status survey activities. Sample locations will be selected randomly, for Class C areas, or by random-start systematic grid for Class A and B areas, supplemented with biased measurements. Biased measurements will be obtained at the locations

of localized contamination. Where samples are taken, each 3-meter core (or other means of collection, such as test pits) will be homogenized and measured.

The horizontal extent of contamination will only be established for judgmental sampling and for samples within a systematic sampling area that exceed the $DCGL_{EMC}$. For the case where the $DCGL_{EMC}$ comparison is made, the value used for the area factor will be determined from the area bounded by the adjacent samples or by the area bounded by additional samples at or below the $DCGL_w$. This approach is consistent with the model used to calculate DCGLs in Chapter 6.

As discussed in Section 2.3.3.1.5, subsurface soils at HNP are divided into 3 classifications: Class A, Class B and Class C. This classification defines the measurement or sample density. There will be 31 measurement locations (approximately one per 500 m^2) in Class A areas. In addition, biased measurements or samples will be obtained at the locations of localized remediation efforts where subsurface leaks are suspected to have created soil contamination. Random measurements or samples will be obtained in Class B and Class C areas. There will be 25 measurement locations in a Class B area. There will be 15 measurement locations in Class C areas. In addition, biased measurements or samples will be obtained in Class B and Class C areas based upon characterization data and professional judgment. If a systematic or random sample location falls on a building foundation, a sample will be obtained at that location unless the building is in contact with bedrock. The range of the number of measurements in Class A, B, and C areas (31 measurements in the Class A area to 15 measurements in Class C areas) corresponds to the range of values for N (for Sign test, or N/2 for WRS), considering $\alpha = 0.05$, $\beta = 0.05$, and $1.0 \leq \Delta/\sigma \leq 3.0$. All samples will be evaluated against the soil DCGLs by using either the Sign or WRS test.

Investigation levels applicable to surface soils (given in Table 5-8) will be applied to subsurface soils. Similarly the area factors for surface soils (given in Table 5-5) will be applied to subsurface soils. That is no sample can exceed the $DCGL_{EMC}$ without an investigation being performed. These investigations would be similar to those performed for surface soils.

Samples will be obtained to a depth of 3 meters or bedrock, whichever is reached first. These samples will be homogenized over the entire depth of the sample obtained. In cases where refusal is met because of bedrock, the sample will be used "as is." In cases where a non-bedrock refusal is met prior to the 3-meter depth, the available sample will be used to represent the 3-meter sample, if the viable sample is at least 1.5 meters in depth. If a non-bedrock refusal is met before the 1.5-meter depth, then a new sample will be obtained within a 3-meter radius from the original location. All samples will be analyzed by gamma spectrometry. Because the mobility of some of the radionuclides, believed to be present, is not well understood, some of the samples will undergo analysis for all hard-to-detect radionuclides. A minimum of 5% of the samples will be analyzed for hard-to-detect radionuclides. During specific investigations, such as the identification of the horizontal extent of contamination, analysis of a larger percentage of samples for hard-to-detect radionuclides will be performed.

5.7.3.2.3 Paved areas

Paved areas that remain at the HNP following decommissioning activities may require surveys for residual radioactivity on the surface, beneath the surface, or both. As part of the survey design and planning process, historical information will be reviewed to determine whether radiological incidents or plant alterations have occurred in the survey unit. Where indications are that impacted soil could have been mixed by grade work prior to paving, this will be factored into final survey design to establish a reasonable depth of disturbed soil for evaluation. If it is determined that the soil beneath pavement has been impacted, the final status survey will incorporate appropriate surveys and sampling.

If residual radioactivity is primarily on or near the surface of the paved area, for purposes of surveying, measurements will be taken as if the area were surface soil. If the residual radioactivity is primarily beneath the paving, it will be treated, for purposes of surveying, as subsurface residual radioactivity.

5.7.3.2.4 Groundwater Assessments

Assessments of any residual activity in groundwater at the Haddam Neck Plant will be via groundwater monitoring wells. The monitoring wells installed at the site will monitor groundwater at both deep and shallow depths. Section 2.3.3.1.6 describes, in depth, the groundwater monitoring to be conducted.

5.7.3.2.5 Bedrock Assessments

Exposed bedrock from the demolition of structures or the remediation of contaminated soils creates another area requiring a methodical radiological assessment.

Several areas of the site will be excavated to bedrock either through the demolition of buildings or the removal of contaminated soils. Initial excavation in the Tank Farm area to bedrock will allow for data to be assessed on the potential magnitude and distribution of contamination within the bedrock. This assessment will include the determination of:

- The degree of contamination on the bedrock surface,
- The degree of contamination migration into bedrock cracks and fissures, and
- The observation of surface conditions of the bedrock.

As remediation progresses, the bedrock surfaces will be cleaned of readily removable material using techniques such as vacuuming and air pressure removal (combined with vacuum collection of removed material). Following remediation, the radiological conditions will be assessed prior to backfilling the excavation. The backfill will ultimately consist of clean fill.

The dose pathway that would apply to such open bedrock excavations will be from potential future groundwater contamination since other pathways such as direct exposure, and plant uptake would not apply to this material (clean backfill provides substantial shielding to the surface and farming plants would not be grown in bedrock). Therefore, the post-remediation field assessment and dose assessment methods focus on the radioactivity inventory potentially available to future uncontaminated groundwater in contact with the remaining radioactivity in bedrock from each bedrock excavation.

The monitoring of the bedrock area will be through the installation and sampling of groundwater monitoring wells. Once the bedrock area is backfilled with clean fill material, the dewatering wells, pumps, used to suppress the groundwater will be turned off allowing the groundwater to return to an equilibrium condition in the unconsolidated backfill materials. Monitoring wells will be installed at locations to provide groundwater samples for monitoring. The installed monitoring wells will be sampled quarterly for at least 18 months to include two springtime periods when groundwater has its greatest impact.

5.7.3.2.6 Surface Water and Sediments

Sediments will be assessed by collecting samples within locations of surface water ingress or by collecting composite samples of bottom sediments, as appropriate. Such samples will be collected using

approved procedures based on accepted methods for sampling of this nature. Sample locations will be established using the methods of Section 5.5.1 of this plan. Scanning in such areas is not applicable.

Note that per the agreement with the Connecticut DPUC, radiological sampling of fish, water and sediment will be performed for the pond (Survey Area 9508).

Also note, per the agreement with the Connecticut DPUC, the sampling density will be adjusted in the area of the canal, from the outfall (the beginning of the canal closest to the plant) to 50 feet past the weir, to be twice the density that would otherwise be required of a Class 2 survey unit.

Sediment samples will be evaluated against the DCGLs for soil. This is considered appropriate given that the action that would result in the greatest radiological impact to future inhabitants of the site would be to dredge up the sediment and use it for farming. If the sediment is left in place, then use of the soil DCGLs is conservative since many of the pathways considered in developing the soil DCGLs (direct exposure, uptake by plants, etc.) would not apply.

Assessment of residual activity levels in surface water drainage systems will be via sampling of sediments, fixed measurements, or both, as appropriate, making measurements at traps and other appropriate access points where activity levels should be representative or bound those on the interior surfaces.

5.7.3.2.7 Storm Drains and Buried Piping Other Than Described in Section 5.7.3.1.5

Most buried piping (including storm drains) will be removed from the site; however, any buried piping that remains following decommissioning activities that has a potential to contain residual activity will be surveyed using the criteria given in Section 5.4.7.5.

5.7.3.3 Surveillance Following Final Status Surveys

Isolation and control measures will be implemented through approved plant procedures and will remain in force throughout final survey activities and until there is no risk of contamination from decommissioning or the survey area has been released from the license. In the event that isolation and control measures established for a given survey unit are compromised, evaluations will be performed and documented to confirm that no radioactive material was introduced into the area that would affect the results of the Final Status Survey.

To provide additional assurance that land areas and the limited number of structures that have successfully undergone FSS remain unchanged until final site release, documented periodic evaluations of the FSS areas will be performed. The periodic evaluation will consist of:

- A walkdown of the areas to check for proper postings,
- Check for materials introduced into the area since the last evaluation,
- Any general disturbance that could change the FSS including the potential for contamination from adjacent decommissioning activities,
- A review of the 50.75(g) files.

Evaluations will be documented and controlled in accordance with site procedures

5.7.3.3.1 Surveillance of Structures

Routine surveys will be performed on structures that have completed their final status survey until the structure is backfilled. These routine operational health physics surveys will be used to verify that the as-left radiological conditions in the area have not changed. These routine surveys will typically include survey locations on the floor and on lower walls, and areas of ingress, egress, and storage. Locations will be selected on a judgmental basis, based on technician experience and conditions present in the survey area at the time of the evaluation, but primarily designed to detect the migration of loose surface contamination from decommissioning activities taking place in adjacent areas and other areas in close proximity that could cause a potential change in conditions.

If the area is suspect following the routine surveillance survey, then a full FSS survey of the affected unit(s) will be performed in accordance with the LTP. The results for the FSS re-survey and investigations surveys will be documented and maintained in the FSS files for the affected survey unit(s). Additionally, for any area that has completed FSS activities, any soil, sediment, or equipment relocated to that area will require a demonstration that the material being introduced does not result in resident radioactivity that is statistically different that that identified in the FSS. Once a structure has been backfilled, the periodic surveillance will be similar to the surveillance employed for open land areas.

5.7.3.3.2 Surveillance of Open Land Areas

Open land areas that have been final status surveyed will be evaluated periodically, not to exceed semi-annually.

If the area is suspect following the evaluation an investigation survey will be performed to confirm the FSS surveys validity. This investigation survey will involve judgmental sampling of the suspect areas. If the results of the investigation survey indicates that contamination is statistically different than the initial FSS results (>2 standard deviations from the mean), then the investigation survey will be increased to include a larger physical area than the initial investigation survey. If the final results of the investigation survey are statistically different than the FSS survey results, then a full FSS survey of the affected areas will be performed in accordance with the LTP. The results of the FSS re-survey and investigation surveys will be documented and maintained in the FSS files for the affected survey units. Additionally, for any area that has completed FSS activities, any soil or sediment relocated to that area will require demonstration that the material introduced does not result in residual radioactivity that is statistically different than that in the FSS.

5.7.3.3.3 Surveillance of Bedrock Areas

Generally, bedrock areas will not remain exposed for a period of time such that surveillance would be necessary. Typically the bedrock area will be assessed for radiological conditions and then backfilled with clean fill material. Any necessary groundwater level adjustments will be made, which include stopping groundwater dewatering from the bedrock area to allow the "normal" groundwater levels to be restored and the installation of any monitoring wells needed to support ongoing radiological groundwater monitoring.

If the bedrock area is suspect, following the evaluation, an investigation assessment will be performed to confirm the radiological assessment's validity. This investigation assessment will involve judgmental sampling of boundary and/or potential access points to bedrock area. If the results of the investigation

assessment are greater than 2 standard deviations from the mean, then the investigation will be increased to include a larger physical area than the initial investigation assessment. If the final results of the investigation assessment are statistically different than the radiological assessment results, then a full radiological assessment of the affected bedrock areas will be performed in accordance with Section 5.7.3. The results of the re-assessment and investigation assessment will be documented and maintained in the bedrock assessment files for the affected bedrock areas.

5.8 Survey Data Assessment

The Data Quality Assessment (DQA) process, being adopted at HNP, is an evaluation method used during the assessment phase of FSS to ensure the validity of FSS results and demonstrate achievement of the survey plan objectives. The level of effort expended during the DQA process will typically be consistent with the graded approach used during the DQO process. The DQA process will include a review of the DQOs and survey plan design, will include a review of preliminary data, will use appropriate statistical testing when applicable (statistical testing is not always required, e.g., when all sample or measurement results are less than the $DCGL_W$), will verify the assumptions of the statistical tests, and will draw conclusions from the data.

Prior to evaluating the data collected from a survey unit against the release criterion, the data are first confirmed to have been acquired in accordance with all applicable procedures and QA/QC requirements. Any discrepancies between the data quality or the data collection process and the applicable requirements are resolved and documented prior to proceeding with data analysis. Data assessment will be performed, by trained personnel, using approved site procedures.

The first step in the data assessment process is to convert all of the survey results to DCGL units. Next, the individual measurements and sample concentrations will be compared to DCGL levels for evidence of small areas of elevated activity or results that are statistical outliers relative to the rest of the measurements (see Section 5.5.3.1). Graphical analyses of survey data that depict the spatial correlation of the measurements are especially useful for such assessments and will be used to the extent practical. The results may indicate that additional data or additional remediation and resurvey may be necessary. If this is not the case, the survey results will then be evaluated using direct comparisons or statistical methods, as appropriate, to determine if they exceed the release criterion. If the release criterion has been exceeded or if results indicate the need for additional data points, appropriate further actions will then be determined.

Interpreting the results from a survey is most straightforward when all measurements are higher or lower than the $DCGL_W$. In such cases, the decision that a survey unit meets or exceeds the release criterion requires little in terms of data analysis. However, formal statistical tests provide a valuable tool when a survey unit's measurements are neither clearly above nor entirely below the $DCGL_W$.

The first step in evaluating the data for a given survey unit is to draw simple comparisons between the measurement results and the release criterion. The result of these comparisons will be one of three conclusions: 1) the unit meets the release criterion; 2) the unit does not meet the release criterion; or 3) no conclusion can be drawn from simple comparisons and thus one of the non-parametric statistical tests must be applied. The initial comparisons made for the results for a given survey unit depend on whether or not the results are to be compared against a background reference area.

If the survey data are in the form of gross (non-radionuclide-specific) measurements or if the radionuclide of interest is present in background in a concentration that is a relevant fraction of the $DCGL_W$, then the initial data evaluation will be as described in Table 5-~~11~~12.

Table 5-~~11~~12
Initial Evaluation of Survey Results
(Background Reference Area Used)

Evaluation Result	Conclusion
Difference between the maximum concentration measurement for the survey unit and the minimum reference area concentration is less than the $DCGL_W$	Survey unit meets the release criterion
Difference between the average concentration measured for the survey unit and the average reference concentration is greater than the $DCGL_W$	Survey unit does not meet the release criterion
Difference between any individual survey result and any individual reference area concentration is greater than the $DCGL_W$ and the difference between the average concentration and the average for the reference area is less than the $DCGL_W$	Conduct either the Wilcoxon Rank Sum test or the Sign test; and the EMC test

If the survey data are in the form of radionuclide-specific measurements and the radionuclide(s) of interest is not present in background in a concentration that is a relevant fraction of the $DCGL_W$, then the initial data evaluation will be as described in Table 5-~~12~~13.

Table 5-~~12~~13
Initial Evaluation of Survey Results
(Background Reference Area Not Used)

Evaluation Result	Conclusion
All measured concentrations less than the $DCGL_W$	Survey unit meets the release criterion
Average concentration exceeds the $DCGL_W$	Survey unit does not meet the release criterion
Individual measurement result(s) exceeds the $DCGL_W$ and the average concentration is less than the $DCGL_W$	Conduct the Sign test and the EMC test

5.8.1 Wilcoxon Rank Sum Test

Gross activity measurements or measurements for which the radionuclide of interest exists in background in concentrations that are a relevant fraction of the $DCGL_W$ may be evaluated using the Wilcoxon Rank Sum (WRS) test. In the WRS test, comparisons are made between the survey results for a given survey unit and reference (background) data for comparable materials. However, for survey units which contain multiple materials having different backgrounds, it may be advantageous to background-subtract gross activity measurements (using paired observation) and apply the Sign test (see Section 5.8.2).

The WRS test tests the null hypothesis that the median concentration in the survey unit exceeds that in the reference area by more than the $DCGL_W$. The null hypothesis is assumed to be true unless the statistical

test indicates that it should be rejected in favor of the alternative. The alternative hypothesis is that the median concentration in the survey unit exceeds that in the reference area by less than the $DCGL_W$. Note that some or all of the survey unit measurements may be larger than some reference area measurements, while still meeting the release criterion. Indeed, some survey unit measurements may exceed some reference area measurements by more than the $DCGL_W$. The result of the hypothesis test determines whether or not the survey unit as a whole is deemed to meet the release criterion. The EMC is used to screen individual measurements.

The WRS test is applied as described in the following steps:

1. Adjust the reference area measurements by adding the $DCGL_W$ to each one.
2. Pool the adjusted reference area measurements and the sample (survey unit) measurements and rank them in increasing order from 1 to the total number of data points (reference measurements plus sample measurements).
3. For any measurements that have the same value, the rank assigned to that set of measurements is the average of their ranks.
4. Sum the ranks of the adjusted reference area measurements.
5. Compare the sum of the adjusted reference area measurements (W_r) with the critical value from Table I.4 of the MARSSIM for the appropriate values of m (the number of reference measurements), n (the number of sample measurements), and α (the decision error rate).

If the value W_r determined from steps 1 through 5 above exceeds the critical value from Table I.4 of the MARSSIM, then the null hypothesis is rejected and the alternate accepted. In other words, the results show that the survey unit meets the release criterion.

Note that the WRS test described in steps 1 through 5 above assumes that there are no "less than" results in the data set, i.e., that all of the data points have a quantitative value rather than "background" or "less than MDC." Though it is not anticipated that data of this nature would be among that collected for a final status survey, if it is encountered and must be used, the method described in Section 8.4.2 of the MARSSIM will be used to assign rank to these values. If more than 40% of the data collected for a final status survey are "less than" values, then the WRS test cannot be used.

5.8.2 Sign Test

Radionuclide specific measurements for which the radionuclide(s) of interest either does not exist in background or is not present in a concentration that is a relevant fraction of the $DCGL_W$ will be evaluated using the Sign test. In addition, the Sign test may be used to evaluate gross activity measurements from survey units containing multiple materials by subtracting the appropriate background using paired measurements.

The null hypothesis tested by the Sign test is the same as that used for the WRS test. As with the WRS test, some individual survey unit measurements may exceed the $DCGL_W$ even when the survey unit as a whole meets the release criterion. In fact, a survey unit average that is close to the $DCGL_W$ might have almost half of its individual measurements greater than the $DCGL_W$. Such a survey unit may still not exceed the release criterion. As with the WRS test, the EMC is used to screen individual measurements.

The Sign test is applied as described in the following steps:

1. List the survey measurements.
2. For each survey unit measurement, subtract the measurement from the $DCGL_W$ and record the differences.
3. Discard any difference that is exactly zero and reduce the total number of measurements (N) by the number of zero differences.
4. Count the number of positive differences. This value is the test statistic S+.
5. Compare the number of positive difference (S+) to the critical values from Table I.3 of MARSSIM for the appropriate values of N (total measurements) and α (decision error rate). (A positive difference corresponds to a measurement below the $DCGL_W$ and contributes evidence that the survey unit meets the release criterion.)

If S+ is greater than the critical value in Table I.3, then the null hypothesis is rejected and the alternate accepted.

Note that "measurements" in Step 1 above refers to the net result in cases where background-subtracted gross activity measurements (using the paired observation methodology) are being evaluated.

Though it is not anticipated, if any of the data collected from a final status survey are reported as "less than MDC" or as background, actual values will be assigned, even if negative, for purposes of applying the Sign test.

5.8.3 Elevated Measurement Comparison

The Elevated Measurement Comparison (EMC) consists of comparing each measurement from the survey unit with the investigation levels discussed in Section 5.5.3. The EMC is performed for both measurements obtained on the systematic-sampling grid and for locations flagged by scanning measurements. Any measurement from the survey unit that is equal to or greater than an investigation level indicates an area of relatively high concentrations that should be investigated, regardless of the outcome of the nonparametric statistical tests. Thus, the use of the EMC against the investigation levels may be viewed as assurance that unusually large measurements will receive proper attention regardless of the outcome of those tests and that any area having the potential for significant dose contributions will be identified. The EMC is intended to flag potential failures in the remediation process. It should not be used as the primary means to identify whether or not a unit meets the release criterion.

If residual radioactivity exists in an isolated area of elevated activity in addition to residual radioactivity distributed relatively uniformly across a survey unit, the unity rule will be used to ensure that the total dose is within the release criterion, i.e.,

$$\frac{\delta}{DCGL_W} + \frac{\overline{C}_{elevated} - \delta}{(AreaFactor) \times DCGL_W} < 1 \qquad \text{(Equation 5-31)}$$

where: δ = average concentration outside the elevated area,
$\overline{C}_{elevated}$ = average concentration in the elevated area.

A separate term will be used in Equation 5-31 for each elevated area identified in a survey unit.

Note that EMC considerations generally apply only to Class 1 survey units, since areas of elevated activity should not exist in Class 2 or Class 3 survey units.

5.8.4 Unity Rule

When radionuclide specific measurements are made in survey units having multiple radionuclides, compliance with the radiological release criterion will be assessed through use of the unity rule, also known as the sum of fractions. The unity rule, represented in the expression below, is satisfied when radionuclide mixtures yield a combined fractional concentration limit that is less than or equal to one, i.e.:

$$\frac{C_1}{DCGL_1}+\frac{C_2}{DCGL_2}+...+\frac{C_n}{DCGL_n}\leq 1 \qquad \text{(Equation 5-32)}$$

where:

C_n = Concentration of radionuclide n
$DCGL_n$ = DCGL for radionuclide n

5.8.5 Data Assessment Conclusions

The result of the data assessment is the decision to reject or not to reject the null hypothesis. Provided that the results of investigations triggered by the EMC were resolved, a rejection of the null hypothesis leads to the decision that the survey unit meets the release criterion. If the data assessment concludes that the null hypothesis cannot be rejected, this may be due to one of two things: 1) the average residual concentration in the survey unit exceeds the $DCGL_W$; or 2) the analysis did not have adequate statistical power. "Power" in this context refers to the probability that the null hypothesis is rejected when it is indeed false. Quantitatively, the power is $1 - \beta$, where β is the Type II error rate (the probability of accepting the null hypothesis when it is actually false). A retrospective power analysis can be used in the event that a survey unit is found not to meet the release criterion to determine if this is indeed due to excess residual activity or if it is due to an inadequate sample size.

Retrospective power analyses, if necessary, will be performed following the methods of MARSSIM Sections I.9 and I.10 for the Sign test and WRS test, respectively. If the retrospective power analysis indicates insufficient power, then an assessment will be performed to determine whether the observed median concentration and/or observed standard deviation are significantly different from the estimated values used during the DQO process. The assessment may identify and propose alternative actions to meet the objectives of the DQOs. These alternative actions may include failing the unit and starting the DQO process over, remediating some or all of the survey unit and starting the DQO process over and adjusting the LBGR to increase sample size. For example, the assessment determines that the median residual concentration in the survey unit exceeds the $DCGL_W$ or is higher than was estimated and planned for during the DQO process. A likely cause of action might be to fail the unit or remediate and resurvey using a new sample design. As another example, the assessment determines that additional samples are necessary to provide sufficient power. One course of action might be to determine the number of additional samples and collect them at random locations. Note, this method may increase the Type I error, therefore agreement with the regulator will be necessary prior to implementation. Another action would be to resample the survey unit with a new (and appropriate) number of samples and/or a new survey design.

There may be cases where the decision was made during the DQO process by the planning team to accept lower power. For instance, during the DQO process the calculated relative shift was found to be less

than 1. The planning team adjusts the LBGR, evaluates the impact on power and accepts the lower power. In this case, the DQA process would require the planning team to compare the prospective power analysis with the retrospective power analysis and determine whether the lower power is still justified and the DQOs satisfied.

5.9 Final Status Survey Reports

Documentation of the final status survey will transpire in two types of reports and will be consistent with Section 14.5 of NUREG-1727 and Section 8.6 of NUREG-1575. An FSS Survey Unit Release Record will be prepared to provide a complete record of the as-left radiological status of an individual survey unit, relative to the specified release criteria. Survey Unit Release Records will be made available to the NRC for review upon request. An FSS Final Report, which is a written report that is provided to the NRC for its review, will be prepared to provide a summary of the survey results and the overall conclusions which demonstrate that the Haddam Neck Plant site, or portions of the site, meets the radiological criteria for unrestricted use.

5.9.1 FSS Survey Unit Release Records

An FSS Survey Unit Release Record will be prepared upon completion of the final status survey for a specific survey unit. Sufficient data and information will be provided in the release record to enable an independent re-creation and evaluation at some future time. The format and content of the FSS Survey Unit Release Record is as follows:

- *Survey Unit Description*, including unit size, descriptive maps, plots or photographs, including reference coordinates and historic changes in description;

- *Classification Basis*, including significant historical site assessment and characterization data used to establish the final classification as well as a statement on the impact groundwater had on the final classification;

- *Data Quality Objectives* stating the primary objective of the survey, and a brief description of the DQO process;

- *Survey Design* describing the design process, including methods used to determine the number of samples or measurements required based on statistical design, number of biased or judgmental samples or measurements required, method of sample or measurement locating, and a table providing a synopsis of the survey design;

- *Survey Implementation* describing survey methods and instrumentation used, accessibility restrictions to sample or measurement location, number of actual samples or measurements taken, documentation activities, Quality Control samples or measurements, and scan data collected in tabular format;

- *Survey Results* including types of analyses performed, types of statistical tests performed, statement of pass or failure of the statistical test(s);

- *Quality Control* results to include discussion of split samples and/or QC replicate measurements;

- *Investigations and Results*;
- *Remediation* activities, both historic and resulting from the final status survey;
- *Changes from the Final Status Survey Plan* including field changes;
- *Data Quality Assessment*;
- *Anomalies* occurring during the survey or in the sample results;
- *Conclusion* as to whether or not the survey unit satisfied the specified release criteria, a discussion of ALARA evaluations performed, and whether or not sufficient power was achieved;
- *Attachments* and enclosures to include supporting maps, diagrams, and sample statistical data.

5.9.2 FSS Final Reports

The ultimate product of the Data Life Cycle is an FSS Final Report which will be, to the extent practical, a stand-alone document with minimal information incorporated by reference. To facilitate the data management process, as well as overall project management, FSS Final Reports will usually incorporate multiple FSS Survey Unit Release Records. To minimize the incorporation of redundant historical assessment and other FSS program information, and to facilitate potential partial site releases from the current license, FSS Final Reports will be prepared and submitted in a phased approach. The format and content of the FSS Final Report is as follows:

- Introduction, including a discussion on the phased approach for submittals;
- FSS Program Overview to include sub-sections on survey planning, survey design, survey implementation, survey data assessment, and Quality Assurance and Quality Control measures;
- Site Information to include sub-sections on site description, survey area/unit description (specific to current phase submittal), summary of historical radiological data, conditions at the time of survey, identification of potential contaminants, and radiological release criteria;
- Final Status Survey Protocol to include sub-sections on Data Quality Objectives, survey unit designation and classification, background determination, final status survey plans, survey design, instrumentation (detector efficiencies, detector sensitivities, instrument maintenance and control and instrument calibration), survey methodology, and quality control surveys;
- Survey Findings to include sub-sections on survey data conversion, survey data verification and validation, evaluation of number of sample/measurement locations, and comparison of findings with DCGLs
- Appendix A: Survey Unit Release Records (specific to each phased submittal);
- Appendix B: FSS Program and Implementing Procedures (initial phased submittal – subsequent submittals contain only revisions or additions to program and/or implementing procedures);

- Appendix C: FSS Technical Basis Documents (initial phased submittal – subsequent submittals contain only revisions or additions to FSS technical basis documents).

5.10 Quality Assurance and Quality Control Measures

Connecticut Yankee Atomic Power Company (CYAPCO) has developed and is implementing a comprehensive Quality Assurance Program to assure conformance with established regulatory requirements, set forth by the Nuclear Regulatory Commission (NRC), and accepted industry standards. The participants in the Connecticut Yankee Quality Assurance Program (CYQAP) assure that the design, procurement, construction, testing, operation, maintenance, repair, and modification of nuclear power plants are performed in a safe and effective manner.

The CYQAP complies with the requirements set forth in Appendix B, of 10 CFR Part 50, along with applicable sections of the Updated Final Safety Analysis Report (UFSAR) for the license application, and is responsive to Regulatory Guide 1.70, which describes the information presented in the Quality Assurance Section of the UFSAR for nuclear power plants. References to specific industry standards for quality assurance and quality control measures governing final status survey activities are reflected in supporting procedures, plans, and instructions.

These Quality Control (QC) and Quality Assurance (QA) measures are integrated into all decommissioning activities, including the development of the LTP and implementation of the final status survey. The CYQAP concepts, as defined in implementing procedures, adequately encompass the risk-significant decommissioning activities. All final status survey activities essential to data quality will be implemented and performed under approved procedures. Effective implementation of administrative controls will be verified through audit activities, with corrective actions being prescribed, implemented and verified in the event any deficiencies are identified. These measures apply to the related services provided by off-site vendors, in addition to on-site sub-contractors.

With regard to the final status survey effort, QA/QC activities will serve to ensure that surveys are performed by trained individuals using approved written procedures and properly calibrated instruments that are sensitive to the suspected contaminant. In addition, QC measures will be taken to obtain quantitative information to demonstrate that measurement results have the required precision and are sufficiently free of errors to accurately represent the site being investigated. QC checks will be performed as prescribed by the implementing procedures required by the CYQAP for both field measurements and laboratory analysis (both on-site and third party). For field measurements, replicate measurements will be made for randomly chosen survey units by a different technician at the same locations as the original measurements. Additionally, the CYAPCO Oversight Organization will be involved in assessing the performance of final status survey activities.

The concepts described in the CYQAP will be applied to the Final Status Survey activities. These activities include the following, as applicable:

Organization

The Director of Nuclear Safety/Regulatory Affairs is responsible for ensuring the implementation of site QA programs and processes. The Nuclear Safety Manager directs and administers independent audits, surveillances, and inspections for the final status survey. Both CYAPCO Independent Oversight and Nuclear Safety have the authority to stop unsatisfactory final status survey activities. An organizational chart of the Final Status Survey Group is provided as Figure 5-4.

Quality Assurance Program

The site characterization program used the same QA practices as employed by the operational radiation protection program. These practices included the use of approved procedures for the calibration, testing and use of both laboratory and portable equipment. Trained and qualified personnel collected data. Samples were controlled by administrative procedures to ensure that sample integrity was maintained. When offsite laboratories were utilized, they were required to perform daily instrumentation checks as well as split samples, blank samples, and cross check samples. Performances of these checks by offsite laboratories were verified periodically by QA auditors.

To support future site characterization and the FSS, quality assurance project plans as well, as Data Quality Objectives, will be developed. Documented procedures will be utilized for implementing quality activities at HNP. Additionally, the assignment of documented responsibilities for the conduct of activities affecting quality is defined. Through implementation of these controls, confidence is established that the performance of the FSS will be accomplished in a manner consistent with CYAPCO Policies. It also establishes the commitment that quality activities are performed by trained qualified personnel and that these activities are verified through audits, surveillances, and inspections.

Design Control

Design control requirements are established to assure that the applicable regulatory bases, codes, technical standards and quality standards are identified in the Final Status Survey. Design controls including independent verification, and design interface control have been implemented to determine the DCGLs, MDCs, area factors, and other DQO and FSS elements.

Procurement Document Control

The procurement of materials, equipment, and services for the FSS are performed in a controlled manner which assures compliance with applicable regulatory requirements, procedures, quality assurance standards and regulations. Service requests will be reviewed for technical adequacy and verification of supplier's quality assurance program will be performed as needed, to assure confidence with services provided. Performance of off-site audits will be used as deemed necessary by administrative controls.

Procedures, Instructions and Drawings

The performance of the FSS will require procedures for personnel training, survey implementation, data collection, chain of custody, instrument calibration and maintenance, verification and record storage. These procedures will be developed to ensure compliance with the License Termination Plan and will meet applicable quality requirements. These requirements

include that the procedures be developed utilizing the guidance of an approved procedure and will receive the appropriate review and approval.

Document Control

As stated above, procedures will be written to control the FSS performance. Additionally, procedures will be provided describing the requirements for the control and storage of survey and sample data developed by implementation of the FSS Plan. The results of the FSS will be maintained as records for a minimum of 3 years as required by 10CFR20.2103(a).

Control of Purchased Material, Equipment, and Services

Vendors may be used for the performance of the FSS and laboratory activities. Quality-related services, such as instrument calibration and laboratory analysis, are procured from qualified vendors whose internal QA program is subject to approval in accordance with the CYAPCO Quality Assurance Program. Additionally, audits and surveillance of these contractors will be performed to provide an adequate level of assurance that the quality activities are being effectively performed.

Inspection

Inspections and verification activities will be delineated in implementing procedures. These programs and procedures will be used to verify sampling and surveying protocols are appropriately utilized. Inspections will also be conducted on off-site laboratories performing sample analysis for the FSS.

Control of Measuring and Test Equipment

Approved procedures will be developed for the use, calibration, and testing of the equipment utilized for the FSS. These procedures will be developed to assure confidence in the data obtained. If additional equipment is procured for the FSS, associated maintenance, calibration, and testing procedures will also be developed. This includes both laboratory equipment and field use equipment. Instrument calibrations will be done periodically in accordance with approved administrative procedures.

Handling, Storage, and Shipping

Some of the material samples will be transported to off-site laboratories for analysis. The process for controlling this material will be sufficient to ensure that a chain of custody is maintained. Additionally, protocols must be established to ensure that there is no cross-contamination between samples and sample packaging. Appropriate controls will be defined in administrative procedures to ensure that sample integrity is maintained.

Nonconforming Materials, Parts, Components, or Services

During the performance of the FSS, non-conforming conditions may be identified with equipment or services. The associated data will be segregated until such time that they are accepted, rejected, or reworked in accordance with an appropriate procedure.

Corrective Action

The existing Corrective Action Program established under the CYQAP will be utilized for the FSS Program to identify conditions adverse to quality and to support the development of corrective actions.

Quality Assurance Records

As stated previously, the FSS records will be maintained in accordance with current administrative controls and will be retained for a minimum of 3 years.

Controls of Special Processes

Procedures will be developed to implement special processes in support of FSS implementation. Validated special processes will be implemented by trained, qualified individuals using approved procedures.

Quality Assurance Audits

Audits of FSS activities will be periodically performed to verify the implementation of quality activities.

Inspection, Test, and Operating Status

Project controls and schedules will be developed and implemented which identify the status of FSS activities. Measures will ensure identification mechanisms are in place to enable accurate determination of FSS status.

5.11 References

5-1 Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use."

5-2 Draft Regulatory Guide-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination," August 1998.

5-3 NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," dated December 1997.

5-4 Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," dated January 1999.

5-5 NUREG-1727, "NMSS Decommissioning Standard Review Plan," dated September 2000.

5-6 NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," December 1997.

5-7 Bechtel Calculation 24265-000-MOC-9000-0007-003, "DCGL Area Factors," R. K. Carr.

5-8 Technical Support Document, BCY-HP-105, Rev. 1, "Dose Evaluation for Buried Piping."

5-9 Technical Support Document, BCY-HP-114, Rev. 0, "Dose Comparison of Imbedded Pipe to Building Structures."

5-10 Information Notice 85-92, "Surveys of Wastes Before Disposal from Nuclear Reactor Facilities."

5-11 IE Circular 81-07, "Control of Radioactively Contaminated Material."

5-12 ISO 7503-1

6 COMPLIANCE WITH THE RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

6.1 Site Release Criteria

6.1.1 Radiological Criteria for Unrestricted Use

The site release criteria for the Haddam Neck Plant (HNP) site will correspond to the radiological criteria for unrestricted use given in 10 CFR 20.1402 (Reference 6-1):

- Dose Criterion: The residual radioactivity that is distinguishable from background radiation results in a Total Effective Dose Equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem/year, including that from groundwater sources; and

- ALARA Criterion: The residual radioactivity has been reduced to levels that are As Low As Reasonably Achievable (ALARA).

6.1.2 Conditions Satisfying the Release Criteria

Levels of residual radioactivity that correspond to the allowable radiation dose and ALARA levels described above are calculated by analysis of various scenarios and pathways (e.g., direct radiation, inhalation, ingestion) through which exposures could be reasonably expected to occur. LTP Section 2.3.3.4 discusses the radionuclides for which Derived Concentration Guideline Levels (DCGLs) and the future groundwater dose must be calculated. These DCGLs and the future groundwater dose calculation methodology form the basis for the following conditions which, when met, satisfy the site release criteria as prescribed in 10 CFR 20.1402:

- The average residual radioactivity in soils, standing above grade buildings and existing groundwater above background is less than or equal to the applicable combined DCGLs.

- In the case of buried concrete, embedded ~~buried~~ piping and the below-grade containment liner, the "future groundwater" dose will be determined using the "Basement Fill" Model. This approach will ensure that the dose from all pathways will be less than the release criteria of 10CFR20.1402. The details of this model are presented in Section 6.8.2

- Individual measurements, representing small areas of residual radioactivity, which exceed the DCGL, do not exceed the elevated measurement comparison $DCGL_{EMC}$. The ~~elevated measurement comparison DCGL~~ (use of the $DCGL_{EMC}$) is described in Section 5.4.7.4.

- Where one or more individual measurements exceed the DCGL, the average residual radioactivity passes the Sign or Wilcoxon Rank Sum (WRS) statistical test. (See Section 5.8 for a detailed discussion application of statistical tests).

- Remediation is performed where it is ALARA to reduce the levels of residual radioactivity to below those concentrations necessary to meet the DCGL or in the case of the Basement Fill

Model, below the future groundwater dose calculated by that models. (See Section 4 and Appendix B for detailed discussions of ALARA considerations).

The methods in MARSSIM (Reference 6-2) and the DCGLs may not be appropriate for complex non-structural components. For those non-structural components and systems to which MARSSIM does not apply (with the exception of those cases discussed in Section 5.4.7.5), site unconditional release limits apply (i.e., no detectable radioactive material). These surveys will be performed in accordance with health physics procedures and are consistent with the requirements of NRC Information Notice 85-92, "Surveys of Wastes before Disposal from Nuclear Reactor Facilities", and IE Circular 81-07, "Control of Radioactively Contaminated Material."

As MARSSIM does not define a protocol for performing volumetric contamination sampling, to be used to calculate the future groundwater dose, protocols for this sampling is defined in Section 5.7.1.5.

CYAPCO will not use concrete debris from the demolition of buildings to backfill basements ~~to~~that remain after release of the buildings from the license. As described in Chapter 1, -demolition debris will be shipped off-site to an appropriate disposal facility. The remaining basements will be backfilled with soil from off-site locations. This backfill soil has been demonstrated to be free from plant related radioactivity over background. Due to this change in the decommissioning strategy, the Concrete Debris Scenario is no longer applicable to building basements. In order to calculate future groundwater dose that was previously calculated using the Concrete Debris DCGLs for basements, the Basement Fill model described in Section 6.8.2 will be used. Should the decommissioning plans at CYAPCO change to include the use of concrete debris for basement fill; the Concrete Debris DCGLs developed and approved as part of the LPT approved in November 2002 will be used to demonstrate compliance.

6.2 Site Characteristics

The following is a description of the physical, geologic, and hydrogeologic characteristics of the area and the relationship of these characteristics to contaminant source areas and potential pathways.

Physical Characteristics

The industrial area of the HNP site is located on the east bank of the Connecticut River on a level, 600 ft wide terrace at an elevation of 21 ft mean sea level (msl). A parking lot occupies the area to the north of the industrial area. The area north of the parking lot is occupied by a pond. To the south, a 5,500 foot-long cooling water discharge canal leads to the river from the southern edge of the industrial area. It is separated from the Connecticut River by a 200 to 1,000 ft wide peninsula flood plain that ranges in elevation from about 5 to 15 ft msl. A steep wooded hill slope rises immediately east of the industrial area to elevations over 300 ft msl. The lowermost 30 to 40 ft of the hillside adjacent to the plant consists of nearly vertical rock cut.

Geologic and Hydrogeologic Characteristics

The geology and hydrogeology of the industrial area is documented in the "Groundwater Monitoring Report" Malcolm Pirnie (1999) (Reference 6-3). Drawings depicting geologic and hydrogeologic characteristics are given in Figures 6-1 and 6-2. A brief discussion of the site characteristics is provided below. Note: As discussed in Chapter 2, the following information was current as of August of 2002. This information has been and will continue to be updated in correspondence with the CT DEP concerning the Phase 2 Hydrogeologic Investigation Work Plan. As the NRC receives copies of all of this correspondence, the information in the LTP will not generally be updated.

The topography of this area originally consisted of a north-south trending promontory approximately 400 ft wide that connected the steep hillside north of this area to a floodplain terrace along the river's edge. The steep hill slope extended southward to the northeastern most third of the Containment Building. The southern part of the promontory consisted of large bedrock outcroppings in the area of the turbine building. Wetlands extended for 1,000 ft or more to the northwest and southeast of the promontory. During construction of the HNP, the steep hill slope to the north and the higher portions of the promontory were cut and the adjacent wetlands were filled. The discharge canal was excavated through the wetland, terrace, and floodplain to the southeast. The subsurface portions of the Containment Building, Primary Auxiliary Building (PAB), Turbine Building, Discharge Tunnel, and Spent Fuel Pool were also excavated down to or below the original bedrock surface.

On either side of the bedrock promontory and on the peninsula are seven layers of unconsolidated sediments: artificial fill, wetland silt and organic matter, gray silt and fine-grained sand (alluvium), gravelly sand, red fine-grained sand, brown sand, and glacial till or cobble gravel. The sediment thickness below the industrial area averages less than 20 ft but increases southeastward to over 100 ft beneath the peninsula.

Bedrock fractures are visible on the hill slope and potentially project into the industrial area. These fractures may be preferential pathways for groundwater migration within the bedrock. The bedrock itself consists of a suite of recrystallized volcanic rocks mapped regionally as the Monson Gneiss and Middletown Formation. These rocks are made of various silicate minerals (quartz, plagioclase, biotite, hornblende, pyroxene, etc.) with essentially no porosity other than fractures.

The shallow groundwater flow beneath the industrial area occurs within the unconsolidated sediments and bedrock. The depth to the water table averages about 10 ft below ground surface (bgs) in this area. Groundwater generally flows southwest and downward near the hill slope, and upwards near the discharge canal and the Connecticut River. Locally, the Containment Building and mat drain sump are important hydrogeologic features. The groundwater flow pattern around the Containment Building was distorted with a component of flow toward the drainage system under the Containment Building. The mat drain sump, located on the southern side of the Containment Building, when operated, removed groundwater and depressed the water table around it. The pumps were shut off for several months but have been restarted. The cooling water discharge tunnels divert the shallow groundwater flowing around the southwestern side of the Containment Building farther to the south. Southwest of tunnels, the shallow groundwater appears to flow southwesterly and directly toward the river.

Contaminant Characteristics

Soil within the industrial area contains residual radioactivity from licensed operations by unplanned liquid releases or long-term accumulation of material in the soil via effluent releases. The impacted soil includes that in current open areas as well as that which will be exposed in the future following demolition of overlying buildings and structures. The areas wherein soil could potentially contain residual radioactivity are identified and described in Chapter 2. Based on the documented release mechanisms and the results of site characterization surveys, the residual radioactivity is generally confined to the surface soil layer, although some subsurface residual radioactivity exists. The surface soils in the industrial area are composed of a silty sand that was imported as artificial fill. Site survey results indicate that there may be localized areas where the soil contamination is deeper, but still restricted to the unsaturated zone.

Site surveys have identified radionuclides that may be present in measurable quantities in site soils and that are likely associated with licensed plant operations. Table 2-12 summarizes these radionuclides and their half-lives.

6.3 Dose Modeling Approach

6.3.1 Overview

To calculate DCGLs, dose models were developed, which translate levels of residual radioactivity into potential radiation doses to the public. Dose models, appropriate to the HNP site, are based on the guidance found in DG-4006 (Reference 6-4), NUREG-1549 (Reference 6-5), and NUREG/CR-5512, Volume 1 (Reference 6-6). A conceptual model was based on the site conditions expected at the time of unrestricted release. Conditions at the HNP site (e.g., pre-existing residual radioactivity in groundwater) required site-specific dose modeling be performed. The approach taken to dose modeling for the HNP site is consistent with the information provided in Chapter 5 and Appendix C of NUREG-1727 (Reference 6-7) for-site specific modeling, including the information regarding source term abstraction and scenarios, pathways, and critical groups.

In addition to calculating DCGLs, a "Basement Fill Model" will be used to determine the future groundwater dose from building basements and other subsurface materials on future uses of the site. This method uses actual characterization data (to determine the radionuclide inventory) and the calculated release rate of the radionuclides from the material to calculate the equilibrium maximum groundwater concentration that will result between back-fill soil and groundwater in the building basements. The future groundwater dose is calculated from the groundwater concentration and the groundwater DCGLs for these sources. This model is explained further in Section 6.8.2.

The dose model is defined by the three factors: 1) the scenario, 2) the critical group and 3) the exposure pathways. The scenarios described in NUREG/CR-5512, Volume 1, address the major exposure pathways of direct exposure to penetrating radiation and inhalation and ingestion of radioactive materials. The scenarios also identify the critical group. The critical group is the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity within the assumptions of the particular scenario. The scenarios and their modeling are specifically designed to be reasonably conservative by generally overestimating rather than underestimating potential dose.

The approach outlined above was used to develop dose models to calculate DCGLs for the following media:

- Soil
- Groundwater, and
- Concrete
  -Buildings Standing
  ~~-Buildings Demolished~~
  - ~~Foundations/Basements~~

It should be noted that the scenarios described in NUREG/CR-5512, Volume 1, were developed to estimate potential radiation doses from radioactive material in standing buildings and soil. These scenarios were adapted to estimate potential radiation doses from groundwater.~~, and concrete from demolished buildings.~~

6.3.2 Resident Farmer Scenario

Scenario Definition:

The resident farmer scenario, described as the "Residential Scenario" in NUREG/CR-5512, Volume 1, was selected to estimate human radiation exposures resulting from residual radioactivity in the soil~~,~~ and groundwater ~~and concrete from demolished buildings and building foundations/basements and~~ to determine corresponding DCGLs.

Critical Group:

Given regional demographic and economic data (References 6-8 and 6-9) the average member of the critical group was determined to be the resident farmer who lives on the plant site following decommissioning, grows all or a portion of his/her diet on site, and uses the water from a groundwater source on the site for drinking water and irrigation. The dose from residual radioactivity in the ~~soil,~~soil, and -groundwater ~~and concrete from demolished buildings and foundations/basements~~ is evaluated for the average member of the critical group as required by 10 CFR Part 20, Subpart E, and described in NUREG-1727, Appendix C and NUREG-1549.

It is unlikely that any other set of plausible human activities could occur onsite that would result in a dose exceeding that calculated for the hypothetical resident farmer. It is more likely that the behavior of future occupants would result in a lower dose. For example it is more likely that the HNP site (currently zoned "industrial") will be reused for a fossil-fired plant, making use of the current infrastructure, or for land conservation. The hypothetical dose from the soil to individual in these settings would be less than for a resident farmer, since the industrial worker would not ingest food derived from onsite. Therefore, the use of the resident farmer as the average member of the critical group is both conservative and bounding for the calculation of soil DCGLs.

Exposure Pathways:

The potential exposure pathways that apply to the resident farmer are listed below and depicted in Figure 6-3. These exposure pathways are based upon those in NUREG/CR-5512, Volume 1:

- Direct exposure to external radiation from the residual radioactivity;
- Internal dose from inhalation of airborne radionuclides; and
- Internal dose from ingestion of
  - Plant foods grown in media containing residual radioactivity and irrigated with water containing residual radioactivity,
  - Meat and milk from livestock fed with fodder grown in soil containing residual radioactivity and water containing residual radioactivity,
  - Drinking water (containing residual radioactivity) from a well,
  - Fish from a pond containing residual radioactivity, and
  - Media containing residual radioactivity.

6.3.3 Building Occupancy Scenario

Scenario Definition:

The building occupancy scenario, based upon NUREG/CR-5512, Volume 1, was selected to estimate human radiation exposure resulting from residual radioactivity in concrete from standing buildings and building foundations/basements that could reasonable be occupied and to determine corresponding DCGLs. CYAPCO will not leave any basements in place that can be reasonably occupied. These DCGLs are also used to bound residual contamination levels on metal surfaces such as the containment liner and embedded piping to be subsequently used in the calculation of the "future groundwater" dose due to contamination on these metal surfaces.

Critical Group:

Given the fact that the buildings associated with the HNP site are commercial, the average member of the critical group is an adult engaging in light industrial work within the buildings following decommissioning of the site. He/she occupies a commercial facility in a normal manner without deliberately disturbing sources of residual radioactivity. The dose from residual radioactivity in the concrete from the standing building is evaluated for the average member of the critical group as required by 10 CFR Part 20, Subpart E, and described in NUREG -1727, Appendix C.

Exposure Pathways:

The potential exposure pathways, described in NUREG/CR-5512, Volume 1, are depicted on Figure 6-4 and listed below:

- Direct exposure to external radiation from
  - Source
  - Material deposited on the floor
  - Submersion in airborne dust
- Internal dose from inhalation of airborne radionuclides
- Internal dose from inadvertent ingestion of radionuclides from the source

6.4 RESidual RADioactivity (RESRAD) and RESRAD-BUILD Codes

The RESRAD family of computer codes is pathway analysis models developed at Argonne National Laboratory (ANL). This family of computer codes includes RESRAD, used to analyze pathways associated with soil, and RESRAD-BUILD, used to analyze pathways associated with buildings.

The RESRAD computer code (Version 5.91) was used in this analysis to consider three major exposure pathways to a resident farmer from residual radioactivity in soil and groundwater:

- Direct exposure to external radiation from media containing residual radioactivity;
- Internal exposure from inhalation of airborne radionuclides; and
- Internal exposure from ingestion of radionuclides.

A newer version of the code released by ANL is RESRAD Version 6.1. This version of the code includes probabilistic modules to examine the sensitivity of input parameters on the resulting dose. A sensitivity

analysis has been performed using the probabilistic modules in RESRAD 6.1. Information obtained from that analysis (identification of sensitive parameters and their correlation to dose, either positive or negative) is then used to select conservative values for the sensitive input parameters for the deterministic runs using RESRAD Version 5.91.

The RESRAD-BUILD computer code (Version 2.37) is used in this analysis to consider five exposure pathways to occupants of a building from residual radioactivity for above-grade building surfaces:

- External exposure directly from the sources;
- External exposure to material deposited on the floor;
- External exposure due to air submersion;
- Inhalation of airborne radioactive particulates; and
- Inadvertent ingestion of radioactive material directly from the sources.

ANL has released a newer version of the code, RESRAD-BUILD Version 3.1. This version of the code includes probabilistic modules to examine the sensitivity of input parameters on the resulting dose. A sensitivity analysis has been performed using the probabilistic modules in RESRAD-BUILD Version 3.1. Information obtained from that analysis (identification of sensitive parameters and their correlation to dose, either positive or negative) was then used to select conservative values for the input parameters for the deterministic runs using RESRAD-BUILD Version 2.37.

For subsurface structures, an inventory-based method for assessing total radioactivity in the subsurface environment is provided. This total inventory is converted into a future groundwater dose using the "Basement Fill Model" and then converted to a future groundwater dose using the groundwater DCGLs as described above. Therefore, the process of parameter selection and sensitivity analysis does not apply to this model.

Information on the use of the codes and their applications are outlined in NUREG/CRs-6676, -6692, -6697 (References 6-10, 6-11, and 6-12), the "Users Manual for RESRAD, Version 6.0" (Reference 6-13), the "Manual for implementing Residual Radioactive Material Guidelines Using RESRAD, Version 5.0" (Reference 6-14) and for RESRAD-BUILD "A Computer Model for Analyzing the Radiological Doses Resulting from the Remediation and Occupancy of Buildings Contaminated with Radioactive Material" (Reference 6-15).

6.5 Parameter Selection Process

The conceptual model underlying the dose model was developed based on site characteristics expected at the time of release of the site. The conceptual model is quantified by a set of parameters. The parameter selection process is outlined schematically in Figure 6-5. The process was developed in accordance with the guidelines presented in NUREG/CR-6755 (Reference 6-16), -6676, -6692 and -6697 and ensures that conservative values are selected. Components of the selection process are discussed in the following sub-sections.

6.5.1 Classification

The parameters were classified as behavioral, metabolic or physical consistent with NUREG/CR-6697, Attachment A. Behavioral parameters depend on the behavior of the receptor and the scenario definition. Metabolic parameters represent the metabolic characteristics of the receptor and are independent of the scenario definition. Physical parameters are the parameters that would not change if a different group of receptors were considered.

6.5.2 Prioritization

The parameters were prioritized in order of importance consistent with NUREG/CR-6697, Attachment B. Prioritization was based on 1) the relevance of the parameter in dose calculations, 2) the variability of the dose as a result of changes in the parameter value, 3) the parameter type and 4) the availability of parameter-specific data. Priority 1 parameters are considered to be high priority; priority 2 parameters are considered to be medium priority; and priority 3 parameters are considered to be low priority.

6.5.3 Treatment

The parameters were treated as "deterministic" or "stochastic" depending on parameter type, priority, and availability of site-specific data and the relevance of the parameter in dose calculations. "Deterministic" modules of the code use single values for input parameters and generate a single value for dose. "Probabilistic" versions of the code use probability distributions for input parameters and generate a range of doses. "Stochastic" parameters are parameters that are defined by a probabilistic distribution.

The behavioral and metabolic parameters were treated as deterministic. The physical parameters for which site-specific data were available were also treated as deterministic. The remaining physical parameters for which no site-specific data were available to quantify were classified as either priority 1, 2, or 3. Priority 1 and 2 parameters were treated as stochastic. The priority 3 physical parameters were treated as deterministic.

6.5.4 Sensitivity Analyses

The purpose of the sensitivity analysis was to determine which of the stochastic parameters have the greatest influence on the resultant dose and associated DCGLs. The analysis was performed using the probabilistic modules of RESRAD, Version 6.1, and RESRAD-BUILD, Version 3.1.

The stochastic parameters were generally assigned distribution types and corresponding distribution statistical parameters from NUREG/CR-6697, Attachment C. Sensitivity analyses were performed on the stochastic parameters using the assigned distributions. To perform the sensitivity analysis the following information was required:

> Sample Specifications: The analyses were run using 300 observations and 1 repetition. The Latin Hypercube Sampling (LHS) technique was used to sample the probability distributions for each of the stochastic input parameters. The correlated or uncorrelated grouping option was used to preserve the prescribed correlations
>
> Input Rank Correlations: Correlation coefficients were assigned between correlated parameters.

Output Specifications: All of the output options were specified.

Sensitivity analyses were performed for each of the radionuclides. The Partial Rank Correlation Coefficient (PRCC) for the peak of the mean dose was used as a measure of the sensitivity of each parameter to the peak of the mean dose.

For the resident farmer scenario, a parameter was identified as sensitive if the absolute value of its PRCC (|PRCC|) was greater than or equal to 0.25 and non-sensitive if the |PRCC| value was less than 0.25. For the building occupancy scenario, a parameter was identified as sensitive if the |PRCC| value was greater than or equal to 0.10 and non-sensitive if the |PRCC| value was less than 0.10. These thresholds were selected based on the guidance included in NUREG/CR-6676.

6.5.5 Parameter Value Assignment

The behavioral and metabolic parameters were assigned values from NUREG/CR-5512, Volume 3, NUREG/CR-6697, or the RESRAD default library.

Physical parameters were assigned values as follows:

- Physical parameters for which site-specific data were available were assigned site-specific values.

- Priority 1 and 2 physical parameters shown to be sensitive ($|PRCC| \geq 0.25$) were assigned conservative values. Depending on whether the parameter was positively or negatively correlated with dose, the 75% or 25% quantile value of the distribution was used, respectively. The mean value of the distribution was also calculated for those parameters positively correlated with dose. If the mean value was greater than the 75% quantile value (positively skewed distribution), the parameter was assigned the mean value.

- Priority 1 and 2 physical parameters shown to be non-sensitive ($|PRCC| < 0.25$) were assigned median values from NUREG/CR-6697, Attachment C.

- Priority 1 and 2 physical parameters shown to be non-sensitive ($|PRCC| < 0.25$) but correlated with a physical parameter shown to be sensitive (see Section 6.5.4) were assigned values based on the conservative value assigned to the sensitive parameter.

- Priority 3 physical parameters were assigned values from NUREG/CR-5512, Volume 3, or from the RESRAD default library.

6.6 DCGLs for Soil

Residual radioactive material is considered to exist in soil underlying portions of the HNP site. The residual radioactivity is considered to be from licensed operations by unplanned liquid releases or long-term accumulation of material in the soil via effluent releases. The affected areas are generally confined to the industrial area of the site and include areas that are currently open and areas that may be open following the demolition of buildings and structures.

6.6.1 Dose Model

The DCGLs for soil were calculated using the resident farmer scenario. The residual radioactive materials were assumed to be contained in a soil layer (surface and subsurface) on the property that can be used for residential and light farming activities. The average member of the critical group is the resident farmer that lives on the plant site, grows all or a portion of his/her diet onsite, drinks water from a groundwater source onsite.

The potential pathways used to estimate human radiation exposure resulting from residual radioactivity in the soil include the following:

- Direct exposure to external radiation from soil containing residual radioactivity;
- Internal dose from inhalation of airborne radionuclides; and
- Internal dose from ingestion of:
    - Plant foods grown in the soil material containing residual radioactivity;
    - Meat and milk from livestock fed with fodder grown in soil containing residual radioactivity and water containing residual radioactivity;
    - Drinking water containing residual radioactivity from a well,
    - Fish from a pond containing residual radioactivity, and
    - Soil containing residual radioactivity.

6.6.2 Conceptual Model

The conceptual model underlying the dose model includes a contaminated zone, an unsaturated zone, and a saturated zone. The contaminated zone is exposed at the ground surface (no cover). Residual radioactivity is confined to the soils in the contaminated zone. The thickness of the contaminated zone is conservatively set at 3 meters. For the purpose of calculating soil DCGLs, the groundwater is assumed to be initially uncontaminated.

The parameters used to quantify the conceptual model are listed in Appendix D, Table D-1. The values/ distributions assigned to each of the parameters and the basis for assigning such values/distributions are shown on the table.

6.6.3 Sensitivity Analysis Results

Parameter distributions assigned in the probabilistic RESRAD, Version 6.1, model is presented in Appendix D, Table D-1. An initial radionuclide concentration of 1 pCi/g was used for the soil comprising the contaminated zone.

The stochastic parameters identified as sensitive ($|PRCC| \geq 0.25$) to the peak of the mean dose for each of the radionuclides are presented in Appendix E, Table E-1. For each radionuclide, the sensitive parameters are listed in order of decreasing sensitivity. Included in the table are the conservative values assigned to each of the sensitive parameters.

6.6.4 DCGL Determination

Parameter values assigned in the deterministic RESRAD Version 5.91 model are presented in Tables E-1 (conservative values assigned to parameters shown to be sensitive) and Appendix F, Table F-1. The soil DCGLs were determined for a radiation dose limit of 25 mrem/yr.

The soil DCGLs calculated for each of the radionuclides are presented in Appendix G, Table G-1. The time to the peak of the mean dose is also included on the table together with the percent contribution to dose from the exposure pathways (water independent and water dependent).

The soil DCGLs are summarized in Table 6-1:

Table 6-1
Base Case DCGLs for Soil

Radionuclide	Soil DCGL (pCi/g)
H-3	4.12E+02
C-14	5.66E+00
Mn-54	1.74E+01
Fe-55	2.74E+04
Co-60	3.81E+00
Ni-63	7.23E+02
Sr-90	1.55E+00
Nb-94	7.12E+00
Tc-99	1.26E+01
Ag-108m	7.14E+00
Cs-134	4.67E+00
Cs-137	7.91E+00
Eu-152	1.01E+01
Eu-154	9.29E+00
Eu-155	3.92E+02
Pu-238	2.96E+01
Pu-239	2.67E+01
Pu-241	8.70E+02
Am-241	2.58E+01
Cm-243	2.90E+01

6.7 DCGLs for Groundwater

Residual radioactivity presently exists in groundwater underlying portions of the HNP site. The affected areas are generally confined to the industrial area of the site, as investigated by Malcolm Pirnie (Reference 6-3).

6.7.1 Dose Model

The resident farmer scenario was selected to estimate human radiation exposures resulting from residual radioactivity in the groundwater and to determine corresponding DCGLs. The residual radioactive materials are assumed to be contained in the groundwater on the property, which is withdrawn via a groundwater source (well) and used for irrigation and drinking water. The average member of the critical group is the resident farmer who lives on the plant site, grows all or a portion of his/her diet onsite, and drinks water from the groundwater source onsite. 0

The potential pathways used to estimate human radiation exposure resulting from residual radioactivity in the groundwater include the following ingestion pathways:

- Plant foods irrigated with water containing residual radioactivity;
- Meat and milk from livestock fed with water containing residual radioactivity; and
- Drinking water containing residual radioactivity from a well.

Groundwater flow directions determined by Malcolm Pirnie (1999) are such that the existing plumes migrate toward the Connecticut River. The flow rate of groundwater, potentially containing residual radioactivity, relative to the flow rate of the Connecticut River is likely very small. Therefore, the aquatic foods ingestion pathway is not considered applicable in this calculation.

6.7.2 Conceptual Model

The conceptual model underlying the dose model was developed based on the site characteristics expected at the time of release of the site. The model assumes that the groundwater contains residual radioactivity at the time of site release and that all sources that contributed to this contamination have since been removed. It is further assumed that the residential farmer installs a well that supplies water for drinking, crop irrigation, and livestock, and that this well is drilled and completed within a portion of the groundwater system that contains residual radioactivity.

The parameters used to quantify the model are presented in Appendix D, Table D-2. The values / distributions assigned to each of the parameters, the basis for assigning such values / distributions and the relevance of the parameters to the dose calculations are included in the table.

The RESRAD code is typically used to calculate radiation doses (and DCGLs) for a source above the water table. To develop a dose model consistent with the conceptual model, it was necessary to establish the parameters below as follows:

- Time since placement of material = 1 year
- Time for calculations = 1 year
- Model for water transport parameters = Mass Balance (MB) model
- Distribution coefficient in the saturated zone = 0 cm^3/g

By doing so, the groundwater (well water) concentrations calculated by RESRAD were found to be greater than or equal to the groundwater concentrations in equilibrium with the contaminated zone, under saturated conditions, and the time to the peak of the mean dose was 0 years.

The equilibrium groundwater concentration associated with the contaminated zone was calculated using the principals of linear sorption theory described in Appendix H of the "Users Manual for RESRAD Version 6.0," from which the following equation was derived:

$$C = \frac{1000\, S_o\, \rho_b}{[1+(K_d\, \rho_b / n)]\; n} \qquad \text{(Equation 6-1)}$$

where,

C = Equilibrium groundwater concentration (pCi/l)
S_o = Initial principal radionuclide concentration in contaminated zone (pCi/g)

ρ_b = Bulk density of contaminated zone (g/cm^3)
K_d = Distribution Coefficient of contaminated zone (cm^3/g)
n = Total porosity of contaminated zone (Fraction~~(%~~)

6.7.3 Sensitivity Analysis Results

Parameter distributions assigned in the probabilistic RESRAD Version 6.1 model are presented in Appendix D, Table D-2. An initial radionuclide concentration of 1 pCi/g was used for the soil comprising the contaminated zone.

The stochastic parameters identified as sensitive ($|PRCC| \geq 0.25$) to the peak of the mean dose for each of the radionuclides are presented in Appendix E, Table E-2. For each radionuclide, the sensitive parameters are listed in order of decreasing sensitivity. Included in Table E-2 are the conservative values assigned to each of the sensitive parameters.

6.7.4 DCGL Determination

Parameter values assigned in the deterministic RESRAD Version 5.91 model are presented in Table E-2, (conservative values assigned to parameters shown to be sensitive) and Appendix F, Table F-2. The groundwater DCGLs were determined for a radiation dose limit of 25 mrem/yr. The groundwater DCGLs were calculated by scaling the groundwater (well water) concentrations calculated by RESRAD against the peak dose to determine the concentration that would give a radiation dose of 25 mrem/yr, as shown in the following equation:

$$DCGL_{GW} = \frac{Conc_{WW}}{Dose_{PEAK}} * 25 \qquad \text{(Equation 6-2)}$$

where,

$DCGL_{GW}$ = DCGL for groundwater (pCi/l)
$Conc_{WW}$ = Groundwater (well water) (pCi/l)
$Dose_{PEAK}$ = Peak Dose (mrem/yr)
25 = Radiation dose limit of 25 mrem/yr

The above derivation of the groundwater DCGLs is applicable to the radionuclides that do not have progeny, as the peak dose occurs at the time of release of the site. For the radionuclides that have progeny, the above derivation is not applicable, as the peak dose may occur subsequent to release of the site, due to the in-growth of progeny, and therefore contributions to dose, with time. For these radionuclides (Eu-152, Pu-238, Pu-239, Pu-241, Am-241 and Cm-243), the groundwater DCGLs were calculated by modeling the decay of a unit source over 1000 yrs in RESRAD, Version 5.91.

In a "new file" in RESRAD, Version 5.91, the parameters from the RESRAD default library were established, together with the 1pCi/g (the units 1pCi/g or 1pCi/l are arbitrary since the only interest is the decay of a unit source) and 1000-year calculation time. A couple of other parameters were established (the precipitation was set to zero and the water-dependent pathways were toggled off) to ensure the model operated as a "closed" system. The resulting concentrations of the parent radionuclides and progeny, as a function of decay time, are presented in Appendix G, Table G-2-1.

The concentrations of the parent radionuclides and progeny, calculated by RESRAD using an initial concentration of 1pCi/g, were converted into dose, outside of RESRAD, by multiplying by effective Dose Conversion Factors (DCF_{eff}'s). The DCF_{eff}'s were calculated outside of RESRAD using the peak dose and groundwater (well water) concentrations for the parent radionuclides. For the progeny, the dose and groundwater concentrations were obtained by re-running the progeny as parent radionuclides in the groundwater model in RESRAD, Version 5.91. The DCF_{eff}'s were calculated as shown in the Equation 6-4 below. Calculation of the DCF_{eff}'s for each of the radionuclides is shown in Table G-2-1, Footnote "C".

$$DCF_{eff} = \frac{Dose_{PEAK}}{Conc_{WW}} \qquad \text{(Equation 6-34)}$$

where,

DCF_{eff}	= Effective Dose Conversion Factor (mrem/yr/pCi/l)
$Dose_{PEAK}$	= Peak Dose at time = 0 yrs (mrem/yr)
$Conc_{WW}$	= Groundwater (well water) (pCi/l)

Following the calculation of dose (using the DCF_{eff}'s), the peak dose was calculated outside of RESRAD by summing the individual doses for the parent radionuclides and progeny ("total" dose) and identifying the highest dose, as shown (in bold) on Table G-2-1. The groundwater DCGLs were calculated outside of RESRAD by scaling the peak dose (based on a radionuclide concentration of 1pCi/l) to obtain a concentration based on a radiation dose limit of 25 mrem/yr. These groundwater DCGL's are presented in Appendix G, Table G-2-1.

The groundwater DCGLs for each of the radionuclides are presented in Appendix G, Table G-2-2, with the percent contribution to dose from the exposure pathways (water dependent). Included in Table G-2-2 are the equilibrium groundwater concentrations associated with the contaminated zone and the groundwater (well water) concentrations for a known concentration of radioactive material in the contaminated zone for each of the radionuclides.

The groundwater DCGLs are summarized in Table 6-2.

Table 6-2
Base Case DCGLs for Groundwater

Radionuclide	Groundwater DCGL (pCi/l)
H-3	6.52E+05
C-14	9.01E+03
Mn-54	2.42E+04
Fe-55	6.54E+04
Co-60	1.14E+03
Ni-63	3.15E+04
Sr-90	2.51E+02
Nb-94	6.75E+03
Tc-99	2.64E+04
Ag-108m	4.24E+03
Cs-134	3.42E+02
Cs-137	4.31E+02
Eu-152	7.33E+03
Eu-154	5.05E+03
Eu-155	3.25E+04
Pu-238	1.51E+01
Pu-239	1.36E+01
Pu-241	4.60E+02
Am-241	1.32E+01
Cm-243	1.94E+01

6.8 DCGLs for Concrete

A few of the building basements and footings, at HNP will remain in place and be surveyed or assessed for residual radioactivity. Presently CYAPCO's plan is to backfill these partial structures with clean material, once the final status survey or assessment of the structure has been completed~~y are released~~ The site dose contribution from these basements will be calculated using the Basement Fill Model as discussed in Section 6.8.2. ~~To ensure that the building~~For buildings to remain after release from the license, the Building Occupancy ~~surface~~ DCGLs will be used. ~~are conservative, two cases were evaluated. Figure 6-6 illustrates the process of determining building surface DCGLs using two scenarios. In one case, the building occupancy scenario was evaluated; and for the other case, the resident farmer (below-grade concrete material) scenario was evaluated. The two scenarios were evaluated separately and the more restrictive DCGL for each radionuclide will be adopted at the time of final status survey, as discussed in Section 5.4.7.1. This ensures that the potential dose to the average member of critical group will be conservatively estimated whether the foundation/basement remains, or the building is demolished and the material will be disposed of at a licensed facility.~~

6.8.1 DCGLs for Concrete: Buildings Standing

6.8.1.1 Dose Model

The DCGLs for building surfaces were calculated using the building occupancy scenario. These DCGLs will be applied during the final status survey of:

- any buildings to remain standing following release from the license of that portion of the site that contains the building, and

- Any structure for which the surface contamination is not accounted for in the volumetric sampling of the structure. Examples of this are the steel liner attached to the inside of the containment building and embedded piping to remain after the release of the building from the license. The likelihood of these structures being occupied is very small. ~~and~~ Conducting the surveys to the Building Occupancy DCGLs allows for the confident use of the contamination levels corresponding to those DCGLs as bounding values in the calculation of "future groundwater" dose using the Basement Fill Model.

The residual radioactivity was assumed to be uniformly distributed over all surfaces of a room, including the floor, ceiling, and four walls. The average member of the critical group is an adult working in the building, engaged in light industrial activities.

The potential pathways used to estimate human radiation exposure resulting from residual radioactivity on the building surfaces include the following:

- External exposure directly from the source;
- External exposure to material deposited on the floor;
- External exposure due to air submersion;
- Inhalation of airborne radioactive particulates and tritium; and
- Inadvertent ingestion of radioactive material directly from the sources.

6.8.1.2 Conceptual Model

The conceptual model underlying the dose model consisted of a room of fixed area (10 m by 10 m by 2.5 m high), uniform concentrations of residual radioactivity on all room surfaces, and the receptor located at the center of the room at a height of 1 m. Two cases were considered for the source type: area (surface) sources and volume sources. Area sources consisted of a thin-layer of residual radioactivity on the surface, consistent with NUREG/CR-5512, Volume 1. Volumetric sources consisted of 0.305 m (12 inches) of concrete to account for the possibility of volumetrically contaminated sources, either by migration of radioactive material into the depth of the source or by neutron activation.

The parameters used to quantify the conceptual model are listed in Appendix D, Table D-3. The values / distributions assigned to each of the parameters and the basis for assigning such values / distributions are also shown on the table.

6.8.1.3 Sensitivity Analysis Results

Parameter distributions assigned in the stochastic model are presented in Appendix D, Table D-3. The stochastic parameters identified as sensitive ($|PRCC| \geq 0.10$) to the peak of the mean dose for each of the radionuclides are presented in Appendix E, Table E-3. For each radionuclide, the sensitive parameters are listed in order of decreasing sensitivity. Included in Table E-3 are the conservative values assigned to each of the sensitive parameters.

6.8.1.4 DCGL Determination

Using the results of the sensitivity analysis, which identified which input parameters were sensitive to dose, conservative input values were selected (see Table E-3). Parameter values assigned in the deterministic model for area sources are presented in Appendix F, Table F-3.

For volume sources, 0.305 m (12 inches) of concrete was assumed for each of the six sources, which modeled an infinite thickness for the radionuclides of interest. In RESRAD-BUILD, the airborne concentration is determined by the parameter erosion rate, instead of the parameters removable fraction and time for source removal. A conservative value (75% quantile) for the erosion rate of 2.8E-7 cm/day based on NUREG/CR-6697, Attachment C, was used for those radionuclides which exhibited sensitivity for that parameter.

Building occupancy DCGLs were calculated using RESRAD-BUILD 2.37. The DCGLs are presented in Table G-3, Appendix G. DCGLs for area sources have units of disintegrations per minute per 100 cm^2 (dpm/100 cm^2). DCGLs for volume sources have units of pCi/g. The building occupancy DCGLs for each of the radionuclides are summarized in Table 6-3:

ʃ

Table 6-3
Base Case Building Surface DCGLs
(Building Occupancy Scenario)

Radionuclide	DCGL for Surface Sources (dpm/100 cm^2)	DCGL for Volumetric Sources (pCi/g)
H-3	3.15E+08	1.47E+03
C-14	1.03E+07	1.18E+08
Mn-54	3.21E+04	9.06E+00
Fe-55	3.49E+07	9.54E+07
Co-60	1.11E+04	2.90E+00
Ni-63	3.60E+07	4.11E+07
Sr-90	1.27E+05	2.38E+03
Nb-94	1.71E+04	4.83E+00
Tc-99	1.45E+07	3.09E+07
Ag-108m	1.65E+04	4.84E+00
Cs-134	1.65E+04	4.93E+00
Cs-137	4.30E+04	1.37E+01
Eu-152	2.34E+04	6.70E+00
Eu-154	2.19E+04	6.11E+00
Eu-155	4.37E+05	3.23E+02
Pu-238	4.87E+03	6.61E+02
Pu-239	4.44E+03	6.02E+02
Pu-241	2.29E+05	3.12E+04
Am-241	4.27E+03	4.16E+02
Cm-243	6.07E+03	7.53E+01

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Table 6-3
Base Case Building Surface DCGLs
(Building Occupancy Scenario)

Radionuclide	DCGL for Surface Sources (dpm/100 cm²)	DCGL for Volumetric Sources (pCi/g)
H-3	3.15E+08	1.47E+03
C-14	1.03E+07	1.18E+08
Mn-54	3.21E+04	9.06E+00
Fe-55	3.49E+07	9.54E+07
Co-60	1.11E+04	2.90E+00
Ni-63	3.60E+07	4.11E+07
Sr-90	1.27E+05	2.38E+03
Nb-94	1.71E+04	4.83E+00
Tc-99	1.45E+07	3.09E+07
Ag-108m	1.65E+04	4.84E+00
Cs-134	1.65E+04	4.93E+00
Cs-137	4.30E+04	1.37E+01
Eu-152	2.34E+04	6.70E+00
Eu-154	2.19E+04	6.11E+00
Eu-155	4.37E+05	3.23E+02
Pu-238	4.87E+03	6.61E+02
Pu-239	4.44E+03	6.02E+02
Pu-241	2.29E+05	3.12E+04
Am-241	4.27E+03	4.16E+02
Cm-243	6.07E+03	7.53E+01

6.8.2 ~~DCGLs for Concrete: Buildings Demolished (Concrete Debris)~~

After completion of final status survey activities of the remaining portions of structures, some concrete debris may remain. However, the current plans are to use clean backfill material and not concrete debris to backfill basements and restore the site to grade elevation. Previously it was assumed that concrete debris generated from the demolition of the buildings and any additional decontaminated concrete structures would be placed in the basements of the buildings, to a depth of approximately 3 feet below the ground surface. And approximately 3 feet of clean backfill soil would be placed over the concrete debris. This concrete debris was considered to contain residual radioactive material. All building basements were assumed to operate as open-systems, allowing in-flow and out-flow of groundwater.

Although current decommissioning plans do not call for the placement of concrete debris in facility basements, the methodology outlined below will be conservatively applied for remaining basement structures.

6.8.2.1 Dose Model

The resident farmer scenario is selected to estimate human radiation exposures resulting from residual radioactivity in concrete debris and to determine corresponding DCGLs. The residual radioactive materials are assumed to be contained in a subsurface layer of concrete debris on property that can be

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used for residential and light farming activities. The average member of the critical group is the resident farmer who lives on the plant site, grows all or a portion of his/her diet onsite, and drinks water from a groundwater source onsite.

The potential pathways used to estimate human radiation exposure resulting from residual radioactivity in the concrete debris include the following:

- Direct exposure to external radiation from the concrete debris containing residual radioactivity;
- Internal dose from inhalation of airborne radionuclides; and
- Internal dose from ingestion of
  - Plant foods grown in the soil cover and irrigated with water containing residual radioactivity;
  - Meat and milk from livestock fed with fodder and water containing residual radioactivity;
  - Drinking water, containing residual radioactivity, from a well,
  - Concrete debris containing residual radioactivity.

Groundwater flow directions determined by Malcolm Pirnie (1999) are such that any radionuclides present in the groundwater underlying the industrial area would migrate toward the Connecticut River. The flow rate of groundwater that may potentially contain residual radioactivity, relative to the flow rate of the Connecticut River, should be very small. Therefore, the aquatic foods ingestion pathway is not considered applicable in this calculation.

6.8.2.2 Conceptual Model

The conceptual model underlying the dose model was developed based on the site characteristics expected at the time of release of the site and the initially planned disposition of concrete debris in the basements of the buildings. Key assumptions associated with that conceptual model are as follows:

(i) All concrete debris contains residual radioactivity and this concrete debris comprises the contaminated zone. The contaminated zone extends below the water table, based on an ambient water table elevation of approximately 10 ft msl (Malcolm Pirnie, 1999).

(ii) Approximately 3 ft of soil fill is initially placed on top of the concrete debris. This soil does not contain residual radioactivity.

(iii) The residential farmer constructs his/her home over a debris-filled basement.

(iv) The well that supplies water for drinking, crop irrigation, and livestock is drilled and completed within the debris-filled basement.

The conceptual model described above bounds site characteristics expected at the time of release under the current decommissioning approach.

The parameters used to quantify the conceptual model are presented in Appendix D, Table D-4. The values / distributions assigned to each of the parameters, the basis for assigning such values / distributions and the relevance of the parameter to the dose calculations are presented in Table D-4.

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The distributions and defining statistical parameters assigned to the distribution coefficients (K_d) and plant transfer factors for the concrete debris were determined from site-specific data contained in a report produced by Batelle, Pacific Northwest Division (PNWD) (Reference 6-17). Batelle performed tests on concrete cores taken from the Containment Building and the Waste Disposal Building and used groundwater from the site for the tests.

The conceptual model underlying the dose model includes a contaminated zone above and below the water table. The RESRAD code is typically used to calculate radiation doses (and DCGLs) for a source above the water table. To develop a dose model consistent with the conceptual model, it was necessary to establish the following parameters:

- Model for water transport parameters = Mass Balance (MB) model
- Number of unsaturated zone strata = 0

By establishing the above parameters, the groundwater (well water) concentrations calculated by RESRAD were found to be greater than or equal to the groundwater concentrations in equilibrium with the concrete debris under saturated conditions. The equilibrium groundwater concentration associated with concrete debris was calculated using Equation 6-1 in Section 6.7.2 for a contaminated zone comprising concrete debris.

6.8.2.3 Sensitivity Analysis Results

Parameter distributions assigned in the probabilistic RESRAD Version 6.1 model are presented in Appendix D, Table D-4. An initial radionuclide concentration of 1 pCi/g was used for the concrete debris comprising the contaminated zone.

The stochastic parameters identified as sensitive (|PRCC| ≥ 0.25) to the peak of the mean dose for each of the radionuclides are presented in Appendix E, Table E-4. For each radionuclide, the sensitive parameters are listed in order of decreasing sensitivity. Included in Table E-4 are the conservative values assigned to each of the sensitive parameters.

6.8.2.4 DCGL Determination

Parameter values assigned in the deterministic RESRAD Version 5.91 model are presented in Table E-4 (conservative values assigned to parameters shown to be sensitive) and Appendix F, Table F-4. The groundwater DCGLs were determined for a radiation dose limit of 25 mrem/yr.

The concrete debris DCGLs calculated for each of the radionuclides are presented in Appendix G, Table G-4-1. The time to the peak of the mean dose is included in the table together with the percent contribution to dose from the exposure pathways (water independent and water dependent). The equilibrium groundwater concentrations associated the concrete debris and the groundwater (well water) concentrations for a known concentration of radioactive material in the concrete debris are presented in Appendix G-4-2. The concrete debris DCGLs are summarized in Table 6-4.

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Table 6-4
Base Case DCGLs for Building Demolished (Concrete Debris)

Radionuclide	Concrete Debris DCGLs (pCi/g)
H-3	9.05E+01
C-14	2.05E+01
Mn-54	5.51E+01
Fe-55	8.96E+01
Co-60	9.07E+01
Ni-63	1.29E+02
Sr-90	3.77E-01
Nb-94	7.74E+00
Tc-99	2.85E+01
Ag-108m	2.59E+01
Cs-134	3.21E+02
Cs-137	6.45E+02
Eu-152	2.27E+02
Eu-154	1.94E+02
Eu-155	9.53E+03
Pu-238	1.14E+01
Pu-239	1.00E+01
Pu-241	1.49E+02
Am-241	4.42E+00
Cm-243	3.83E+00

6.8.3 Concrete DCGL Conversion

Table 6-3 shows the DCGLs calculated for building surfaces using the building occupancy scenario, and Table 6-4 shows the DCGLs calculated for concrete debris using the resident farmer scenario. Note that the units for surface sources from the building occupancy scenario are in units of dpm/100cm^2, whereas, for volumetric sources from the resident farmer concrete debris scenario the units are pCi/g. In order to determine operational DCGLs, as discussed in Section 5.4.7.1, a method of converting the volumetric DCGLs to surface DCGLs (and vice versa) is needed.

This conversion is performed by assuming that the entire quantity of radioactivity within the volume occupied by the available fill area up to 3 feet below grade, V, is distributed on the internal surface area of the building, A_b. The value of A_b is determined based on the surface area of concrete prior to demolition. This conversion is performed as follows.

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$$DCGL_s(dpm/100cm^2) = DCGL_v(pCi/g) * \frac{V(m^3)}{A_b(m^2)} * \frac{10^6 cm^3}{m^3} * \rho(g/cm^3) * (1-n) * \frac{1m^2}{10^4 cm^2} * \frac{2.22dpm}{pCi} * 100$$

(Equation 6-4)

where: ρ is the density for concrete, or 2.4 g/cm^3, and
n is the porosity for the buried concrete debris, or 0.3.
100 is used to convert from cm^2 to 100 cm^2

Substituting these values and combining terms provides the following relationship:

$$DCGL_s(dpm/100cm^2) = 37{,}296 * DCGL_{conc}(pCi/g) * \frac{V(m^3)}{A_b(m^2)}$$ (Equation 6-5)

The minimum value of V/A_b has been evaluated and documented in the Technical Support Document (Reference 6-18). The minimum value of V/A_b was determined to be for the Containment Building where

V = 5790 m^3, and
A_b = 16764 m^2.

The value of A_b represents the surface area of concrete within the containment building. The values presented in LTP Table 2-10 represent all structural surfaces within the containment, including steel surfaces that will be included in the final status survey.

Using minimum values therefore ensures that additional conservatism is applied to all plant structures subject to concrete demolition. Substituting these values yields the following relationship for the $DCGL_s$:

$$DCGL_s(dpm/100cm^2) = 12{,}881 * DCGL_{conc}(pCi/g)$$ (Equation 6-6)

The volumetric concrete debris DCGL values are converted to surface activity DCGL values using the above relationship. These surface activity DCGL values are conservative since all of the concrete debris within the backfill volume is assumed to contain residual radioactivity at a level corresponding to the volumetric DCGL. This is unlikely since a large fraction of internal surfaces of the plant structures contain very low levels of residual radioactivity. This total activity is then distributed uniformly over all internal surfaces of the building. The resulting surface DCGL is conservative considering that no allowances are made for the large area occupied by the non-contaminated surface areas and continues to bound the site characteristics expected at site release considering the current decommissioning approach.

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6.8.2 Future Groundwater Dose Subsurface Structures and Basements/Footings: Basement Fill Model

Equation 5-6 will to be used to demonstrate compliance with license termination criteria for land areas potentially affected by groundwater contamination. This equation has three dose components that must total at most the unrestricted release criteria. These dose components are:

- Dose due to residual radioactivity in soil
- Dose due to existing groundwater contamination
- Dose due to "future groundwater" from the burial of concrete structures. This component is from radioactivity being released to the groundwater contained in the building basements

The first two dose components are determined using the Soil and Groundwater DCGLs as described in other portions of this LTP. The future groundwater dose is determined using the Basement Fill Model and includes radioactivity from building basement/footing concrete, the containment liner and embedded piping that will remain on-site after release of site buildings from the NRC license.

6.8.2.1 General Dose Calculation Model

The Basement Fill Model uses the total radioactivity inventory from buried plant structures and embedded piping as its primary input. In this model, the maximum annual quantity of radioactivity released to the saturated zone (water volume below the water table) is calculated individually for each structure and component within the saturated zone. The resulting groundwater concentration is calculated using:

- measured distribution coefficients (K_d) for the selected backfill material (Reference 6-~~22~~ 20),
- a conservative diffusion and buildup factor (Reference 6-~~21~~ 19), and,
- a dilution volume represented by the volume of the containment building in the saturated zone.

Once the future groundwater concentrations are calculated for each radionuclide and from each structure/component, the individual groundwater dose values are calculated using the groundwater DCGLs and then summed. The various simplifying and conservative assumptions, as discussed below, result in a very conservative dose analysis from the future groundwater component.

The basement fill model as detailed in reference 6-~~21~~ 19 establishes the following relationship to calculate the maximum future groundwater concentration, $C_{w,i}$. from radionuclide i and structure/component s as follows:

$$C_{w,i} = \frac{B_i \sum_{s=1}^{N} (s) CFR_{s,i} A_{s,i}}{R_i V} \quad \text{(eq 6-4)}$$

Where:

$CFR_{s,i}$: the cumulative fractional release for each subsurface structure, s, and each radionuclide, i,
$A_{s,i}$: is the total activity of radionuclide i contained in the structure/component s,
B_i: the radionuclide specific buildup factor (this accounts for an increase in groundwater concentrations for radionuclides that diffuse relatively slowly compared to tritium),
R_i: the radionuclide specific retardation factor (this accounts for the adsorption of radionuclides on the backfill material in the basement, calculated from the backfill soil distribution coefficients

determined in the Brookhaven National laboratory study of Kds for CYAPCO backfill soil (Ref 6-20~~22~~)), and

V: the dilution volume, assumed to be the water volume of containment below the water table. This value is taken as 1.37E6 liters and is larger than the assumed resident farmer annual pumping rate.

In this model, the radionuclide specific values of R_i and B_i are based on analysis performed on the specific material to be used as backfill. Analysis samples from two types of backfill material to be used were used to measure the distribution coefficients for the selected radionuclides and the values of R_i and B_i and calculated parameter values. From each of the two types of backfill, the most conservative (i.e. lowest) value was selected. The retardation values, R (unitless) are calculated as:

$$R = 1 + \rho K_d / \eta \quad \text{(eq 6-5)}$$

Where:

ρ is the bulk density of the backfill soil = 1.56 from Table F-1, and
η is the effective porosity of the soil =0.35 from Table F-1.

The calculation of buildup factors as provided in reference 6-~~21~~ 19 accounts for the groundwater flow velocity (from the site hydrogeologic parameters), the retardation values for each radionuclide, and radioactive decay for each radionuclide over time.

Table 6-4 summarizes the values of K_d, R, and B_i as used in this model and in equations ~~6-c~~ 6-4 and 6-5 for each structure/component.

Table 6-4: Parameter Values for Equations ~~6-a~~ 6-4 and ~~6-b~~ 6-5

Radionuclide	K_d	R	B
H-3	0.06	1.26	1.91
Fe-55	1200	5350	1.62
Co-60	22	99	1.65
Sr-90	10	45.6	2.84
Cs-137	45	202	2.86
Eu-152	825	3678	1.93

Reference 6-~~21~~ 19 provides an initial assessment of the concrete structures that will remain following license termination. In this assessment, a concentration of 1 pCi/g for each radionuclide and for each structure was used as an initial starting point. In the implementation of this model, the actual average concrete concentrations will be used to obtain the value of $C_{w,I}$, as described in section 5.7.1.6.
Using the above value for $C_{w,I}$, the future groundwater dose, $H_{future\ GW}$, due to concrete in the containment and spent fuel pool basements is determined from the groundwater DCGLs, $DCGL_{GW,I}$ as follows:

$$H_{future\,GW} = 25 \sum \frac{C_{w,I}}{DCGL_{GW,I}} \quad \text{(eq. 6-6)}$$

It should be noted that the above methodology assumes that concrete, the metal liner and the rebar embedded in the concrete in areas of activation have the same radionuclide concentrations. As discussed in section 5.7.1.6, this factor will be confirmed during characterization sampling. Should the rebar or metal have higher concentrations then the concrete, the higher concentrations will be used in the future groundwater dose calculation.

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As discussed in the ~~r~~eference 6-~~19~~ 17 this method is very conservative due to following simplifying assumptions:

- The literature diffusion rates used are the highest values from the range given in the literature
- The use of the Buildup Factor assigns the highest available radioactivity inventory to the first year even though, for several radionuclides, this maximum occurs in different years.
- The radioactivity is assumed to diffuse from both the inside and the outside of all concrete masses into the same dilution volume. This conservative assumption does not account for additional dilution that will occur from groundwater flow around and through the subsurface basements as penetration holes will be placed in these structures to encourage such flow.

6.8.2.2 Future Groundwater Dose Calculation for Basement Concrete

From equation 6-~~1~~ 4, the cumulative fractional release, $CFR_{s,i}$, is needed for all structures/components, s, and radionuclides, i, contained in the saturated zone (i.e. water table). Reference 6-~~21~~ 18 provides a physical inventory of these concrete structures. This inventory includes all dimensions along with the estimated volumes and masses. For each major structure, the CFR values are calculated from these dimensions and from a conservative selection of diffusion coefficients using:

$$CFR_{s,i} = \frac{2fSA_s(D_i t/\pi)^{0.5}}{V_s} \quad \text{(eq. 6-7)}$$

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) = 1 year or 3.17E7 sec

Based on a literature review of available data and on a review of available experimental data from CYAPCO concrete, ~~r~~eference 6-21 19 selects conservative values of D_i as provided in Table 6-5.

Table 6-5: Concrete Diffusion Coefficients Used in the Basement Fill Model

Radionuclide	Selected Diffusion Coefficient, D_i (cm^2/s)
H-3	5.5×10^{-7}
Fe-55	5.0×10^{-11}
Co-60	4.0×10^{-11}
Sr-90	5.2×10^{-10}
Cs-137	3.0×10^{-09}
Eu-152	1.0×10^{-11}

Using the parameter values above with equations 6-4 and 6-6, the calculated groundwater concentrations and dose are conservative as a result of the following simplifying assumptions.

- The concrete surfaces are assumed to be represented by a semi-infinite geometry.

- No credit is taken for the barrier to diffusion that the containment liner provides. (Although the liner will be pierced by holes that amount to a very small percentage of the liner area, it would still provide a substantial barrier to groundwater/concrete interaction for many years.)
- The Brookhaven Study assumes that 2.5 feet of grout will be placed above the activated concrete region of the In-Core Sump. The depth grout placed above the activated region will actually be 5 feet, thereby providing additional resistance to transport.

~~Although~~ The analysis provided by Reference 6-~~34~~ 19 is specifically for the containment and spent fuel ~~pit~~ pool basements taken together. Other adjacent subsurface structures such as the cable vault portion of the containment and other footings will be inventoried in a similar manner and added to the containment dilution volume. This approach will also be used for other basements that remain (i.e. "B" Switchgear and the discharge tunnels) although the released radioactivity from these additional basements will be assumed to migrate and be included with the discharge tunnel inventories..

6.8.2.3 Future Groundwater Dose Calculation for Surface Contamination on the Containment Liner

As mentioned previously, the containment liner below four feet below grade will remain in place after the containment building is released from the license. This section describes the method to be used to account for this added surface contamination source.

For the containment liner, a characterization and final status survey will be implemented using the building occupancy DCGLs adjusted based on the results of an ALARA evaluation or administrative level not to exceed 25 mrem/yr. These surveys will establish the appropriate radionuclide mix for this surface. For purposes of the basement fill model, this radionuclide mix will be used to calculate a total inventory assuming that the average surface radioactivity concentration is equal to the building occupancy values. Using this inventory, a CFR value of 1.0 will be used along with the total surface area, and, equations 6-4 and 6-6 will be used to calculate the future groundwater dose and concentrations. These values will be added to their respective values for the concrete case discussed in 6.8.2.2. This is conservative approach since it assumes

1. that the total inventory of radioactivity for each radionuclide is released to the groundwater/backfill soil system assumed in the containment basement in the first year after release of this area from the NRC license, and
2. that all surfaces are contaminated to a level equivalent to the building occupancy DCGL .

Using this calculational approach, the surveys conducted for this source will be performed using the MARSSIM guidance as described in Chapter 5 for surfaces.. The results of these surveys will be used to determine average activity concentrations and radionuclide distributions.

The above methodology applies to the containment liner. For all other buildings that have basement surfaces, no liner will remain in place. The contamination on the surfaces of these basements are accounted for in the volumetric sampling and subsequently in the calculation of future groundwater dose due to volumetric contamination. Separate calculations of dose due to this surface contamination on exposed building surfaces are therefore not required.

6.8.2.4 Future Groundwater Dose Calculation for Surface Contamination on Embedded Piping

The last source to be evaluated in determining the future groundwater dose from buried structures and components is that resulting for the surface contamination contained on embedded piping to remain after termination of the license.

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Embedded piping is that which is present in the containment or spent fuel pool basement and will not be removed or grouted. This source will be included in the calculation of the containment interior groundwater concentration. As in the case of the containment liner, surface activity surveys will be performed in accordance with Section 5.4.7, using building occupancy DCGLs determined by an ALARA evaluation or administrative limit. Using the radionuclide mix from these and other characterization surveys, each pipe will be assumed to be contaminated to levels equivalent to a total contamination level corresponding to building occupancy DCGL being used. Using this inventory, a CFR value of 1.0 will be used along with the total surface area, and, equations 6-4 and 6-6 will be used to calculate the future groundwater dose and concentrations. These values will be added to their respective values for the concrete case discussed in 6.8.2.2. This is conservative approach since it assumes

1. that the total inventory of radioactivity for each radionuclide is released to the groundwater/backfill soil system assumed in the containment basement in the first year after release of this area from the NRC license, and
2. that all surfaces are contaminated to a level equivalent to the building occupancy DCGL.

6.8.2.5 Summary of All Future Groundwater Dose Calculations

Sections 6.8.2.2 to 6.8.2.4 show the method to be used determine the future groundwater doses due to the individual sources, buried concrete, containment liner and embedded piping. The individual doses will next be summed to determine the total future groundwater dose from all sources. This calculated dose will be used to supply the "future groundwater" dose component of the compliance equation (6-7). In the case of the containment basement, the future groundwater dose will be calculated as given in the 5-1 methodology above. This calculation will be performed concurrent with the survey of the containment and be available for NRC review prior to the backfilling of the containment basement.

This approach will also be used for other basements that remain (i.e. "B" Switchgear and the discharge tunnels) although the released radioactivity from these additional basements will be assumed to migrate and be included with the discharge tunnel inventories. x

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Table 6-5
Conversion of Base Case
Concrete Debris DCGLs

Radio-nuclide	Base Case DCGLs (pCi/g)	Base Case DCGLs (dpm/100 cm^2)
H-3	9.05E+01	1.17E+06
C-14	2.05E+01	2.64E+05
Mn-54	5.51E+01	7.10E+05
Fe-55	8.96E+01	1.15E+06
Co-60	9.07E+01	1.17E+06
Ni-63	1.29E+02	1.66E+06
Sr-90	3.77E-01	4.86E+03
Nb-94	7.74E+00	9.97E+04
Tc-99	2.85E+01	3.67E+05
Ag-108m	2.59E+01	3.34E+05
Cs-134	3.21E+02	4.13E+06
Cs-137	6.45E+02	8.31E+06
Eu-152	2.27E+02	2.92E+06
Eu-154	1.94E+02	2.50E+06
Eu-155	9.53E+03	1.23E+08
Pu-238	1.14E+01	1.47E+05
Pu-239	1.00E+01	1.29E+05
Pu-241	1.49E+02	1.92E+06
Am-241	4.42E+00	5.69E+04
Cm-243	3.83E+00	4.93E+04

6.9 Operational DCGLs

Future groundwater dose calculated using the Basement Fill model

Since additional scenarios, beyond those described above, may be created by combining pathways from different scenarios (e.g., resident farmer with the ~~buried debris scenario~~), a method to assess doses from these combined pathways is necessary. Additionally, any initial residual radioactivity in groundwater that exists will also contribute to total dose. For example, a resident farmer may locate his residence and raise crops on soil containing residual radioactivity and use groundwater that is in contact with ~~the~~ a buried basements, ~~dispositioned concrete debris~~, which may also contain residual radioactivity. Soil and ~~building surface~~ groundwater DCGLs for these combined scenarios will be determined on an operational basis, using the base case DCGLs for soil, and groundwater, ~~and building surfaces~~ and future groundwater dose, calculated in Sections 6.6, 6.7, and 6.8. Section 5.4.7.1 describes, in detail, the methodology to account for all of these contributions.

along with the future groundwater dose calculated using the Basement Fill Model

6.10 References

6-1 Code of Federal Regulations, Title 10, Section 20.1402, "Radiological Criteria for Unrestricted Use."

6-2 NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," dated December 1997.

6-3 "Groundwater Monitoring Report," Connecticut Yankee Atomic Power Company, Haddam Neck, Connecticut, Malcolm Pirnie, Inc., September 1999.

6-4 Draft Regulatory Guide 4006, "Demonstrating Compliance with the Radiological Criteria for License Termination," August 1998.

6-5 NUREG-1549, "Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination," July 1998.

6-6 NUREG/CR-5512, Volume 1, "Residual Radioactive Contamination from Decommissioning, Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent," October 1992.

6-7 NUREG-1727, "NMSS Decommissioning Standard Review Plan," September 2000.

6-8 1997 Census of Agriculture, Volume I, Geographic Area Series, Table 1. County Summary Highlights: 1997.

6-9 Connecticut Town Profiles 1998-1999, Connecticut Department of Economics and Community Development, Research Section, Public and Government Relations Division.

6-10 NUREG/CR-6676, "Probabilistic Dose Analysis Using Parameter Distributions Developed for RESRAD and RESRAD-BUILD Codes," May 2000.

6-11 NUREG/CR-6692, "Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes", November 2000.

6-12 NUREG/CR-6697, "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes", December 2000.

6-13 "Users Manual for RESRAD, Version 6.0," July 2001.

6-14 "Manual for Implementing Residual Radioactive Material Guidance using RESRAD, Version 5.0", September 1993.

6-15 Yu et al., "RESRAD-BUILD: A Computer Model for Analyzing the Radiological Doses Resulting from the Remediation and Occupancy of Buildings Contaminated with Radioactive Materials," ANL/EAD/LD-3, Argonne National Laboratory, November 1994.

6-16 NUREG/CR-6755, Technical Basis for Calculating Radiation Doses for the Building Occupancy Scenario Using the Probabilistic RESRAD-BUILD 3.0 Code, February 2002.

6-17 Batelle, Pacific Northwest Division (PNWD), "Radonuclide Desorption and Leaching Tests for Concrete Cores from Haddam Neck Nuclear Plant Facilities", March 2002 ✓

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6-18 Technical Support Document, BCY-HP-0018, Building Concrete Surface Area and Volume Determination

6.19 - Technical Support Document CY-HP-0184, Subject: Estimates for Release of Radionuclides from Potentially Contaminated Concrete at the Haddam Neck Plant

6.20 - Technical Support Document CY-HP-0185, Subject Kd Values of Backfill Material for Connecticut Yankee.

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Figure 6-6
Process for Determining Building Surface DCGLs