CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT
362 INJUN HOLLOW ROAD • EAST HAMPTON, CT 06424-3099

December 2, 2013
CY-13-049
10 CFR 50.71(e)(4) and 10 CFR 50.4

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555 - 0001

Connecticut Yankee Atomic Power Company
Haddam Neck Plant Independent Spent Fuel Storage Installation
NRC License No. DPR-61 (NRC Docket Nos. 50-213 and 72-39)

Subject: Biennial Update to the Haddam Neck Plant License Termination Plan

Pursuant to the requirements of 10 CFR 50.71(e)(4) and 10 CFR 50.4, Connecticut Yankee Atomic Power Company provides Revisions 8 and 9 to the Haddam Neck Plant (HNP) License Termination Plan (LTP) (Enclosure 1). These revisions address the changes made to the HNP LTP, since the submittal of the last biennial update on December 1, 2011 (Reference a). Attachment 1 provides a summary and rationale for the changes. Attachment 2 provides instructions for removal and insertion of affected pages for Revisions 8 and 9 of the HNP LTP.

This letter contains no commitments.

If you have any questions regarding this submittal, please do not hesitate to contact Brantley Buerger at (860) 267-6426 ext. 303.

I state under penalty of perjury that the foregoing is true and correct.

Executed on December 2, 2013.

Respectfully,

Wayne Norton
CYAPCO President and Chief Executive Officer

NMSS26
NMSS

Attachments and Enclosures

Attachment 1 – Summary of Proposed Changes to the Haddam Neck Plant License Termination Plan Made in Revisions 8 and 9

Attachment 2 – Instructions for Removal and Insertion, Haddam Neck Plant License Termination Plan, Revisions 8 and 9

Enclosure 1 – Revisions 8 and 9 to the Haddam Neck Plant License Termination Plan

References

a. Letter from W. Norton (CYAPCO) to Document Control Desk (NRC), Biennial Update of the License Termination Plan, dated December 1, 2011 (CY-11-032) (Accession No. ML11346A017).

cc: W. M. Dean, NRC Region I Administrator
M. S. Ferdas, Chief, Decommissioning Branch, NRC, Region 1
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ATTACHMENT 1 TO CY-13-049

SUMMARY OF CHANGES TO THE HADDAM NECK PLANT

LICENSE TERMINATION PLAN

MADE IN REVISIONS 8 AND 9

ATTACHMENT 1 TO CY-13-049
SUMMARY OF CHANGES TO THE HADDAM NECK PLANT LICENSE TERMINATION PLAN MADE IN REVISIONS 8 AND 9

Revision	Section #	Proposed Change	Reason for Change
Revision 8	1.1, 1.3, 1.3.3, 1.4.1, 1.6, 2.1, 2.2.2, 2.3.2, 2.3.3.1, 2.3.3.1.2, 2.3.3.1.3, 2.3.3.1.4, 2.3.3.2, 2.3.3.3, Table 2-5, Tables 2-11A, 11B, and 11C, 2.2 References, 3.1, 3.2, 3.5, 3.6, 4.1, 4.3, 4.4, 5.1, 5.4.3, 5.4.7.1, 5.7.6.1, 5.11, 6.1.1, 6.2, 6.10, 7.1. 7.3, 8.1.3.1, 8.1.3.2, 8.2.1.1, 8.2.1.2, 8.3	Updated to reflect that the decommissioning of the HNP is complete, with the exception of the HNP ISFSI and the applicable land areas.	On September 1, 2004, February 27, 2006, and November 26, 2007, the NRC issued Safety Evaluation Reports that reduced the land areas that remained within the control of the 10 CFR 50 License to only those areas associated with the ISFSI. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.
Revision 8	1.1, 2.3.3.1.4	The discussions regarding the property boundary, site boundary (i.e., boundary of area controlled in accordance with the 10 CFR 50 License), controlled area, exclusion area, and the acreage within the control of the 10 CFR 50 License were updated.	The discussions regarding the various boundaries and areas are consistent with the applicable regulations. The area that remains within the license was previously identified as approximately 4.6 acres. This was changed to approximately 5.7 acres, based on an estimate developed with a software program utilizing a scaled representation of the site. This does not impact the portion of the Survey Units that remain within the control of the 10 CFR 50 License as defined in the NRC's Safety Evaluation Report dated November 26, 2007. This change does not modify the land that remains under the control of the 10 CFR 50 License, it corrects the calculation of the acreage of that land area.

Revision	Section #	Proposed Change	Reason for Change
Revision 8	1.2, 1.4.2, 2.1, 2.2.1, 2.2.2, 2.2.3, 2.2.4.2.2, 2.2.4.2.3, 2.2.4.2.4, 2.3.1, 2.3.3.1, 2.3.3.1.1, 2.3.3.1.2, 2.3.3.1.3, 2.3.3.1.4, 2.3.3.1.7, 2.3.3.2, 2.3.3.3, 2.3.3.5, Table 2-5, Table 2-10, Tables 2-11A, 11B, and 11C, 3.1, Table 3-1, 3.6, 4.1, 5.1, 6.2, Figure 6-1, 8.1.3.1, 8.1.3.2, 8.2.1.1, 8.2.1.2, 8.2.3, 8.2.4, 8.2.5.2, 8.2.6.2, 8.2.8.1, 8.2.9, 8.2.10, 8.2.11, 8.2.12, 8.2.16	Updated to reflect the current activities of the HNP ISFSI.	Changes were made to reflect the current practices at the ISFSI. These changes are consistent with approved procedures or other license basis documents.
Revision 8	1.3.3	Eliminated the list of specific low-level waste disposal sites.	The general statement regarding the need for access to low-level waste sites is sufficient at this time. The decommissioning of the ISFSI is not expected to occur for numerous years, thus, the names, owners, and locations of the sites that will be available at that time is not known.
Revision 8	7.2, 7.3 and Appendix C	Updated to reflect the new cost estimates regarding decommissioning and storage of spent nuclear fuel and Greater than Class C (GTCC) waste approved by FERC in July 2013. The cost estimate assumes that the storage period will be extended from 2022 to 2031 with license termination in 2033. In addition, the decommissioning cost estimate assumes that all of the concrete and steel from the VCCs and ISFSI storage will be shipped offsite as low-level radioactive waste.	The decommissioning cost estimate was submitted to the NRC in December 2012 as part of the Decommissioning Funding Plan. In addition, the Federal Energy Regulatory Commission approved the new decommissioning cost estimate and a new cost estimate for the management of spent nuclear fuel and GTCC Waste in July 2013.
Revision 8	3.5.2, Table 3-3, 8.1.3.2, 8.2.4,	Updated to reflect the environmental impacts associated with the change in schedule for storage of spent nuclear fuel and GTCC waste and change in methodology regarding disposal of the materials comprising the Vertical Concrete Casks and the ISFSI Storage Pad as low-level radioactive waste.	The changes update the environmental impact associated with decommissioning the ISFSI and the longer time period that the spent nuclear fuel and GTCC waste will be stored onsite. The environmental impact remains bounded by the previous assessment.

Revision	Section #	Proposed Change	Reason for Change
Revision 8	2.3.3.1.2, 4.1, 5.1, 5.11, Table 5-7, 6.1.1, 12.5, 12.5.1,	Editorial or administrative changes were made.	These changes are non-substantive changes that do not modify the intent of the document.
Revision 8	3.5.1, Table 3-2, 8.1.3.2,	Updated to address the impacts of decommissioning the ISFSI on occupational health and safety.	The ISFSI structures, systems, and components are not expected to be significantly contaminated at the time of decommissioning. During decommissioning of the ISFSI, the material comprising the Vertical Concrete Casks and the ISFSI storage pad may be shipped off-site as low-level radioactive waste. This activity will be managed, so that doses to workers and the public are minimized and federal regulations regarding doses and dose rates are met.
Revision 8	1.1, 2.1, 3.1, 4.1, 5.1, 6.1, 7.1,	Updated to reflect that some historical information regarding the decommissioning of the HNP has been maintained in the LTP.	In August 2007, the NRC issued a Safety Evaluation Report that reduced the land areas that remained within the control of the 10 CFR 50 License to only those areas associated with the ISFSI. The LTP is only applicable to those land areas. However, some of the historical information regarding the HNP decommissioning may provide value during the decommissioning of the HNP ISFSI; thus, the information was retained.
Revision 8	2.3.3.1.4, 2.3.3.1.6, 2.3.3.1.7, Tables 2-6, through 2-9,	The status of the Groundwater Monitoring Program is updated to reflect the NRC conclusions presented in the Safety Evaluation Report dated November 26, 2007.	This information was updated to reflect the NRC conclusions presented in the Safety Evaluation Report dated November 26, 2007.
Revision 8	8.2.9, 8.3	Updated to include references to an exemption to 10 CFR 50.47 and 10 CFR 50, Appendix E granted by the NRC in March 2013, and re-issued in August 2013.	Letter from M. D. Lombard (NRC) to B. Buerger (CYAPCO), Response to Exemption Request for Portions of Title 10 of the Code of Federal Regulations Part 50 Appendix E, and Section 50.47 of Title 10 of the Code of Federal Regulations for the Yankee Rowe Plant (TAC No. L24663)," dated March 18, 2013. Letter from J. M. Goshen (NRC) to B. Buerger (CYAPCO), Revised Response to Exemption Request for Portions of Title 10 of the Code of Federal Regulations Part 50 Appendix E, and Section 50.47 of Title 10 of the Code of Federal Regulations for the Yankee Rowe Plant (TAC No. L24663)," dated August 15, 2013.
Revision 9	1.1	Reference to Figure 1-1 is changed to 6-1.	Figure 1-1 was deleted in a previous revision; thus, the current reference is incorrect. The information referenced in the text is now provided in Figure 6-1.

ATTACHMENT 1 TO CY-13-049
SUMMARY OF CHANGES TO THE HADDAM NECK PLANT LICENSE TERMINATION PLAN MADE IN REVISIONS 8 AND 9

Revision	Section #	Proposed Change	Reason for Change
Revision 9	2.2.2	The following statement was added to Section 2.2.2: The classification for these Survey Units is based on historical surveys conducted prior to the storage of the spent fuel and GTCC at the ISFSI. Following the removal of the spent fuel and GTCC waste from the site, the areas in Survey Units 9523-0000, 9528-000, and 9528-0004 will be reclassified to the appropriate classification (Class 1, 2, or 3) based on the criteria and guidance provided within the LTP.	Section 2.2.2 is modified to clarify that classifications presented in Table 2-10 for the remaining survey units are historical, and that new classifications in accordance with the LTP will be established for those areas following the removal of spent fuel and Greater than Class C waste from the site.
Revision 9	Figure 6-1	Add a revision bar to reflect the change made in Revision 8. The page will still be identified as Revision 8.	This is an administrative change. The figure was properly identified as Revision 8.

ATTACHMENT 2 TO CY-13-049

INSTRUCTIONS FOR REMOVAL AND INSERTION

HADDAM NECK PLANT LICENSE TERMINATION PLAN

REVISIONS 8 AND 9

ATTACHMENT 2 TO CY-13-049
INSTRUCTIONS FOR REMOVAL AND INSERTION
HADDAM NECK PLANT LICENSE TERMINATION PLAN
REVISIONS 8 AND 9

To simplify the issuance of Revisions 8 and 9 of the Haddam Neck Plant License Termination Plan, Sections 1 through 8 of the LTP are reissued in their entirety. Appendices A, B, and D through H are not affected by this revision. Thus, they are not reissued. The List of Effective Pages defines the applicable Revision # for each page.

Remove	**Insert**
Pages i though x	Pages i through x
Pages LEP-1 through LEP-3	Pages LEP-1 through LEP-3
Pages 1-1 through 1-14	Pages 1-1 through 1-14
Pages 2-1 through 2-111	Pages 2-1 through 2-51
Figures 2-1 through 2-7	Figures 2-1 through 2-7
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Pages 4-1 through 4-8	Pages 4-1 through 4-8
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1 GENERAL INFORMATION

1.1 Purpose

The objective for decommissioning the Haddam Neck Plant (HNP) site is to reduce residual radioactivity to levels that permit release of the site for unrestricted use and for termination of the 10CFR50 license, in accordance with the Commission's site release criteria set forth in 10CFR20, Subpart E. The purpose of this HNP License Termination Plan (LTP) is to satisfy the requirements of 10CFR50.82, "Termination of License" (Reference 1-1) using the guidance provided in Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" (Reference 1-2) and Draft Regulatory Guide-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination" (Reference 1-3). In September of 2000, the NRC incorporated much of the guidance of DG-4006 into various sections of NUREG-1727 (Reference 1-4). References to the corresponding sections of NRUEG-1727 (in which the guidance of DG-4006 have been incorporated) has been given in specific sections of this LTP, as appropriate.

The LTP describes the decommissioning activities that will be performed, the process for performing the final status surveys, and the method for demonstrating that the site meets the criteria for release for unrestricted use. The LTP contains specific information on:

- Historical Site Assessment and Site Characterization;
- Remaining Decommissioning Activities;
- Site Remediation Plans;
- Final Status Survey Design and Implementation Plan;
- Dose Modeling Scenarios;
- Update to the Site-Specific Decommissioning Cost Estimate; and
- Supplement to the Environmental Report.

Each section of the LTP is summarized in Section 1.3.

As of October of 2007, CYAPCO had completed all activities required to remove all areas of the site other then those related to the Independent Spent Fuel Storage Installation (ISFSI) from the Haddam Neck Plant NRC license, Docket No. 50-213 (License No. DPR-61). The NRC approved the removal of the non-ISFSI areas from the license via a partial site release in References 1-22 through 1-24. The approximately 5.7 acre sized area related to the ISFSI is shown in Figure 6-1. The areas that remain in the license include portions of Survey Units 9523-0000, 9528-0000, and 9528-0004. as described in Section 2. CYAPCO has subsequently revised the description of the licensed area of the site in the HNP Updated Final Safety Analysis Report (UFSAR) and Quality Assurance Plan (QAP).

The ISFSI related areas of site will remain in the HNP license until:

- The spent nuclear fuel has been shipped from the site for storage/or disposal
- The facilities in the areas shown in Figure 1-1 have been decommissioned
- A Final Status Survey has been conducted and the results submitted to the NRC

- The NRC approves the partial site release of remaining areas covered by the HNP license

The HNP LTP which follows describes the status of the plant, decommissioning activities, site release criteria and final status survey plans that describe the site as it exists through the decommissioning process.

With the exception of decommissioning activities at the HNP Independent Spent Fuel Storage Installation (ISFSI) to be undertaken when all fuel and Greater than Class C (GTCC) waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

The information included within this section includes historical information that will be maintained in its current form. This information will be reviewed, and revised as necessary, at the time of initiating the decommissioning activities for the ISFSI and associated land areas to ensure that appropriate information is available for the implementation of final status survey activities for the ISFSI and termination of the Part 50 License for the HNP site.

1.2 Historical Background

The HNP site is located on the east bank of the Connecticut River, approximately 21 miles south-southeast of Hartford, at 362 Injun Hollow Road, Haddam, Middlesex County, Connecticut. The HNP site is owned by Connecticut Yankee Atomic Power Company, CYAPCO (Reference 1-5). Figures depicting the site area and buildings are provided at the end of LTP Chapter 2.

HNP, Docket No. 50-213 (License NO. DPR-61), began commercial operation in January 1968. The plant incorporated a 4-loop closed-cycle pressurized water type nuclear steam supply system (NSSS); a turbine generator and electrical systems; engineered safety features; radioactive waste systems; fuel handling systems; instrumentation and control systems; the necessary auxiliaries; and structures to house plant systems and other onsite facilities. HNP was designed to produce 1,825 MW of thermal power and 590 MW of gross electrical power (Reference 1-6).

On December 4, 1996, HNP permanently shut down after approximately 28 years of operation. On December 5, 1996, CYAPCO notified the Nuclear Regulatory Commission (NRC) of the permanent cessation of operations at the HNP and the permanent removal of all fuel assemblies from the Reactor Pressure Vessel and their placement in the Spent Fuel Pool (Reference 1.7). Following the cessation of operations, CYAPCO began to decommission the HNP. The Post Shutdown Decommissioning Activities Report (PSDAR) was submitted, in accordance with 10CFR50.82(a)(4), on August 22, 1997 (Reference 1-8), and was accepted by the NRC (Reference 1.9). On January 26, 1998, CYAPCO transmitted an Updated Final Safety Analysis Report to reflect the plant's permanent shutdown status (Reference 1-10), and on June 30, 1998, the NRC amended the HNP Facility Operating License to reflect this plant condition (Reference 1-11). On October 19, 1999, the Operating License was amended to reflect the decommissioning status of the plant and long-term storage of the spent fuel in the spent fuel pool (Reference 1-12). Additional licensing basis documents were also revised and submitted to reflect long-term fuel

storage in the spent fuel pool (Defueled Emergency Plan, QA program, and Operator Training Program).

In April of 1999, CYAPCO contracted Bechtel Power Corporation, as the Decommissioning Operations Contractor (DOC), to perform the decommissioning activities at the HNP. CYAPCO continued to perform Spent Fuel Pool Island Operations and provide oversight of the activities performed by the DOC, until June 2003, when CYAPCO terminated the DOC contract. CYAPCO managed the decommissioning of the HNP using staff augmentation and subcontractors for specialty work. Currently, CYAPCO is managing the storage of spent fuel and GTCC waste at the HNP ISFSI. Following the removal of the spent fuel and GTCC waste from the site by the Department of Energy, the decommissioning activities associated with the ISFSI and associated land areas will be undertaken.

1.3 Plan Summary

Termination of the NRC license and environmental closure of the HNP site are closely related activities, completion of which will allow the site to be released for future use. The License Termination Plan (LTP) describes the processes to be used in meeting the requirements for terminating the NRC license. An integrated site closure plan has been prepared to include the processes to be used for non-radiological cleanup and release of the site. This information was submitted to the appropriate regulatory agencies. An integrated approach to site release processes will be used to the extent practicable.

The decommissioning will be conducted by performing radiological and hazardous environmental surveys to allow for controlled demolition of structures, removal of the wastes generated from the site, performing the Final Status Survey (FSS) of the remaining foundations/basements and/or soils, and the use of appropriate backfill materials to restore the site to grade elevation. Soils identified to be contaminated above release limits will be removed, an FSS or assessment performed, the area restored to grade using an appropriate backfill material and an FSS satisfactorily completed if required.

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 site. With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site. Due to a change in the approach to waste disposal, some previously identified survey areas were removed from Table 2-10. Appendix H contains historical information from Table 2-10. If areas are later determined to require an FSS (due to a change in waste disposal approach), the classification information provided in Appendix H will be used.

1.3.1 General Information

This LTP has been prepared for the HNP in accordance with the requirements of 10CFR50.82(a)(9). The LTP is being maintained as a supplement to the HNP Updated Final Safety Analysis Report. Each of the sections required by 10CFR50.82(a)(9) are outlined in the subsections below. Note that figures are located at the end of the corresponding Chapters.

1.3.2 Site Characterization

Chapter 2 discusses site characterization activities. The site characterization for HNP includes the results of surveys and evaluations conducted to determine the extent and nature of the contamination at the site. The initial characterization, performed in accordance with the guidelines of the "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," (Reference 1-13) began in 1997 and was completed in 1999. This initial characterization included a Historical Site Assessment (HSA) a review of historical documents, and measurements, samples, and analyses to further define the current conditions of the site. The effort also evaluated hazardous and state-regulated non-radioactive materials at the site that may require remediation and disposal.

The HSA consisted of a review and compilation of the following information: historical records, plant and radiological incident files, operational survey records, and annual environmental reports to the NRC. Personnel interviews were conducted with present and former plant employees and contractors to obtain additional information regarding operational events that caused contamination in areas or systems not designed to contain radioactive or hazardous materials.

Information from previous surveys was reviewed for historical information regarding radiological conditions throughout the site. The current HNP Radiation Protection Program requires that site radiological conditions be assessed and documented by performing operational surveys and evaluations throughout the decommissioning process. The radiological data collected during this process will supplement the initial characterization data and provide a basis for developing plans for remediation and final status surveys.

The information developed during the initial HNP characterization program represents a radiological and hazardous material assessment based on the knowledge and information available at the end of 1999. The objectives of this initial characterization program were:

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

2. To identify the potential and known sources of radioactive contamination in systems, on structures, in surface or subsurface soils, and in ground water;

3. To determine the initial classification of each survey area;

4. To develop the initial radiological and hazardous material information to support decommissioning planning including building decontamination, demolition, and waste disposal;

5. To develop the information to support Final Status Survey design including instrument performance standards and quality requirements; and

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

Operational radiation surveys and additional characterization measurements and samples obtained during cleanup activities will be used to confirm the area classification and effectiveness of the cleanup activities before completing the Final Status Survey.

The site characterization and historical site assessment efforts are summarized in two documents: "Connecticut Yankee Haddam Neck Plant Characterization Report" (Reference 1-14) and the "Haddam Neck plant Historical Site Assessment Supplement" (Reference 1-15).

The LTP includes a summary of information contained in References 1-14 and 1-15. 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.

1.3.3 Identification of Remaining Site Dismantlement Activities

CYAPCO is conducting decontamination and dismantlement activities at the HNP site consistent with activities discussed in the HNP PSDAR. Chapter 3 of the LTP describes those dismantlement and decontamination activities that remain at the HNP. Also included in this section are estimates of radiation dose to workers from decommissioning activities and projected volumes of radioactive waste.

CYAPCO's primary goals are to decommission the HNP safely and to maintain the safe storage of spent fuel and Greater Than Class C (GTCC) waste. On March 30, 2005, all spent fuel and GTCC waste have been removed from the Spent Fuel Pool and stored in an Independent Spent Fuel Storage Installation (ISFSI) at the HNP site. To the extent practical, impacted facility materials and surfaces that remain will be decontaminated to allow for beneficial reuse. Materials that cannot be decontaminated will be sent to an offsite radioactive waste processor to recycle or to a Low-Level Waste (LLW) disposal site. Completion of decommissioning the HNP site depends on the availability of low-level waste disposal sites. With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

One of two types of radiological surveys will be performed on systems, structures, or components that will be demolished and the wastes generated shipped to a LLW storage facility or a licensed clean material landfill, as appropriate:

1. If a Structure, System, or Component (SSC) is known to be contaminated and is to be demolished, a Contamination Verification Survey (CVS) will be performed to ensure that contamination levels are within the established levels to permit controlled demolition.

2. If the SSC is not suspected to be contaminated, an Unrestricted Release Survey (URS), in compliance with the HNP Radiation Protection Program, will be performed to document that the SSC meets the criteria for unrestricted release.

If a contaminated SSC is to remain on site, it will be decontaminated to the required levels, and a final status survey will be performed and documented. This survey will confirm that the site meets the release criteria. The final status survey results for each survey area will be complied into a release record documenting the as-left radiological conditions demonstrating compliance with site remediation criteria. Several release records will be compiled in a series of reports by area(s). These reports, each made up of several release records will be made available for NRC inspection. Following completion of the final status survey and in the absence of any NRC inspection finding the report deficient, surveyed areas may be released from NRC license control.

1.3.4 Site Remediation Plans

Chapter 4 of the LTP describes various methods that can be used during HNP decommissioning to reduce the levels of radioactivity to those which meet the NRC radiological release criteria, that is, does not exceed 25 mrem/yr Total Effective Dose Equivalent (TEDE) and is As Low As Reasonably Achievable (ALARA). This section describes the methodology that will be used to demonstrate that the residual radioactivity has been reduced to a level that is ALARA in compliance with the NRC requirements.

An ALARA analysis determines when cleanup, beyond that required to meet the 25 mrem/yr TEDE dose limit, is appropriate. Figure 4-1 shows the ALARA evaluation process. Generic ALARA screening values may be determined at the planning stage, prior to the start of cleanup, or after some or all of the characterization work is complete. Survey unit-specific ALARA evaluations may be performed later in the remediation and survey processes.

These ALARA evaluations establish remediation levels at which additional cleanup actions are to be taken to reduce residual radioactivity. These different types of cleanup actions may include but are not limited to, chemical decontamination, wiping, vacuuming, scabbling, or high-pressure washing. The methodology and equations to be used for calculating remediation levels are those provided in NRC's Draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination," which was subsequently included in Appendix D to NUREG-1727 (Reference 1-4).

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 advance 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 the 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, and the 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, 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, and embedded piping that is contained in foundations or footings that are 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 (or with the discharge tunnel if they are assessed after the completion of the containment basement fill model).

For building footings, an alternate criteria to the concrete debris Derived Concentration Guideline Levels (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 (or with the discharge tunnel if they are assessed after the completion of the containment basement fill model) 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 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, and ground water and the building occupancy scenario for site buildings. 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 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 with10CFR50.82(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. Buildings to remain after release from the NRC License which are decontaminated at or below the DCGLs, could be allowed to remain standing after the final status survey. Consideration of the building occupancy scenario 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 most site buildings (Chapter 3). With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

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 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 generally 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/basements and soil. The dose modeling approach, described in Chapter 6, evaluates potential exposures resulting from any remaining concrete structures, footings/basements to ensure that the doses are bounded by the conservative DCGLs (after accounting for future groundwater dose) 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.

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;
   - Haddam Neck ISFSI 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 10CFR30, 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 unit(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 control) 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 guidelines levels (as discussed in Section 6 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 1 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 (DRP-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 Operations 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, 2998, 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 NRUEG-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 NRUEG-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 (CYAPC) letter to the 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.

1-22 Letter from T. Smith (USNRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of East Site Grounds from Part 50 License," dated September 1, 2004.

1-23 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of Phase II Areas from Part 50 License," dated February 27, 2006.

1-24 Letter from K. McConnell (USNRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of Land from Part 50 License," dated November 26, 2007.

2 GENERAL INFORMATION

2.1 Purpose

This section provides site characterization information that describes the site as it existed partially through the decommissioning process. On September 1, 2004, February 27, 2006, and November 26, 2007 (References 2-14 through 2-16), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

The information included within this section includes historical information that will be maintained in its current form. This information will be reviewed, and revised as necessary, at the time of initiating the decommissioning activities for the ISFSI and associated land areas to ensure that appropriate information is available for the implementation of final status survey activities for the ISFSI and termination of the Part 50 License for the HNP site.

Initial site characterization of the Haddam Neck Plant (HNP) began following the permanent cessation of operations in the fall of 1977, 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 of 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 was 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 "quality controls" 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, or 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 the 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 the 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, including the HNP ISFSI. CYAPCO has replaced the future groundwater dose calculation method for basements 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 as defined in 1999. 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 2000 (Reference 2-2), and a Historical Site Assessment Supplement (Reference 2-3) was issued in August of 2001. Chapter 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 2-2 and 2-3. 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 as defined in 1999 has been divided into areas with numeric designations. Figures 2-1 through 2-7 provide the area identification numbers for buildings and grounds within the owner controlled area as defined in 1999. The area designations form the basis for survey units presented in Table 2-10. The majority of these classifications are historical, because the only Survey Units that remain within the 10 CFR 50 License are Survey Units 9523-0000, 9528-0000, and 9528-0004. The classification for these Survey Units is based on historical surveys conducted prior to the storage

of the spent fuel and GTCC at the ISFSI. Following the removal of the spent fuel and GTCC waste from the site, the areas in Survey Units 9523-0000, 9528-000, and 9528-0004 will be reclassified to the appropriate classification (Class 1, 2, or 3) based on the criteria and guidance provided within the LTP.

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 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 plant. 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 9522) or the South East Landfill Area (Survey Area 9535). All these areas received characterization surveys. Materials from construction activities that took place outside of the Industrial Area or Radiological Controlled Area generally were released to offsite locations. These locations were identified and surveyed under the Offsite Material Recovery Program (Section 2.2.4.2.4 contains further details). The NRC released these areas were released from the 10 CFR 50 License in letters dated February 27, 2006 and November 26, 2007 (References 2.15 and 2.16).

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 using instrumentation calibrated and maintained in accordance with station procedures utilizing National Institute of Standards and Technology (NIST) traceable calibration sources. The program consisted 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 HNP. 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 buildings, 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.

Table 2-1
Typical Instruments Used at HNP

Instrument	**Efficiency (Nominal)**	**MDA**	**Probe Type**	**Detection**	**Units**	**Characteristics**
Eberline ASP-1	NA	NA	GM (HP-270)	βγ	mr/hr	Shipping instrument currently calibrated for gamma. This instrument can use various probes.
Eberline E-130A	NA	NA	GM	Bγ	mr/hr	Dose rating laundry bags.
Eberline E-140	>12%	700	GM Pancake	Bγ	cpm	Frisker, battery operated
Eberline E-520	NA	NA	GM (HP-270)	Bγ	mr/hr	Shipping instrument currently calibrated for gamma
Eberline E-600 SHP-100CGS	15%	1000		Bγ	Cpm	Scaler/count rate instrument with various probes.
SHP-300	NA	NA		Bγ	μrem/hr	
SHP-360	10%	60		B	cpm	
SPA-3	15%	18000		γ	cpm	
239-1F Floor Monitor	20%	Varies With Back-ground	Gas Flow	B	cpm	
Data Logger capability	30%			α	cpm	
Eberline E-530N	NA	NA	GM Tube	γ	mr/hr	Shielded directional Probe for high dose rates
RSS-112 Reuter Stokes	NA	NA	Pressurized Ion Chamber	γ	μr/hr	Low dose rate monitoring
Eberline MP-2	NA	NA	NA	NA	Pulses	Used to calibrate Count rate Instruments and AMS-3
Eberline PS-1	12-17%	700	GM Pancake	Bγ	cpm	Emergency Plan Instrument. (Digital Scaler) Used also to calibrate the Gamma-10.
Eberline PS-2-2	12-17%	700	GM Pancake	Bγ	cpm	Emergency Plan Instrument. (Digital Scaler) Used also to calibrate the Gamma-10.
Eberline RM-14	> 12%	700	GM Pancake	Bγ	cpm	Frisker, AC powered

Table 2-1
Typical Instruments Used at HNP

Instrument	Efficiency (Nominal)	MDA	Probe Type	Detection	Units	Characteristics
Ludlum 19	NA	NA	Scintillation	γ	µr/hr	Used for boundary surveys, dose rates of trash, etc.
Ludlum 2200	17% B 11% α	<100 B < 100 α	HP-210 B 43-2 α 43-1 α	Bα	dpm	Smear counting. (Approx 12 43-2 Probes and 4 43-1 Probes.
Bicron Electra 1B with DP6BD Probe	> 20% B	1000	dual phosphor scintillation	Bα	cpm	--100 cm^2 sized Probe --Detects α & B Simultaneously or Independently.
Bicron Electra 1B with DP6DD Probe	> 12% B 8% α	1000	dual Phosphor scintillation	Bα	cpm	Same as the DP6BD probe. This probe has a Double mylar window.
NE Technologies SAM-9	10-30%	<5000	Scintillation	γ	dpm	Small article monitor.
Bicron Micro-Rem	NA	NA	Scintillation	γ	µr/hr	Used for yard Surveys and truck Surveys. Pulser Instrument.
Exploranium Gamma Ray GR-130 Minispec In Situ Monitor	NA	NA	NaI	γ	Selectable cm to m/hr	This instrument is used only for qualitative analysis and store up to 122 complete 256 channel spectra in memory. This instrument can be used in an operational mode as a survey meter and dose meter also. Data logger capability downloadable to a computer via serial link.
ISOCS – In Situ Object Counting System Canberra	40%	NA	NaI	γ	Variable	Collimator. Analyzes spectra data and generates Reports.

Table 2-1
Typical Instruments Used at HNP

Instrument	**Efficiency (Nominal)**	**MDA**	**Probe Type**	**Detection**	**Units**	**Characteristics**
CM-11	30-34% B 17-19% α	1000 100	DE-11A Gas Flow Proportional	Bα	dpm	Used mainly at control point for radon. 100 cm^2 sized probe. One is on a portable cart. Pulser instrument.
APC II Counter	22% α 35% B	Variable	Gas Flow Proportional	Bα	Counts	Low background scaler/counter
XLB-1 Counter Ludlum	24% α 38% B	Variable	Gas Flow Proportional	Bα	Counts	Low background scaler/counter

2.2.4 Results

2.2.4.1 Routine Releases

Normal operations at HNP resulted in releases of radioactive material through both liquid and gaseous pathways. Releases were monitored in accordance with the requirements of the plant Technical Specifications. The routine gaseous release pathway was via the main stack, adjacent to the Containment Building. Routine liquid releases were made after sampling and analysis of the liquid in the test tanks. Effluents were quantified and reported to the NRC in the Semiannual and later Annual Radioactive Effluent Release Reports, as required by the plant Technical Specifications. Analysis of the routine gaseous releases from the normal effluent pathways indicates typical short-lived radionuclides and inert gases typical of nuclear power plant operations. These gaseous releases did not result in site contamination and, therefore, do not impact the site relative to decommissioning activities.

2.2.4.2 Operational Events

Information reviewed during the HSA identified several events that involved atmospheric releases, unplanned liquid releases, facility contamination and release of radioactive material. The following sections describe the major events in each of these categories and the site areas impacted by the events.

2.2.4.2.1 Atmospheric Releases

During the initial years of operation at HNP, several events involving unplanned releases of airborne radioactivity occurred due to equipment failures or operator errors. During 1971 through 1979, these releases occurred in various areas of the plant, with the resulting radioactive discharges to the environment primarily involving inert gases and iodine. These releases were documented in station abnormal occurrence (AO) reports or plant incident reports (PIR), with the discharges included in the Semi-Annual Effluent Release Reports. As indicated by and discussed in the Site Characterization Report, these occurrences involved short-lived radioactive gases and iodines that did not result in contamination of site areas that would impact decommissioning activities. Examples of typical gaseous releases are listed in Table 2-2. Additional information is provided in the "Haddam Neck Plant Historical Site Assessment Supplement" and in the HNP 50.75(g)(1) files, as well as Survey Area files associated with areas impacted by various events.

Table 2-2
Examples of Unplanned Gaseous Release Events

Date	CYAPCO Reference Document No.	Location of Event	Survey Area Number (Figs 2-1 through 2-6)	Medium	Event Description
5/10/71	CYH-1686, Dated 5/10/71, Letters dated 6/11/71, 6/18/71, 6/22/71, 4/27/99	Containment	9312-0001	Gas I-131	Iodine release due to Operator error with penetration seals.
5/19/72	AO-72-2	Ion Exchange Cubicles	9312-0005	Gas Xe-133 & Xe-135	Unplanned airborne release from demineralizer due to operator error
4/26/73	PIR 73-134	Steam Generator	9312-0001	Gas	Primary to Secondary Leak, High RMS Alarm-Air Ejector Monitor
6/21/73	AO-73-6	Ion Exchange Cubicles	9312-0005	Liquid/Gas	500 gallons leaked to the Aerated Drains Tank due to failed valve. Gas release.
3/30/76	LER 76-8/3-L	Waste Gas Decay Tank Cubicle	9312-0010	Gas	Waste Gas Decay Tank rupture disk failed
9/18/77	LER 77-21/1P	"A" Waste Gas Decay Tank	9312-0010	Gas	"A" Waste Gas Decay Tank rupture disk actuated
12/16/79	PIR 79-125	Main Stack	9302, 9312, 9514, 9527	Particulate, Stack Release	Degassifier rupture disk release of primary coolant to main stack

Atmospheric releases that have impacted the site area include a series of events that occurred in 1979. Stack discharges involving particulate activity occurred in February and December of 1979 as a result of the failure of a level controller in the letdown degassifier, flooding of the degassifier, and actuation of the degassifier rupture disk. The letdown liquid (primary coolant) then overflowed the degassifier. The discharge line from the rupture disk was routed directly to the main discharge stack. Efforts to clean the stack following the first incident may also have resulted in particulate releases. Surveys identified a number of localized areas of elevated activity within the Radiologically Controlled Area, within the fenced area of the plant site, and in the parking lot and hillside east of the plant. The results of the site surveys are documented in Reference 2-4. The majority of the radioactive particles were found on the roof areas of buildings close to the stack, and on the ground within the Radiologically Controlled Area. The extent of area outside the Radiologically Controlled Area impacted by the releases includes the parking lot north of the industrial area, the hillside east of the plant out to 200 meters, areas adjacent to the discharge canal, the lower parking lot and Information Center. At that time, the isolated spots of contamination were remediated. The parking lot has since been expanded (with the previous asphalt removed) and paved. Surveys of the parking lot conducted more recently (1997) have identified no contamination.

Atmospheric releases, following the particulate releases of 1979, included gaseous and iodine releases documented in the Annual Effluent Reports submitted in accordance with the plant Technical Specifications. The releases predominantly consisted of inert gases and radioiodines with short half-lives, radionuclides that do not impact the site relative to decommissioning.

2.2.4.2.2 Liquid Releases

In addition to routine releases, spills through the storm drain system affected Survey Area 9522, Southeast Site Grounds, and ultimately reached the discharge canal, Survey Area 9106. Sampling of the discharge canal occurred in the fall of 1997. Sediment samples were obtained at seven (7) locations down to a depth of two (2) feet. In the northern canal, none of the samples had plant-related radioactivity levels greater than the corresponding DCGL. In the southern canal, no plant-related radioactivity was detected.

A review of the available Annual Radiological Environmental Operating reports (AREOR) indicates that licensed material (e.g., Cs-134, Co-60) has been identified in the past, in the vicinity of discharge in the southern canal. All measurements indicated concentrations at a small fraction of the applicable soil DCGL (i.e., <5%) with the exception of a single measurement performed in September 1975, which indicated a concentration of 0.54+/0.09 pCi/g for Co-60. However, decay correcting this concentration to the time of LTP submittal results in an expected concentration that is a small fraction of the soil DCGL.

The information available at the time of initial classification support classification of the southern unit of the Discharge Canal (from the Canal Road to the river) as a Class 3 area. The 1999 AREOR reported a concentration of 0.50+/-0.10 pCi/g for Co-60, which is approximately 15% of the Co-60 "base case" soil DCGL. Given the higher potential for residual radioactivity, the entire discharge canal was reclassified as a Class 2. This example demonstrates the CYAPCO Survey Area/Unit classification process, which includes a unit reclassification process that drives the most appropriate classification.

In addition to the spills identified above, the HSA identified a number of leaks and unplanned liquid releases that have occurred during the operational lifetime of HNP. The majority of the occurrences were confined to the Radiologically Controlled Area. The leaks and unplanned releases were associated with equipment failures and operational events associated with components within the Containment Building, Primary Auxiliary Building, outside storage tanks, and the radioactive waste processing systems. The most significant unplanned liquid release occurred in 1984, the result of a failure of the reactor cavity seal. Although this event did not result in any release outside of the buildings it is considered the most significant because of the large volume of water involved. This water, released to the basement of containment, was pumped through the purification system to the Refueling Water Storage Tank (RWST). Smears of the seal ring indicated 350 mR/hr gamma, 5.4 rad/hr beta with 120,000 dpm alpha. RWST radiation levels ranged from 65 to 200 mR/hr as a result of relocating water during this period. Surveys in the basement of Containment on August 23, 1984, indicated contamination levels to 295 mrad/hr beta and 6300 dpm/100 cm^2 alpha. Sampling the yard drains concluded that no liquid radioactivity was released to the environment.

Examples of typical events involving radioactive liquid leaks and unplanned releases are presented in Table 2-3. A record of unplanned liquid releases is maintained on site in accordance with the requirements of 10CFR50.75(g) and is identified in the affected area assessment in the Haddam Neck Plant Characterization Report (Reference 2-2) and in the Historical Site Assessment Supplement (Reference 2-3).

The principal impact of these events is to the ground within the Radiologically Controlled Area (RCA). Migration of radioactivity to subsurface soils has occurred in the area of the Primary Auxiliary Building and the Refueling Water Storage Tank, adjacent to the Containment Building. Radioactivity has also been detected in groundwater wells on site. Assessments of radioactivity in soils and groundwater are further discussed in Sections 2.3.3.1.3 through 2.3.3.1.7. Radioactive materials from leaks have also impacted the area in the southeast corner of the protected area, a leach field south of the protected area (but within the owner controlled area) as defined in 1999, and drain systems leading from the RCA. All of these areas were within the owner controlled area as defined in 1999.

Table 2-3
Examples of Unplanned Liquid Release Events

Date	CYAPCO Reference Document No.	Location of Event	Survey Area Number (Figs 2-1 through 2-7)	Medium	Event Description
11/1/73	AO-73-11	RWST	9312-0005	Liquid	Valve Leak, 270 liters of liquid released to storm drain.
1/28/76	PIR 76-15	"A" Recycle Test Tank	9312-0005	Liquid	15 gal. of liquid leaked from tank to diked area
5/22/76	LER 76-13/990	PAB Below Drumming Room Floor	9312-0006	Liquid	Leakage from drain line below floor
2/24/77	LER 77-2	"A" Recycle Test Tank	9312-0005	Liquid	1000 gal. of radioactive water released to diked area around tank.
2/23/79	PIR 79-27	Main Stack	9312-0001	Liquid	Manway leakage following SG blowdown rupture disk actuation. 20 gal to yard area.
3/6/79	PIR 79-38	Main Stack	9312-0001	Liquid	Manway leakage following SG blowdown rupture disk actuation
3/28/83	PIR 83-37	Septic Tank	9520-0001	Liquid	84 gal. of water from Chem Lab to Septic Tank
8/21/84	PIR 84-136	Containment Building	9312-0001	Liquid	Reactor cavity seal ring failure. 200,000 gals of water drained to lower levels of Containment Building
2/24/89	PIR 89-35	Leach Field, 115 kV yard	9522, 9312	Liquid	50 gal. release from SFB floor drain, line discharges to 115 kV yard
9/14/90	PIR 90-239	RWST	9312-0005	Liquid	6 gallon per day leak from RWST identified from inventory monitoring
8/12/91	PIR 91-149	Pipe Trench	9312-0006	Liquid	400 gal. release from open valve to pipe trench

2.2.4.2.3 Contamination of Buildings

The HNP design included several structures, engineered and constructed to contain radioactive material. These structures included the Containment Building, the Primary Auxiliary Building, the Service Building, the Waste Storage Building, Ion Exchange Structure, Spent Resin Facility, and structures containing tanks for storage of radioactive liquids. Operations and maintenance activities in these buildings have resulted in surface contamination typical of nuclear power plants. Additionally, the HSA has identified a number of events that affected the radiological status of these structures. Following events resulting in internal structure contamination, decontamination activities were implemented based on ALARA considerations. The decontamination efforts performed during normal operation were conducted to reduce occupational radiation exposure and were not undertaken to achieve site release conditions. The HSA did not identify any events that created an unexpected scope of contamination based on the design intent of these structures. Information gathered during the HSA did indicate that on two occasions-one in 1979 and one in 1989-the plant operated with failed fuel at a level that resulted in an increase in the level of alpha emitting radionuclides in the Reactor Coolant System. Events, as well as routine maintenance activities during these periods, increased the alpha emitting component of the source term.

The HSA identified primary-to-secondary leakage events resulting from steam generator tube leakage. The events occurred during several operating cycles, with the first leakage identified in 1973 and the final events occurring in 1990. The leakage has resulted in measurable radioactivity in small areas of the secondary system piping, primarily in the high-pressure steam components within the Turbine Building.

2.2.4.2.4 Release of Radioactive Materials

The Historical Site Assessment identified several events in which radioactive material was found outside the Radiologically Controlled Area. The primary locations of discovery were the southwest peninsula, Survey Area 9520, and the shooting range landfill area, Survey Area 9535. These areas have been remediated and are no longer controlled as "Radioactive Material Areas" with restricted access. During plant operations, the peninsula area was used for storage of materials. The materials were typically associated with maintenance activities performed during outages or plant modifications. Documentation indicates that the radioactive material was detected in 1980, 1985, and 1989. Surveys performed in 1998, indicated some previously stored materials contained detectable radioactivity. Since the shutdown of the plant, these stored materials removed from the peninsula area were evaluated using the same process as material leaving the Radiologically Controlled Area. Surveys for free release were required prior to that material leaving the industrial area.

Other instances of minor levels of contamination being identified in areas outside the Radiologically Controlled Area but within the site boundary are documented in the Historical Site Assessment. The levels indicated in surveys are detectable, but typically are below the DCGLs. The areas affected were areas of high personnel traffic, such as the Administration Building or the Steam Generator Mock-up Building. Upon discovery, remediation occurred and more extensive surveys of the areas were performed. In 1997, an extensive survey was

conducted of the material within those support buildings housing materials that may have, at one time, been in the plant. Any material or object with any indication of radioactive material was returned to the Radiologically Controlled Area for disposition.

Surveys of approximately 12,000 items located outside of the Industrial Area indicated only 23 items that had contamination that was greater than or equal to 1000 dpm as measured with a HP210 probe. The maximum contamination level found on an individual item as 8,000 dpm. None of these items exhibited detectable removable contamination. Approximately 175,000 items located outside of the RCA but inside of the Industrial Area were surveyed. Of these, only 105 items had detectable contamination greater than or equal to 1000 dpm by direct frisk. The maximum contamination level found on an individual item was 45,000 dpm. Three items with removable contamination were found in the operations tool cage of the Turbine Building. These were 1) on the inside of a small pipe reducer bushing-2,600 dpm beta and 180 dpm alpha, 2) inside a ½-inch pipe elbow-200 dpm beta and 30 dpm alpha, and 3) inside a 1/4 –inch pipe to ferruled adapter-50 dpm beta and no alpha. During the course of the project, no other removable contamination was identified.

Table 2-4
Summary of Unrestricted Release Confirmatory Survey Program

Area	# of Items Surveyed	Items > or =1000 dpm	Maximum Contamination Level (β and γ in dpm)
Outside the Industrial Area			
Recycle Building	1667	1	2000
Warehouse	8716	21	8000
Miscellaneous	1381	1	1000
Within the Industrial Area			
Sea Van Containers	23822	30	15000
Fabrication Shop	777	4	15000
Maintenance Shop	68051	23	20000
Electrical Shop	15270	3	4000
Butler Bldg –Clean	469	0	
Control Room	75	1	3000
Turbine Bldg Areas	23126	15	10000
Gen Const Bldg	6330	11	6000
Weld Shop	8548	9	45000
I&C Shop	29137	9	15000
Records Vault	34	0	

Additional events have been documented, indicating the release of potentially contaminated material from the site. The material in question has been primarily construction materials such as concrete blocks and excavated soils. These materials were addressed through the Offsite Material Recovery Program that was completed in 2003. Materials identified as "plant related" radioactive and nonradioactive materials were appropriately dispositioned.

2.3 Initial Site Characterization

2.3.1 Introduction

Radiological characterization of the HNP site has been on-going since the plant began operation in 1968. Radiological surveys and sampling for radionuclides have been conducted as part of a routine surveillance program in support of the plant radiological safety program, the environmental monitoring program, and in response to operational events. At the time of final shutdown and the cessation of nuclear operations in 1996, a substantial amount of radiological information existed. This information was evaluated to determine the need for additional data. The radiological information base was compared to the results of the HSA to determine the type and amount of new or updated information that is necessary. Although radiological

characterization will continue throughout the decommissioning process, an initial amount of characterization data was necessary to support preparation of the LTP.

The site characterization process focused on data for structures, systems and the site environs, considering radiological, hazardous and state-regulated materials. Groundwater is included in the assessment of the site environs. In addition, activation analyses were performed on components and structures subjected to neutron flux, to support planning efforts.

The information provided in this section summarizes the characterization of the HNP site. The data is based on surveys, samples and analysis performed through the end of 1999 and is the planning basis for remediation activities, establishment of area classifications, and the development of the Final Status Survey Plan. The figures included in this section depict the plant site at the time of plant shutdown.

2.3.2 Methodology

A Data Quality Objective (DQO) approach was applied to the characterization process. This approach focused the effort on gathering sufficient information to achieve the objectives identified in Section 2.1. The site characterization process began with the consolidation of information gathered during the HAS, radiological survey and sample analysis data maintained within the HNP document control program, and current site radiological data maintained by the Radiation Protection Department. This information has been augmented with the results from surveys performed in support of Decommissioning Operations Contractor (DOC) proposal development. The HAS provides the basis for identifying suspect areas outside of those designed to contain radioactivity or those expected to be impacted by normal operations of a nuclear plant.

To facilitate the evaluation of information, available data were sorted by building or structure, and by land areas. A second sort of information was developed based on systems. A unique characterization report was developed for each building or structure with a cross reference to those systems contained within those areas. The characterization process was controlled by site procedures. These procedures established a consistent approach to the evaluation of each area for radiological, hazardous and state-regulated materials that are known to exist or are potential contaminants in structures, systems or soils within that area. The procedural process included the evaluation or determination of:

- the structure or area bounded by the evaluation;
- systems contained within the area, if any;
- HAS identification of events that may have impacted the area;
- area use, present and historic;
- materials of construction;
- presence of radioactive or hazardous material storage areas;
- radiological survey information including extent of area containing radioactivity, radionuclides, exposure rates and contamination levels, both present and historic; and
- potential for migration of contaminants from contiguous areas.

The evaluation of each area included a walkdown of the area by a professional experienced in radiological hazards and a second professional experienced in hazardous and state-regulated materials.

Fuel cladding failures occurring during operations, primarily affected the Reactor Coolant System, as well as liquid systems that interfaced with the Reactor Coolant System. As the result of leaks during operation and refueling outage maintenance activities, it is acknowledged that Transuranics (TRUs) may have been released from these systems to surrounding rooms or cubicles. These rooms, cubicles and systems were monitored extensively during outages for personnel protection purposes, and the resulting data obtained from monitoring activities were used to classify them as Class 1. The upper areas of the Containment and Containment dome were initially classified as Class 2. These areas were demolished and disposed of as radioactive waste.

During the planning of characterization activities, Transuranics activity was considered in survey design by reviewing the operational history and operational surveys of the area. If TRU activity was suspected in an area, proper instruments were selected, smears counted for alpha activity and soil samples analyzed for Am-241, a predominant alpha emitter at the HNP.

Upon completion of the records review and the walkdown of each area, a characterization report for that area was completed. The characterization report for each area contains:

- a description of the area;
- survey units;
- summary of radiological, hazardous, and state-regulated material conditions within the area;
- classification of the area in accordance with the categories defined by MARSSIM;
- radiologically impacted systems within the area; and
- additional sampling and analysis necessary to support remediation.

The area characterization reports have been compiled in the "Connecticut Yankee Haddam Neck Plant Characterization Report" dated January 6, 2000 (Reference 2-2). Additional information is provided in the "Haddam Neck Plant Historical Site Assessment Supplement" (Reference 2-3).

2.3.3 Site Characterization/HSA Results

2.3.3.1 Radiological Status

On September 1, 2004, February 27, 2006, and November 26, 2007 (References 2-14 through 2-16), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

Currently, CYAPCO is managing the storage of spent fuel and GTCC waste at the HNP ISFSI. Following the removal of the spent fuel and GTCC waste from the site by the Department of Energy, the decommissioning activities associated with the ISFSI and associated land areas will be undertaken. There is the potential for activation of concrete and steel that comprise the Vertical Concrete Casks and the ISFSI storage pad. The decommissioning cost estimate assumes that all of this material will be shipped to a low-level radioactive waste storage facility.

2.3.3.1.1 Systems

An extensive review of systems was conducted to determine those systems that contain radioactive materials or in which radioactive material was detected at some time during the operating history of the plant. Systems that were identified as "affected" required additional surveys to define the extent and magnitude of radioactivity. For those systems that may have been impacted due to steam generator tube leakage or other operational events in the past, but for which subsequent samples have not identified radioactivity, the "affected" status is maintained. A listing of historical plant systems and their status relative to the potential for radioactivity was developed. The assessment considered the internal portions of the systems. Systems that may be assessed as "unaffected" and are located in contaminated areas may themselves be externally contaminated and may be considered for remediation or disposal as radioactive waste.

For those systems designed to contain radioactivity, such as the Reactor coolant System and Radioactive Waste Processing Systems, the associated radiological conditions were continuously changing, with the most recent information necessary to support radiation protection activities maintained by the site Radiation Protection Department. These systems were evaluated for remediation or disposal as radioactive waste based on economic evaluation of the alternatives.

Several components, such as the gland seal and turbine casing, were identified as "affected" based on primary-to-secondary leakage identified in operating cycles as recent as 1990. These components contained low levels of radioactivity. The extent of the contamination was further defined as the systems were disassembled and the internal surfaces become accessible. These items are identified in the sections of the characterization report associated with the areas containing the systems.

Table 2-5
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2.3.3.1.2 Buildings

Radiological surveillances performed routinely at the HNP, including the HNP ISFSI, are designed to ensure compliance with 10CFR20 requirements regarding posting of areas, and to provide a basis for establishing controls for the safety of workers involved in those areas. The radiological information from the Contamination Verification Surveys (CVS) provides the basis for the demolition of Systems, Structures, or Components (SSCs) at the site. Once the initial remediation processes have adequately reduced the ambient radiation levels for controlled demolition, access and control of the SSC will be formally turned over to the demolition contractor. In general, the demolition contractor demolished the SSCs to 4 feet below grade

elevation and left any remaining foundations at or less than 4 feet below grade if all of the structure and foundation are removed. In some of the structures all of the forms and foundation concrete were removed within the 4 feet below grade elevation. In structures where portions of the foundations remained once demolition activities reach grade or the 4 feet below grade elevation, the remaining concrete was part of the final status survey of the land area after the area is backfilled with clean material. When basements remain below the demolition elevation the Basement Fill Model was applied to the portions that are to remain below the saturated zone.

The extent and nature of radioactive material in the primary structures on site, is identified in the "Connecticut Yankee Haddam Neck Plant Characterization Report" and Historical Site Assessment.

On September 1, 2004, February 27, 2006, and November 26, 2007 (References 2-14 through 2-16), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

Currently, CYAPCO is managing the storage of spent fuel and GTCC waste at the HNP ISFSI. Following the removal of the spent fuel and GTCC waste from the site by the Department of Energy, the decommissioning activities associated with the ISFSI and associated land areas will be undertaken. There is the potential for activation of concrete and steel that comprise the Vertical Concrete Casks and the ISFSI storage pad. The decommissioning cost estimate assumes that all of this material will be shipped to a low-level radioactive waste storage facility.

2.3.3.1.3 Radiologically Controlled Area Grounds

The Radiologically Controlled Area (RCA) grounds consisted of paved areas around the Containment Building, Primary Auxiliary Building, Refueling Water Storage Tank and waste storage tanks, and the Fuel Building.

On November 26, 2007 (Reference 2-16), the NRC issued a Safety Evaluation Report that released the majority of the site from the 10 CFR 50 License, including the radiologically controlled area grounds. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

2.3.3.1.4 Non-Radiologically Controlled Area Grounds

The Historical Site Assessment identified that some material contaminated with radioactive material was placed in the shooting range landfill area along with construction debris. The assessment identified that between 1974 and 1996, construction materials from approximately 32 site projects had been placed on the landfill. Examples of materials identified are discharge canal dredging spoils, excavated soils, construction debris and sand. The materials originated from areas both within and outside the Radiologically Controlled Area boundary. A separate landfill

area southeast of the borrow pit was permitted by the State of Connecticut for disposal for bulk wastes.

When certain construction projects were undertaken, bulk materials that were generated were transported and placed in piles in the shooting range area. As multiple projects were completed, the piles increased in height to the current estimated height of approximately 3 meters from the original elevation. Some bulk materials were deposited in the landfill area without a proper survey to evaluate concentrations or quantities of radioactive material. This issue was discussed in Inspection Report 50-213/97-08 and in CYAPCO's December 5, 1997, reply to the notice of violation contained in the inspection report. LER 97-017 also discussed this issue and was submitted on November 18, 1997. Based upon the Historical Site Assessment, there are no known areas on the HNP site that were excavated for the purpose of burial of radioactive material.

The site characterization group performed radiological surveys of the landfill area with confirmatory surveys conducted by ORISE (Oak Ridge Institute for Science and Engineering). The initial characterization surveys were completed in 1997. Results of the radiological survey and ground penetrating radar survey established the size of the landfill area to be approximately 5000 square meters and an approximate thickness of 3 meters. This area was Survey Area 9535 located about 1 mile southeast of the Containment Building on higher elevations between the Salmon River and the discharge canal within the property boundary.

On February 27, 2006 (Reference 2-15), the NRC issued a Safety Evaluation Report that released the Survey Units for Survey Area 9535 from the 10 CFR 50 License.

In December 1997, surveys were performed of paved areas within the industrial area but outside the Radiologically Controlled Area (specifically Survey Areas 9302, 9304, 9306 and 9308 (9522). These surveys identified seven discrete areas of contamination ranging from 10,000 to 65,000 dpm, each limited to less than 100 cm^2. The seven discrete areas were found in Survey Areas 9302, 9306 and 9308 (9522). Information concerning these surveys is provided in Reference 2-6. The radioactivity identified was remediated prior to completion of the survey. On November 26, 2007 (Reference 2-16), the NRC issued a Safety Evaluation Report that released these areas of the site from the 10 CFR 50 License.

Land areas adjacent to the industrial area had been surveyed in response to the events occurring in 1979, discussed in section 2.2.4.2.1. Subsequent surveys were performed in the land areas adjacent to the industrial area fence. The criteria for surface area coverage was to survey to a distance from the plant twice that of the furthest point where localized areas of elevated activity were detected. These areas of localized elevated activity were detected and removed from the area close to the industrial area fence, with few areas found beyond 100 meters. The details of the surveys and the results are presented in Reference 2-4.

Additionally in 1989, Warehouse #2 and the Office Building #3/PAP were built and the primary parking lot was re-configured. On November 26, 2007 (Reference 2-16), the NRC issued a Safety Evaluation Report that released these areas of the site from the 10 CFR 50 License.

Table 2-6
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Table 2-7
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2.3.3.1.5 Subsurface 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. Subsurface radioactivity refers to residual radioactivity that is underneath structures such as building floors/foundations or material that is covered with clean soil or some other unaffected layer(s).

The historical site assessment was consulted to identify those survey areas where the potential exists for subsurface radioactivity. 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. Soil data from both the historical site assessment and any pertinent characterization data will be used to establish a bounding depth profile for any potential sub-surface radioactivity. However, the assessment of all subsurface soil contamination is not currently complete. Soil in difficult to access areas such as under tanks will be deferred until later in the decommissioning process, when access will be more readily available.

During 1998 and 1999 over one hundred subsurface soil samples, in some cases down to six feet in depth, were collected in Survey Areas 9302, 9304, 9306 and 9308 (9522) in support of plant modifications and site characterization activities. None of the samples had plant-related radioactivity levels greater than the corresponding base case soil DCGL. During the same time period, over two hundred subsurface soil samples, in some cases down to approximately two meters in depth, were collected inside the RCA in Survey Areas 9307 (9522), 9310 (9522), 9312 and 9227 (9522) (Bus-10 Pad and ground underneath). Some isolated locations showed Co-60 and Cs-137 activity levels up to several hundred pCi/g each, under the Bus-10 pad (Survey Area 9227 (9522)). Most of the sampling was performed in Survey Areas 9310 and 9227 in support of Spent Fuel Pool isolation. The soil was removed until residual radioactivity was below the generic soil screening DCGLs, adjusted to 10 mrem/yr. Groundwater dose contribution was not included in this evaluation.

The subsurface soils at CY are divided into 3 classifications: Class A, Class B and Class C. Class A soils have had known contaminating events and have a high potential to be at or exceed the DCGL. Class B soils have had contaminating events or may have been impacted by events in Class A soils but are not expected to exceed the DCGL. Contamination levels in Class C soils are expected to be a small fraction of the DCGL.

The classification guides the number of measurements or samples to be taken. Class A soils will receive the highest number of samples while Class C will receive the lowest. There are no limitations on the size of the area. Subsurface soils in the Radiological Control Area, with the exception of the soil beneath Survey Area 9308 (9522), are considered as Class A (the size of this area is reflected in the dose modeling for soil). Due to large numbers of subsurface samples already collected and analyzed in Survey Area 9308 (9522), subsurface soils beneath Survey Area 9308 (9522) and in the remainder of the Industrial Area outside the RCA are considered Class B. Areas northwest and southeast of Class B are considered Class C. Figure 2-6 provides the designation and location of affected subsurface soils in the vicinity of the plant power block.

Subsurface soils contaminated above the applicable DCGLs will be removed. Groundwater dewatering wells, approved by the State of Connecticut, were installed at strategic locations to draw down groundwater as necessary to support contaminated soil and structure removal activities.

2.3.3.1.6 Groundwater

The Groundwater Monitoring Plan for compliance with the HNP License Termination Plan contains the details of the Groundwater Monitoring Program. This plan describes the groundwater characterization efforts going forward to support groundwater monitoring for the LTP.

On November 26, 2007 (Reference 2-16), the NRC issued a Safety Evaluation Report that released the majority of the site from the 10 CFR 50 License. It provides the following conclusions regarding the groundwater monitoring program:

> "The staff reviewed CYAPCO's confirmation of groundwater compliance dated July 31, 2007, (ADAMS Accession Numbers ML072060467 and ML072060517) for the HNP. In that document, CYAPCO demonstrated license termination compliance for the groundwater at the HNP site as specified in its Revised Groundwater Monitoring Plan dated March 21, 2007, (ADAMS Accession Number ML070860743). CYAPCO's LTP allocates 0.08 milliSieverts per year (mSv/yr) (8 millirem per year (mrem/yr)), for groundwater dose.
>
> "This demonstration included the following items:
>
> - The performance of six quarters of radiological sampling data for 60 monitoring wells (December 2005 - June 2007),
> - The cumulative dose from existing groundwater at each of the monitoring wells does not exceed 0.08 mSv/yr (8 mrem/yr), and
> - The substances of concern (H-3, Sr-90, and Cs-137) have concentrations in the groundwater that are usually stable or decreasing. Variations are further discussed below.

"Results from the quarterly radiological sampling indicate that no groundwater from the 60 monitoring wells at the end of the 18 month sampling period exceeded a cumulative dose of 0.08 mSv/yr (8 mrem/yr). The largest cumulative dose from the substances of concern (SOCs) during the June 2007 sampling event was 0.0198 mSv/yr (1.98 mrem/yr) at monitoring well MW-137. The majority of this dose comes from Cs-137, which is very immobile in the saturated zone. This dose is approximately 25 percent of the 0.08 mSv/yr (8 mrem/yr) allocation.

"NRC staff agrees with CYAPCO's assessment that the current H-3 and Sr-90 plumes are related to the remediation activities at the Fuel Building during the summer and fall of 2006. The Sr-90 peaks occur later than the H-3 peaks because H-3 is more mobile (lower distribution coefficient Kd) in the groundwater than Sr-90. Also, these plumes are currently down gradient from the remediated source area and they will be likely discharged into the Connecticut River by late 2007. Finally, there is no information to support any increase in dose from Cs-137 at monitoring well MW-137 or any other well. Therefore, the cumulative dose from the SOCs should not exceed the 0.08 mSv/yr (8 mrem/yr) dose allocation for any monitoring well in the future.

"CYAPCO did not achieve a stable or downward trend in the SOCs at all the monitoring wells by the end of the 18 month sampling period. However, NRC staff determined that the upward trends that were statistically significant (5 monitoring wells for H-3 and 4 monitoring wells for Sr-90) do not represent a dose or risk concern, since the dose was well below the 0.08 mSv/yr (8 mrem/yr) dose allocation.

"The NRC has reviewed the licensee's groundwater sampling documents and analysis and agrees that the 0.08 mSv/yr (8 mrem/yr) LTP dose allocation has been met and therefore, compliance with the release criteria for groundwater has been achieved.

"CYAPCO's LTP allocated 0.02 mSv/yr (2 mrem/yr) in dose contribution from two activated subsurface concrete structure areas as future groundwater dose. By letter dated May 30, 2007, CYAPCO provided dose contribution calculations from the two subsurface concrete areas (ADAMS Accession Numbers ML071870492, ML071870494, ML071870497, ML071870499, ML071870503, ML071870504, and ML071870507). The resulting dose was 0.0158 mSv/yr (1.58 mrem/yr) for the former radiologically controlled area and 0.0023 mSv/yr (0.23 mrem/yr) for the river intake and discharge area. NRC staff reviewed CYAPCO's calculations and agrees that the 0.02 mSv/yr (2 mrem/yr) LTP dose allocation has been met."

Table 2-8
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Table 2-9
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2.3.3.2 Initial Area Classification

The current classifications for the HNP site are presented in Table 2-10 for site grounds (surface and subsurface) and structures. The majority of these classifications are historical, because the only Survey Units that remain within the 10 CFR 50 License are Survey Units 9523-0000, 9528-0000, and 9528-0004. Survey area definitions were initially established for decommissioning and were used and expanded upon during subsequent site characterization activities. Some area boundaries were redefined to make use of logical physical boundaries and site landmarks. Many areas are further subdivided into survey units. A survey unit is a physical area consisting of structures or land areas of specified size and shape for which a separate decision will be made as to whether or not residual contamination in that area exceeds the release criterion. Note that survey areas for subsurface soils include any sub-surface features that are present such as piping and drain systems. The only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004. Survey Areas 9523 and 9528 are presented on Figure 2-7.

The current decommissioning approach is generally to remove the above-grade portions of site buildings and structures. The above-grade portions had been previously identified as survey areas and had been given MARSSIM classifications as a part of the original LTP. Their survey area designations have been subsequently removed as final status activities are no longer planned. Survey units have been redefined to ensure building footprints are contained as much as possible in the redefined survey units. All previous survey units classifications have been maintained or conservatively changed to a more restrictive MARSSIM classification, e.g., Class 3 to 2 or 1.

Classification of a survey area has a minimum of two stages: (1) initial classification and (2) final classification. Initial classification is performed only once, at the time of identification of the survey unit using the information available. Final classification is performed and verified as an objective of the final status survey plan.

Although it is expected that the existing survey areas will require no modification with regard to boundaries or classification, the characterization process is iterative. When additional information is obtained during the decommissioning process through characterization surveys, remediation surveys (performed to track the effectiveness of decontamination techniques), or turnover surveys (discussed below), the data will be assessed using the DQO process to verify that the current classification of the survey area is appropriate, to guide reclassification of the survey area, and/or to guide the design of subsequent surveys. Approved site procedures govern the process of classification and mandate appropriate documentation of the classification results.

Correct classification of a survey unit is crucial for appropriate final status survey design, and the potential for making decision errors increases when a survey unit is incorrectly classified. Thus, the initial assumption for classifying a survey unit is that the survey unit contains residual radioactivity levels greater than the applicable Derived Concentration Guideline Levels (DCGLs) and, thus is a Class 1 area. Available information is subsequently used to support classification of an area or unit as Class 2, Class 3, or non-impacted. Approved site procedures guide this determination process and mandate appropriate documentation for the determinations made.

The definitions of Class 1, 2, and 3 (per MARSIM) are as follows:

The DCGL established for the average residual contamination in a survey unit is $DCGL_W$.

- Class 1 Areas: Areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiation surveys) above the $DCGL_W$. Examples of Class 1 areas include: 1) site areas previously subjected to remedial actions, 2) locations where leaks or spills are known to have occurred, 3) former burial or disposal sites, 4) waste storage sites, and 5) areas with contaminants in discrete solid pieces of material and high specific activity.

- Class 2 Areas: Areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the $DCGL_W$. To justify changing the classification from Class 1 to Class 2, there should be measurement data that provides a high degree of confidence that no individual measurement would exceed the $DCGL_W$. Other justifications for reclassifying an area as Class 2 may be appropriate based on site-specific considerations. Examples of areas that may be classified as Class 2 for the final status survey include: 1) locations where radioactive materials were present in an unsealed form, 2) potentially contaminated transport routes, 3) areas downwind from stack release points, 4) upper walls and ceilings of buildings or rooms subjected to airborne radioactivity, 5) areas handling low concentrations of radioactive materials, and 6) areas on the perimeter of former contamination control areas.

- Class 3 Areas: Any impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_W$, based on site operating history and previous radiation surveys. Examples of areas that might be classified as Class 3 include buffer zones around Class 1 or Class 2 areas, and areas with very low potential for residual contamination but insufficient information to justify a non-impacted classification.

Class 1 areas have the greatest potential for contamination and therefore receive the highest degree of survey effort for the final status survey, followed by Class 2 areas, and then by Class 3 areas. Non-impacted areas do not require any level of survey coverage because they have no potential for residual contamination. As a survey progresses, reevaluation of classifications may be necessary based on newly acquired survey data. The final status survey plan includes a process by which measurements that approach investigation levels, defined as fractions of the DCGLs (and discussed further in Section 5.5.3), are reviewed to see if reclassification of an area(s) is necessary.

The classifications, provided in Table 2-10, were conservatively chosen based upon the review of a large volume of historical radiological survey data (routine and non-routine), collected over the plant's operating history; a review of the historical information maintained under 10 CFR 50.75(g); and a review of the scoping survey information compiled for the decommissioning of the HNP site. The radiological survey information in particular represents a substantial volume of information.

A supplement to the original Historical Site Assessment (HSA) was developed to provide more detailed information on the radiological status of structures and land at the Haddam Neck Plant. This supplement provides the additional information that was used to establish the initial MARSSIM classifications of survey areas and survey units as shown in LTP Table 2-10.

The HSA Supplement (and LTP Table 2-11B) summarizes a comprehensive review of numerous plant records such as plant incident reports, condition reports, investigations, radiological surveys (1967 to 2000), annual effluent reports, interviews with past and present employees and site walk-downs.

In all cases, radiological survey data was obtained by trained technicians using calibrated instrumentation. Since the data set spans a large period of time (33 yrs), the procedures used in the calibration and other quality control processes have likely changed. However, most of the data presented in Table 2-11B is a result of surveys conducted within a 5-year period where the calibration and quality control processes were consistent. All instruments were calibrated against NIST traceable standards and had daily operability checks and in the case of laboratory sample analysis participate in third-party quality control performance testing. In all cases however, radiological surveys and laboratory sample results have been generated by trained technicians and reviewed and approved by supervision. These processes ensure the validity of the collected data. In general, the table is a summary of numerous routine and decommissioning support surveys. As a result, some radiation data in the table (e.g., <0.2 or <1 mR/hr) is limited due to the instrumentation used for the surveys. This is also the reason for the large standard deviations in contamination levels. Between 1997 and 1999 limited site characterization surveys were performed. Results of samples from these surveys are listed in the in the Concentration columns of Table 2-11B.

The survey of many inaccessible or not readily accessible areas or surfaces was deferred. Examples of areas where surveys were deferred include soils under structures, contaminated sumps, pipe trenches, and the Containment dome. The decision to defer the survey of an area was based on one or more of the following conditions:

- ALARA considerations (e.g., the area is either a high radiation or high contamination area and additional data would likely not change the survey area or unit MARSSIM classification of the location or surrounding areas),
- safety considerations (e.g., difficulty of access to the upper reaches of the Containment dome due to height above the charging floor),
- historical data shows that the area could be classified without further characterization,
- access for characterization would require significant deconstruction of adjacent systems, structures or other obstacles the removal of which could result in an unsafe condition or interfere with continued operation of required components, or
- the ability to use engineering judgement in assigning the area a MARSSIM classification based on physical relationship to surrounding areas and the likelihood of the area to have radiological conditions represented by the conditions in these adjacent areas.

As access is gained to areas that were previously inaccessible, additional characterization data will be collected, evaluated and stored with other radiological survey data in a survey history file for the survey unit. Sampling for this additional characterization data will be chosen to include several locations such as cracks, floors and walls to establish the variability and extent of the contamination. This data will be used to establish the radionuclides present and variability in the radionuclide mix for both Easy-To-Detect (ETD) and Hard-To-Detect (HTD) radionuclides. In addition, as the decommissioning progresses, data from operational events caused by equipment failures or personnel errors, which may affect the radiological status of survey unit(s), will be captured by CYAPCO's Corrective Action Program. These events will be evaluated and, when appropriate, stored in the 50.75(g) database. This additional characterization data will be used in validating the initial classification and in planning for the final status survey of each survey unit.

As decommissioning proceeds, areas will, as necessary, be decontaminated to remove loose surface decontamination (as well as fixed contamination) to levels that will meet the conditions for controlled demolition or unrestricted release conditions for demolition. When an SSC is ready for demolition, a documented formal turnover from CYAPCO to the demolition contractor is made for access and control of the area. Following the demolition and when an area is believed to be ready for final status survey, a "turnover assessment" will be performed. If the results of this assessment indicate that the Final Status Survey acceptance criteria will be met, then physical and administrative control of the area will be transferred to Final Status Survey personnel for preparation, design, and performance of the FSS. Otherwise, additional remediation may be required. This assessment may include a "turnover survey," primarily for Class 1 and 2 areas within the Industrial Area and in land areas outside of the industrial area, that are impacted by existing groundwater contamination. This "turnover survey" process will

include a MARISSM-type survey in which a combination of scanning and fixed measurements will be obtained and evaluated against the FSS criteria for the survey unit. If the results of this survey indicate that the FSS acceptance criteria will be met, then physical and administrative control of the area will be transferred to Final Status Survey personnel for preparation, design, and performance of the FSS. Otherwise, additional remediation may be required. The "turnover survey" process together with any additional characterization and remediation survey performed, represent at least one, but possibly several, opportunities to collect additional survey data prior to conducting the FSS. For each survey type (characterization, remediation, turnover, and final status) a documented survey plan will be 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 to defining data quality requirements. For example, scoping and characterization survey plans intended to collect data may only require a survey objective and the instrumentation and analyses specifications necessary to met that survey objective. Remediation and final status 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. These survey plans will contain the appropriate data assessment to ensure that several objectives are met. These objectives include:

- Appropriate instrument selection to ensure the proper sensitivity relative to the applicable DCGLs,
- Appropriate instrument quality control measures to ensure operability,
- Appropriate survey techniques, as described in NUREG-1507, to ensure that the field measurement techniques are consistent with the calibration methodologies,
- Appropriate sample collection and analysis to determine spatial variability and variability in radionuclide ratios,
- Data analysis criteria to identify follow-up actions such as remediation and the collection of additional samples, and
- Appropriate classification of the survey area.

Table 2-10
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	**Survey Area Code Description**	**Survey Unit Code**	**Survey Unit Code Description**	**MARSSIM Class.**	**Floor Area**	**Total Area**	**Ratio (Total Area: Floor Area)**	**Figure No.**
9106	Discharge Canal	0001	Sediment to Ave. High Water Level	1	N/A	1,917	N/A	2-2
		0002	Sediment to Ave. High Water Level	2	N/A	5,520	N/A	2-2 2-3
		0003	Sediment to Ave. High Water Level	2	N/A	8,292	N/A	2-3
		0004	Sediment to Ave. High Water Level	2	N/A	9,900	N/A	2-3
		0005	Sediment to Ave. High Water Level	2	N/A	9,632	N/A	2-4
		0006	Sediment to Ave. High Water Level	2	N/A	9,716	N/A	2-4 2-5
		0007	Sediment to Ave. High Water Level	2	N/A	8,692	N/A	2-5

Table 2-10
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Class.	Floor Area	Total Area	Ratio (Total Area: Floor Area)	Figure No.
		0008	Sediment to Ave. High Water Level	2	N/A	9,763	N/A	2-5
		0009	Sediment to Ave. High Water Level	2	N/A	9,933	N/A	2-5
		0010	Sediment to Ave. High Water Level	2	N/A	9,512	N/A	2-5
		0011	Sediment to Ave. High Water Level	2	N/A	6,394	N/A	2-5
		0012	Peninsula Wetlands	2	N/A	7,272	N/A	2-4
		0013	Peninsula Wetlands	2	N/A	9,011	N/A	2-5
		0014	Canal Sediment	1	N/A	1,870	N/A	2-2
		0015	Canal Sediment Within SU 9106-007	1	N/A	1,170	N/A	2-5

Table 2-10
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	**Survey Area Code Description**	**Survey Unit Code**	**Survey Unit Code Description**	**MARSSIM Class.**	**Floor Area**	**Total Area**	**Ratio (Total Area: Floor Area)**	**Figure No.**
9302	Northwest Protected Area Grounds	0000	Land Area	3	N/A	3,490	N/A	2-2
9304	Southwest Protected Area Grounds	0001	Land Area	3	N/A	4,800	N/A	2-2
		0002	Land Area	1	N/A	1,268	N/A	2-2
9306	South Central Protected Area Grounds	0000	Land Area	2	N/A	5,878	N/A	2-2
9312	Northeast Protected Area Grounds (Former Radiologically Controlled Area)	0001	Containment	1	N/A	1,975	N/A	2-2
		0002	Spent Fuel Bldg	1	N/A	1,975	N/A	2-2
		0003	Southwest 115kV Switchyard	1	N/A	1,486	N/A	2-2

Table 2-10
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	**Survey Area Code Description**	**Survey Unit Code**	**Survey Unit Code Description**	**MARSSIM Class.**	**Floor Area**	**Total Area**	**Ratio (Total Area: Floor Area)**	**Figure No.**
9312		0004	North of the PAB and Tank Farm	1	N/A	1,586	N/A	2-2
		0005	Tank Farm	1	N/A	1,460	N/A	2-2
		0006	PAB Footprint	1	N/A	1,754	N/A	2-2
		0007	RRF Bldg/115kV East Footprint	1	N/A	1,566	N/A	2-2
		0008	115kV West Side GW Treatment Facility	1	N/A	1,351	N/A	2-2
		0009	East Trench North	1	N/A	1,511	N/A	2-2
		0010	East Trench South/MWST A&B	1	N/A	1,365	N/A	2-2

Table 2-10
MARSSIM Classifications

Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Class.	Area (m^2) Floor Area	Area (m^2) Total Area	Ratio (Total Area: Floor Area)	Figure No.
9313	Central Site Grounds	0000	Land Area	2	N/A	8,266	N/A	2-2
9504	Bypass Road and Secondary Parking lot (Add N. Portion of 9502) (Info Only)	0000	Land Area	2	N/A	5,692	N/A	2-1
9506	North Site Grounds (Non-Protected Area) (Add EOF Parking Area from 9510) (Info Only)	0000	Land Area	3	N/A	11,244	N/A	2-1
9508	Pond	0000	Land Area and Pond Sediment	3	N/A	10,831	N/A	2-1
9512	Northwest Site Grounds (Non-Protected Area) (Remove Truck Monitor and Roadway) (Info Only)	0000	Land Area	3	N/A	30,701	N/A	2-1
9514	Primary Parking Lot West (Add the S. Portion of 9502) Add Truck Monitor and Roadway From 9512) (Info Only)	0000	Land Area	3	N/A	18,757	N/A	2-1
	Primary Parking Lot East	0001	Land Area	2	N/A	8,739	N/A	2-1

Table 2-10
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Class.	Floor Area	Total Area	Ratio (Total Area: Floor Area)	Figure No.
9520	Southwest Site Storage Area	0001	Land Area, North Peninsula	2	N/A	8,062	N/A	2-2
		0002	Land Area, North Peninsula	2	N/A	9,845	N/A	2-2 2-3
		0003	Land Area, North Peninsula	2	N/A	8,106	N/A	2-3
		0004	Land Area, North Peninsula	1	N/A	1,983	N/A	2-3
		0005	Land Area, North Peninsula	1	N/A	1,887	N/A	2-3
		0006	Land Area, North Peninsula	1	N/A	1,808	N/A	2-2
9521	Southeast Pond	0000	Land Area and Pond Sediment	3	N/A	23,100	N/A	2-7

Table 2-10
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	**Survey Area Code Description**	**Survey Unit Code**	**Survey Unit Code Description**	**MARSSIM Class.**	**Floor Area**	**Total Area**	**Ratio (Total Area: Floor Area)**	**Figure No.**
9522	Southeast Site Grounds	0001	Land Area	2	N/A	6,972	N/A	2-2
		0002	Land Area	1	N/A	1,913	N/A	2-2
		0003	Land Area 150m East of Load distribution Tower	1	N/A	1,997	N/A	2-2
		0004	Land Area	1	N/A	1,386	N/A	2-2
		0005	East Side Land of Canal	1	N/A	1,947	N/A	2-2
		0006	South RCA Land Area	1	N/A	1,987	N/A	2-2
		0007	South RCA Land Area	1	N/A	1,952	N/A	2-2

Table 2-10
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	**Survey Area Code Description**	**Survey Unit Code**	**Survey Unit Code Description**	**MARSSIM Class.**	**Floor Area**	**Total Area**	**Ratio (Total Area: Floor Area)**	**Figure No.**
9523	Southeast Wetland Area	0000	Land Area	3	N/A	106,000	N/A	2-7
9524	South Side Grounds (Non-Protected Area)	0000	Land Area	3	N/A	110,000	N/A	2-7
9525	Southeast Site Road	0000	Land Area	3	N/A	28,000	N/A	2-7
9526	Northeast Mountain Side	0000	Land Area	3	N/A	44,700	N/A	2-7
		0001	Land Area	2	NA	6,536	N/A	2-7
		0002	Land Area	2	N/A	6,069	N/A	2-7
9527	East Mountain Side	0001	Land Area	2	N/A	8,600	N/A	2-7
		002	Land Area	2	N/A	9,740	N/A	2-7

Table 2-10
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Class.	Floor Area	Total Area	Ratio (Total Area: Floor Area)	Figure No.
		0003	Land Area	2	N/A	8,200	N/A	2-7
		0004	Land Area	2	N/A	5,400	N/A	2-7
		0005	Land Area	2	N/A	5,777	N/A	2-2
		0006	Land Area	1	N/A	790	N/A	2-2
9528	Southeast Mountain Side	0000	Land Area	3	N/A	575,488	N/A	2-7
		0002	Land Area	2	N/A	9,752	N/A	2-7
		0003	Land Area	2	N/A	9,447	N/A	2-7

Table 2-10
MARSSIM Classifications

					Area (m²)			
Survey Area Code	**Survey Area Code Description**	**Survey Unit Code**	**Survey Unit Code Description**	**MARSSIM Class.**	**Floor Area**	**Total Area**	**Ratio (Total Area: Floor Area)**	**Figure No.**
		0004	Land Area	2	N/A	3,058	N/A	2-7
9530	Central Peninsula Area	0001	Land Area Bounded by and Immediately Adjacent to the Road	2	N/A	8,000	N/A	2-3 2-4
		0002	Western Half of Diked Area and Immediately Surrounding Sides	2	N/A	5,000	N/A	2-4
		0003	Eastern Half of Diked Area and Immediately Surrounding Sides	2	N/A	5,000	N/A	2-4
		0004	Open Land Areas	3	N/A	97,000	N/A	2-4
9531	South End of Peninsula	0000	Land Area	3	N/A	118,000	N/A	2-5
9532	East Side Grounds (Non-Protected Area)	N/A	N/A	Non-impacted	N/A	375,600	N/A	2-7

Table 2-10
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	**Survey Area Code Description**	**Survey Unit Code**	**Survey Unit Code Description**	**MARSSIM Class.**	**Floor Area**	**Total Area**	**Ratio (Total Area: Floor Area)**	**Figure No.**
9535	South East Landfill Area	0001	Land Area	1	N/A	1,900	N/A	2-7
		0002	Land Area	2	N/A	3,600	N/A	2-7
9536	Construction Piles Near Rifle Range	0000	Land Area	2	N/A	2,200	N/A	2-7
9537	Permitted Landfill Area	0000	Land Area	2	N/A	2,200	N/A	2-7
9538	Material Storage Area	0000	Land Area	2	N/A	4,200	N/A	2-7
9539	ISFSI Haul Road	0001	ISFSI Haul Road (Northern Section)	2	N/A	9,332	N/A	2-3 2-4
		0002	ISFSI Haul Road (Southern Section)	2	N/A	7,239	N/A	2-4

Table 2-10
MARSSIM Classifications

					Area (m²)			
Survey Area Code	**Survey Area Code Description**	**Survey Unit Code**	**Survey Unit Code Description**	**MARSSIM Class.**	**Floor Area**	**Total Area**	**Ratio (Total Area: Floor Area)**	**Figure No.**
9801	Subsurface soils in Radiologically Controlled Area	0000	Subsurface Soil	A*	N/A	15,788	N/A	2-6
9802	Subsurface area associated with the West Industrial Site Grounds (Non-Protected Area)	0000	Subsurface Soil	B*	N/A	18,292	N/A	2-6
9803	Subsurface area associated with the North Grounds (Non-Protected Area)	0000	Subsurface Soil	C*	N/A	35,485	N/A	2-6
9804	Subsurface area associated with Southeast Grounds (Non-Protected Area)	0000	Subsurface Soil	A*	N/A	11,568	N/A	2-6
9805	Subsurface soils associated with peninsula (excluding area 9531)	0000	Subsurface Soil	C*	N/A	104,207	N/A	2-6
9806	Subsurface soils associated with surface soil portions of survey area 9535 (South East Landfill)	0000	Subsurface Soil	A*	N/A	4250	N/A	2-6
9807	Subsurface Soils associated with 9520-0004	0000	Subsurface Soil	B*	N/A	1983	N/A	2-6

*MARISSM does not cover media such as subsurface soil, which is considered beyond its scope. LTP Section 5.7.3.2.1 discusses the criteria applied during the classification of subsurface soils.

Table 2-11A
Deleted

Table 2-11B
Nominal Radiological Data
Supporting Classifications for Land Areas

Survey Area Code	Survey Area Code Description	Radiation Levels		Sample analysis Results			
		(mR/hr)		Concentration (pCi/g)			Comments
		Min	Max	Medium	Nuclide	Max	
9523	Southeast Wetland Area						
9528	Southeast Mountain Side			Soil	Cs-137	1.898	

Table 2-11C
Deleted

2.3.3.3 Non-Impacted Area Assessment

Non-impacted areas are those areas having no reasonable potential for residual contamination. Non-impacted areas are typically identified during initial classification using historical data and past or current radiological surveillance. Non-impacted areas should have no history of using, storing, or burying radioactive materials. Records and surveillances, including those required by 10CFR50.75(g)(1), should show that unplanned liquid releases, discharges and other occurrences have not resulted in the spread of contamination in these areas.

The Connecticut Yankee Haddam Neck Characterization Report has classified the East Site Grounds (Survey Area 9532) as non-impacted. This area consisted of approximately ninety-three (93) acres of uninhabited, undeveloped land located about a third of a mile (0.29 miles) from the RCA (Radiologically Controlled Area).

In a letter dated April 29, 2004, CYAPCO provided written notification to the NRC of the intent to release the East Site Grounds (Survey Area 9532) from its Part 50 license. The NRC approved the release of the Survey Area from the 10 CFR 50 License in a letter dated September 1, 2004 (Reference 2-14).

2.3.3.4 Radionuclide Suite Selection

As a part of site characterization, the list of radionuclides expected to be encountered during decommissioning was established. The radionuclides listed in Table 2-12 were identified using available waste characterization data from the HNP site.

Table 2-12
Radionuclides Potentially Present at HNP

Radionuclide	Half-life[a] (years)
H-3	**12.33**
C-14	**5,730**
Mn-54	0.8561
Fe-55	**2.685**
Co-60	5.271
Ni-63	**100**
Sr-90	**28.8**
Nb-94	2.0×10^{4}
Tc-99	**2.14×10^{5}**
Ag-108m	1.27×10^{2}
Cs-134	2.062
Cs-137	30.17
Eu-152	13.3
Eu-154	8.5
Eu-155	4.96
Pu-238	**87.74**
Pu-239*	**2.41×10^{4}**
Pu-241	**14.4**
Am-241	432.2
Cm-243*	**28.5**

* One DCGL was established for the Pu-239/240 pair and the Cm-243/244 pair, as laboratory radiochemical analyses do not report concentrations of these radionuclides separately. **Bold** indicates those radionuclides that are considered to be hard to detect.

Available regulatory documents, addressing radionuclides existing in bio-shield wall concrete and rebar and surface contamination, were reviewed in order to determine if other dose-significant radionuclides existed and should be included. NUREG/CR-3474, "Long-Lived Activation Products in Reactor Material" (Reference 2-9), Tables 5.4 and 5.6, and NUREG/CR-0130, "Technology, Safety and Cost of Decommissioning a Reference Pressurized Water Reactor Power Station" (Reference 2-10), Tables 7.3-5, 7.3-11 and 7.3-14, were used in this determination. A number of the radionuclides listed in the regulatory guidance have been found to be not significant to HNP based upon their extremely low dose contribution. The methodology used to reach that conclusion follows.

The radionuclide inventory values, from the NUREGs referenced above, were used to determine the activity ratio of a given nuclide to that of Co-60. Co-60 was used as a reference nuclide, since it is a predominant beta/gamma emitting nuclide at the HNP. For situations in which a radionuclide was included more than once in the references, the highest ratio was selected as being representative of the relative activity for that nuclide. This ratio (representing activity at the time of reactor shutdown) was adjusted to represent decay to the current period (i.e., April 1, 2002). Doses, relative to Co-60, were calculated for the following exposure pathways:

- Inhalation Exposure
- Ingestion Exposure
- External Exposure to a Plane Source
- External Exposure to an Infinitely Thick Soil Contamination Source

These relative doses were calculated by multiplying the applicable dose conversion factors (from Federal Guidance Reports FGR-11 and FGR-12) (References 2-12 and 2-13) by the relative decayed activity. Table 2-13 provides a summary of the relative doses from each of the pathways, the maximum relative dose for each nuclide, and dose fraction relative to the total dose for each nuclide (Reference 2-11). This table is sorted in descending order by the dose fraction for each radionuclide relative to the total dose.

Table 2-13
Summary of Radionuclide Analysis

	Fraction of Co-60 Dose				Maximum Pathway	
Radionuclide	Inhalation	Ingestion	Plane	Infinite	Max Dose Relative to Co-60	Fraction of Total Dose
Cs-137	2.72E+00	3.46E+01	2.26E-03	8.64E-04	3.46E+01	6.48E-01
Am-241	4.28E+00	2.85E-01	2.47E-05	5.69E-06	4.28E+00	8.03E-02
Pu-238	3.65E+00	2.42E-01	7.26E-07	1.90E-08	3.65E+00	6.84E-02
Cm-243	2.60E+00	1.73E-01	9.85E-05	6.66E-05	2.60E+00	4.88E-02
Sr-90	2.21E+00	1.96E+00	4.49E-05	1.61E-05	2.21E+00	4.14E-02
Eu-152	1.74E+00	4.14E-01	8.06E-01	7.44E-01	1.74E+00	3.26E-02
Cs-134	1.04E-01	1.34E+00	3.19E-01	2.88E-01	1.34E+00	2.52E-02
Co-60	1.25E-11	3.35E-11	1.02E-10	9.26E-11	1.00E+00	1.87E-02
Pu-239	7.60E-01	5.08E-02	6.05E-08	7.05E-09	7.60E-01	1.42E-02
Fe-55	2.69E-01	4.93E-01	0.00E+00	0.00E+00	4.93E-01	9.25E-03
Eu-154	4.51E-01	1.22E-01	1.75E-01	1.63E-01	4.51E-01	8.45E-03
H-3	1.03E-02	8.33E-02	0.00E+00	0.00E+00	8.33E-02	1.56E-03
Pu-241	6.09E-02	4.10E-03	1.33E-09	5.88E-10	6.09E-02	1.14E-03
Hf-178m	3.26E-02	2.26E-03	2.84E-03	2.32E-03	3.26E-02	6.11E-04
U-233	1.68E-02	2.91E-04	8.26E-09	2.34E-09	1.68E-02	3.15E-04
Ni-63	1.23E-02	9.16E-03	0.00E+00	0.00E+00	1.23E-02	2.31E-04
Na-22	3.85E-04	4.68E-03	9.81E-03	9.26E-03	9.81E-03	1.84E-04
Ca-41	7.15E-04	5.49E-03	0.00E+00	0.00E+00	5.49E-03	1.03E-04
Sm-151	1.78E-03	1.87E-04	2.78E-08	7.87E-10	1.78E-03	3.33E-05
Mn-54	1.56E-04	5.22E-04	1.75E-03	1.61E-03	1.75E-03	3.29E-05
Ba-133	3.10E-04	1.10E-03	1.47E-03	1.06E-03	1.47E-03	2.75E-05
Eu-155	1.40E-03	4.18E-04	1.85E-04	8.28E-05	1.40E-03	2.62E-05
C-14	1.57E-04	1.27E-03	1.13E-07	1.36E-08	1.27E-03	2.39E-05
Ho-166m	1.13E-03	9.53E-05	2.30E-04	2.02E-04	1.13E-03	2.11E-05
Zn-65	5.09E-05	2.92E-04	1.28E-04	1.24E-04	2.92E-04	5.48E-06
Sb-125	5.64E-05	1.05E-04	1.83E-04	1.53E-04	1.83E-04	3.43E-06
Ca-45	4.68E-05	1.81E-04	3.03E-08	5.96E-09	1.81E-04	3.40E-06
Nb-94	1.38E-04	1.92E-05	4.73E-05	4.33E-05	1.38E-04	2.58E-06
Cl-36	9.52E-05	1.07E-04	2.72E-07	1.40E-07	1.07E-04	2.00E-06
Ag-108m	5.98E-05	1.31E-05	3.14E-05	2.74E-05	5.98E-05	1.12E-06
Ni-59	4.69E-05	2.95E-05	0.00E+00	0.00E+00	4.69E-05	8.78E-07
Ce-144	3.81E-05	1.74E-05	1.93E-07	9.86E-08	3.81E-05	7.14E-07
Mo-93	2.76E-05	1.06E-05	4.83E-07	7.74E-09	2.76E-05	5.18E-07
Ag-110m	5.78E-06	6.31E-06	1.78E-05	1.67E-05	1.78E-05	3.33E-07
Nb-93m	8.79E-06	1.27E-06	2.63E-08	4.22E-10	8.79E-06	1.65E-07
Tc-99	5.97E-06	8.51E-06	5.21E-09	1.21E-09	8.51E-06	1.60E-07
Pm-145	6.35E-06	8.02E-07	6.33E-07	8.25E-08	6.35E-06	1.19E-07

Table 2-13
Summary of Radionuclide Analysis

	Fraction of Co-60 Dose				Maximum Pathway	
Radionuclide	Inhalation	Ingestion	Plane	Infinite	Max Dose Relative to Co-60	Fraction of Total Dose
Tb-158	2.42E-06	3.39E-07	6.81E-07	6.04E-07	2.42E-06	4.55E-08
Co-57	6.15E-07	6.52E-07	7.26E-07	4.58E-07	7.26E-07	1.36E-08
Zr-93	4.26E-07	1.79E-08	0.00E+00	0.00E+00	4.26E-07	7.99E-09
Sn-121m	3.08E-07	3.37E-07	1.22E-08	7.08E-10	3.37E-07	6.31E-09
Se-79	1.00E-09	7.19E-09	1.96E-13	2.55E-14	7.19E-09	1.35E-10
Cs-135	1.91E-10	2.41E-09	1.30E-13	2.17E-14	2.41E-09	4.52E-11
Sm-146	1.42E-09	2.86E-11	0.00E+00	0.00E+00	1.42E-09	2.67E-11
I-129	1.08E-10	1.39E-09	1.49E-12	1.08E-13	1.39E-09	2.60E-11
Mn-53	6.15E-10	1.08E-09	0.00E+00	0.00E+00	1.08E-09	2.03E-11
Pb-205	4.86E-11	1.64E-10	1.73E-13	1.18E-15	1.64E-10	3.08E-12
S-35	6.75E-11	1.62E-10	4.27E-14	5.48E-15	1.62E-10	3.04E-12
Co-58	1.25E-11	3.35E-11	1.02E-10	9.26E-11	1.02E-10	1.91E-12
Sc-46	2.32E-12	4.06E-12	1.40E-11	1.34E-11	1.40E-11	2.63E-13
Zr-95	3.46E-13	4.49E-13	9.86E-13	8.93E-13	9.86E-13	1.85E-14
Sb-124	1.07E-14	3.49E-14	6.76E-14	6.70E-14	6.76E-14	1.27E-15
Fe-59	1.38E-15	5.06E-15	9.70E-15	9.59E-15	9.70E-15	1.82E-16
Sr-89	1.97E-15	3.57E-15	1.00E-17	5.82E-18	3.57E-15	6.70E-17
Ru-103	7.26E-21	2.01E-20	3.49E-20	3.00E-20	3.49E-20	6.55E-22
Nb-95	6.72E-22	2.42E-21	8.06E-21	7.32E-21	8.06E-21	1.51E-22
Te-129m	1.49E-22	5.41E-22	2.19E-23	1.56E-23	5.41E-22	1.01E-23
Ce-141	1.09E-25	2.87E-25	8.37E-26	5.22E-26	2.87E-25	5.38E-27
Cr-51	7.77E-27	2.78E-26	6.66E-26	5.47E-26	6.66E-26	1.25E-27
P-33	1.43E-27	4.60E-27	2.56E-30	4.92E-31	4.60E-27	8.62E-29
Cs-136	5.93E-52	7.39E-51	1.57E-50	1.45E-50	1.57E-50	2.95E-52
Ba-140	2.57E-54	5.29E-53	1.15E-53	9.56E-54	5.29E-53	9.91E-55
Y-90	1.60E-239	1.66E-238	9.41E-241	6.13E-241	1.66E-238	3.11E-240
La-140	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

The data provided in Table 2-13 allows the selection of significant nuclides based on both the expected relative abundance and the dose potential for each nuclide. The criterion for considering a radionuclide as being dose-significant was set at a fraction of total dose significance of 0.1% or greater. Although they do not meet the criteria for selection based upon dose significance, C-14, Mn-54, Ni-63, Nb-94, Tc-99, Ag-108m and Eu-155, listed in Table 2-12, are considered important for inclusion for HNP, as these radionuclides have been reported in plant waste streams by easy-to-detect methods (i.e., gamma spectroscopy) or are anticipated as a result of activation.

Based upon this information, no additional radionuclides were selected for inclusion, and the list of radionuclides in Table 2-12 is considered to be the list of radionuclides of concern for HNP.

2.3.3.5 Hazardous Material Status

The characterization process included the identification of hazardous materials and State of Connecticut regulated materials. The characterization process coupled the radiological and hazardous material evaluations such that the resultant characterization report for each area included an assessment of materials known to be present as well as those where further analyses are needed to fully define the existence and scope of materials present. The hazardous material characterization effort used the same site procedure, following the methodology described in Section 2.3.2. As indicated in that section, a critical element of the characterization effort included a walk-down of each area by a professional experienced in hazardous and state-regulated materials.

The review of historical records and the familiarity of personnel performing the characterization with plant operations identified that the major hazardous materials encountered at HNP are asbestos, lead, PCBs and mercury. These materials are typically contained in building materials, paints, light bulbs, light fixtures, switches, electrical components and high voltage cables. In addition to the above materials, temporary RCRA waste storage areas were maintained on site in compliance with federal requirements. These storage areas are identified in the area characterization reports, with further evaluation required to determine the extent, if any, of hazardous material contamination in those areas. An example of a building containing a RCRA waste storage area (90 day storage) was the North Warehouse, Building No. 160.

Full details of hazardous and state regulated materials identified in each survey area, and the additional actions and evaluations necessary to ensure the appropriate definition of the extent of the hazardous materials is presented in the "Connecticut Yankee Haddam Neck Plant Characterization Report," dated January 6, 2000.

2.4 References

2-1. NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," dated December 1997.

2-2. "Connecticut Yankee Haddam Neck Plant Characterization Report," dated January 6, 2000.

2-3. "Haddam Neck Plant Historical Site Assessment Supplement," dated August 14, 2001.

2-4 Investigation of the Source of Radioactive Contamination Found on the Connecticut Yankee Site March 10-30, 1980, dated April 1980.

2-5 Results of Phase 2 PCB and Radiological Characterization Study, CY Letter HP-98-423, dated July 28, 1998.

2-6 Executive Summary of Radiation Surveys Performed at Connecticut Yankee Atomic Power Station, dated January 22, 1998, Millennium Services, Inc.

2-7 Groundwater Monitoring Report, Connecticut Yankee Atomic Power Station Haddam Neck, Connecticut, by Malcom Pirney, Inc., dated July 1999, revised September 1999.

2-8 "Haddam Neck Plant Phase 2 Hydrogeologic Investigation Work Plan," May 2002.

2-9 NUREG/CR-3474, "Long-Lived Activation Products in Reactor Materials, dated August 1984.

2-10 NUREG/CR-0130, "Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station," June 1978.

2-11 Technical Support Document, BCY-HP-0023, Revision 1, "Radionuclide List for DCGL Calculation in Support of the LTP."

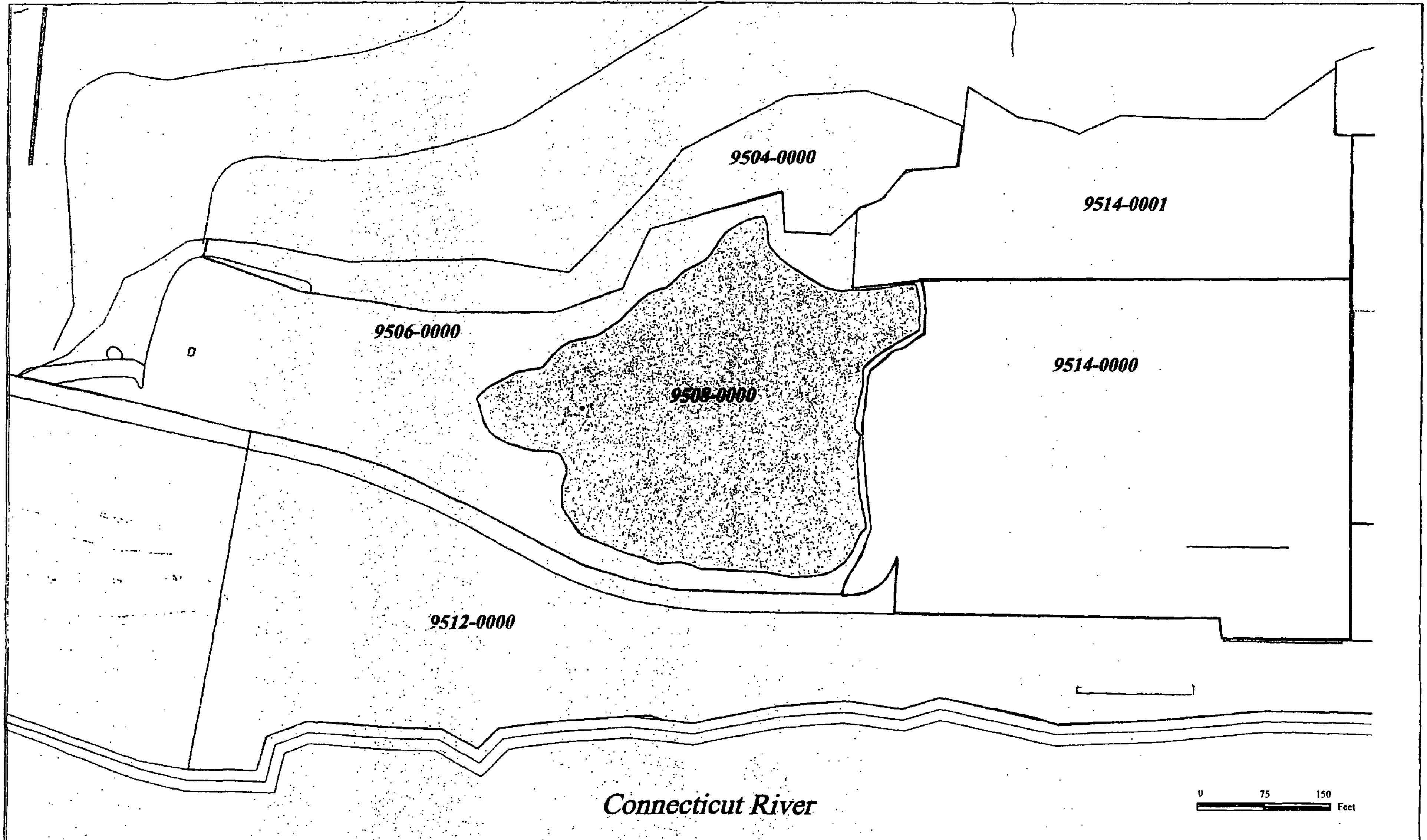
2-12 Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", 1988.

2-13 Federal Guidance Report 12, "External Exposures to Radionuclides in Air, Water and Soil", 1993.

2-14 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of East Site Grounds from Part 50 License," dated September 1, 2004.

2-15 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of Phase II Areas from Part 50 License," dated February 27, 2006.

2-16 Letter from K. McConnell (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant - Release of Land from Part 50 License," dated November 26, 2007.



Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY

CY UPDATE SURVEY UNIT BOUNDARIES

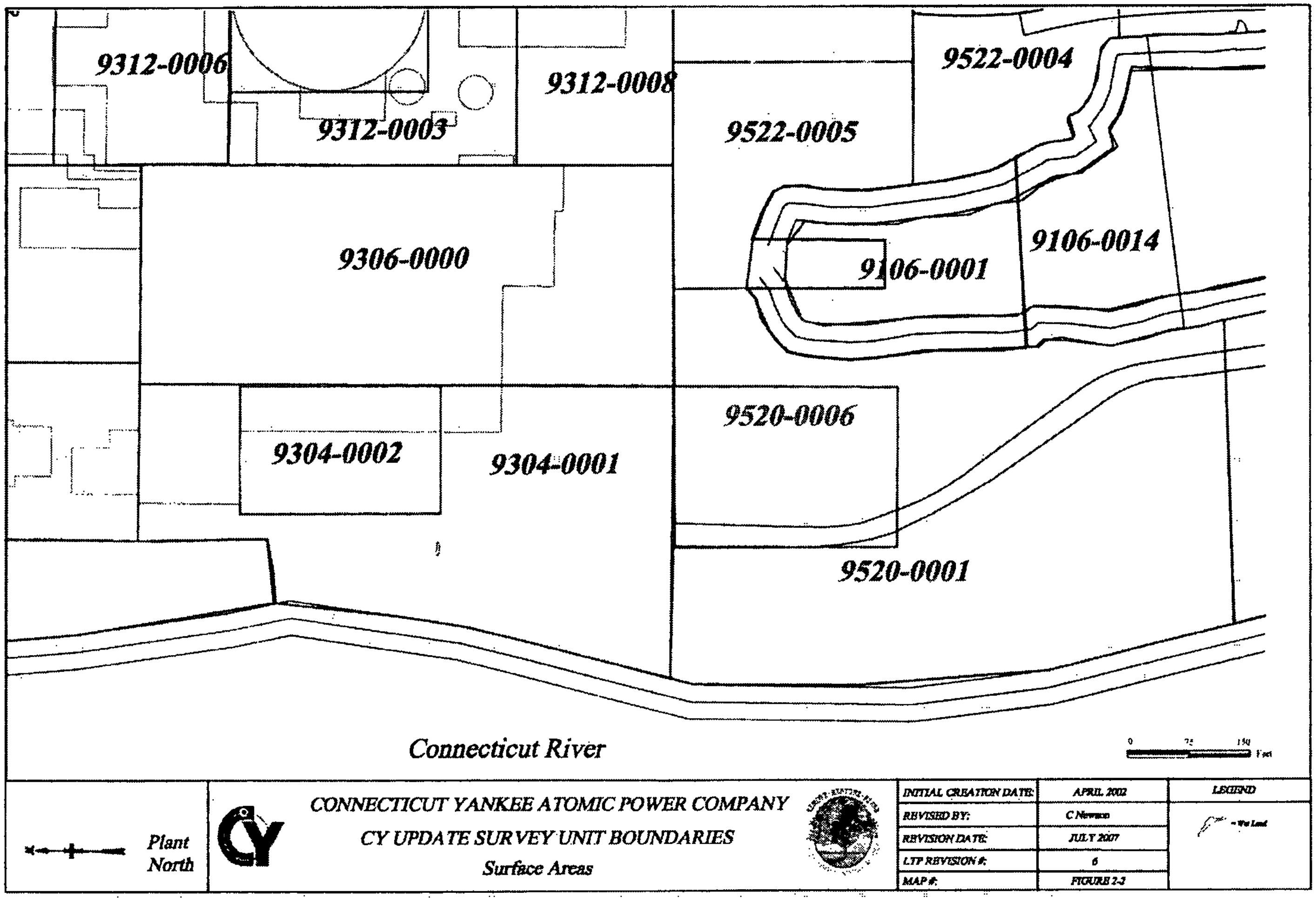
Surface Areas

INITIAL CREATION DATE:	APRIL 2002
REVISED BY:	J McCARTHY
REVISION DATE:	FEBRUARY 2007
LTP REVISION #:	5
MAP #:	FIGURE 2-1

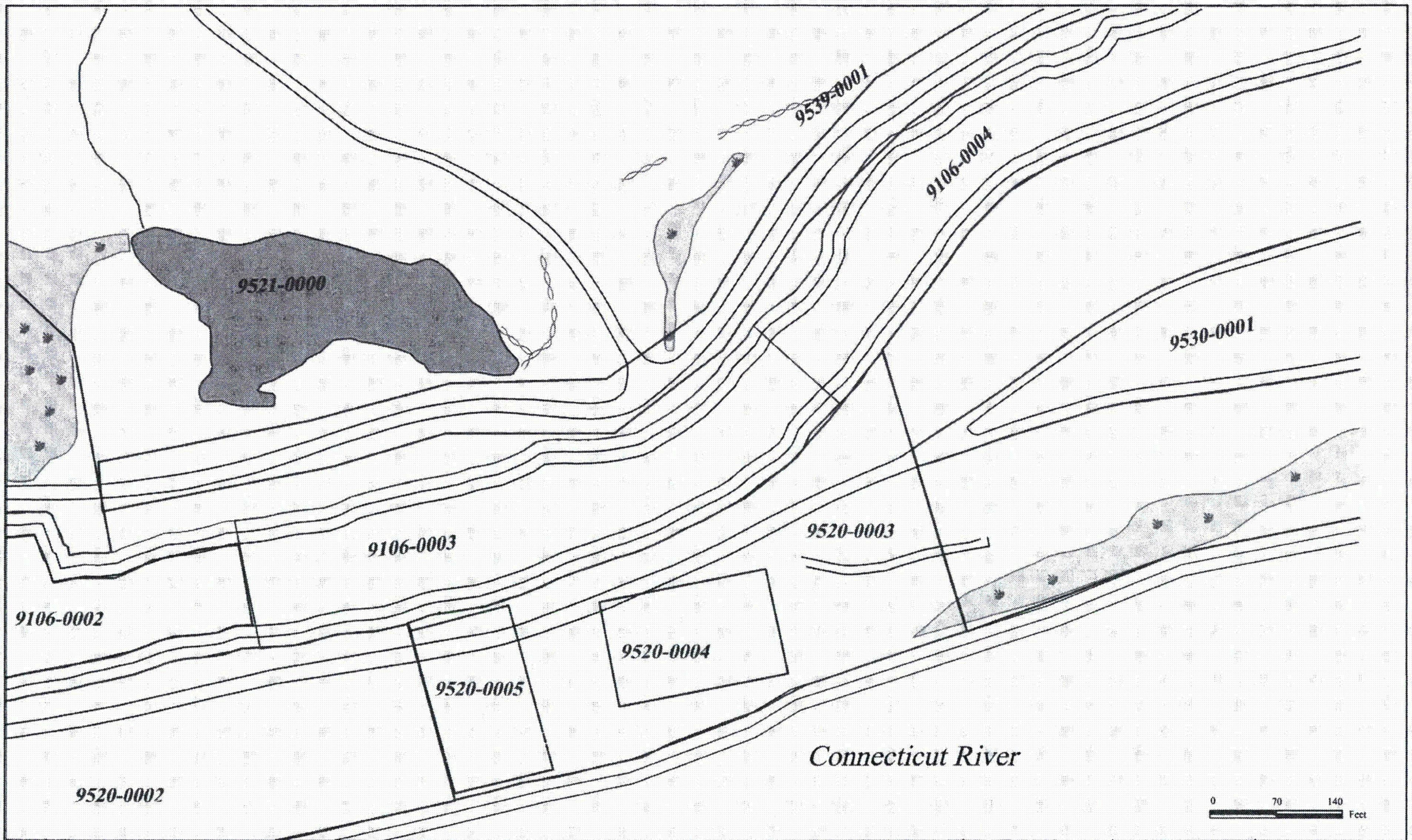
LEGEND

= Pond

= Wet Land



9312-0006
9312-0008
9312-0003
9522-0004
9522-0005
9306-0000
9106-0014
9106-0001
9520-0006
9304-0002
9304-0001
9520-0001
Connecticut River
0 75 150 Feet
Plant North
CONNECTICUT YANKEE ATOMIC POWER COMPANY
CY UPDATE SURVEY UNIT BOUNDARIES
Surface Areas
INITIAL CREATION DATE: APRIL 2002
REVISED BY: C Newson
REVISION DATE: JULY 2007
LTP REVISION #: 6
MAP #: FIGURE 2-3
LEGEND



Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY

CY UPDATE SURVEY UNIT BOUNDARIES

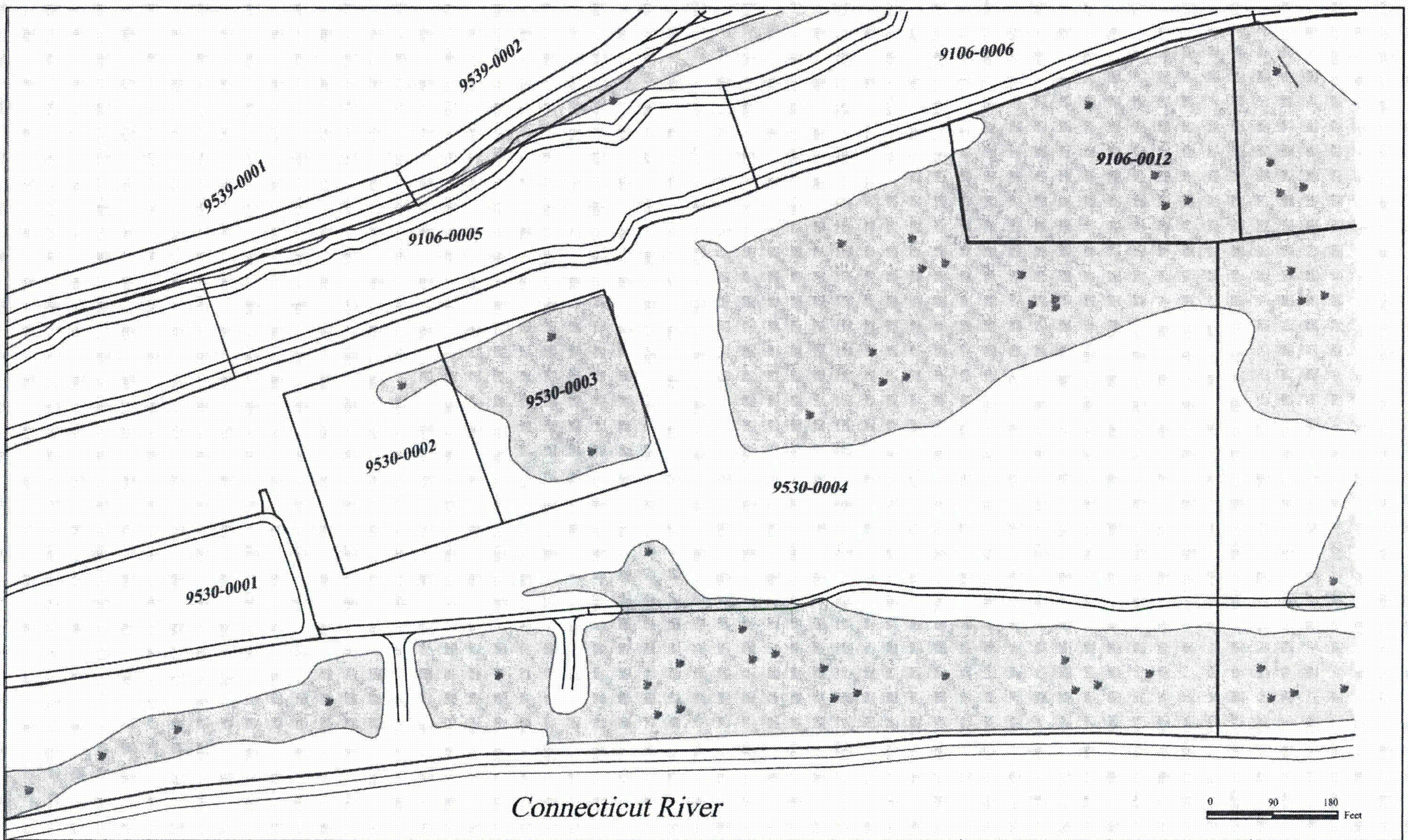
Surface Areas

INITIAL CREATION DATE:	APRIL 2002
REVISED BY:	J McCARTHY
REVISION DATE:	OCTOBER 2006
LTP REVISION #:	4
MAP #:	FIGURE 2-3

LEGEND

= Pond

= Wet Land



Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY

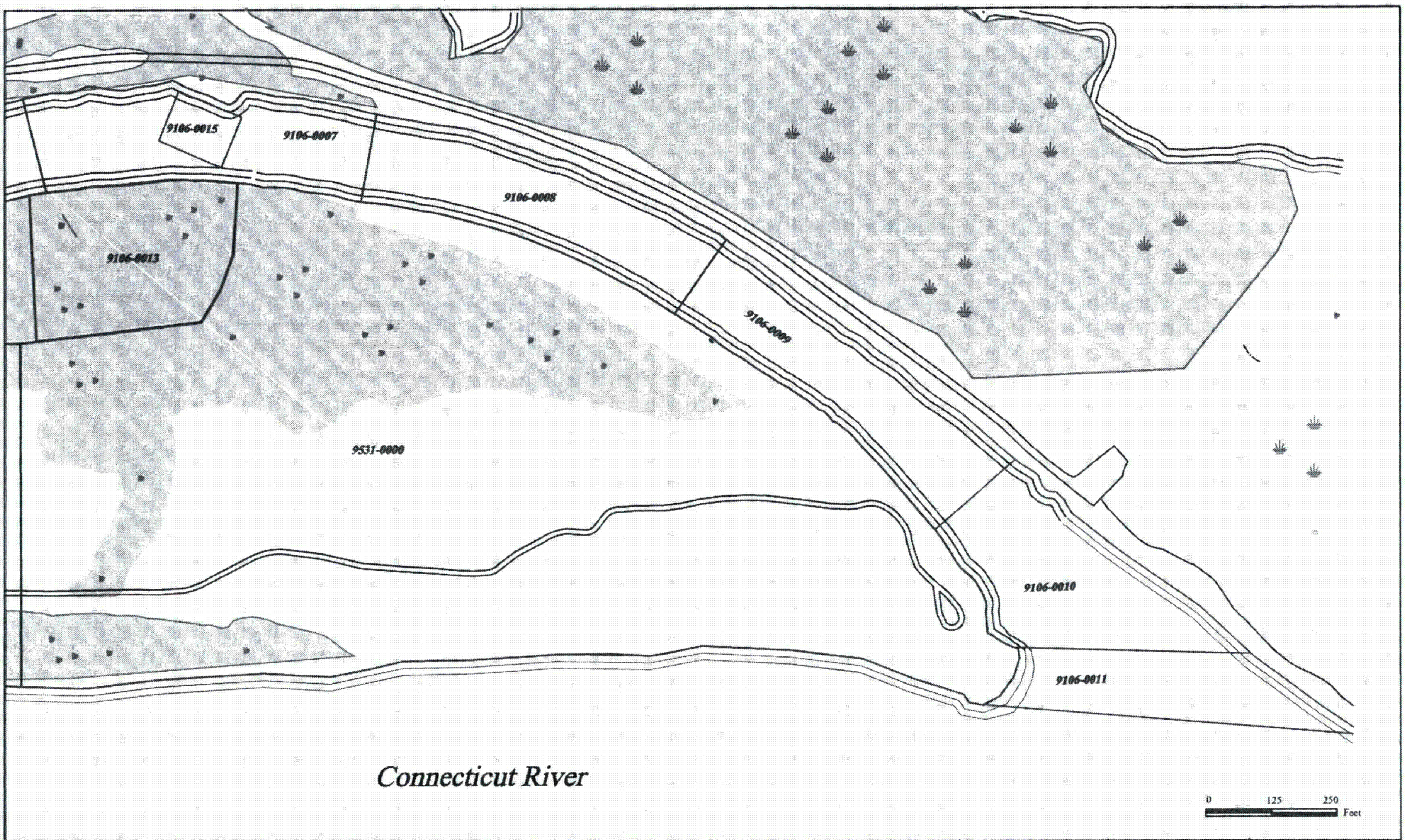
CY UPDATE SURVEY UNIT BOUNDARIES

Surface Areas

INITIAL CREATION DATE:	APRIL 2002
REVISED BY:	J McCARTHY
REVISION DATE:	OCTOBER 2006
LTP REVISION #:	4
MAP #:	FIGURE 2-4

LEGEND

= Wet Land



Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY

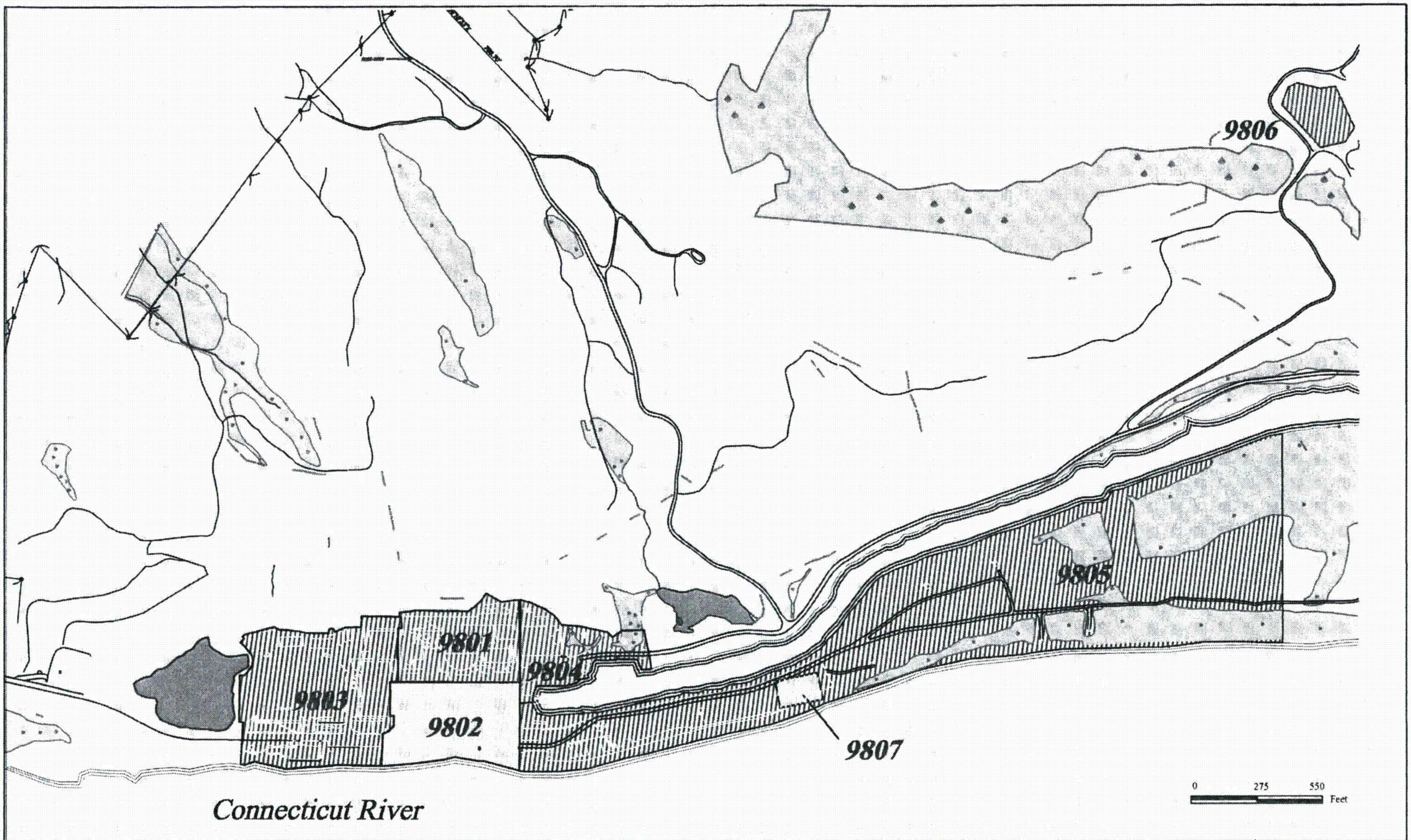
CY UPDATE SURVEY UNIT BOUNDARIES

Surface Areas

INITIAL CREATION DATE:	APRIL 2002
REVISED BY:	J McCARTHY
REVISION DATE:	OCTOBER 2006
LTP REVISION #:	4
MAP #:	FIGURE 2-5

LEGEND

= Wet Land



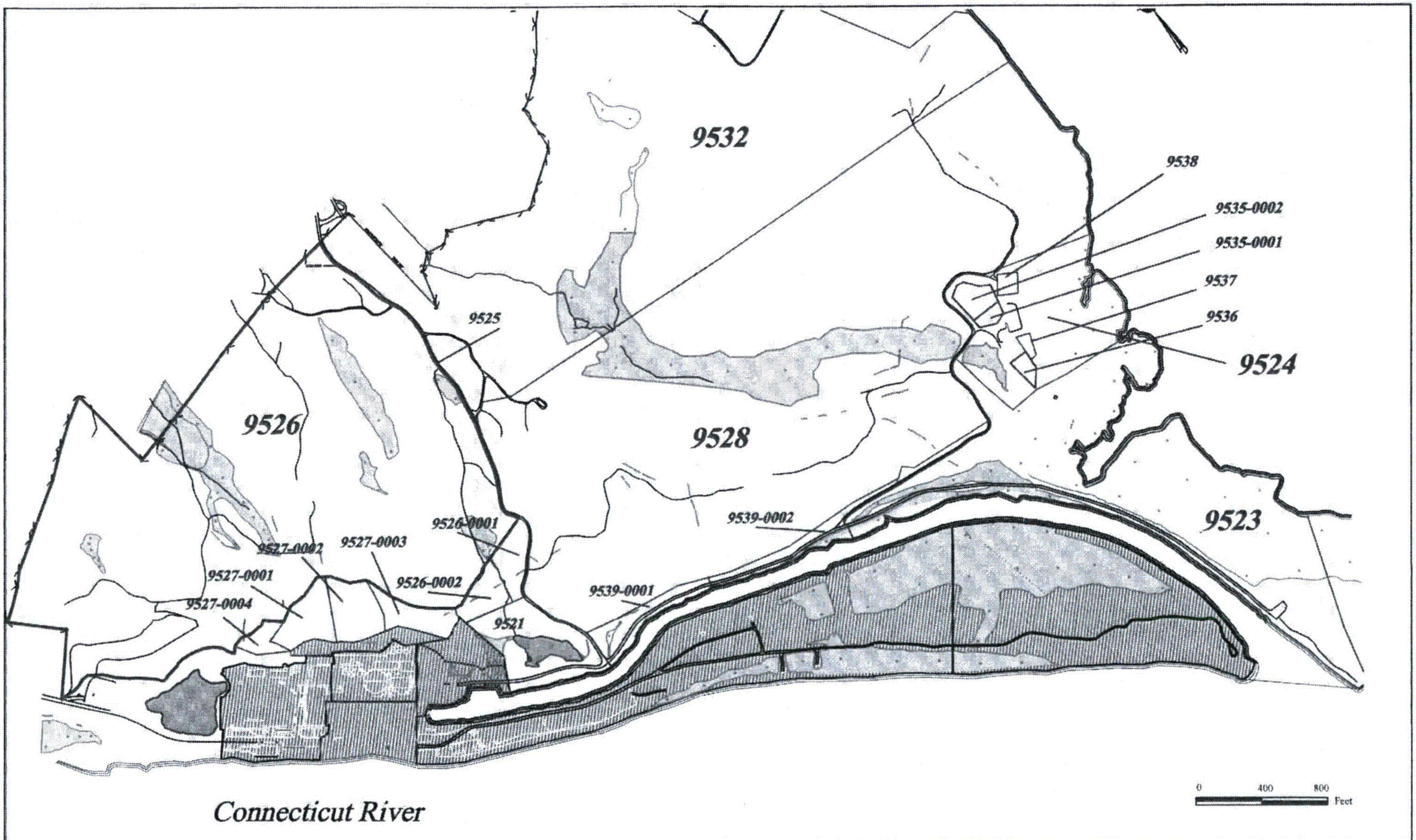
Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
Subsurface Areas

INITIAL CREATION DATE:	APRIL 2002
REVISED BY:	J McCARTHY
REVISION DATE:	OCTOBER 2006
LTP REVISION #:	4
MAP #:	FIGURE 2-6

LEGEND

= CLASS A
= CLASS B
= CLASS C



Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
Surface Areas

INITIAL CREATION DATE:	APRIL 2002
REVISED BY:	C Newson
REVISION DATE:	JULY, 2007
LTP REVISION #:	6
MAP #:	FIGURE 2-7

LEGEND

= Refer to Figures 2-1 through 2-6

3 IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

3.1 Introduction

This section provides the remaining site dismantlement activities based on the site as it existed partially through the decommissioning process. On September 1, 2004, February 27, 2006, and November 26, 2007 (References 3-9 through 3-11), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

The information included within this section includes historical information that will be maintained in its current form. This information will be reviewed, and revised as necessary, at the time of initiating the decommissioning activities for the ISFSI and associated land areas to ensure that appropriate information is available for the implementation of final status survey activities for the ISFSI and termination of the Part 50 License for the HNP site.

In accordance with 10CFR50.82 (a)(9)(ii)(B) (Reference 3-1), the LTP must identify the major dismantlement and decontamination activities that remain. The information includes those areas and equipment that need further remediation and an estimate of the radiological conditions that may be encountered. Included are estimates of associated occupational radiation dose and projected volumes of radioactive waste. These activities are undertaken pursuant to the current 10CFR50 license, are consistent with the PSDAR, and do not depend upon LTP approval to proceed.

CYAPCO's primary goals are to decommission the HNP safely and to maintain the continued safe storage of spent fuel. CYAPCO has decontaminated and dismantled the HNP in accordance with the DECON alternative, as described in the NRC's Final Generic Environmental Impact Statement. Currently, CYAPCO is managing the storage of spent fuel and GTCC waste at the HNP ISFSI. Following the removal of the spent fuel and GTCC waste from the site by the Department of Energy, the decommissioning activities associated with the ISFSI and associated land areas will be undertaken. Completion of the DECON option is contingent upon continued access to one or more low level waste disposal sites.

CYAPCO is conducting decontamination and dismantlement activities at the HNP site in accordance with the HNP PSDAR (Reference 3-2). Decommissioning activities

associated with the HNP ISFSI will be coordinated with the appropriate Federal and State regulatory agencies in accordance with plant administrative procedures.

Decommissioning activities at Haddam Neck will be conducted in accordance with the Haddam Neck UFSAR, Technical Specifications, existing Part 50 License and the requirements of 10CFR50.82(a)(6) and (a)(7). If an activity requires prior NRC approval under 10CFR50.59(c)(2) or a change to the Haddam Neck Plant Technical specifications, or license, a submittal will be made to the NRC for review and approval before implementing the activity in questions. Decommissioning activities are conducted under the scrutiny of the existing CYAPCO Radiation Protection Program, Industrial Safety Program, and Waste Management Program. Such activities will be conducted in accordance with these programs which are well established and frequently inspected by the NRC. Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during operations, especially those during major maintenance and outage evolutions.

Decontamination and dismantlement activities continue to be performed, as described in Section 3.3, while taking into account the specific system considerations as discussed in Sections 3.4.1 and 3.4.2. These sections provide an overview and describe the major remaining components of contaminated plant systems and, as appropriate, a description of specific equipment remediation considerations. Table 3-1 contains a list of major systems and components that remain to be removed.

Table 3-1
Status of Major HNP ISFSI Systems, Structures, and Components as of October 2013

SSC	SSC Status
ISFSI Pad	In place
Vertical Concrete Casks	In place
Transportable Storage Canisters	In place

3.2 Spent Fuel Pool Island Activities

On March 30, 2005, all spent fuel and GTCC waste have been removed from the Spent Fuel Pool and stored in an Independent Spent Fuel Storage Installation (ISFSI) at the HNP site.

3.3 Completed and Ongoing Decommissioning Activities and Tasks

On September 1, 2004, February 27, 2006, and November 26, 2007 (References 3-9 through 3-11), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

The Spent Fuel and GTCC waste have been removed from the Spent Fuel Pool and transferred to the ISFSI. With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site. At the completion of D&D activities, a comprehensive final radiation survey will be performed.

3.4 Future Decommissioning Activities and Tasks

Refer to Section 3.3.

3.5 Radiological Impacts of Decommissioning Activities

With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site. Decommissioning activities at the HNP ISFSI will be conducted in accordance with the Haddam Neck UFSAR, Technical Specifications, existing 10 CFR Part 50 License and the requirements of 10CFR50.82(a)(6) and (a)(7). If an activity requires prior NRC approval under 10CFR50.59(c)(2) or a change to the Haddam Neck Plant Technical Specifications or license, a submittal will be made to the NRC for review and approval before implementing the activity in question.

The decommissioning activities described herein are conducted under the provisions of the CYAPCO Radiation Protection Program and Radioactive Waste Management Program. These programs continue to be implemented as described in the HNP UFSAR. The Radiation Protection Program implements the regulatory requirements of 10CFR20 (Reference 3-7) through approved plant procedures established to maintain radiation exposures ALARA. The Radioactive Waste Management Program controls generation, characterization, processing, handling, shipping, and disposal of radioactive waste per the

approved CYAPCO Radiation Protection Program, Process Control Program, and plant procedures.

The current radiation protection program (described in UFSAR Chapter 12), waste management program (described in UFSAR Chapter 11), and Radiological Effluent Monitoring and Offsite Dose Calculation Manual (described in Section 6.6.3 of the HNP technical Specifications) will be used to protect the workers and the public during the various decontamination and decommissioning activities. These well established programs are routinely inspected by the NRC to ensure that workers, the public, and the environment are protected during facility decommissioning activities. It is also important to note that most decommissioning activities involve very similar radiation protection and waste management considerations as those encountered during plant operations. As described in the PSDAR, the HNP ISFSI decommissioning will be accomplished with no significant adverse environmental impacts in that:

- No site-specific factors pertaining to the HNP would alter the conclusion of the Final Generic Environmental Impact Statement (FGEIS).
- Radiation dose to the public will be minimal.
- Decommissioning is not an imminent health or safety concern and will generally have a positive environmental impact.

3.5.1 Occupational Exposure

Detailed exposure estimates and exposure controls for specific activities are developed during detailed planning per Radiation Protection Program procedures. The total radiation exposure impact for decommissioning and spent fuel management was estimated to total approximately 935 person-rem ($\pm$ 10%), as given in the PSDAR. Since that estimate was made, a significant amount of the decommissioning tasks have been completed. An estimate of the total occupational exposure as of March 2004 established that the total occupational exposure estimate was within 10% of the original estimate. This estimate utilized the actual occupational exposure associated with the decommissioning tasks that have had been completed and estimates for the tasks to be performed. The maximum estimate of the total occupational exposure remains within the 1,115 person-rem exposure estimate of Supplement 1 to NUREG-0586 (Section 4.3.8, Table 4-1) for a pressurized water reactor (PWR). The occupational dose associated with the decommissioning of the ISFSI is not expected to contribute significantly to the overall occupational dose associated with decommissioning the HNP site.

Table 3-2
Deleted

3.5.2 Radioactive Waste Projections

The Radioactive Waste Management Program is used to control the characterization, generation, processing, handling, shipping, and disposal of radioactive waste during decommissioning. For the remaining ISFSI structures and systems, activated systems, structures, and components represent the largest volume of low level radioactive waste expected to be generated during decommissioning. Other forms of waste generated during decommissioning include:

1. Contaminated water;
2. Used disposable protective clothing;
3. Expended abrasive and absorbent materials;
4. Expended resins and filters;
5. Contamination control materials (e.g., strippable coatings, plastic enclosures); and
6. Contaminated equipment used in the decommissioning process.

Table 3-3 provides projections of waste quantities for decommissioning. These waste quantities are those reflected in the PSDAR. The total volume of HNP low-level radioactive waste for disposal has been estimated at 1,158,000 cubic feet. Actual waste volumes and classifications may vary, but the total quantity is not expected to exceed 1,158,000 cubic feet. In addition, the decommissioning cost estimate prepared in 2012 assumed that the materials that comprise the Vertical Concrete Casks (VCCs) and the ISFSI storage pad would be shipped offsite as low-level radioactive waste (LLRW). This increases the total amount of material that will be removed from site.

Decommissioning planning at CYAPCO incorporates the assumption that cost-effective waste volume reduction methods are limited. It also assumes some significantly contaminated or activated materials are sent directly to a disposal facility. However, alternate processing methods consistent with the approaches described herein may be evaluated and used during decommissioning.

Table 3-3
Projected Waste Quantities Associated with the HNP

Radwaste	Volume
Commodities	41,292
Containment	426,506
Misc Packaged Radwaste Awaiting Shipment	33,333
Misc. R2 Structures Disposal	101,454
PAB	69,330
Service Building	55,828
Spent Fuel Building	59,962
Terry Turbine Building	316
Waste Disposal Building	22,096
Containment Liner (Mixed PCB/radwaste)	12,407
Contaminated Roofs	17,949
Dredge Spoils	83,333
Rad Soil	187,142
Rad Asphalt	46,568
Rad Subtotal	**1,157,516**

3.6 References

3-1 Code of Federal Regulations, Title 10, Part 50.82, "Termination of License."

3-2 Letter, CY-97-075 from CYAPCO to the USNRC, "Haddam Neck Plant Post Shutdown Decommissioning Activities Report," dated August 22, 1997.

3-3 Not used.

3-4 Not used.

3-5 Not used.

3-6 Not used.

3-7 Code of Federal Regulations, Title 10 Part 20, "Standards for Radiation Protection."

3-8 Revision 3 to the PSDAR, dated October 2013.

3-9 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of East Site Grounds from Part 50 License," dated September 1, 2004.

3.10 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of Phase II Areas from Part 50 License," dated February 27, 2006.

3-11 Letter from K. McConnell (USNRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of Land from Part 50 License," dated November 26, 2007.

4 SITE REMEDIATION PLANS

4.1 Introduction

This section provides site remediation plans that are based on the site as it existed partially through the decommissioning process. On September 1, 2004, February 27, 2006, and November 26, 2007 (References 4-5 through 4-7), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

The information included within this section includes historical information that will be maintained in its current form. This information will be reviewed, and revised as necessary, at the time of initiating the decommissioning activities for the ISFSI and associated land areas to ensure that appropriate information is available for the implementation of final status survey activities for the ISFSI and termination of the Part 50 License for the HNP site.

In accordance with 10CFR50.82(a)(9)(ii)(C) (Reference 4-1), the LTP must provide the "plans for site remediation." These plans must include the provisions to meet the criteria from Subpart E of 10CFR20 (Reference 4-2) before the site may be released for unrestricted use:

- Annual total effective dose equivalent to the average member of the critical group not to exceed 25 mrem, and
- The dose to the public msut be "as low as reasonably achievable," or ALARA.

Decontamination and dismantlement activities will be conducted in accordance with the CY Radiation Protection, Safety and Waste Management Programs, which are well established and frequently inspected. Changes have been made to these programs for D&D activities, and any future changes that may be made will be documented and processed with existing plant administrative procedures using 10CFR50.59 and the guidance contained in Regulatory Guide 1.187.

This section describes the methodologies and criteria that will be used to perform remediation activities of residual radioactivity and to demonstrate compliance with the ALARA criteria, required by 10CFR20. More specific detail regarding remediation activities may be found in Chapter 3.

4.2 Remediation Levels and ALARA Evaluations

When dismantlement and decontamination actions are completed, residual radioactivity may remain on building surfaces and on site soils. Residual radioactivity must satisfy the provisions of 10CFR20, Subpart E. As depicted on Figure 4-1, the ALARA cleanup levels for the HNP decommissioning may be established at one of two levels:

(1) a predefined generic ALARA screening, or
(2) a survey unit-specific ALARA evaluation.

In either case, the ALARA evaluation uses an action level, referred to as a remediation level. This remediation level corresponds to a residual radioactivity concentration at which the averted collective radiation dose converted into dollars is equal to the costs of remediation (e.g., risk of transportation accidents converted into dollars, worker and public doses associated with the remediation action converted into dollars, and the actual costs to perform the remediation activity).

If the value of further dose reduction from remediation is greater than the "costs" of the action, then the remediation action being evaluated is cost-effective and should be performed. Conversely, if the value of further dose reduction is less than the costs, the levels of residual radioactivity are considered ALARA and therefore further remediation action would not be required. The methodology and equations used for calculating remediation levels are consistent with those provided in Draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination" (Reference 4-3), as incorporated in Appendix D of NUREG-1727 (Reference 4-4). These are provided in Appendix B of the LTP. Documentation of ALARA evaluations will be included in the final status survey documentation for each survey area.

The selection of appropriate instrumentation for post-remediation surveys is important from a planning and financial risk management perspective. Post-remediation surveys serve as a check to see if the remediation target is achieved. As shown on Table 5-11 of this LTP, if small handheld beta-gamma detectors are used to determine if remedial actions have been successful, the nominal frisk MDC is estimated to be between 1200 to 3200 dpm/100 cm^2. Based upon site characterization data, the predominant beta-gamma emitting radionuclides at HNP are Co-60 and Cs-137. The corresponding building surface DCGLs for Co-60 and Cs-137 as shown in Table G-3 are 11,100 and 43,000 dpm/100 cm^2 respectively. Therefore, the MDCs of the handheld detectors are approximately only 3 to 30 percent of the predominant beta-gamma emitting radionuclide DCGLs. In other cases, the actual final status survey instrumentation may be used to evaluate remedial actions.

4.2.1 Generic ALARA Screening Levels

As discussed in DG-4006 (and Appendix D of NUREG-1727), soil remediation beyond the DCGLs is not likely to be cost-beneficial due to the high costs of waste disposal.

This has been confirmed in a generic ALARA evaluation for soils. Thus soil will be at ALARA levels when it meets the site-specific DCGLs discussed in Chapters 5 and 6.

For building surfaces, a generic ALARA screening value will be calculated using conservative estimates for building remediation costs. This generic ALARA screening value will be calculated using the guidance of DG-4006, after additional characterization has been undertaken and remediation methods have been evaluated for their effectiveness. This value will represent the level, expressed as a percentage or fraction of the DCGL, for which the benefit of further remediation of structures is greater than the associated costs.

Upon completion of post-remediation surveys and satisfaction of the 25 mrem/yr TEDE criteria, the level of residual radioactivity in the survey area will be compared against the appropriate generic ALARA screening value. Where the level of residual radioactivity is lower than the generic ALARA screening value, the remediation is clearly ALARA, no further remediation is required, and final status surveys can proceed. Where the level of residual radioactivity is greater than the generic ALARA screening value, a survey-unit ALARA evaluation is performed to determine the unit-specific ALARA remediation level for comparison.

4.2.2 Survey-Unit Specific ALARA Evaluation

In cases where levels of residual radioactivity are above the generic ALARA screening levels described above, survey unit-specific ALARA evaluations will be performed using approved site procedures. These survey unit-specific ALARA evaluations will be performed using data from post-remediation surveys in accordance with DG-4006 and will take into account:

- Radiation doses and environmental impacts for the decommissioning process and from the residual radioactivity remaining onsite following the decommissioning, and
- Other costs and risks associated with the decontamination and decommissioning of the site.

Once the total cost, $Cost_T$, for a survey-unit specific remediation action has been calculated, a remediation level, expressed as a fraction of a DCGL, can be determined and the ALARA evaluation can be performed using the process described in DG-4006.

The remediation levels represent the radioactivity concentrations at which a remediation action is cost beneficial and, therefore, do not represent maximum or "not-to-exceed" concentrations. The ALARA criteria are met by performing the remediation action and not necessarily by achieving results below a specific remediation level. An ALARA analysis ensures that the efforts to remove residual contamination are commensurate with the risk that exists with leaving the residual contamination in place, even if the target

remediation levels are not achieved. However the residual contamination must be low enough to assure the annual dose to the average member of the critical group does not exceed 25 mrem/yr TEDE.

4.2.3 Groundwater ALARA Evaluation

As discussed in Section 1.6 of Appendix D to NUREG-1727, if there is residual radioactivity from site operations in groundwater, it may be necessary to calculate the collective dose from consumption of the groundwater. Sampling of the groundwater at the HNP Site indicates that residual radioactivity does exist. Dose modeling, as discussed in Chapter 6, assumes that the aquifer does not supply a large population, but only the resident farming family. Currently there is no population deriving its drinking water from a downstream supply, and based upon current knowledge of the aquifer onsite, it is doubtful that this aquifer would be used as a drinking water source for a large population. Wells providing water of potable quality cannot supply high well pumping rates needed to support a large population, and wells that can supply higher pumping rates are located in the unconsolidated sediments and provide poor quality water. These conditions will be re-confirmed during the program of ongoing groundwater monitoring, as described in Section 2.3.3.1.6.2. However, if it is determined that drinking water for a large population could be supplied by groundwater onsite, the collective dose for that population will be included in the ALARA calculation as indicated in NUREG-1727, Appendix D.

4.3 Remediation Actions

Remediation actions may be required to reduce the residual radioactivity levels below the applicable cleanup criteria as provided in Chapters 5 and 6. The specific remedial actions depend on the type of area under consideration. These area types are categorized as one of the following:

1. Structures (including building interiors and exteriors, major freestanding exterior structures, exterior surfaces of plant systems, and paved exterior ground surfaces);

2. Soils; and

3. Nonstructural plant systems (including interior surfaces of process piping and components).

On September 1, 2004, February 27, 2006, and November 26, 2007 (References 4-5 through 4-7), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

4.3.1 Structures

Using the demolition and bulk disposal approach to decommissioning, concrete from contaminated structures will be remediated to levels meeting the radiological criteria for controlled demolition. The debris materials will be disposed of at a licensed radioactive material disposal facility. Nonstructural materials will be assessed using the process in Section 5.6.

Methods for remediating structures may include a variety of techniques ranging from water washing to surface material removal. A number of factors determine the choice of the remediation method for a given area, including: the size of the contaminated area, the extent of contamination, surface material, depth of contamination, and accessibility.

Remediation activities for an area may include wiping, vacuuming, and washing with low- or high-pressure applications. Surfaces may also be remediated using surface removal techniques such as scabbling or grinding.

For concrete surfaces, remediation methods may include core drilling, concrete sawing, or scabbling. Scabbling removes the concrete surface by bush heads, rotopeen devices, flappers, or similar devices and is effective for removing contamination that resides close to the surface. Abrasive blasting may also be used as an effective technique for contamination removal from surfaces that are not necessarily smooth. Also, chipping, jackhammering, and other similar aggressive methods may be needed for removal of concrete surfaces as deep as the first mat of reinforcing steel. Strippable coatings can be used to remove contaminants from surfaces where more aggressive methods may not be appropriate or when other techniques are not successful.

4.3.2 Soils

During 1998 and 1999 greater than one hundred subsurface soil samples, in some cases down to six feet in depth, were collected outside of the RCA in Survey Areas 9302, 9304, 9306, and 9522 in support of plant modifications and site characterization activities. None of the samples had plant related radioactivity levels greater than the corresponding base-case soil DCGLs. During the same time period, over two hundred subsurface soil samples, in some cases down to four feet in depth, were collected inside the RCA in Survey Area 9312 (including Bus-10 Pad and ground underneath). Some isolated spots showed Co-60 and Cs-137 activity levels up to several hundred pCi/g each. Most of the sampling was performed in Survey Area 9312, Unit 0002 in support of the spent fuel pool project. Soil under bus 10 was removed until that soil which remained was at or below the NRC's generic soil screening DCGLs, adjusted to 10 mrem/yr. Continued

characterization of site soils has resulted in identification of contaminated areas and the depth of the contamination. This will continue throughout the decommissioning.

Soils not meeting the applicable DCGLs will be removed and disposed of as radioactive waste. Offsite fill or on-site materials shown to meet the soil DCGLs will be used to replace the excavated materials. As discussed previously in Chapter 2, the ongoing site characterization process establishes the location, depth and extent of soil contamination. As needed, additional investigations will be performed to ensure that any soil contamination profiles that may change during the remediation actions are adequately identified and characterized. In cases where offsite fill will be used to replace the excavated materials, a direct radiation survey will be conducted of either each load of fill or of the site from which the material will be obtained. This will be done as a documented survey to ensure that the background radiation levels (from the presence of naturally occurring radioactive material) from this fill material will not be significantly higher than that from the onsite material. This survey may be performed using CYAPCO's vehicle monitoring system (Bicron SM6000), by a handheld scintillation-based survey meter (sensitive to changes in ambient background radiation levels), sample collection or analysis, or in-situ gamma spectroscopy measurements. Based upon the results of this survey, either background radiation levels will be accounted for in subsequent final status surveys or the material will be rejected for use.

4.3.3 Nonstructural Systems

Chapter 3 describes the systems to be decontaminated, demolished, and disposed of. Contaminated plant systems and components may be sent to an offsite processing facility or to a low-level radioactive waste disposal facility. Slightly contaminated systems may be decontaminated onsite and released. Nonstructural systems and components will be surveyed and released using existing plant procedures and process (i.e., "free release criteria"), with the exception of those cases discussed in Section 5.4.7.6.

Remediation methods typically used for system decontamination include chemical decontamination, wiping, washing, vacuuming, or abrasive blasting. Selection of the preferred method is based on the specific situation. Other remediation technologies may be considered and used, as appropriate.

4.4 References

4-1 Code of Federal Regulations, Title 10, Part 50.82, "Termination of License."

4-2 Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use."

4-3 Draft Regulatory Guide-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination," August 1998.

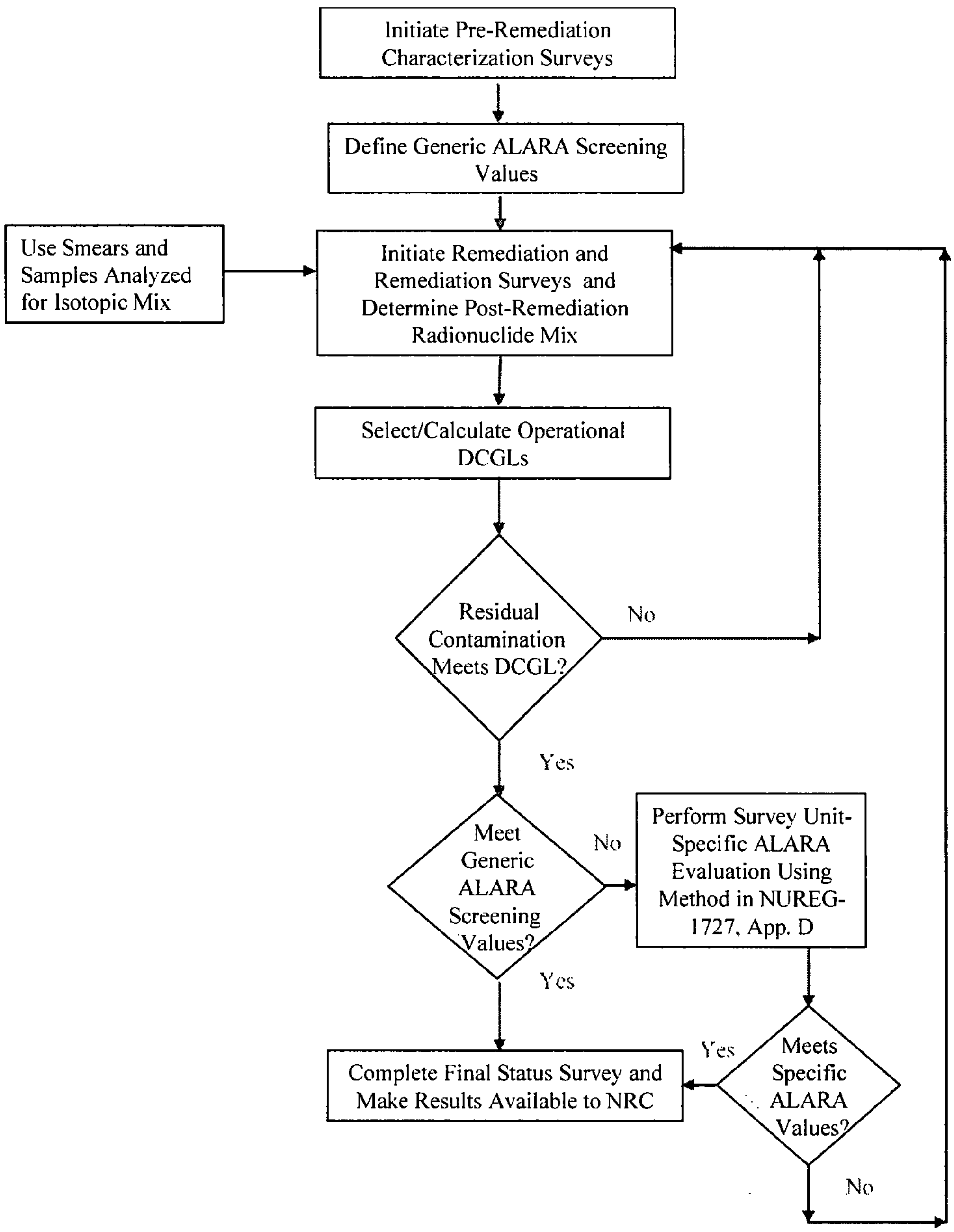
4-4 NUREG-1727, "NMSS Decommissioning Standard Review Plan," September 2000.

4-5 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of East Site Grounds from Part 50 License," dated September 1, 2004.

4-6 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of Phase II Areas from Part 50 License," dated February 27, 2006.

4-7 Letter from K. McConnell (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant - Release of Land from Part 50 License," dated November 26, 2007.

Figure 4-1
Survey Unit ALARA Evaluation Process



5 FINAL STATUS SURVEY PLAN

5.1 Introduction

This section provides a final status survey plan that is based on the site as it existed partially through the decommissioning process. On September 1, 2004, February 27, 2006, and November 26, 2007 (References 5-14 through 5-16), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

The information included within this section includes historical information that will be maintained in its current form. This information will be reviewed, and revised as necessary, at the time of initiating the decommissioning activities for the ISFSI and associated land areas to ensure that appropriate information is available for the implementation of final status survey activities for the ISFSI and termination of the Part 50 License for the HNP site.

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); NRUEG-1575, "The Multi-Agency Radiation Survey and Site 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 NRUEG-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 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 some cases be used to calculate the "future groundwater" dose in building basements/footings 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 performing 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.

The Data Quality Assessment (DQA) approach is applied to FSS results to ensure their validity and to demonstrate that the objectives of the FSS are met. Data assessment includes data Verification and Validation (V&V), review of survey design bases, and data analysis. For a given survey unit, the survey data are evaluated to determine if the residual activity

levels in the unit meet the applicable release criterion and if any areas of elevated activity exist. In some cases, data evaluation will simply serve to show that all of the measurements made in a given survey unit were below the applicable $DCGL_W$. If so, demonstrating compliance with the release criterion is a simple matter and requires little in the way of analysis. In other cases, residual radioactivity may exist where measurement results both above and below the $DCGL_W$ are observed. In these cases, statistical tests must be performed to make a decision as to whether the unit meets the release criterion. The statistical tests that may be required to make decisions regarding the residual activity levels in a survey unit relative to the applicable $DCGL_W$ must be considered in the survey design to ensure that a sufficient number of measurements are collected.

MARSSIM specifies two non-parametric statistical tests to be applied to final status survey data to evaluate whether a set of measurements demonstrates compliance with the release criterion for a given survey unit. These statistical tests are discussed in detail in Section 5.8.

Quality assurance and control measures are employed throughout the final status survey process to ensure that all decisions are made on the basis of data of acceptable quality. Quality assurance and control measures are applied to ensure:

- the plan is correctly implemented as prescribed,
- DQOs are properly defined and derived,
- all data and samples are collected by individuals with the proper training following approved procedures,
- all instruments are properly calibrated,
- all collected data are validated, recorded, and stored in accordance with approved procedures,
- all required documents are properly maintained, and,
- if necessary, corrective actions are prescribed, implemented and followed up.

These measures apply to any services provided in support of final status survey.

Survey results will be converted to appropriate units (i.e., either dpm/100 cm^2 or pCi/g) and compared to investigation levels to determine appropriate follow-up action. Measurements exceeding investigation levels will be verified and investigated and, following confirmatory measurement(s), the affected area may be remediated and/or reclassified and a re-survey performed consistent with the guidance in MARSSIM (Section 8.5.3, "If the Survey Unit Fails") and commensurate with the classification and extent of contamination.

Documentation of the final status survey will transpire in two types of reports. 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 submitted to the NRC, 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. These reports are discussed in detail in Section 5.9.

The documentation describing the final status survey for a given survey unit will include:

- a physical description of the survey area which encompasses the unit(s) (in many cases, the survey areas and survey units will be the same);
- the characterization data associated with the area, including any required investigations, re-classifications or subdivisions;
- the classification history of the unit;
- the remediation activities (if any) performed in the survey unit;
- results and discussion of any ALARA evaluations performed;
- a discussion of the survey design (combination of scans, fixed measurements, samples, number of measurements, grid spacing, etc.);
- tabular and graphical depictions of survey results;
- discussions of data assessments, including graphical depictions; and
- conclusion that survey unit meets all applicable criteria.

It is anticipated that final status survey results will be documented and made available to the NRC for survey areas rather than for individual survey units. Reports will be compiled after final status survey activities for all of the survey units for a given area are completed. This approach should minimize the submittal of redundant historical assessment information and provide for a logical approach to perform reviews and independent verification.

5.4 Survey Planning

5.4.1 Data Quality Objectives

The DQO process is incorporated as an integral component of the data life cycle at HNP. The DQO process is used in the planning phase for scoping, characterization, remediation, and final status survey plan development using a graded approach. Survey plans that are complex or that have a higher level of risk associated with an incorrect decision (such as final status surveys) would require significantly more effort than a survey plan used to obtain data relative to the extent and variability of a contaminant. This process, described in MARSSIM, is a series of planning steps found to be effective in establishing criteria for data quality and developing survey plans. DQOs allow for systematic planning and are specifically designed to address problems that require a decision to be made and provide alternate actions. Furthermore, the DQO process is flexible in that the level of effort associated with planning a survey is

based on the complexity of the survey and nature of the hazards. Finally, the DQO process is iterative allowing the survey planning team to incorporate new knowledge and modify the output of previous steps to act as input to subsequent steps. The Final Status Survey Quality Assurance Project Plan (FSSQAPP) provides a general description of the application of the DQO process to the different elements of the final status survey.

The DQO process consists of performing the following seven steps:

- State the Problem
- Identify the Decision
- Identify the Inputs to the Decision
- Define the Boundaries of the Decision
- Develop a Decision Rule
- Specify Tolerable Limits on Decision Errors
- Optimize the Design for Obtaining Data

The actions taken to address these DQO process steps during the planning of a final status survey for a particular survey area are addressed below.

- **State the Problem**

  The first step of the planning process consists of defining the problem. This step provides a clear description of the problem, identification of planning team members (especially the decision-makers), a conceptual model of the hazard to be investigated and the estimated resources. The problem associated with FSS is to determine whether an area meets the radiological release criterion of 10CFR20.1402.

- **Identify the Decision**

  This step of the DQO process consists of developing a decision statement based on a principal study question (i.e., the stated problem) and determining alternative actions that may be taken based on the answer to the principal study question. Alternative actions identify those measures to resolve the problem. The decision statement combines the principal study question and alternative actions into an expression of choice among multiple actions. For FSS the principal study question is "Does residual radioactive contamination present in the survey unit exceed the release criteria?" The alternative actions may include no action, investigation, resurvey, remediation and reclassification.

- **Identify Inputs to the Decision**

  The information required depends on the type of media under consideration (e.g., soil, water, concrete) and whether existing data are sufficient or new data are needed to make the decision. If the decision can be based on existing data then the source(s) will be documented and evaluated to ensure reasonable confidence that the data are acceptable. If new data are needed, then the type of measurement (e.g., scan, direct measurement and sampling) will be determined.

Sampling methods, sample quantity, sample matrix, type(s) of analyses and analytic methods and measurement process performance criteria, including detection limits, are established to ensure adequate sensitivity relative to the action level and to minimize bias. Action levels provide the criterion for choosing among alternative actions (e.g., whether to take no action, perform confirmatory sampling). These action levels may be radioactivity concentration (pCi/g) or measurement device response (count rate corrected for background).

- **Define the Boundaries of the Study**

  This step of the DQO process includes identification of the target population of interest, the spatial and temporal features of the population pertinent to the decision, time frame for collecting the data, practical constraints and the scale of decision making. In FSS, the target population is the set of samples or direct measurements that constitute an area of interest (i.e., the survey unit). The medium of interest (e.g., soil, water, concrete, steel) is specified during the planning process. The spatial boundaries include the entire area of interest including soil depth, area dimensions, contained water bodies and natural boundaries, as needed. Temporal boundaries include those activities impacted by time-related events including weather conditions, seasons (i.e., more daylight available in the summer, operation of equipment under different environmental conditions, resource loading and work schedule.

- **Develop a Decision Rule**

  This step of the DQO process develops the binary statement that defines a logical process for choosing among alternative actions. The decision rule is a clear statement using the "If...then..." format and includes action level conditions and the statistical parameter of interest (e.g., mean of data). Decision statements can become complex depending on the objectives of the survey and the radiological character of the affected area.

- **Specify Tolerable Limits on Decision Errors**

  This step of the DQO process incorporates hypothesis testing and probabilistic sampling distributions to control decision errors during data analysis. Hypothesis testing is a process based on the scientific method that compares a baseline condition to an alternate condition. The baseline condition is technically known as the null hypothesis. Hypothesis testing rests on the premise that the null hypothesis is true and that sufficient evidence must be provided for rejection.

  The primary consideration during FSS will be demonstrating compliance with the release criteria. The following statement may be used as the null hypothesis at HNP: "The survey unit exceeds the release criteria."

Decision errors occur when the data set leads the decision-maker to make false rejections or false acceptances during hypothesis testing. The α error (Type I error) is set at 0.05 (5%). The β error may be variable depending upon the objectives of the surveys. A nominal value of 0.05 (5%) has been established for the β error (Type II error). Another output of this step is assigning probability limits to points above and below the gray region where the consequences of decision errors is considered acceptable. The upper bound corresponds to the release criteria. The Lower Bound of the Gray Region (LBGR) is determined in this step of the DQO process. LBGR is influenced by a parameter known as the relative shift. The relative shift is set between (and including) 1 and 3. If the relative shift is not between (or including) 1 and 3, then the LBGR is adjusted.

Graphing the probability that a survey unit does not meet the release criteria may be used during FSS. This graph, known as a power curve, may be performed retrospectively (i.e., after FSS) using actual measurement data. This retrospective power curve may be important when the null hypothesis is not rejected (i.e., the survey unit does not meet the release criteria) to demonstrate that the DQOs have been met.

- **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. For the subsurface areas that will be evaluated using the Basement Fill Model, area classifications and survey unit area do not apply.

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 Table 2-11B, associated 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. The majority of these classifications are historical, because the only Survey Units that remain within the 10 CFR 50 License are Survey Units 9523-0000, 9528-0000, and 9528-0004.

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. The primary benchmark and coordinate system uses Global Positioning System (GPS) technology based on the Connecticut State Plane North American Datum (NAD) 1927. Other 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.

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. Generally, background will be included in reported measurements.

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 meets 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
<u>NaI</u> 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 of 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 surface 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 results 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 that may cause the spread of contamination in the area of consideration or any adjacent areas are completed or properly controlled 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 Remedial 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.

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

- Known contaminated decommissioning activities in the area are complete and any additional decommissioning activities identified shall pose a very low risk to add contamination to an area, including removal, as necessary, of items (e.g., equipment mounts, wall hangers, and exposed studs) that could interfere with final survey activities. Examples of activities that may be performed following FSS with isolation and controls in place are: The use of the foot bridge in the Pond to perform FSS surveys and the removal of the foot bridge following the FSS; performing an FSS of an area where a culvert requires replacement where there is a very low risk of contaminating the area; performing an FSS of an area and providing a vehicle travel path through the area for "clean vehicles" (vehicles not used in an RCA or having been surveyed (clean exterior) from the RCA.

- 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 structure 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.

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 RAS 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 21ft-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. 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.

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 semi-annually 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 Concentration Guideline 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 building surfaces. 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. 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 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 may 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).

To show compliance with the base case DCGL, 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. 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 future groundwater radioactivity from the building basements, foundations and footings. 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 m2 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, the dose from the residual radioactivity from radionuclide i is:

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

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" 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 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 the 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.

The future groundwater component of equation 5-1 can be further stated as follows:

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

For building basements, foundations and footings to remain, the future groundwater dose will be calculated by the Basement Fill Model. The dose calculation method for future groundwater is discussed in Section 6.8. Table 5-3 lists the survey units affected by present and future groundwater.

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 capture zone, the influence boundary distance of detectble 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 additional groundwater characterizationand completion of the capture zone analysis and future groundwater. The capture zone analysis determined a maximum zone of influence of 100 meters around a groundwater monitoring well, (see Figure 5-3 and 5-3.1) and reference 5-13, Estimated Zone of Influence/Capture Zone for Hypothetical Water Supply Wells in Post-Closure Dose Modeling, CH2MHILL Technical Memorandum, dated January 11, 2005. These figures depict the capture zone around the wells which have shown detectable contamination at the perimeter of the industrial area and around peninsula wells.

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 capture zone or detection of groundwater contamination at locations outside the zone. The Phase 2 Hydrogeologic Work Plan, as described in Section 2.3.1.6, will provide additional characterization of groundwater that will be used to better define the groundwater contamination plume.

Prior to the request to release of any portion of the site from the license, CYAPCO will prepare and make available for inspection a capture zone analysis provided to the NRC in January 2005, based on data collected as part of the Phase 2 Hydrogeological Work Plan, to better define the capture zone distance, Reference 5-13. 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 used the hydrogeological conditions and parameters assumed in the Resident farmer Scenario as described in Chapter 6 of the LTP.

Table 5-3
Survey Units Affected by Groundwater Contamination

Existing	**Future Groundwater**
Groundwater Zone of Influence	Concrete
9302	9302
9304-1&2	9304-1&2
9306	9306
9312-1 Thru 10	9312-1 Thru 10
9313	9313
9522-5, 6 & 7	9522-5, 6 & 7
9801	9801
9802	9802
9106-1, 2, 3, 4 & 14	N/A
9512	N/A
9514	N/A
9520-1, 2, 3, 4 & 5	N/A
9521	N/A
9522-1, 2, 3 & 4	N/A
9527-1, 2, 3, 4, 5 & 6	N/A
9528-0, 2 & 3	N/A
9530-1, 2, 3 & 4	N/A
9539-0001, 0002	N/A
9803	N/A
9804	N/A
9805	N/A
9807	N/A

On September 1, 2004, February 27, 2006, and November 26, 2007 (References 5-14 through 5-16), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004. Table 5-3 has been retained to establish the impact on those areas.

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

(Equation 5-5)

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

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 to the total dose from existing contamination in groundwater
$f_{FutureGW}$ is the fraction of the base case DCGL 25 mrem/yr dose that is calculated by the Basement Fill Model.

The use of this equation requires that only one variable be unknown. Therefore, values for Future Groundwater dose and $f_{ExistingGW}$ will need to be selected in order to calculate f_{Soil}. As 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 following example is provided to illustrate the use of the operational DCGLs for 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, Future Groundwater Dose (fraction of 25 mrem/yr) = 0.5

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 DGL Example for Cs-137
Using Fractional Values from Above

	Base Case DCGL/Dose	Operational DCGL/Dose
Soil (pCi/g)	7.91E+00	2.37E+00
Existing Groundwater (pCi/1)	4.31E+02	8.62E+01
Future Groundwater Dose (mrem/yr)	25	12.5

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.

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

- 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 *i th* 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		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 Basements, Foundations and Footings

After completion of final status survey activities of the remaining portions of structures, some subsurface concrete may remain in the form of building basements, foundations and 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 from groundwater contamination from the leaching of radionuclides from these structures. The dose model for the calculation of this "future groundwater" dose is call the Basement Fill Model and is discussed in Section 6.8. The sampling to be performed to determine 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 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.

Table 5-7
Release Limits for Buried Piping, dpm/100cm^2

Radionuclide	1"	2"	2 1/2 "	3"	4"	8"	10"	12"	14"	16"	18"	20"	24"
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
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
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
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
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
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
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
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
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
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
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
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
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
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
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
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
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
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
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
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

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 runs 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 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 will be 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, 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 mixed 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 as 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 DC-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 using the WRS test is computed by the following equation:

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

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 using 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 5-12 or 5-13. 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:

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

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:

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

Equation 5-14 can be rewritten as follows:

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

Substituting equation 5-18 into 5-10 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:

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

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

ε_i = 0.3 for Co-60
ε_i = 0.38 for Cs-137
ε_s = 0.25 for Co-60
ε_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

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

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 guide 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.

Sample locations may be modified or revised in cases where sampling is impractical (e.g., rock outcroppings where there is no soil) or where there is obstruction, inaccessibility, etc. The FSS survey design will provide a process to relocate sample locations as necessary using the DQO process. The process may be a simple matter of relocating the sample to a nearby adjacent location (e.g., within a circle of one (1) square meter radius) or randomly selecting a new location (e.g., using commercially available software tools). In addition, for Class 2 areas, additional scanning will be performed in the area bounding the original sample location to the extent possible (e.g., obstruction, inaccessibility) when a new location has been determined. Class 3 areas are exempt from the additional scanning requirement because elevated areas of residual radioactivity are not expected and the FSS survey design uses a random grid pattern. Class 1 areas are exempt from the additional scanning requirement because survey design requires 100% scanning coverage.

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	$> 0.5 \times DCGL_W$	Detectable over background.

For Class 1 survey units, measurements above the $DCGL_W$ are not necessarily expected. 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 are 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 collected 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 21')	Concrete Samples Collected to Date (08/31/2006)	Additional Concrete Samples to be Collected	Minimum Total Number Of Samples To be Used for the Inventory Calculation
Containment Mat	14	0	14
Containment Walls	10	0	10
In Core Sump	9	0	9
Spent Fuel Pool	12	0	12
Cable Vault	13	0	13
"B" Switchgear Building	8	0	8
Discharge Tunnels/Structure	10	0	10
Intake Structure	8	0	8

The mechanism that has caused volumetric contamination in concrete is in many cases specific to certain radionuclides. 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 and Eu-152 among others. To adequately assess the volumetric 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.

On September 1, 2004, February 27, 2006, and November 26, 2007 (References 5-14 through 5-16), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004. The information in Table 5-10 has been retained for historical purposes.

With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

5.7.2 Survey Instrumentation

5.7.2.1 Survey Instrument Data Quality Objectives

The data quality objectives process include 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 establishes 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 may 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
ε_i = 0.38 cpm/dpm
ε_s = 0.25 (from ISO 7503-1) emissions per disintegration
A = 100 cm^2

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

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 MARSSIM 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"x 2" 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 (900 cpm/μR/h from MARISSM 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 5-11 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 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 Chapter 6 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 5-11
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 Basement

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 spill 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.

Basement concrete that is 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.

5.7.3.1.2.2 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 the buildings generally 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 will be sampled or assessed as follows:

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 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 the 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 than 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. Following any required remediation approximately 4 feet of backfill soil would be placed over the footings.

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; and
- Buildings shown by characterization sampling to be free of residual radioactivity;

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

5.7.3.1.3 Sewer Systems, Plumbing

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 are 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 below.

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.

5.7.3.1.5 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, they were not obtained in areas 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.

In-situ gamma spectroscopy may be used to perform remediation surveys for activated concrete to determine the radionuclide concentration to be used in the "Basement Fill Model" calculation. 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.

Such surveys would be conducted so that 100% of the affected volume is covered in overlapping measurements. Embedded materials (such as rebar) and activated piping and activated portions of the liner will be treated as concrete for purposes of calculating future groundwater dose with the Basement Fill Model. Assessments for any "hard-to-detect" radionuclides that may be present in activated concrete will be by either direct measurements (core-bores or equivalent) or by establishing surrogate concentrations 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 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 may 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. 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 was 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 well 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.

The sediment samples taken for the discharge canal and permanent wetland areas will be homogenized over depth to the original canal excavation elevation or at least three (3) feet. The discharge canal is influenced approximately two (2) feet by tidal affect. The boundary of the discharge canal survey units is the average high water level. Samples taken into the bank of the discharge canal will be homogenized over a depth of three (3) feet.

This approach will insure that the material deposited above the original canal depth during operation will be sampled. The minimum three (3) feet of depth corresponds to the normal depth to which a clam shell dredge would dig. This is an appropriate minimum sample depth as dredging is the exposure scenario discussed above.

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 than 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 5.12.

Table 5-12
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 5-13.

Table 5-13
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 or 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 average 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 $DCGL_W$ may 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.:

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

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 mean 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 mean 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 may 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 may 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 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);

- Additional appendices will be added as necessary (e.g., Technical Basis Documents containing radiological assessment results, etc.)

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

5.13 Estimated Zone of Influence/Capture zone for Hypothetical Water Supply Wells in Post-Closure Dose Modeling, CH2MHill, Technical Memorandum, dated January 11, 2005.

5-14 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of East Site Grounds from Part 50 License," dated September 1, 2004.

5-15 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of Phase II Areas from Part 50 License," dated February 27, 2006.

5-16 Letter from K. McConnell (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant - Release of Land from Part 50 License," dated November 26, 2007.

Figure 5-1

Site Grounds Grid Map

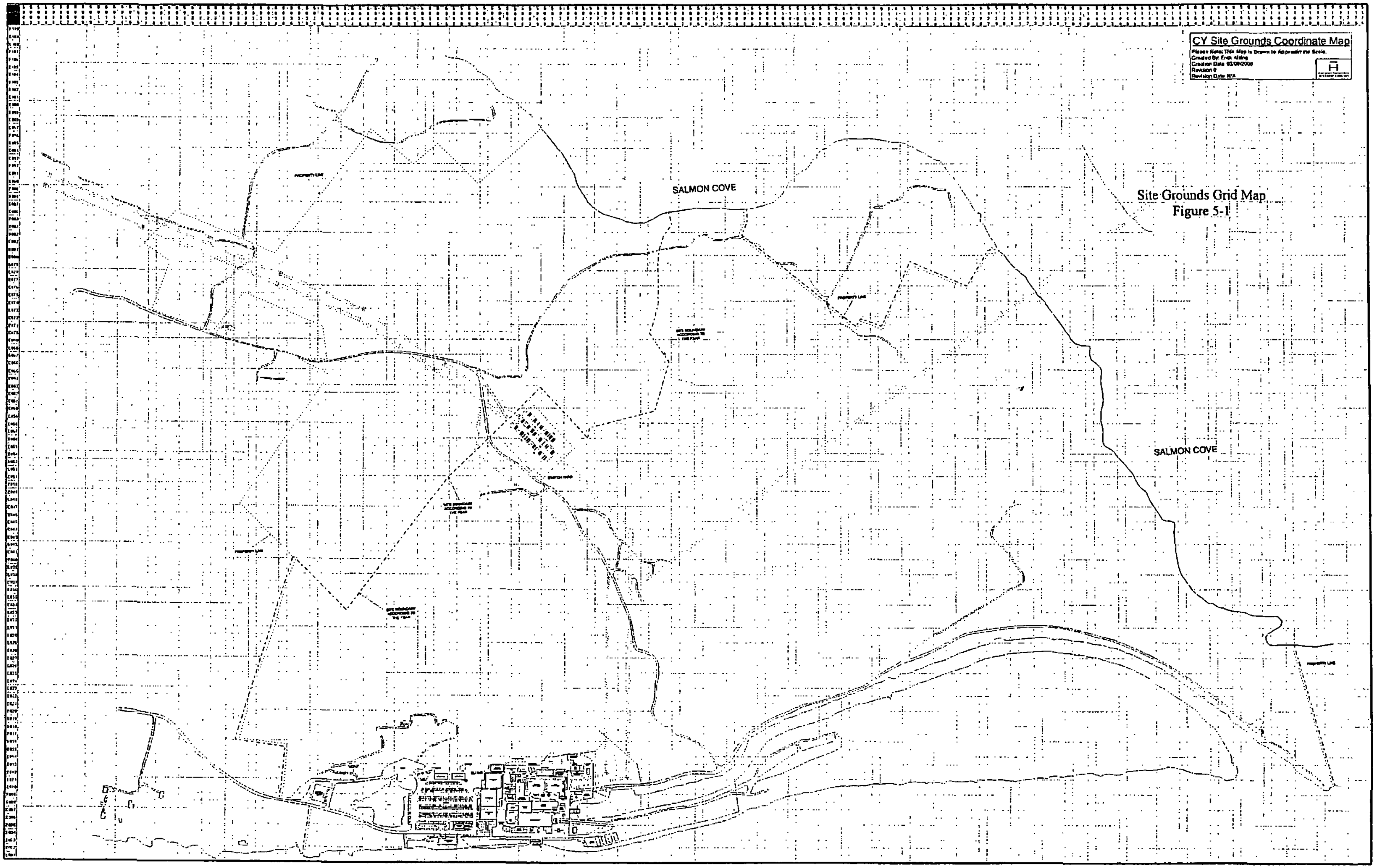


Figure 5-2

Site Grid Map
(Site View)

Site Area Grid Map
Figure 5-2

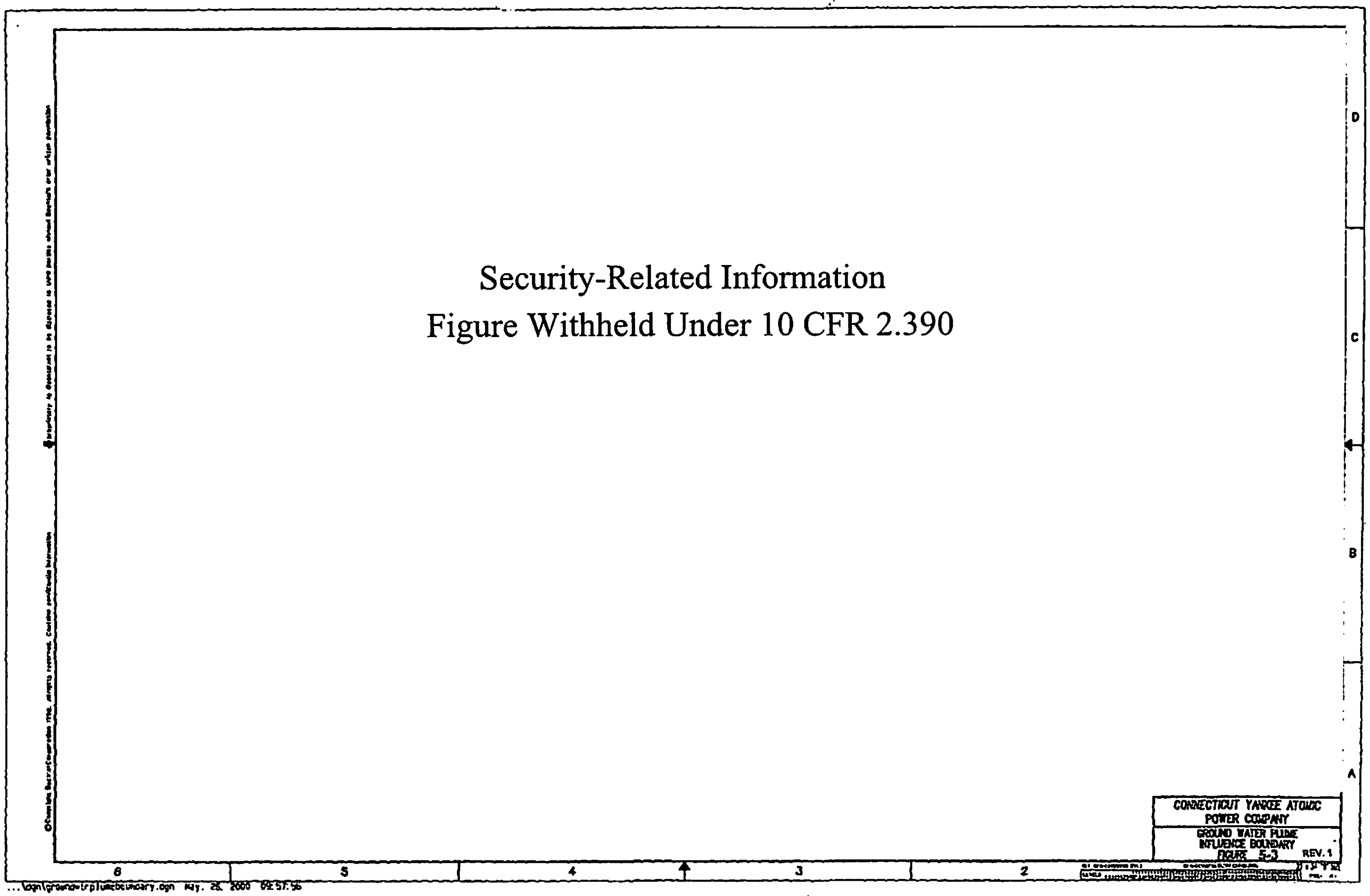
Security-Related Information
Figure Withheld Under 10 CFR 2.390

CONNECTICUT RIVER

Please Note: This Map Is Drawn to Approximate Scale
Created By Erick Alsing
Date: 03/09/2000
Revision 0
Revision Date: N/A
SCALE

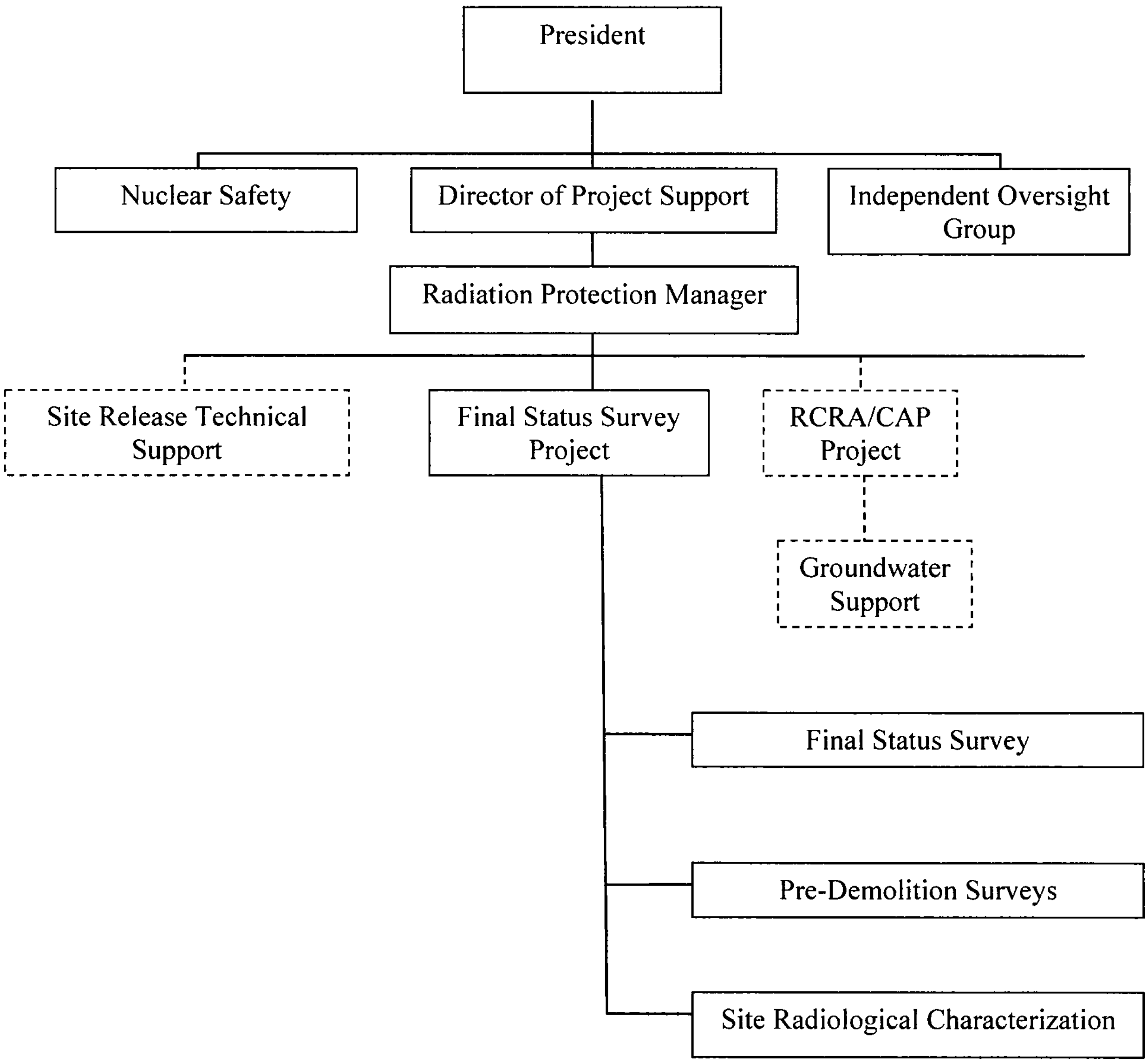
Figure 5-3

Ground Water Plume Influence Boundary



Security-Related Information
Figure Withheld Under 10 CFR 2.390

Figure 5-4
Final Status Survey Organization



6 COMPLIANCE WITH THE RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

6.1 Site Release Criteria

6.1.1 Radiological Criteria for Unrestricted Use

This section provides methods for showing compliance with the radiological criteria of license termination that are based on the site being partially through the decommissioning process. On September 1, 2004, February 27, 2006, and November 26, 2007 (References 6-21 through 6-23), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

The information included within this section includes historical information that will be maintained in its current form. This information will be reviewed, and revised as necessary, at the time of initiating the decommissioning activities for the ISFSI and associated land areas to ensure that appropriate information is available for the implementation of final status survey activities for the ISFSI and termination of the Part 50 License for the HNP site.

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 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 10 CFR 20.1402. 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 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 model. (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 are defined in Section 5.7.1.5.

CYAPCO will not use concrete debris from the demolition of buildings to backfill basements 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 from offsite sources has been demonstrated to be free from plant related radioactivity over background. Additional materials that may be used as backfill are those items determined to be less than the fifteen (15) mrem/year operational DCGL for soil by an in-situ object counting system and/or sampling. Examples of materials found suitable are rip rap stone from the discharge canal, soil removed from excavations to access contaminated soil in the saturated zone and asphalt for reuse on roadways. 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.

On September 1, 2004, February 27, 2006, and November 26, 2007 (References 6-21 through 6-23), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

Physical Characteristics

The industrial area of the HNP was 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 occupied 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 plane 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. The HNP ISFSI is located near the south end of the HNP site and just north of the ridge that parallels the discharge canal.

Geologic and Hydrogeologic Characteristics

The geology and hydrogeology of the industrial area of the HNP was 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 flood plane 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 of the HNP 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 terminated and removed during the decommissioning. The cooling water discharge tunnels divert the shallow groundwater flowing around the southwestern side of the Containment Building farther to the south. Southwest of the tunnels, the shallow groundwater appears to flow southwesterly and directly toward the river.

Contaminant Characteristics

Soil within the industrial area of the HNP 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

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.

6.3.2 Resident Farmer Scenario

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 to determine corresponding DCGLs.

<u>Critical Group:</u>

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, and groundwater 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 individuals 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.

<u>Exposure Pathways:</u>

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 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 are 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 (Reference 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.

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 Equation 6-3 below. Calculation of DCF_{eff}'s for each of the radionuclides is shown in Table G-2-1, Footnote "C".

$$DCF_{eff} = \frac{Dose_{PEAK}}{Conc_{WW}} \qquad \text{(Equation 6-3)}$$

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 from off-site or on-site material shown to meet the soil DCGLs once the final status survey or assessment of the structure has been completed. The site dose contribution from these basements will be calculated using the Basement Fill Model as discussed in Section 6.8.2. Presently, the only remaining building after license termination will be the Emergency Operations Facility (EOF) which will be free released in accordance with applicable station procedures.

6.8.1 DCGLs for Concrete: Buildings Standing

6.8.1.1 Dose Model

Presently the only building to remain on-site is the Emergency Operations Facility (EOF). The plan is to free release this structure in accordance with current regulations. The structure is located outside the industrial area at the extreme north boundary of the site boundary.

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. The likelihood of these structures being occupied is very small and conducting the surveys to the Building Occupany DCGLs allows for the confident use of the contamination levels corresponding to those DGLs 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 G-3.

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 in-core instrumentation area portion of 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 6-20),
- a conservative diffusion and buildup factor (Reference 6-19), 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 6-19 establishes the following relationship to calculate the maximum future groundwater concentration, $C_{w,i}$. from radionuclide i and structure/component s as follows:

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

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), 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. Reference 6-20 was revised to add an additional soil source to be used as leach fill material. The calculations provided here are for the initial two soild sources used as backfill. When the additional soil source is used the K_d, R and B values will be adjusted in Table 6-3 to reflect the soil type being used in the calculations. The retardation values, R (unitless) are calculated as:

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

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 6-19 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 6-3 summarizes the values of K_d, R, and B_i as used in this model and in equations 6-4 and 6-5 for each structure/component.

Table 6-3: Parameter Values for Equations 6-4 and 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 6-19 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:

$$H_{futureGW} = 25\sum\frac{C_{w,i}}{DCGL_{GW,i}} \qquad \text{(Equation 6-6)}$$

It should be noted that the above methodology assumes that concrete, the metal liner and the rebar, piping and other metal items 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.

As discussed in the Reference 6-19 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 (i.e., containment basement). This conservative assumption does not account for additional dilution that will occur from groundwater flow around the subsurface basements.

6.8.2.2 Future Groundwater Dose Calculation for Basement Concrete

From equation 6-4, 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 6-19 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:

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

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, Reference 6-19 selects conservative values of D_i as provided in Table 6-4.

Table 6-4: 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 incore instrument area containment liner provides.
- The Brookhaven Study assumes that 2.5 feet of grout (flowable fill) will be placed above the activated concrete region of the In-Core Sump. The depth of grout placed above the activated region will actually be 5 feet, thereby providing additional resistance to transport.

The analysis provided by Reference 6-19 is specifically for the containment and spent fuel 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 footings assessed after completion of the spent fuel pool Basement Fill Model Calculation will be included with the discharge tunnel inventory. 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 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.

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. Surface activity surveys will be performed in accordance with Section 5.4.7.4 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 a 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.

6.8.2.4 Summary of All Future Groundwater Dose Calculations

Sections 6.8.2.2 and 6.8.2.3 show the method to be used to determine the future groundwater doses due to the individual sources, buried concrete, the in-core instrumentation area steel 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 (5-1). In the case of the containment basement, the future groundwater dose will be calculated as given in the 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.

6.9 Operational DCGLs

Since additional scenarios, beyond those described above, may be created by combining pathways from different scenarios (e.g., resident farmer with the future groundwater dose calculated using the Basement Fill Model), 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 a buried basement, which may also contain residual radioactivity. Soil and groundwater DCGLs for these combined scenarios along with the future groundwater dose calculated using the Basement Fill Model will be determined on an operational basis, using the base case DCGLs for soil and groundwater 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.

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

6-18 Not Used.

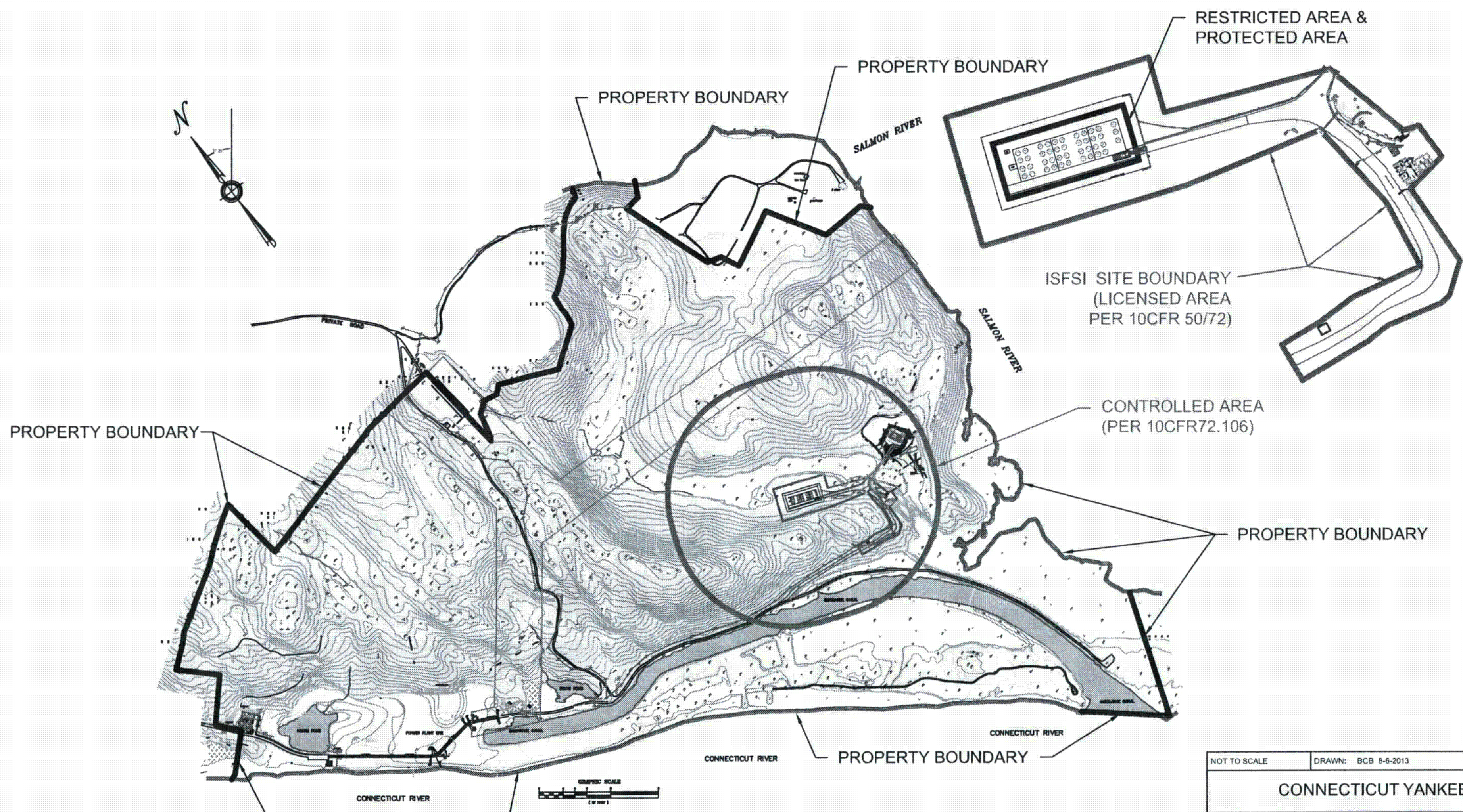
6-19 Technical Support Document CY-HP-0184, Subject: Estimates for Release of Radionuclides from Potentially Contaminated Concrete at the Haddam Neck Nuclear Plant.

6-20 Technical Support Document CY-HP-0185, Revision 1, Subject: K_d Values of Backfill Material for Connecticut Yankee.

6-21 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of East Site Grounds from Part 50 License," dated September 1, 2004.

6-22 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of Phase II Areas from Part 50 License," dated February 27, 2006.

6-23 Letter from K. McConnell (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant - Release of Land from Part 50 License," dated November 26, 2007.



NOTE 1: THE ISFSI SITE CONSISTS OF APPROXIMATELY 5.7 ACRES.

NOTE 2: THE CONTROLLED AREA IS DEFINED AS 300 METERS FROM THE PROTECTED AREA FENCELINE.

NOTE 3: THE PROPERTY CONSISTS OF APPROXIMATELY 525 ACRES, AND IS BOUNDED BY THE CONNECTICUT RIVER TO THE SOUTHWEST AND THE SALMON RIVER TO THE EAST AND NORTHEAST.

NOT TO SCALE	DRAWN: BCB 8-6-2013
CONNECTICUT YANKEE	
GENERAL SITE PLAN	
SHEET 1 OF 1	
CY	FIGURE 6-1

Figure 6-2

Industrial and Peninsula Area Cross Section

Security-Related Information

Figure Withheld Under 10 CFR 2.390

CONNECTICUT YANKEE ATOMIC POWER COMPANY

INDUSTRIAL AND PENINSULA AREA CROSS SECTION FIGURE 6-2

Figure 6-3
Exposure Pathways Considered
in the Resident Farmer Scenario

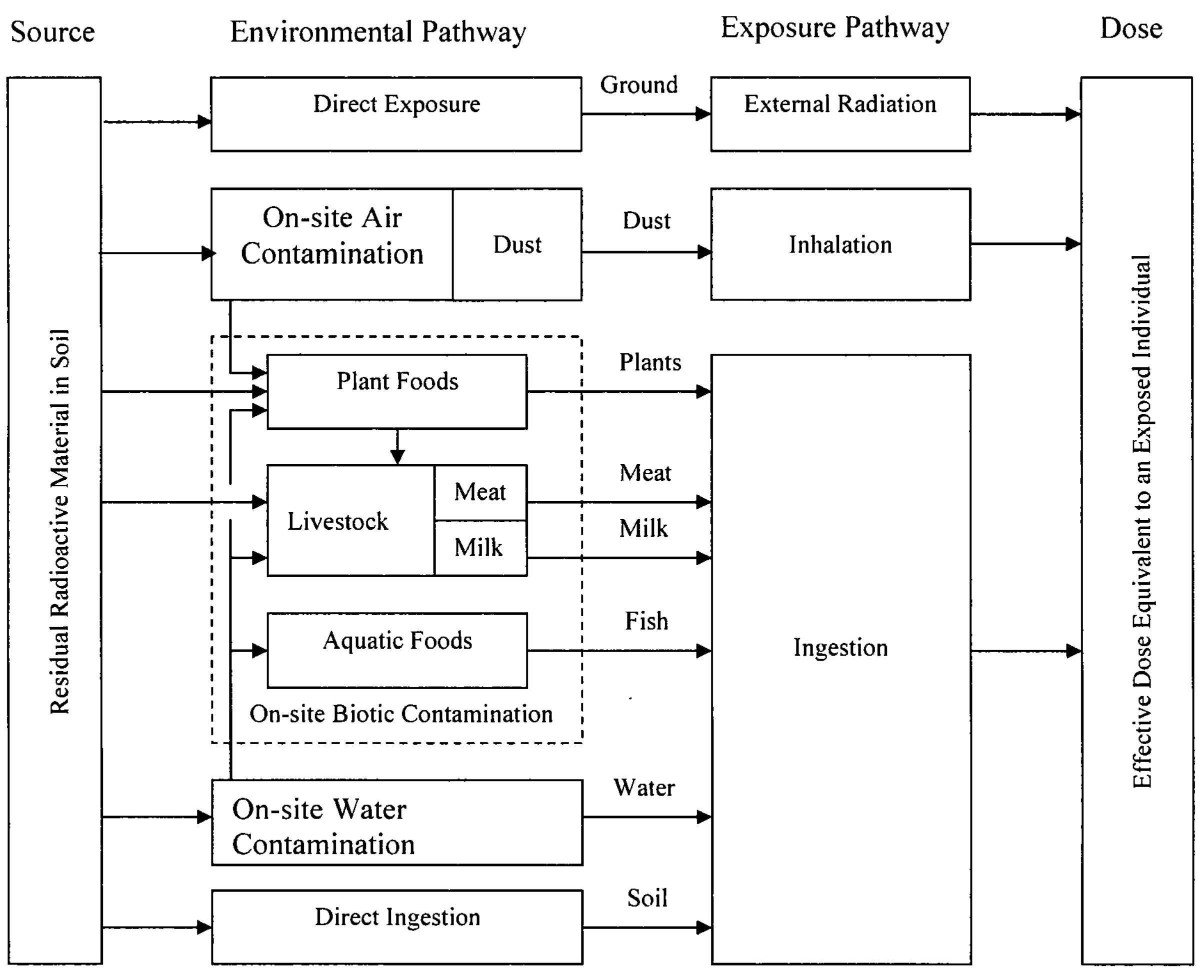
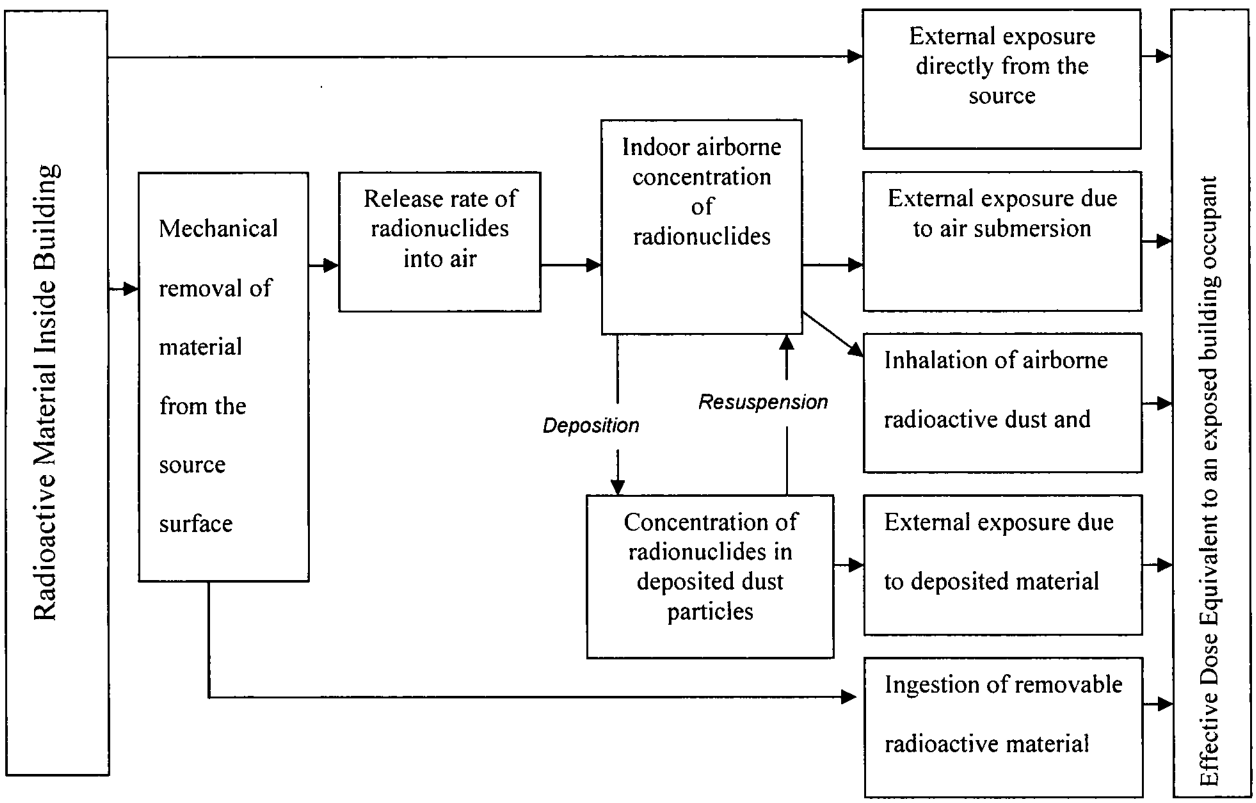
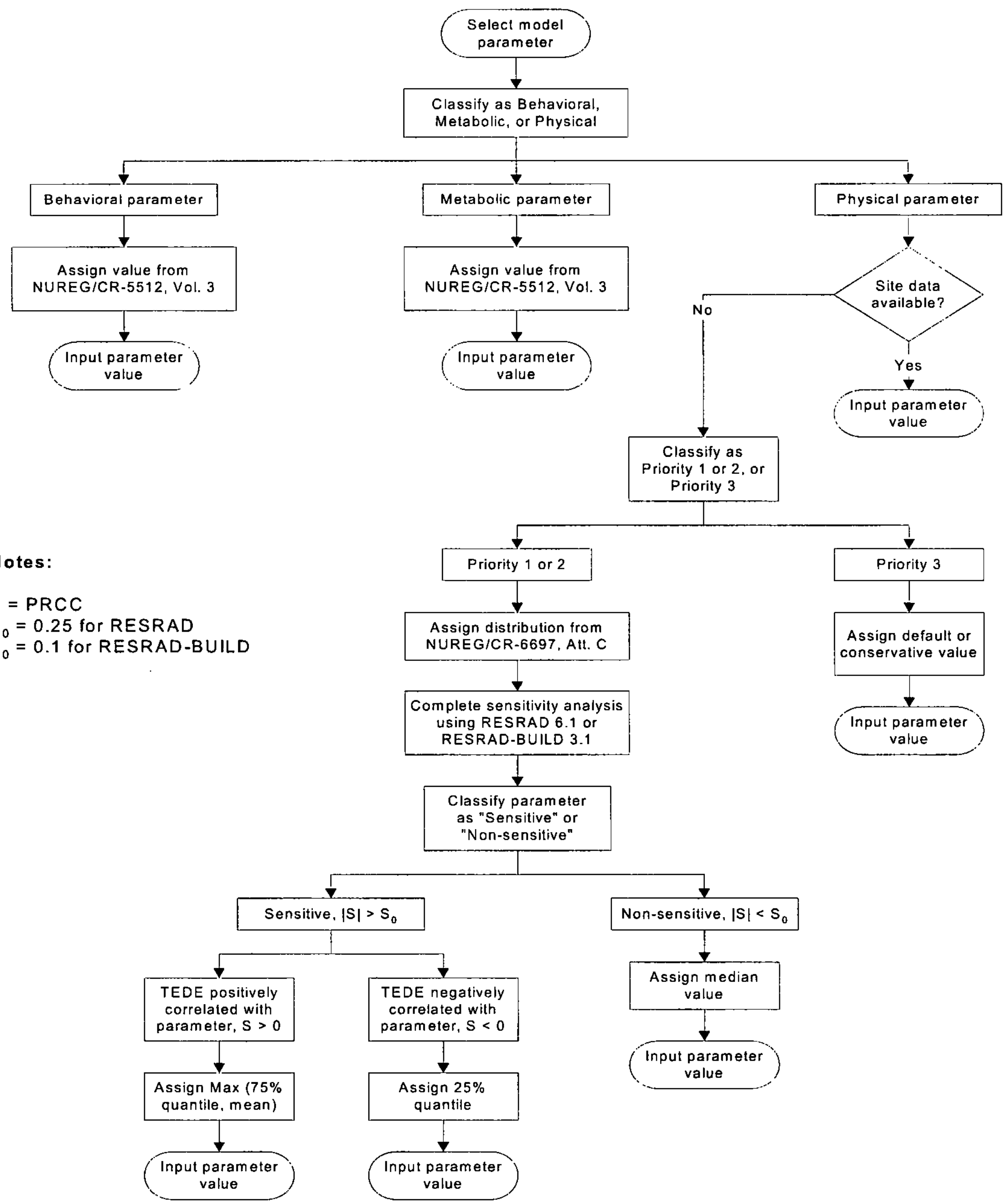


Figure 6-4
Exposure Pathways Considered in the Building Occupancy Scenario



Based on Figure 2.1 of NUREG/CR-6755; modified to reflect exposure pathways considered at HNP

Figure 6-5
Parameter Selection Process



Notes:

S = PRCC
S_0 = 0.25 for RESRAD
S_0 = 0.1 for RESRAD-BUILD

7 UPDATE OF SITE-SPECIFIC DECOMMISSIONING COSTS

7.1 Introduction

This section provides an update of site-specific decommissioning costs that are based on the site as it existed partially through the decommissioning process. On September 1, 2004, February 27, 2006, and November 26, 2007 (References 7-2 through 7-4), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

With the exception of decommissioning activities at the ISFSI to be undertaken when all fuel and GTCC waste have been removed from the site, all decommissioning and dismantlement activities have been completed at this site.

The information included within this section includes historical information that will be maintained in its current form. This information will be reviewed, and revised as necessary, at the time of initiating the decommissioning activities for the ISFSI and associated land areas to ensure that appropriate information is available for the implementation of final status survey activities for the ISFSI and termination of the Part 50 License for the HNP site.

In accordance with 10 CFR 50.82(a)(9)(ii)(F) and Regulatory Guide 1.179, the site specific cost estimates and funding plans are provided. Regulatory Guide 1.179 discusses the details of the information to be presented.

The License Termination Plan (LTP) must:

> Provide an estimate of the remaining decommissioning costs, and compare the estimated costs with the present funds set aside for decommissioning. The financial assurance instrument required by 10CFR50.75 (Reference 7-1) must be funded to the amount of the cost estimate. If there is a deficit in the present funding, the LTP must indicate the means for ensuring adequate funds to complete the decommissioning.

The decommissioning cost estimate should include an evaluation of the following cost elements:

- Cost assumptions used, including contingency
- Major decommissioning activities and tasks
- Unit cost factors
- Estimated decontamination and equipment and structure removal
- Estimated cost of radioactive waste disposal including disposal surcharges
- Estimated final survey costs
- Estimated total costs

The cost estimate should focus on the remaining work, detailed activity by activity, including costs of labor, materials, equipment, energy, and services.

7.2 Decommissioning Cost Estimate

The current Federal Energy Regulatory Commission (FERC) approved decommissioning cost estimate (December 2012) and is based on the April 30, 2013 Stipulation and Settlement Agreement between CYAPCO and the Connecticut Public Utilities Regulatory, the Connecticut Department Office of Consumer Counsel, the Maine Public Utilities Commission, the Maine Office of Public Advocate, the Massachusetts Department of Public Utilities, and the Attorney General of Massachusetts.

This cost estimate includes the cost associated with the projected ISFSI decommissioning costs and a funding assumption of 15 years of operations costs to manage spent fuel and GTCC waste. A funding mechanism provides that damage awards and settlement proceeds that CYAPCO receives in future phases of its litigation with the Department of Energy (DOE) will be applied to maintain the adequacy of the Nuclear Decommissioning Trust (NDT) to cover 15 years of ISFSI operations (as well as all other projected decommissioning costs). In addition, CYAPCO has the right to resume collection of decommissioning charges from its customers subject to the submittal of a proposal under section 205 of the Federal Power Act, if needed.

CYAPCO has an account within its NDT entitled, "ISFSI Radiological Decom," that segregates the funds for radiological decommissioning of the ISFSI from the larger balance of funds for ongoing management of spent fuel and GTCC waste held in the NDT.

The assumptions of the current decommissioning cost estimate are discussed in the Decommissioning Funding Plan submitted to the NRC on December 17, 2012 in accordance with 10 CFR 72.30(b)(2) (Reference 7-5). The decommissioning cost estimate incorporates the most recent assumptions with respect to the remaining decommissioning activities and related costs (i.e., those associated with the HNP ISFSI). The total un-escalated cost estimate for decommissioning the ISFSI, including contingency is $19.4 million, which includes $17.7 million for radiological removal and $1.8 million for non-radiological removal. The decommissioning cost estimate is in 2013 dollars.

ISFSI operations will continue until DOE removes the spent fuel and GTCC waste, allowing for the decommissioning of the ISFSI. CYAPCO expects that the ISFSI operating costs will continue to cover a number of categories, including payments for the storage of wet fuel at the General Electric facility in Morris, Illinois, regulatory fees, and costs for insurance, labor, security, materials and supplies, miscellaneous expenses, outside services, property taxes, regulatory fees, rentals and leases and utilities. The un-escalated cost estimate for the management of spent fuel and GTCC waste from 2013 through 2031, including contingency, is $248.3 million. The cost estimate is in 2013 dollars. This is based on the estimate submitted to FERC on May 1, 2013 (Reference 7-6).

The total un-escalated cost estimate is approximately $267.7 million for decommissioning the ISFSI and managing the storage of spent fuel and GTCC waste for the time period of 2013 through 2033.

CYAPCO will continue to inform the NRC regarding the status of this funding by complying with the obligations defined in: 1) 10 CFR 50.75(f)(1) and (2) to submit an annual Decommissioning Funding Status Report; 2) 10 CFR 50.82(a)(8)(v) to submit an annual financial assurance status report regarding decommissioning funding; 3) 10 CFR 72.30(c) to resubmit the decommissioning funding plan at intervals not to exceed three years; and 4) 10 CFR 50.82(a)(8)(vii) to submit an annual report regarding the status of the funding for managing irradiated fuel.

7.3 References

7-1 Code of Federal Regulations, Title 10, Part 50.75, "Reporting and Recordkeeping for Decommissioning Planning."

7-2 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of East Site Grounds from Part 50 License," dated September 1, 2004.

7.3 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of Phase II Areas from Part 50 License," dated February 27, 2006.

7.4 Letter from K. McConnell (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant - Release of Land from Part 50 License," dated November 26, 2007.

7-5 Letter from C. Pizzella (CYAPCO) to Document Control Desk (NRC), "Independent Spent Fuel Storage Installation Decommissioning Funding Plan," dated December 17, 2012.

7-6 Letter from Alston & Bird LLP to FERC, "Connecticut Yankee Atomic Power Company Docket No. ER13-____-000," dated May 1, 2013.

8 SUPPLEMENT TO THE ENVIRONMENTAL REPORT

8.1 Introduction

8.1.1 Overview

The Connecticut Yankee Power Company Decommissioning Environmental Review (Reference 8-1), dated 1997, was prepared and submitted in conjunction with the HNP Post-Shutdown Decommissioning Activities Report (Reference 8-2). The Environmental Review was previously provided to Federal and State agencies. The report concluded that the environmental impacts of decommissioning activities are bounded by previously issued environmental impact statements—NUREG-0586, "Final Generic Environment Impact Statement (FGEIS) on Decommissioning of Nuclear Facilities," (Reference 8-3); Final Environmental Statement, Haddam Neck Nuclear Power Plant, Docket No. 50-213, October, 1973 (Reference 8-4); and "Environmental Assessment for Proposed License Extension," dated November 23, 1987 (Reference 8-5).

The purpose of this section of the LTP is to describe any new information on significant environmental impacts associated with site-specific license termination activities and to determine if these impacts are within the scope of the environmental impacts previously evaluated either generically or on a site-specific basis by:

1. the environmental impact statement developed in support of the original facility,
2. the environmental impacts described in conjunction with the Decommissioning Plan (and PSDAR) related to decommissioning activities, or
3. The Final Generic Environmental Impact Statement addressing decommissioning (NUREG-0586).

The NRC has issued guidance associated with the impacts of decommissioning, including Supplement 1 to NUREG-0586 (Reference 8-6). Supplement 1 to NUREG-0586 focuses on the impacts of decommissioning nuclear power reactors licensed by the NRC, unlike the 1988 FGEIS, which took a broad look at decommissioning of a variety of sites and activities.

Supplement 1 to NUREG-0586 is intended to consider, in a comprehensive manner, all aspects related to the radiological decommissioning of nuclear reactor facilities. Supplement 1 uses an approach that defines a measure of significance and severity of potential environmental impacts and an applicability of these impacts to a variety of facilities. The significance of an impact is described as being SMALL, MODERATE, or LARGE. The applicability of impacts is described as being generic or site-specific. These terms are clearly defined in Section 4 of Supplement 1 to NUREG-0586.

Table H-1, located in Appendix H to Supplement 1 of NUREG-0586, provides a listing of activities for which the NRC has generically determined that no environmental impacts exist. Because these activities have already been determined not to result in environmental impacts, no further review is required in connection with the LTP.

Table H-2 provides a summary of the decommissioning activities and associated environmental issues that have been determined to have *potential* impacts. As stated in Section 4.3 of Supplement 1 to the NUREG-0586, if these plant-specific impacts fall within the scope of the environmental impacts previously identified and evaluated by the NRC staff, these activities can be performed without further evaluation. The issues identified in Table H-2 to be evaluated for plant-specific impacts are:

- Onsite/offsite land use
- Water use
- Water quality
- Air quality
- Aquatic ecology
- Terrestrial ecology
- Threatened and endangered species
- Radiological
- Radiological accidents
- Occupational
- Socioeconomics
- Environmental justice
- Cultural impacts
- Aesthetics
- Noise
- Transportation
- Irretrievable resources.

According to Supplement 1 to NUREG-0586, the NRC assessed the impacts of each of these issues using data from previous studies and environmental reviews in addition to information obtained during site visits and provided by plants undergoing decommissioning. The NRC then examined the cumulative impacts of decommissioning activities and other past, present, and reasonably foreseeable future activities at the sites. After analyzing the issues, the NRC determined the impact of each and assigned a significance level (SMALL, MODERATE, or LARGE).

The NRC also determined whether the analysis of the environmental issues could be applied to all plants. Each environmental issue identified was assigned one of the following two categories: generic or site-specific.

Generic issues met the following three criteria:

1. The environmental impacts associated with the issue have been determined to apply to all plants, or, for some issues, to a group of plants of a specific size, specific locations, or having a specific type of cooling system or site characteristic.

2. A single significance criterion (SMALL, MODERATE, or LARGE) has been assigned to describe the impacts.

3. Mitigation of adverse impacts associated with the issue has been considered in the analysis, and it has been determined that additional plant-specific mitigation measures are likely not to be sufficiently beneficial to warrant implementation.

If one or more of the above criteria cannot be met, the issue is considered to be "site-specific" and a site-specific evaluation of the issue is required. Table 8-1 summarizes the NRC's findings with respect to applicability and impact of the identified environmental issues pertinent to decommissioning.

Decommissioning and license termination activities at HNP fall within the range of activities evaluated for the FGEIS and NUREG-0586, Supplement 1. For those issues identified as "generic" in Table 8-1, the NRC's prior conclusions bound environmental impacts at HNP from decommissioning and license termination.

The LTP addresses the issues identified in Table 8-1 as "site-specific." In addition, consistent with regulatory guidance, the review focuses on any new information or significant environmental change associated with site-specific termination issues. Impacts associated with site-specific termination activities have been compared to previously analyzed decommissioning and termination activities, in this LTP and its references. The proposed termination activities related to the end use of the site do not result in significant environmental changes that are not bounded by the site-specific decommissioning activities described in the Decommissioning Plan, PSDAR, the FGEIS, or NUREG-0586.

Note that the review and conclusion in this Section relate only to activities and impacts associated with termination of the NRC license. CYAPCO is conducting other site characterization at HNP for non-radiological remediation and site restoration, which are not part of the license termination activities and are outside of the scope of NRC regulation. The non-radiological remediation activities are addressed in the RCRA Corrective Action Program (CAP) that is being reviewed by the EPA. The Property Transfer Program regulations from the Connecticut Department of Environmental Protection will be used to achieve site closure. This program will incorporate the remediation activities that occurred under the LTP and RCRA CAP. Other agencies, such as the Connecticut Department of Public Health, the Army Corps of Engineers and the Haddam Wetlands Commission are also routinely involved in aspects of non-radiological site remediation.

8.1.2 Proposed Site Conditions at the Time of License Termination

The HNP site is intended to be released for unrestricted use, under the radiological release criteria of 10CFR20.1402 (Reference 8-7) upon termination of its NRC license. Sections 3 and 4 of this LTP discuss in greater detail the activities that have been completed, those ongoing and remaining, and the proposed final state of the site.

8.1.3 Remaining Dismantlement and Decommissioning Activities

8.1.3.1 General Description of Remaining Dismantlement and Decommissioning Activities

On September 1, 2004, February 27, 2006, and November 26, 2007 (References 8-20 through 8-22), the NRC issued Safety Evaluation Reports that released the majority of the site from the 10 CFR 50 License. As a result, the only areas of the site that remain within the control of the 10 CFR 50 License are those areas associated with the HNP ISFSI, portions of Survey Units 9523-0000, 9528-0000, and 9528-0004.

Currently, CYAPCO is managing the storage of spent fuel and GTCC waste at the HNP ISFSI. Following the removal of the spent fuel and GTCC waste from the site by the Department of Energy, the decommissioning activities associated with the ISFSI and associated land areas will be undertaken. CYAPCO continues to implement the DECON alternative as the most appropriate alternative for decommissioning the HNP site. Evaluation of the environmental effects of the DECON alternative is contained in NUREG-0586 and its supplement.

8.1.3.2 Conclusions Regarding Environmental Impact Included in the PSDAR

The PSDAR includes a discussion of environmental impacts from decommissioning. This information was based upon NUREG-0586; "The Connecticut Yankee Power Company Decommissioning Environmental Review," dated 1997; the "Final Environmental Statement, Haddam Neck Nuclear Power Plant, Docket No. 50-213," dated October 1973; and the "Environmental Assessment for Proposed License Extension," dated November 23, 1998.

- The PSDAR concluded that the impacts due to decommissioning would be bounded by the previously issued environmental impacts statements. This was principally due to the following reasons: The postulated impacts associated with the method chosen, DECON, have already been considered in Supplement 1 to NUREG-0586.

- There are no unique aspects of the plant or HNP ISFSI or decommissioning techniques to be utilized that would invalidate the conclusions reached in Supplement 1 to NUREG-0586.

- The methods to be employed to dismantle and decontaminate the site (including the HNP ISFSI) are standard construction-based techniques fully considered in Supplement 1 to NUREG-0586.

- The site-specific person-rem estimate for all decommissioning activities has been conservatively calculated using methods similar to those used in Supplement 1 to NUREG-0586.

Specifically, the review concluded that the HNP and HNP ISFSI decommissioning will result in generally positive environmental effects, in that:

- Radiological sources that create the potential for radiation exposure to site workers and the public will be minimized.

- The site will be returned to a condition that will be acceptable for unrestricted use.

- The thermal impact on the Connecticut River from facility operations will be eliminated.

- Noise levels in the vicinity of the facility will be reduced.

- Hazardous material and chemicals will be removed.

- Local traffic will be reduced (fewer employees, contractors and materials shipments than required to support an operating nuclear power plant).

Furthermore, the HNP and HNP ISFSI decommissioning will be accomplished with no significant adverse environmental impacts in that:

- No site specific factors pertaining to HNP and HNP ISFSI will alter the conclusions of Supplement 1 to NUREG-0586
- Radiation dose to the public will be minimal.
- Radiation dose to decommissioning workers will be a fraction of the operating exposure.
- Decommissioning is not an imminent health or safety problem and will generally have a positive environmental impact.

Revisions 0 through 2 of the PSDAR estimated the total occupational exposure (excluding public and transportation dose) for the proposed decommissioning activities to be approximately 935 person-rem. Since the original estimate was made, a significant amount of the decommissioning tasks have been completed. An estimate of the total occupational exposure as of May 2006 is within 10% of the original estimate (Reference 8-8). This estimate used the actual occupational exposure associated with completed decommissioning tasks and provides estimates for those tasks yet to be completed. The maximum estimate of the total occupational exposure is still bounded by the 1,115 person-rem exposure estimate in the FGEIS for the reference pressurized reactor. The occupational dose associated with the decommissioning of the ISFSI is not expected to contribute significantly to the overall occupational dose associated with decommissioning the HNP site.

Radiation exposure due to transportation of radioactive waste has been conservatively estimated to be approximately 71 person-rem. This value is bounded by the estimate in Supplement 1 to NUREG-0586 for occupational exposure for transport of radioactive material. All of the material associated with the Vertical Concrete Casks and the ISFSI storage pad is assumed in the decommissioning cost estimate to be shipped offsite as Low Level Radioactive Water (LLRW). This waste is considered to be very low activity waste, and is not addressed in the transportation dose analysis.

Radiation exposure to offsite individuals for expected conditions, or from postulated accidents is bounded by the Environmental Protection Agency's Protective Action Guidelines and NRC regulations. The public exposure due to radiological effluents will continue to remain well below the 10CFRPart 20 limits and the ALARA dose objectives of 10CFR50, Appendix I. This conclusion is supported by the HNP Annual Effluent

Release Reports in which individual doses to members of the public are calculated for station liquid and gaseous effluents.

The total volume of HNP low-level waste (LLW) was estimated to be 283,117 cubic feet in Revisions 0 and 1 of the PSDAR. The scenario associated with this estimate involved license termination with many site buildings remaining onsite. Since that time, the decommissioning approach has been modified, based upon lessons learned at other decommissioning facilities. The modified approach assumes that these buildings will be demolished and disposed of as radwaste, prior to license termination. The change in decommissioning approach increased the estimated volume of LLW waste to approximately 1,158,000 cubic feet. This estimate exceeds the FGEIS LLW volume for the reference PWR by approximately 79% and the estimate in Supplement 1 to NUREG-0586 by 228%. Thus, the bases for the conclusions in the FGEIS were investigated to determine if the newly estimated volume at HNP would invalidate the conclusions in the FGEIS.

The change in the waste volume is a result of the decision to demolish the buildings and ship a large portion of them as radioactive waste, prior to license termination. The additional source term would be that which would have met the 25 mrem/yr criteria and would have remained onsite after license termination. Appendix K to Supplement 1 to NUREG-0586 classifies this type of waste as "Very Low Activity Waste" and states that "the activity estimates for very low level activity waste are sufficiently small that the activity may be neglected in the evaluation of radiological impacts of transportation of LLW." Approximately 1,086,000 cubic feet of waste from the HNP and the waste from the Vertical Concrete Casks and ISFSI storage pads is considered to be very low level activity waste, and thus can be ignored in the evaluation of radiological impacts of transportation of this waste. Thus, the increase in the waste volume due to the change in decommissioning approach does not increase the occupational, public, or "on-looker" dose for the decommissioning.

The FGEIS evaluates the impact of LLW waste from decommissioning in the context of the commitment of radwaste disposal space and dose to the public. The commitment of radwaste disposal space is related to the amount of LLW generated and requiring disposal. The FGEIS (Section 4.4) estimates the commitment of LLW disposal space based upon a volume of 18,340 cubic meters (647,600 cubic feet) of LLW, assuming shallow-land burial in standard trenches. The FGEIS concluded that 2 acres of radioactive waste disposal space would be required for the disposal. The FGEIS concluded that the environmental impact of the disposal would not be significant because the amount of radioactive waste disposal is small when compared to the amount of acreage associated with use of the plant site. The basis and conclusions remain valid for HNP. Based upon the amount of LLW estimated to require disposal from the HNP (1,158,000 cubic feet), approximately 3.5 acres of LLW disposal space would be required. This value is increased by the decision to rip and ship all of the material associated with the Vertical Concrete Casks and the ISFSI storage pad. However, the

total disposal space that would be utilized represents a small fraction, when compared to the 525 acres associated with the HNP site.

No significant environmental impacts are anticipated in the event that LLW is required to be temporarily stored onsite because adequate storage space exists and LLW storage will be in accordance with all applicable federal and state regulations.

The non-radiological environmental impacts from decommissioning are temporary and are not significant. The largest occupational risk associated with decommissioning HNP is related to the risk of industrial accidents. The primary environmental effects are short term: small increases in noise levels and fugitive dust in the immediate vicinity of the site, as well as truck traffic to and from the site for hauling equipment and waste. No socioeconomic impacts, other than those associated with the cessation of operations (loss of jobs and taxes) have been identified. Also, no significant impacts to local culture, terrestrial or aquatic resources, such as the Connecticut River have been identified.

8.2 Analysis of Site-Specific Issues

8.2.1 Onsite-Offsite Land Uses

8.2.1.1 Onsite Land Uses

The environmental impacts associated with onsite land uses have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of onsite land uses is documented in Section 4.3.1 of Supplement 1 to NUREG-0586.

The HNP site is approximately 525 acres. A small fraction of that area had been developed for plant and HNP ISFSI use. Decommissioning activities utilize the same areas used during initial construction and operations. The use of a small fraction of the total site area land impacted by decommissioning and the re-use of areas used during initial construction are consistent with the NRC's assumptions in Supplement 1 to NUREG-0586, and thus there are no significant environmental impacts associated with HNP decommissioning.

CYAPCO has identified no new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

8.2.1.2 Offsite Land Uses

Only areas within the existing site boundary will be used to support decommissioning and license termination activities (such as temporary storage areas and staging areas). As discussed previously in this section, and in detail in Chapter 5, isolation and control measures will be instituted to prevent the spread of contamination. These measures will also be monitored to ensure their effectiveness. Thus, no environmental impacts associated with the use of offsite lands are anticipated from HNP and HNP ISFSI decommissioning and license termination activities.

8.2.2 Water Use

The environmental impacts associated with water use, during decommissioning, have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of water use is documented in Section 4.3.2 of Supplement 1 to NUREG-0586.

During plant operation, approximately 372,000 gpm was diverted from the Connecticut River and used to cool plants systems in a "once-through" condenser cooling system (Reference 8-9). Environmental reviews associated with operation of the plant identified no significant environmental impacts associated with water use for the operating facility.

The systems that contributed to water usage have been removed from operation, dismantled, and decommissioned.

Use of water for decontamination of systems such as the Reactor Coolant System and the Spent Fuel Pit are addressed in the FGEIS. Other water usage, such as for dust abatement, are similar to those that occurred during construction of the plant. Potable water for the decommissioning contractor staff was provided via bottled water, and sanitary services are provided by contracted services through the use of enclosed sanitary systems.

In summary, the conditions for HNP and HNP ISFSI decommissioning are consistent with the assumptions of Supplement 1 to the NUREG-0586, and thus there are no significant environmental impacts associated with water use during the decommissioning of the HNP. CYAPCO has not identified any new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

8.2.3 Water Quality

The environmental impacts associated with surface water quality have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of surface water quality is documented in Section 4.3.3 of Supplement 1 to NUREG-0586.

All discharges from the HNP were controlled under the National Pollutant Discharge Elimination System (NPDES) permit (Reference 8-10) or stormwater permits. These permits were issued by the U.S. Environmental Protection Agency (EPA). The Offsite Dose Calculation Manual (Reference 8-11) also addresses limitations on radiological doses to members of the public from liquid effluent and requires that they be maintained below the limits in:

- 10CFR50, Appendix I;
- 10CFR20, Appendix B, Table II, Column 2 (pre-1994 version); and
- 40CFR190.

Radiological impacts are being assessed and monitored by use of on- and offsite groundwater monitoring wells for aquifers that discharge to the Connecticut River. A detailed discussion about future groundwater assessments and historical data are provided in Chapter 2 of this LTP.

As previously discussed, site buildings are generally being removed to four feet below grade, and any remaining basements are being remediated to meet the appropriate Derived Concentration Guideline Levels (DCGLs). Contaminated concrete debris from demolition of the buildings will be removed from the site and disposed of at an appropriate facility. This contaminated debris will not be used as backfill at the site, and thus does not have the potential to affect ground or surface water quality.

The conditions for HNP and HNP ISFSI decommissioning are consistent with the assumptions of Supplement 1 to the NUREG-0586, and thus there are no significant environmental impacts associated with surface water quality during the decommissioning of HNP. CYAPCO has not identified any new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

8.2.4 Air Quality

The environmental impacts of decommissioning associated with air quality have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of air quality is documented in Section 4.3.4 of Supplement 1 to the NRUEG-0586.

Supplement 1 to the NUREG-0586 identifies the following decommissioning activities as having the potential for non-radiological impacts on air quality:

- Worker transportation to and from the site,
- Dismantling of systems and removal of equipment,
- Movement and open storage of materials onsite,
- Demolition of buildings and structures, and
- Shipment of material and debris to offsite locations.

<u>Worker transportation:</u> Consistent with the assumptions in the FGEIS, the work force at the HNP site decreased from the time the plant ceased operation. The work force will further decrease as decommissioning nears completion. There will and have been occasional increases during specific decontamination and decommissioning activities. The work force during decommissioning is smaller than that associated with plant construction and refueling at HNP. Accordingly, the adverse changes in air quality, associated with changes in worker transportation, will not be detectable and are not destabilizing.

<u>Dismantling systems and removal of equipment:</u> Generation of particulate matter associated with the physical activities of dismantlement and by the release of gases from systems during removal are potential sources that could impact air quality. Methods and provisions are available to minimize fugitive dust (e.g., wet suppression) and to minimize

airborne contamination in buildings (e.g., isolation of areas and HEPA filtration). Localfiltration systems may also be used when activities are located in areas that are not ventilated to the plant stack, and are likely to generate airborne radioactivity. Thus, it is highly unlikely that particulate matter generated during decommissioning and released to the environment will be detectable offsite. Any refrigerants will be disposed of in accordance with the applicable state and federal regulations.

Movement and open storage of materials onsite: Movement of equipment and open storage of materials during decommissioning may result in fugitive dust. Provisions as discussed in Chapter 3 and identified above can mitigate these effects. Thus, it is highly unlikely that particulate matter generated as a result of movement or storage of material onsite will be detectable offsite.

Demolition of buildings or structures: As discussed in the FGEIS, demolition of structures and buildings on the HNP site may result in a temporary increase in fugitive dust. The controlled dismantlement and packaging of site components and structures will minimize the potential for fugitive dust from becoming an ambient air quality concern during decommissioning. Fugitive dust from demolition of buildings and structures generally involves large particles that settle quickly. Dust and smaller particles will be controlled using mitigation methods such as wet suppression. Thus, it is highly unlikely that particulate matter generated as a result of building or structure demolition will be detectable offsite.

Shipments of material to an offsite location: Material, debris, and equipment will be removed from the site during decommissioning. The remaining number of shipments to be sent during decommissioning are those associated with the Vertical Concrete Casks and ISFSI Storage pad. These shipments will occur over one or two years, and thus the average number of shipments per day will be relatively small. As stated in the FGEIS, it is unlikely that the emissions associated with the small number of daily shipments would be detectable offsite.

Air effluent released from the site is monitored in accordance with the Radiological Effluent Monitoring Manual (REMM) which sets limits on doses caused by effluents, based upon the ALARA (as low as reasonably achievable) objectives of 10CFR50.34a, 10CFR50.36a, and Section IV.A of Appendix I to 10CFR50. Effluents are reported annually to the NRC.

Based upon the above considerations, it has been determined that the conclusions of the FGEIS are applicable to HNP, and decommissioning of HNP will not noticeably affect offsite air quality. CYAPCO has not identified any new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

8.2.5 Aquatic Ecology

8.2.5.1 Activities within the Operational Area

The environmental impacts associated with aquatic ecology for decommissioning activities within the operational area have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of aquatic ecology for activities within the operational area is documented in Section 4.3.5 of Supplement 1 to NUREG-0586

8.2.5.2 Activities Outside of the Operational Area

The FGEIS identifies generation of runoff due to ground disturbances and surface erosion as having the potential to impact aquatic resources. Provisions will be made to reduce surface erosion and runoff by appropriate environmental controls and implementation of the site stormwater pollution prevention program.

It is understood that decommissioning of shoreline and in-water structures has the potential to impact aquatic habitats and biota. CYAPCO consulted with regulatory and resource agencies to obtain permits and plan activities to minimize the duration and extent of these impacts. Regardless, impacts would be limited to those areas previously disturbed during construction and operation, and these areas would be expected to re-colonize as they did following initial construction. Thus, even considering the removal of shoreline and in-water structures, the impacts of decommissioning on aquatic ecology are minimal.

CYAPCO has not identified any new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

8.2.6 Terrestrial Ecology

8.2.6.1 Activities within the Operational Area

The environmental impacts of decommissioning associated with terrestrial ecology for activities within the operational area have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of terrestrial ecology for activities within the operational area is documented in Section 4.3.6 of Supplement 1 to the FGEIS.

8.2.6.2 Activities Outside the Operational Area

Only areas within the existing site boundary that were used during plant operations are being used to support decommissioning and license termination activities (such as temporary storage areas and staging areas). These areas are within those areas that were disturbed during initial construction and operation. The FGEIS states that terrestrial habitats disturbed during the construction of the site often continue to be of low habitat quality during operation and decommissioning.

As discussed previously in this section, and in detail in Chapter 5, isolation and control measures will be instituted to prevent the spread of contamination, and these measures will be monitored to ensure their effectiveness. Because the HNP site has been in active decommissioning since the decision to permanently close the facility was made, it is reasonable to conclude that areas disturbed during the construction and operation of the plant have not become new sensitive areas with respect to terrestrial biota. Thus, no environmental impacts associated with the use of offsite lands are anticipated from HNP and HNP ISFSI decommissioning and license termination activities related to the end use of the site.

8.2.7 Threatened and Endangered Species

CYAPCO has reviewed the Natural Diversity Data Base (Reference 8-12) compiled by the CTDEP. In addition, a report has been prepared for CYAPCO's contractor, CLF Ventures, Inc., by Ickthyological Associates, Inc., to assess the aquatic resources on the Haddam Neck Plant site (Reference 8-13).

The report by Ickthyological Associates identified twelve species as being either endangered, threatened, or special species of concern in Connecticut. These were two fish species (the shortnose sturgeon and the Atlantic sturgeon); the bald eagle; six species of macroinvertebrates (the bronze copper, the midland clubtail, little bluet, the tidewater mucket, the eastern pond mussel, and the wooland pond snail and possibly the yellow lampmussel); and three plants (the swamp cottonwood, the smooth hedge-nettle, and the arrow leaf). The species present at the HNP site are: the eastern pondmussel, the tidewater mucket, and the swamp cottonwood.

A survey performed by certified biologists was conducted on April 2, 2004. The survey found two relic shells of the tidewater mucket and one of the eastern pondmussel in the vicinity of the Intake Structure. One live pondmussel was found approximately 50 feet upstream of the Intake Structure and approximately 20 feet from the shore. The live pondmussel was photographed and then returned to the river bottom close to its original location. It was noted the survey also identified the presence of 100 to 200 of each of three common native species of fresh water mussel in the Intake Structure.

Based upon the survey results and on discussions with wildlife biologists, it has been concluded that there is no significant viable population of the tidewater mucket or the eastern pondmussel in the vicinity of the Intake Structure.

Dr. Priscillia Baillie, botanist and ecologist, later identified in a July 10, 2003, report to Tighe and Bond (Reference 8-14) that the species of arrow leaf located on the HNP site was the more common *Sagittaria montevidentsis* ssp. *Sponiosus* rather than the state-listed plant *Sagittaris subulata.*

With respect to other listed plant species, it was noted in Dr. Baillie's report that proposed decommissioning plans to deepen the canal would remove non-native invasive plant species from the canal and would remove the soft substrate, which is rich in organics. The effect of the removal of these harmful invasive species would more than offset the loss a relatively small number of native plants in the area. Removal of the sediments would curtail fill-in of the canal and would prevent the consumption of much of the available oxygen in the water by decomposition of the organic materials. The Swamp Cottonwood are located away from the areas affected by active decommissioning (including the ISFSI and ISFSI haul road). Thus, there would be no significant environmental impact on the identified listed plant species.

Field walk downs are performed in areas of the site undergoing decommissioning or remediation to verify that additional endangered or threatened species are not present.

Thus, decommissioning and license termination activities at the HNP site does not adversely impact threatened or endangered species.

8.2.8 Radiological

8.2.8.1 Activities Resulting in Occupational Doses to Workers

The environmental impacts associated with radiological activities resulting in occupational doses to worker have been determined by the NRC to be generically applicable with a SMALL impact, because of the existence of guidance regulating doses to workers (10CFR20) which remain applicable to the HNP and HNP ISFSI. The NRC's analysis of the environmental impacts of radiological activities resulting in occupational doses to workers is documented in Section 4.3.8 of Supplement 1 to NUREG-0586.

8.2.8.2 Activities Resulting in Doses to the Public

The environmental impacts associated with radiological activities resulting in doses to the public have been determined by the NRC to be generically applicable with a SMALL impact, because of the existence of guidance regulating and documenting doses to members of the public (10CFR20). The NRC's analysis of the environmental impacts of radiological activities resulting in doses to the public is documented in Section 4.3.8 of Supplement 1 to NUREG-0586. CYAPCO has not identified any new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

Potential doses to the public following license termination are not covered by the Supplement to the FGEIS but were evaluated during promulgation of rulemaking for the radiological criteria for license termination (10CFR20.1402). The basis for public health and safety considerations associated with the license termination rule is discussed in NUREG-1496.

8.2.9 Radiological Accidents

The environmental impacts associated with radiological accidents have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of radiological accidents is documented in Section 4.3.9 of Supplement 1 to NUREG-0586. CYAPCO has not identified any new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

The NRC concluded that radiological impacts, due to accidents, are considered to be undetectable and non-destabilizing, in the National Environmental Policy Act (NEPA) sense, if the doses remain within regulatory limits. The HNP UFSAR provides a summary of the evaluation of plant transients that have a potential impact on both occupational and public safety and health. The risk of accidents resulting in a significant radiological release during decommissioning activities is considerably less than during plant operations.

The analysis of decommissioning events includes all phases of decommissioning activities: decontamination, dismantlement, packaging, storage, radioactive materials handling, and license termination activities (including final status surveys). The following radiological events were identified in the UFSAR (July 2005 revision) as having the potential to affect public health and safety:

- Radioactive Waste System Failure

CYAPCO requested and received an exemption from the emergency preparedness requirements of 10CFR50.47 (Reference 8-16); however, approval of the exemption request was predicated on the absence of any accidents where the offsite dose consequences could exceed the EPA protective action guidelines (PAGs). Releases resulting from accidents postulated in the decommissioning accident analysis were evaluated using the EPA PAGs as an upper limit and found to be bounded by this criterion. Use of the EPA PAGs as an administrative limit also ensures that postulated accident offsite doses are significantly less than the 10CFR100 reference values.

CYAPCO received another exemption from the emergency preparedness requirements of 10CFR50.47 on March 18 and August 15, 2013 (References 8-23 and 8-24). The NRC's Safety Evaluation Report stated: "In accordance with 10 CFR 50.82, the 10 CFR Part 50 licensed area for the HNP has been reduced to a small area surrounding the ISFSI. In this condition, the HNP poses a significantly reduced risk to public health and safety from design basis accidents or credible beyond design basis accidents since these cannot result in radioactive releases which exceed EPA PAGS at the site boundary. Because of this reduced risk, compliance with all the requirements in 10 CFR 50.47 and 10 CFR Part 50, Appendix E is not appropriate. The requested exemption from portions of 10 CFR 50.47 and 10 CFR Part.50, Appendix E is needed to continue implementation of the HNP ISFSI Emergency Plan that is appropriate for a stand-alone ISFSI and is commensurate with the reduced risk posed by the facility. The requested exemption will allow spent fuel to continue to be stored safely without imposing burdensome and costly new requirements that provide no increased safety benefit."

Thus, because the dose consequences resulting from radiological events, identified as having the potential to affect public health and safety, are below the EPA PAGs and the criteria of 10CFR100, the associated impacts on the environment are minimal. The NRC defined in their Safety Evaluation Report dated November 26, 2007 (Reference 8-22) that "the criteria of 10 CFR Part 100 no longer apply to this site and need not be addressed."

8.2.10 Occupational Issues

The environmental impacts of occupational issues have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of occupational issues is documented in Section 4.3.10 of Supplement 1 to NUREG-0586. CYAPCO has not identified any new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

As Supplement 1 to the NUREG-0586 indicates, the Occupational Safety and Health Act of 1970 was enacted to protect the health of workers, and applicable regulations are administered by the Occupational Safety and Heath Administration (OSHA). The HNP and HNP ISFSI are subject to 29 CFR 1910 and 1926 for worker health and safety

protection under OSHA regulations. These requirements are implemented under existing plant programs and procedures.

8.2.11 Socioeconomic Impacts

The environmental impacts of socioeconomic impacts have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of socioeconomic impacts is documented in Section 4.3.12 of Supplement 1 to NUREG-0586.

The impacts that are observed by the community are primarily those resulting from plant closure rather than from decommissioning, although some decommissioning activities began very shortly after closure. These impacts occur through changes in employment levels and local demands for housing and infrastructure or through decline of the local tax base and the ability of local government entities to provide public services. Supplement 1 to NUREG-0586 states that decommissioning, itself, has no impact on the tax base and no detectable impact on the demand for public services.
Additionally Supplement 1 to NUREG-0586 concludes that the effects of employment changes on population growth are:

1. not detectable if population changes (reductions or increases) are less than 3% per year,
2. detectable but not destabilizing if the population change is between 3% and 5%, and
3. de-stabilizing if the population change is greater than 5% per year.

Table 8-2 shows the change in population over the last two decades. For the decade 1990 to 2000, which includes the period of shutdown and partial decommissioning, the population in the vicinity of the site increased less than an average of 3% per year during this ten-year period (Reference 8-17). It is notable that the population continues to increase, and in most cases does so consistent with the previous decade. As can be seen, the average annual population change, based upon the data from 1990 and 2000, does not exceed the NRC's threshold of 3%, and thus signifies that the changes are neither detectable nor destabilizing. Thus no significant socioeconomic impacts are associated with HNP and HNP ISFSI decommissioning and license termination activities related to the end use of the site.

8.2.12 Environmental Justice

Environmental Justice was addressed in the HNP Decommissioning Plan (Reference 8-1). An evaluation of the demographic data regarding low income and minority populations along the LLW transportation routes was performed to address Environmental Justice concerns. The results of the analysis support the conclusions that

minority populations and low income populations are not disproportionately impacted by LLW waste shipments from the HNP site.

These conclusions remain valid. The types of decommissioning and license termination activities, conducted or planned at HNP and the HNP ISFSI, are not significantly different than those described in the Decommissioning Plan and the assumptions related to affected populations remain valid, considering the information from the 2000 Census, presented above. Thus, there are no environmental justice impacts introduced by decommissioning or license termination.

8.2.13 Cultural and Historic Resources Impacts

8.2.13.1 Activities Within the Operational Area

The environmental impacts associated with cultural and historic resource impacts from activities within the operational area have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of cultural and historic resource impacts from activities within the operational area is documented in Section 4.3.14 of Supplement 1 to NUREG-0586. CYAPCO has not identified any new information or significant environmental changes associated with the site-specific termination activities related to the end use of the site.

8.2.13.2 Activities Outside the Operational Area

At CYAPCO's request, a review was performed by American Cultural Specialists, LLC, to identify and evaluate the archaeological resources that may exist on the plant site, particularly in the area of the ISFSI (Reference 8-18). The conclusion reached was that the site activities overall, and specifically those planned in the area of the ISFSI would present no impact on the area's cultural resources.

Archaeological research continues onsite outside the operational area. At the direction of the State Historic Preservation Office, CYAPCO will characterize and investigate, as appropriate, the archaeologically sensitive areas on site. Results of any investigations will be reported to the State Historic Preservation Office.

The State [of Connecticut] Historic Preservation Office reviewed the documents prepared by American Cultural Specialists, as well as reviewing the Venture Smith Archaeological Site, Haddam, Connecticut, National Register of Historic Places Nomination: Report on Background Research, dated May 15, 2002, prepared by the Public Archaeology Survey Team, Inc. In addition, a Staff Archaeologist from the State Historical Preservation Office conducted onsite inspections of the area and associated areas, including the haul road; monitoring station, bulky waste burial pit, and the burial pit access road.

The onsite inspection identified a diffuse scatter of prehistoric and historic archaeological artifacts. However, the State Historical Preservation Office believe that the artifacts lack stratigraphic context and scientific integrity and represent isolated finds that do not provide substantial information on the prehistory or history of the area. Thus, the Office agrees with the conclusions of the American Cultural Specialists that no further archaeological investigations are required in the ISFSI area and that the proposed activities will have no effect upon the state's archaeological heritage. (Reference 8-19)

8.2.14 Aesthetics

The environmental impacts associated with aesthetics have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of aesthetics is documented in Section 4.3.15 of Supplement 1 to NUREG-0586.

Aesthetic resources include natural and man-made landscapes and the way the two are integrated. As a part of construction and operation of the facility, the landscape was previously altered. Decommissioning activities will be conducted onsite, both inside and outside of existing buildings (in the case of dismantlement or shipping activities). The NRC has concluded that any visual intrusion resulting from decommissioning will be temporary and would serve to reduce the aesthetic impacts of the facility. CYAPCO will use best management practices to control many of the potentially adverse impacts of decommissioning on aesthetics (such as dust and noise), as discussed in other sections.

CYAPCO has not identified any new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

8.2.15 Noise

The environmental impacts associated with noise have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of noise is documented in Section 4.3.16 of Supplement 1 to NUREG-0586.

The plant is quite remote from well-traveled roads. To the northeast, just beyond the yard perimeter fence, the terrain rises steeply forming hills that parallel the Connecticut River. The heavily wooded terrain rises abruptly to hill crests of approximately 200 to 300 feet above mean sea level.

During operations, only at locations directly across the Connecticut River and in the immediate plant area could noise level, attributable to plant operations, be detected. With decommissioning, these noises have been eliminated. Decommissioning activities will, in general, be intermittent and temporary, and limited to a relatively small portion of the entire HNP site. Noise is attenuated by the mature forests surrounding the plant. During fall and winter, absence of foliage will allow some additional transmission of noise to the

areas north and west of the plant. The presence of the Connecticut River will allow some transmission of noise over the water before attenuation by forest. Because decommissioning activities are expected to add minimally to ambient noise beyond the perimeter security fence, noise will have a negligible effect on the environment.

8.2.16 Transportation

The environmental issue of transportation has been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of transportation is documented in Section 4.3.17 of Supplement 1 to NUREG-0586.

The number of shipments and the volume of waste shipped are greater during decommissioning than during operations. In Supplement 1 to the NUREG-0586, the public health and safety impacts of transportation of radioactive wastes are evaluated on the basis of compliance with regulations. The NRC has concluded that compliance with regulations is adequate to protect the public against unreasonable risk from the transportation of radioactive materials. The supplement to the FGEIS notes that the evaluation leading to that conclusion was based, in part, on information in NUREG-0170 and that recent re-evaluation of transportation risks, using updated information and assessment tools, found that risks are lower than those estimated in NUREG-0170. Because HNP and the HNP ISFSI will comply with all applicable regulations when shipping radioactive wastes from decommissioning, the effects of transportation of that radioactive waste on public health and safety are considered to be neither detectable nor destabilizing.

Non-radiological impacts of transportation include increased traffic and wear and tear on roadways. Because the average number of daily shipments from the site will be relatively small, there will be no significant effect on traffic flow or road wear. Additionally, because of the industry's emphasis on training and adherence to established procedures, truck accident rates for activities at nuclear facilities has been lower than the national average for similar activities. The NRC has concluded that impacts of transportation accidents would neither be detectable nor destabilizing.

Thus, transportation of wastes associated with the HNP and HNP ISFSI decommissioning and license termination activities do not present significant adverse impacts.

8.2.17 Irretrievable Resources

The environmental issue of irretrievable resources has been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of irretrievable resources is documented in Section 4.3.18 of Supplement 1 to NUREG-0586.

Supplement 1 to the NUREG-0586 indicates that land associated with a site released for unrestricted use is available for other uses, regardless of whether or not the decommissioning process returned the land to an open space or to an industrial complex. Thus the land resource would not be considered "irretrievable." The Supplement to the NUREG-0586 evaluated other irretrievable resources such as the materials/equipment used to decontaminate the facilities and the fuel used for construction machinery and for transporting wastes and concluded these resources are minor.

CYAPCO plans to release the land for unrestricted use. Thus, the impact of decommissioning and license termination on irretrievable resources is neither detectable nor destabilizing.

8.3 References

8-1 Connecticut Yankee Atomic Power Company, Decommissioning Environmental Report., dated August 1997.

8-2 Letter CY-97-075 from CYAPCO to USNRC, "Haddam Neck Plant Post-Shutdown Decommissioning Activities Report," dated August 22, 1997, as revised.

8-3 NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated August 1988.

8-4 USNRC, Final Environmental Statement, Haddam Neck (Connecticut Yankee) Nuclear Power Plant, Docket No. 50-213, October 1973.

8-5 Letter, USNRC to CYAPCO, "Environmental Assessment for Proposed License Extension," dated November 23, 1987.

8-6 Supplement 1 to NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated November 2002.

8.7 Title 10 to the Code of Federal Regulations, Subpart E to Part 20.

8-8 Revision 2 to the PSDAR, dated April 28, 2004

8-9 Haddam Neck Plant UFSAR, revisions through February 2004.

8-10 State of Connecticut Department of Environmental Protection, NPDES Permit issued to Connecticut Yankee Atomic Power Company, expiration date September 29, 2005.

8-11 Radiological Effluent Monitoring Manual for the Haddam Neck Plant, Revision 17.

8-12 CTDEP to CYAPCO letter, "Natural Diversity Data Base," dated April 24, 2000.

8-13 "Assessment of and Management Recommendations for the Aquatic Resources of the Connecticut Yankee Property, Haddam Neck, Connecticut," final report, prepared by Ichthyological Associates, Inc., dated June 4, 2003.

8-14 Letter from Priscilla W. Baillie to Kurt Prochorena, Tighe & Bond, Inc., "Re: Connecticut Yankee Decommissioning Project, Reference Number 1260891," dated July 10, 2003.

8-15 Letter CY-04-113, from Gerard van Noordennen, CYAPCO, to Mike Grzywinski, DEP, "Haddam Neck Plant Structures, Dredging & Fill Permit Application, Intake Structure Demolition, Additional Information," dated May 25, 2004,

8-16 Fredrichs, USNRC, to Mellor, CYAPCO, "Exemption from a Portion of 10 CFR Section 50.54(q) and Approval of Defueled Emergency Plan at Haddam Neck Plant (RAC No. 99015) dated August 28, 1998.

8-17 US Census Bureau, "Population Housing Units, Area, and Density: 2000, Connecticut Place and County Subdivision."

8-18 Phase IB Archaeology Reconnaissance Survey of the Proposed ISFSI Location on the Connecticut Yankee Property in Haddam Neck, CT," prepared by American Cultural Specialists, LLC, dated May 16, 2002.

8-19 Letter from John W. Shannahan, Director and State Historic Preservation Officer, Connecticut Historical Commission to K.J. Heider, CYAPCO, dated May 20, 2002.

8-20 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of East Site Grounds from Part 50 License," dated September 1, 2004.

8-21 Letter from T. Smith (NRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of Phase II Areas from Part 50 License," dated February 27, 2006.

8-22 Letter from K. McConnell (USNRC) to W. Norton (CYAPCO), "Haddam Neck Plant – Release of Land from Part 50 License," dated November 26, 2007.

8-23 Letter from M. Lombard (USNRC) to B. Buerger (CYAPCO), "Response to Exemption Request for Portions of Title 10 of the Code of Federal Regulations Part 50 Appendix E, and Title 10 of the Code of Federal Regulations Part 50.47 for the Haddam Neck Plant (TAC No. L24663)," dated March 18, 2013.

8-24 Letter from J. Goshen (USNRC) to B. Buerger (CYAPCO), "Revised Response to Exemption Request for Portions of Title 10 of the Code of Federal Regulations Part 50 Appendix E, and Title 10 of the Code of Federal Regulations Part 50.47 for the Haddam Neck Plant (TAC No. L24663)," dated August 15, 2013.

Table 8-1

Summary of Environmental Impacts from Decommissioning

Issue	Generic	Impact	LTP Section
Onsite-Offsite Land Uses			8.2.1
• Onsite Land Uses	Yes	Small	8.2.1.1
• Offsite Land Uses	No	Site-Specific	8.2.1.2
Water Use	Yes	Small	8.2.2
Water Quality	Yes	Small	8.2.3
Air Quality	Yes	Small	8.2.4
Aquatic Ecology			8.2.5
• Activities within the operational area[1]	Yes	Small	8.2.5.1
• Activities outside the operational area	No	Site-Specific	8.2.5.2
Terrestrial Ecology			8.2.6
• Within the operational area	Yes	Small	8.2.6.1
• Outside the operational area	No	Site-Specific	8.2.6.2
Threatened and Endangered Species	No	Site-Specific	8.2.7
Radiological			8.2.8
• Activities resulting in occupational doses to workers	Yes	Small	8.2.8.1
• Activities resulting in doses to the public	Yes	Small	8.2.8.2
Radiological accidents	Yes	Small	8.2.9
Occupational issues	Yes	Small	8.2.10
Cost	N/A	N/A[2]	7
Socioeconomic	Yes	Small	8.2.11
Environmental Justice	No	Site-Specific	8.2.12
Cultural and Historic Resource Impacts			8.2.13
• Activities within the operational area	Yes	Small	8.2.13.1
• Activities outside the operational area	No	Site-Specific	8.2.13.2
Aesthetics	Yes	Small	8.2.14
Noise	Yes	Small	8.2.15
Transportation	Yes	Small	8.2.16
Irretrievable Resources	Yes	Small	8.2.17

[1] The operational area is defined as the portion of the plant site where most or all of the site activities occur, such as reactor operation, materials and equipment storage, parking, substation operation, facility service, and maintenance. This includes areas within the protected area fences, the intake, discharge, cooling, and associated structures as well as surrounding paved, graveled, maintained landscape, or other maintained areas.

[2] A decommissioning cost assessment is not a specific National Environmental Policy Act (NEPA) requirement.

Table 8-2

Population Changes in the Vicinity of HNP

Municipality	1990 (Ref 8-1)	% change from prior decade	2000 (Ref 8-17)	% change in decade including shutdown
Chester	3417	11.4	3742	9.5
Colchester	10980	41.5	14551	32.5
Deep River	4332	8.5	4610	6.4
Durham	5732	11.5	6627	15.6
East Haddam	6676	18.8	8333	24.8
East Hampton	10428	21.7	13352	28.0
Essex	5904	16.3	6505	9.2
Haddam	6769	6.0	7157	5.7
Hebron	7079	29.8	8610	21.6
Killingworth	4814	21.1	6018	25.0
Lyme	1949	7.0	2016	3.4
Madison	15485	10.4	17858	15.3
Marlborough	5535	16.6	5709	3.1
Middlefield	3925	3.4	4203	7.1
Middletown	42762	9.5	43167	1.0
Portland	8418	0.4	8732	3.8
Salem	3310	41.8	3858	8.3
Westbrook	5414	3.0	6292	16.2

APPENDIX C

Deleted

Refer to Section 7 for a discussion of the decommissioning cost estimate for the HNP ISFSI