November 25, 2002

Mr. Kenneth J. Heider
Vice President - Operations and Decommissioning
Connecticut Yankee Atomic Power Company
362 Injun Hollow Road
East Hampton, Connecticut 06424-3099

SUBJECT: HADDAM NECK PLANT - ISSUANCE OF AMENDMENT RE: APPROVAL OF LICENSE TERMINATION PLAN (LTP) (TAC NO. MA9791)

Dear Mr. Heider:

The Commission has issued the enclosed Amendment No. 197 to Facility Operating License No. DPR-61 for the Haddam Neck Plant. The amendment consists of changes to the Technical Specifications in response to your application dated July 7, 2000, as supplemented by letters dated June 14, July 31, August 15, August 22, September 6, September 7, 2001, August 20, and October 10, 2002. Calculations to support the LTP were also provided in letters dated May 9, June 26, and August 15, 2002.

The amendment adds a license condition which approves the LTP for the Haddam Neck Plant, and provides the criteria by which you may make changes to the LTP without prior U.S. Nuclear Regulatory Commission approval.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Jack Donohew, Senior Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-213

Enclosures: 1. Amendment No. 197 to DPR-61
2. Safety Evaluation

cc w/encls: See next page

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<u>CONNECTICUT YANKEE ATOMIC POWER COMPANY</u>

<u>DOCKET NO. 50-213</u>

<u>HADDAM NECK PLANT</u>

<u>AMENDMENT TO FACILITY OPERATING LICENSE</u>

Amendment No. 197
License No. DPR-61

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by the Connecticut Yankee Atomic Power Company (CYAPCO or the licensee) dated July 7, 2000, as supplemented by letters dated June 14, July 31, August 15, August 22, September 6, September 7, 2001, and May 9, June 26, and August 15, August 20, and October 10, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the rules and regulations of the Commission as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the regulations of the Commission and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by the addition of paragraph 2.C.(7) to Facility Operating License No. DPR-61. Paragraph 2.C.(7) has been added as indicated in the attachment to this license amendment.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Operating License

Date of Issuance: November 25, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 197

FACILITY OPERATING LICENSE NO. DPR-61

DOCKET NO. 50-213

Revise Facility Operating License No. DPR-61 by removing the page identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
Operating License	Operating License
4	4
---	5

(5) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Haddam Neck Plant Physical Security Plan," with revisions submitted through January 24, 1989; "Haddam Neck Plant Guard Training and Qualification Plan," with revisions submitted through January 27, 1983; and "Haddam Neck Plant Safeguards Contingency Plan," with revisions submitted through December 9, 1983. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

(6) Fuel Movement

The movement of special nuclear material used as reactor fuel into the containment is prohibited.

(7) License Termination Plan (LTP)

The License Termination Plan dated July 7, 2000, as revised in August 2002 (Revision 1) is approved by NRC License Amendment No. 197.

In addition to those criteria specified in 10CFR50.59, 10CFR50.82(a)(6), and 10CFR50.82(a)(7), changes to the approved License Termination Plan shall require NRC approval prior to being implemented, if the change:

(a) Increases the radionuclide-specific derived concentration guideline 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 I decision error above the level stated in the LTP (discussed in Section 5.5.1.1 of the LTP);

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

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

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

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

Prior to a request to release a survey area from the license, the licensee shall have performed a Capture Zone Analysis and have assured that the ground water dose contribution is included for all applicable survey areas per the process described in Section 5.4.7.1 of Revision 1 of the LTP. |

D. This license is effective as of the date of issuance and authorizes ownership and possession of this facility until the Commission notifies the licensee in writing that the license is terminated. The licensee shall:

1. Take actions necessary to decommission and decontaminate this facility and continue to maintain this facility, including, where applicable, the storage, control and maintenance of the spent fuel, in a safe condition; and

2. Conduct activities in accordance with all other restrictions applicable to this facility in accordance with NRC regulations and the specific provisions of this 10 CFR 50 facility license.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by A. Giambusso

A. Giambusso, Deputy Director for
Reactor Projects
Directorate of Licensing

Enclosure:
Appendices A and B - Technical Specifications
Date of Issuance: December 27, 1974

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 197

TO FACILITY OPERATING LICENSE NO. DPR-61

CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT

DOCKET NO. 50-213

1.0 INTRODUCTION

By letter dated July 7, 2000, as supplemented by the letters dated June 14, July 31, August 15, August 22, September 6, and September 7, 2001, and May 9, June 26, August 15, August 20, and October 10, 2002; the Connecticut Yankee Atomic Power Company (CYAPCO, or the licensee) requested a change to the operating license for the Haddam Neck Plant. The proposed change would add a license condition which would approve the License Termination Plan (LTP) for the Haddam Neck Plant, and provide the criteria by which the licensee may change the LTP without prior U. S. Nuclear Regulatory Commission (NRC) approval.

The licensee's supplemental letters of June 14, July 31, August 15, August 22, September 6 and 7, 2001, and May 9, June 26, August 15 and 20, and October 10, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 13, 2000 (65 FR 77915). The supplemental letters of June 14, July 31, August 15 and 22, and September 6 and 7, 2001, were in response to the NRC staff's requests for additional information dated February 1 and March 19, 2001. These responses, and subsequent discussions with the NRC staff, were incorporated into Revision 1 of the LTP, which was provided by the licensee in its supplemental letter of August 22, 2002.

On August 23, 2000, the NRC staff published a notice of receipt and availability for public comment of the LTP in the *Federal Register* (65 FR 51345) and held a public meeting on the LTP on October 17, 2000, in Higganum, Connecticut, on the LTP. The NRC staff decided to close the formal public comment period on December 29, 2000 (65 FR 77394). The licensee discussed its application with the NRC staff in the meeting held on November 15, 2001, and provided the additional information in the supplemental letter of February 20, 2002. The meeting summary was issued by the NRC staff on November 27, 2001.

On December 5, 1996, the licensee notified the NRC staff of the permanent cessation of power operations of the Haddam Neck Plant and the permanent removal of all fuel assemblies from the reactor vessel to the spent fuel pool. Following the cessation of power operations, the licensee prepared to decommission the plant. The Post Shutdown Decommissioning Activities report was submitted in accordance with Section 50.82(a)(4) of Title 10 of the *Code of Federal*

Regulations (10 CFR) on August 22, 1997. On January 26, 1998, the licensee transmitted an Updated Final Safety Analysis Report to reflect the plant's shutdown status, and on June 30, 1998, the NRC staff amended the Facility Operating License to reflect the permanent shutdown status of the plant.

2.0 REGULATORY REQUIREMENTS

In accordance with the requirements of 10 CFR 50.82(a)(9), the licensee submitted a LTP for its facility. Under the provisions of 10 CFR 50.82(a)(10), the NRC approves an LTP by license amendment. Thus, the licensee has requested the addition of a new License Condition 2.C.(7) to the Haddam Neck Operating License. The new license condition would incorporate the approved "License Termination Plan" into the Haddam Neck Plant license and allow the licensee to make certain changes to this approved LTP without prior NRC approval. The new License Condition would appear as follows:

7. License Termination Plan (LTP)

The License termination Plan dated July 7, 2000, as revised in August 2002 (Revision 1) is approved by NRC License Amendment No.

In addition to those criteria specified in 10CFR50.59, 10CFR50.82(a)(6), and 10CFR50.82(a)(7), changes to the approved License Termination Plan shall require NRC approval prior to being implemented, if the change:

(a) Increases the radionuclide-specific derived concentration guideline 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 I decision error above the level stated in the LTP (discussed in Section 5.5.1.1 of the LTP);

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

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

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

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

> Prior to a request to release a survey area from the license, the licensee shall have performed a Capture Zone Analysis and have assured that the ground water dose contribution is included for all applicable survey areas per the process described in Section 5.4.7.1 of Revision 1 of the LTP.

3.0 TECHNICAL EVALUATION

In accordance with 10 CFR 50.82(a)(9), the licensee submitted its LTP. This section of the regulations requires the LTP to contain the following information: (1) a site characterization; (2) identification of remaining dismantlement activities; (3) plans for site remediation; (4) detailed plans for the final radiation survey; (5) a description of the end use of the site, if restricted; (6) an updated site-specific estimate of remaining decommissioning costs; and (7) a supplement to the environmental report, pursuant to 10 CFR 51.53, describing any new information or significant environmental change associated with the licensee's proposed termination activities. In addition, the licensee requested the authority to: (1) remove certain portions of the site from the license once NRC has approved the LTP; and (2) make certain changes, if necessary, to the LTP once NRC has approved this document. The following is the NRC staff's evaluation of this information.

3.1 Site Characterization

The site characterization survey is the radiation survey conducted to determine the nature and extent of radiological contamination at a site. The purpose of the site characterization survey is: to permit planning for remediation activities; to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected at the site; to provide information to design the final site survey (i.e., identify survey unit classifications for impacted areas); and to provide input into dose modeling. Surveys and sampling conducted during site characterization are based on the area's Historical Site Assessment (HSA), scoping survey, and judgmental measurements. According to NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM) (Section 2.4.4), if an area could be classified as a Class 1 or Class 2 for the final status survey (FSS), based on the HSA and scoping survey results, a characterization survey is warranted.

The licensee plans to use a series of surveys to demonstrate compliance with the criteria specified in 10 CFR Part 20, Subpart E, for unrestricted release of the Haddam Neck site, consistent with the Radiation Survey and Site Investigation (RSSI) process and the Data Quality Objective (DQO) process as recommended by MARSSIM. The DQO process is a systematic procedure developed by the Environmental Protection Agency to determine the type, quantity, and quality of data to make Agency decisions. The procedure is designed to define the criteria for data collection design including when and where to collect samples, the tolerable level of error, and how many samples to collect. Characterization is one of the steps in the RSSI process that relies on the Data Life Cycle. The Data Life Cycle is a process of planning a survey using the DQO process, implementing it, and assessing the results before making a decision. It is the basis for the performance-based guidance in MARSSIM and it is used in an iterative fashion within the RSSI process. Licensees may use data developed from site characterization as final site survey data, providing these data meet the DQOs appropriate for FSS. The NRC staff will review changes made to DQOs as part of its ongoing inspection process.

The licensee states, in Section 2.3.1 of the LTP, that radiological characterization of the Haddam Neck site has been ongoing since the plant began operation in 1968. To date, the licensee has not yet fully characterized the Haddam Neck site. The licensee has conducted an HSA and a limited site characterization for the Haddam Neck site.

The licensee conducted an HSA that consisted of a review and compilation of site historical records [e.g., 10 CFR 50.75(g) records, radiological incident files, operational survey records, and annual environmental reports to NRC]. Personnel interviews were conducted with present and former plant employees and selected contractors to determine operational events that caused contamination in areas or systems not designed to contain radioactive or hazardous materials.

Between 1997 and 1999 the licensee performed a limited site characterization survey. The licensee refers to this effort as an initial site characterization of the Haddam Neck site performed to the guidelines of MARSSIM. The purpose of this effort was to estimate the extent of on-site contamination in controlled radiological areas and on adjoining licensee controlled property that would require remediation to support decommissioning. The licensee used the results to develop the initial decommissioning plans, schedules, and cost estimates.

Initial characterization information was provided in November 2000 to the NRC staff in the Haddam Neck Characterization Report, dated January 6, 2000. This report consisted of general information, called “characterization reports” by the licensee, for each of the site buildings and subsections of the site grounds. Each characterization report contains a boundary description of the area, a general description of the radiological and hazardous material within the area, impacted systems within an area, and the licensee’s recommendations for further samples or surveys. However, many of these characterization reports do not include references to site characterization data per area/room, building, or survey area.

The licensee provided more detailed information on the radiological status of structures and land, as a result of a request for additional information (RAI) from the NRC staff. It produced a document titled “HSA Supplement” dated August 14, 2001, to supplement the original “Haddam Neck Plant Characterization Report." The licensee used this additional information to establish the initial MARSSIM classifications of survey areas and survey units (Table 2-10 of the LTP). The HSA Supplement and Tables 2-11A, 2-11B, and 2-11C of the LTP summarize a comprehensive review of numerous plant records such as: plant incident reports, condition reports, investigations, radiological (routine and decommissioning support) surveys (1967-2000), annual effluent reports, interviews with past and present employees, and site walk-downs. As noted in Section 2.3.3.2 of the LTP, the licensee indicated that data presented in Tables 2-11A, 2-11B, and 2-11C of the LTP were the results of surveys conducted within the last 5 years. The licensee indicated that the calibration and quality control processes for these surveys were consistent and similar to those used today. However, the licensee did not produce DQOs for these data.

The licensee plans to develop more extensive characterization efforts and expand the information collected from its initial characterization efforts. Concurrent with site characterization, the licensee is conducting decommissioning activities. In Section 2.3.3.2 and Section 5.9 of the LTP, the licensee committed to document its radiological survey plans, using the DQO process and associated data evaluation reports produced from subsequent RSSI surveys (e.g., operational, characterization, and remedial action support surveys). This information and supporting data will be available on site for NRC staff review during inspections. Additionally, in Section 5.9 of the LTP, the licensee commits to provide, in the FSS documentation for each survey unit, a description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity. The NRC staff plans to review this information as part of its ongoing inspection effort. The NRC staff will review the licensee's characterization survey plans and associated reports to: (1) assess whether the licensee has sufficient information to characterize the nature and full extent of radiological contamination at the site; (2) assess whether the licensee demonstrated, on a statistical basis, that the characterization data are sufficiently representative of the waste and contaminated environmental media; (3) determine whether the licensee has demonstrated that the characterization results are sufficient to support evaluation of reasonable decommissioning approaches; and (4) assess whether the licensee adequately used characterization data in the FSS design. If the NRC staff finds that the licensee's characterization is not adequate, it will document its findings in an inspection report and pursue follow-on corrective action. In conclusion, the licensee's continuing efforts to finalize the site characterization will be a focus of future NRC inspections.

In Section 2.3.3.2 of the LTP, the licensee described the process by which decommissioning will proceed and areas will, as necessary, be decontaminated to levels that will meet the FSS acceptance criteria. The licensee plans to perform a "turnover assessment" when it believes that an area is ready for the FSS. If the results of the turnover assessment indicate that the FSS acceptance criteria will be met, physical and administrative control of the area will be transferred to the FSS personnel for preparation, design, and performance of the FSS. Otherwise, the licensee may decide that additional remediation is necessary. Further, the licensee indicated that the turnover assessment may include a "turnover survey," primarily for Classes 1 and 2 survey units within the Industrial Area (IA), and in land areas outside the IA that are impacted by existing ground water contamination. The turnover survey process, together with any additional characterization and remediation surveys performed, represent at least one, but possibly several, opportunities to collect and evaluate additional survey data before conducting the FSS for a survey unit. The NRC staff expects the documented turnover assessments, as well as the results of turnover surveys, when performed, to be available for NRC review during inspections.

The types of surveys and sampling that the licensee plans to conduct include: (1) surface activity measurements on interior of buildings (surface structures); (2) more extensive sampling and surveys to determine potential migration of radionuclide contamination in hard-to-reach or not readily accessible areas (e.g., cracks, crevices, areas beneath buildings, construction joints, etc.); (3) surface activity measurements on interior surfaces of embedded piping; (4) detailed surveys and sampling to supplement information developed during the initial site characterization (i.e., internal surfaces of secondary side systems, activated concrete near the reactor vessel, direct radiation measurements, etc.); and (5) surveys and sampling of systems, structures, and the environment that were not conducted during the initial characterization [e.g., open land areas, subsurface soil, sediment, ground water, and structural surfaces potentially contaminated with transuranic (TRU) and difficult-to-measure /hard-to-detect (HTD) radionuclides].

A derived concentration guideline (DCGL) is the concentration of a radionuclide which, if distributed uniformly across a survey unit, would result in an estimated dose equal to the applicable dose limit. The criteria for unrestricted release are specified in 10 CFR Part 20, Subpart E. The licensee has developed "base case" DCGLs for each radionuclide, provided in Table 6.1 (soil); Table 6.2 (groundwater); Table 6.3 (building surfaces); and Table 6.4 (building demolished or concrete debris). When multiple radionuclides are present, the unity rule applies thereby ensuring that the dose contribution from all radionuclides does not exceed 25 mrem/yr. At the Haddam Neck site, there are multiple radionuclides. Also, there are a number of areas of the site where exposure from multiple radionuclides could occur. In order to meet 10 CFR Part 20, Subpart E, criteria for unrestricted release, the exposures for all sources (multiple radionuclides in these credible multiple exposure areas and areas of existing groundwater contamination) must not exceed 0.25 mSv/yr (25 mrem/yr) and must be as low as is reasonably achievable (ALARA). In Section 5.4.7 of the LTP, the licensee discusses how it will account for the total dose in such areas.

The licensee will calculate additional, "operational derived concentration guideline levels" $DCGLs_{op}$ or "reduced" DCGLs (discussed in Sections 5.4.7 and 6.9 of the LTP) when contamination from multiple pathways may be present. A $DCGL_{op}$ is a term established by the licensee specific to Haddam Neck. Use of $DCGLs_{op}$ is a method of accounting for the potential contribution to dose from other possible exposure pathways, such as ground water, for a specific survey unit. The licensee will derive the $DCGLs_{op}$ of a survey unit before performing the FSS of soils or buildings in survey units where existing ground-water contamination may impact the potential dose. In no case will a $DCGL_{op}$ be greater than its corresponding base case DCGL. This determination will be provided in a technical support document and will be applied to the affected survey units.

In Section 5.4.7 of the LTP, the licensee stated that difficult-to-measure radionuclides are present. This will necessitate the use of surrogate radionuclides and will be based on a representative radionuclide mix established for each survey unit. In cases where measurable activity still exists, the licensee expects 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 and will be based on a representative radionuclide mix established for each survey unit. Scaling factors will be selected from available composite waste stream analyses or similar assays. Such analyses will be performed periodically and documented in support of waste characterization needs. However, for those cases when survey units do not have measurable activity distinguishable from background at the time of the FSS, the licensee will select a representative radionuclide mix, based on the historical characterization information for the survey unit of interest or for units with similar history and physical characteristics. In such cases, it is important that the NRC review establish that the selected mix is representative. The NRC staff plans to review the licensee’s rationale for its selection of representative radionuclide mixes. This review will be performed as part of the inspection process.

The licensee’s site characterization process focuses on structures, systems, and the site environs, considering radiological, hazardous, and State regulated-materials. Groundwater and subsurface soil contamination are included in the assessment of the site environs. Quality Assurance (QA) and Quality Control (QC) measures include the appropriate training and qualifications, instrumentation, procedures, records, audits and surveillance, and data collection for the site characterization program, to ensure data quality in accordance with the Haddam Neck QA Program and for compliance with 10 CFR Part 50, Appendix B.

The NRC staff finds the site characterization process acceptable, based on the information described above. As more characterization information is developed, it will be available on site. The NRC staff will review the licensee's characterization plans and supporting reports, as part of NRC's ongoing inspection process, to ensure that the basis for the FSS design and implementation and supporting data are adequate for the licensee to ultimately demonstrate compliance with the requirements of 10 CFR Part 20, Subpart E.

The FSS will be conducted using guidance in MARSSIM to demonstrate compliance with the criteria specified in 10 CFR Part 20, Subpart E, for unrestricted release of the Haddam Neck site. The types of surveys and sampling described for complete characterization are acceptable but will require further validation by the NRC staff to ensure that the methodology and data are adequate as this information becomes available. This validation will occur as part of NRC's ongoing inspection process.

In accordance with 10 CFR 50.82(a)(9)(ii)(A), the licensee provided the radiological conditions of the site in Sections 2.2 and 2.3 of the LTP. Summaries of the most significant radiological events were described in Section 2.2.4.2 of the LTP. As part of the NRC staff's effort to verify the completeness of the historical data presented by the licensee, the results of an earlier NRC staff report, "NRC Historical Review Team-Radiological Control and Area Contamination Issues at Haddam Neck," dated March 26, 1998, were reviewed. The NRC staff noted that in the NRC Historical Review Team's report the NRC staff concluded that events in 1979, when the plant operated with failed fuel, resulted in approximately 40 discrete areas of the site where fission product activity could have resulted in skin contamination with doses near the quarterly occupational dose limits specified in 10 CFR Part 20. This caused the NRC staff to ask the licensee to address the potential dose impact of the 1979 events of concern on decommissioning. In response the licensee provided a technical evaluation entitled, "Evaluation of the 1980 Particulate Activity for Impact on the PSR (Partial Site Release) Survey," dated September 9, 2001. The licensee calculated the dose to an individual, in the remote possibility that one of these particles was undetected today and could be ingested after a proposed PSR. The individual dose that was calculated was less than 10 microSievert (µSv) [one milliroentgen-equivalent-man (mrem)], should such an unlikely event occur. When this potential dose is compared with the average annual dose equivalent due to natural background in the U.S. of 3,000 µSv (300 mrem) it is insignificant.

The licensee has determined which radionuclides could potentially be found in the environment as a result of Haddam Neck operations. The results of the licensee's assessment and base case DCGLs for these radionuclides are provided in Tables 6.1 (soil), 6.2 (ground water), 6.3 (building surfaces), and 6.4 (concrete debris), of the LTP. The NRC staff's evaluation of the licensee's assessment is presented in Section 3.5.3.1 of this evaluation.

Classification is a process by which a survey unit is described according to its radiological characteristics. Section 2 of the LTP discusses in detail the HSA for the Haddam Neck site and the initial classifications assigned to all the site structures and grounds. Characterization is an ongoing effort throughout the decommissioning process, and survey unit classifications may be modified based on new characterization information or impacts from decommissioning activities. The licensee is allowed to change the classification of a survey unit from a less restrictive classification to a more restrictive classification based on new information; however, if the licensee decides to change the classification of a survey unit from a more restrictive classification to a less restrictive classification, the licensee will seek NRC approval before implementing this type of classification change (see Section 1 of this evaluation).

The initial site classification for the Haddam Neck site began in 1997 and has expanded during subsequent site characterization activities following the MARSSIM guidance. The licensee will evaluate area classifications throughout the dismantlement and decommissioning process as radiological conditions change and additional information and data are obtained. In accordance with the LTP, the licensee will finalize the classification of each survey unit during the development of FSS packages for that survey unit. The final classification designation and verification will be included, along with characterization data, classification history of the unit, and discussion of survey design, in the FSS Report for each survey unit. The FSS report for each Haddam Neck survey unit will be submitted to NRC for review. The NRC staff will examine the rationale for assumptions, classification designations, and characterization through the inspection process. These reports will serve as the basis for NRC terminating Facility Operating License Number DPR-61.

3.1.1 Facility Radiological Status

As described in Section 1.2 of the LTP, the Haddam Neck plant permanently shut down after approximately 28 years of operation. Operations ceased and all fuel assemblies were permanently removed from the Reactor Pressure Vessel and placed in the Spent Fuel Pool. After cessation of operations, the licensee began to decommission the Haddam Neck site. Air and liquid effluents were reported to NRC on a regular basis. Operational events, corrective action system documents, and licensee event reports were reviewed by the licensee to determine the facility's radiological history. Oral interviews with plant personnel were conducted by the licensee to develop clear understanding of the radiological status of the site. The HSA Supplement and the initial characterization report were also developed to support the RSSI process. Further, the licensee maintains decommissioning records in accordance with 10 CFR 20.2103(a) and 10 CFR 50.75(g).

The NRC staff finds the facility radiological status acceptable based on the information described above.

3.1.1.1 Structures

The primary structures on site are designed to house systems containing radioactive material and to function as an environmental barrier against releases of this material. These structures include: the Containment Building, the Primary Auxiliary Building (PAB), the Service Building, the Waste Storage Building, Ion Exchange Structure, Spent Resin Facility, and structures containing tanks for storage of radioactive liquids. These buildings have been identified as impacted areas, with the majority of survey units designated as Class 1. There are no Class 3 survey units within the primary buildings.

Operations and maintenance activities performed in these buildings have resulted in surface contamination typical of nuclear power plants. The routine radiological surveillances that provided the basis for establishing controls for 10 CFR Part 20 worker safety now serve as the basis for the licensee's initial remediation program in each building at the site. As a result of initial remediation efforts, more radiological surveys will be conducted to assist in the design of building remediation efforts. This information will be used later in design of the FSS, as described above.

Additionally, a number of events were identified that affected the radiological status of these structures, specifically failed fuel (degraded fuel cladding allowing material to leak into the primary coolant) and primary-to-secondary leakage in the steam generators (primary coolant leaking into the secondary coolant system). In 1979 and 1989, the plant operated with failed fuel at a level that resulted in an increase in the level of alpha-emitting radionuclides in the primary coolant system and liquid systems that interface with the primary coolant system, as well as areas inside buildings that house these systems. There were several primary-to-secondary leakage events resulting from steam generator tube leakage. These events occurred during several operating cycles, with the first leakage event identified in 1973 and the final events occurring in 1990. The leakage from these events resulted in measurable radioactivity in small areas of the secondary system piping, primarily in the high-pressure steam components within the Turbine Building.

In Section 2.3.3.1.2 of the LTP, the licensee discusses the extent and nature of contamination in primary structures on site and references Section 7.0 of the HSA Supplement for a detailed discussion of the radiological impact on each building.

Results of surveys from the Turbine Building show no indications of alpha-emitting radionuclides from failed fuel, with the exception of the Auxiliary Boilers. Surveys of accessible areas of system internals have shown removable contamination ranging from 0.37 kilobecquerel (kBq/g) [0.01 picocuries, (pCi/g)], up to several pCi/g of Cs-137 and Co-60. As documented in the HSA Supplement, fixed radioactive material has been identified in levels up to approximately 10,000 disintegrations per minute (dpm)/100 square centimeters (cm^2). Isotopic analysis has identified Cs-137 as the principal radionuclide resulting from the primary to secondary leakage events. Additionally, in 1996, scoping surveys of the operating floor and grade level of the Turbine Building for both beta- and gamma-emitting radionuclides identified only one small area in excess of Inspection and Enforcement Circular 81-07 (IEC 81-07), "Control of Radioactively Contaminated Material," guidance. All survey units within the Turbine Building are initially identified as Class 2 impacted areas, because of historical steam generator tube leaks and the proximity to Class 1 survey units.

Based on the design basis of the PAB, events that have occurred within the PAB, and the present status of areas controlled as contaminated areas, much of the interior surface of the PAB is expected to contain radioactivity above the DCGLs. Results of core boring in the PAB indicate that contamination typically only penetrated to a depth of about 1.3 cm (0.5 inches); however, contamination up to a depth of 5.1 cm (2 inches) has been identified in an area that was expected to be among the most contaminated in the PAB. The HSA Supplement provides a summary of the core-bore results. Specifically, three core-bore samples were taken. The licensee identified Co-60, Cs-137, and Cs-134. No neutron activation analysis was performed on any components of the PAB, because of the distance and extensive shielding from the reactor.

Initial characterization of the PAB roof was performed in 1998, based on a history of contamination near the ventilation ducting and the use of epoxy paint to cover areas of fixed contamination in the tar and fixed-stone covering. Results of the PAB roof characterization (historical) showed that only one of five material samples reported licensed material (Co-60) above background. However, as the sample geometry of roof material samples was not correct, the licensee considered these results qualitative. The NRC staff agrees. However, the licensee will take additional samples as recommended by MARSSIM.

Based on the design basis of the Containment Building, events that have occurred within the Containment Building, and the present status of areas controlled as contaminated areas, much of the interior surface of the Containment Building is expected to contain radioactivity above the DCGLs. Beta/gamma contamination levels in the Containment Building range from less than 1000 dpm/100 cm^2 up to hundreds of thousands of dpm/100cm^2. Alpha contamination levels range from less than 50 dpm/100 cm^2 to several thousands of dpm/100 cm^2. Radiation levels in the Containment Building range from less than 1.3 coulombs per kilogram (C/kg) [5 milliroentgen per hour (mR/hr)] up to several thousand mR/hr. Some components, equipment, structural steel, and concrete have become radioactive because of neutron activation. The HSA Supplement summarizes the concrete core-bore results that were taken in the Containment Building. In Section 2.3.3.1.2.3 of the LTP, the licensee has committed to obtain additional cores to establish radioactivity levels of materials subject to neutron flux after the reactor vessel and other highly radioactivity components have been removed. The licensee will collect site-specific data to characterize the nature and extent of radioactive contamination for reactor vessel/components and concrete shield structures near the reactor vessel. The NRC staff will review these data as part of its ongoing inspection effort.

Paint samples were taken in 1998 and assessed for polychlorinated biphenyls, Resources Conservation Recovery Act (RCRA) metals, and radioactivity. A number of paint-chip and concrete-chip samples were collected from Containment and analyzed by gamma spectroscopy. The results of this characterization effort are summarized in Section 2.3.3.1.2.3 of the LTP. Based on these data, the licensee found that the average concentration of radioactivity in paint on the steel liner is about 44 Bq/g (1200 pCi/g) on the charging floor level, and about 11 Bq/g (300 pCi/g) on the grade level --the primary radionuclides being Cs-137, Co-60 and Cs-134. Additionally, the concentrations of radioactivity in paints on equipment vary greatly, in both total activity and radionuclide distribution. Thus, the licensee concluded that radioactivity in paint/concrete samples is greater than the radioactivity in the underlying concrete samples, with greater concentration on the floor than on the walls.

Upper walls and ceiling of the containment were subject to airborne radioactivity during the lifetime of the unit. The licensee noted that industry experience with the deposition of radioactive material on ceilings and vertical surfaces has shown that these surfaces have a lower contamination potential primarily because of the gravitational settling. Additionally, the licensee states that there is no history of events leading to the spraying of primary coolant directly onto portions of the upper containment walls or dome, and the design of the containment charging floor makes direct communication between primary coolant and the containment upper structure unlikely. Therefore, isolated pockets of contamination are not expected on the upper walls or ceiling of the containment. Concrete core bores and the containment inner steel liner samples were obtained from 10 locations during the alternate containment access installation project performance in 1998. The analyses of these samples provided qualitative results (limited radionuclide identification, not complete radionuclide identification and quantification). The licensee indicated in Section 2.3.3.1.2.3 of the LTP that the steel liner of the containment enclosure will be removed and sent offsite for disposal.

The Radwaste Reduction Facility is a structure used for staging and packaging various radioactive and RCRA mixed-waste streams. The Radwaste Reduction Facility contained radiologically contaminated items, both as radioactive waste and processing equipment, that were internally contaminated. The facility only contains such support systems such as electrical and ventilation. Historical surveys of the building indicated contamination levels range from non-detectable up to 2000 dpm/100 cm^2 - beta/gamma. The licensee concludes that the floors and drains of the facility represent the primary concerns for residual contamination.

All other structures within the IA have been identified as impacted areas, with the specific initial classification based on the function of the area, historical events, and radiological survey results. Structures outside the IA are identified as Class 3 areas, unless specific events or radiological data indicate that a higher classification is required. In accordance with the guidance provided in MARSSIM, if an area could be classified as a Class 1 or Class 2 for the FSS, based on the HSA and scoping survey results, a characterization survey is warranted. The licensee has committed to characterize such areas. The NRC staff will review the characterization of these areas as part of its ongoing inspection efforts.

According to the information provided in Section 5.7.3.1.5 of the LTP, characterization of embedded piping is planned to be conducted before FSS design is completed. The licensee defines embedded pipe as piping in concrete or piping (penetrations), that is impractical to remove. The NRC staff learned, during a site visit in June 2002, that the licensee expects the embedded piping will be accessible for survey purposes.

At Haddam Neck, there is buried pipe below grade but not embedded in concrete (e.g., storm drains, culverts). Examples of such piping include a yard drain system, storm drains, and septic system. The licensee has evaluated most of the buried pipe on site. Through a historical review, the licensee will identify all buried pipe and then perform a cost evaluation (survey and remediation versus removal with soil sampling). The licensee expects that most of the buried piping will be removed.

In Section 5.7.3.1.2 of the LTP, the licensee states that it will survey for activity beneath surfaces (cracks, crevices, paint, and paved surfaces); sewer systems; plumbing and floor drains; interiors of ventilation ducts; underground and embedded piping; activated concrete; and interiors and exteriors of both systems and equipment. The NRC staff has determined that the approach the licensee has proposed to characterize both structures and internal surfaces of embedded piping is acceptable. However, the NRC staff will, through the inspection process, review the associated RSSI data, to ensure that the above practice was implemented properly and data are adequate, until complete characterization information becomes available.

3.1.1.2 Systems

The licensee conducted a review of the Haddam Neck systems to determine which systems contain radioactive materials, and what radioactive material was detected, at some time, during the operating history of the plant. Systems that were identified as “affected” require additional surveys to define the extent and magnitude of radioactivity. Table 2-5 of the LTP provides a listing of plant systems and its status relative to the potential for being radioactively contaminated. The assessment considered the internal portions of the systems. The licensee indicated that, for those systems designed to contain radioactivity, the associated radiological conditions are continuously changing. The site Radiation Protection Department maintains the most recent information necessary to support radiation protection activities. These systems will be evaluated for remediation or disposal as radioactive waste, based on an economic evaluation of the alternatives. Several components have been identified as “affected,” based on primary-to-secondary leakage in operating cycles, as recent as 1990. These components contain low levels of radioactivity. The extent of contamination will be further defined as systems are disassembled and the internal surfaces become accessible.

Each plant system will be evaluated for the potential of both removable and fixed contamination, by direct surveys and/or analyses of swipes or metal scrapings. Radioactivity content on structural surfaces, from contamination, will be estimated by measuring dose rates from piping; using beta-gamma radiation survey instruments; counting of swipes and scrapings; and isotopic gamma spectroscopy of scrapings. Calculations will be used to estimate waste volumes for decommissioning planning and initial cost estimates.

Potential internal surface contamination of secondary side system piping, valves, or components will be assessed by: (1) direct surface activity measurements, (2) swipes to measure removable surface contamination, and (3) scrapings collected for gamma spectroscopy analyses, when there are direct surface activity measurements. Based on information collected from initial characterization, further assessment and characterization of secondary side systems are planned to determine whether systems are to be remediated, left in place, or removed for disposal.

The NRC staff has determined that the methodology used to characterize potential internal surface contamination of systems is consistent with the guidance in MARSSIM and is therefore acceptable.

3.1.1.3 Activation

The licensee has committed to characterize areas in the containment building that have the potential to have been exposed to neutron flux. The areas subject to the highest neutron activation are currently inaccessible. As the decommissioning progresses and high-dose-rate components are removed, additional characterization of activated concrete and other components will take place. In lieu of actual data, the licensee has reviewed neutron activation data from Maine Yankee, Trojan, and Yankee Nuclear Power Station. Data from these reactors indicate that the radionuclides that are present in the highest concentrations are H-3 and Fe-55. Other radionuclides such as C-14, Co-60, Eu-152, and Ni-63 are also present. Based on these data, the licensee included the activation products Eu-152 and Eu-154 in the list of radionuclides expected to be present in the Haddam Neck soils and structural materials. The licensee will typically use in-situ gamma spectroscopy to survey activated concrete, after any required remediation, to demonstrate that the concrete meets the applicable volumetric DCGLs. The licensee indicated, in Section 5.7.3.1.6 of the LTP, that these surveys will be conducted so that 100 percent of the affected volume will be covered in overlapping measurements. The licensee plans to treat embedded material, such as rebar, as concrete for the purposes of assigning DCGLs. Difficult-to-detect radionuclides that may be present in the concrete will be assessed by either direct measurements (core bores or equivalent) or by establishing surrogate DCGLs for these radionuclides relative to a radionuclide that is easily measured using gamma spectroscopy. Surrogate ratios will be established using characterization data for the survey unit of interest. Because characterization is still ongoing at the site, the licensee is unable to provide specific ratios, at this time, that will be used in developing the surrogate ratio DCGL values for the FSS. Therefore, as part of the inspection process, the NRC staff will review the licensee's assessment of any surrogate relationships and its specific ratios for inclusion in the FSS design.

The NRC staff has determined that the licensee's activation characterization strategy is consistent with the guidance in MARSSIM and is therefore acceptable.

3.1.1.4 Surface Soil

Through the HSA process, the licensee identified events involving unplanned liquid releases that have impacted the Radiologically Controlled Area (RCA). Historical surveys identified radioactive contamination in excess of the DCGLs. Consequently, the grounds within the RCA are classified as Class 1 survey units.

The licensee identified events where radioactively contaminated surface soil was found outside the RCA. The primary locations of discovered contamination associated with these events were the southwest peninsula; Survey Area 9520 (the Southwest Site Storage Area); and the shooting range. These areas are controlled today as Radioactive Material Areas. Land areas adjacent to the IA were surveyed as a result of events occurring in 1979 that resulted in particles of radioactive contamination being distributed in these areas, as discussed in Section 2.2.4.2.1 of the LTP. The licensee has subsequently conducted additional surveys in these areas. The licensee's strategy for surveying these areas was to survey land areas a distance from the plant twice that of the point where radioactive particles were detected. Most of the particles detected and removed were located close to the IA fence. There were only a few particles found beyond 100 meters (330 feet). The details of the results are found in an investigation report referenced in Section 2.4 (reference 2-4) of the LTP.

As described in Section 2.3.3.1.4 of the LTP, because of warehouse construction activities after the 1979 event, the primary parking lot in the area in question was re-configured and the parking lot asphalt and near-surface soil were removed. During discussions with the licensee, the licensee informed NRC that: (1) all of the parking lot asphalt of the previous parking lot was removed, except for a small area; (2) after installation of a new parking lot, the licensee surveyed and took core-bore samples of the parking lot; and (3) the licensee compared the survey results to its proposed DCGLs in the LTP, to support the initial classification.

Other instances of minor levels of contamination (typically below the proposed DCGLs) onsite were identified through the HSA. These areas were areas of high personnel traffic. On discovery, the licensee remediated the contamination in these areas and performed more extensive surveys.

Nominal radiological data for each survey unit that the licensee used to support its initial survey unit classifications for surface soil are also presented in Table 2-11B of the LTP. Also, the maximum and minimum radiation levels as well as Co-60 and Cs-137 maximum concentrations for each survey unit were provided.

In Section 2.3.3.2 of the LTP , the licensee noted that characterization for the Haddam Neck Site is an iterative process. The licensee has committed to use the DQO process, along with developing sampling and analysis plans during the RSSI process, to characterize these surface soil survey units. The NRC staff will review the RSSI data during the inspection process. The licensee plans to analyze samples, using either radiochemical techniques, gamma, or alpha, spectroscopy. These analyses will be performed by a laboratory operating under an audited QA program. The analytical laboratory's minimum detectable concentrations (MDCs) for both the alpha and gamma spectroscopy instruments to be used to quantify radionuclide concentrations will be consistent with MARSSIM guidance.

The licensee will survey surface soil areas in accordance with the guidance provided in MARSSIM, through combinations of sampling, scanning, and in situ measurements, as appropriate to characterize surface soil areas. The licensee has committed to obtain additional information, related to QA, of both field and laboratory activities, consistent with the guidance provided in Section 4.9 of MARSSIM.

The NRC staff finds the licensee's surface soil characterization strategy is consistent with the guidance in MARSSIM and is therefore acceptable.

3.1.1.5 Subsurface Soil

The licensee, through the HSA process, initially identified those areas where the potential exists for subsurface radioactive contamination. 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; onsite storage areas where radioactive materials have been identified; and areas containing spoils from past dredging of the discharge canal. However, the licensee states that it has not completed its assessment of all subsurface contamination. The licensee has also committed to further characterize soil in order to include soil in difficult-to-assess areas, such as under buildings. During 1998 and 1999, the licensee conducted subsurface soil samples, in some cases down to 6 feet in depth, in support of plant modification and site characterization activities. The licensee states, in Section 2.3.3.1.5 of the LTP, that none of these samples had plant-related radioactivity levels greater than the corresponding base case DCGLs. During the same time period, over 200 subsurface soil samples were collected, down to a depth of 2 meters in some cases, inside the RCA. Some isolated locations showed Co-60 and Cs-137 activity levels up to several hundred pCi/g each. Table 2-11C of the LTP provides nominal radiological data that the licensee used to support its initial classification of subsurface areas.

MARSSIM does not address subsurface soils. The licensee has divided the subsurface soil at Haddam Neck into three classifications, which it designated as: Classes A, B, and C. Class A soils have had known contaminating events and have 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.

Subsurface soils in the RCA, with the exception of soil beneath Survey Area 9308, are considered Class A. Subsurface soils beneath Survey Area 9308 and in the remainder of the IA outside the RCA are considered Class B. The licensee has initially classified areas northwest and southeast of Class B areas as Class C. Figure 2-21 in the LTP shows the location of the affected subsurface soil identified by the licensee.

The licensee states, in Section 5.7.3.1.2 of the LTP, that interior surfaces of below-grade foundations will be surveyed and decontaminated in the same manner as surfaces above grade. Exterior surfaces of below-grade foundations will be evaluated, using HSA and other pertinent records to determine the potential for subsurface contamination on the exterior surfaces of below-grade foundations. If there is a potential for subsurface contamination, the licensee plans on performing biased sampling. The licensee proposes two sampling approaches: (1) core boring through the foundation with biased soil sampling and (2) gamma well logging (requiring soil excavation) in biased locations next to the exterior of buildings.

The licensee will survey subsurface soil areas in accordance with Section 5.7.3.2.2 of the LTP, through combination of systematic and biased measurements. The licensee has committed to using the DQO process for subsurface areas. The licensee states that this process will be “similar” to the DQO process used for other surveys at the Haddam Neck Plant; however, there may be “differences in design input parameters,” to satisfy objectives of the plan. As with surface soil, the NRC staff will review the DQOs and survey plans for subsurface soil as part of the inspection process. Although the licensee has specified a specific number of samples for each survey class a priori (up front), the licensee must demonstrate (post priori) that these numbers are statistically significant for each survey unit. Because of limited characterization data, it cannot be determined, at this point in time, whether the range of number of measurements in Classes A, B, and C areas (31 measurements in the Class A area to 15 measurements in the Class C areas) will be appropriate to meet DQOs (specifically, an alpha error of 5 percent). The range of number of measurements that the licensee has identified cannot be used as an indicator of compliance until the variability of the data is determined. If the variability of the data is great, it is likely that the data quality objectives will not be met and result in the need for additional action, such as remediation or additional samples and modification of the FSS sampling plan. The NRC staff will review FSS design changes as part of the inspection process and will review the DQOs to see if changes are adequately reflected.

The use of sampling or in-situ gamma spectroscopy, by well logging or other advanced technology is acceptable provided the MDC meets the criteria shown in Section 5.7.2.1 of the LTP. If advanced survey technology is used, the licensee has agreed to provide a technical basis document for NRC review and approval before the survey is implemented.

The licensee plans to evaluate all subsurface samples against the surface soil DCGLs by using either the Sign or Wilcoxon Rank Sum (WRS) test. Investigation levels applicable to surface soils will be applied to subsurface soils. Similarly, the area factors for surface soils will be applied to subsurface soils. The licensee states, in Section 5.7.3.2.2 of the LTP, that a minimum of 5 percent of the samples will be analyzed for hard-to-detect radionuclides. During specific investigations, such as identification of horizontal extent of contamination, analysis of a larger percentage of samples for hard (difficult)-to-detect radionuclides will be performed. The NRC staff will review the licensee’s data to determine whether: (1) a minimum of 5 percent of the samples is analyzed for hard (difficult)-to-detect radionuclides for each survey unit and (2) subsurface soil characterization and demonstration of compliance with surface soil DCGLs for each subsurface survey unit are addressed separately from surface soil survey units.

The NRC staff finds the licensee’s subsurface soil characterization strategy and initial subsurface area classification acceptable.

3.1.1.6 Ground Water

The LTP and supporting references [i.e., (1) “Groundwater Monitoring Report” -- September 1999 by Malcolm Pirnie, Inc.; (2) “Groundwater Monitoring Report” -- January 2001 by CYAPCO; and (3) “Phase 2 Hydrogeologic Investigation Work Plan” -- May 2002 by Malcolm Pirnie, Inc.] indicate that plant-generated radionuclides have impacted the ground water at the Haddam Neck site. The licensee’s initial evaluation, using existing monitoring wells, was not sufficient to adequately evaluate the spatial, both vertical and horizontal, impact of plant-generated spills and leaks of radionuclides on the ground water within and adjacent to the IA.

Therefore, the licensee will supplement its existing hydrogeological data in the industrial area with additional radionuclide characterization of the ground-water system. The licensee has developed a Phase 2 Hydrogeologic Work Plan that calls for additional radionuclide characterization of the ground water in the IA. The Connecticut Department of Environmental Protection (CTDEP) required the development of this work plan. Both the CTDEP and the NRC staff provided guidance to the licensee in the development of the work plan and technical evaluation of the work plan.

The Haddam Neck site consists of about 2,124,675 square meters (525 acres) of predominantly wooded land, excluding the IA, parking lot, and the northern peninsula area. The site is bounded by the Salmon River on the east and northeast and by the Connecticut River to the south and southwest. The IA, parking lot, the 1,676 meter (5,500 foot) discharge canal, and the 152-to-305-meter (500-to-1,000-foot)-wide peninsula occupy a terrace and flood plain of the Connecticut River. Construction of the IA required removal of a north-south trending bedrock promontory that projected from the steep hillside north of the IA into the terrace and flood plain, and excavation into the unconsolidated materials and bedrock of the terrace and flood plain.

The geology of the IA and nearby areas is variable and complex. The unconsolidated materials, with increasing depth are: (1) sand and/or rock fill; (2) wetland organic silt and river alluvium (gray interbedded clay, silt, and sand); (3) gravelly sand; (4) red fine sand; (5) red-brown fine sand; and (6) glacial till. The bedrock in the IA and nearby areas is the Connecticut Yankee Complex. It is a metamorphic gneiss and amphibolite with mineral layering that strikes north-south and dips from vertical to 65 degrees to the east.

The construction of the plant structures and the discharge canal within the IA and nearby areas, together with the geology in this area, has generated a complex hydrogeologic system. The impact of the tidal Connecticut River on nearby ground-water levels further complicates the ground-water flow in this area.

Currently, the Haddam Neck site has approximately 24 monitoring wells in the IA and northern peninsula area; 10 monitoring wells in the parking lot; two monitoring wells in the Emergency Operations Facility area; 3 monitoring wells in the water supply area of the peninsula; and 8 monitoring wells in the landfill area.

Existing radiological analyses for monitoring wells in the IA indicate that H-3, Cs-137, and Sr-90 are present in the ground water above background levels. Recent (samples collected in December 2001) ground-water concentrations of H-3 in wells MW-110D and MW-101D are 21,300 and 20,600 pCi/L, respectively. Cs-137 has been currently detected only in well

MW-103S (29-76 pCi/L). Thus far, Sr-90 has been analyzed for and detected in wells MW-103S (2.6 pCi/L), MW-105S (143 pCi/L), and MW-106S (6.6 pCi/L). Also, boron, a non-radionuclide, has been observed in several monitoring wells above background levels. Boron is a plant-generated contaminant that is indicative of plant spills and leaks that have reached the ground water.

Additional monitoring wells are needed to delineate the horizontal and vertical extent of H-3, Cs-137, and Sr-90 ground-water plumes within and near the IA. The licensee must also address the hydraulic characteristics of the unconsolidated and bedrock water-bearing units, and it must analyze for all potential plant-generated radionuclides in the ground water.

The licensee has agreed to perform the hydrogeologic and radiological work outlined in the Phase 2 Hydrogeologic Investigation Work Plan that will provide the above information. This work plan includes the following items: (1) development of a three-dimensional conceptual model of the site, using "GMS," a ground-water modeling software program (the conceptual model will be generated from existing hydrogeological and radiological data and from new data collected during this study); (2) investigation of the effects of tidal changes in the Connecticut River and discharge canal on ground-water levels in the unconsolidated water-bearing units; (3) evaluation of the hydraulic characteristics of the unconsolidated and bedrock units, using geophysical bore-hole procedures, slug tests, and tidal and ground-water level fluctuations where appropriate; (4) installation of monitoring wells in both the unconsolidated and bedrock water-bearing units; and (5) sampling and analyzing ground water from the existing and new monitoring wells (procedures for the sampling frequency, ground-water level measurements, and the suite of radionuclides and chemical parameters to be analyzed are provided in the work plan).

The successful performance of the items listed in the work plan will provide sufficient information on the ground-water flow patterns, ground-water flow rates, and current concentrations of plant-generated radionuclides in the ground water that are essential in evaluating compliance with radiological criteria for license termination. The NRC staff will periodically evaluate the licensee's progress in performing the work items listed in the work plan during inspections.

The plant-generated radionuclides will be used in the development of operational DCGLs ($DCGLs_{op}$) as described in Section 5.4.7.1 of the LTP. Furthermore, results from the Phase 2 Hydrogeologic Work Plan will be used in the capture zone analysis to ensure that the area of impact has been established for implementation of the $DCGLs_{op}$.

Successful implementation of the licensee's Phase 2 Hydrogeologic Work Plan will provide the radionuclide concentrations in the ground water that will then be used in developing the operational DCGLs. The NRC staff has determined that the licensee's ground-water characterization strategy with respect to radionuclide fate and transport is consistent with industry practices. Therefore, this approach to derive the radionclide concentrations in the ground water is acceptable.

3.1.1.7 Surface Water

Surface waters at the Haddam Neck site consist of: (1) the Connecticut River on the southwest boundary of the site; (2) a discharge canal that parallels the Connecticut River; (3) the Salmon River and Salmon Cove, which discharge into the Connecticut River; (4) Dibble Creek, which discharges into Salmon Cove; and (5) two small surface-water ponds. The Salmon River, Salmon Cove, Dibble Creek, and the surface-water ponds have not been impacted by plant-generated radionuclides; however, the Connecticut River and discharge canal have been. The power plant, during its operation, withdrew once-through cooling water from the Connecticut River, through an intake structure at the edge of the river. The cooling water effluent was discharged into the discharge canal that flows parallel to the river, with its outflow located approximately 1,676 meters (5,500 feet) downstream of the intake.

Because the Connecticut River is tidally controlled at the site, stream flow at the site is a combination of upstream basin discharge and tidal interchange. The average annual daily flow at Haddam Neck is approximately 481 cubic meters per second (cms) [17,000 cubic feet per second (cfs)]. Tidal flow at the site averages about 425 cms (15,000 cfs), but it may be as great as 623 cms (22,000 cfs). Saline water extends only as far north as East Haddam, about 3.2 kilometers (2 miles) south of the plant. No drinking water intakes exist on the Connecticut River in the vicinity of the site; local water supply needs are provided by wells or tributary-stream reservoirs.

The licensee has sampled the Connecticut River at a control and an indicator site each quarter, as part of its radiological environmental management program. The control site is at Middleton, Connecticut, approximately 14.4 kilometers (9.0 miles) northwest (upstream) of the reactor site, and the indicator site is at the East Haddam bridge about 2.9 kilometers (1.8 miles) southeast (downstream) of the reactor site. The surface water is analyzed for gamma isotopes and tritium, and there has been no surface-water contamination attributable to the site, except tritium concentrations within 1.5 times two standard deviations (2σ error). Since 1994, the tritium concentrations have been below the minimum detectable activity.

The NRC staff has determined that the plant's impact on the surface water is minimal because of the dilution effect of the Connecticut River on the ground-water discharge into the river and on the surface-water effluent from the plant.

3.1.1.8 Sediment

The licensee will assess sediment by collecting and evaluating samples within locations of surface-water ingress or by collecting and evaluating composite samples of bottom sediments, as appropriate. Scanning in such areas is not appropriate, because of wet conditions. Sample locations will be established according to Section 5.5.1 of the LTP. The licensee will adjust the sampling density in the area of the canal, from the outfall to 15 meters (50 feet) past the weir. The sampling density for this area will be twice the density that would otherwise be required for a Class 2 survey unit. Sediments will be evaluated against the soil DCGLs. Through the inspection process, the NRC staff will verify that the LTP requirements that apply to surface-soil sampling and analysis and data assessment have been applied to sediments.

The NRC staff has determined that the licensee's sediment-characterization strategy is consistent with the guidance in MARSSIM and is therefore acceptable.

3.1.1.9 Pavement

The RCA consists of paved areas around the containment building, PAB, reactor-water-storage tank, waste-storage tanks, and the spent fuel building. The HSA identified several events involving unplanned liquid releases that have radiologically impacted the RCA. Portions of the RCA have been posted as contaminated because of system leakage. Several documented events that occurred during the course of plant operations have led to contamination of other portions of the RCA. By reviewing the results of radiological surveys performed during the operational phase of Haddam Neck, several areas within the RCA, which have been identified by the licensee, contained radioactive material in excess of the DCGLs. Therefore, the licensee has initially classified the RCA ground survey units as Class 1.

In December 1997, surveys of paved areas in the IA, but outside the RCA, were performed. The surveys were conducted using a Position Sensitive Proportional Counter. Essentially all paved areas not restricted by trailers or snow piles were surveyed. Seven discrete areas of contamination ranging from 10,000 to 65,000 dpm were identified. Each area was limited to less than 100 cm^2. The licensee remediated these areas before it completed the survey. The licensee references the survey in Section 2.4 of the LTP (i.e., "Executive Summary of Radiation Surveys Performed at Connecticut Yankee Atomic Power Station," dated January 22.1998, Millennium Services, Inc., Reference 2-6, page 2-126).

The licensee stated in Section 5.7.3.2.2 of the LTP that paved areas that remain at the Haddam Neck site after 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, the licensee will review historical information, to determine whether radiological incidents or plant alterations have occurred in the survey unit. If there are indications of impacted soil that could have been mixed by grade work before paving, this will be considered in the FSS design to establish a reasonable depth of disturbed soil evaluation. If it is determined that soil beneath the pavement has been impacted, the FSS will incorporate appropriate surveys and sampling. The licensee states that, if the residual activity is primarily on or near the surface of the paved area, for purposes of surveying, measurements will be taken as if the area were surface soil. If the residual radioactivity is primarily beneath the paving, the licensee will treat it, for the purposes of surveying, as subsurface residual radioactivity. The NRC staff will review the RSSI data used to characterize such areas. Based on the licensee's above measurement approach and the RSSI data, if the residual radioactivity at the time of characterization is primarily beneath the pavement, then NRC will, through the inspection process, review the FSS design for such areas against the licensee's subsurface-soil FSS strategy. This includes areas with multiple layers of paving material. Also, NRC will, through the inspection process, examine the licensee's RSSI data covering the areas noted above that were restricted by trailers or snow piles to verify that the survey unit classification and FSS design are appropriate.

The NRC staff has determined that the licensee's pavement characterization strategy is acceptable.

3.1.1.10 Exposure Rate Survey

In Tables 2-11A and 2-11B of the LTP, the licensee provided limited exposure rate data for structures and areas, respectively. The type of instrument used, the distance from the source, and the detection sensitivity were not noted. Instrumentation used to obtain exposure rate data must be consistent with the guidance in Section 6.0 of MARSSIM. The NRC staff will review the licensee's exposure rate data during the RSSI process and in the FSS Report as part of the inspection process to determine whether they are acceptable and whether instrument use and data analysis are consistent with MARSSIM guidance.

3.1.2 Site Characterization - Summary Finding

The NRC staff has reviewed the information provided in the Haddam Neck LTP, according to Section B.2 of NUREG-1700, “Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans.” Based on this review, the NRC staff has determined that the licensee has met the objective of providing adequate site characterization information, as required by 10 CFR 50.82(a)(9)(ii)(A). Although the licensee has not yet fully characterized the site, it has committed to do so by using the RSSI process according to MARSSIM. The NRC staff finds the licensee’s characterization strategy acceptable.

3.2 Remaining Dismantlement Activities

In accordance with the requirement of 10 CFR 50.82(a)(9)(ii)(B), the licensee provided the status of dismantlement and decontamination of the Haddam Neck Plant major systems, structures, and components, as of May 2000. Also, in accordance with the guidance provided in NUREG-1700 and Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," the licensee provided a radioactive waste characterization, an estimate of the quantity of radioactive material that will be shipped for offsite burial, an estimate of personnel exposures, and the methods that will be used to control the spread of contamination while performing these dismantlement activities.

The licensee estimated that there will be about 6743 cm (283,117 cf) of radioactive waste generated as a result of decommissioning activities. In addition, the licensee estimates the total radiation exposure to complete the decommissioning of the facility to be about 10 person-Sievert (Sv) (1007 person-rem). This estimate includes exposure from transportation of the waste. The licensee indicated that the remaining dismantlement activities will continue to be conducted under the existing Haddam Neck Plant Radiation Protection Program and Radioactive Waste Management Program.

The NRC staff has reviewed the information in the LTP for the Haddam Neck Nuclear Plant, according to Section B.3 of NUREG-1700. Based on this review, the NRC staff has determined that the licensee has identified, in sufficient detail, the remaining dismantlement activities necessary to complete decommissioning of the facility, as required by 10 CFR 50.82(a)(9)(ii)(B). Further, the NRC staff has determined that the remaining dismantlement activities can be completed in accordance with 10 CFR 50.59.

3.3 Plans for Site Remediation

In accordance with the requirement of 10 CFR 50.82(a)(9)(ii)(C), the licensee provided its plans for completing the radiological remediation of the site. The licensee plans to remediate the site, including structures and systems that remain on site, to the criteria specified in 10 CFR Part 20, for unrestricted use. To meet these criteria, the licensee plans to use typical remediation methods, which include chemical decontamination, wiping, washing, vacuuming, scabbling, spalling, and abrasive blasting. For radiologically contaminated structures and systems the licensee plans to either: (1) remove them and send them to an offsite processing facility or to a low-level radioactive waste facility for disposal; or (2) decontaminate them onsite and ensure that any residual radioactivity remaining meets the release criteria for unrestricted use.

Included in the criteria specified in 10 CFR Part 20 for release for unrestricted use is that, in addition to the remaining residual radioactivity being less than or equal to 0.25 mSv/yr (25 mrem/yr) above background, the remaining residual radioactivity must be reduced to levels that are ALARA. The licensee also provided its ALARA analysis process. From its ALARA analysis the licensee will calculate the remediation levels based on cost of remediation versus the benefits derived from additional remediation. The licensee's formulas for calculating the remediation levels conform to the guidance provided in NUREG-1727, "NMSS Decommissioning Standard Review Plan."

The NRC staff has reviewed the information in the LTP for the Haddam Neck Plant according to Section B.4 of NUREG-1700. Based on this review, the NRC staff has determined that the licensee has a process that will adequately identify site areas requiring remediation and has in place an organization to safely perform the remediation as required by 10 CFR 50.82(a)(9)(ii)(C). Further, the NRC staff has determined that there is reasonable certainty that the licensee can complete remediation at this site and meet the criteria specified in 10 CFR Part 20 for unrestricted use.

3.4 Final Site Survey

The FSS is the radiation survey performed after an area has been fully characterized; remediation, as applicable, has been completed; and the licensee believes that the area is ready to be released for unrestricted use. The purpose of the FSS is to demonstrate that the area meets the radiological criteria for license termination. The FSS design entails an iterative process that requires appropriate site classification—based on the potential residual radionuclide concentration levels relative to the DCGLs—and formal planning. An integrated design is developed that addresses selection of appropriate survey and laboratory instrumentation and procedures and a statistically based measurement and sampling plan for collecting and evaluating the FSS data. Sections 5.4, "Survey Planning"; 5.5, "Final Status Survey Design Elements - Surface Soils and Structures"; 5.6, "Survey Protocol for Non-Structural Systems and Components"; 5.7, "Survey Implementation and Data Collection"; of the LTP, were reviewed, to ensure that the design was appropriate and all applicable variables had been addressed. In many cases, information was not available for the licensee to complete all aspects of the design. The licensee will have to gather this information as the decommissioning progresses. For those survey design aspects that could not be completed, the licensee has committed to follow appropriate guidance contained in NUREG-1575 (MARSSIM), or, alternatively, will prepare a technical basis document for deviations from the guidance for NRC review and approval. This approach to final survey design is acceptable.

The NRC staff recognizes that not all the information required to properly design the FSS will be available when a licensee submits its LTP for review. The NRC staff, therefore, conducts performance-based in-process inspections of the licensee's final site program at various stages in its decommissioning process. The purpose of the inspections is to verify the implementation of the commitments made by the licensee in the LTP and to review the procedures, methodology, equipment, training and qualifications, and QC. Implementation of DCGLs, embedded piping surveys, detection sensitivities, instrument calibration, reference background areas, area factors, QA/QC, and other areas, may be subjects for future inspections.

The licensee used initial site characterization data (historical data or scoping data), together with process knowledge and operational and routine surveillance survey records, as the principal means for initially classifying site areas as non-impacted or impacted. Table 2.10 of the LTP lists each survey unit, the size of the survey unit, and the initial classification, as either Class 1, 2, or 3, for surface soil and structures and either Class A, B, or C, for subsurface soil. Survey unit sizes that the licensee designated as Class 1, 2, or 3 were evaluated relative to the recommendations provided in MARSSIM and were within the recommended area limits. As explained in Section 5.5 of the LTP, MARSSIM does not directly address FSS design for subsurface soils. Survey areas for subsurface soils include any subsurface features that are present, such as piping and drain systems. Subsurface survey units that the licensee has identified thus far are depicted in Figures 2-1 through 2-21. The licensee has committed to use the principles of MARSSIM to guide the design of subsurface surveys. Subsurface survey considerations are found in Section 5.7.3.2.2 of the LTP.

The licensee, in Section 5.4.1 of the LTP, has committed to use the DQO process throughout the data life cycle for a survey unit. The licensee states it will use the DQO process in the planning phase for scoping, characterization, remediation, and FSS plan development, using a graded approach.

The DQO process consists of seven steps. The output from each step influences the choices that will be made later in the process. However, the process is iterative, allowing the survey planning team to incorporate new knowledge and modify the output of prior steps. Section 5.4.1 of the LTP discusses the steps and the actions the licensee plans to take in addressing the steps of the DQO process. The NRC staff finds the licensee's DQO process consistent with MARSSIM.

Although the licensee states that it has a Final Status Survey Quality Assurance Project Plan that provides a detailed description of the application of the DQO process to the different elements of the FSS, it is unclear from Section 5.4.1 of the LTP whether the licensee intends to produce DQOs. DQOs are qualitative and quantitative statements derived from the DQO process that clarify the study objectives, define the appropriate type of data, and specify tolerable levels of potential decision errors that will be used as the basis for establishing the quality and quantity of data needed to support decisions. Usually, for each survey unit or group of similar survey units, these statements are documented in one place and referred to as the DQO. The licensee does state, in Section 5.10 of the LTP, that, to support site characterization and the FSS, QA project plans as well as DQOs will be developed. However, the format of the DQO is unspecified. During the RAI comment-resolution process, the licensee did not have a consolidated record of all the outputs to the DQO process nor could this information be easily assembled. Because the DQO process is iterative, throughout the DQO process for each survey unit or group of similar survey units, the NRC staff will review the licensee's documentation that clarifies the study's technical and quality objectives, defines the appropriate type of data, and specifies tolerable levels of potential decision errors that will be used as the basis for establishing the quality and quantity of data needed to support decisions. In addition, the NRC staff will verify if the licensee has documented the minimum information (outputs) required from the DQO process with the methods described in MARSSIM. The NRC staff will refer to the minimum outputs required, which are listed in Section 2.3.1 of MARSSIM.

The statistical tests discussed in MARSSIM are the WRS test and the Sign test. The WRS test is typically selected when the radionuclides of concern are present in background, or gross measurements are to be made. The WRS test also requires the identification of appropriate background reference areas from which the same sample or measurement type is collected as was collected from the survey unit. The reference area data are adjusted for the DCGL and then the two data sets are compared to demonstrate release criteria compliance. Alternatively, the Sign test may be selected if the radionuclides of concern are not present in background or are present at a small fraction of the DCGL. Finally, consistent with NUREG-1507, the Sign test may be used to evaluate gross activity measurements from survey units containing multiple materials, by subtracting the appropriate background and using paired measurements. Sections 5.4.4 and 5.4.6 of the LTP state that the licensee may compare the total radionuclide concentrations, including background, to the release criteria. Should conditions require application of the WRS test, Section 5.4.5 of the LTP establishes the methods that the licensee would use to select background reference areas.

The input parameters for sample size calculations include the DCGL, the Lower Bound of the Gray Region (LBGR) -- which generally provides an estimate of the mean concentration in the survey unit, but may be adjusted to optimize design -- and an estimate of the radionuclide variability. These parameters, together with decision errors, are used to calculate the required number of statistical samples, and the information is usually available from planning or from preliminary surveys (scoping, characterization, remedial action support). For initial planning purposes, if site-specific data are not available, the licensee has decided to set the LBGR at 50 percent of the DCGL and set the default decision errors at 0.05; however, the licensee has not discussed how it will estimate the radionuclide variability. MARSSIM recommends that, if the preliminary surveys are not available, it may be necessary to: (1) perform some limited measurements (about 5 to 20) to estimate the distribution, or (2) make a reasonable estimate based on available site knowledge. If the licensee decides to make some limited measurements, it is important that the data used to estimate the standard deviation use the same technique that will be used during the FSS. When preliminary data are not obtained, it may be reasonable to assume a coefficient of variation on the order of 30 percent based on experience. Eventually, before FSS is implemented, there should be preliminary data that can be used for the above purpose.

Because survey units are not finalized until the planning stage of the FSS, the NRC staff recognizes that before the FSS planning is complete, there may be some difficulty in determining which individual measurements from a preliminary survey may later represent a particular survey unit at the time of FSS. The importance of choosing appropriate values of standard deviation, for the survey unit and the reference area, for incorporation in the FSS plan, at the completion of the FSS planing phase, must be emphasized. If the value is grossly underestimated, the number of data points will be too few to obtain the desired power level for the test, and a resurvey may be recommended. If, on the other hand, the value is overestimated, the number of data points determined will be unnecessarily large.

The principal decision error of concern to the NRC staff for survey design inputs is the Type I or α error. This error occurs when a survey unit is determined to meet the release criteria when in fact it does not. The MARSSIM default value of 0.05 for the Type I or α error is acceptable.

MARSSIM allows the use of advanced survey technologies as long as these techniques meet the applicable requirements for data quality and quantity. The decision to use advanced technologies and other methods may not be a matter of convenience for a licensee, but a genuine attempt to use the best instruments, which can minimize the potential for residual radioactivity to exceed the release criterion. For example, by using a system that can sample the entire population or surface, a licensee can determine if there exists any combination of areas and activity that could potentially exceed the dose limit.

The licensee has elected to defer the details regarding the use of advanced instrumentation to technical basis documents, which will be available to NRC staff for inspection. In Chapter 5 of the LTP, there are eight references to technical support documents. More specifically, if the licensee wants to employ advanced survey techniques or other instruments and methods such as in situ gamma spectrometry, in situ object counting systems, and systems capable of traversing ducting or piping, the licensee has agreed to do so only after the NRC staff has reviewed the licensee's technical support document describing the technology "to be used" and how the technology meets the objectives of the survey. NRC staff will inspect this documentation to determine: (1) whether the technical basis document is of high quality, with enough detail to make an accurate evaluation of its contents; (2) whether the technical basis is acceptable; (3) how the licensee intends to evaluate the data, especially if different statistical techniques or methods are to be used; (4) how the DQOs for the advanced technology or methods dovetail into MARSSIM; and (5) whether the objective of the survey can be met. Also, NRC staff will inspect to determine whether the licensee has met the conditions that the licensee has established in Section 5.5 of the LTP, regarding the implementation of advanced survey techniques. The NRC staff finds acceptable the licensee's technical basis documentation submittal approach, for using advanced technologies and methods.

The licensee has further committed to evaluate the acceptability of the selected input parameters when assessing final status data. The statistical survey planning approach discussed in the LTP is acceptable.

The LTP discussion of the reference system that will be used for structures, systems, and land areas is provided in Sections 5.4.3 and 5.4.5. Overall, the proposed grid sizes are appropriate for the survey unit classification and the type of survey unit (i.e., structure, land area, or system). Grid coordinates will serve as the basis for the random-start systematic sample-location selection. The recommended guidance has been adapted for subsurface soils and is also acceptable.

FSS-meter/detector selection, calibration, and evaluation are discussed in Section 5.7.2 of the LTP. The meter/detector selection process must ensure that the instrumentation used for RSSI surveys will adequately respond to the radiation emitted from the various radionuclides of concern, that the instrumentation is sufficiently sensitive to detect these radiation at levels less than the DCGLs, and the instrumentation is calibrated in a manner that accounts for the radionuclide mixture, the expected radiation energies of the mixture, and surface efficiencies. The instrumentation presented in Table 5-10 of the LTP is appropriate for the primary detectable radionuclides, for surveys of structures and land areas. Section 5.7.2.3 provides a discussion of the expected calibration sources that will be used and accounts for other modifying factors, specifically surface efficiency. The licensee will use National Institute of Standards and Technology traceable calibration sources that are similar in energy to the primary radionuclides of concern. The static MDC calculations in Section 5.7.2.4 are appropriate and the MDCs for both scanning and static measurements are less than the DCGLs for those beta-gamma emitters that are detectable using field instrumentation.

The discussion of review of instrument response checks in Section 5.7.2.4, concluded that the provisions for confirming the constancy of the instrumentation before use each day is acceptable. Typically, the licensee performs an instrument response check before issue and after use each day; however the licensee states that the DQO process determines the frequency of response checks. Should a response check fail, the instrument is removed from service and any data collected since the last acceptable check are evaluated and may be discarded.

The FSS process, discussed in Section 5.7.2 of the LTP, addresses methods for performing surface scans, surface activity measurements, soil and bulk material sampling, and special measurements. The licensee will evaluate the adequacy of the scanning techniques by calculating a scanning MDC, the concentration that a specific instrument or technique can be expected to detect 95 percent of the time under actual conditions of use, expressed as a fraction of the elevated measurement comparison DCGL ($DCGL_{EMC}$) when multiple radionuclides are present, as described in Section 5.5.1.5 of the LTP. Surface-scan descriptions recognize the importance of surface-to-detector distance and scan speed to achieve an adequate scan sensitivity. The licensee provided a discussion, in Section 5.7.2 of the LTP, for evaluating the required-scan MDC to the actual-scan MDC. If adequate sensitivity is not achieved, additional measurements or samples are required. The scan coverage is based on survey unit classification, with Class 1 survey units receiving 100 percent scan coverage and Classes 2 and 3 receiving coverages of 10 to 50 percent (or greater), and up to 10 percent, respectively. The licensee states, in Section 5.5.1.6, that measurement/sampling locations are to be determined, based on a random-start systematic pattern for Classes 1 and 2 survey units and randomly for Class 3 survey units. Additional measurements or samples will also be collected from areas of elevated radioactivity detected while scanning and from judgmental locations. These proposed methodologies for surveys are acceptable and generally follow MARSSIM guidance.

MARSSIM does not apply to non-structural systems and components. The licensee will use the current site-release guidance [i.e., IEC 81-07, Information Notice (IN) 88-22, "Disposal of Sludge From Onsite Sewage Treatment Facilities at Nuclear Power Stations," and IN 85-92, "Surveys of Wastes Before Disposal from Nuclear Reactor Facilities,"] for determining non-detectable limits for non-structural systems and components, except for buried piping. The licensee clearly states that those non-structural systems and components 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 that will remain onsite will be surveyed to the limits set forth in Table 5-7. The licensee has committed to perform “full-length surveys” for this piping, typically using conventional methods and instrumentation. Detection sensitivity will be at least equivalent to the release limits given in Table 5-7 of the LTP, at the 95 percent confidence level. Detection sensitivity will be computed using the methods described in Section 5.7.2.5 of the LTP. The NRC staff’s evaluations of both buried and embedded piping are provided in Section 3.5.1 of this report.

The LTP does not discuss in detail the process for sample handling and analysis. In Section 5.7.1.5 of the LTP, the licensee notes that it will control sample tracking using a chain of custody record for all sample activities, to ensure sample integrity.

Section 5.10 of the LTP addresses QA requirements that apply to onsite and offsite laboratories. Onsite sample analysis capabilities are not specified. Gamma spectroscopy, liquid-scintillation counting, and gross alpha/beta proportional counting analyses are appropriate for the primary radionuclides of concern at the site. Other radionuclides, such as Sr-90 and TRU, will require wet-chemistry analysis and may be performed at an offsite laboratory. The licensee does not note the sensitivity of the analytical procedures. The sensitivities should be between 10 to 50 percent of the DCGLs, to be acceptable. The licensee states, in Section 5.10, that to support site characterization and the FSS, QA project plans will be developed.

The survey data assessment process and investigation levels are discussed in Sections 5.6 and Sections 5.4.4.1 through 5.4.4.5. The licensee's data assessment, as described in the LTP, involves data validation, graphical data reviews, basic statistical evaluations, and statistical data testing, when applicable. The data validation is presented as an eight-step process, to ensure data quality and defensibility. Any discrepancies identified must be investigated. The graphical data reviews that the licensee intends to use serve to identify spatial patterns and potential anomalies that would indicate additional investigation is required. The basic statistical evaluation serves as a method to evaluate the adequacy of survey design and lists specific acceptance criteria. Finally, data testing would be performed when necessary, using either the WRS or Sign test, or advanced survey technologies and methods, as discussed above. These approaches are acceptable.

The licensee has established investigation-level requirements, a process for evaluating the results of the investigation, and follow-up actions. Requirements for investigation are related to the survey unit classification and the DCGL. Elevated activities detected while scanning or from measurements are investigated. The licensee states that the investigation may result in remediation, reclassification of a given survey unit to a higher level, and/or evaluation of the elevated area to a $DCGL_{EMC}$. These processes are acceptable as a means to ensure data quality and adequate investigations of anomalies, evaluation of the final status survey design, and assurance that the release criteria are satisfied for each survey unit.

When reviewing FSS results, the NRC staff will examine whether the licensee has measured and/or appropriately accounted for each of the radionuclide contaminants (Table 6.1 for soil; Table 6.2 for ground water; Table 6.3 for building surfaces; and Table 6.4 for concrete debris) when presenting dose compliance information for each survey unit. Additionally, whenever the licensee accounts for HTD radionuclides through surrogate analyses, the NRC staff will examine whether the licensee has verified whether the activity ratios (activity ratio for difficult-to-detect radionuclides to easy-to-detect radionuclides) remain valid for use during the FSS in accordance with Sections 5.4.7.1, 5.4.7.2, 5.4.7.3, 5.5, and 5.7.3 of the LTP.

Section 5.9 of the LTP provides a brief description of the FSS documentation. The information that is to be compiled for each survey unit includes a history file and release record. At project completion, the licensee will prepare an FSS report summarizing the ALARA evaluations, survey data results, and overall conclusions, as they relate to the radiological criteria for release for unrestricted use. The planned presentation of the site's final radiological status is acceptable, although the adequacy of the site documentation cannot be determined until the licensee has had an opportunity to begin compiling FSS records.

The NRC staff has reviewed the information in the LTP for the Haddam Neck Nuclear Plant according to Section B.5 of NUREG-1700. Based on this review the NRC staff has determined that the licensee has conformed to 10 CFR 50.82(a)(9)(ii)(D) in that the final radiation survey plan in the LTP provides assurance that residual radioactive contamination levels will meet the criteria specified in 10 CFR Part 20 for unrestricted use.

3.5 Compliance with Radiological Criteria for License Termination

The development of residual radionuclide concentration levels that will be used to demonstrate compliance with the regulations for releasing the site for unrestricted use (10 CFR 20.1402) is discussed in Section 6 of the LTP. Two primary scenarios were considered in developing the radionuclide-specific base-case DCGLs for the Haddam Neck site: a building occupancy scenario and a residential farming scenario. CYAPCO has elected to use the RESRAD computer code, Version 5.91, to model doses resulting from exposures to contaminated soil, contaminated concrete debris from demolished buildings, and contaminated ground water, and to use the RESRAD-BUILD computer code, Version 2.37, to model doses resulting from exposures to contaminated building structures.

As discussed in NUREG-1727, the question is either: “How could humans be exposed either directly or indirectly to residual radioactivity?" or “What is the appropriate exposure scenario?” Each exposure scenario must address the following questions:

(1) How does the residual radioactivity move through the environment?

(2) Where can humans be exposed to the environment concentrations?

(3) What are the exposure group’s habits that will determine exposure? (e.g., what do they eat and where does it come from? How much? Where do they get water and how much? How much time do they spend on various activities? etc.)

In most situations, there are numerous possible scenarios of how future human exposure groups could interact with residual radioactivity. The compliance criteria in 10 CFR Part 20 for decommissioning do not require an investigation of all (or many) possible scenarios; their focus is on the dose to members of the critical group. The critical group is defined (in 10 CFR 20.1003) as "the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances."

By combining knowledge about the sources of residual radioactivity and the scenario questions, the analyst can develop exposure pathways. Exposure pathways are the routes that residual radioactivity uses to travel from its source, through the environment, until it interacts with a human. They can be fairly simple (e.g., surface soil residual radioactivity emits gamma radiation which results in direct exposure to the individual standing on the soil), or they can be fairly involved (e.g., the residual radioactivity in the surface soil leaches through the unsaturated soil layers into underlying aquifer, and the water from the aquifer is pumped out by the exposed individual for use as drinking water, which results in the exposed individual ingesting the environmental concentrations). Exposure pathways typically fall into three principal categories identified by the manner in which the exposed individual interacts with the environmental concentrations resulting from the residual radioactivity; the three principal categories are ingestion, inhalation, or external (i.e., direct) exposure pathways.

The exposure pathways for many of the exposure groups can be bounded by a smaller number of possible exposure groups. For example, at a site with surface soil residual radioactivity, two possible exposure groups are (1) a gardener who grows a small fraction of his or her fruits and vegetables in the soil and (2) a resident farmer who grows a larger fraction of his or her own food, (i.e., the site supplies not only vegetables, but also meat and milk). In this case, the resident farmer bounds the gardener exposure group (because it both incorporates the gardener's pathways and also includes other routes of exposure). Therefore, the gardener exposure group does not need to be analyzed and the compliance calculation's scenario would involve the resident farmer.

As required by 10 CFR 20.1302(b), expected doses are evaluated for the average member of the critical group, which is not necessarily the same as the maximally exposed individual. This is not a reduction in the level of protection provided to the public, but an attempt to emphasize the uncertainty and assumptions needed in calculating potential future doses, while limiting boundless speculation on possible future exposure scenarios. While it is possible to actually identify with confidence the most exposed member of the public in some operational situations (through monitoring, time-studies, distance from the facility, etc.), identification of the specific individual who might receive the highest dose some time (up to 1000 years) in the future is impractical, if not impossible. Speculation on his or her habits, characteristics, age, or metabolism could be endless. The use of the "average member of the critical group" acknowledges that any hypothetical "individual" used in the performance assessment is based, in some manner, on the statistical results from data sets (i.e., the breathing rate is based on the range of possible breathing rates) gathered from groups of individuals. While bounding assumptions could be used to select values for each of the parameters (e.g., the maximum amount of meat, milk, vegetables, possible exposure time), the result could be an extremely conservative calculation of an unrealistic scenario and may lead to excessively low allowable residual radioactivity levels.

Calculating the dose to the critical group is intended to bound the individual dose to other possible exposure groups because the critical group is a relatively small group of individuals, due to their habits, actions, and characteristics, who could receive among the highest potential dose at some time in the future. By using the hypothetical critical group as the dose receptor, coupled with prudently conservative models, it is highly unlikely that any individual would actually receive doses in excess of that calculated for the average member of the critical group. The description of a critical group's habits, actions, and characteristics should be based on credible assumptions and the information or data ranges used to support the assumptions should be limited in scope to reduce the possibility of adding members of less exposed groups to the critical group. An analysis of the average member of the critical group's potential exposure should also include, in most cases, some evaluation of the uncertainty in the parameter values used to represent physical properties of the environment.

Because of the definitions in 10 CFR Part 20**,** when calculating for compliance with the requirements of Subpart E, the intake-to-dose conversion factors used to calculate internal exposures can be found in Federal Guidance Report No. 11, Environmental Protection Agency (EPA) EPA-520/1-88-20, which are based primarily on adults. As stated in EPA's Draft Guidance for Exposure of the General Public (EPA 1994) *Federal Register* Notice (FR 66414, dated December 23, 1994, on "Federal Radiation Protection Draft Guidance for Exposure of the

General Public"), which proposes a public dose limit of 1.0 mSv (100 mrem) per year from all sources:

> "These dose conversion factors are appropriate for application to any population adequately characterized by the set of values for physiological parameters developed by the [International Committee on Radiological Protection] and collectively known as "Reference Man." The actual dose to a particular individual from a given intake is dependent upon age and sex, as well as other characteristics. As noted earlier, implementing limits for the general public expressed as age and sex dependent would be difficult .... More importantly, the variability in dose due to these factors is comparable in magnitude to the uncertainty in our estimates of the risks which provide the basis for our choice of the [public dose limit]. For this reason EPA believes that, for the purpose of providing radiation protection under the conditions addressed by these recommendations, the assumptions exemplified by Reference Man adequately characterize the general public, and a detailed consideration of age and sex is not generally necessary." [sic]

Since age-based dose conversion factors are not being used, the same dose conversion factors are applied to all individuals. Only in rare scenarios will a non-adult individual receive a higher dose (i.e., intake more radioactive material) than an adult individual in a similar exposure scenario.

By integrating the exposure scenario, source term, and knowledge about the applicable environmental transport routes involved in the exposure pathways, a conceptual model of the features and processes at the site can be created. The conceptual model is a qualitative description of the important environmental transport and exposure pathways and their interrelationships. Using this description, a mathematical model quantifying it, or using an off-the-shelf computer code that implements the same (or similar) conceptual model, needs to be identified. Generally, a single mathematical model can be used for several different conceptual models by varying either the boundary conditions or the various parameters.

Going from a conceptual model to a mathematical model involves a number of assumptions and simplifications. For example, one part of a conceptual model of surface soil residual radioactivity involves the leaching of radionuclides through the soil and into the aquifer. In reality, the soil between the surface and the aquifer is usually formed by numerous layers of different types of soils with varying thickness across a site. For the purposes of dose modeling, the conceptual model is more focused on knowing how much activity is entering (and leaving) each major environmental compartment (such as the aquifer) than to precisely predict the level of activity in the intervening material (e.g., any single soil layer between the surface and the aquifer). Therefore, the mathematical model may view the intervening soil layers as one layer or just a few layers, depending on the difficulty of justifying effective parameters that will mimic the real behavior. Users of off-the-shelf codes should be aware of and consider the appropriateness of the assumptions made in the computer model they are using.

Selection of parameter values (or ranges) for features, events, and processes depends not only on the site conditions and the exposure scenario, but also on the computer code (or mathematical model) being used. Nearly any data set will need to be transformed into one appropriate to the situation. This can be as straightforward as generating a site-wide effective soil density value or as complex as converting resuspension factor data into resuspension rates. The NRC staff has already factored these issues in the data used in the screening analyses, but licensees using site-specific information should justify their values.

3.5.1 Site-Release Criteria

The site-release criteria for the Haddam Neck site correspond to the 10 CFR 20.1402 criteria for unrestricted use. The residual radioactivity, including that from ground-water sources, that is distinguishable from background, must not cause the total effective dose equivalent (TEDE) to an average member of the critical group to exceed 0.25 mSv/yr (25 mrem/yr). The residual radioactivity must also be reduced to levels that are ALARA.

The compliance approach being used for the Haddam Neck site relies on generating derived values of what concentration would be equivalent to 0.25 mSv/yr (25 mrem/yr). To calculate what concentrations of residual radioactivity could be left on site, doses to a hypothetical person are calculated. As discussed in 2.5.2, the LTP calculates the dose to a hypothetical adult who is either an industrial worker in the still-standing buildings or a farmer who moves onto the site immediately after license termination and grows crops and raises animals for the food supply for his family. This hypothetical person is meant to represent the average member of the critical group. It should also be noted that the time of greatest exposure, based on the radionuclides involved at the Haddam Neck site, would be, in most cases, immediately after license termination and the overall probability of a residential farmer using the site in the short-term after license termination is small. In addition, the DCGL approach assumes that the entire site is right at the concentration that results in 0.25 mSv/yr (25 mrem/yr) to the average member of the critical group. Because of both the conservative approach in selecting the exposure scenario and the conservative assumption that the entire site is at the calculated DCGL, this compliance approach provides the NRC staff with reasonable assurance that it is highly unlikely that anyone will actually receive a dose approaching 0.25 mSv/yr (25 mrem/yr).

The LTP establishes DCGLs on the basis of a dose criterion of 0.25 mSv/yr (25 mrem/yr) to an average member of the critical population group for three environmental media: soil, ground water, and concrete (demolished and standing building). The LTP discusses how the survey-unit-specific $DCGLs_{op}$ will be implemented on the basis of the soil, ground water, and concrete DCGLs, by considering contributions from all applicable pathways.

Buried pipe in contact with the saturated zone is expected to remain at the site following decommissioning. The total length of this buried pipe has been estimated to be less than 305 meters (1,000 feet). To ensure the potential dose contribution from this buried pipe will be inconsequential, the licensee has proposed to apply a lower release limit than that which will be applied to other media with residual radioactivity. For buried piping in contact with the saturated zone, the licensee determined surface activity levels corresponding to a release limit of 10 µSv/yr (1 mrem/yr). The NRC staff considers use of the 10 µSv/yr (1 mrem/yr) release limit as an appropriate level for excluding the dose contribution for this piping, as this is well within the expected uncertainty of the analysis used to develop the DCGLs. Table 3.5.1 provides the surface concentrations resulting in 10 µSv/yr (1 mrem/yr) dose. The method used to derive these surface concentrations is provided in Section 5.4.7.5 of the LTP. The approach calculates volumetric concentrations for each radionuclide resulting in a dose of 10 µSv/yr (1 mrem/yr) by considering all water-dependent pathway doses obtained from the concrete debris scenario, and converts the volumetric concentrations for concrete debris to surface concentrations for this buried piping by using the most conservative pipe diameter at the site [i.e.,10.2 cm (4 inch) diameter]. The evaluation of the dose calculation for the concrete debris scenario is discussed in Section 3.5.5 of this report. Because the pathways for potential radiation exposure from concrete debris and buried piping are the same in the saturated zone, and the conversions from volumetric concentrations for concrete debris to surface concentrations for buried piping are conservative, the proposed method is judged to be acceptable. If concentration limits in Table 3.5.1 are exceeded, the buried piping will be either dug up and removed or decontaminated and left in place.

Table 3.5.1 Concentration Limits for Buried Piping

Radionuclide	Surface Concentration Limits Equivalent to 1 mrem/yr (dpm/100 cm^2)	Radionuclide	Surface Concentration Limits Equivalent to 1 mrem/yr (dpm/100 cm^2)
H-3	5.21E+03	Cs-134	8.35E+04
C-14	7.77E+04	Cs-137	9.66E+04
Mn-54	5.31E+04	Eu-152	2.68E+05
Fe-55	6.17E+04	Eu-154	1.87E+05
Co-60	3.21E+05	Eu-155	1.20E+06
Ni-63	1.52E+05	Pu-238	7.50E+02
Sr-90	1.82E+02	Pu-239/240	6.82E+02
Nb-94	1.37E+05	Pu-241	1.14E+04
Tc-99	2.44E+04	Am-241	3.33E+02
Ag-108m	1.37E+06	Cm-242/243	4.61E+02

Piping embedded in concrete that runs through structures may also remain following decommissioning. The licensee plans to apply building-surface DCGLs in assessing the need to remediate this piping. Because the expected exposure would be different in general, building-surface DCGLs are not applicable for use with piping. The licensee has provided an analysis that shows the application of building-surface DCGLs will be bounding for the Haddam Neck site as long as the gross activity beta-to-alpha ratio is ≥15:1. If necessary, the licensee will use scaling factors to establish gross activity levels via radionuclide-specific measurements or assessments. The licensee has committed to remove piping that are greater than 61 cm (24 inches) in diameter and are found to have a gross activity beta-to-alpha ratio < 15:1. The NRC staff has determined that the licensee's analysis appropriately demonstrates that use of the building-surface DCGLs for the embedded piping will adequately ensure that the 0.25 mSv/yr (25 mrem/yr) dose limit will not be exceeded. The majority of the piping expected to be left at the site will be inaccessible because of their limited diameters or locations within the building; therefore, use of DCGLs, based on an assumption of someone spending a significant amount of time exposed to the radioactivity (as assumed for the building-surface DCGLs), is conservative. Further, the licensee has committed to only leaving piping that is extremely difficult to separate from the concrete; therefore, removal of the piping from the concrete is unlikely because of the expense and effort that would be involved. NRC through the inspection process will review: (1) any scaling factors used for acceptability and (2) verify that the interior surface of the piping/penetration was surveyed.

The approach of including doses from all possible exposure pathways resulting from residual radioactivity in soil, ground water, and building surfaces toward compliance with the 0.25 mSv/yr (25-mrem/yr) dose criterion is determined to be conservative and, therefore, acceptable.

3.5.2 Dose Modeling Scenarios

The LTP considers two primary scenarios for developing DCGL values: a residential farming scenario for considering contamination in soil, ground water, and concrete debris from demolished buildings, and a building occupancy scenario for considering contamination in building structures. For both scenarios, the NRC staff considers the licensee's use of a hypothetical adult as adequately representing the average member of the critical group at the site. While children may be considered members of the critical group, for example, children of the residential farmer, their overall doses are likely to be lower than the total dose received by an adult. Therefore, the use of the adult as the average member of the critical group provides reasonable assurance that any actual doses will be less than the unrestricted release limit.

As discussed above, assessments of residual radioactivity tend to use adults as the average member of the critical group. This is because the scenarios have combined a number of pathways, or methods, such as breathing, eating food, drinking water, and spending time outdoors, through which a hypothetical adult tends to receive the highest doses. Only in rare scenarios will a hypothetical infant or child receive a significantly higher calculated dose than an adult in a similar exposure scenario. In general, these scenarios tend to be ones where someone could only get exposed to the residual radioactivity through a much more limited set of pathways. One example is when the only pathway is through milk, since children generally drink more milk than adults. If milk was the only pathway that could expose the individual to a dose, then the child would be a better representation of the average member of the critical group. But in most situations, especially ones involving multiple pathways and multiple radionuclides, the total dose of the adult is greater than or similar to that of a child. Therefore, for most multiple pathway scenarios, such as the scenarios used in the LTP, the average member of the critical group should usually be assumed to be an adult, with adult habits and characteristics.

During the review of the LTP, the NRC staff investigated the results where a hypothetical infant or child was used in the scenario to verify that doses to infants and children were consistent with those to the assumed adult average member of the critical group. The primary residual radioactivity (e.g., Co-60 and Cs-137) at the Haddam Neck site results in external radiation being the most important route of exposure. To see if more detailed calculations would be necessary, the NRC staff maintained the same exposure time as that used for the adult. This assumption would skew the outcome since the exposure time for an adult would normally be significantly greater than that for a child. The results of the screening approach by the NRC staff found that the results for infant or child were similar to the more detailed modeling done for the adult. Because the results of a more detailed calculation would be expected to be lower, since the exposure time for a child would be adjusted downward, the NRC staff did not calculate further the dose to an infant or a child.

The NRC staff analysis primarily used default probabilistic parameter values in RESRAD version 6.2. The NRC staff modified the dose conversion factors to use the appropriate age-specific dose conversion factors from International Commission on Radiological Protection (ICRP) ICRP-72 (ICRP, 1995) to calculate internal dose to an infant or child in the comparison discussed above. The NRC staff modified the amount of soil ingestion, breathing rate, and food intake to correspond to an infant and child, based on the age-specific information in NCRP Report No. 129 (NCRP, 1999). In addition, the NRC staff increased the external dose calculated by RESRAD by 30 percent for an infant and 20 percent for a child, as suggested by NCRP Report No. 129, as a conservative estimate of the effect of properly accounting for the effective height of the infant or child. Because the external exposure is the most important pathway, a more detailed and realistic analysis that would likely reduce the assumed outdoor exposure time for an infant or child would result in lower total doses than the screening analysis. Therefore, the use of the adult as the average member of the critical group provides reasonable assurance that any actual doses will be less than the unrestricted release limit.

3.5.2.1 Scenario for Soil DCGLs

For soil DCGLs, a resident farmer was considered to represent the average member of the critical group. This selection was based on demographic and economic data for the towns of Haddam and East Haddam and Middlesex County, Connecticut. Those data demonstrate that agriculture is a feasible lifestyle and means of employment in the area and will likely continue to be for the foreseeable future. In addition, the flat topography of the industrial area at the Haddam Neck site, which is adjacent to the Connecticut River, would be conducive to agricultural activities. Although in the near future the site is most likely to be used by industries to take advantage of the on-site building infrastructure, the resident farmer scenario is more reflective of the demographic and economic environments surrounding the site. Therefore, the choice of the resident farmer scenario is reasonable and is consistent with the guidance of NUREG-1727, NUREG/CR-5512, and NUREG-1549. Because the exposure pathways associated with the resident farmer scenario cover all the likely routes of exposures, it would also result in more restrictive DCGLs than other scenarios. Therefore, the resident farmer scenario is considered acceptable for developing soil DCGLs.

3.5.2.2 Scenario for Ground-Water DCGLs

The scenario considered for deriving the ground-water DCGLs is also the resident farmer scenario. For the same reasons discussed under “Scenario for Soil DCGLs,” above, the resident farmer scenario is reasonable and acceptable for deriving ground-water DCGLs.

3.5.2.3 Scenario for Concrete DCGLs

Potential radiation exposures after unrestricted release of the Haddam Neck site could result from reusing the buildings left standing or living close to the buried concrete debris from demolished buildings. DCGLs for contaminated buildings, standing or demolished, needed to be established. The licensee considered two scenarios for determining DCGLs for concrete. For the case of on-site buildings left standing, the industrial building occupancy scenario was considered; for the case of on-site buildings demolished and the debris used as backfill for a basement, the resident farmer scenario was considered. The more restrictive DCGL derived for each radionuclide was selected as the final DCGL for concrete, which would be used as the guideline level for decontamination activities.

3.5.2.3.1 Standing Buildings

An industrial building occupancy scenario was selected to develop DCGLs for standing buildings. This scenario considers potential radiation exposures to an industrial worker working in a standing building. Considering the internal designs of the buildings that could possibly be left standing, reusing the buildings as residential houses or converting them to apartment complexes is unlikely. Therefore, in terms of reusing the standing buildings, the industrial occupancy scenario is reasonable and likely.

3.5.2.3.2 Demolished Buildings

A scenario involving exposure to concrete debris was developed to reflect the fact that some buildings may be demolished and the concrete debris used as backfill material and left onsite. The concrete debris would be covered by 0.9 meter (3 feet) of soil to maintain a flat surface. In the LTP, a resident farmer was assumed to establish living quarters above the concrete debris. This scenario results in the most restrictive DCGLs and is consistent with the demographic and economic environments surrounding the site; therefore, it is determined to be reasonable and acceptable for determining DCGLs.

3.5.3 DCGLs for Soil

DCGLs for soil were developed for the 22 radionuclides listed in Table 3.5.2. The radionuclides were identified by previous site characterization data and with reference to the

Table 3.5.2, Radionuclides for Which DCGL Values Were Derived for Soil, Ground Water, and Concrete Media

H-3	Ni-63	Cs-134	Pu-238
C-14	Sr-90	Cs-137	Pu-239/240
Mn-54	Nb-94	Eu-152	Pu-241
Fe-55	Tc-99	Eu-154	Am-241
Co-60	Ag-108m	Eu-155	Cm-243/244

NUREG/CR-0130 and NUREG/CR-3474 reports for potential dose-significant radionuclides. The same suite of radionuclides was used to develop ground water and concrete DCGLs.

Table 3.5.3 lists the DCGL values for soil presented in the LTP. Single DCGL values were established for two separate pairs of radionuclides, Pu-239/240Pu and Cm-243/244, primarily because radiochemical analyses do not report concentration of these radionuclides separately. The licensee reported DCGL value for Pu-239 from the Pu-239/240 pair and Cm-243 from the Cm-243/244 pair. Choosing Pu-239 from the Pu-239/240 pair and for Cm-243 from the

Cm-243/244 pair is acceptable because dose conversion factors (external exposure, inhalation, and ingestion) for Pu-239 and Cm-243 are more conservative than for Pu-240 and Cm-244, respectively. Although only a single DCGL value has been provided for each radionuclide pair, both radionuclides will be assumed to be present when determining compliance with the dose limit. The evaluation of the DCGLs is discussed in the following sections.

Table 3.5.3 DCGL Values for Soil

Radionuclide	DCGL (pCi/g)	Radionuclide	DCGL (pCi/g)
H-3	4.1×10^{2}	Cs-134	4.7×10^{0}
C-14	5.7×10^{0}	Cs-137	7.9×10^{0}
Mn-54	1.7×10^{1}	Eu-152	1.0×10^{1}
Fe-55	2.7×10^{4}	Eu-154	9.3×10^{0}
Co-60	3.8×10^{0}	Eu-155	3.9×10^{2}
Ni-63	7.2×10^{2}	Pu-238	3.0×10^{1}
Sr-90	1.6×10^{0}	Pu-239/240	2.7×10^{1}
Nb-94	7.1×10^{0}	Pu-241	8.7×10^{2}
Tc-99	1.3×10^{1}	Am-241	2.6×10^{1}
Ag-108m	7.1×10^{0}	Cm-243/244	2.9×10^{1}

3.5.3.1 Contaminant Characteristics

The soil DCGLs were determined for those radionuclides found contributing to site contamination. Contaminant characteristics were determined by an initial site characterization effort. As summarized in Section 2 of the LTP, initial site characterization at Haddam Neck included a historical site assessment and a characterization report that documented site conditions in the latter half of 1999. Potential sources of contamination included both routine and accidental airborne and liquid releases.

Routine airborne releases were stated not to result in site contamination (Section 2.2.4.1 of the LTP). Routine liquid releases were monitored, but their effect on site conditions was not mentioned. Several events led to unplanned airborne releases. Except for a particulate release in 1979, these unplanned events involved inert gases and short-lived radionuclides that were considered not to result in site contamination (Section 2.2.4.2.1 of the LTP). A number of spills through the storm drain system, leaks, and unplanned liquid releases have led to soil and ground-water contamination (Section 2.2.4.2.2 of the LTP). In addition, small amounts of contamination have been detected in the nonradiologically controlled-area grounds (Section 2.3.3.1.4 of the LTP).

Table 3.5.2 lists 22 radionuclides for which DCGLs were developed. The NRC staff reviewed the information provided by CYAPCO to ensure that the list is complete. The radionuclides listed in Table 3.5.2 include the significant radionuclides that could contribute to residual doses from Haddam Neck. In identifying radionuclides to be considered for developing DCGL values, the licensee used the available waste characterization data. The list was compared with the regulatory guidance for radionuclides of concern in bioshield wall concrete, rebar, and surface contamination provided in NUREG/CR-3474 ("Long-Lived Activation Products in Reactor Material," Tables 5.4 and 5.6), and NUREG/CR-0130 ("Technology, Safety and Cost of Decommissioning," Tables 7.3-5, 7.3-11, and 7.3-14).

Sixty-five radionuclides, with the exception of noble gases, are listed in the above-mentioned NUREGs. The licensee has demonstrated that the dose contribution would be insignificant for a number of those radionuclides. Table 3.5.4 lists radionuclides with insignificant dose contributions on the basis of relative abundance and dose potential. In that list, Y-90 is the short-lived daughter of a longer-lived parent, Sr-90, and is included in the LTP as an associated

Table 3.5.4 NUREG Radionuclides Contributing Insignificant Doses

Na-22	Co-57	Zr-93	I-129	Ce-144
P-33	Co-58	Nb-94	Te-129m	Pm-145
S-35	Ni-59	Zr-95	I-131	Sm-146
Cl-36	Ni-63	Nb-95	I-133	Sm-151
Ca-41	Zn-65	Tc-99	Ba-133	Tb-158
Ca-45	Se-79	Ru-103	Cs-135	Ho-166m
Sc-46	Sr-89	Ag-108m	Cs-136	Hf-178m
Cr-51	Y-90	Ag-110m	Ba-140	Pb-205
Mn-53	Mo-93	Sn-121m	La-140	U-233
Mn-54	Nb-92m	Sb-124	Ce-141	
Fe-59	Nb-93m	Sb-125	Ce-143	

radionuclide of Sr-90. Although dose contributions from C-14, Mn-54, Ni-63, Nb-94, Tc-99, Ag-108m, and Eu-155 were found to be small, they are included in the LTP for developing DCGLs because these radionuclides have been identified in waste streams at other sites.

In accordance with guidance in Appendix E of NUREG-1727, radionuclides contributing less than 10 percent of the dose limit can be screened out. The NRC staff's own independent assessment concluded that the percentage used by the licensee to screen out radionuclides was more conservative than that recommended as guidance in NUREG-1727; therefore, the licensee's basis for screening out radionuclides is acceptable, and, moreover, these radionuclides were not detected in any of the waste streams at the Haddam Neck site.

3.5.3.2 Scenario Definition and Exposure Pathways

A residential farming scenario was considered to develop soil DCGLs. Potential exposure pathways considered for the resident farmer included direct external exposure from contaminated soil, internal exposure from inhalation of airborne radionuclides, and internal exposure from ingestion of: (1) plant foods grown in the contaminated soil and irrigated with contaminated water; (2) meat and milk from livestock fed with contaminated fodder and water; (3) drinking water from a contaminated well; (4) fish from a contaminated pond; and (5) contaminated soil. These pathways reflect a subsistence farming practice and are feasible considering the physical, geological, and hydrogeologic characteristics of the Haddam Neck site; therefore, they are determined to be applicable for the Haddam Neck site.

3.5.3.3 Application of RESRAD for Dose Modeling

The residential farming scenario considers potential radiation exposure from residual soil contamination after unrestricted release of the Haddam Neck site. The residual radionuclides have the potential of running off from the contaminated area, leaching to ground water, being taken up by plant roots, and becoming suspended in the air. The RESRAD computer code was used to model the potential radiation exposure, and soil concentration corresponding to the dose limit of 0.25 mSv/yr (25 mrem/yr) was derived for each radionuclide.

3.5.3.3.1 Selection of Input Parameter Values

The licensee used the process outlined in Figure 6-5 of the LTP to choose conservative values for input parameters that have great influence on radiation dose results. The selection process, which consists of five steps, was developed in accordance with the approach presented in NUREG/CR-6755, NUREG/CR-6676, NUREG/CR-6692, and NUREG/CR-6697.

In the first step of the selection process, the RESRAD input parameters were classified as one of the following three types: behavioral, metabolic, or physical, which is consistent with NUREG/CR-6697. 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 parameters related to specific site attributes and thus are not dependent on the assumed receptor group.

In the second step of the selection process, parameters were prioritized in order of their respective importance in dose modeling, according to NUREG/CR-6697. Four attributes were considered in determining the priority of a parameter: (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. The parameters that have a large influence on dose results and are site-specific (i.e., physical) were assigned a higher priority than parameters that have small influence on dose results and/or are behavioral or metabolic parameters. Three levels of priority (1, 2, and 3) were used.

According to the priority, parameter type, availability of site-specific data, and relevance in dose calculation, a parameter was treated as either deterministic or probabilistic in the third step. Deterministic parameters were assigned fixed values without further analysis. The values for probabilistic parameters were determined by their correlation with the resulting dose.

Probabilistic sensitivity analysis that was incorporated into the RESRAD code (version 6.1) was used to study the correlation.

Step four of the selection process involved using RESRAD to conduct a sensitivity analysis for the probabilistic parameters. In the sensitivity analysis, each probabilistic parameter was assigned a generic distribution obtained from NUREG/CR-6697, Attachment C, whereas the deterministic parameters were assigned fixed values that were site-specific, recommended by NRC, or RESRAD defaults. The partial rank correlation coefficient (PRCC) reported by RESRAD was used as the index to characterize a probabilistic parameter's sensitivity. If the absolute value of PRCC was greater than 0.25, the parameter was characterized as sensitive; otherwise, the parameter was characterized as insensitive. The threshold value of 0.25 was consistent with the approach in NUREG/CR-6676.

After the sensitivity analysis was conducted, the last step of the selection process was to assign values to the input parameters. Behavioral and metabolic parameters were assigned values from NUREG/CR-5512, NUREG/CR-6697, or the RESRAD default library. Physical parameters were assigned values according to the following rules:

1. Parameters for which site-specific data were available were assigned site-specific values.

2. Priority 1 and 2 parameters shown to be sensitive (with absolute PRCC values greater than 0.25) were assigned conservative values, either 75th (positive correlation) or 25th (negative correlation) quantile value of its generic distribution. If the 75th quantile value is less than the mean value of the distribution, the mean value was assigned to the parameter.

3. Priorities 1 and 2 parameters shown to be insensitive (absolute values of PRCC less than 0.25) were assigned median value from their generic distributions.

4. Priorities 1 and 2 parameters shown to be insensitive, but correlated with parameters shown to be sensitive, were assigned values based on the conservative values assigned to the sensitive parameters.

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

Parameter values resulting from the selection process were listed in Table F-1 of the LTP. The selection process takes into account the site-specific physical environment, sensitivities of parameters, and a receptor's behavioral pattern and metabolic conditions. The process used is consistent with NRC guidance and is expected to result in derivation of conservative DCGLs.

The values for behavioral and metabolic parameters are primarily from NUREG/CR-5512. When values were not available from NUREG/CR-5512, RESRAD default values were used. Using NUREG/CR-5512 values is consistent with the guidance in NUREG-1727 and using the RESRAD default values is expected to result in conservative dose estimates.

The contaminated area of 15,600 square meters was based on the largest Class A subsurface survey area and is determined to be reasonable and acceptable because: (1) it is a site-specific value; (2) dose results are generally not sensitive to the size of the contaminated area when the area is large; and (3) dose results from the meat and milk pathways would not be affected by the choice of the contaminated area because it is assumed that 100 percent of the consumed meat and milk are contaminated. The depth of the contaminated area was assigned a uniform distribution (minimum value 0.15 m and maximum value = 3.0 m). For radionuclides found to be insensitive to depth of the contaminated area, the value was set at the median of the distribution (1.575 m), and for radionuclides found to be sensitive to depth of the contaminated area the value was set at the 75th quantile of its distribution (2.29 m). In reality, the preliminary site characterization data showed that most of the soil contamination is limited to the surface layer. Therefore, the depth of the contaminated area is set at a conservative value. According to the "Groundwater Monitoring Report," the overburden lithology within the industrial area can be generally described as a silty (loamy) sand. Therefore, values for soil density, total porosity, effective porosity, hydraulic conductivity, and soil-specific exponential parameters were obtained from literature for this type of soil. The use of site-specific data is encouraged and acceptable. Other parameters with site-specific values include humidity, evapotranspiration coefficient, wind speed, precipitation, runoff coefficient, watershed area, saturated zone hydraulic gradient, and number of unsaturated zones. The remaining parameters were defined as probabilistic parameters, and their values were determined by using Steps 4 and 5 of the proposed selection process. Because the selection process is determined to be conservative and acceptable, parameter values determined using the process are also conservative and acceptable.

3.5.3.3.2 Sensitivity Analysis

Probabilistic sensitivity analysis was used to study the influence of the input parameters on dose results, identify the important parameters, and assign parameter values, as discussed in the parameter selection process. The parameter distributions used in the sensitivity analysis are the generic distributions from NUREG/CR-6697; the actual distribution is likely to be narrower because the generic distributions are based on national data. Determination of sensitive parameters was based on the values of the PRCC calculated by the RESRAD code for each individual parameter. The PRCC is a gauge of the correlation between the peak radiation dose and the parameter value. The larger the absolute value of the PRCC, the greater the influence of the parameter value on the peak dose. When the PRCC value is positive, the peak dose increases with an increased parameter value. When the PRCC value is negative, the peak dose decreases when the parameter value is increased. On the basis of previous studies (NUREG/CR-6755, NUREG/CR-6697, NUREG/CR-6692, and NUREG/CR-6697) on uncertainty analysis of the RESRAD code, the NRC staff concluded that the PRCC is the most representative among several coefficients of correlation between the peak dose and the parameter value.

The use of 75th or 25th quantile values for sensitive parameters in deterministic calculations would most likely generate conservative dose values, i.e., the peak dose would most likely be greater than the 75th quantile value of the corresponding peak dose distribution obtained from probabilistic calculations. The use of the mean value to replace the 75th quantile value for some sensitive parameters in case the former is greater than the latter provides another layer of conservatism. Therefore, it is determined that the sensitivity analyses conducted for the LTP were comprehensive and the method used to select parameter values is acceptable and will result in conservative DCGLs.

3.5.3.4 Comparison of the Peak Dose Results

The DCGL value for each radionuclide was derived on the basis of the peak dose of that radionuclide obtained from deterministic calculation. As mentioned in the previous section, it is most likely that the peak dose used to derive the DCGL value would be greater than the 75th quantile value of the corresponding distribution obtained from probabilistic calculations. The NRC staff has performed independent probabilistic calculations using the same parameter assignments as used in the licensee's sensitivity analysis and has confirmed the above expectation.

The NRC staff, based on its evaluation, determined that the soil DCGLs are conservative and meet the requirement of limiting radiation dose to an average member of the critical group to less than 0.25 mSv/yr (25 mrem/yr).

3.5.4 DCGLs for Ground Water

Ground-water contamination was identified at the Haddam Neck site, and, according to the LTP, the affected areas are generally confined to the industrial area of the site. To derive DCGLs for ground water, RESRAD's feature of accepting input ground-water concentrations was used. This feature allows the calculation of radiation doses associated with the use of ground water. Table 3.5.5 lists the derived ground-water DCGLs based on the dose limit of 0.25 mSv/yr (25 mrem/yr) for individual radionuclides. Evaluation of the DCGLs is discussed in the following sections.

Table 3.5.5 DCGL Values for Ground Water

Radionuclide	DCGL (pCi/L)	Radionuclide	DCGL (pCi/L)
H-3	6.5×10^5	Cs-134	3.4×10^2
C-14	9.0×10^3	Cs-137	4.3×10^2
Mn-54	2.4×10^4	Eu-152	7.3×10^3
Fe-55	6.5×10^4	Eu-154	5.1×10^3
Co-60	1.1×10^3	Eu-155	3.3×10^4
Ni-63	3.2×10^4	Pu-238	1.5×10^1
Sr-90	2.5×10^2	Pu-239/240	1.4×10^1
Nb-94	6.8×10^3	Pu-241	4.6×10^2
Tc-99	2.6×10^4	Am-241	1.3×10^1
Ag-108m	4.2×10^3	Cm-243/244	1.9×10^1

3.5.4.1 Contaminant Characteristics

A total of 22 radionuclides were identified as relevant to the decontamination activities at the Haddam Neck site. The selection of these radionuclides was judged to be acceptable, and the reasons were discussed in detail in Section 3.5.3.1. In reality, site characterization conducted so far has found only the concentrations of H-3, Sr-90, Tc-99, and Cs-137 to be above the reporting limits. Nevertheless, DCGL values have been calculated for all 22 radionuclides. If additional radionuclides are detected as part of the "Phase 2 Hydrogeologic Work Plan" (described in Section 2.3.3.1.6 of the LTP), the $DCGLs_{op}$ as described in Section 5.4.7.1 of the LTP will have to consider these radionuclides.

3.5.4.2 Scenario Definition and Exposure Pathways

A residential farming scenario was used to derive the ground-water DCGLs. The resident farmer was assumed to withdraw ground water contaminated with residual radionuclides and use it for irrigation and drinking. The water was also assumed to be used to raise livestock later slaughtered and consumed. Because the flow rate of ground water is very small, relative to the flow rate of the Connecticut River, to which the ground-water plume is migrating, potential concentrations of radionuclides in the river would be very small; therefore, the exposure pathway of ingesting contaminated aquatic foods was not evaluated. The exposure pathways selected in RESRAD dose modeling were: (1) ingestion of plant foods irrigated with ground water; (2) ingestion of meat and milk from livestock watered with ground water and fed fodder irrigated with ground water; and (3) ingestion of ground water. The pathways selected for dose modeling are judged to be acceptable.

3.5.4.3 Application of RESRAD for Dose Modeling

The RESRAD code is typically used to analyze potential radiation exposures resulting from a soil source that lies above the ground-water table. To use it to analyze the radiation dose resulting from existing ground-water contamination, a hypothetical soil source has to be established, and a non-zero material placement time and initial ground-water concentration have to be input. Under such circumstances, the dose calculated for the water dependent pathways at time zero would result entirely from contamination in ground water and would correspond to a ground-water concentration calculated for time zero.

3.5.4.3.1 Input Parameter Selection

To appropriately consider ingrowth and decay of radionuclides when developing DCGLs for ground water using the RESRAD computer code, it was necessary to adjust some of the parameters. These adjustments also ensured that radiation doses resulted entirely from contaminated ground water. The water-independent dose from plant, meat, and milk ingestion pathways was suppressed. To achieve this, the depth of roots, livestock intake of soil, and mass loading for foliar deposition were set to zero. Moreover, the LTP specifically set: (1) the time since placement of material to 1 year; (2) the time for dose calculations to 1 year; (3) a non-zero initial ground-water concentration; and (4) the number of unsaturated zones to zero. In addition, the Mass Balance (MB) model was selected as the ground-water transport model. Settings 1 and 3 cause the code to derive appropriate distribution coefficients for the contaminated and saturated zones so that a non-zero ground-water concentration would be calculated for time 0 and would match the input value. The default ground-water model in RESRAD is the nondispersion (ND) model that assumes the hypothetical well is located at the down-gradient edge of the contaminated zone. With the ND model, not all contaminants reaching the water table will be drawn into the hypothetical well, as this depends on the assumed configuration of the contaminant zone and the well pumpage rate. However, with the MB model the hypothetical well is assumed to be located at the center of contamination and all contaminants reaching the water table are assumed to be drawn into the well. Thus, the MB model was used to eliminate migration of radionuclides in the saturated zone (i.e., ground water). Setting the time for dose calculation to 1 year (setting 2) would result in the occurrence of the peak dose within 1 year, thereby suppressing any influence on peak dose from the hypothetical soil source.

Besides the special settings, the values of the input parameters related to the calculation of dose for the water-dependent pathways were selected by using the same selection process discussed in Section 3.5.3.3.1. These parameters include some dietary and nondietary parameters related to the ingestion pathway and the plant, meat, and milk transfer factors. This selection process was determined to be consistent with NRC guidance, and the parameter values selected were expected to result in conservative dose results. Detailed discussions on the evaluation of the selection process are provided in Section 3.5.3.3.1. In conclusion, the input parameter values selected for dose calculation are determined to be adequate and acceptable.

3.5.4.3.2 Sensitivity Analysis

Sensitivity analysis was incorporated into the selection process for input parameter values. The use of the PRCC to identify sensitive parameters and the assignment of 75th quantile, 25 percent quantile, or mean values for sensitive parameters, was considered to be a conservative approach and acceptable. Detailed discussions are provided in Section 3.3.3.3.2.

3.5.4.3.3 Ingrowth and Decay Consideration

The DCGLs are guidelines that provide sufficient protection of human health so that potential radiation exposure resulting from the use of contaminated ground water would not exceed 0.25 mSv/yr (25 mrem/yr) for an average member of the critical population group within 1,000 years. The RESRAD dose calculations discussed so far are limited to a time frame of 1 year. To extend the time frame to 1,000 years, the dose results need to be modified by considering the influence of radioactive ingrowth and decay.

The LTP accomplished this modification by using the effective dose conversion factor, which is the peak dose of each individual radionuclide, based on a ground-water concentration of 37 mBq/liter (1 pCi/liter). The effective dose conversion factor can be calculated by scaling the peak dose previously calculated with the ground-water concentration at time 0. The effective dose conversion factors were used to replace the default ingestion dose conversion factors for all radionuclides of concern, and another run of RESRAD was performed. Under these conditions, the peak dose calculated for each radionuclide was the modified effective dose conversion factor accounting for the influence of ingrowth and decay within 1,000 years. The peak dose obtained was then scaled to find the initial ground-water concentration, for each radionuclide, that would result in a peak dose of 0.25 mSv/yr (25 mrem/yr). The ground-water concentrations obtained were the derived DCGLs.

The approach that was used for considering the influence from ingrowth and decay is determined to be appropriate. Therefore, the resulting DCGLs are determined to be conservative and acceptable.

3.5.4.4 Comparison of the Peak Dose Results

The peak dose for an individual radionuclide obtained by using the selected parameter values from a deterministic analysis is most likely to be greater than the 75th quantile value of the corresponding peak dose distribution from a probabilistic analysis. Independent probabilistic analyses have confirmed this expectation. This supports the conclusion that the proposed ground-water DCGL values are conservative.

3.5.5 DCGLs for Concrete

For deriving DCGLs for concrete, the LTP considers a building occupancy scenario as well as a residential farming scenario because once the contaminated buildings are released, CYAPCO may choose to demolish the structures and bury the concrete debris onsite. The two scenarios were evaluated separately, and two sets of DCGLs were developed. The more restrictive DCGL for each radionuclide will be adopted at the time of the final status survey. For the building occupancy scenario, depending on the extent of contamination, radiation exposures could result from surface sources (i.e., contamination limited to the surface of concrete) -- or volumetric sources (i.e., contamination extends beyond the surface of concrete). Table 3.5.6 lists the DCGLs developed for the building occupancy scenario.

Table 3.5.6 DCGLs for Building Occupancy Scenario

Radionuclide	DCGLs for Surface Sources (dpm/100 cm^2)	DCGLs for Volumetric Sources (pCi/g)	Radionuclide	DCGLs for Surface Sources (dpm/100 cm^2)	DCGLs for Volumetric Sources (pCi/g)
H-3	3.2×10^8	1.5×10^3	Cs-134	1.7×10^4	4.9×10^0
C-14	1.0×10^7	1.2×10^8	Cs-137	4.3×10^4	1.4×10^1
Mn-54	3.2×10^4	9.1×10^0	Eu-152	2.3×10^4	6.7×10^0
Fe-55	3.5×10^7	9.5×10^7	Eu-154	2.2×10^4	6.1×10^0
Co-60	1.1×10^4	2.9×10^0	Eu-155	$4.4 x 10^5$	3.2×10^2
Ni-63	3.6×10^7	4.1×10^7	Pu-238	4.9×10^3	6.6×10^2
Sr-90	1.3×10^5	2.4×10^3	Pu-239/240	4.4×10^3	6.0×10^2
Nb-94	1.7×10^4	4.8×10^0	Pu-241	2.3×10^5	3.1×10^4
Tc-99	1.5×10^7	3.1×10^7	Am-241	4.3×10^3	4.2×10^2
Ag-108m	1.7×10^4	4.8×10^0	Cm-243/244	6.1×10^3	7.5×10^1

Table 3.5.7 lists the DCGLs for the residential farming scenario involving buried concrete debris. The DCGLs are evaluated in Section 3.5.5.2 for the building occupancy scenario and in Section 3.5.5.3 for the residential scenario involving concrete debris. Section 3.5.5.1 discusses contaminant characteristics that are the same for both scenarios.

Table 3.5.7 DCGLs for Residential Scenario Involving Concrete Debris

Radionuclide	DCGLs ($dpm/100\ cm^2$)	DCGLs (pCi/g)	Radionuclide	DCGLs ($dpm/100\ cm^2$)	DCGLs (pCi/g)
H-3	1.2×10^6	9.1×10^1	Cs-134	4.1×10^6	3.2×10^2
C-14	2.6×10^5	2.1×10^1	Cs-137	8.3×10^6	6.5×10^2
Mn-54	7.1×10^5	5.5×10^1	Eu-152	2.9×10^6	2.3×10^2
Fe-55	1.2×10^6	9.0×10^1	Eu-154	2.5×10^6	1.9×10^2
Co-60	1.2×10^6	9.1×10^1	Eu-155	1.2×10^8	9.5×10^3
Ni-63	1.7×10^6	1.3×10^2	Pu-238	1.5×10^5	1.1×10^1
Sr-90	4.9×10^3	3.8×10^{-1}	Pu-239/240	1.3×10^5	1.0×10^1
Nb-94	1.0×10^5	7.7×10^0	Pu-241	1.9×10^6	1.5×10^2
Tc-99	3.7×10^5	2.9×10^1	Am-241	5.7×10^4	4.4×10^0
Ag-108m	3.3×10^5	2.6×10^1	Cm-243/244	4.9×10^4	3.8×10^0

3.5.5.1 Contaminant Characteristics

DCGLs were developed for the same 22 radionuclides discussed in Section 3.5.3.1. The selection of radionuclides was based on the previous site characterization data and with reference to the NUREG/CR-0130 and NUREG/CR-3474 reports. It is determined that the list of radionuclides is complete and covers all the radionuclides that potentially occur at the site. Detailed discussions on this finding are provided in Section 3.5.3.1. For a contaminated building, it was assumed that the floor, ceiling, and four walls were all contaminated so that conservative dose results could be obtained from dose modeling. Contamination can be limited to the concrete surfaces or can extend to the entire concrete thickness.

3.5.5.2 Building Occupancy Scenario

For buildings that are left standing after the site is released, an industrial occupancy scenario was assumed, to derive the DCGLs.

3.5.5.2.1 Scenario Definition and Exposure Pathways

Because of the designs of the buildings at Haddam Neck, reuse of the buildings as residence is unlikely. The reasonable assumption is that buildings would be reused by other industries to take advantage of the existing infrastructures. As a result, the building occupancy scenario considers an adult working in the building, engaged in light industrial activities. The potential exposure pathways to residual contamination in concrete include: (1) external radiation exposure directly from the concrete source; (2) external radiation exposure to concrete material eroded away and dispersed in the air; (3) external radiation exposure to the deposition of eroded material on the floor; (4) inhalation of airborne radioactive particulate and tritium; and (5) inadvertent ingestion of radioactive material directly from the sources. The assumption of an industrial occupancy scenario is considered reasonable, and the exposure pathways considered are appropriate.

3.5.5.2.2 Application of RESRAD-BUILD for Dose Modeling

The RESRAD-BUILD computer code was used to conduct dose modeling for the building occupancy scenario. The code was designed specifically for use in such a situation. RESRAD-BUILD can be used to estimate radiation exposures from both surface and volumetric sources.

Input Parameter Selection: The input parameter values were selected with the parameter selection process evaluated in Section 3.5.3.3.1. The selection process takes into account the influence of the parameter values on the potential dose results -- it then, based on the influence of the parameters, assigns conservative values to the sensitive parameters, to ensure calculation of conservative doses. The selection process was determined to be appropriate, and the derived DCGLs based on the parameter values selected in this way would be conservative. Table F-3 in the LTP lists the final parameter values.

Potential radiation doses are very sensitive to the erosion rate of radioactive materials because the eroded materials are dispersed into the air and result in inhalation exposure. When still in the air, the dispersed radioactive materials would result in external radiation exposure through submersion; when deposited on the floor, the eroded materials form a secondary surface source and would also contribute to external radiation exposure. Although subsequent radiation exposures could result from erosion of radioactive materials, direct external radiation from the original source would decrease because of the loss of materials. Between the two opposite effects caused by erosion and with the wide range of its potential value, selecting a conservative or defensible erosion rate is difficult. The probabilistic sensitivity analysis was used in the LTP to resolve this dilemma. For surface sources, the “removable fraction” and the “lifetime for source removal” are the two input parameters that determine the erosion rate. As shown in Table F-3 of the LTP, the removable fraction was set to 0.1, in accordance with the guidance in NUREG/CR-5512; the lifetime value was varied from radionuclide to radionuclide, ranging from 18,000 to 52,000 days (corresponding to 25^{th} and 75^{th} quantile values). Although it is unrealistic to have different lifetimes, the selected value results in conservative dose results for each individual radionuclide. Therefore, the approach is determined to be acceptable. For volumetric sources, the erosion rate itself is an input parameter. A 75^{th} quantile value (2.8 x10^{-7} cm/day) was used for those radionuclides exhibiting sensitivity.

Sensitivity Analysis: The built-in capability of the RESRAD-BUILD code to conduct probabilistic sensitivity analyses was used to study the sensitivity of input parameters. The procedure used was the same as that discussed in Section 3.5.3.3.2 for the RESRAD code, except the threshold value of PRCC was changed from 0.25 to 0.1. This choice of the PRCC value to identify sensitive parameters was consistent with NUREG/CR-6676 and was determined to be acceptable. The probabilistic sensitivity analysis considered the sensitivity of a parameter within its potential range, in conjunction with the range of other parameters; therefore, it is considered to be more comprehensive and more appropriate than a deterministic sensitivity analysis when the range of the input parameter value is wide and the parameter value is quite uncertain. The choice of using 25^{th} quantile, 75^{th} quantile, or the mean value for sensitive parameters is expected to result in conservative dose results. The sensitivity analysis in the LTP is determined to be appropriate and acceptable.

3.5.5.2.3 Comparison of the Peak Dose Results

The peak dose for an individual radionuclide obtained using the selected parameter values from a deterministic analysis is most likely to be greater than the 75^{th} quantile value of the corresponding peak dose distribution from a probabilistic analysis. Independent probabilistic analyses using the same parameter distributions listed in Table D-3 of the LTP have confirmed this expectation. This finding supports the conclusion that the proposed DCGL values for concrete for the building occupancy scenario are conservative.

3.5.5.3 Residential Farming Scenario Involving Concrete Debris

A residential farming scenario was used to evaluate risks from buildings that are demolished after the site is released. This scenario considers the potential radiation exposure resulting from buried concrete debris, from the basement of the demolished buildings, which is assumed to be used as backfill materials.

3.5.5.3.1 Scenario Definition and Exposure Pathways

The concrete debris containing residual radioactive materials was assumed to be covered with about 0.9 meter (3 feet) of soil and left on site. A farmer is assumed to move to the site after it is released and live on the land above the concrete debris, grow all or a portion of his food on the site, and drink water from a ground-water source that extends to the buried debris. Potential exposure pathways include direct radiation from the concrete debris, inhalation of airborne radionuclides, and ingestion of contaminated plant foods, meat and milk, ground water, and concrete debris. Because the ground water would flow to the Connecticut River (according to the ground-water study), potential radionuclide concentration in the river, a likely surface water source, would be very small because of the large dilution of the ground water once it is discharged to the river. Therefore, the ingestion of contaminated aquatic food was not included in dose calculations. The selected pathways are representative of a resident farmer and are judged to be appropriate.

3.5.5.3.2 Application of RESRAD for Dose Modeling

The radiation dose that might be incurred by the residential farmer was calculated with the RESRAD computer code. Because the concrete debris would extend to the saturated zone, assignment of special values to some parameters is needed to obtain the correct dose results.

Input Parameter Selection: To obtain radiation dose estimates for the buried concrete, which would extend beyond the ground water table, the MB model was selected, and the number of unsaturated zone was set to 0. Without any unsaturated zone, the contaminated zone (i.e., the concrete debris) would be in contact with ground water. The selection of the MB model would exclude transport of radionuclides and the resulting dilution of their concentrations in the saturated zone. These two settings are necessary and consistent with the conceptual model.

Site-specific distribution coefficient (K_d) values were used for the concrete debris in dose calculations. The K_ds represent the partitioning of a solute in solution (i.e., radionuclide) between the solid matrix and water. Concrete cores taken from the Containment Building and the Waste Disposal Building and ground-water samples taken from the Haddam Neck site were analyzed in laboratories to measure the K_d values. The range of potential K_d values for most of the radionuclides of concern was obtained from laboratory measurements, and potential distributions of the K_d values were developed. For those radionuclides for which measurement data were not available, K_d values and distributions were set to those of chemically similar radionuclides through chemical analogy. Because plant transfer factors are strongly correlated with K_d values, potential bounding values were calculated by using the bounding K_d values and the correlation with the K_d parameter, as suggested in NUREG/CR-6697. Uniform or log uniform distributions were then assumed for the plant transfer factors. Because the properties of concrete and soil are very different and literature data for concrete are very limited, the use of measurement data is the best approach and is acceptable. Although the measurement data are subject to uncertainty, the use of sensitivity analysis helps in selecting a conservative value for use in final dose calculations.

Because the area of the concrete debris would be limited to the footprint area of the Containment Building basement, 1533 square meters, it was decided that contamination fractions of plant food, meat, and milk that the resident farmer consumed would be less than 1. The default RESRAD setting that calculates contamination fractions on the basis of ratio of the contaminated area to the area needed to produce 100 percent of the needed food products was used. This approach was judged to be acceptable.

The parameter selection process discussed in Section 3.5.3.3.1 was applied to select values for the other input parameters. Because the process would result in conservative dose results, the input parameter values selected in this way were judged to be acceptable. The final input parameter values are listed in Table F-4 of the LTP.

Sensitivity Analysis: Probabilistic sensitivity analysis was conducted to study the influence of input parameters and decide their values. The procedure used was the same as that evaluated in Section 3.5.3.3.2. This procedure is judged to be comprehensive and acceptable.

3.5.5.3.3 Conversion of DCGLs

The DCGLs derived using the RESRAD dose modeling results were for the concrete debris, which was modeled as an underground volumetric contamination source. The DCGLs are expressed in units of picocuries per gram. For use in the final status survey, corresponding DCGLs for the buildings before demolition are needed. The DCGLs for the buildings would be the same as the DCGLs for the concrete debris if the buildings were volumetrically contaminated. If the contamination of the buildings is limited to the surfaces, conversion of the concrete debris DCGLs to the building-surface DCGLs is needed and can be obtained by using a proper surface-to-volume conversion factor. The conversion factor used in the LTP was developed by assuming that all the concrete debris was contaminated. This assumption is conservative because it is more likely that only some portions of the building structure are contaminated. Building-surface DCGLs were obtained by evenly distributing the total amount of radionuclides in concrete debris to the surfaces of the buildings. The final DCGLs are listed in Table 3.5.7 and are determined to be conservative and acceptable.

3.5.5.3.4 Comparison of the Peak Dose Results

The NRC staff conducted independent probabilistic analysis, using the values and distributions of input parameters listed in Table D-4 of the LTP. Initially, it was expected that the deterministic peak dose of each radionuclide obtained, using the selected parameter values listed in Table F-4, would exceed the 75th quantile value of the corresponding peak dose distribution from probabilistic analysis. However, for Cs-134, Cs-137, Eu-152, Eu-154 and Eu-155, this was not the case. For Eu-154, the corresponding quantile value is only about 25 percent. Further investigation revealed that the "thickness of cover material" parameter is responsible for this discrepancy. By fixing the cover thickness parameter to 0.5 meter (1.6 feet) instead of using a uniform distribution in the probabilistic analysis, the deterministic peak dose would exceed the 50th quantile value of the corresponding distribution.

3.5.6 Development of Operational, Gross-Activity, Surrogate-Ratio, and Elevated-Measurement-Comparison DCGLs

DCGLs developed for the LTP included DCGLs$_{op}$ for a given survey unit; gross-activity DCGLs; surrogate-ratio DCGLs; and DCGLs$_{EMC}$. The representative radionuclide mix will be established for each survey unit on the basis of: (1) gamma-ray spectroscopy and alpha spectroscopy (where appropriate) or equivalent analyses; (2) scaling factors (to establish the activity contribution for difficult-to-measure radionuclides); and (3) historical characterization information (for areas with no measurable activity). The licensee has proposed to use the DCGLs$_{op}$ in cases where potential exposures can result from more than one of the following sources: contaminated soil; contaminated concrete debris; and existing contaminated ground water. The surrogate-ratio DCGLs adjust DCGLs for easy-to-detect radionuclides, to consider estimated dose contributions from difficult-to-detect radionuclides.

3.5.6.1 Operational DCGLs

The LTP (Section 5.4.7.1) discusses the method to develop DCGLs$_{op}$ for survey units where someone could be exposed to residual radioactivity in multiple media. The DCGLs$_{op}$ will ensure that the 0.25 mSv/yr (25 mrem/yr) dose limit is not exceeded even though different DCGLs will be used for the various media. In areas with residual radioactivity in multiple media, the total dose could be the sum of the dose from the soil contamination (all pathways included); the dose associated with existing ground-water contamination (water-dependent pathways only); and the dose associated with using water obtained from a well located on the downgradient edge of the concrete debris (water-dependent pathways only). The areas of the site where these three exposures could occur concurrently are where building debris may be buried and existing ground- water contamination may be present. The licensee has reported that this area encompasses approximately 15,600 square meters and includes the industrial area of the site.

Table 5.4 of the LTP provides the survey areas where DCGLs$_{op}$ will be applied. These include areas where existing ground-water concentrations of tritium have been measured or predicted to be $\geq$ 1000 pCi/L. This area also includes survey units that fall within an area encompassing 100 meters of the existing tritium plume (i.e., a 100-meter buffer zone). Since the ground-water characterization efforts are still ongoing, the survey areas affected may change. For example, hydrogeologic information may show that a future well could have a radius of influence greater than 100 meters, in which case a larger buffer zone would be needed because a well located outside of the contaminated area would be able to withdraw contaminated ground water. The hydrogeologic data could also identify contamination in other areas of the site. For these reasons, a condition of the license amendment will require, after completion of the ground-water characterization, the licensee to perform a well-capture-zone analysis, and to ensure that the ground-water dose contribution has appropriately considered all contamination.

The $DCGLs_{op}$ for building surfaces would be the lower of the DCGLs for either the building-occupancy-scenario DCGLs or the building $DCGLs_{op}$ developed using equation 5-5 of the LTP. To use equation 5-5, the fraction of dose from the concrete debris attributable to water-dependent pathways (f_w) must be known. These are listed in Table 3.5.8.

Table 3.5.8 Dose Fraction from Water-Dependent Pathways for Building Debris

Radionuclide	f_w (%)	Radionuclide	f_w (%)
H-3	92.1	Cs-134	20.84
C-14	1.35	Cs-137	35.37
Mn-54	5.52	Eu-152	4.45
Fe-55	7.68	Eu-154	5.51
Co-60	1.49	Eu-155	42.03
Ni-63	4.45	Pu-238	80.57
Sr-90	10.70	Pu-239	77.80
Nb-94	0.33	Pu-241	69.56
Tc-99	6.19	Am-241	70.43
Ag-108m	0.08	Cm-243	44.07

Although the method proposed for deriving the $DCGLs_{op}$ seems restrictive, the NRC staff will need to determine the acceptability of the $DCGLs_{op}$ used in the final status survey as part of its review of that survey.

3.5.6.2 Gross Activity DCGLs

The LTP proposes the use of gross-activity DCGLs ($DCGL_{GA}$), generally for alpha or beta surface activity, for demonstrating compliance for areas with multiple radionuclides, rather than determining individual radionuclide activities. Development of $DCGL_{GA}$ values requires determination of the relative fraction of the total activity contributed by each radionuclide within the survey unit. The LTP states that the radionuclide mix will be established for each survey unit on the basis of gamma-ray spectrometry and alpha spectrometry (where appropriate) or equivalent analysis of representative samples. Radionuclides present in a survey unit in concentrations greater than 5 percent of their respective DCGLs would be used in establishing the $DCGLs_{GA}$ for a survey unit. The aggregate of all radionuclides not included in deriving the $DCGLs_{GA}$, based on the percentage of their respective DCGLs, will not exceed 10 percent. This practice is similar to the process presented in NUREG-1727, according to which radionuclides that contribute less than 10 percent of the dose limit can be screened out. Although the method proposed for deriving the $DCGLs_{GA}$ is acceptable, the NRC staff will need to determine the acceptability of the relative fractions derived for each survey unit as part of its review of the final status survey, to ensure compliance with the dose limit. Further, if $DCGLs_{GA}$ are not used, the NRC staff will need to ensure that for survey units with multiple radionuclides the sum of fraction of the dose contribution for each radionuclide does not exceed one.

3.5.6.3 Surrogate-Ratio DCGLs

The LTP proposes the use of surrogate-ratio DCGLs in areas where difficult-to-detect radionuclides may be present. The surrogate ratio DCGLs are computed on the basis of the activity ratio between difficult-to-detect radionuclides and the easy-to-measure radionuclides. Use of this approach can save time and resources by enabling the licensee to measure the concentration of one radionuclide and relate the concentration of other radionuclides, based on this ratio. This procedure also allows the DCGLs specific to hard-to-detect radionuclides in a mix to be expressed in terms of several or a single radionuclide that is more readily measured. The use of surrogate-ratio DCGL values requires that either a sufficient number of measurement, spatially separated throughout the survey unit be made to establish a consistent ratio, or that the data collected be reviewed to establish the ratio and the DQO process be used to select an appropriate ratio from the data.

According to the LTP, before performing an FSS the beta/gamma-to-alpha ratio will be established for a survey unit, by using at least six samples. The samples will be collected from the same unit or from other similar units. These samples will be analyzed for TRU radionuclides (using gross-alpha or alpha spectroscopy techniques) and for beta-gamma activity (using gross-beta analysis and/or gamma spectroscopy techniques). If any TRU radionuclide contributes more than 5 percent of its DCGLs, or if the aggregate of all TRU radionuclide contributions is more than 10 percent of that from the samples collected for the survey unit, the surrogate DCGLs will be determined using Equation 5.8 of the LTP.

First, gross DCGLs for easy-to-detect and difficult-to-detect radionuclides would be established from the samples collected. To establish gross DCGLs for easy-to-detect and difficult-to-detect radionuclides if the percent coefficient of variation (CV) of the average gross DCGLs has a value of 25 percent or less, average DCGLs will be applied to the survey unit; otherwise either the lowest DCGLs from the samples will be applied or additional analysis will be performed. Once gross DCGLs for easy-to-detect and difficult-to-detect radionuclides are established, Equation 5.8 of the LTP will be applied to calculate the surrogate DCGLs. If the percent CV of the average surrogate DCGLs has a value of 25 percent or less, average surrogate DCGLs will be applied to the survey unit. Otherwise, either the lowest surrogate DCGLs from the samples will be applied or additional analysis will be performed.

According to the LTP, surrogate DCGLs will only be applied in survey units where an individual TRU radionuclide contributes more than 5 percent of its DCGLs, or if the aggregate of all TRU radionuclide contributions is more than 10 percent. This practice is acceptable in lieu of the guidance provided in NUREG-1727, according to which radionuclides that contribute less than 10 percent of the dose limit can be screened out. Therefore, the method proposed for deriving the surrogate-ratio DCGLs is acceptable; however, the NRC staff will also need to determine the acceptability of the actual concentration ratios used in deriving the surrogate-ratio DCGLs as part of its review of the FSS.

3.5.6.4 Elevated Measurement Comparison DCGLs

Area factors are needed for elevated-measurement comparisons during scanning in Class 1 areas. The number of static measurements needs to be adjusted if the sensitivity of the scanning technique is not capable of detecting levels of residual radioactivity below the DCGLs. Area factors are also needed to identify small areas with elevated residual radioactivity that may require further investigation.

The licensee calculated area factors for both the resident farmer and the building-occupancy scenarios. Area factors for the resident farmer scenario were computed by running the RESRAD computer code repeatedly, with changing areas of contamination and other parameters (such as length parallel to aquifer flow, fraction of the plant, meat, and milk from site) affected by the area of contamination. Area factors were computed for all radionuclides of concern at the site considering all potential pathways of exposure. The area factors for all radionuclides of concern at the site for the resident farmer scenario are listed in Table 5-5 of the LTP. RESRAD-BUILD Version 2.37 was used to compute area factors for the building occupancy scenario for all radionuclides of concern at the site. For calculating area factors for the building occupancy scenario, only one source was modeled instead of six sources as used for base-case DCGLs. The area of the source was varied from 1 to 100 square meters. Areas larger than 100 square meters would use base-case DCGLs. The area factors for the building occupancy scenario are listed in Table 5-6 of the LTP.

The NRC staff independently verified area factors, using RESRAD Version 5.9 for the resident farmer scenario, and RESRAD-BUILD Version 2.37 with all pathways active for the building-occupancy scenario, and found no discrepancies. Maintaining consistency between the derivation of the base-case DCGLs and $DCGL_{EMC}$ values gives reasonable assurance that doses from exposure to smaller areas with elevated residual radioactivity would not exceed the dose limit. The area factors will be used for Class 1 areas at the site.

3.5.7 ALARA Demonstration

The ALARA cleanup levels for the Haddam Neck decommissioning are discussed in Section 4.2 and Appendix B of the LTP. According to the information provided in the LTP, the ALARA cleanup levels would be established at a predefined generic ALARA screening value, and if these levels are not met for a specific survey unit, a survey-unit-specific ALARA analysis would be conducted by the process described in draft Regulatory Guide 4006, "Demonstrating Compliance with the Radiological Criteria for License Termination." The licensee’s ALARA analysis would ensure, apart from the requirement that the TEDE to the average member of the critical group does not exceed 0.25 mSv/yr (25 mrem/yr), that the efforts to remove residual contamination are commensurate with the risk that exists from leaving the residual contamination in place.

The licensee has indicated that currently no population is deriving its drinking water from a downstream supply, and, based on current knowledge of the aquifer on-site, it is doubtful that this aquifer would be used as a drinking water source for a large population. If it is determined, during the program of ongoing ground-water monitoring, that drinking water for a large population could be supplied by ground water on site, the collective dose for that population would be included in the ALARA calculations. This is acceptable according to the guidance in NUREG-1727, Appendix D, Section 1.6, because if a site has residual radioactivity in ground water from site operations, it would be necessary to calculate the collective dose from consumption of the ground water.

The NRC staff has reviewed the information submitted by the licensee to demonstrate that the preferred decommissioning option is ALARA, as required in 10 CFR Part 20, Subpart E, in accordance with the criteria in the NMSS Decommissioning Standard Review Plan, Section 7.0 "ALARA Analysis". On the basis of this review, the NRC staff concludes that the preferred option selected (either predefined generic ALARA screening value or a survey-unit-specific ALARA analysis) would provide reasonable assurance that the remediation would result in residual radioactivity levels that are ALARA.

The licensee has committed to showing compliance during remediation by either meeting the concentration limits established on the basis of a generic ALARA screening value or setting appropriate remediation goals, based on a survey-unit-specific ALARA analysis, with established protocols to optimize the remediation activities during decommissioning. A process similar to the one given in Appendix D of the NUREG-1727 would be used by the licensee in a survey-unit-specific ALARA analysis. The specific process and equations that would be used are given in Appendix B of the LTP and are acceptable. The site procedures that would be used for ALARA evaluations were determined to be appropriate. However, the NRC staff will need to determine the acceptability of the ALARA evaluations as part of its review of the final status survey.

3.6 Site End Use

Section 50.82(a)(9)(ii)(E) requires a licensee to provide a description of the planned end use of the site if the licensee proposes to have its license terminated under restricted conditions. The licensee has proposed to have its license terminated with no restrictions on the use of the site, under the provisions of 10 CFR 20.1402. Therefore, the licensee is not required to provide a description of the planned end use of the site.

3.7 Cost Estimate

In the cost estimate included in the Haddam Neck Nuclear Power Plant LTP dated March 2001, and supplements dated September 6, 2001, and March 7, 2002, the licensee estimated the remaining cost, after January 1, 2000, to complete decommissioning, in 2000 dollars as $226.2 million. The $226.2 million estimated cost included $128.0 million for dismantlement and decommissioning; $78.6 million for radioactive waste disposal; and $19.6 million for cost to complete the final survey. In accordance with NRC regulations, fuel storage and site restoration costs are not considered decommissioning costs, and the $226.2-million-cost estimate excluded spent fuel and site restoration costs. The total cost estimate for site remediation is $564.0 million, which includes an overall contingency cost of $39 million; $112.4 million for long-term spent-fuel costs through 2023, including $44.3 million to construct and transfer spent fuel to the independent spent fuel storage installation; $15.3 million for site restoration; and $226.2 million for the remaining decommissioning.

The review of the decommissioning cost estimate for Haddam Neck was based on a comparison of several activities to be conducted at Haddam Neck, with similar activities conducted at other nuclear facilities either undergoing decommissioning or that have completed decommissioning -- and an evaluation of the licensee's cost assumptions used for estimating major decommissioning activities and tasks, including a review of the dismantlement and decontamination costs, waste disposal costs, and final survey cost. The NRC staff used the "2000 Means Building Construction Cost Data," the "2000 Means Facility Cost Data," and the "1998 Means Labor Rates for the Construction Industry," as references in its analysis. A comparison to similar activities conducted at Rancho Seco Nuclear Generating Station; Fort St. Vrain (FSV); Yankee Rowe Nuclear Power Station; Shoreham Nuclear Power Station (SNPS); and Trojan Nuclear Power Plant, were also conducted. Where costs were escalated to year 2000 dollars, an annual inflation rate of 3.5 percent was used.

Many activities and tasks to dismantle/decommission Haddam Neck are similar to activities and tasks conducted at other nuclear generating plants that are undergoing decommissioning or have been decommissioned. The NRC staff reviewed several areas to assess the reasonableness of the estimated activity costs to decontaminate Haddam Neck. For example, the cost of the final status survey was compared to the Trojan final survey estimate, and both FSV and SNPS actual final survey costs, and after escalation, the Haddam Neck final survey cost was found to be in a reasonable range.

The NRC staff also compared the specific factors that were identified in Section 7.3, "Decommissioning Funding" of the LTP, used to develop the LTP cost estimate, to cost factors in NUREG-1700, and the cost factors for developing the Trojan and FSV estimates. The specific factors used for the Haddam Neck cost estimate compared favorably with other facilities, and were found to be reasonable.

The NRC staff also reviewed the cost allocated for waste disposal, $78.6 million, and the basis for the cost estimate. Section 7.2.4, "Radiological Waste Disposal" based the disposal costs on the cost for disposal at the Barnwell facility, although a large part of the waste is Class A waste, which will either be sent to GTS Duratek for waste processing, or Envirocare facility, where disposal charges are significantly less than Barnwell. The portion of waste being shipped to Barnwell consists mainly of Classes B and C waste. Based on the assumption that all waste would be shipped to Barnwell, the NRC staff concluded the estimate for waste disposal to be conservative.

The NRC staff reviewed the cost estimate of $19.6 million for the Haddam Neck FSS and found it to be reasonable when compared to the FSS cost estimate for Trojan, and the actual FSS cost for SNPS, and FSV. For SNPS, the final survey cost was approximately $14.0 million, and for FSV, the cost was approximately $22.0 million. The FSS cost for Trojan is estimated to be $17.8 million, and the Trojan estimate was based on the MARRSIM survey technique. Based on the plant configuration, the Haddam Neck FSS cost is reasonable.

The NRC staff reviewed the composite labor rate for removal and decontamination activities identified in the September 6, 2001, submittial, "Decommissioning Study of the Haddam Neck Plant" dated March 1999, Section 3, "Basis and Assumptions" with the labor rates in the "1998 Means Labor Rates for the Construction Industry" and escalated the Means rates to $2,000 and found the Haddam Neck labor rate to be reasonable.

The NRC staff reviewed the information in the LTP for the Haddam Neck Nuclear Plant, against Section B.7 of NUREG-1700, and similar activities conducted at plants undergoing or having completed decommissioning, and based on this review, the NRC staff determined that the licensee has met the requirements of 10 CFR 50.82(a)(9)(ii)(F) with respect to providing an updated site-specific estimate of the remaining decommissioning costs. The NRC staff recognizes that all the funds to meet the cost of decommissioning of Haddam Neck have not been set aside, but the proposed funding plan, included as Table 7.2, "Decommissioning/Spent Fuel Trust Analysis" allocates sufficient funds to complete decommissioning of Haddam Neck.

3.8 Environmental Report

In accordance with the requirements of 10 CFR 50.82(a)(9)(ii)(G), a licensee is required to provide a supplement to the environmental report, pursuant to 10 CFR 51.53, describing any new information or significant environmental changes associated with the licensee's proposed license termination activities. The licensee, by letter dated July 7, 2000, submitted its Haddam Neck Plant LTP. In the LTP, the licensee incorporated, by reference, a report titled "Decommissioning Environmental Review" dated August 1997. Based on this report the licensee concluded that the environmental impacts of decommissioning activities are bounded by previously issued environmental reviews, such as the "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," NUREG-0586, dated August 1988; "Environmental Assessment for Proposed License Extension" dated November 23, 1987; and "Final Environmental Statement, Haddam Neck Nuclear Power Plant" dated October 1973. Under the provisions of 10 CFR 51.21, the NRC staff prepared an environmental assessment (EA) to determine the impacts of the proposed LTP amendment on the environment. In this EA, published in the *Federal Register* on November 4, 2002 (67 FR 67212), the NRC staff found that approval of the LTP would not cause any significant impacts on the human environment and is protective of human health.

The NRC staff has reviewed the information in the LTP for the Haddam Neck Plant, according to Section B.8 of NUREG-1700. Based on this review and the EA prepared by the NRC staff, the NRC staff has determined that the licensee has met the requirements of 10 CFR 50.82(a)(9)(ii)(G) and 10 CFR Part 51.53.

3.9 Phase Release

In Section 1.4.2 of the LTP, the licensee indicated that it may want to remove areas from the license once decommissioning and remediation tasks are complete and the licensee can demonstrate that the area and any associated buildings will have no adverse impact on the site in the aggregate to meet the 10 CFR Part 20, Subpart E, criteria for unrestricted release. Before doing so, the licensee will review and assess the impact of releasing the specific area and any buildings on the following: Updated Final Safety Analysis Report, Technical Specifications, Environmental Monitoring Program, Offsite Dose Calculation Manual, Defueled Emergency Plan, Security Plan, Post Shutdown Decommissioning Activities Report, LTP, Ground-Water Monitoring Program, 10 CFR Part 100 Siting Criteria, and Environmental Report. When the licensee has determined that a portion of the site can be removed from the license without any adverse impact on the ability of the site in the aggregate to meet the 10 CFR Part 20, Subpart E, criteria for unrestricted release, the licensee will submit a letter of intent to remove a portion of the Haddam Neck property from the 10 CFR Part 50 license to NRC no later than 60 days before the anticipated date for release. This letter will include the basis for the determination of whether the area is impacted (i.e., the area may contain residual radioactivity in excess of natural background or fallout levels due to facility operations) or non-impacted (i.e., there is no reasonable possibility of residual contamination). If the area is impacted the licensee's letter will, among other information, include an FSS based on methods described in the LTP. This will allow the NRC staff to review the licensee's basis for considering the area suitable for release and allow NRC to make a determination of whether a confirmatory survey will be performed. The NRC staff will notify the licensee of the results of the review.

The NRC staff finds the licensee's proposed process for releasing property from the license acceptable.

3.10 LTP Change Procedure

The licensee has proposed that it be authorized to make certain changes to the NRC-approved LTP without NRC approval if these changes do not:

- Require NRC approval pursuant to 10 CFR 50.59, 10 CFR 50.82(a)(6), and 10 CFR 50.82(a)(7);

- Increase the radionuclide-specific derived concentration guideline levels (as discussed in Section 6 of the LTP) or area factors (as discussed in Section 5.4.7.4 of the LTP);

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

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

- Change 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 3.3.3.2 of the LTP, and definitions for the different classifications for subsurface soils are provided in Section 3.3.3.1.5 of the LTP;

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

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

Based on its evaluation of the LTP for the Haddam Neck Plant, the NRC staff concludes that authorizing the licensee to make certain changes to the LTP during the final site remediation is acceptable, subject to the above listed conditions.

3.11 Conclusion

Based on the above evaluation, the NRC staff concludes that the proposed LTP for the Haddam Neck Plant and the proposed amendment to incorporate the LTP in Facility Operating License No. DPR-61, with criteria to allow the licensee to change the LTP without prior Commission review and approval, is acceptable.

3.12 Summary of Areas Requiring Further Validation

As a result of its review of the LTP, the NRC staff has determined that there are several areas, related to the LTP, that will need to be validated either as part of the NRC staff's ongoing inspection effort at the Haddam Neck site, or during the FSS. These areas are:

- The licensee has committed to document its radiological survey plans, using the DQO process, and associated data evaluation reports produced from subsequent RSSI surveys (e.g., operational, characterization, and remedial-action support surveys). This information and supporting data will be available on site for NRC review during inspections. The NRC staff expects to review this information for each survey unit before FSS implementation (Section 3.1).

- The FSS will be conducted using guidance in MARSSIM to demonstrate compliance with the criteria specified in 10 CFR Part 20, Subpart E, for unrestricted release of the Haddam Neck site. The types of surveys and sampling described for complete characterization will require further NRC staff validation to ensure that the methodology and data are adequate when this information becomes available (Section 3.1).

- For cases of survey units for which there is no measurable activity distinguishable from background at the time of the FSS, the licensee will select a representative radionuclide mix, based on the historical characterization information for the survey unit of interest or for units with similar history and physical characteristics. The NRC staff will review the licensee's rationale for its selecting representative radionuclide mixes. This review will be performed as part of the inspection process (Section 3.1).

- The turnover survey process, together with any additional characterization and remediation surveys performed, represents at least one, but possibly several, opportunities to collect and evaluate additional survey data before conducting the FSS for a survey unit. The NRC staff expects the documented turnover assessments, as well as the results of turnover surveys, when performed, to be available for NRC review during inspections (Section 3.1).

- Licensees may use data developed from site characterization as FSS data, providing these data meet the DQOs appropriate for FSS. The NRC staff will review DQOs and changes made to DQOs as part of NRC's ongoing inspection process (Section 3.1).

- The licensee has committed to obtain additional cores to establish radioactivity levels of materials subject to neutron flux after the reactor vessel and other highly radioactive components have been removed. The licensee will collect site-specific data to characterize the nature and extent of radioactive contamination for reactor vessel/components and concrete shield structures near the reactor vessel. The NRC staff will review these data as part of NRC's ongoing inspection effort (Section 3.1.1.1).

- In accordance with the guidance provided in MARSSIM, if an area could be classified as a Class 1 or Class 2 for the FSS, based on the HSA and scoping survey results, a characterization survey is warranted. The licensee has committed to fully characterize such areas. The NRC staff will review the characterization of these areas, as part of its ongoing inspection efforts (Section 3.1.1.1).

- In Section 5.7 of the LTP, the licensee will survey for activity beneath surfaces (cracks crevices, paint, and paved surfaces); sewer systems; plumbing and floor drains; interiors of ventilation ducts; underground and embedded piping; activated concrete; and interiors and exteriors of both systems and equipment. The NRC staff has determined that the approach the licensee proposed to characterize both structures and internal surfaces of embedded piping is acceptable. However, the NRC staff will go through the inspection process review for the associated RSSI data, to ensure that the methodology was implemented accordingly and data are adequate until complete characterization information becomes available (Section 3.1.1.1).

- Surrogate ratios will be established using characterization and/or post-remediation data for the survey unit of interest. Because characterization is still ongoing at the site and remediation is not completed, the licensee is unable to provide specific DCGL values for the FSS. The DQO process should be used to assess the use of surrogates. Therefore, as part of the inspection process, the NRC staff will review the licensee's assessment of specific ratios for inclusion in the FSS design (Section 3.1.1.3).

- The licensee states that the DQO process applied to subsurface soils will be similar to the DQO process used for other surveys at the Haddam Neck Plant; however, there may be differences in design input parameters to satisfy objectives of the plan. As with surface soil, the NRC staff will review the DQOs and survey plans for subsurface soil as part of the inspection process (Section 3.1.1.5).

- The NRC staff will review the licensee's exposure rate data during the RSSI process and in the FSS Report, as part of the inspection process, to determine whether they are acceptable and instrument use and data analysis are consistent with MARSSIM guidance (Section 3.1.1.10).

- The NRC staff will review FSS design changes related to the use of advance techniques as part of the inspection process and will review the DQOs to see if changes are adequately reflected (Section 3.1.1.5).

- NRC will inspect to determine whether the licensee has met the conditions that the licensee has established, in Section 5.5 of the LTP, regarding the implementation of advanced survey techniques (Section 3.1.1.5).

- Survey procedures for plant systems and embedded piping have not been completely described—although the LTP provides specific commitments the licensee intends to satisfy. These commitments include 100 percent surface scans of Class 1 systems and embedded piping, with 30 measurements to be performed at accessible points. Classes 2 and 3 survey-unit scanning coverage will vary, depending on accessibility and historical information. Scanning and measurement activities are to be supplemented with indirect measurement techniques and scale sampling. The survey methods for plant systems and embedded piping, as presented in the LTP, are recognized as an acceptable general approach, but will require further validation during the FSS (Sections 3.1.1.1 and 3.4).

- When reviewing FSS results, the NRC staff will examine whether the licensee has measured and/or accounted for each of the radionuclide contaminants (Table 6.1 for soil, Table 6.2 for ground water, Table 6.3 for building surfaces, and Table 6.4 for concrete debris) when presenting dose compliance information for each survey unit. Additionally, whenever the licensee accounts for HTD radionuclides through surrogate analyses, the NRC staff will examine if the licensee has verified whether the activity ratios (activity ratio for difficult-to-detect radionuclides to easy-to-detect radionuclides) remain valid for use during the FSS in accordance with Section 5.4.7.3 of the LTP and consistent with Section 4.3.2 of MARSSIM (Section 3.4).

- The NRC staff will review the gross activity beta-to-alpha ratio for embedded piping remaining at the site with diameters greater than 24-inches to ensure that the ratio is greater than or equal to 15 to 1. For buried piping remaining at the site, located in the saturated zone, the NRC staff will ensure that the radioactivity remaining in the piping does not exceed the limits specified in Table 3.5.1. Further, if the licensee uses scaling factors to establish gross-activity levels via radionuclide-specific measurements or other assessments, the NRC staff will: (1) review any scaling factors used for acceptability and (2) verify that the entire interior surface of the piping/penetration was surveyed (Section 3.5.1).

- The NRC staff will review information from the ongoing ground-water characterization study to ensure that the appropriate mix of radionuclides have been considered in the development of the $DCGLs_{op}$ and that the area for which these DCGLs are applied is appropriate (Section 3.5.6.1).

- The licensee stated in the LTP that relative fractions for application of $DCGLs_{GA}$ will be based on composite samples from specific areas collected before the FSS. The NRC staff will determine the acceptability of the relative fractions derived for each survey unit as part of its review of the FSS to ensure compliance with the dose limit. Further, if $DCGLs_{ga}$ are not used, the NRC staff will ensure that for survey units with multiple radionuclides the sum of fractions of the dose contribution for each radionuclide does not exceed one. Accordingly, the NRC staff will determine the acceptability of specific $DCGLs_{GA}$ used in the FSS as part of its review of the FSS (Section 3.5.6.2).

4.0. FINAL NO SIGNIFICANT CONSIDERATION DETERMINATION

The objective for decommissioning the Haddam Neck site is to reduce the residual radioactivity to levels that permit release of the site for unrestricted use and for termination of the 10 CFR Part 50 license, held by the licensee for the plant, in accordance with the site release criteria in 10 CFR Part 20. The purpose of the LTP is to satisfy the requirements of 10 CFR 50.82, "Termination of license," using the guidance provided in draft Regulatory Guide 4006, "Demonstrating Compliance with the Radiological Criteria for License Termination." The LTP describes the decommissioning activities to be performed by the licensee, the process for performing the final status surveys, and the methods for demonstrating that the site meets the criteria for release for unrestricted use. Therefore, the LTP involves the reduction of radioactivity at the site, and does not involve in any way the operation of the plant and the generation of new radioactive material. Based on this and on the evaluation given above on the proposed amendment to approve the LTP, and the criteria whereby the licensee may change the plan without prior review and approval of the Commission, the NRC staff provides its final determination on the no significant hazards consideration criteria in 10 CFR 50.92 for the proposed amendment.

1. Does the amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   The bounding radioactivity release event is given in the Haddam Neck Plant Updated Final Safety Analysis Report. The proposed amendment (1) approves a plan to reduce the radioactivity at the site and (2) allows the plan to be changed without NRC staff review and approval if the change does not:

   a. Increase the radionuclide-specific DCGLs or area factors,

   b. Increase the probability of making a Type I decision error above the level stated in the LTP,

   c. Increase the investigation level thresholds for a given survey unit classification,

   d. Change the classification of a survey unit from a more restrictive classification to a less restrictive classification,

   e. Reduce the coverage requirements for scan measurements, or

   f. Involve reliance upon statistical tests other than the WRS or Sign Test for data evaluation.

   Because the plan, including the change criteria, do not allow an increase in radioactivity at the site, there cannot be an increase in the probability or consequences of an accident previously evaluated. Therefore, the NRC staff concludes that the LTP amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   The LTP and LTP change criteria are not initiators of an accident. Therefore, the LTP amendment does not create a new failure mechanism that has not been previously evaluated, or affect plant systems, structures, or components in a way not previously evaluated. Therefore, the NRC staff concludes that the amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the amendment involve a significant reduction in a margin of safety?

   The LTP is the means for the licensee to perform remediation activities to reduce residual radioactivity at the site and demonstrate compliance with the radiological criteria for license termination provided in 10 CFR 50.82 and 10 CFR 20.1402, "Radiological criteria for unrestricted use." These regulations provide criteria for acceptable doses to the public in unrestricted areas, and the LTP is designed to comply with this criteria.

Based on this, the NRC staff concludes that the amendment does not involve a significant reduction in a margin of safety.

Therefore, based on the above, the NRC staff concludes that the proposed amendment does not involve a significant hazards consideration as defined by 10 CFR 50.92.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on November 4, 2002 (67 FR 67212).

Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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