CONNECTICUT YANKEE ATOMIC POWER COMPANY

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

July 7, 2000
CY-00-077
Re: 10CFR50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Haddam Neck Plant
Submittal of License Termination Plan and Proposed Revision To Operating License

In accordance with 10CFR50.82(a)(9), Connecticut Yankee Atomic Power Company (CYAPCO) hereby submits the License Termination Plan for the Haddam Neck Plant. This License Termination Plan demonstrates that the remaining decommissioning activities will be performed in accordance with the requirements of Title 10 to the Code of Federal Regulations, will not be inimical to the common defense and security or to the health and safety of the public, and will not have a significant effect on the quality of the environment.

Accompanying the License Termination Plan, CYAPCO submits, pursuant to 10 CFR 50.90, an application to amend the Haddam Neck Plant license. This proposed license amendment adds a license provision that provides criteria by which the need for NRC approval of changes to the approved License Termination Plan is determined.

Attachment 1 provides the background and reason for the proposed change, a description of the proposed change, a no significant hazards consideration determination, and an environmental impact consideration determination. Attachment 2 provides a copy of the affected page to Facility Operating License No. DPR-61 with changes annotated. Attachment 3 provides the Haddam Neck Plant License Termination Plan for NRC review and approval. In accordance with 10CFR50.82(a)(9)(i), the Haddam Neck Plant License Termination Plan is being submitted as a supplement to the UFSAR, and is being maintained accordingly.

This change does not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in the margin of safety.

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The Plant Operations Review Committee and the Nuclear Safety Assessment Board have reviewed the proposed amendment request.

Additionally, as appropriate, the Haddam Neck Plant License Termination Plan incorporates preliminary comments from the Environmental Protection Agency, the Connecticut Department of Environmental Protection, the Connecticut Department of Public Health, and the Community Decommissioning Advisory Committee.

In accordance with 10CFR 50.91(b), CYAPCO is providing the State of Connecticut with a copy of the proposed amendment request.

The proposed change has been reviewed in accordance with 10 CFR 50.92 and has been determined to not constitute a Significant Hazards Consideration (SHC). In addition, the proposed change has been reviewed in consideration of 10 CFR 51.22; and it has been determined that the proposed change meets the criteria for a categorical exclusion from requiring an environmental impact statement.

If the NRC staff should have any questions regarding this submittal, please contact Mr. G. P. van Noordennen at (860) 267-3938.

Sincerely,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

Kenneth J. Heider

K. J. Heider
Vice President -Operations and Decommissioning

Attachments

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Subscribed and sworn to before me

this 7th day of July, 2000

Gerard P. van Noordennen

Date Commission Expires: December 31, 2002

Attachment 1

Background and Reason for Change

On December 5, 1996, Connecticut Yankee Atomic Power Company (CYAPCO) notified the NRC of the permanent cessation of operations of its Haddam Neck Plant (HNP) and the permanent removal of all fuel assemblies from the reactor vessel and their emplacement in the Spent Fuel Pool. Following the cessation of operations, CYAPCO has been preparing to decommission the HNP. The Post Shutdown Decommissioning Activities Report was submitted, in accordance with 10CFR50.82(a)(4), on August 22, 1997. On January 26, 1998, CYAPCO transmitted an Updated Final Safety Analysis Report (UFSAR) to reflect the plant's shutdown status, and on June 30, 1998, the NRC amended the HNP Facility Operating license to reflect the plant's permanent shutdown status.

Section 50.82(a)(9) to Title 10 of the Code of Federal Regulations requires that a licensee must submit an application for the termination of the site's Part 50 license. The application for termination of the license must be accompanied or preceded by a license termination plan to be submitted for NRC approval. The license termination plan is to be a supplement to the plant's UFSAR or an equivalent document and is required to be submitted at least two years before the date of license termination. The proposed license change provided in this attachment is submitted to satisfy the requirements of 10 CFR 50.82(a)(10) for approval of the License Termination Plan by license amendment. CYAPCO is not, at this time, submitting its application for termination of license.

Description of Change

CYAPCO proposes to amend the license to include a provision to allow CYAPCO to make changes to the approved License Termination Plan without prior NRC approval, similar to the flexibility afforded to licensees in making changes to the facilities or procedures, as described in the UFSAR.

The change method includes four change criteria elements. Thus, CYAPCO proposes to amend its license to incorporate a new license condition, License Condition C.7 as follows:

"(7) License Termination Plan (LTP)

The License Termination Plan dated July ___ , 2000 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;

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

(c) Increases the investigation level thresholds for a given survey unit classification; or

(d) Changes the classification of a survey unit from a more restrictive classification to a less restrictive classification."

No Significant Hazards Consideration Determination

CYAPCO has reviewed the proposed change to the Operating License in accordance with the requirements of 10 CFR 50.92, "Issuance of Amendment," and concluded that the change does not involve a significant hazards consideration (SHC). The proposed change does not involve an SHC because the change would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated.*

Currently, the bounding airborne radioactivity event given in the Haddam Neck Plant UFSAR is the resin container accident. Whereas previously doses associated with gaseous waste system accidents would have bounded those associated with solid waste system failures, the small amount of radioactivity contained within the gaseous radioactive waste system with the plant in the permanently defueled condition results in this system's failure no longer being bounding. The curie content of the resin container was based on the actual radioactivity inventory collected on the resin from the reactor coolant system decontamination. This corresponded to approximately 90% of the NRC Class C burial limits. Consistent with NUREG-0782 for a resin fire, one percent of the activity of the container was assumed to be released to the environment. The 1% bounds the potential airborne release fraction from various resin incidents, such as an exothermic reaction during dewatering, dropping of a high integrity container, or a resin spill. Other airborne particulate radwaste or radioactive material accidents considered in the UFSAR but bounded by the resin container fire are as follows:

- a fire in the radwaste storage facility,
- a drop of a component (e.g., steam generator, reactor vessel, or heat exchanger) being removed from the site,

- a van of radioactive waste materials consumed by a fire while stored in the yard area on-site,
- a radiological HEPA filter rupture,
- segmentation of components or structures during loss of local engineering controls,
- an oxyacetylene tank explosion, or
- an explosion of liquid propane gas leaked from a front-end loader.

The UFSAR also discusses a fuel handling accident in the fuel building, involving the drop of a spent fuel assembly onto the fuel racks. The postulated drop assumes the rupture of all fuel rods in the associated assembly. The probability or consequences of this accident would not be increased during any future fuel transfer operations in the spent fuel pool related to decommissioning. Transfer of the spent fuel to canisters for dry cask storage will involve additional restrictions contained in the cask certificate of compliance in order to maintain decommissioning activities within the assumption and consequences of the fuel handling accident.

The requested license amendment is consistent with plant activities described in the Post Shutdown Decommissioning Activities Report (PSDAR) and the HNP Decommissioning UFSAR. Accordingly, no systems, structures, or components that could initiate the previously evaluated accidents or are required to mitigate these accident are adversely affected by this proposed change. Therefore, the proposed change does not involve an increase in the probability or consequences of any previously evaluated accident.

2. *Create the possibility of a new or different kind of accident from any accident previously evaluated.*

Accident analyses related to decommissioning activities are addressed in the UFSAR. The requested license amendment is consistent with the plant activities described in the HNP Decommissioning UFSAR and the PSDAR. Thus, the proposed change does not affect plant systems, structures, or components in a way not previously evaluated. No new failure mechanisms will be created by this activity, and the proposed activity does not create the possibility of a new or different kind of accident than those previously evaluated.

3. *Involve a significant reduction in a margin of safety.*

The License Termination Plan (LTP) is a plan for demonstrating compliance with the radiological criteria for license termination as provided in 10CFR20.1402. The margin of safety defined in the statements of consideration for the final rule on the Radiological Criteria for License Termination is described as the margin between the 100 mrem/yr public dose limit established in 10CFR20.1301 for licensed operation and the 25 mrem/yr dose limit to the average member of the critical group at a site considered acceptable for unrestricted use (one of the criteria of 10CFR20.1402). This margin of safety accounts for the potential effect of multiple sources of radiation exposure to the critical group. Since the License Termination Plan was designed to comply with the radiological criteria for license termination for unrestricted use, the LTP supports this margin of safety.

In addition, the LTP provides the methodologies and criteria that will be used to perform remediation activities of residual radioactivity to demonstrate compliance with the ALARA criterion of 10CFR20.1402.

Additionally, the LTP was designed with recognition that (a) the methods in MARSSIM (Multi-Agency Radiation Survey and Site Investigation Manual) and (b) the building surface contamination levels are not directly applicable to use with complex nonstructural components. Therefore, the LTP states that nonstructural components remaining in buildings (e.g., pumps, heat exchangers, etc.) will be evaluated against the criteria of RG 1.86 to determine if the components can be released for unrestricted use. The LTP also states that materials, surveyed and evaluated as a part of normal decommissioning activities and prior to implementation of the final status survey, will be surveyed for release using current site procedures to demonstrate compliance with the "no detectable" criteria. Such materials that do not pass these criteria will be controlled as contaminated.

Also, as previously discussed, the bounding accident for decommissioning is the resin container accident. Since the bounding decommissioning accident results in more airborne radioactivity than can be released from other decommissioning events, the margin of safety associated with the consequences of decommissioning accidents is not reduced by this activity.

Thus, the proposed change does not involve a significant reduction in the margin of safety.

<u>Conclusion</u>

Thus, the HNP License Termination Plan does not involve a significant hazards consideration as defined in 10CFR50.92.

Environmental Impact Considerations

This amendment request satisfies the criteria specified in 10CFR51.22(c)(9) for a categorical exclusion from the requirements to perform an environmental assessment or to prepare an environmental impact statement. The criteria of 10CFR51.22(c)(9) are addressed as follows:

(i) The amendment involves no significant hazards consideration.

As discussed in the "No Significant Hazards" Section above, this activity does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of effluents that may be released offsite.

The proposed license amendment is consistent with the plant activities described in the HNP Operating License. The environmental impacts associated with doses to members of the public related to decommissioning activities and site release for unrestricted use were considered in NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," and NUREG-1496, "Generic Environmental Impact Statement in Support of the Rulemaking on Radiological Criteria for License Termination." In support of the HNP PSDAR, CYAPCO performed an environmental review of site-specific decommissioning activities. The PSDAR concluded that the environmental impacts associated with HNP site-specific decommissioning activities will be bounded by previously issued environmental impact statements issued for the HNP Operating License or issued generically by the NRC. The release of effluents from the plant will continue to be controlled by plant procedures throughout the decommissioning, and the activities at HNP will continue to be performed in accordance with the HNP Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM). In addition, because of the decay of short-lived radionuclides, the number of nuclides which could potentially be released in effluents has decreased.

Additionally, the LTP was designed with recognition that (a) the methods in MARSSIM (Multi-Agency Radiation Survey and Site Investigation Manual) and (b) the building surface contamination levels are not directly applicable to use with complex nonstructural components. Therefore, the LTP states that nonstructural components remaining in buildings (e.g., pumps, heat exchangers, etc.) will be evaluated against the criteria of RG 1.86 to determine if the components can be released for unrestricted use. The LTP also states that materials, surveyed and evaluated as a part of normal decommissioning activities and prior to implementation of the final status survey, will be surveyed for release using current site procedures to demonstrate compliance with the "no detectable" criteria. Such materials that do not meet these criteria will be controlled as contaminated.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

In support of the HNP PSDAR, CYAPCO performed an environmental review of site-specific decommissioning activities. Information from that environmental review was included in a HNP Supplement to the Environmental Report that was prepared and submitted in conjunction with the PSDAR. By reference, that HNP Supplement to the Environmental Report is included in the License Termination Plan (LTP).

As discussed in Chapter 8 of the LTP, the total occupational radiation exposure (excluding public and transportation dose) impact for the proposed decommissioning activities (including completed decontamination of the Reactor Coolant System) has been estimated in the PSDAR at approximately 935 person-rem, which is less than the 1,115 person-rem exposure limits for a PWR from NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," (FGEIS) for a PWR. This estimate is primarily based on January 1997 plant dose rate surveys with no credit for (1) decay in place of radionuclides (such as Co-60), (2) sequenced removal of higher dose components first, (3) aggressive ALARA program initiatives, (4) increased worker efficiency with experience, or (5) smaller scale decontamination activities.

As discussed in Chapter 8 of the LTP, the total occupational radiation exposure due to transportation of radioactive waste has been estimated in the PSDAR at approximately 61 person-rem, which is less than the 100 person-rem exposure limits of the FGEIS. The estimated radiation exposure to the general public due to transportation is approximately 11 person-rem, which is less than the corresponding FGEIS value of 21 person-rem.

For each defined survey area and building, the LTP also describes the process and modeling to demonstrate compliance with the radiological criteria of 10CFR20.1402 for unrestricted future use of the HNP site. These criteria are: (a) Residual radioactivity that is distinguishable from background radiation results in a TEDE to an average member of the critical group that does not exceed 25 mrem/year and (b) Residual radioactivity levels are consistent with ALARA (as low as reasonably achievable) criteria defined by the LTP.

The LTP includes the requirement that, prior to the demolition of each defined building, the building must meet the unrestricted release criteria of 10CFR20.1402. Consequently, the resulting radiation exposures from either the demolition of the building and/or the use of the resulting debris will conform to the requirements of 10CFR20.1402.

The foregoing discussions demonstrate that there is no significant increase in individual or cumulative occupational radiation exposure.

Conclusion

Thus, this activity satisfies the criteria provided in 10CFR51.22(c)(9) for categorical exclusion from the requirements of an environmental impact statement or environmental assessment.

Attachment 2

(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 ___ , 2000 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;

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

(c) Increases the investigation level thresholds for a given survey unit classification; or

(d) Changes the classification of a survey unit from a more restrictive classification to a less restrictive classification.

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

Attachment 3

Connecticut Yankee Atomic Power Company

Date of Distribution: 7-13-00

Notice of Receipt of Plant Procedure **License Terminiation Plan (LTP)**

Change No.: 00-01

To: NRC HQ Office

Copy No.: COPY # 195

Please revise your controlled copy per instructions below:

Initial distribution of License Terministration Plan (LTP)

ATTACH:

REMOVE:

REPLACE

I have read and am aware of the provisions of the above listed documents.This acknowledges receipt of the revisions listed above. In addition, all superseded pages have been removed and destroyed.

Signature: _____ Date: _____

Please Return This Sheet to the Administrative Office, Connecticut Yankee Within Thirty (30) Days.

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Haddam Neck Plant
License Termination Plan

Connecticut Yankee Atomic Power Company

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No figures.

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1 GENERAL INFORMATION

1.1 Purpose

The objective for decommissioning the Haddam Neck Plant (HNP) site is to reduce residual radioactivity to levels that permit release of the site for unrestricted use and for termination of the 10CFR50 license. The purpose of this HNP License Termination Plan (LTP) is to satisfy the requirements of 10CFR50.82, "Termination of License" (Reference 1-1) using the guidance provided in Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" (Reference 1-2) and Draft Regulatory Guide-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination." (Reference 1-3)

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

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

Each section of the LTP is summarized in Section 1.3.

1.2 Historical Background

The HNP is located on the east bank of the Connecticut River, approximately 21 miles south-southeast of Hartford, at 362 Injun Hollow Road, Haddam, Middlesex County, Connecticut. HNP is owned by Connecticut Yankee Atomic Power Company (CYAPCO). (Reference 1-4)

HNP, Docket No. 50-213 (License No. DPR-61), began commercial operation in January 1968. The plant incorporated a 4-loop closed-cycle pressurized water type nuclear steam supply system (NSSS); a turbine generator and electrical systems; engineered safety features; radioactive waste systems; fuel handling systems; instrumentation and control systems; the necessary auxiliaries; and structures to house plant systems and other onsite facilities. HNP was designed to produce 1,825 MW of thermal power and 590 MW of gross electrical power. (Reference 1-5)

On December 4, 1996, HNP permanently shut down after approximately 28 years of operation. On December 5, 1996, CYAPCO notified the Nuclear Regulatory Commission (NRC) of the permanent cessation of operations at the HNP and the permanent removal of all fuel assemblies from the Reactor Pressure Vessel and their placement in the Spent Fuel Pool (Reference 1-6). Following the cessation of operations, CYAPCO began to decommission the HNP. The Post Shutdown Decommissioning Activities Report (PSDAR) was submitted, in accordance with 10CFR50.82 (a)(4), on August 22, 1997 (Reference 1-7), and was accepted by the NRC (Reference 1-8). On January 26, 1998, CYAPCO transmitted an Updated Final Safety Analysis Report to reflect the plant's permanent shutdown status (Reference 1-9), and on June 30, 1998, the NRC amended the HNP Facility Operating License to reflect

this plant condition (Reference 1-10). On October 19, 1999, the Operating License was amended to reflect the decommissioning status of the plant and long-term storage of the spent fuel in the spent fuel pool. (Reference 1-11) Additional licensing basis documents were also revised and submitted to reflect long-term fuel storage in the spent fuel pool (Defueled Emergency Plan, Security Plan, QA program, and Operator Training Program).

In April of 1999, CYAPCO contracted Bechtel Power Corporation, as the decommissioning operations contractor (DOC), to perform the decommissioning activities at HNP. CYAPCO continues to perform Spent Fuel Pool Island Operations and provides oversight of the activities performed by the DOC.

1.3 Plan Summary

Termination of the NRC license and environmental closure of the HNP site are closely related activities, completion of which will allow the site to be released for future use. The License Termination Plan describes the processes to be used in meeting the requirements for terminating the NRC license. A Site Environmental Closure Plan is also being prepared to describe the processes to be used for non-radiological cleanup and release of the site. This information will be submitted to the appropriate regulatory agencies. An integrated approach to site release processes will be used to the extent practicable.

1.3.1 General Information

This LTP has been prepared for the HNP in accordance with the requirements of 10CFR50.82(a)(9). The LTP is being maintained as a supplement to the HNP Updated Final Safety Analysis Report to support the application for a license amendment to meet 10CFR50.82(a)(9) and 10CFR50.90. Each of the sections required by 10CFR50.82(a)(9) are outlined in the subsections below. Note that figures are located at the end of the corresponding section.

1.3.2 Site Characterization

Section 2 discusses site characterization activities. The site characterization for HNP includes the results of surveys and evaluations conducted to determine the extent and nature of the contamination at the site. The initial characterization, performed in accordance with the guidelines of the "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," (Reference 1-12) began in 1997 and was completed in 1999. This initial characterization included a Historical Site Assessment (HSA), a review of historical documents, and measurements, samples, and analyses to further define the current conditions of the site. The effort also evaluated hazardous and state-regulated non-radioactive materials at the site that may require remediation and disposal.

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

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

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

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

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

3. To determine the initial classification of each survey area or unit as non-impacted or impacted Class 1, 2, or 3 as defined in MARSSIM;

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

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

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

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

As a result of the HSA and site characterization, approximately 93 acres of the plant site have been initially identified as "non impacted" as defined in MARSSIM. For those portions of the site that have been identified to be impacted, 53% of the survey areas have been initially identified as Class 1, 27% of the survey areas have been initially identified as Class 2, and 20% of the survey areas have been initially identified as Class 3. Section 2.3.3.2 defines these classification levels. Table 2-6 provides the area classifications for the various survey areas of the HNP site. The results of the surveys are being used to identify areas of the site that require decontamination, as well as to identify the cleanup methods and plan for their associated costs.

1.3.3 Identification of Remaining Site Dismantlement Activities

CYAPCO has begun decontamination and dismantlement activities at the HNP site consistent with activities discussed in the HNP PSDAR. Section 3 of the LTP describes those dismantlement and decontamination activities that remain at the HNP as of May 2000. Also included in this section are estimates of radiation dose to workers from decommissioning activities and projected volumes of radioactive waste.

CYAPCO's primary goals are to decommission the HNP safely and to maintain the safe storage of spent fuel. To the extent practical, impacted facility materials and surfaces will be decontaminated to allow beneficial reuse. Materials that cannot be decontaminated will be sent to an offsite radioactive waste processor to recycle or to a low-level waste disposal site. Completion of decommissioning the HNP site depends on the availability of low-level waste disposal sites. Currently, HNP has access to low-level waste disposal facilities in Barnwell, South Carolina, and in South Clive, Utah.

Spent Fuel Pool Island

One of the significant activities that CYAPCO has performed is the creation of the Spent Fuel Pool Island. This involved separating the systems and components required to support storage of spent fuel in the Spent Fuel Pool from systems that no longer support current and planned decommissioning activities. This minimizes the effects that decommissioning activities have on safe spent fuel storage.

Future Decommissioning Activities and Tasks

The remaining decontamination activities can be placed into several classifications which may be performed concurrently. These include major component removal (e.g., steam generators and reactor pressure vessel), contaminated system removal, clean systems removal, decontamination of site buildings and cleanup of the site land areas.

Decontamination of plant structures can occur at the same time as equipment removal. Decontamination techniques may range from water washing to removal of a layer of building surface material. Contaminated equipment and structural material may be packaged and either shipped to a processing facility, or shipped directly to a low level radioactive waste disposal facility.

Decontamination and dismantlement activities are currently expected to continue until Spring of 2003, with the exception of those activities related to the operation of the Spent Fuel Pool Island. The spent fuel may continue to be stored in the existing Spent Fuel Pool or may be transferred to an independent spent fuel storage installation (ISFSI). Transfer of spent fuel from the Spent Fuel Building to dry storage casks at the onsite ISFSI is planned to begin in 2003 and to be completed in 2004. Following the removal or decontamination of contaminated systems, components, and structures in an area, a comprehensive final radiation survey will be completed. This survey will confirm that the site meets the cleanup criteria. The final status survey results will be compiled in a series of reports by area(s) and will be made available for NRC inspection. Following completion of the final status survey and in the absence of any NRC inspection action finding the report deficient, buildings may be demolished and the concrete debris used on site as backfill.

1.3.4 Site Remediation Plans

Section 4 of the LTP describes various methods that can be used during HNP decommissioning to reduce the levels of radioactivity to that which meet the NRC radiological release criteria (i.e., does not exceed 25 mrem/yr total effective dose equivalent (TEDE) and is as low as reasonably achievable (ALARA). This section describes the methodology that will be used to demonstrate that the residual radioactivity has been reduced to a level that is ALARA in compliance with the NRC requirements.

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

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

1.3.5 Final Status Survey Plan

The primary objectives of the final status survey are to:

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

The purpose of the Final Status Survey Plan is to describe the methods to be used in planning, designing, conducting, and evaluating final status surveys at the HNP site to demonstrate that the site meets the NRC's radiological criteria for unrestricted use. Section 5 of the LTP describes the Final Status Survey plan which is consistent with the guidelines of MARSSIM. The HNP survey plan allows for the use of advanced technologies as long as the survey quality is equal to or better than traditional methods described in MARSSIM. Since MARSSIM is not readily applicable to complex nonstructural components within buildings, the criteria in Regulatory Guide 1.86 (RG 1.86), "Termination of Operating Licenses for Nuclear Reactors" (Reference 1-13), will be applied to nonstructural components and systems at the time of final status survey. Components meeting the criteria in RG 1.86 may be released for unrestricted use after completion of the final status survey for the corresponding area. The plan also describes methods and techniques used to implement isolation controls to prevent recontaminating remediated areas. The HNP Final Status Survey Plan incorporates measures to ensure that final survey activities are planned and communicated to regulatory agencies to allow the scheduling of inspection activities by these agencies if so desired.

1.3.6 Compliance with the Radiological Criteria for License Termination

Section 6 together with Section 5, Final Status Survey Plan, describes the process to demonstrate compliance with the radiological criteria of 10CFR20.1402 (Reference 1-14) for unrestricted future use for the HNP site. CYAPCO has selected the Residual Radiation (RESRAD) computer code to model dose from soils and ground water, and its counterpart, RESRAD-BUILD, to model dose from structures.

Two primary scenarios have been selected as input to the RESRAD codes for calculating the radionuclide-specific derived concentration guideline levels (DCGLs). DCGLs are the concentration and surface radioactivity limits that will be the basis for performing the final status survey. These models are the resident farmer scenario for site soils and ground water, and the building occupancy scenario for site buildings. Since concrete buildings may be demolished after acceptance of the final status surveys, two future potential uses of concrete debris have been evaluated to ensure that the reuse is adequately bounded by doses calculated in the LTP. The first evaluation considered the use of concrete debris as backfill on site. This evaluation uses the resident farmer scenario to calculate impacts from the concrete including the conservative assumption that future drinking water originates in a well located in the buried debris. The second considers future excavation and reuse of the concrete debris. The results of these two additional scenarios have been analyzed to ensure the most limiting radionuclide-specific DCGLs are used to calculate operational DCGLs for building surface surveys.

It is recognized that the methods in MARSSIM and the building surface DCGLs are not directly applicable to use with complex nonstructural components. Therefore, nonstructural components remaining in buildings (e.g., pumps, heat exchangers, etc.) will be evaluated against the criteria of

RG 1.86 to determine if the components can be released for unrestricted use. Materials, surveyed and evaluated as a part of normal decommissioning activities and prior to the implementation of the final status survey, will be surveyed for release using current site procedures to demonstrate compliance with the "no detectable" criteria. Materials which do not pass this criteria will be controlled as radioactive materials.

1.3.7 Update of Site-Specific Decommissioning Costs

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

1.3.8 Supplement to the Environmental Report

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

- the PSDAR,
- the previously issued environmental assessment,
- the environmental impact statement,
- NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities (FGEIS)" (Reference 1-15), and
- NUREG-1496, "Generic Environmental Impact Statement in Support Rulemaking for Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities." (Reference 1-16).

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

As discussed in Section 6, the DCGLs for site buildings are calculated using the building occupancy scenario as the primary modeling scenario. Because use of the demolished, decontaminated buildings as backfill is being considered, additional modeling scenarios have been considered as discussed in Section 6.5 (i.e., resident farmer for concrete debris and excavator). Buildings which are decontaminated at or below the DCGLs could be allowed to remain standing after the final status survey. These buildings could then be demolished and the debris dispositioned in a number of different manners. Consideration of the building occupancy scenario (as well as other scenarios) in determining the DCGL is compatible with the information in SECY 00-41 (Reference 1-17). SECY 00-41 concluded that the building occupancy and resident farmer scenarios, as well as assumptions used in the FGEIS to estimate public dose, are sufficiently conservative to bound such a condition. Section 8 also provides a summary description of the process CYAPCO will use to ensure that the non-radiological aspects of decommissioning meet state and federal requirements for release of the site.

1.4 Decommissioning Approach

1.4.1 Overview

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

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

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

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

Remediation of some site structures and soils will be performed, based upon the input of the initial site characterization and the cleanup levels determined by dose modeling. Title 10 of the CFR, Section 20.1402 has a dual criteria, namely 25 mrem/yr TEDE and ALARA. Accordingly additional remediation activities are evaluated to determine the cost/benefit of remediation beyond that which is necessary to meet the DCGLs. If the additional remediation activities are determined to be appropriate, they will be performed. Once areas have been remediated to the required level, administrative controls will be put into place to prevent recontamination of the areas. (Section 4)

The Final Status Survey Plan (Section 5) describes the methodology by which plant areas and buildings will be verified to be at or below the DCGLs, and thus meet the site release criteria for unrestricted use. Once final status surveys are performed for a specific area or building, the data collected will be documented in a report and made available to the NRC as evidence of completion of activities and acceptability of the area for unrestricted release. CYAPCO plans to communicate the schedules for these final status surveys, to the NRC so that independent confirmatory surveys can be scheduled and performed, as necessary.

CYAPCO may pursue demolition activities once final status survey results for an area or group of areas are completed. The final status survey results will be compiled in a series of reports by area(s) and will be made available for NRC inspection. CYAPCO may choose to demolish the surveyed structure(s) and use the concrete for onsite fill. It is important to note that CYAPCO plans to demonstrate that buildings meet the criteria for release for unrestricted use prior to their demolition and use as backfill on the site. The dose modeling approach described in Section 6 evaluates potential exposures resulting from the concrete debris to ensure that the doses are bounded by the conservative DCGLs specified in the plan. CYAPCO

does not intend to use on-site burial, disposal or incineration of any low-level radioactive waste. Materials remaining onsite will meet the appropriate DCGLs for unrestricted release, and thus is not low-level radioactive waste.

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

1.4.2 Phased Release Approach

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

1. Following completion of decommissioning activities and final status survey of a survey unit, CYAPCO will compile a final status survey report to address the area or building, where decommissioning and remediation tasks are complete and the criteria of 10CFR20.1402 has been met. The results of these surveys are documented in a report and made available to the NRC for its inspection. The final status survey report will contain the following:

    - a description and location of the building or the area to be released;
    - certification that dismantlement/decommissioning activities, as described in the LTP, have been completed for the subject building or area;
    - an evaluation of the potential for possible recontamination of the area and a description of controls in place to prevent such recontamination;
    - final status survey results for the building or area, as demonstration of compliance with the LTP release criteria; and
    - expected timing for removing the area from the 10CFR50 license.

2. Once a building or area has been verified ready for release, no additional surveys or decontamination of the subject building or area will be required unless administrative controls to prevent recontamination are known or suspected to have been compromised. Following completion of the final status survey and in the absence of any NRC inspection action finding the report deficient, the subject building may be demolished and the associated debris dispositioned as construction debris in accordance with state and federal requirements. The subject area may be removed from the license either before or after demolition activities.

3. CYAPCO will review and assess the impacts on the following documents before releasing an area from the license:

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

- Security Plan;
- Post Shutdown Decommissioning Activities Report;
- 10CFR100 Siting Criteria; and
- Environmental Report.

The reviews will include the impacts on the discharge of effluents and the limits of 10CFR 20, as they pertain to the public. After any impacts have been resolved, CYAPCO will inform the NRC of its intent to remove the subject area from the license.

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

1.5 License Termination Plan Change Process

CYAPCO is submitting this License Termination Plan as a supplement to the Updated Final Safety Analysis Report. Accordingly, the License Termination Plan will be updated in accordance with 10CFR50.71(e). Once the LTP has been approved, the following change criteria will be used, in addition to those criteria specified in 10CFR50.59, 10CFR50.82(a)(6), and 10CFR50.82(a)(7). Changes to the LTP require NRC approval prior to being implemented, if the change:

(a) Increases the radionuclide-specific derived concentration guideline levels (as discussed in Section 6) or area factors (as discussed in Section 5.4.6);

(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);

(c) Increases the investigation level thresholds for a given survey unit classification (as given Table 5-10); or

(d) Changes the classification of a survey unit from a more restrictive classification to a less restrictive classification (as discussed in Section 5.4.2).

This change process will be reflected in the application for the proposed license amendment accompanying the LTP.

1.6 References

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

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

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

1-4 Haddam Neck Facility Operating License (DPR-61) issued December 27, 1974, as amended December 14, 1999.

1-5 Haddam Neck Updated Final Safety Analysis Report, dated August 4, 1998.

1-6 Letter B16066 from CYAPCO to the USNRC, "Haddam Neck Plant Certifications of Permanent Cessation of Power Operation and that Fuel Has Been Permanently Removed from the Reactor," dated December 5, 1996.

1-7 Letter CY-97-075 from CYAPCO to the USNRC, "Haddam Neck Plant Post Shutdown Decommissioning Activities Report," dated August 22, 1997.

1-8 USNRC Memorandum from Fairtile to Weiss dated January 28, 1998, regarding CYAPCO Post-Shutdown Decommissioning Activities Report.

1-9 Letter CY-98-005 from CYAPCO to the USNRC, "Decommissioning Updated Final Safety Analysis Report," dated January 26, 1998.

1-10 USNRC Safety Evaluation, related to Amendment No. 193 to Facility Operating License No. DPR-61, Connecticut Yankee Atomic Power Company, Connecticut Yankee Atomic Power Station, Docket 50-213, dated June 30, 1998.

1-11 USNRC Safety Evaluation, related to Amendment No. 195 to Facility Operating License No. DPR-61, Connecticut Yankee Atomic Power Company, Connecticut Yankee Atomic Power Station, Docket 50-213, dated October 19, 1999.

1-12 NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual)," dated December 1997.

1-13 Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," June 1974.

1-14 Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use."

1-15 NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated August 1988.

1-16 NUREG-1496, "Generic Environmental Impact Statement in Support Rulemaking for Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities," dated

1-17 SECY 00-41, "Use of Rubblized Concrete Dismantlement to Address 10CFR Part 20, Subpart E, Radiological Criteria for License Termination," February 14, 2000.

2 SITE CHARACTERIZATION

2.1 Introduction

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

The objectives of the characterization program were:

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

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

3. To determine the initial classification of each survey area or unit as non-impacted or impacted Class 1, 2, or 3, as defined in Section 2.2 of MARSSIM;

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

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

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

Characterization efforts at the HNP decommissioning project will be an iterative process spanning all aspects of the remediation activities. The information developed during the initial HNP characterization program represents a radiological and hazardous material assessment based on the knowledge and data available at the end of 1999. This information was sufficient to satisfy the objectives listed above. Additional measurements and samples obtained during the remediation process will continue to be assessed to ensure adequacy of area classification and effectiveness of the Final Status Survey to show

compliance with the established derived concentration guideline levels (DCGLs), in accordance with the guidelines of MARSSIM.

2.2 Historical Site Assessment

2.2.1 Introduction

The HSA for the HNP commenced in 1998, under the direction of the CYAPCO Radiation Protection Department staff. The process for conducting the HSA was established in accordance with MARSSIM guidelines. The HSA focused on historical events and routine operational processes that resulted in contamination of the plant systems, onsite buildings, exterior grounds and subsurface areas within the Radiologically Controlled Area (RCA); and grounds and subsurface areas external to the RCA, but within the owner controlled area. The HSA, as part of the initial characterization program, was conducted to support the objectives detailed in Section 2.1.

In 1999, the HSA process became the task of Bechtel Power Corporation, as the Decommissioning Operations Contractor (DOC). The HSA was completed in the fall of 1999. The HSA and Initial Characterization Reports were issued in January of 2000 (Reference 2-2). Section 2 of the License Termination Plan provides a summary of findings from the HSA and the information that is the basis for area classifications, input into the development of DCGLs, development of remediation plans, and design of the Final Status Survey. The scope of the HSA included potential contamination from radioactive materials, hazardous materials, and state-regulated materials.

2.2.2 Methodology

The HSA was designed to evaluate input from two separate sources — plant records and personnel interviews. The review of plant records consisted of routine radioactive effluent release reports, non-routine reports submitted to the NRC under provisions of the technical specifications, 10CFR20, or 10CFR50; plant incident reports or condition reports; and findings documented in accordance with other assessment processes such as the Quality Assurance Program (QAP) and oversight activities. The information obtained through this process forms the input data for the records that are maintained on site to satisfy the requirements of 10CFR50.75(g)(1). The objective of the document reviews was to identify events that caused the contamination of systems, buildings, external surfaces, subsurface areas, or waterways, via atmospheric releases, liquid releases, or release of solid radioactive material. For each event, available supporting documentation regarding event description, facility and system design, radiological surveys and analysis, remediation efforts, and post remediation surveys was collected and reviewed. The CYAPCO nuclear records management system was the primary source of plant record information gathered during the HSA process.

To facilitate correlation of the impact of an event to physical locations on the plant site and to provide a means to correlate subsequent survey data, the owner-controlled area has been divided into areas with numeric designations. Figures 2-1 through 2-19 provide the area identification numbers for buildings and grounds within the owner controlled area. The area designations form the basis for survey units presented in Table 2-6.

In addition to the review of plant records, interviews with individuals involved in nuclear operations at HNP were conducted. Personnel interviewed included selected present and former employees and contractors involved in operations, maintenance, and radiation protection activities at the site. Information regarding unplanned releases or other events that could have resulted in site contamination was obtained through site staff and all-hands meetings, the daily plant newsletter, and Northeast Utilities system-wide

publications. The effort was designed to ensure that historical events were identified that had an impact on the radiological or hazardous material status of the site. Information gathered from the interviews was reviewed and included as appropriate in the 10CFR50.75(g) database.

2.2.3 Instrumentation Selection, Use, and Minimum Detectable Concentration

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

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

Table 2-1
Typical Instruments Used at HNP

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

Table 2-1
Typical Instruments Used at HNP

Instrument	Efficiency (Nominal)	MDA (dpm)	Probe Type	Detection	Units	Characteristics
Eberline PS-2-2	12 – 17%	700	GM Pancake	βγ	cpm	Emergency Plan instrument. (Digital scaler) Used also to calibrate the Gamma-10.
Eberline RM-14	> 12%	700	GM Pancake	β	cpm	Frisker, AC powered.
Ludlum 19	NA	NA	Scintillation	γ	μr/hr	Used for boundary surveys, dose rates of trash, etc.
Ludlum 2200	17% β 11%α	<1000β < 100 α	HP-210 β 43-2 α 43-1 α	βα	dpm	Smear counting. (Approx. 12 43-2 probes and 4 43-1 probes.
Bicron Electra 1B with DP6BD Probe	> 20% β	1000	dual phosphor scintillation	βα	cpm	--100 cm^2 sized probe --Detects α & β simultaneously or independently.
Bicron Electra 1B with DP6DD Probe	> 12% β 8% α	1000	dual phosphor scintillation	βα	cpm	Same as the DP6BD probe. This probe has a double mylar window.
NE Technologies SAM-9	10 – 30%	<5000	Scintillation	γ	dpm	Small article monitor.
Bicron Micro-Rem	NA	NA	Scintillation	γ	μr/hr	Used for yard surveys and truck surveys. Pulser instrument.

Table 2-1
Typical Instruments Used at HNP

Instrument	Efficiency (Nominal)	MDA (dpm)	Probe Type	Detection	Units	Characteristics
Exploranium Gamma Ray GR-130 Minispec In Situ Monitor	NA	NA	NaI	γ	selectable cpm to mr/hr	This instrument is used only for qualitative analysis and store up to 122 complete 256 channel spectra in memory. This instrument can be used in an operational mode as a survey meter and dose meter also. Data logger capability downloadable to a computer via serial link.
ISOCS – In Situ Object Counting System	40% (intrinsic)	variable	Intrinsic germanium	γ	variable	Collimator. Analyzes spectra data and generates reports.
CM-11	30-34% β 17-19% α	1000 100	DE-11A Gas Flow Proportional	$\beta\alpha$	dpm	Used mainly at control point for radon. 100 cm^2 sized probe. One is on a portable cart. Pulser instrument.
APC II Counter	22% α 35% β	variable	Gas Flow Proportional	$\beta\alpha$	Counts	low background scaler/counter
XLB-1 Counter	24% α 38% β	variable	Gas Flow Proportional	$\beta\alpha$	Counts	low background scaler/counter

2.2.4 Results

2.2.4.1 Routine Releases

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

2.2.4.2 Operational Events

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

2.2.4.2.1 Atmospheric Releases

During the initial years of operation at HNP, several events involving unplanned releases of airborne radioactivity occurred due to equipment failures or operator errors. During 1971 through 1979, these releases occurred in various areas of the plant, with the resulting radioactive discharges to the environment primarily involving inert gases and iodine. These releases were documented in station abnormal occurrence (AO) reports or plant incident reports (PIR), with the discharges included in the Semi-Annual Effluent Release Reports. As indicated by and discussed in the Site Characterization Report, these occurrences involved short-lived radioactive gases and iodines that did not result in contamination of site areas that would impact decommissioning activities. These events are listed in Table 2-2.

Table 2-2
Unplanned Gaseous Release Events

Date	CYAPCO Reference Document No.	Location of Event	Survey Area Number (Figs. 2-1 through 2-19)	Medium	Event Description
5/10/71	CYH-1686, dated 5/10/71, Letters dated 6/11/71, 6/18/71, 6/22/71, 4/27/99	Containment	9000	Gas I-131	Iodine release due to operator error with penetration seals.
5/19/72	AO 72-2	Ion Exchange Cubicles	9114	Gas Xe-133 & Xe-135	Unplanned airborne release from Demineralizer due to operator error
4/26/73	PIR 73-134	Steam Generator	3111	Gas	Primary to Secondary Leak, High RMS Alarm – Air Ejector Monitor
6/21/73	AO 73-6	Ion Exchange Cubicles	9114	Liquid/Gas	500 gallons leaked to the Aerated Drains Tank due to failed valve. Gas release.
3/30/76	LER 76-8/3-L	Waste Gas Decay Tank Cubicle	3111	Gas	Waste Gas Decay Tank rupture disk failed
9/18/77	LER 77-21/1P	"A" Waste Gas Decay Tank	2000	Gas	"A" Waste Gas Decay Tank rupture disk actuated
12/16/79	PIR 79-125	Main Stack	2402, 9120, 9302, 9307, 9310, 9312, 9514, 9527	Particulate, Stack Release	Degassifier rupture disk release of primary coolant to main stack

Atmospheric releases that have impacted the site area include a series of events that occurred in 1979. Stack discharges involving particulate activity occurred in February and December of 1979 as a result of the failure of a level controller in the letdown degassifier, flooding of the degassifier, and actuation of the degassifier rupture disk. The letdown liquid (primary coolant) then overflowed the degassifier. The discharge line from the rupture disk was routed directly to the main discharge stack. Efforts to clean the stack following the first incident may also have resulted in particulate releases. Surveys identified a number of radioactive particles within the Radiologically Controlled Area, within the fenced area of the plant site, and in the parking lot and hillside east of the plant. The results of the site surveys are documented in Reference 2-3. The majority of the radioactive particles were found on the roof areas of buildings close to the stack, and on the ground within the Radiologically Controlled Area. The extent of area outside the Radiologically Controlled Area impacted by the releases includes the parking lot north of the industrial area, the hillside east of the plant out to 200 meters, and areas adjacent to the discharge canal.

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

2.2.4.2.2 Liquid Releases

The HSA identified a number of leaks and unplanned liquid releases that have occurred during the operational lifetime of HNP. The majority of the occurrences were confined to the Radiologically Controlled Area. The leaks and unplanned releases were associated with equipment failures and operational events associated with components within the Containment Building, Primary Auxiliary Building, outside storage tanks, and the radioactive waste processing systems. The most significant unplanned liquid release occurred in 1984, the result of a failure of the reactor cavity seal. The resultant release of approximately 200,000 gallons of water was contained in the lower level of the Containment Building and subsequently pumped to the RWST. No radioactivity was detected in any liquid pathway outside containment as a result of that event. A summary of events involving radioactive liquid leaks and unplanned releases is presented in Table 2-3. A record of unplanned liquid releases is maintained on site in accordance with the requirements of 10CFR50.75(g) and also identified in the affected area assessment in the Haddam Neck Plant characterization report (Reference 2-2).

The principal impact of these events is to the ground within the Radiologically Controlled Area (RCA). Migration of radioactivity to subsurface soils has occurred in the area of the Primary Auxiliary Building and the Refueling Water Storage Tank, adjacent to the Containment Building. Radioactivity has also been detected in ground water wells on site. Assessments of radioactivity in soils and ground water are further discussed in Sections 2.3.3.1.3, 2.3.3.1.4, 2.3.3.1.5. Radioactive materials from leaks also may have impacted an area in the southeast corner of the protected area, a leachfield south of the protected area (but within the owner controlled area), and drain systems leading from the RCA. All of these areas were within the owner controlled area.

Table 2-3
Unplanned Liquid Release Events

Date	CYAPCO Reference Document No.	Location of Event	Survey Area Number(s) (Figs. 2-1 through 2-19)	Medium	Event Description
11/1/73	AO 73-11	RWST	9110, 2110	Liquid	Valve Leak, 270 liters of liquid released to storm drain
1/28/76	PIR 76-15	"A" Recycle Test Tank	9108	Liquid	15 gal. of liquid leaked from tank to diked area
5/22/76	LER 76-13/990	PAB Below Drumming Room Floor	2228	Liquid	Leakage from drain line below floor
2/24/77	LER 77-2	"A" Recycle Test Tank	9108	Liquid	1000 gal. of radioactive water released to diked area around tank
2/23/79	PIR 79-27	Main Stack	9307	Liquid	Manway leakage following SG blowdown rupture disk actuation. 20 gal. to yard area.
3/6/79	PIR 79-38	Main Stack	9120, 9307	Liquid	Manway leakage following SG blowdown rupture disk actuation
3/28/83	PIR 83-37	Septic Tank	9520	Liquid	84 gal. of water from Chem Lab to Septic Tank
8/21/84	PIR 84-136	Containment Building	3002	Liquid	Reactor cavity seal ring failure. 200,000 gals of water drained to lower levels of Containment Building
2/24/89	PIR 89-35	Leach Field, 115 kV yard	9102, 9308	Liquid	50 gal. release from SFB floor drain, line discharges to 115 kV yard
9/14/90	PIR 90-239	RWST	9110	Liquid	6 gallon per day leak from RWST identified from inventory monitoring
8/12/91	PIR 91-149	Pipe Trench	2110	Liquid	400 gal. release from open valve to pipe trench

2.2.4.2.3 Contamination of Buildings

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

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

2.2.4.2.4 Release of Radioactive Materials

The Historical Site Assessment identified several events in which radioactive material was found outside the Radiologically Controlled Area. The primary locations of discovery were the southwest peninsula, survey area 9520, and the shooting range landfill area, survey areas 9535 - 9538. These areas are currently controlled as "Radioactive Material Areas" with restricted access. During plant operations, the peninsula area has been used for storage of materials. The materials have been typically associated with maintenance activities performed during outages or plant modifications. Documentation indicates that the radioactive material was detected in 1980, 1985, and 1989. Surveys performed in 1998, indicated some previously stored materials contained detectable radioactivity. Since the shutdown of the plant, these stored materials removed from the peninsula area have been evaluated using the same process as material leaving the Radiologically Controlled Area. Surveys for free release have been required prior to that material leaving the industrial area.

Other instances of minor levels of contamination being identified in areas outside the Radiologically Controlled Area but within the site boundary are documented in the Historical Site Assessment. The levels indicated in surveys are detectable, but typically are below the DCGLs. The areas affected were areas of high personnel traffic, such as the Administration Building or the Steam Generator Mock-up Building. Upon discovery, remediation occurred and more extensive surveys of the areas were performed. In 1998, an extensive survey was conducted of the material within those support buildings housing materials that may have, at one time, been in the plant. Any material or object with any indication of radioactive material was returned to the Radiologically Controlled Area for disposition.

Additional events have been documented, indicating the release of potentially contaminated material from the site. The material in question has been primarily construction materials such as concrete blocks and excavated soils. These materials are being addressed through the Material Recovery Program that has been ongoing since 1997.

2.3 Initial Site Characterization

2.3.1 Introduction

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

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

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

2.3.2 Methodology

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

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

- the structure or area bounded by the evaluation;
- systems contained within the area, if any;
- HSA identification of events that may have impacted the area;
- area use, present and historic;
- materials of construction;

- presence of radioactive or hazardous material storage areas;
- radiological survey information including extent of area containing radioactivity, radionuclides, exposure rates and contamination levels, both present and historic; and
- potential for migration of contaminants from contiguous areas.

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

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

- a description of the area;
- survey units (as identified Table 2-6 and shown on Figures 2-1 through 2-19) contained within the scope of the report;
- summary of radiological, hazardous, and state-regulated material conditions within the area;
- classification of the area in accordance with the categories defined by MARSSIM;
- radiologically impacted systems within the area; and
- additional sampling and analysis necessary to support remediation.

The area characterization reports have been compiled in the "Connecticut Yankee Haddam Neck Plant Characterization Report" dated January 6, 2000. The following information provides a summary of the information contained in that report.

2.3.3 Results

2.3.3.1 Radiological Status

2.3.3.1.1 Systems

An extensive review of systems was conducted to determine those systems that contain radioactive materials or in which radioactive material was detected at some time during the operating history of the plant. Systems that are identified as "affected" require additional surveys to define the extent and magnitude of radioactivity. For those systems that may have been impacted due to steam generator tube leakage or other operational events in the past, but for which subsequent samples have not identified radioactivity, the "affected" status is maintained. Table 2-4 provides a listing of plant systems and their status relative to the potential for radioactivity. The assessment considers the internal portions of the systems. Systems that might be assessed as "unaffected" and are located in contaminated areas may themselves be externally contaminated and may be considered for remediation or disposal as radioactive waste.

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

Several components, such as the gland seal and turbine casing, have been identified as "affected" based on primary-to-secondary leakage identified in operating cycles as recent as 1990. These components

contain low levels of radioactivity. The extent of the contamination, although appearing to be limited to small portions of the high pressure steam portion of the system, will be further defined as the systems are disassembled and the internal surfaces become accessible. These items are identified in the characterization reports associated with the areas containing the systems.

Table 2-4
Radiological Status of HNP Systems

System Name	Survey Area #	Contamination Potential	Comments
Auxiliary Boiler	0001	A	Identified radioactivity through sampling, 9/9/97
Auxiliary Feedwater	0002	B	
Blowdown	0003	B	
Boric Acid	0004	B	
Boron Recovery	0005	A	
Charging	0006	A	
Chemical Addition	0007	B	System consists entirely of addition tank.
Circulating Water	0008	C	
Closed Cooling Water	0009	A	Identified radioactivity through sampling, 9/3/97
Component Cooling Water	0010	A	Identified radioactivity through sampling, 9/3/97
Condensate	0011	B	
Containment Air Recirculation	0012	A	
Containment Heating	0013	A	
Containment Iodine Removal	0014	A	
Containment Leak Monitoring	0015	A	
Containment Purge	0016	A	
Containment Rod Drive Mechanism Cooling	0017	A	
Control Air	0018	B	
Demineralized Water	0019	B	Grouped system with Water Treatment for survey purposes as systems are closely aligned.
Diesel Generator	0020	C	
Domestic Water	0021	C	

A – System contains radioactive materials, known to be contaminated.
B – Radioactivity detected in portions of the system, surveys required to determine full scope of contamination
C – System has no history of radioactive contamination, no samples that have indicated detectable contamination.

Table 2-4
Radiological Status of HNP Systems

System Name	Survey Area #	Contamination Potential	Comments
Feedwater	0022	B	
Feedwater Heater, Extraction Steam, Drains, Vents	0023	B	
Fire Protection	0024	C	
Floor/Equipment Drains	0025	A	
High Pressure Safety Injection	0026	A	
High Pressure Steam Dump	0027	B	
Isolated Phase Bus Duct Cooling	0028	C	
Letdown	0029	A	
Low Pressure Safety Injection	0030	A	
Low Pressure Steam Dump	0031	B	
Main Generator Seal Oil	0032	C	
Main Steam	0033	B	
Main Turbine & Auxiliaries	0034	A	Survey data indicates areas of radioactivity
Misc. Ventilation Systems	0035	B	Includes Turbine Bldg, Cable Vault, New Diesel Gen Bldg, Aux Feed Pump Enclosure, Office Bldg and Service Bldg.
Nitrogen/Hydrogen/ Carbon Dioxide	0036	B	
PAB Ventilation	0037	A	
Post Accident Sampling	0038	A	
Primary Grade Water	0039	B	
Purification	0040	A	
Radiation Monitoring	0041	A	
RCP Seal Water Injection	0042	A	
Reactor Coolant	0043	A	
Residual Heat Removal	0044	A	
Roof Drains	0045	B	

A – System contains radioactive materials, known to be contaminated.
B – Radioactivity detected in portions of the system, surveys required to determine full scope of contamination
C – System has no history of radioactive contamination, no samples that have indicated detectable contamination.

Table 2-4
Radiological Status of HNP Systems

System Name	Survey Area #	Contamination Potential	Comments
Sampling	0046	A	
Septic	0047	B	
Service Air	0048	B	
Service Water	0049	B	
Spent Fuel Building Exhaust	0050	A	
Spent Fuel Pool Cooling	0051	A	
Storm Drains	0052	B	
Turbine Lube/Control Oil	0053	C	
Vacuum Priming	0054	C	
Waste Disposal	0055	A	
Waste Gas	0056	A	
Waste Water Treatment	0057	B	
Water Treatment	0058	C	

A – System contains radioactive materials, known to be contaminated.
B – Radioactivity detected in portions of the system, surveys required to determine full scope of contamination
C – System has no history of radioactive contamination, no samples that have indicated detectable contamination.

2.3.3.1.2 Buildings

Radiological surveillance performed routinely at the HNP is designed to ensure compliance with 10CFR20 requirements regarding posting of areas, and to provide a basis for establishing controls for the safety of workers involved in those areas. The radiological information provides the basis for the remediation program in each of the buildings at the site. Once the initial remediation processes have adequately reduced the ambient radiation levels, fixed surface contamination surveys will be conducted to assist in the design of the building remediation efforts.

The extent and nature of radioactive material in the primary structures on site are discussed in the following paragraphs.

2.3.3.1.2.1 Turbine Building

Primary to secondary leakage has resulted in measurable radioactivity in small areas of the secondary system piping, primarily in the high-pressure steam components within the turbine building. Surveys of accessible areas of the systems have shown fixed radioactive material in levels up to approximately 10,000 dpm/100 cm^2. Isotopic analysis has identified that Cs-137 is the principal radioactive radionuclide that has carried over in the steam following primary to secondary leakage. No alpha emitting radionuclides have been identified in any surveys for either fixed or removable radioactivity.

Scoping surveys performed in the turbine building, covering more than 30,000 square feet of the surfaces of the operating floor and grade level in 1996 identified only one small area of elevated activity. That area, near the normal entrance/egress path to the Radiologically Controlled Area, was remediated at the time of the survey.

2.3.3.1.2.2 Primary Auxiliary Building

The Primary Auxiliary Building (PAB) is designed to house systems containing radioactive materials. The building is designed to contain and control leakage occurring during routine operations as well as unusual conditions. The radiological status of the building is maintained by the radiation protection department staff through surveys performed in support of daily plant activities.

With the exceptions of the service water, primary de-ionized water, control air, fire protection, nitrogen gas and service air, all of the systems within the PAB are radiologically contaminated. Contamination levels in several of these systems are such that high radiation areas exist in their vicinity. Most of the cubicles that contain major systems are posted as contaminated areas identifying removable radioactive material.

The PAB contains pipe trench and pump pit areas, where conditions in these areas include high dose rates, possibility for high airborne contamination levels and alpha-emitting radionuclides. The lower level of the PAB under the boron recovery equipment is contaminated due to past releases of evaporator bottoms. The concrete surfaces making up these pit and trench areas will require remediation efforts.

The PAB, fuel building and containment air handling systems all contain contaminated filter elements that will have to be removed and disposed of once these systems are declared abandoned.

The roof of the PAB has been radiologically impacted by historical plant events. Contamination of the PAB roof has occurred on multiple occasions due to emissions from the PAB roof exhaust and identified

problems with the exhaust ducting. In 1980, an epoxy coating was used to fix contamination identified on the PAB roof.

Historically, leaks have been found at the junction between the steam generator blowdown line and the service water discharge line beneath the floor of the PAB drumming room. On at least one occasion, a leak has resulted in contamination of the soil beneath the drumming room floor.

Based on the building design basis, events that have occurred within the building, and the present status of areas that are controlled as contaminated areas, much of the interior surfaces of the PAB are expected to contain radioactivity above the DCGL. Core bore information has identified that typical penetration depth is about 0.5 inch, however a depth of up to 2 inches has been identified in an area expected to contain contamination among the highest in the building. All of the PAB is considered a Class 1 area.

2.3.3.1.2.3 Containment Building

The Containment Building houses numerous systems containing primary coolant as well as radioactively contaminated support systems. System leakage and maintenance activities over the operating life of the plant have resulted in radiological conditions similar to the containment buildings at other pressurized water reactors of similar vintage. Radiological conditions in the Containment Building have warranted designation of some areas as Contaminated Areas (greater than 1000 dpm/100 cm^2 removable contamination) and Locked High Radiation Areas (greater than 1000 mrem/hr). The surfaces within this area are classified as potentially contaminated. Some components, equipment, structural steel and concrete have become radioactive due to neutron activation. Additional characterization for activation will be conducted as the areas become accessible.

During 1998, a site characterization study was performed for polychlorinated biphenyls (PCBs), RCRA metals, and radioactivity in paints used on the structures on primary and secondary sides of the HNP. Many paint-chip and concrete-chip samples from containment were collected and analyzed by gamma spectral analysis. The purpose of the sampling program was to determine the extent of remediation required, and the waste management requirements due to potential PCB or RCRA metals in paint used in the Containment Building. Complete results of this study are provided in Reference 2-4. A summary of the results is as follows:

- The average concentrations of radioactivity in paint on the steel liner are about 1200 pCi/g on the charging floor level and about 300 pCi/g on the grade level. The primary radionuclides are Cs-137, Co-60 and Cs-134.

- The concentrations of radioactivity in paints on equipment vary markedly in both total activity and radionuclide distribution, depending on location and use. For example, the core barrel lift rig contained approximately 25,000 pCi/g of Co-60 in paint while the polar crane contained 30 pCi/g of Co-60 in paint.

- The total radioactivity in floor paint averaged approximately 8200 pCi/g and is essentially the same from the charging floor, grade level and lower level.

- The total radioactivity in wall paint averaged approximately 490 pCi/g and is essentially the same from the charging floor, grade level and lower level.

- The radioactivity in the paint/concrete samples is greater than the radioactivity in the underlying concrete samples. The radioactivity in the paint/concrete and concrete only samples is greater on the floors than in the corresponding samples from the walls.

Based on the building design basis, and the operation history as well as the present status of areas that are controlled as contaminated areas, much of the interior surfaces of the containment building are expected to contain radioactivity above the DCGL. All of the containment building is considered a Class 1 area. Radiological conditions within this area are changing on a frequent basis due to ongoing dismantlement activities.

2.3.3.1.2.4 Radwaste Reduction Facility

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. Therefore, the potential for residual contamination exists throughout the building. The Radwaste Reduction Facility historically contained a variety of equipment such as a waste shredder/compactor previously used for waste processing. The shredder/compactor is known to be internally contaminated. This equipment is typical of equipment used throughout the life of the facility since the primary purpose was the sorting and volume reduction of radioactive material. Additionally, there is a permitted lead work booth that is radiologically contaminated and may represent a mixed waste concern. The floors and the floor drains of the facility represent the primary concerns for residual contamination. The Radwaste Reduction Facility contains no plant related process systems (such as service air, control air, etc.). The systems within the building are support systems such as electrical service and ventilation. The interior of the Radwaste Reduction Facility has been identified as a Class 1 area.

2.3.3.1.3 Radiologically Controlled Area Grounds

The Radiologically Controlled Area ground consists of paved areas around the containment building, primary auxiliary building, RWST and waste storage tanks, and the spent fuel building. The Historical Site Assessment identified events involving unplanned liquid releases that have radiologically impacted the area. Portions of the area have been posted as contaminated (removable contamination greater than 1000 dpm/100 cm^2) due to system leakage. The contents of the tanks caused radiation areas to exist. The paved areas served as the pathways for personnel movement between buildings and for vehicles moving materials, including radioactive waste. The area has also been used for temporary processing equipment in support of operations and maintenance activities during outages. Radiological surveys performed during the plant operating years have identified areas of removable contamination. For example, a temporary cask washdown areas was operated in the areas south of the Spent Fuel Building. Contaminated pavement and soil resulting from its use have been identified and remediated. Fixed contamination measurements have been limited due to the relatively high backgrounds in the area.

The area contains several drains for rainwater runoff. During several events involving leakage of radioactive materials in the area, samples of materials from the bottom of the drains identified detectable radioactivity. Details regarding those events impacting the grounds within the Radiologically Controlled Area are found in the Connecticut Yankee Haddam Neck Plant Characterization Report dated January 6, 2000. Historical surveys identified radioactive material in excess of the anticipated DCGLs. Consequently, grounds within the Radiologically Controlled Area were classified as Class 1 areas.

2.3.3.1.4 Non-Radiologically Controlled Area Grounds

The Historical Site Assessment identified that some material contaminated with radioactive material was placed in the shooting range landfill area along with construction debris. The assessment identified that between 1974 and 1996, construction materials from approximately 32 site projects had been placed in the landfill. Examples of materials identified are discharge canal dredging spoils, excavated soils,

construction debris and sand. The materials originated from areas both within and outside the Radiologically Controlled Area boundary. The landfill area had been permitted by the State of Connecticut for disposal of bulk wastes. No evidence of authorized or unauthorized burial of radioactive materials on the HNP site exists.

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

Radiological results indicated the presence of both Co-60 and Cs-137, with maximum concentrations in soil of 5.0 pCi/g and 52.9 pCi/g respectively. A study was completed to determine the extent and magnitude of the plant-derived radionuclides or hazardous materials that might have accompanied the construction debris into the landfill area. This radiological characterization consisted of:

- Gamma scan surveys centered on 9 m^2 areas;
- Exposure rate measurements at 1 m above the surface; and
- Soil samples collected for analysis from the surface, and at depths within the ranges 0-1 m, 1-2 m, and 2-3 m

The results of the survey identified 8 samples with positive Co-60 results between the minimum detectable activity of 0.15 pCi/g (environmental LLD) and a maximum of 2.8 pCi/g. Additionally, 40 samples measured positive Cs-137 results between the minimum detectable activity of 0.18 pCi/g (environmental LLD) and a maximum of 31.4 pCi/g. Only 10 of the 40 positive Cs-137 results were greater than the local background level of 1.68 pCi/g due to historical fallout from weapons testing. The gamma scan results showed a variance of a factor of 4 or 5 times the lowest levels that are indicative of background. The exposure rate measurements ranged from 9.1 to 15.6 μrem/hr. The scan and exposure rate measurements both indicate the presence of low levels of radioactive material but are not able to establish if the variance is due to plant derived radionuclides or represent variations in naturally occurring radionuclides.

A hazardous materials assessment was also performed in the survey area. Two composite samples were analyzed for the Envirocare Suite of analytes. A 10 CFR 61 series of analyses were also performed on the composite samples. The test results indicated that all the analyses were below the Connecticut Department of Environmental Protection Remediation Standards for soil except for the chemical thallium. Additional investigations will be performed. The level of thallium is above the residential limit, but below the industrial limit. Co-60 and Cs-137 were the only plant-derived radionuclides present in the composite samples.

In December 1997, surveys were performed of paved areas within the industrial area (designated as the restricted area) but outside the Radiologically Controlled Area. The surveys were conducted using a Position Sensitive Proportional Counter. The survey, performed over 7,386 square meters of surface area, consisted of approximately 3,000,000 measurements. Essentially all paved areas not restricted by trailers or snow piles were surveyed. The surveyed area is approximately 80% of the paved areas and includes all of the normally traveled areas.

The paved area survey identified 7 discrete areas of contamination ranging from 10,000 to 65,000 dpm. Each area was limited to less than 100 cm^2. Identified radioactivity was remediated prior to completion of the survey. The details of the survey are documented in Reference 2-6.

Land areas adjacent to the industrial area had been surveyed in response to the events occurring in 1979, discussed in Section 2.2.4.2.1. Subsequent surveys were performed in the land areas adjacent to the industrial area fence. The criteria for surface area coverage was to survey to a distance from the plant twice that of the point where radioactive particles were detected. Particles were detected and removed in areas close in to the industrial area fence, with few particles found beyond 100 meters. The details of the surveys and the results are presented in Reference 2-3.

2.3.3.1.5 Ground Water

As part of the site characterization efforts and to address issues related to leakage of radioactive liquids from the Refueling Water Storage Tank (RWST), a ground water monitoring sampling and analysis plan was developed in March of 1998. An investigation was conducted through May of 1999, with the primary purpose of radiological characterization of ground water in four geographically separate areas of interest. The areas of interest are the power plant area, the peninsula water supply area, the Emergency Operations Facility area, and the landfill area.

As called for in the sampling and analysis plan, monitoring wells were installed to characterize the water table and shallow bedrock aquifers near potential contaminant source areas. Two rounds of sampling, March and April of 1999, were analyzed for dissolved boron, tritium, gross alpha, gross beta and gamma spectroscopy. Tritium was quantified above detection limits (Minimum Detectable Activities of 700 to 1000 pCi/l) at 11 of 40 monitoring well locations and in the reactor building mat drain sump. The elevated tritium concentrations, detected only in those wells in the plant area, ranged from 1,180 pCi/l (located between the Spent Fuel Pool and the Containment Building), to 138,700 pCi/l (located between the Containment Building and the Primary Auxiliary Building). Cs-137 was detected in 1 of 40 monitoring well locations, again in the power plant area. No other gamma emitting radionuclides were above the minimum detectable levels. Sr-90, a pure beta emitting radionuclide, is not expected to be detectable in the ground water. The chemical behavior of strontium is similar to that of cesium, and the activity ratio of Cs-137 to Sr-90 in the plant is greater than 100 to 1. Boron, a naturally present chemical element as well as a power plant chemical additive, was found at concentrations above background at the same locations where elevated tritium was found (maximum of 9,590 ug/l) and the values roughly correlated.

Complete sampling results, along with results of geologic and hydrogeologic characterization activities, are documented in Reference 2-5. This report has been submitted to the Connecticut DEP and has been provided to the Nuclear Regulatory Commission and the Environmental Protection Agency. The primary conclusions of that report are as follows.

- The general flow of ground water is from north to south, or from hillside to river across the plant site, primarily within the fractured bedrock.
- The ground water migrates down under and around the deep foundations of the plant structures in complex patterns and then moves upward again to the river.
- Areas such as the Emergency Operations Facility, water supply well and landfill areas show no tritium above minimum detectable activity and no boron above background levels. No contamination in any of the drinking water wells has been detected.

- Ground water beneath and immediately around the power plant buildings has been affected by boron and tritium.
- The likely source of tritium and boron in the ground water is the RWST. This source was eliminated when the RWST was drained.
- More ground water data and monitoring wells are needed to identify the bottom and core of the plume.

A more detailed discussion of the extent of tritium and cesium contamination in the power plant area is provided below.

Tritium

Tritium is a radionuclide created in the reactor water as a normal by-product. Because it behaves like water, its hydrologic behavior and subsurface migration are the same as ground water – it is not subject to the physical/chemical factors affecting the movement of dissolved chemicals, such as volatilization, retardation, or chemical degradation.

Figure 2-20 shows the observed tritium concentrations and extent of tritium as reported in Reference 2-5. Based on observed concentrations and their spatial distributions, there have been three distinct sources of tritium:

- The RWST located north of the Containment Building. Tritium has been observed at both MW-103S and MW-103D. Tritium has not been observed at sampling points upgradient of the RWST (i.e., MW-101).
- A potential source near well cluster MW-105, located south of the Containment Building.
- A minor, potential source near well cluster MW-102 and MW-114S.

The tritium from the former RWST source appears to be flowing downward and towards the Containment Building, with much of it currently captured by the mat drain sump. However, due to the complexity of flow patterns around the Containment Building, this source may have contributed to other downgradient plumes.

From MW-105, a portion of the tritium is migrating in both the shallow and deep ground water horizons southeastward past MW-106 to MW-107. The direction of the tritium appears to be controlled by the presence of the discharge tunnel that blocks shallow ground water movement to the west. A downward component has carried some of the tritium into the bedrock at well MW-106D.

A second, larger portion of the tritium from the MW-105 area migrates downward underneath the discharge canal. It continues southwestward and southward towards well clusters MW-109 and MW-110, which are adjacent to the Connecticut River. The presence of local downward gradients, discharge canal and foundation walls combine to divert the plume into the bedrock under the turbine building where it eventually is carried upward by gradient toward the river, where it eventually discharges.

Cesium

Cs-137, observed only at well MW-103S, is the only other plant-related radionuclide detected in the ground water. The presence of Cs-137 coincides with the tritium findings and apparently originated from the RWST. Cs-137 concentrations in soil have been observed to decrease with depth, indicating a residual surficial source. The Cs-137 is not as mobile as tritium and should be retained in soils near the release area, which is consistent with the observed distribution. Ground water flow directions and the absence of any positive gamma spectroscopy findings at the Containment mat drain sump may indicate that the plume is not migrating towards the sump and/or is being mixed with the large amount of ground water being drawn into the sump.

Concentration Trends

Additional sampling and analysis of monitoring wells in the power plant area were performed in September 1999. Table 2-5 summarizes these results along with data from the two previous sampling rounds for key monitoring locations. As the data indicate, both H-3 and Cs-137 concentrations are generally declining with time. This trend is consistent with the removal of the source of contamination (i.e., draining of the RWST). Ground water monitoring will continue on a periodic basis during the decommissioning process to confirm this trend.

Table 2-5
Temporal Trends in H-3 and Cs-137 Ground Water Concentrations

Date	H-3 (pCi/l)				Cs-137 (pCi/l)
	MW-103S	MW-103D	MW-105S	MW-105D	MW-103S
Mar 1999	2,580	22,180	138,700	4,590	75.9
Apr 1999	9,260	17,550	67,400	2,450	32.7
Sep 1999	2,980	19,660	23,480	3,030	28.7

2.3.3.2 Initial Area Classification

The initial classifications provided in Table 2-6 were selected based on the review of a large volume of historical radiological survey data (routine and non-routine) over the plant's operating history, review of the historical information maintained under 10 CFR 50.75(g), and review of the scoping survey information compiled for the decommissioning of the HNP site. The radiological survey information in particular represents a substantial volume of information.

The initial classifications for the HNP site are presented using the existing breakdown and numbering scheme for site grounds and structures. These area definitions were initially established for decommissioning and were used and expanded upon during subsequent site characterization activities. Many areas are further subdivided into survey units. A survey unit is a physical area consisting of structures or land areas of specified size and shape for which a separate decision will be made as to whether or not residual contamination in that area exceeds the release criterion. A survey unit does not include areas that have different classifications, and its characteristics should be generally consistent with the exposure pathway modeling that is used to establish its corresponding DCGLs. The survey areas identified in Table 2-6 have been established based on logical physical boundaries and site landmarks for the purpose of documenting and conveying initial characterization information for the HNP site. The survey areas are sub-divided into survey units to the extent necessary in the final status survey plan, though it is expected that in general the survey areas will meet the requirements for survey units with little

or no modification required. Note that survey areas for open land or yard areas are implied to include any sub-surface features that are present such as piping and drain systems. The survey areas are depicted in the maps presented as Figures 2-1 through 2-19. The classifications provided apply to all survey units within the survey area.

Classification is the process by which an area or survey unit is described according to its radiological characteristics. The significance of classification is that this process determines the final status survey design and the procedures used to develop this design. In classifying areas, those that have no reasonable potential for residual contamination are classified as non-impacted areas. These areas have no radiological impact from site operations and are typically identified early in decommissioning. Areas with some potential for residual contamination are classified as impacted areas. Impacted areas are further divided into one of three classifications:

- Class 1 areas: Areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys) above the DCGL. Examples of Class 1 areas include: 1) site areas previously subjected to remedial actions, 2) locations where leaks or unplanned releases are known to have occurred, 3) waste storage sites, and 4) areas with contaminants in discrete solid pieces of material with high specific activity. Note that areas containing contamination in excess of the DCGL prior to remediation would generally be classified as Class 1 areas unless ample evidence exists to show that a lower classification is justified.

- Class 2 areas: These areas have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the DCGL. Examples of areas that might be classified as Class 2 for the final status survey include: 1) locations where radioactive materials were present in an unsealed form (e.g., process facilities), 2) potentially contaminated transport routes, 3) areas downwind from stack release points, 4) upper walls and ceilings of some buildings or rooms subjected to airborne radioactivity, 5) areas where low concentrations of radioactive materials were handled, and 6) areas on the perimeter of former contamination control areas.

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

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

The classifications for the various areas of the HNP site are presented in the tables below. Each table presents the survey area number, the description (identification) of the area, the MARSSIM classification, and remarks regarding the selection of the individual area classifications. Many of the classifications were conservatively chosen using historical site information. For example, all of the survey units within the containment structure, the waste disposal building, the PAB and the RCA yard were assigned to Class

1. Class 1 is appropriate for these areas given their known radiological history and past uses. Other areas were designated as Class 3 despite the fact that none of the historical radiological survey data or other historical site documents reviewed for these areas showed that any radiological contamination had ever been detected. The decision to make these areas Class 3 (as opposed to non-impacted) is based on the proximity of these areas to areas known or suspected to be impacted and the fact that personnel move freely between various areas of the site. In addition, the history of radioactive particles being found in uncontrolled areas in the vicinity of the main plant area precluded a non-impacted designation. Examples of these Class 3 areas include the Emergency Operations Facility, the Information Center, and the Administration Building (Reference 2-7).

Table 2-6
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Floor Area	Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
1102	Fuel Building Laydown Area	0001	Floor Area, Walls and Ceiling - Section 1	1	80	250	3.1	GAD1000
		0002	Floor Area, Walls and Ceiling - Section 2	1	70	275	3.9	GAD1000
		0003	Floor Area, Walls and Ceiling - Section 3	1	65	315	4.8	GAD1000
1104	Fuel Building Fuel Cask Decon Area	0000	Floor Area, Walls and Ceiling	1	40	245	6.1	GAD1000
1106	Fuel Building Skimmer Pump and Sump Area	0000	Floor Area, Walls and Ceiling	1	30	160	5.3	GAD1000
1202	Fuel Building New Fuel Storage Area	0000	Floor Area, Walls and Ceiling	1	90	340	3.8	GAD1000
1204	Fuel Building Exhaust Filters and Fan Area	0000	Floor Area, Walls and Ceiling	1	20	110	5.5	GAD1000
1302	Fuel Building Patio Area	0001	Roof Area	1	N/A	65	N/A	GAD1000
		0002	Roof Area	1	N/A	65	N/A	GAD1000
1304	Fuel Building New Fuel Storage Area	0000	Floor Area, Walls and Ceiling	1	80	320	4.0	GAD1000
1306	Fuel Building Cask Laydown Area	0000	Floor Area, Walls and Ceiling	1	80	320	4.0	GAD1000
1308	Fuel Building Spent Fuel Pool Pit	0001	Floor Area, Pool Sides and Bottom, Walls and Ceiling - Section 1	1	100	655	6.6	GAD1000
		0002	Floor Area, Pool Sides and Bottom, Walls and Ceiling - Section 2	1	100	655	6.6	GAD1000
1404	Fuel Building Roof Area	0001	Roof Area - Section 1	1	N/A	90	N/A	GAD9002
		0002	Roof Area - Section 2	1	N/A	90	N/A	GAD9002
		0003	Roof Area - Section 3	1	N/A	90	N/A	GAD9002
		0004	Roof Area - Section 4	1	N/A	90	N/A	GAD9002
		0005	East & West Exterior Walls	3	N/A	1,000	N/A	N/A
		0006	North & South Exterior Walls	3	N/A	460	N/A	N/A
2002	Auxiliary Building RHR Pump Room A	0000	Floor Area, Walls and Ceiling	1	25	275	11.0	GAD2000
2004	Auxiliary Building RHR Pump Room B	0000	Floor Area, Walls and Ceiling	1	25	275	11.0	GAD2000
2006	Auxiliary Building RHR Heat Exchanger Room	0000	Floor Area, Walls and Ceiling	1	30	390	13.0	GAD2000
2008	Auxiliary Building Primary Drain Tank Pump Room	0000	Floor Area, Walls and Ceiling	1	35	380	10.9	GAD2000
2010	Auxiliary Building Primary Drain Tank Room	0000	Floor Area, Walls and Ceiling	1	30	320	10.7	GAD2000
2012	Auxiliary Building Aerated Drain Tank Room	0000	Floor Area, Walls and Ceiling	1	25	285	11.4	GAD2000
2104	Auxiliary Building Pipe Chase Under Hallway	0001	Floor Area, Walls and Ceiling - Section 1	1	85	300	3.5	N/A
		0002	Floor Area, Walls and Ceiling - Section 2	1	70	215	3.1	N/A
2106	Auxiliary Building Pipe Chase Under Valve Room	0001	Floor Area, Walls and Ceiling - Section 1	1	60	155	2.6	N/A
		0002	Floor Area, Walls and Ceiling - Section 2	1	60	155	2.6	N/A
2108	Auxiliary Building Boric Acid Evaporator Area TK EV1-1A, EV2-1A	0000	Floor Area, Walls and Ceiling	1	80	265	3.3	GAD2000
2110	Auxiliary Building Pipe Chase East & West	0000	Floor Area, Walls and Ceiling	1	75	250	3.3	N/A

Table 2-6
MARSSIM Classifications

Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Area (m^2) Floor Area	Area (m^2) Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
	Outside							
2202	Auxiliary Building Hallway	0001	Floor Area, Walls and Ceiling - Section 1	1	80	270	3.4	GAD2000
		0002	Floor Area, Walls and Ceiling - Section 2	1	80	270	3.4	GAD2000
2204	Auxiliary Building Component Cooling Area	0000	Floor Area, Walls and Ceiling	1	100	320	3.2	GAD2000
2206	Auxiliary Building Boric Acid Evaporator Area	0000	Floor Area, Walls and Ceiling	1	80	300	3.8	GAD2000
2208	Auxiliary Building Boric Acid Mix Tank Area	0000	Floor Area, Walls and Ceiling	1	100	320	3.2	GAD2000
2210	Auxiliary Building "B" Charging Pump Area	0000	Floor Area, Walls and Ceiling	1	35	220	6.3	GAD2000
2212	Auxiliary Building "A" Charging Pump Area	0000	Floor Area, Walls and Ceiling	1	35	220	6.3	GAD2000
2214	Auxiliary Building Metering Pump Area	0000	Floor Area, Walls and Ceiling	1	35	240	6.9	GAD2000
2216	Auxiliary Building Purification Pump Area	0000	Floor Area, Walls and Ceiling	1	100	430	4.3	GAD2000
2218	Auxiliary Building Primary Water Transfer Pump Area	0000	Floor Area, Walls and Ceiling	1	40	200	5.0	GAD2000
2220	Auxiliary Building Sample Room	0000	Floor Area, Walls and Ceiling	1	20	150	7.5	GAD2000
2222	Auxiliary Building Steam Generator Blowdown Room	0000	Floor Area, Walls and Ceiling	1	45	240	5.3	GAD2000
2224	Auxiliary Building HPSI Cubicle Area	0000	Floor Area, Walls and Ceiling	1	60	180	3.0	GAD2000
2226	Auxiliary Building LPSI Cubicle Area	0000	Floor Area, Walls and Ceiling	1	60	160	2.7	GAD2000
2228	Auxiliary Building Drumming Room	0001	Floor, Walls and Ceiling - Section 1	1	70	460	6.6	GAD2000
		0002	Floor, Walls and Ceiling - Section 2	1	100	536	5.4	GAD2000
2302	Auxiliary Building Component Cooling Area	0000	Floor Area, Walls and Ceiling	1	90	355	3.9	GAD2000
2304	Auxiliary Building Boric Acid Evaporator Area	0000	Floor Area, Walls and Ceiling	1	75	350	4.7	GAD2000
2306	Auxiliary Building Boric Acid Mix Tank Area	0000	Floor Area, Walls and Ceiling	1	80	320	4.0	GAD2000
2308	Auxiliary Building Volume Control Tank Room	0000	Floor Area, Walls and Ceiling	1	40	220	5.5	GAD2000
2310	Auxiliary Building Purge and Dilution Fan Areas	0000	Floor Area, Walls and Ceiling	1	90	320	3.6	GAD2000
2312	Auxiliary Building Service Water Strainer Area	0000	Floor Area, Walls and Ceiling	1	25	170	6.8	GAD2000
2314	Auxiliary Building HEPA Filter and Hall Area	0001	Floor Area, Walls and Ceiling - Section 1	1	85	310	3.6	GAD2000
		0002	Floor Area, Walls and Ceiling - Section 2	1	80	205	2.6	GAD2000
		0003	Floor Area, Walls and Ceiling - Section 3	1	70	180	2.6	GAD2000
		0004	Floor Area, Walls and Ceiling - Section 4	1	70	195	2.8	GAD2000
		0005	Floor Area, Walls and Ceiling - Section 5	1	95	250	2.6	GAD2000
		0006	Floor Area, Walls and Ceiling - Section 6	1	75	200	2.7	GAD2000
		0007	Floor Area, Walls and Ceiling - Section 7	1	65	190	2.9	GAD2000
		0008	Floor Area, Walls and Ceiling - Section 8	1	35	70	2.0	GAD2000
2316	Auxiliary Building Boric Acid Storage Room	0001	Floor Area, Walls and Ceiling - Section 1	1	60	155	2.6	GAD2000

Table 2-6
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Floor Area	Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
		0002	Floor Area, Walls and Ceiling - Section 2	1	60	215	3.6	GAD2000
2402	Auxiliary Building Roof Area	0001	Roof Area by Ventilation Duct- Section 1	1	N/A	75	N/A	GAD9002
		0002	Roof Area by Ventilation Duct- Section 2	1	N/A	75	N/A	GAD9002
		0003	Roof Area by Ventilation Duct- Section 3	1	N/A	75	N/A	GAD9002
		0004	Remaining Roof Area - Section 1	2	N/A	455	N/A	GAD9002
		0005	Remaining Roof Area - Section 2	2	N/A	775	N/A	GAD9002
		0006	Exterior Walls	3	N/A	2,260	N/A	N/A
3002	Containment Enclosure Under Reactor Vessel	0000	Floor Area and Walls	1	15	85	5.7	GAD3001
3004	Containment Enclosure Sump Area Under Reactor Vessel	0000	Floor Area and Walls	1	20	90	4.5	GAD3001
3101	Containment Enclosure #4 Outer Annulus Lower Level NE	0001	Floor Area, Walls and Ceiling - Section 1	1	60	240	4.0	GAD3100
		0002	Floor Area, Walls and Ceiling - Section 2	1	60	240	4.0	GAD3100
3102	Containment Enclosure #1 Outer Annulus Lower Level NW	0001	Floor Area, Walls and Ceiling - Section 1	1	60	240	4.0	GAD3100
		0002	Floor Area, Walls and Ceiling - Section 2	1	60	240	4.0	GAD3100
3103	Containment Enclosure #2 Outer Annulus Lower Level SW	0001	Floor Area, Walls and Ceiling - Section 1	1	60	240	4.0	GAD3100
		0002	Floor Area, Walls and Ceiling - Section 2	1	60	240	4.0	GAD3100
3104	Containment Enclosure #3 Outer Annulus Lower Level SE	0001	Floor Area, Walls and Ceiling - Section 1	1	60	240	4.0	GAD3100
		0002	Floor Area, Walls and Ceiling - Section 2	1	60	240	4.0	GAD3100
3105	Containment Enclosure Containment Sump Area	0000	Floor Area, Walls and Ceiling	1	35	185	5.3	GAD3100
3107	Containment Enclosure Cable Vault Outside Containment	0001	Floor Area, Walls and Ceiling - Section 1	1	70	290	4.1	GAD3100
		0002	Floor Area, Walls and Ceiling - Section 2	1	70	290	4.1	GAD3100
3111	Containment Enclosure Loop #1 Inner Annulus Lower Level NE	0001	Floor Area, Walls and Ceiling - Section 1	1	60	225	3.8	GAD3100
		0002	Floor Area, Walls and Ceiling - Section 2	1	55	220	4.0	GAD3100
3112	Containment Enclosure Loop #2 Inner Annulus Lower Level NW	0001	Floor Area, Walls and Ceiling - Section 1	1	75	255	3.4	GAD3100
		0002	Floor Area, Walls and Ceiling - Section 2	1	75	255	3.4	GAD3100
3113	Containment Enclosure Loop #3 Inner Annulus Lower Level SW	0001	Floor Area, Walls and Ceiling - Section 1	1	75	255	3.4	GAD3100
		0002	Floor Area, Walls and Ceiling - Section 2	1	75	255	3.4	GAD3100
3114	Containment Enclosure Loop #4 Inner Annulus Lower Level SE	0001	Floor Area, Walls and Ceiling - Section 1	1	75	290	3.9	GAD3100
		0002	Floor Area, Walls and Ceiling - Section 2	1	75	270	3.6	GAD3100

Table 2-6
MARSSIM Classifications

Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Area (m^2) Floor Area	Area (m^2) Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
3201	Containment Enclosure #1 Outer Annulus Ground Level NE	0000	Floor Area, Walls and Ceiling	1	100	500	5.0	GAD3200
3202	Containment Enclosure #2 Outer Annulus Ground Level NW	0001	Floor Area, Walls and Ceiling - Section 1	1	60	350	5.8	GAD3200
		0002	Floor Area, Walls and Ceiling - Section 2	1	60	370	6.2	GAD3200
3203	Containment Enclosure #3 Outer Annulus Ground Level SW	0001	Floor Area, Walls and Ceiling - Section 1	1	60	375	6.3	GAD3200
		0002	Floor Area, Walls and Ceiling - Section 2	1	60	350	5.8	GAD3200
3204	Containment Enclosure #4 Outer Annulus Ground Level SE	0001	Floor Area, Walls and Ceiling - Section 1	1	60	380	6.3	GAD3200
		0002	Floor Area, Walls and Ceiling - Section 2	1	60	350	5.8	GAD3200
3205	Containment Enclosure Containment Foyer Area Ground Level	0000	Floor Area and Walls	1	65	235	3.6	GAD3200
3206	Containment Enclosure Containment Hatch Area Ground Level	0000	Floor Area and Walls	1	95	195	2.1	GAD3200
3211	Containment Enclosure Loop #1 Inner Annulus Mid Ground NE	0000	Floor Area and Walls	1	70	370	5.3	GAD3200
3212	Containment Enclosure Loop #2 Inner Annulus Mid Ground NW	0000	Floor Area and Walls	1	100	455	4.6	GAD3200
3213	Containment Enclosure Loop #3 Inner Annulus Mid Ground SW	0000	Floor Area and Walls	1	100	420	4.2	GAD3200
3214	Containment Enclosure Loop #4 Inner Annulus Mid Ground SE	0000	Floor Area and Walls	1	80	395	4.9	GAD3200
3301	Containment Enclosure #1 Outside Crane Charging Floor	0001	Floor Area and Containment Enclosure Wall up to el. 56' 6" - Section 1	1	60	110	1.8	GAD3300
		0002	Floor Area and Containment Enclosure Wall up to el. 56' 6" - Section 2	1	60	110	1.8	GAD3300
3302	Containment Enclosure #2 Outside Crane Charging Floor	0001	Floor Area and Containment Enclosure Wall up to el. 56' 6" - Section 1	1	60	110	1.8	GAD3300
		0002	Floor Area and Containment Enclosure Wall up to el. 56' 6" - Section 2	1	60	110	1.8	GAD3300
3303	Containment Enclosure #3 Outside Crane Charging Floor	0001	Floor Area and Containment Enclosure Wall up to el. 56' 6" - Section 1	1	60	110	1.8	GAD3300
		0002	Floor Area and Containment Enclosure Wall up to el. 56' 6" - Section 2	1	60	110	1.8	GAD3300
3304	Containment Enclosure #4 Outside Crane Charging Floor	0001	Floor Area and Containment Enclosure Wall up to el. 56' 6" - Section 1	1	60	110	1.8	GAD3300
		0002	Floor Area and Containment Enclosure Wall up to el. 56' 6" - Section 2	1	60	110	1.8	GAD3300
3311	Containment Enclosure #1 Inside Crane	0000	Floor Area	1	70	70	1.0	GAD3300

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MARSSIM Classifications

Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Area (m^2) Floor Area	Area (m^2) Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
	Charging Floor							
3312	Containment Enclosure #2 Inside Crane Charging Floor	0000	Floor Area	1	95	95	1.0	GAD3300
3313	Containment Enclosure #3 Inside Crane Charging Floor	0000	Floor Area	1	85	85	1.0	GAD3300
3314	Containment Enclosure #4 Inside Crane Charging Floor	0000	Floor Area	1	80	80	1.0	GAD3300
3315	Containment Enclosure Removable Grating for RX Head Staging	0000	Floor Area	1	40	40	1.0	GAD3300
3320	Containment Enclosure CTMT Rx Refuel Canal to Spent Fuel Pit	0000	Floor and Walls	1	35	500	14.3	GAD3300
3322	Containment Enclosure CTMT Reactor Refueling Cavity	0000	Floor and Walls	1	90	460	5.1	GAD3300
3324	Containment Enclosure CTMT Reactor Vessel Area	0000	Wall Area and Supports	1	40	210	5.3	GAD3300
3326	Containment Enclosure Upper Core Package Storage Area	0000	Floor Area	1	10	10	1.0	GAD3300
3403	Containment Enclosure Inside Surfaces	0001	Dome - Quadrant 1	2	N/A	675	N/A	N/A
		0002	Dome - Quadrant 2	2	N/A	675	N/A	N/A
		0003	Dome - Quadrant 3	2	N/A	675	N/A	N/A
		0004	Dome - Quadrant 4	2	N/A	675	N/A	N/A
		0005	Shell (el. 56' 6" and up) - Section 1	2	N/A	840	N/A	N/A
		0006	Shell (el. 56' 6" and up) - Section 2	2	N/A	840	N/A	N/A
		0007	Shell (el. 56' 6" and up) - Section 3	2	N/A	840	N/A	N/A
3502	Containment Enclosure Outside Surfaces	0001	Dome - Quadrant 1	2	N/A	725	N/A	GAD9002
		0002	Dome - Quadrant 2	2	N/A	725	N/A	GAD9002
		0003	Dome - Quadrant 3	2	N/A	725	N/A	GAD9002
		0004	Dome - Quadrant 4	2	N/A	725	N/A	GAD9002
		0005	Shell - Quadrant 1, East Section	2	N/A	525	N/A	N/A
		0006	Shell - Quadrant 1, North Section	2	N/A	525	N/A	N/A
		0007	Shell - Quadrant 2	3	N/A	1,050	N/A	N/A
		0008	Shell - Quadrant 3	3	N/A	1,050	N/A	N/A
		0009	Shell - Quadrant 4	3	N/A	1,050	N/A	N/A
4102	Turbine Building North Floor Area	0000	Floor Area and Walls	2	450	1,446	3.2	GAD4100
4104	Turbine Building Oil Room, Heater Drains, Emergency Power	0000	Floor Area, Walls and Ceiling	2	810	1,650	2.0	GAD4100
4106	Turbine Building Air Compressor Area	0000	Floor Area, Walls and Ceiling	2	170	460	2.7	GAD4100
4108	Turbine Building Steam Generator Feed Pump Area	0000	Floor Area, Walls and Ceiling	2	325	800	2.5	GAD4100

Table 2-6
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Floor Area	Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
4110	Turbine Building Chemistry/Closed Cooling Water Area	0000	Floor Area, Walls and Ceiling	2	290	750	2.6	GAD4100
4112	Turbine Building Water Treatment Area	0000	Floor Area, Walls and Ceiling	2	365	980	2.7	GAD4100
4114	Turbine Building Condenser Pump and South Floor Area	0000	Floor Area, Walls and Ceiling	2	755	1,660	2.2	GAD4100
4116	Turbine Building Hoist/Equipment Laydown Area	0000	Floor Area and Walls	2	165	330	2.0	GAD4100
4118	Turbine Building Condenser "A" Water Box "A & B" Area	0000	Floor Area, Walls and Ceiling	2	480	1,040	2.2	GAD4100
4120	Turbine Building Condenser "B" Water Box "C & D" Area	0000	Floor Area, Walls and Ceiling	2	580	1,260	2.2	GAD4100
4121	Turbine Building Secondary Chem Lab	0000	Floor Area, Walls and Ceiling	2	85	270	3.2	GAD4100
4202	Turbine Building North End Open Area	0000	Wall Area and Supports	2	N/A	320	N/A	GAD4200
4204	Turbine Building Oil Reservoir Area	0000	Structure Area and Walls	2	N/A	250	N/A	GAD4200
4206	Turbine Building S/G Feedwater Heater 2A and 2B Area	0000	Floor Area, Walls and Ceiling	2	425	1,410	3.3	GAD4200
4208	Turbine Building S/G Feedwater Heater 1A and 1B Area	0000	Floor Area, Walls and Ceiling	2	270	840	3.1	GAD4200
4210	Turbine Building Steam Generator Feedwater Control Valve Area	0000	Floor Area, Walls and Ceiling	2	165	600	3.6	GAD4200
4212	Turbine Building South End/Turbine Hall	0000	Wall Area and Supports	2	N/A	640	N/A	GAD4200
4216	Turbine Building S/G Feedwater Heater 6B and 5B Area	0000	Floor Area, Walls and Ceiling	2	570	1,290	2.3	GAD4200
4218	Turbine Building S/G Feedwater Heater 6A and 5A Area	0000	Floor Area, Walls and Ceiling	2	800	1,810	2.3	GAD4200
4302	Turbine Building 30" Main Steam Line Area	0000	Floor Area, Walls and Ceiling	2	1,000	2,350	2.4	GAD4300
4304	Turbine Building 24" Main Steam Line Area	0000	Floor Area, Walls and Ceiling	2	715	1,820	2.5	GAD4300
4306	Turbine Building MSRHR 1A and 1B Area Reheater	0000	Floor Area, Walls and Ceiling	2	980	2,040	2.1	GAD4300
4308	Turbine Building MSRHR 1C and 1D Area Reheater	0000	Floor Area, Walls and Ceiling	2	900	2,080	2.3	GAD4300
4402	Turbine Building Laydown Area North Floor	0000	Floor Area and Walls	2	900	3,100	3.4	GAD4400
4404	Turbine Building Steam Generator Feedwater Heater 3A Area	0000	Floor Area and Walls	2	190	720	3.8	GAD4400
4406	Turbine Building Steam Generator Feedwater Heater 4A Area	0000	Floor Area and Walls	2	200	885	4.4	GAD4400
4408	Turbine Building Steam Generator Feedwater Heater 3B Area	0000	Floor Area and Walls	2	170	650	3.8	GAD4400
4410	Turbine Building Steam Generator Feedwater Heater 4B Area	0000	Floor Area and Walls	2	180	690	3.8	GAD4400

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MARSSIM Classifications

Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Area (m^2) Floor Area	Area (m^2) Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
4412	Turbine Building H.P. Turbine Area	0000	Floor Area	2	290	290	1.0	GAD4400
4414	Turbine Building L.P. #1 Turbine Area	0000	Floor Area	2	290	290	1.0	GAD4400
4416	Turbine Building L.P. #2 Turbine Area	0000	Floor Area	2	315	315	1.0	GAD4400
4418	Turbine Building Generator Area	0000	Floor Area	2	330	330	1.0	GAD4400
4420	Turbine Building Exciter Area	0000	Floor Area	2	60	60	1.0	GAD4400
4422	Turbine Building Laydown Area South Floor	0000	Floor Area and Walls	2	320	780	2.4	GAD4400
4424	Turbine Building Open Hoist Area	0000	Wall Area	2	N/A	460	N/A	GAD4400
4502	Turbine Building Overhead Crane Area	0001	Ceiling Area - Section 1	2	N/A	810	N/A	N/A
		0002	Ceiling Area - Section 2	2	N/A	810	N/A	N/A
		0003	Ceiling Area - Section 3	2	N/A	810	N/A	N/A
		0004	Ceiling Area - Section 4	2	N/A	810	N/A	N/A
4603	Turbine Building Roof Area	0001	Roof Area - Section 1	2	N/A	810	N/A	GAD9002
		0002	Roof Area - Section 2	2	N/A	810	N/A	GAD9002
		0003	Roof Area - Section 3	2	N/A	810	N/A	GAD9002
		0004	Roof Area - Section 4	2	N/A	810	N/A	GAD9002
		0005	Exterior Walls	3	N/A	8,760	N/A	N/A
5102	Service Building "A" Diesel Generator Area	0000	Floor Area, Walls and Ceiling	3	160	550	3.4	GAD5101
5104	Service Building "B" Diesel Generator Area	0000	Floor Area, Walls and Ceiling	3	160	550	3.4	GAD5101
5106	Service Building Clean Locker Room Area	0000	Floor Area, Walls, and Ceiling	2	220	750	3.4	GAD5100
5108	Service Building Hot Locker Room Area	0000	Floor Area, Walls, and Ceiling	2	145	490	3.4	GAD5100
5110	Service Building HP Control Point and Office Areas	0000	Floor Area, Walls, and Ceiling	2	160	695	4.3	GAD5100
5112	Service Building Woman's Locker Room Area	0000	Floor Area, Walls, and Ceiling	2	90	280	3.1	GAD5100
5114	Service Building Hot Chemistry Area	0001	Floor Area, Walls, and Ceiling - Section 1	1	70	205	2.9	GAD5100
		0002	Floor Area, Walls, and Ceiling - Section 2	1	70	205	2.9	GAD5100
5118	Service Building Maintenance Decon Area	0000	Floor Area, Walls, and Ceiling	1	70	340	4.9	GAD5100
5120	Service Building Machine Shop Clean Area	0000	Floor Area, Walls, and Ceiling	2	120	560	4.7	GAD5100
5122	Service Building Machine Shop Hot Area	0001	Floor Area, Walls, and Ceiling - Section 1	1	95	420	4.4	GAD5100
		0002	Floor Area, Walls, and Ceiling - Section 2	1	85	325	3.8	GAD5100
5124	Service Building Maintenance Clean Shop Area	0000	Floor Area, Walls, and Ceiling	2	585	2,100	3.6	GAD5101
5126	Service Building "A" Auxiliary Boiler Area	0000	Floor Area, Walls, and Ceiling	2	130	530	4.1	GAD5101
5128	Service Building "B" Auxiliary Boiler Area	0000	Floor Area, Walls, and Ceiling	2	130	530	4.1	GAD5101
5130	Service Building East Hallway	0000	Floor Area, Walls, and Ceiling	2	100	830	8.3	GAD5100
5132	Service Building Health Physics Facility 1st Floor	0000	Floor Area, Walls, and Ceiling	2	145	770	5.3	GAD5100
5134	Service Building Health Physics Facility 2nd Floor	0000	Floor Area, Walls, and Ceiling	2	145	770	5.3	GAD5100
5202	Service Building Switch Gear Area	0000	Floor Area, Walls, and Ceiling	3	900	2,490	2.8	GAD5200

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					Area (m^2)			
Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Floor Area	Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
5302	Service Building Control Room Area	0000	Floor Area, Walls, and Ceiling	3	505	1,250	2.5	GAD5200
5304	Service Building Computer, Operations, Security Area	0000	Floor Area, Walls, and Ceiling	3	195	600	3.1	GAD5200
5306	Service Building Machine and Equipment Area	0000	Floor Area, Walls, and Ceiling	2	210	575	2.7	GAD5200
5308	Service Building Instrument & Controls Shop	0000	Floor Area, Walls, and Ceiling	2	225	610	2.7	GAD5200
5402	Service Building Roof	0001	Roof Area - Section 1	2	N/A	1,000	N/A	GAD9002
		0002	Roof Area - Section 2	2	N/A	565	N/A	GAD9002
		0003	Roof Area - Section 3	2	N/A	565	N/A	GAD9002
		0004	Roof Area - Section 4	2	N/A	590	N/A	GAD9002
		0005	Exterior Walls	3	N/A	2,100	N/A	N/A
5502	CW System Trench	0001	Unit 1 Discharge Tunnel Surface Area	2	290	950	3.3	N/A
		0002	Unit 2 Discharge Tunnel Surface Area	2	470	2,110	4.5	N/A
6002	Waste Disposal Building Hall Area Lower Level	0000	Floor Area, Walls and Ceiling	1	25	230	9.2	GAD6000
6004	Waste Disposal Building Area Outside Reboiler Room	0000	Floor Area, Walls and Ceiling	1	25	190	7.6	GAD6000
6006	Waste Disposal Building Bottoms Pump and Reboiler Area	0000	Floor Area, Walls and Ceiling	1	15	105	7.0	GAD6000
6008	Waste Disposal Building Sump Trench Area Lower Level	0000	Floor Area, Walls and Ceiling	1	20	145	7.3	GAD6000
6010	Waste Disposal Building-Waste Decay Tank A, B, C Area	0000	Floor Area, Walls and Ceiling	1	20	135	6.8	GAD6000
6012	Waste Disposal Building Surge Tank Area Lower Level	0000	Floor Area, Walls and Ceiling	1	10	100	10.0	GAD6000
6102	Waste Disposal Building Hall Area	0000	Floor Area, Walls and Ceiling	1	10	40	4.0	GAD6000
6202	Waste Disposal Building Hallway Area	0000	Floor Area, Walls and Ceiling	1	30	195	6.5	GAD6000
6304	Waste Disposal Building Evaporator Area	0000	Floor Area, Walls and Ceiling	1	25	175	7.0	GAD6000
6306	Waste Disposal Building Radwaste Liquid Evaporator	0000	Floor Area, Walls and Ceiling	1	15	100	6.7	GAD6000
6308	Waste Disposal Building Degassifier Transfer Pump Area	0000	Floor Area, Walls and Ceiling	1	30	185	6.2	GAD6000
6312	Waste Disposal Building Degassifier and Associated Valves	0000	Floor Area, Walls and Ceiling	1	20	155	7.8	GAD6000
6404	Waste Disposal Building Evaporator Area	0000	Floor Area, Walls and Ceiling	1	25	175	7.0	GAD6000
6406	Waste Disposal Building Liquid Evaporator Area	0000	Floor Area, Walls and Ceiling	1	15	100	6.7	GAD6000
6408	Waste Disposal-Waste Gas Compressor A&B Area	0000	Floor Area, Walls and Ceiling	1	30	185	6.2	GAD6000

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					Area (m^2)			
Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Floor Area	Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
6412	Waste Disposal Building Degassifier Area and Associated Valves	0000	Floor Area, Walls and Ceiling	1	20	155	7.8	GAD6000
6502	Waste Disposal Building Roof Area	0001	Roof Area - Section 1	1	N/A	60	N/A	GAD6000
		0002	Roof Area - Section 2	1	N/A	60	N/A	GAD6000
		0003	Exterior Walls	3	N/A	870	N/A	N/A
7002	CW Circ Pump A&B Head Area	0000	Floor Area, Walls and Ceiling	3	90	300	3.3	GAD7000
7004	CW Circ Pump C&D Head Area	0000	Floor Area, Walls and Ceiling	3	90	300	3.3	GAD7000
7102	CW Circ Pump Motor A&B Area	0000	Floor Area, Walls and Ceiling	3	125	395	3.2	GAD7000
7104	CW Circ Pump Motor C&D Area	0000	Floor Area, Walls and Ceiling	3	125	395	3.2	GAD7000
7106	CW Hypochloride Tank Area	0000	Floor Area, Walls and Ceiling	3	55	250	4.5	GAD7000
7108	CW Intake and Screen Area	0000	Floor Area	3	180	180	1.0	GAD7000
7202	CW Roof Area	0000	Roof Area and Exterior Walls	3	N/A	595	N/A	GAD7000
8100	FW/STM Penetration Building Upper Level	0000	Floor Area, Walls and Ceiling	1	90	330	3.7	N/A
8200	FW/STM Penetration Building Mid Level	0000	Floor Area, Walls and Ceiling	1	90	330	3.7	N/A
8300	FW/STM Penetration Building Lower Level	0000	Floor Area, Walls and Ceiling	1	90	210	2.3	N/A
9102	YD 115KV Switchyard Area	0001	Trench and Adjoining Land Area	1	N/A	120	N/A	GAD9000
		0002	Land Area	2	N/A	1,220	N/A	GAD9000
9104	YD Main Transformer Area	0000	Land Area	3	N/A	930	N/A	GAD9000
9106	Discharge Canal	0001	Land Area From Discharge Structure to Old Security Guard Shack and Canal Road	2	N/A	8,360	N/A	GAD9000
		0002	Land Area From Old Security Guard Shack and Canal Road To the CT River	3	N/A	33,640	N/A	GAD9000
9108	YD North Tank Farm Area	0000	Land Area	1	N/A	200	N/A	GAD9001
9110	YD South Tank Farm Area	0000	Land Area	1	N/A	150	N/A	GAD9001
9112	YD Boron Storage Tank Area	0000	Land Area	1	N/A	60	N/A	GAD9001
9114	YD Ion Exchange Area	0000	Land Area	1	N/A	300	N/A	GAD9001
9116	YD Resin Slurry Area	0000	Land Area	1	N/A	60	N/A	GAD9001
9118	YD Fuel Oil Tank Area	0000	Standing Structure and Land Area	3	N/A	200	N/A	GAD9000
9120	YD Primary Vent Stack	0000	Structure and Land Area	1	N/A	10	N/A	N/A
9122	YD Primary Water Storage Tank Area	0000	Land Area	1	N/A	300	N/A	GAD9001
9124	YD Backup Primary Water Storage Tank Area	0000	Land Area	1	N/A	300	N/A	GAD9001
9126	YD Large Yard Crane Area	0000	Land Area	1	N/A	1,400	N/A	GAD9002
9128	YD Demin Water Storage Tank Area	0000	Land Area	1	N/A	340	N/A	GAD9001
9202	Switchgear Building "B"	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	3	300	2,300	7.7	GAD9001
9208	Administration Building	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	3	495	4,300	8.7	GAD9001
9214	Shutdown Auxiliary Feed Pump House	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	2	25	220	8.8	GAD9001

Table 2-6
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Floor Area	Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
9226	Radwaste Reduction Facility	0001	Floor, Ceiling and Walls From the Eastern Entrance to a Point 15 Foot West	1	100	460	4.6	N/A
		0002	Floor, Ceiling and Walls From Point 15 Foot West to a Point 30 Foot West	1	100	405	4.1	N/A
		0003	Floor, Ceiling and Walls From Point 30 Foot West to a Point 45 Foot West	1	100	230	2.3	N/A
		0004	Floor, Ceiling and Walls From Point 45 Foot West to West Wall	1	100	305	3.1	N/A
		0005	Floor, Ceiling and Walls From the Southwestern Rollup Door to the Opposite Shield Wall	1	75	445	5.9	N/A
		0006	Roof	2	N/A	465	N/A	N/A
		0007	Exterior Walls	3	N/A	665	N/A	N/A
9227	Bus10 Pad and Ground Underneath	0001	Bus10 Pad and Ground Underneath	1	N/A	100	N/A	N/A
		0002	Bus 13 Structure	3	100	410	N/A	N/A
9228	Unconditional Release Facility	0000	Structure Including Floor, Walls, Ceiling, Roof and External Walls	2	45	340	7.6	GAD9001
9234	HP Project Trailer	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Walls	2	14	150	10.7	GAD9001
9236	HP Count Module	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Walls	2	55	480	8.7	GAD9001
9302	Northwest Protected Area Grounds	0000	Land Area	3	N/A	4,200	N/A	GAD9000
9304	Southwest Protected Area Grounds	0000	Land Area	3	N/A	3,000	N/A	GAD9000
9306	South Central Protected Area Grounds	0000	Land Area	2	N/A	2,450	N/A	GAD9000
9307	PAB / Service Building Alleyway	0000	Land Area	1	N/A	1,000	N/A	GAD9000
9308	Southeast Protected Area Grounds	0000	Land Area	2	N/A	4,400	N/A	GAD9000
9310	East Protected Area Grounds	0001	Land Area From the Spent Fuel Building to the RadWaste Reduction Facility	1	N/A	1,120	N/A	GAD9000
		0002	Land Area From the RadWaste Reduction Facility to the East RCA Boundary	1	N/A	1,350	N/A	GAD9000
9312	Northeast Protected Area Grounds	0001	Land Area From the North RCA Gate to Security Fence	1	N/A	1,190	N/A	GAD9000
		0002	Land Area From Security Fence to Spent Fuel Building	1	N/A	1,230	N/A	GAD9000
9313	Central Site Grounds	0000	Land Area	3	N/A	600	N/A	GAD9000
9402	Emergency Operations Facility	0000	Structure Including Floor, Walls, Ceiling and Exterior Walls	3	1,300	5,300	4.1	GAD9001
9403	Emergency Operations Center Roof	0000	Structure Roof	3	N/A	1,300	N/A	GAD9002
9404	North Warehouse	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	3	400	2,900	7.3	GAD9001

Table 2-6
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Floor Area	Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
9406	South Warehouse	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	3	400	2,900	7.3	GAD9001
9408	Miscellaneous Trailer Complex	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	3	N/A	1,600	N/A	GAD9001
9410	Steam Generator Mockup Building	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	3	N/A	460	N/A	GAD9001
9412	Training Stores Office Building	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	3	480	2,620	5.5	GAD9001
9414	Warehouse #1	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	3	2,000	10,800	5.4	GAD9001
9416	Warehouse #2	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	3	1,500	8,500	5.7	GAD9001
9418	Office Building #3 and PAP	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	3	900	7,500	8.3	GAD9001
9420	Office Trailer	0000	Structure Including Floor, Walls and Ceiling	3	220	650	3.0	GAD9001
9422	Information Center	0000	Structure Including Floor, Walls, Ceiling and Exterior Walls	3	1,200	3,800	3.2	GAD9001
9423	Information Center Roof	0000	Structure Roof	3	N/A	1,200	N/A	GAD9002
9424	All Buildings Contained in the Southwest Site Storage Area	0000	Structure Including Floor, Walls, Ceiling, Roof and Exterior Surfaces	3	130	470	3.6	GAD9001
9502	Northeast Site Grounds (Non-Protected Area)	0000	Land Area	3	N/A	8,700	N/A	GAD9000
9504	Bypass Road / Secondary Parking Lot	0000	Land Area	3	N/A	2,800	N/A	GAD9000
9506	North Site Grounds (Non-Protected Area)	0000	Land Area	3	N/A	3,800	N/A	GAD9000
9508	Pond	0000	Land Area and Pond Sediment	3	N/A	10,000	N/A	GAD9000
9510	Access Road	0000	Paved Road	3	N/A	2,300	N/A	GAD9000
9512	Northwest site Grounds (Non-Protected Area)	0000	Land Area	3	N/A	19,500	N/A	GAD9000
9514	Primary Parking Lot	0000	Paved Lot	3	N/A	20,000	N/A	GAD9000
9518	Southwest Site Grounds (Non-Protected Area)	0000	Land Area	2	N/A	5,900	N/A	GAD9003
9520	Southwest Site Storage Area	0001	Land Area From Security Fence to Load Distribution Tower	2	N/A	7,500	N/A	GAD9003
		0002	Land Area From Load Distribution Tower East to 150m	2	N/A	7,000	N/A	GAD9003
		0003	Land Area 150m East of Load Distribution Tower to Gate 3	2	N/A	7,000	N/A	GAD9003
9521	Southeast Pond	0000	Land Area and Pond Sediment	3	N/A	23,100	N/A	GAD9003
9522	Southeast Site Grounds (Non-Protected Area)	0001	Land Area Above Transmission Towers	2	N/A	6,200	N/A	GAD9000
		0002	Land Area Below Transmission Towers	2	N/A	5,900	N/A	GAD9000
9523	Southeast Wetland Area	0000	Land Area	3	N/A	106,000	N/A	GAD9003

Table 2-6
MARSSIM Classifications

					Area (m^2)			
Survey Area Code	Survey Area Code Description	Survey Unit Code	Survey Unit Code Description	MARSSIM Classification	Floor Area	Total Area	Ratio (Total Area: Floor Area)	GAD Map No.
9524	South Site Grounds (Non-Protected Area)	0000	Land Area	3	N/A	110,000	N/A	GAD9003
9525	Southeast Site Road	0000	Paved Road	3	N/A	28,000	N/A	GAD9003
9526	Northeast Mountain Side	0000	Land Area	3	N/A	444,700	N/A	GAD9003
9527	East Mountain Side	0001	Land Area From Contour Line 120' to Upper Fence and Eastern Ridge Edge	2	N/A	9,800	N/A	GAD9000
		0002	Land Area From Contour Line 120' to Lower Fence and Eastern Ridge Edge	2	N/A	9,600	N/A	GAD9000
		0003	Land Area From Ridge Edge to 9522A Boundary	2	N/A	9,300	N/A	GAD9000
9528	Southeast Mountain Side	0000	Land Area	3	N/A	553,000	N/A	GAD9003
9530	Central Peninsula Area	0001	Land Area Bounded by and Immediately Adjacent to the Road	2	N/A	9,700	N/A	GAD9003
		0002	Western Half of Diked Area and Immediate Surrounding Sides	2	N/A	7,000	N/A	GAD9003
		0003	Eastern Half of Diked Area and Immediate Surrounding Sides	2	N/A	7,000	N/A	GAD9003
		0004	Remaining Land Area	3	N/A	70,800	N/A	GAD9003
9531	South End of Peninsula	0000	Land Area	3	N/A	118,000	N/A	GAD9003
9532	East Site Grounds (Non-Protected Area)	N/A	N/A	Non-impacted	N/A	375,600	N/A	GAD9003
9535	South East Landfill Area	0000	Land Area	2	N/A	8,600	N/A	GAD9003
9536	Construction Piles Near Rifle Range	0000	Land Area	2	N/A	2,200	N/A	GAD9003
9537	Permitted Landfill Area	0000	Land Area	2	N/A	2,200	N/A	GAD9003
9538	Material Storage Area	0000	Land Area	2	N/A	4,200	N/A	GAD9003

2.3.3.3 Non-Impacted Area Assessment

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

The Connecticut Yankee Haddam Neck Characterization Report has classified the East Site Grounds (survey area 9532) as non-impacted. This area consists of approximately ninety-three (93) acres of uninhabited, undeveloped land located about a third of a mile (0.29 miles) from the RCA (Radiologically Controlled Area). The East Site Grounds are bounded by steep, wooded hillsides to the east, an open clearing for power distribution lines to the south and west, and an access road (Wood Road) from the substation to the discharge canal to the north. Access to the interior of the South East Grounds area is limited to a gated road (Cove Road) to the east, abandoned or seldom used logging paths and trails, and the power transmission clearing to the west. A walk down and visual inspection of the East Site Grounds area indicates the land was not used to store materials from the Haddam Neck Plant. There were no identified soil disturbances that would indicate dumping or burial of materials.

Historical data do not indicate that plant operations had an impact on the East Site Grounds area. Historical data and radiological surveys have identified contamination from plant operations on the east hillside from the RCA boundary out to a distance of 200 meters or roughly an eighth of a mile (0.12 miles). Following identification of the contamination further surveys were conducted to a distance of approximately 400 meters, with no additional plant related radioactivity identified. Given the topography of the eastern hillside in general, and the distance from the RCA to the nearest boundary of the East Site Grounds area (0.29 miles), past occurrences from plant operations would not have had any radiological impact on the East Site Grounds.

Radiological environmental monitoring and sampling is performed in the East Site Grounds area in accordance with the Radiological Effluent Monitoring and Off-site Dose Calculation Manual. Radiological analyses are performed with gamma exposure measuring devices, on samples of air particulates and iodine, and on broad leaf vegetation. The Haddam Neck Station Annual Radiological Reports show no long-lived radionuclides other than Sr-90 and Cs-137 above the MDL (Minimum Detectable Level). These radionuclides (Sr-90 and Cs-137) are measured at levels consistent with those found throughout the central Connecticut area and are attributed to past atmospheric nuclear weapons testing. It is important to note that the Haddam Neck Station Annual Radiological Reports consider all data statistically valid, including negative values, zeros, numbers below the MDL and those values with reporting errors greater than two standard deviations. The Haddam Neck Station Annual Radiological Reports present all valid data for strictly counting statistics purposes and to indicate background biases.

The historical data, use and topography of the land and radiological environmental monitoring results support the classification of the East Site Grounds area as a non-impacted area. The East Site Grounds area may be used as a background reference area for soil samples taken as a part of the FSS (Final Status Survey).

2.3.3.4 Hazardous Material Status

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

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

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

2.4 References

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

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

2-3 Investigation of the Source of Radioactive Contamination Found on the Connecticut Yankee Site March 10-30, 1980, dated April 1980

2-4 Results of Phase 2 PCB and Radiological Characterization Study, CY Letter HP-98-423, dated July 28, 1998.

2-5 Groundwater Monitoring Report, Connecticut Yankee Atomic Power Station Haddam Neck, Connecticut, dated July 1999, revised September 1999, Malcom Pirney, Inc.

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

2-7 "Initial Classification for Survey Areas at Connecticut Yankee," Bechtel Document No. 24265-100-V00-HPRA-G0002.

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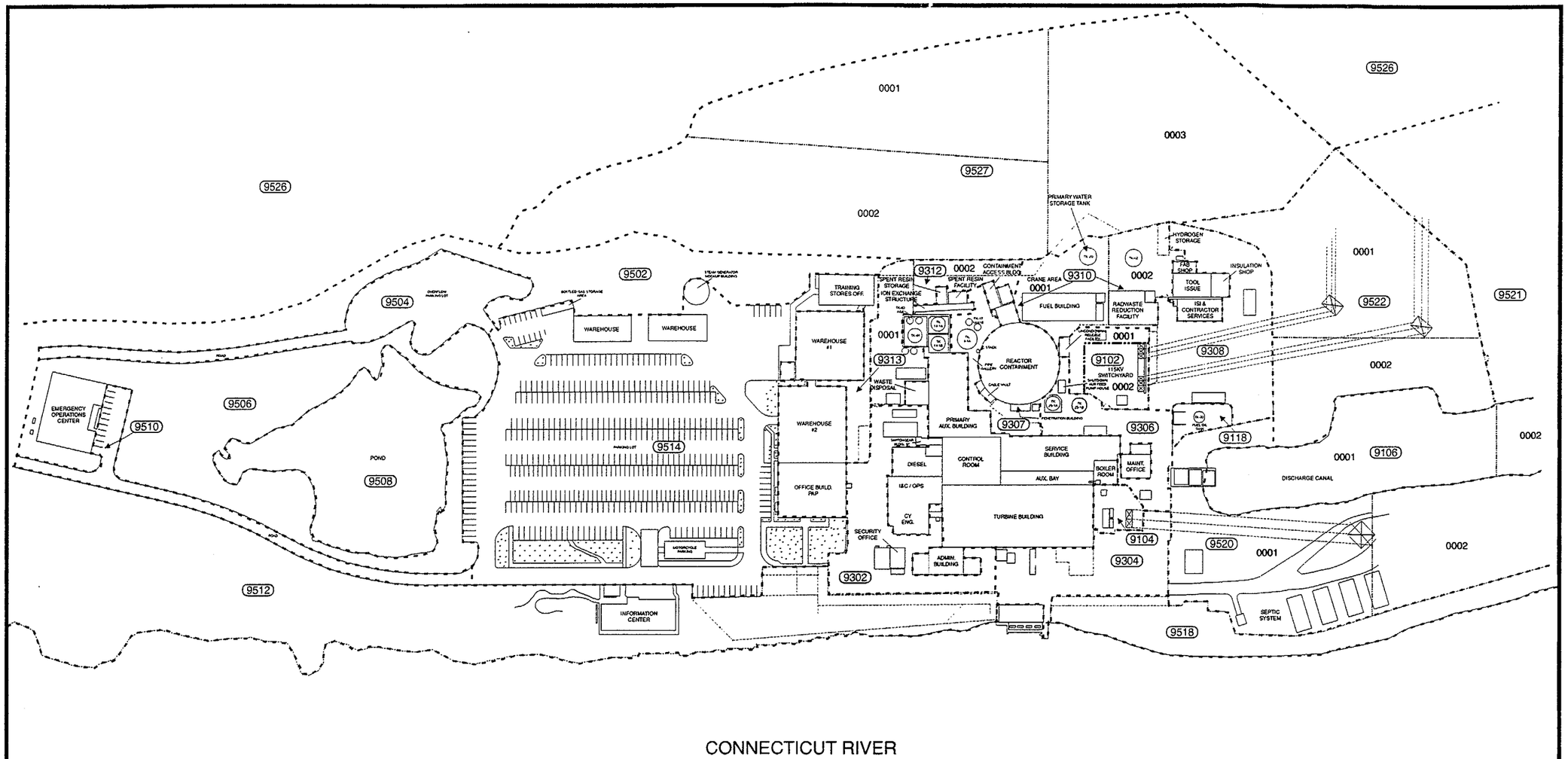

Site Grounds
Figure 2-1

Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
SITE GROUNDS

INITIAL CREATION DATE:	04/25/1997
CREATED BY:	E. ALSING
REVISION DATE:	03/29/2000
REVISION #:	15
MAP #:	GAD9000

LEGEND:

FLOOR WALL INTERSECTION	———
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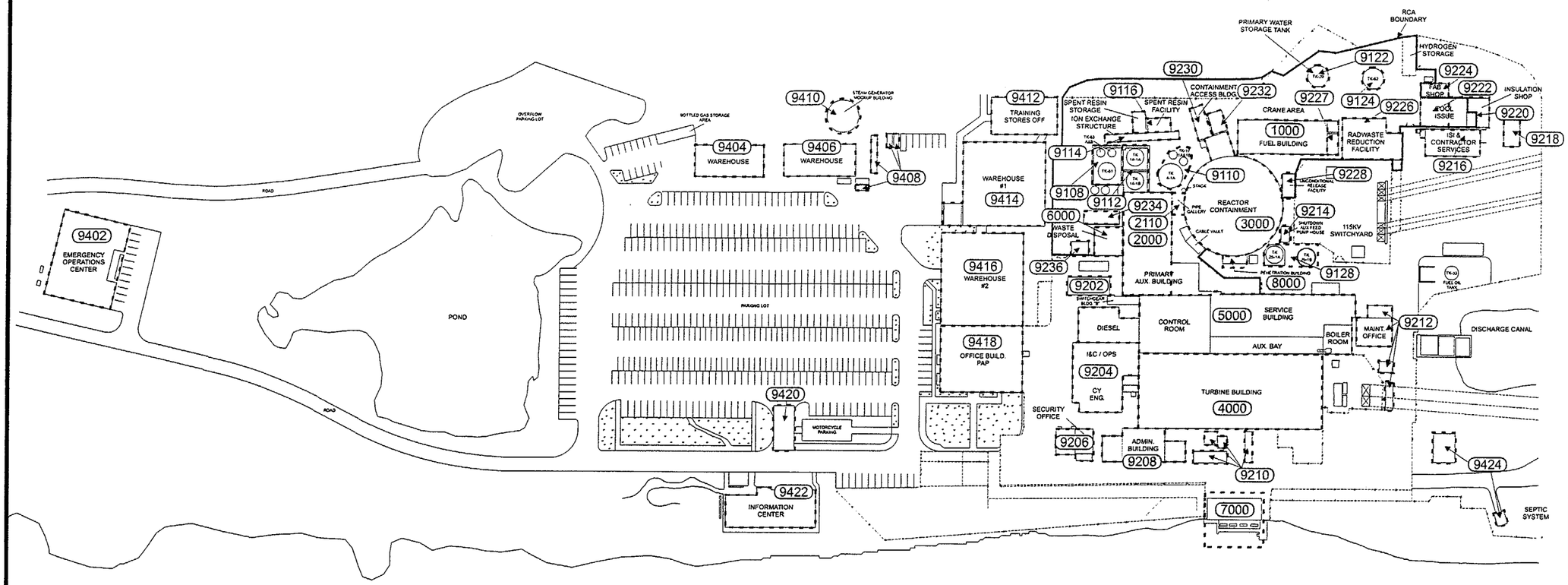
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—·—·—
SURVEY UNIT NUMBER	####



Buildings

Site Buildings
Figure 2-2

Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
SITE BUILDINGS

INITIAL CREATION DATE:	05/08/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/22/2000
REVISION #:	7
MAP #:	GAD9001

LEGEND:

FLOOR WALL INTERSECTION	———
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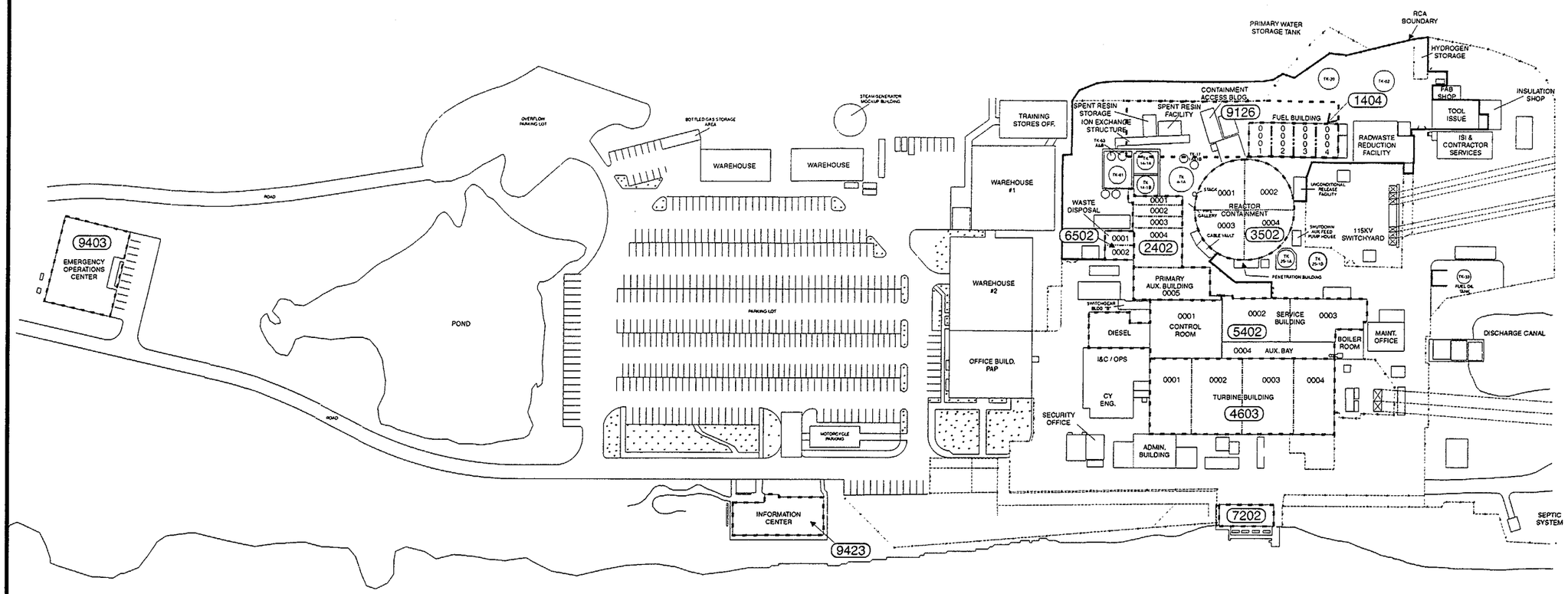
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—·—·—
SURVEY UNIT NUMBER	####



Buildings

Structure Exteriors
Figure 2-3

Plant North

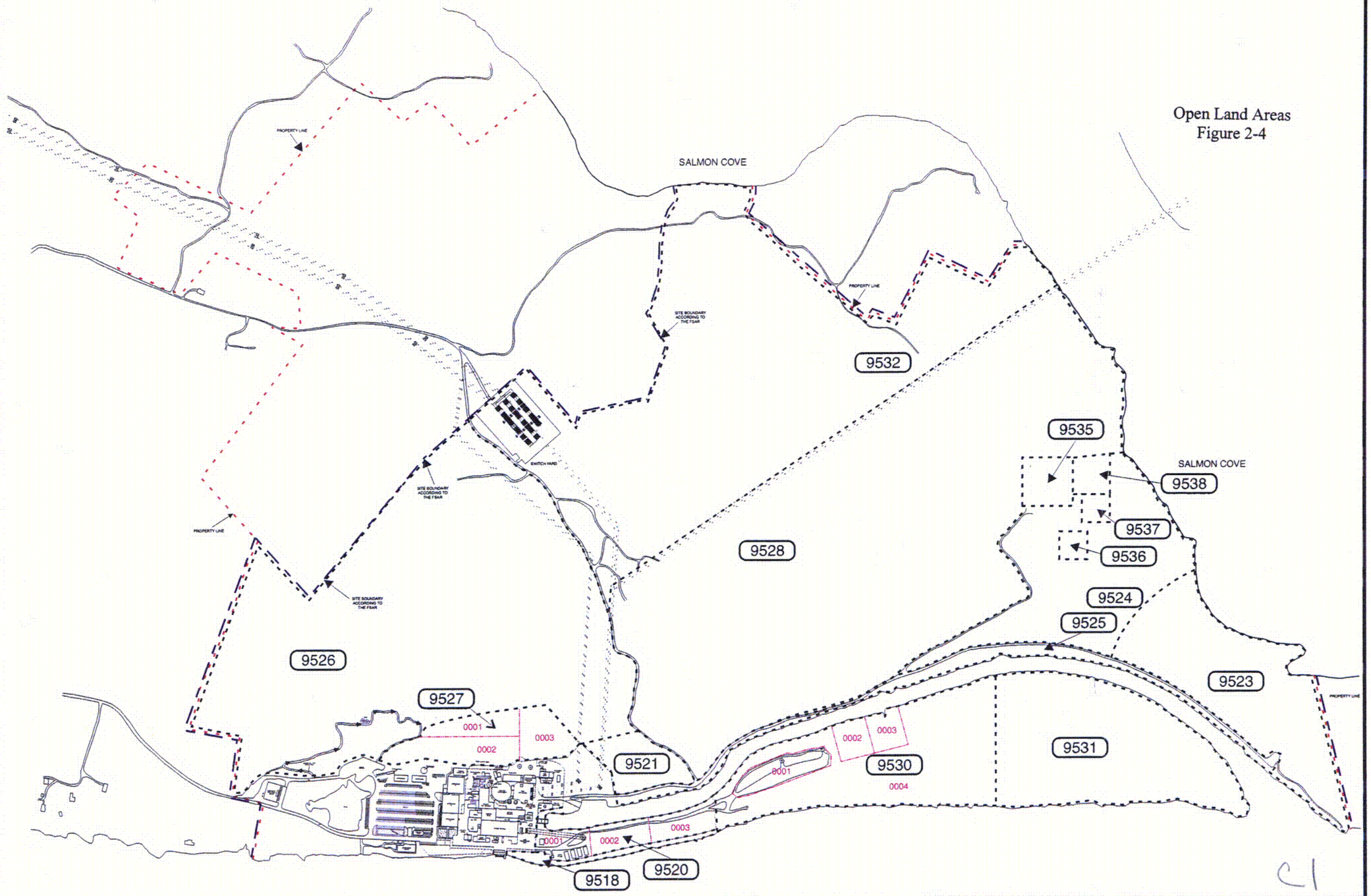
CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
STRUCTURE EXTERIORS

INITIAL CREATION DATE:	09/08/1997
CREATED BY:	E. ALSING
REVISION DATE:	03/15/2000
REVISION #:	5
MAP #:	GAD9002

LEGEND:

FLOOR WALL INTERSECTION	———
SURVEY AREA BOUNDARY	- - -
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—·—·—
SURVEY UNIT NUMBER	####

Open Land Areas
Figure 2-4



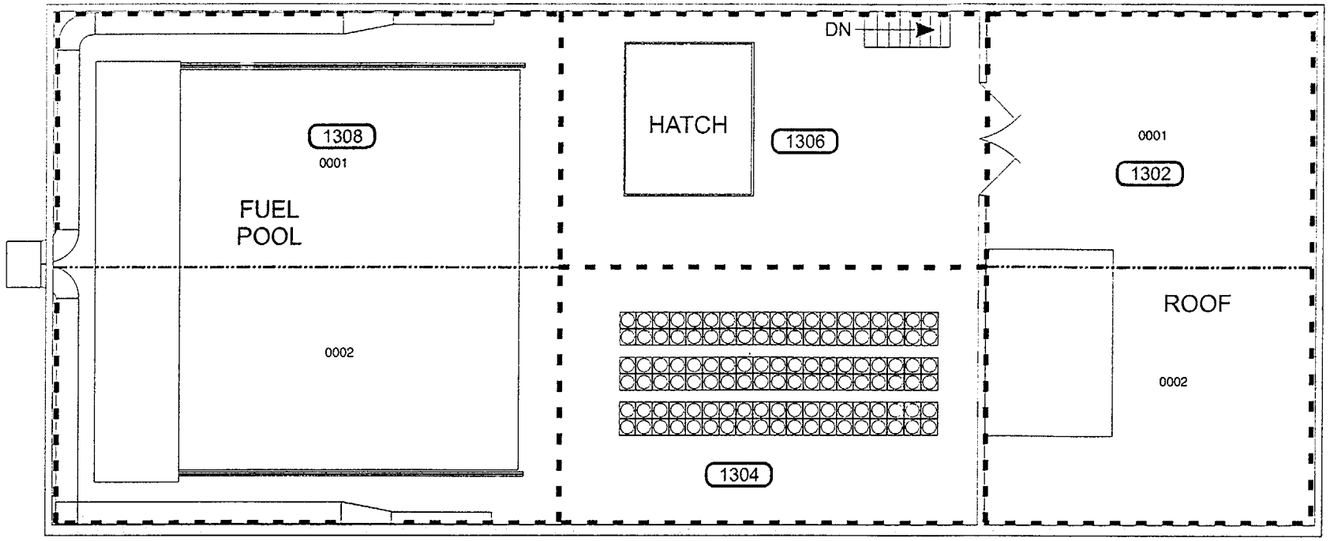
Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
OPEN LAND AREAS

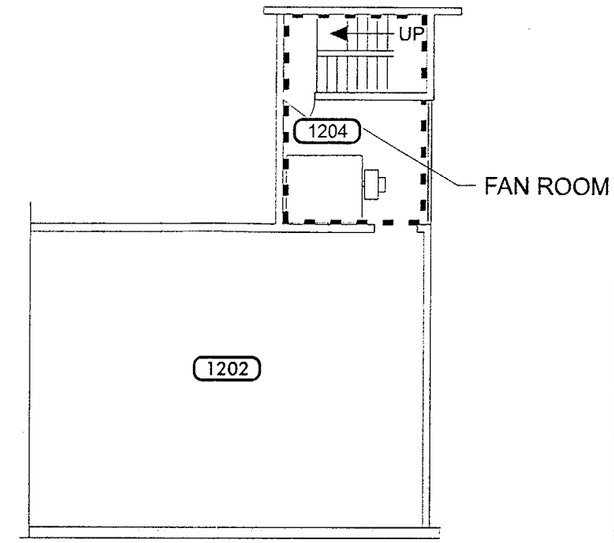
INITIAL CREATION DATE:	09/24/97
CREATED BY:	E. ALSING
REVISION DATE:	01/06/2000
REVISION #:	8
MAP #:	GAD9003

LEGEND:

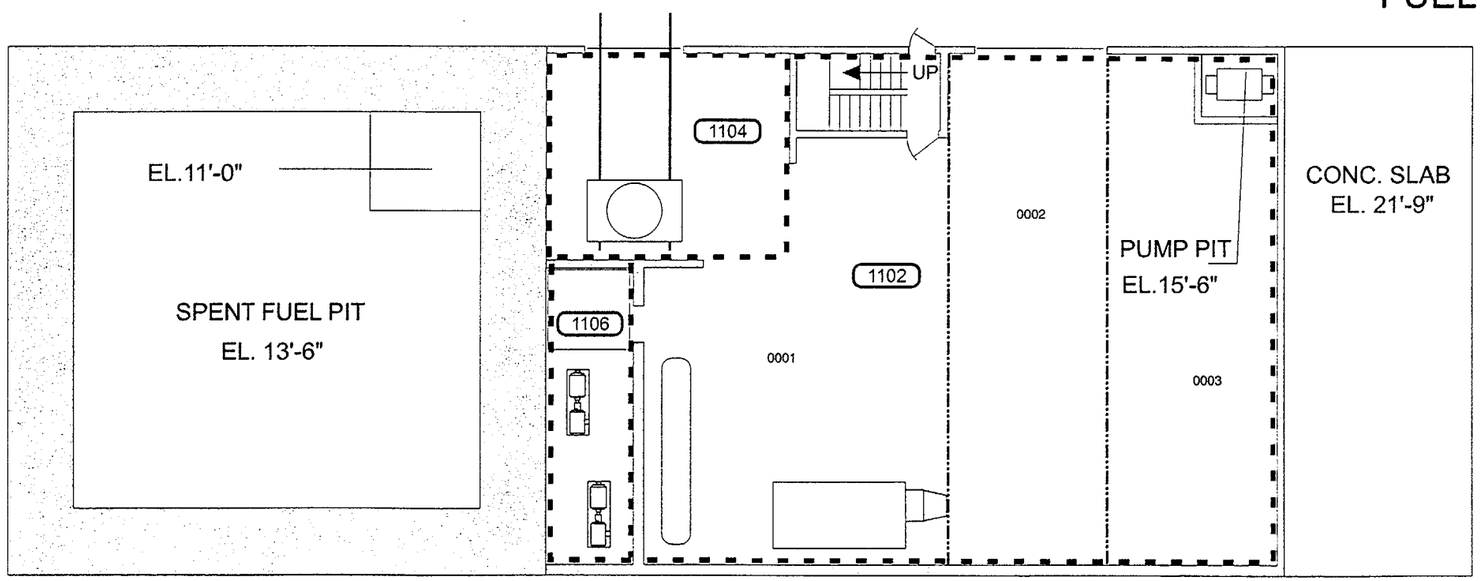
SITE PROPERTY LINE	
SITE BOUNDARY	
SURVEY AREA BOUNDARY	
SURVEY AREA	#
SURVEY UNIT BOUNDARY	
SURVEY UNIT NUMBER	####



FUEL BLDG. EL. 47'-0"



FUEL BLDG. EL. 35'-0"



FUEL BLDG. EL. 21'-6"

Fuel Building
Elevations 21'-6", 35'-0" & 47'-0"
Figure 2-5

Areas not shown

1402	5 Ton Cranes Area
1402	Fuel Building Roof

Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
FUEL BUILDING
ELEVATION(s) 21'-6", 35'-0" & 47'-0"

INITIAL CREATION DATE:	05/20/1997
CREATED BY:	E. ALSING
REVISION DATE:	03/06/2000
REVISION #:	2
MAP #:	GAD1000

LEGEND:

FLOOR WALL INTERSECTION	———
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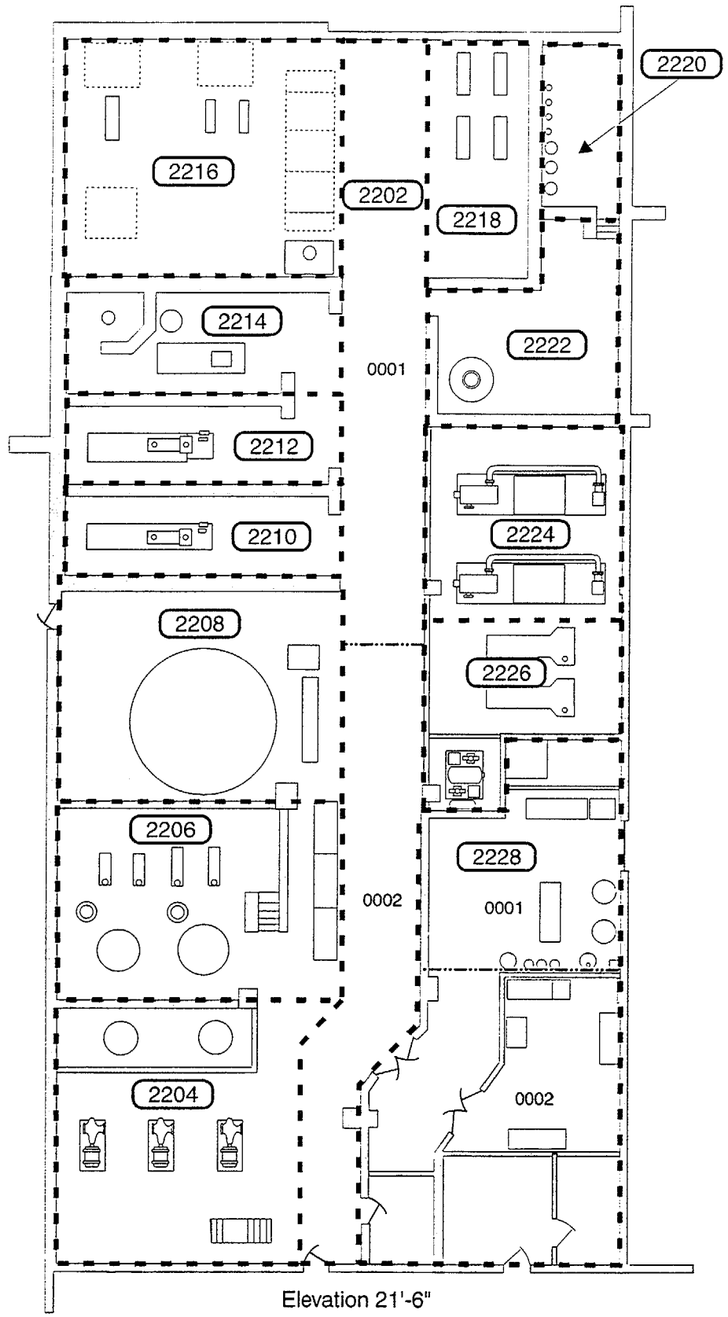
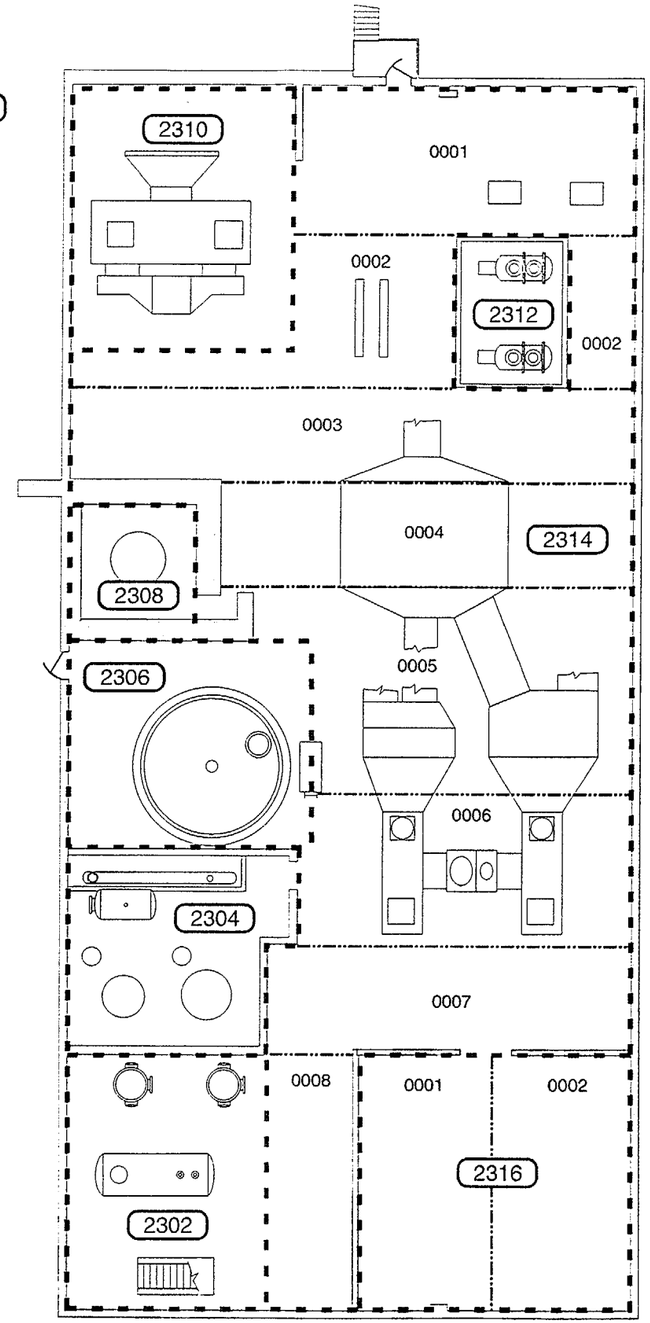
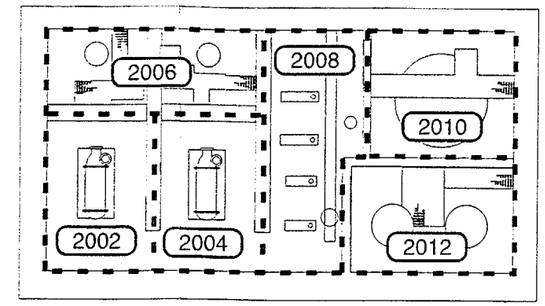
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—·—·—
SURVEY UNIT NUMBER	####



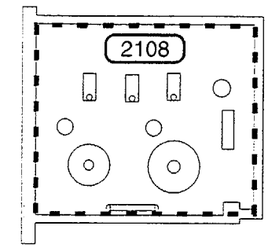
Elevation 21'-6"



Elevation 35'-6"



Elevation 19'-0"



Elevation 15'-6"

Primary Auxiliary Building
Elevations 15'-6", 19'-0", 21'-6" & 35'-6"
Figure 2-6

Not Shown

2102	PAB Mezzanine @ 15'-6" Elev
2104	PAB Pipe Chase Under Hallway
2110	PAB East & West Outside Pipe Trench

Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
PRIMARY AUXILIARY BUILDING
ELEVATION(s) 15'-6", 19'-0", 21'-6" & 35'-6"

INITIAL CREATION DATE:	06/18/97
CREATED BY:	E. ALSING
REVISION DATE:	02/15/2000
REVISION #:	5
MAP #:	GAD2000

LEGEND:

FLOOR WALL INTERSECTION	———
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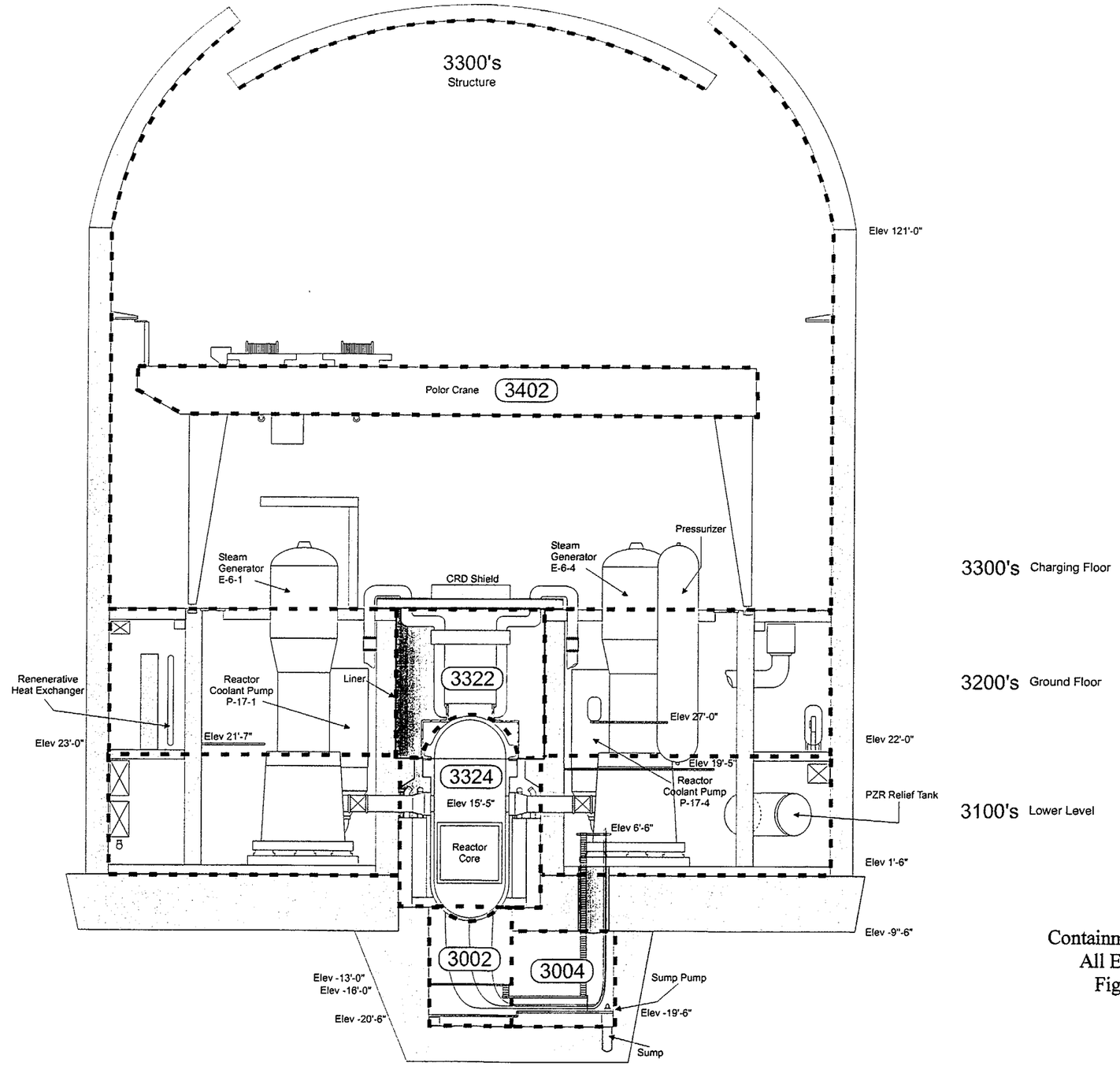
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—··—··—
SURVEY UNIT NUMBER	####



Containment Building
All Elevations
Figure 2-7

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
CONTAINMENT BUILDING
ALL ELEVATIONS

INITIAL CREATION DATE:	05/07/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/22/2000
REVISION #:	1
MAP #:	GAD3001

LEGEND:

FLOOR WALL INTERSECTION	———
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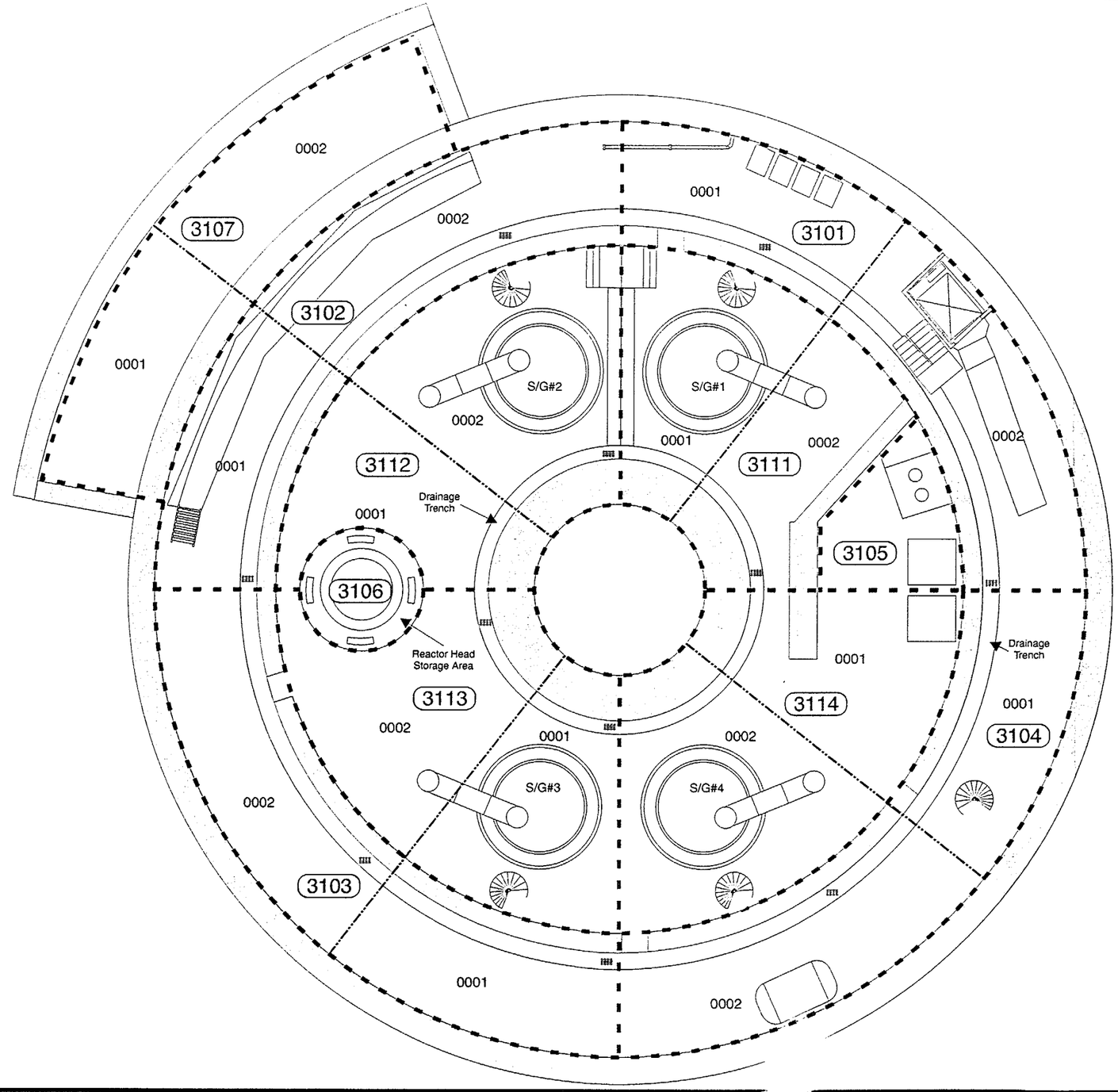
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—·—·—
SURVEY UNIT NUMBER	####



Containment Building
Ground Floor
Elevation 1'-6"
Figure 2-8

N

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
CONTAINMENT BUILDING GROUND FLOOR
ELEVATION 1'-6"

IAL CREATION DATE:	04/25/1997
REATED BY:	E. ALSING
REVISION DATE:	02/16/2000
REVISION #:	2
MAP #:	GAD3100

LEGEND:	
FLOOR WALL INTERSECTION	———
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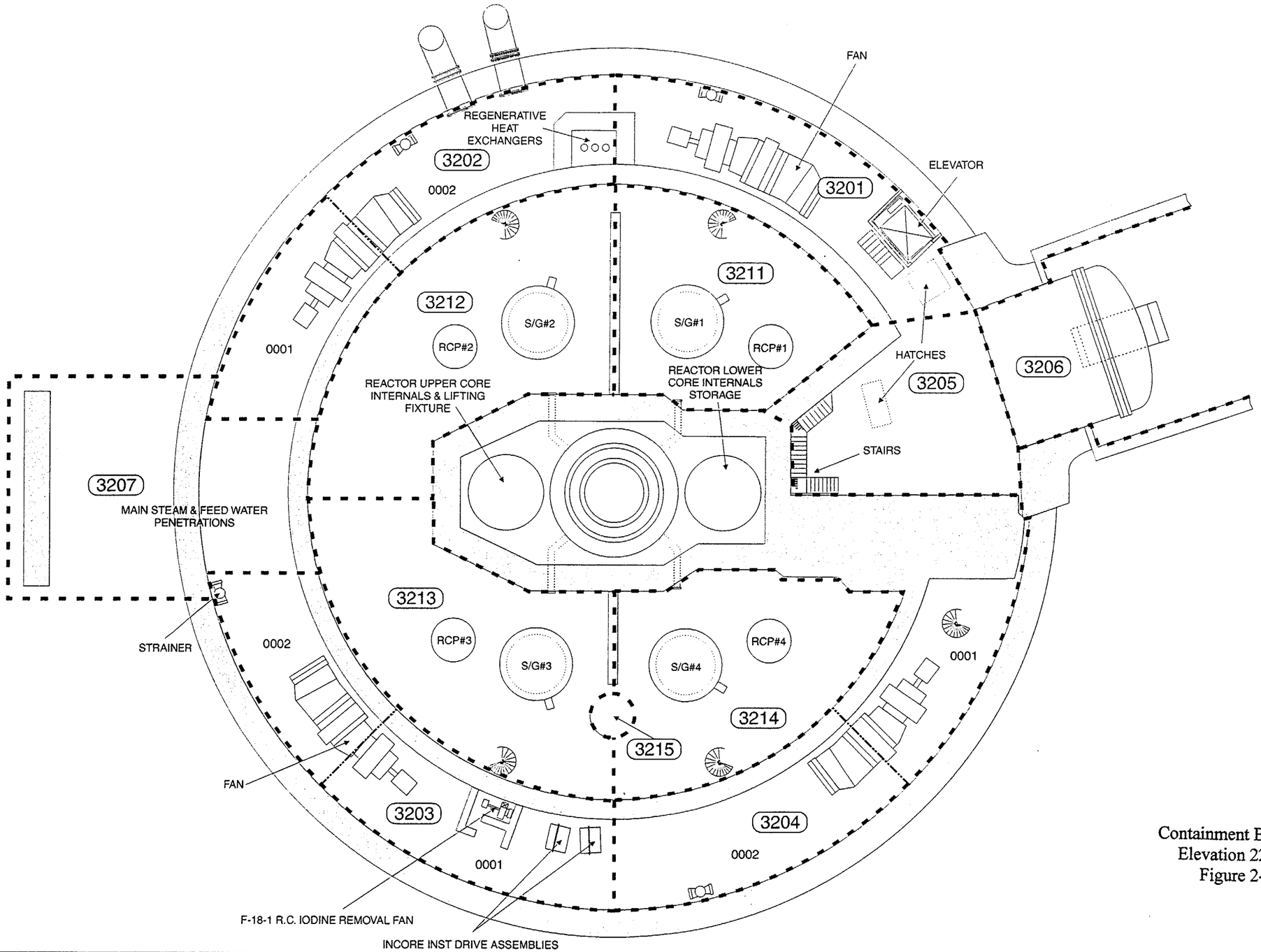
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—·—·—
SURVEY UNIT NUMBER	####



Containment Building
Elevation 22'-0"
Figure 2-9

N

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
CONTAINMENT BUILDING
ELEVATION 22'-0"

INITIAL CREATION DATE:	05/01/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/16/2000
REVISION #:	1
MAP #:	GAD3200

LEGEND:

FLOOR WALL INTERSECTION	———
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SURVEY AREA	#
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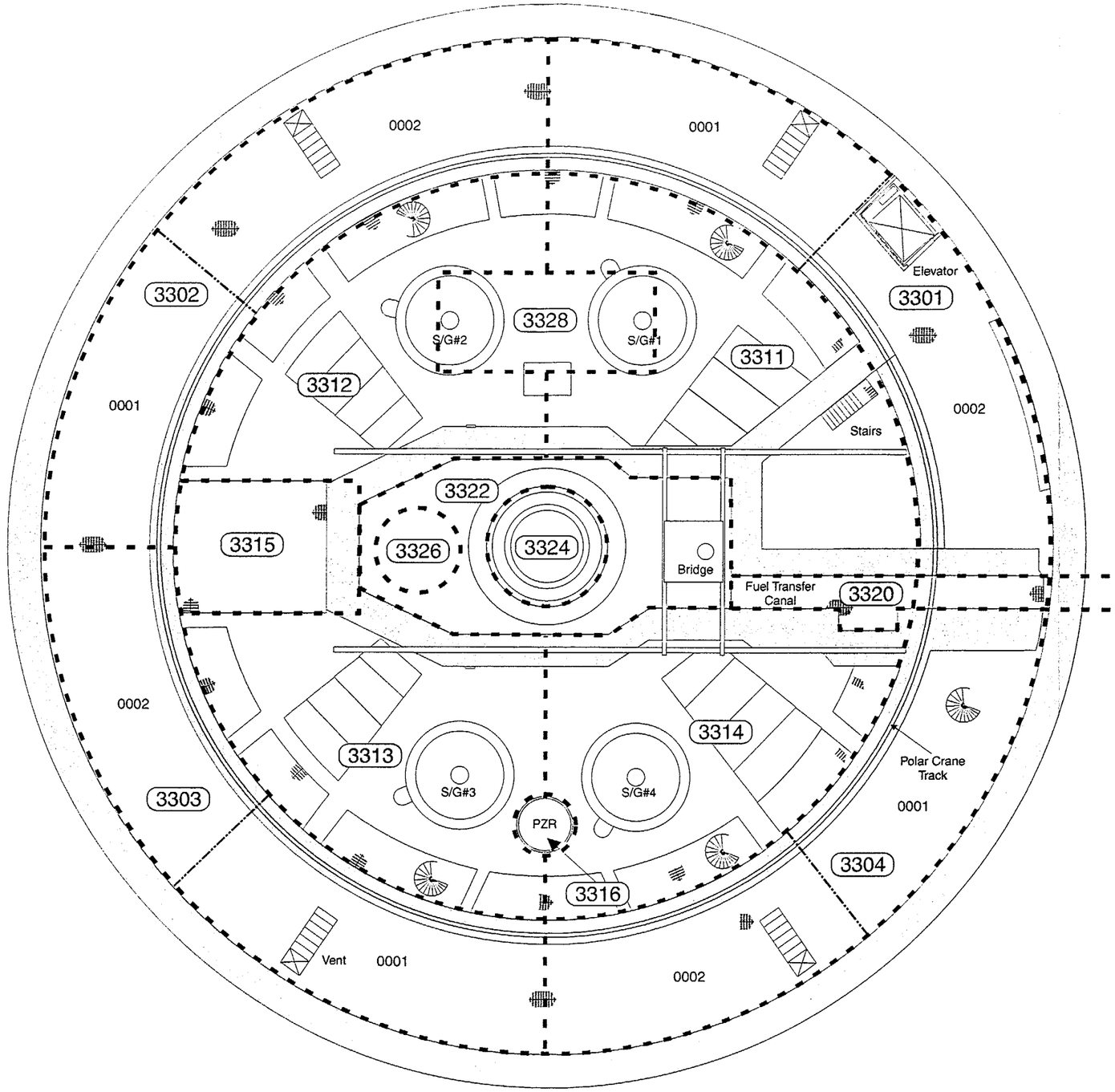
SURVEY UNIT NUMBER	####



Containment Building
Elevation 48'-6"
Figure 2-10

N

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
CONTAINMENT BUILDING
ELEVATION 48'-6"

INITIAL CREATION DATE:	05/06/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/15/2000
REVISION #:	2
MAP #:	GAD3300

LEGEND:

FLOOR WALL INTERSECTION	———
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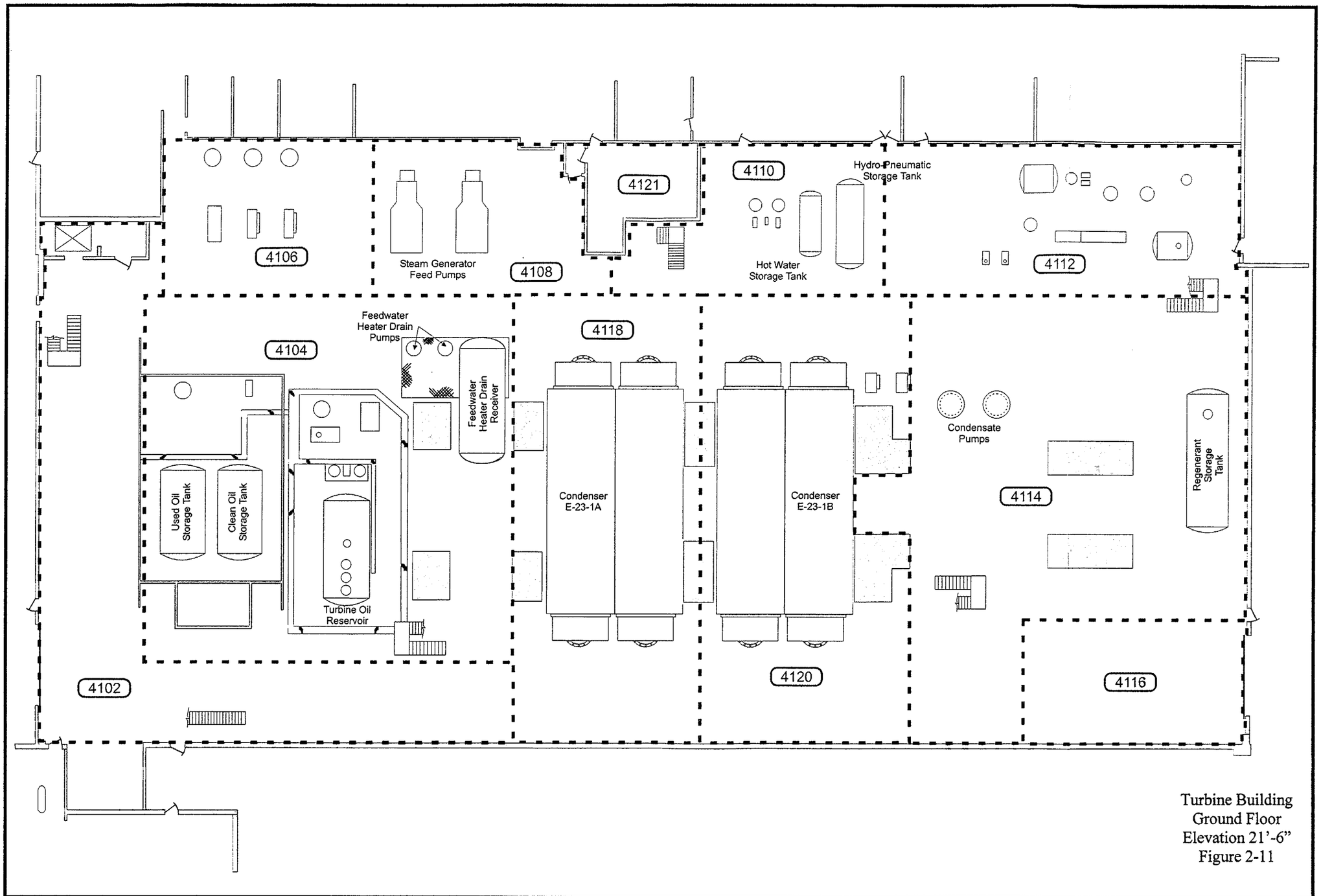
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—·—·—
SURVEY UNIT NUMBER	####



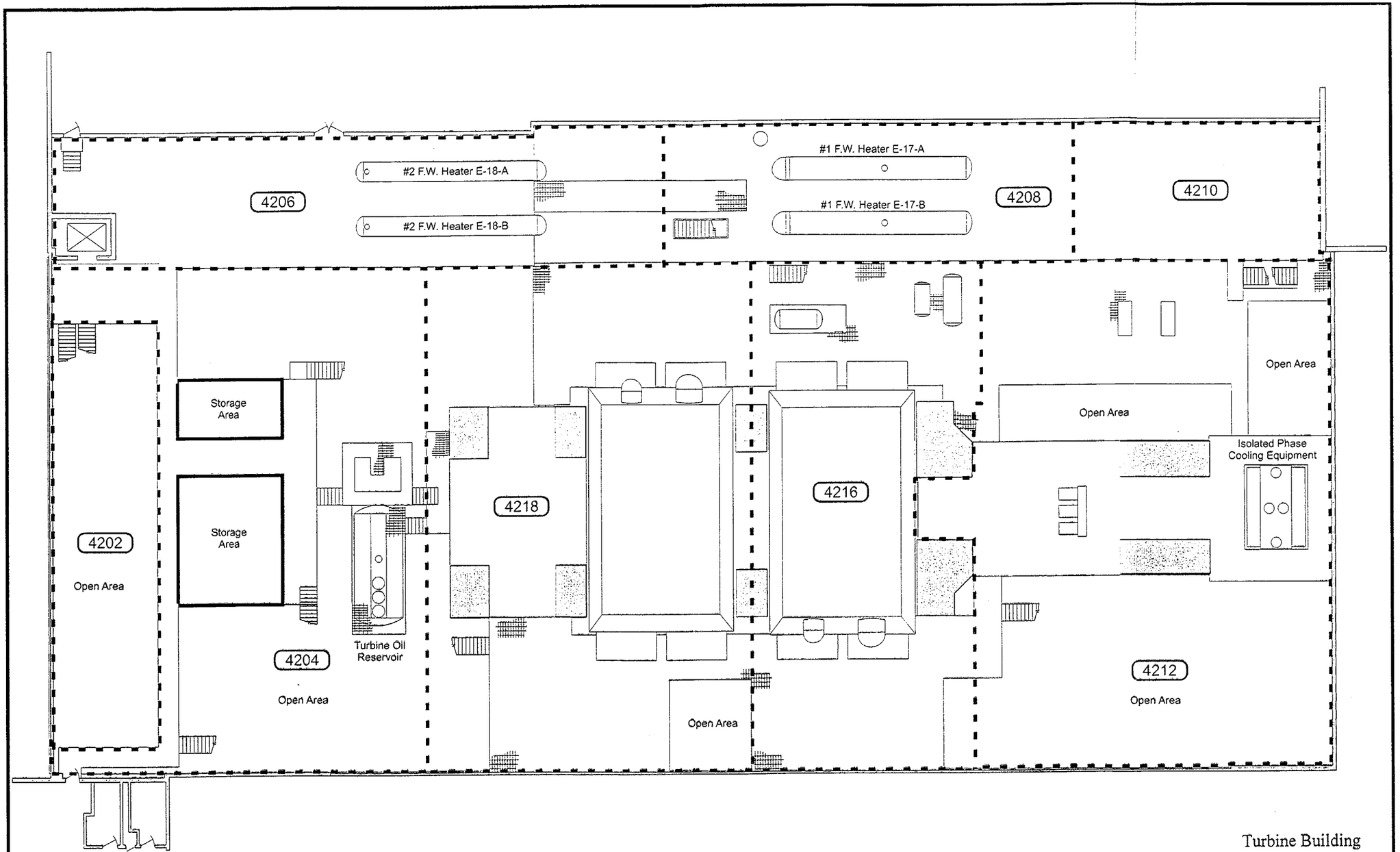
Turbine Building
Ground Floor
Elevation 21'-6"
Figure 2-11

Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
TURBINE BUILDING GROUND FLOOR
ELEVATION 21'-6"

INITIAL CREATION DATE:	05/27/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/22/2000
REVISION #:	3
MAP #:	GAD4100

LEGEND:	
FLOOR WALL INTERSECTION	———
SURVEY AREA BOUNDARY	- - -
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—·—·—
SURVEY UNIT NUMBER	####



Turbine Building
Second Floor
Elevation 37'-6"
Figure 2-12

Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
TURBINE BUILDING SECOND FLOOR
ELEVATION 37'-6"

INITIAL CREATION DATE:	05/28/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/22/2000
REVISION #:	1
MAP #:	GAD4200

LEGEND:

FLOOR WALL INTERSECTION	
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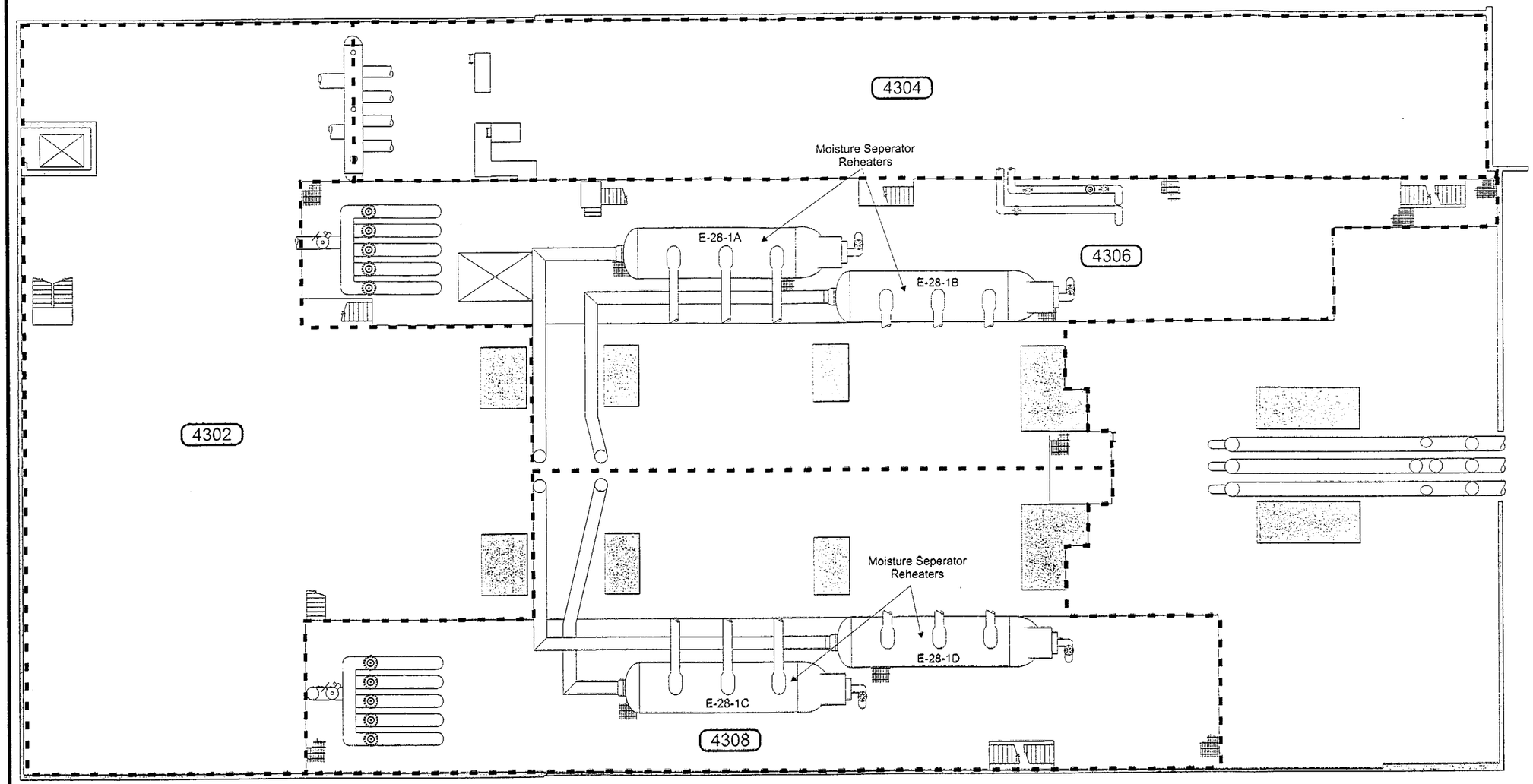
SURVEY AREA	#
SURVEY UNIT BOUNDARY	
SURVEY UNIT NUMBER	####



Turbine Building
Elevation 47'-0"
Figure 2-13

Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
TURBINE BUILDING
ELEVATION 47'-0"

INITIAL CREATION DATE:	06/03/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/22/2000
REVISION #:	1
MAP #:	GAD4300

LEGEND:

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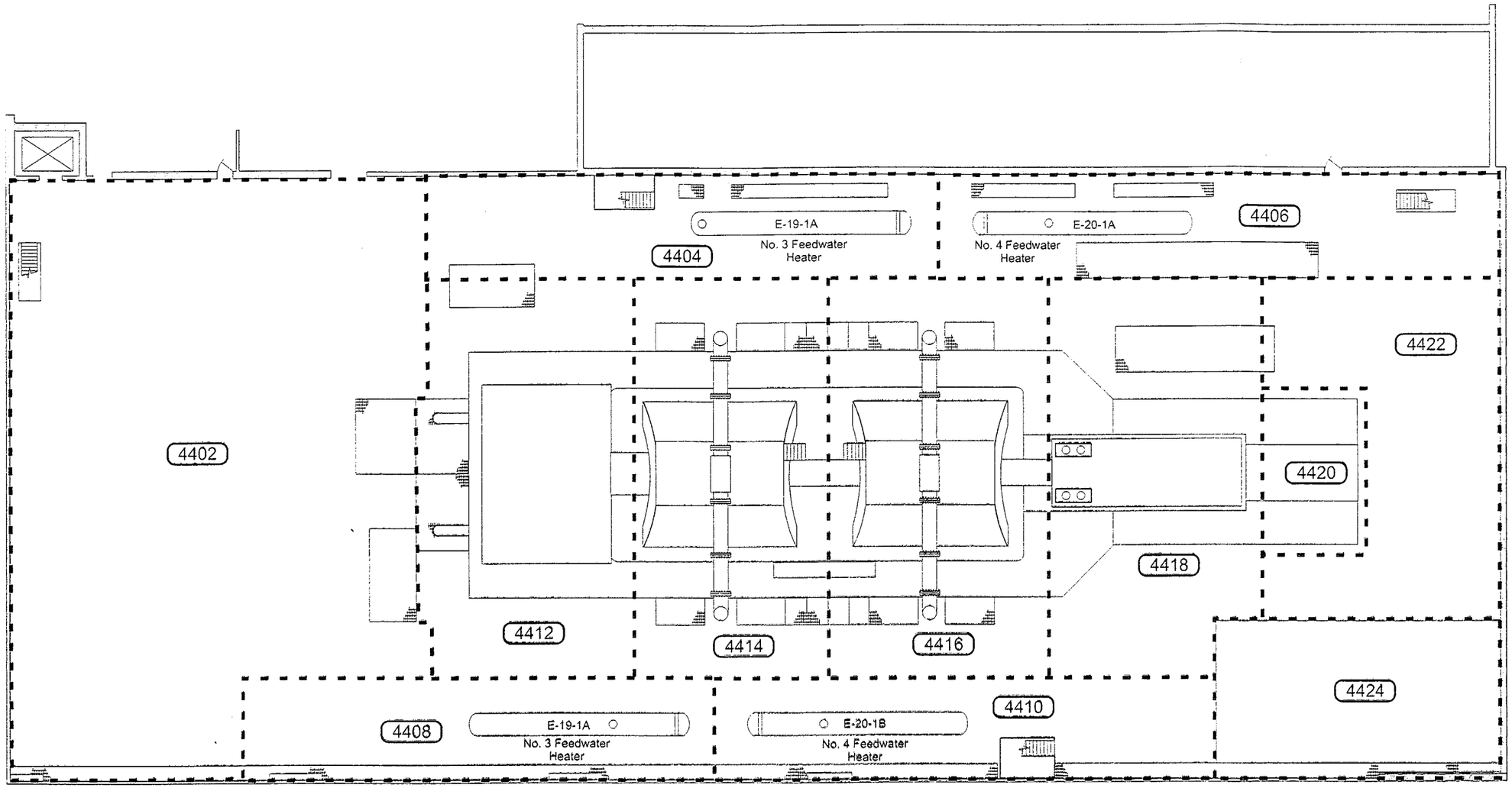
SURVEY AREA BOUNDARY	
SURVEY AREA	#
SURVEY UNIT BOUNDARY	
SURVEY UNIT NUMBER	####



Turbine Building
Elevation 59'-6"
Figure 2-14

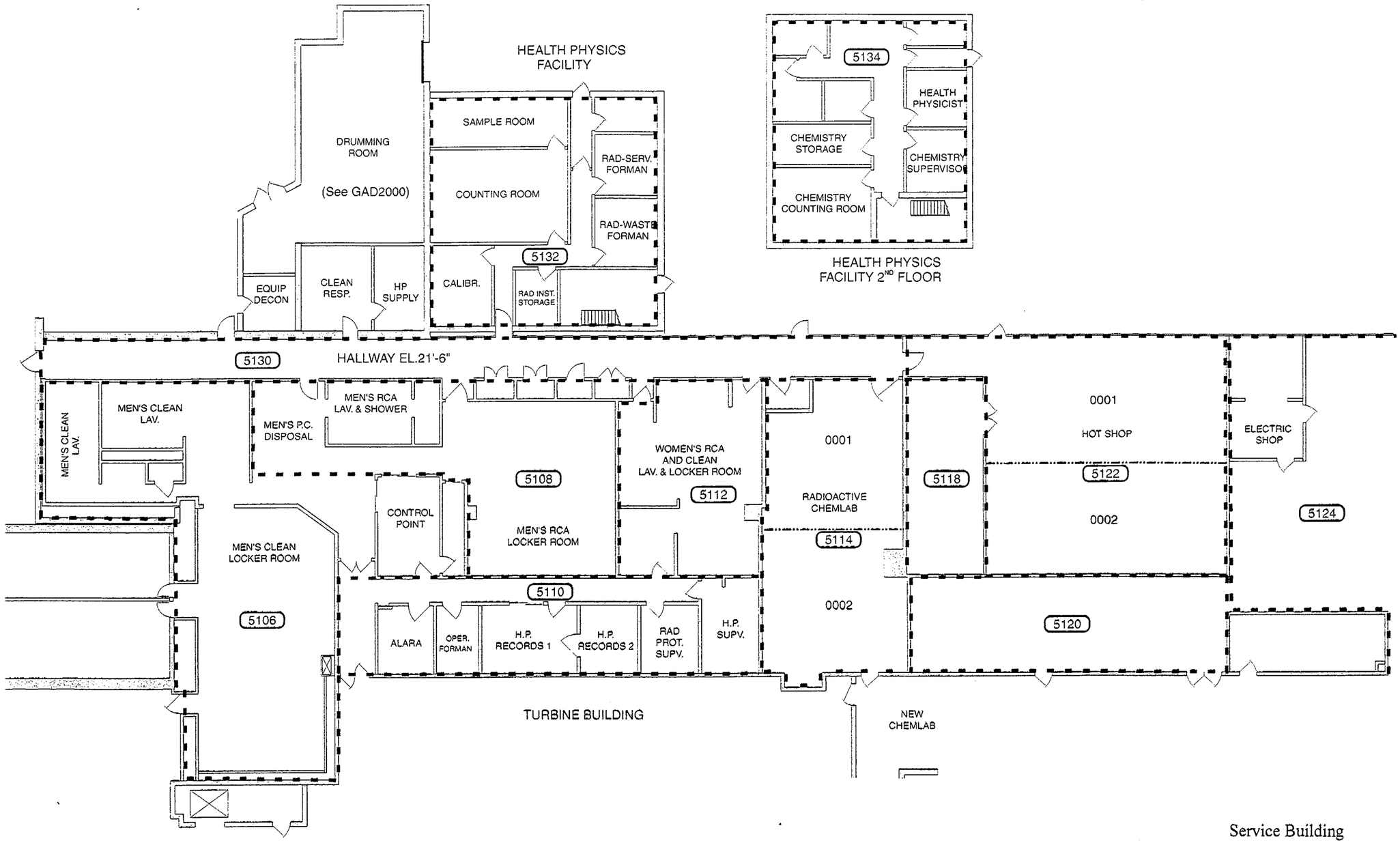
Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
TURBINE BUILDING
ELEVATION 59'-6"

INITIAL CREATION DATE:	06/05/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/22/2000
REVISION #:	1
MAP #:	GAD4400

LEGEND:

FLOOR WALL INTERSECTION	
SURVEY AREA BOUNDARY	
SURVEY AREA	#
SURVEY UNIT BOUNDARY	
SURVEY UNIT NUMBER	####



Service Building
Elevations 21'-6" & 35'-4"
Figure 2-15

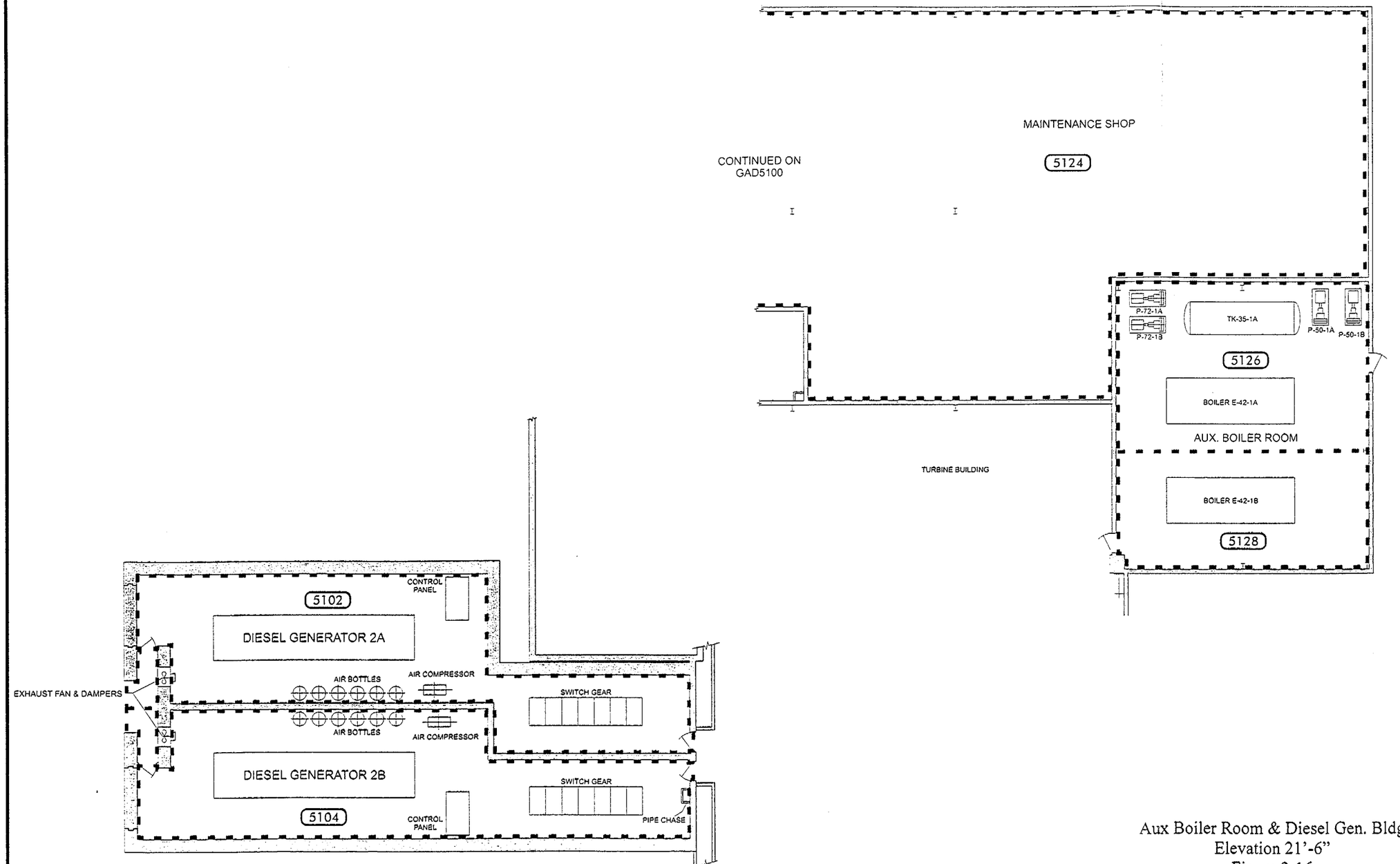
Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
SERVICE BUILDING
ELEVATION 21'-6" & 35'-4"

INITIAL CREATION DATE:	05/15/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/15/2000
REVISION #:	4
MAP #:	GAD5100

LEGEND:

FLOOR WALL INTERSECTION	———
SURVEY AREA BOUNDARY	- - -
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—·—·—
SURVEY UNIT NUMBER	####



Aux Boiler Room & Diesel Gen. Bldg
Elevation 21'-6"
Figure 2-16

Plant North

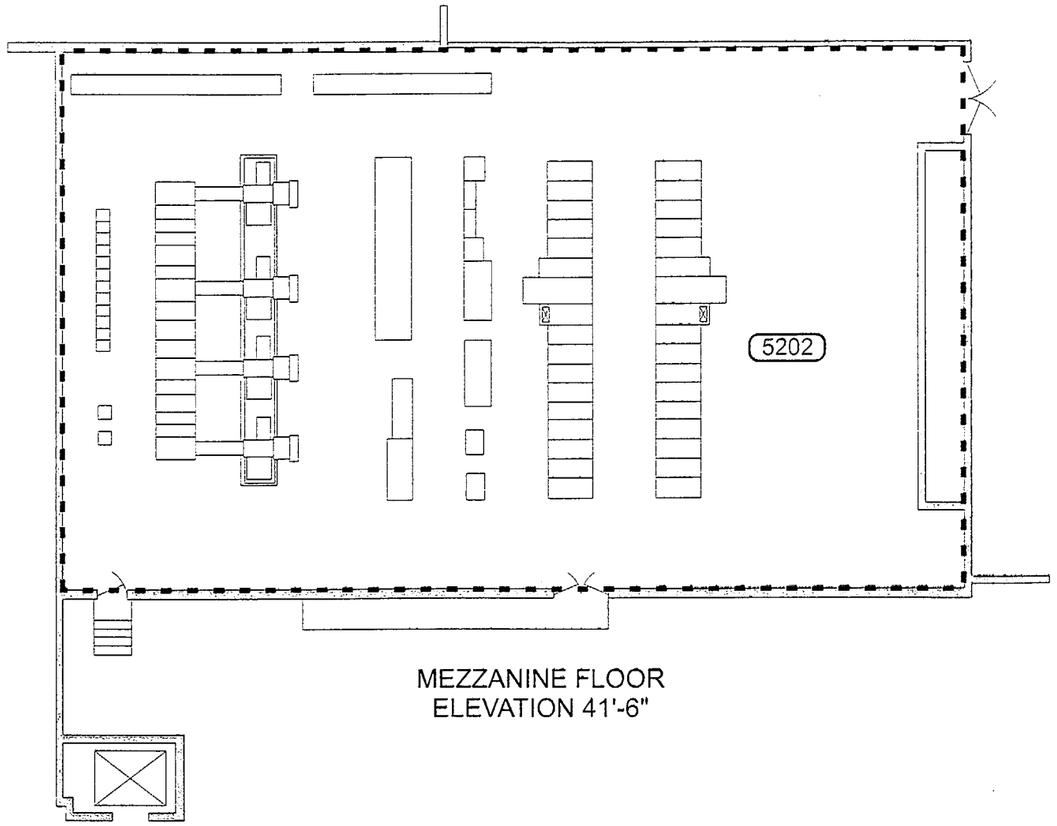
CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
AUX BOILER ROOM & DIESEL GEN. BLDG
ELEVATION 21'-6"

INITIAL CREATION DATE:	05/22/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/22/2000
REVISION #:	1
MAP #:	GAD5101

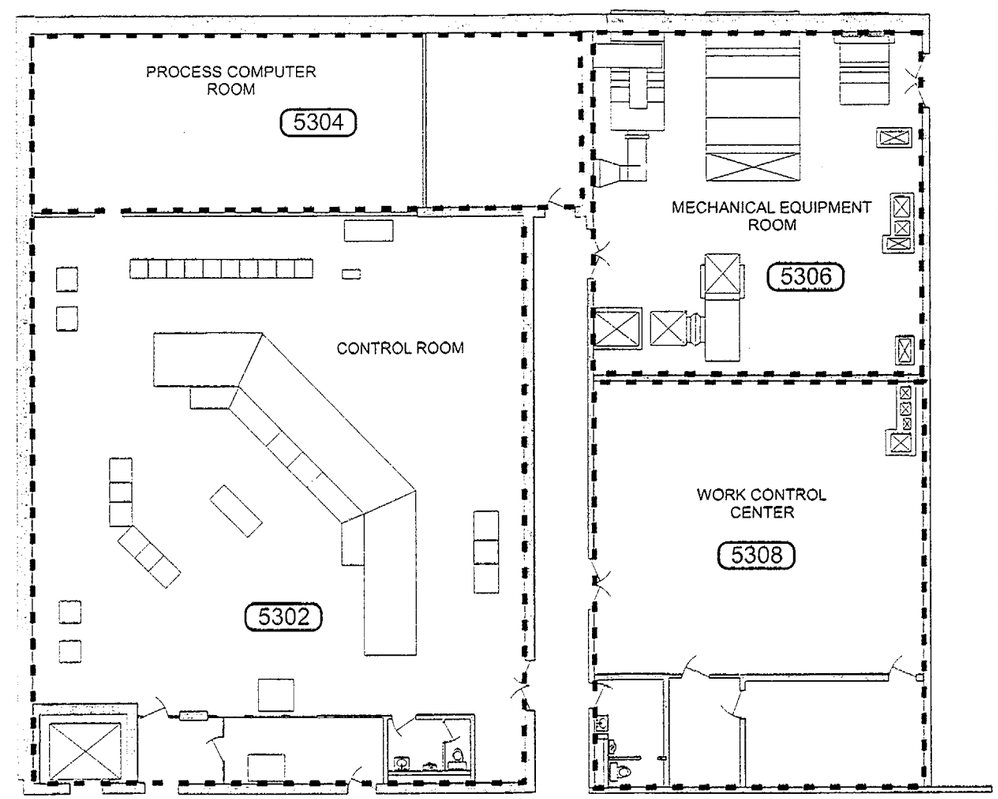
LEGEND:

FLOOR WALL INTERSECTION	
SURVEY AREA BOUNDARY	
SURVEY AREA	#
SURVEY UNIT BOUNDARY	
SURVEY UNIT NUMBER	####

Service Building Mezzanine
& Operating Floors
Elevations 41'-6" & 59'-6"
Figure 2-17



MEZZANINE FLOOR
ELEVATION 41'-6"



OPERATING FLOOR
ELEVATION 59'-6"

Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY

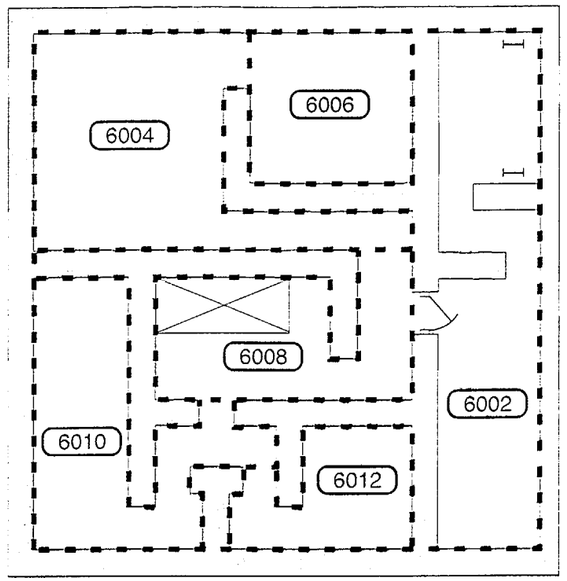
GENERAL ARRANGEMENT DRAWING

SERVICE BUILDING MEZZANINE & OPERATING FLOORS ELEVATION 41"-6" & 59'-6"

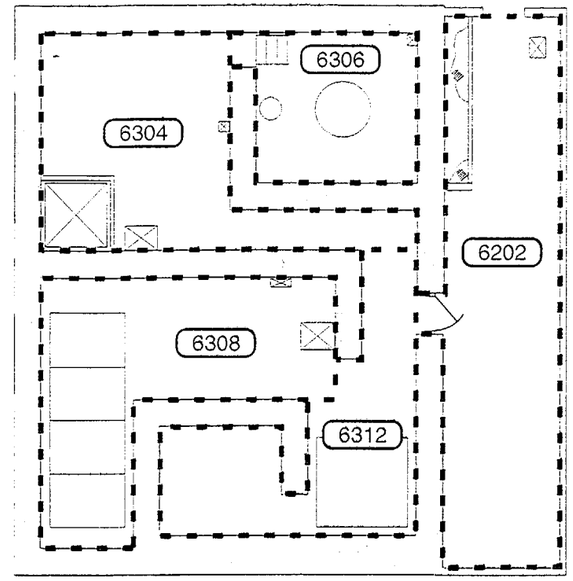
INITIAL CREATION DATE:	06/04/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/22/2000
REVISION #:	1
MAP #:	GAD5200

LEGEND:

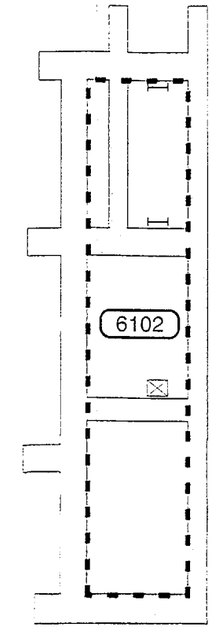
FLOOR WALL INTERSECTION	———
SURVEY AREA BOUNDARY	- - -
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—·—·—
SURVEY UNIT NUMBER	####



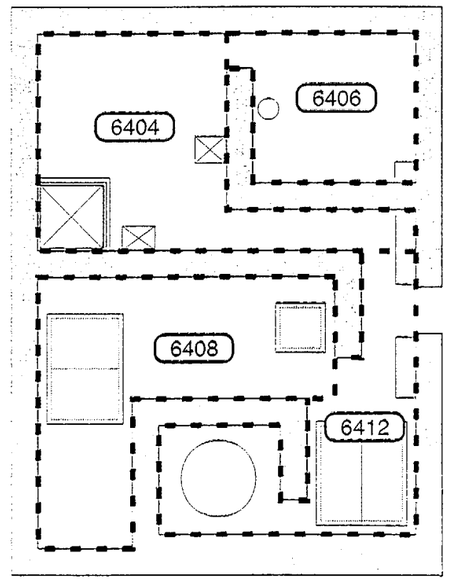
ELEVATION 0'-0"



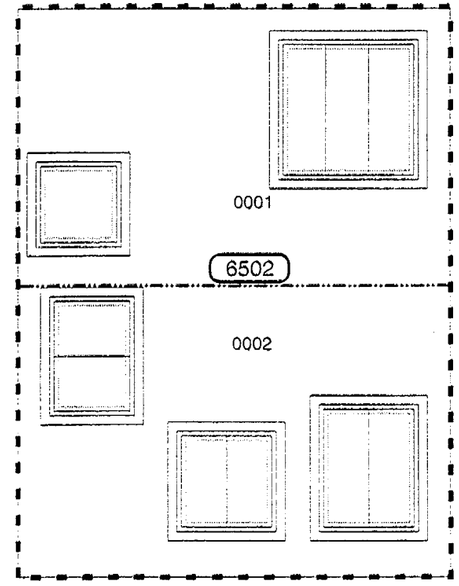
ELEVATION 18'-6" & 12'-6"



ELEVATION 12'-6"



ELEVATION 35'-6"



ROOF

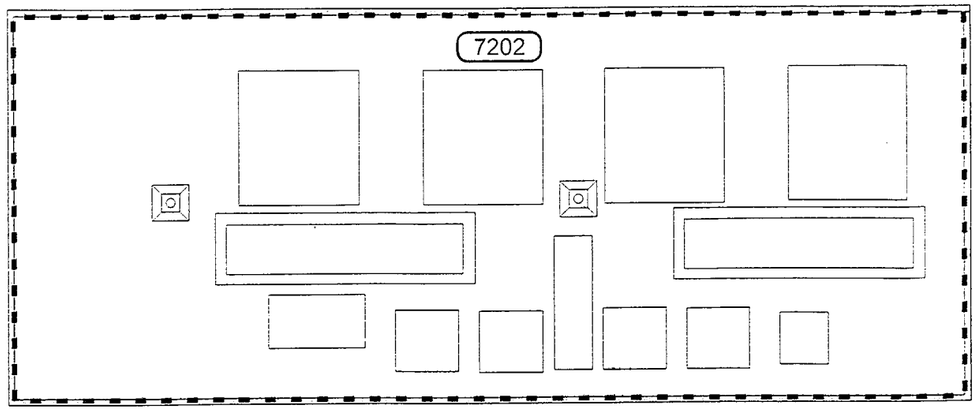
Waste Disposal Building
Elevations 0'-0", 12'-6", 18'-6", 35'-6" & Roof
Figure 2-18

Plant North

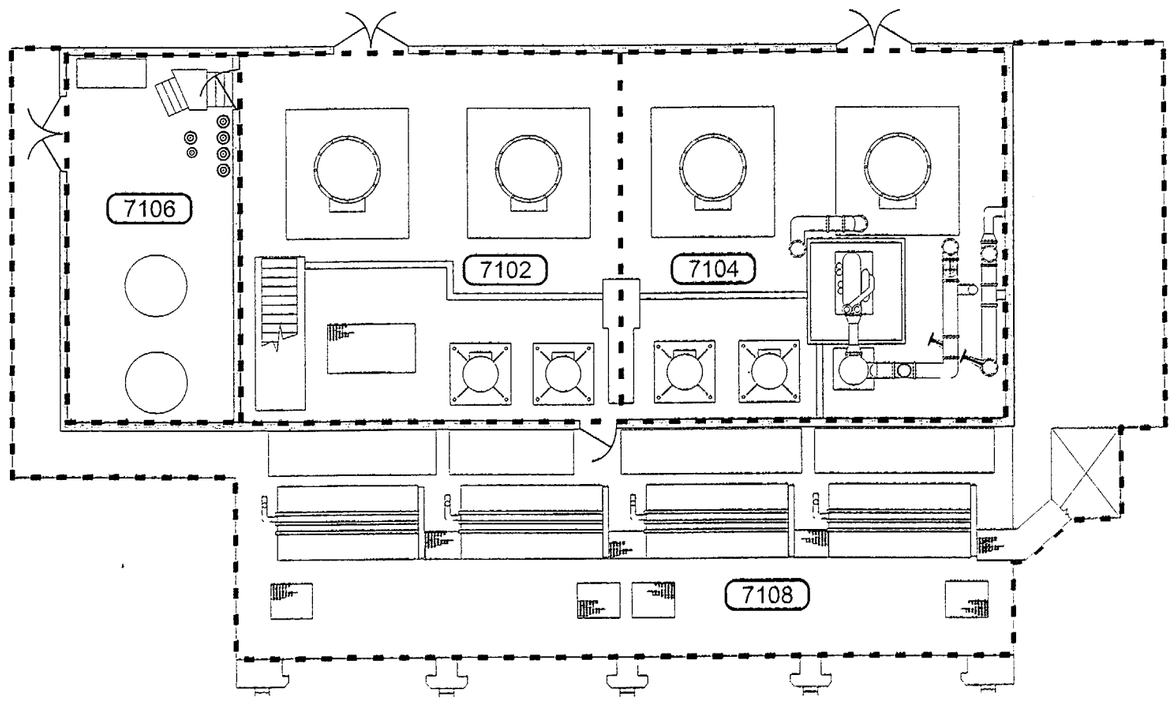
CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
WASTE DISPOSAL BUILDING
ELEVATION(S) 0'-0",12'-6",18'-6",35'-6" & ROOF

INITIAL CREATION DATE:	06/24/97
CREATED BY:	E. ALSING
REVISION DATE:	02/15/2000
REVISION #:	1
MAP #:	GAD6000

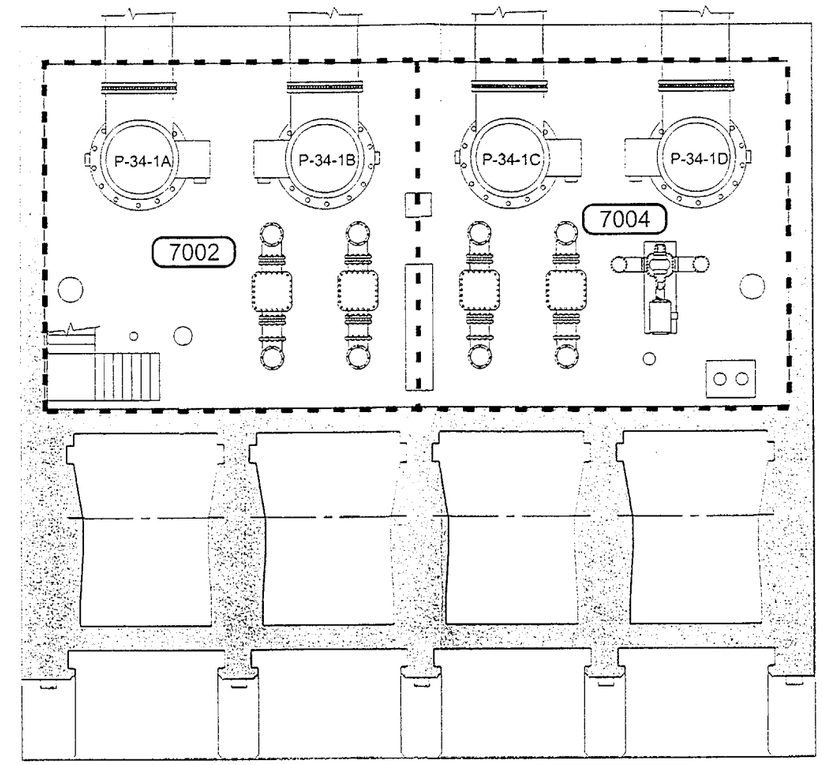
LEGEND:	
FLOOR WALL INTERSECTION	
SURVEY AREA BOUNDARY	
SURVEY AREA	#
SURVEY UNIT BOUNDARY	
SURVEY UNIT NUMBER	####



ROOF ELEVATION 36'-6"



ELEVATION 21'-6"



ELEVATION 8'-0"

Screenwell Building
Elevations 8'-0", 21'-6" & 36'-6"
Figure 2-19

Plant North

CONNECTICUT YANKEE ATOMIC POWER COMPANY
GENERAL ARRANGEMENT DRAWING
SCREENWELL BUILDING
ELEVATION(S) 8'-0", 21'-6" & 36'-6"

INITIAL CREATION DATE:	06/19/1997
CREATED BY:	E. ALSING
REVISION DATE:	02/22/2000
REVISION #:	1
MAP #:	GAD7000

LEGEND:

FLOOR WALL INTERSECTION	———
SURVEY AREA BOUNDARY	- - -
SURVEY AREA	#
SURVEY UNIT BOUNDARY	—·—·—
SURVEY UNIT NUMBER	####

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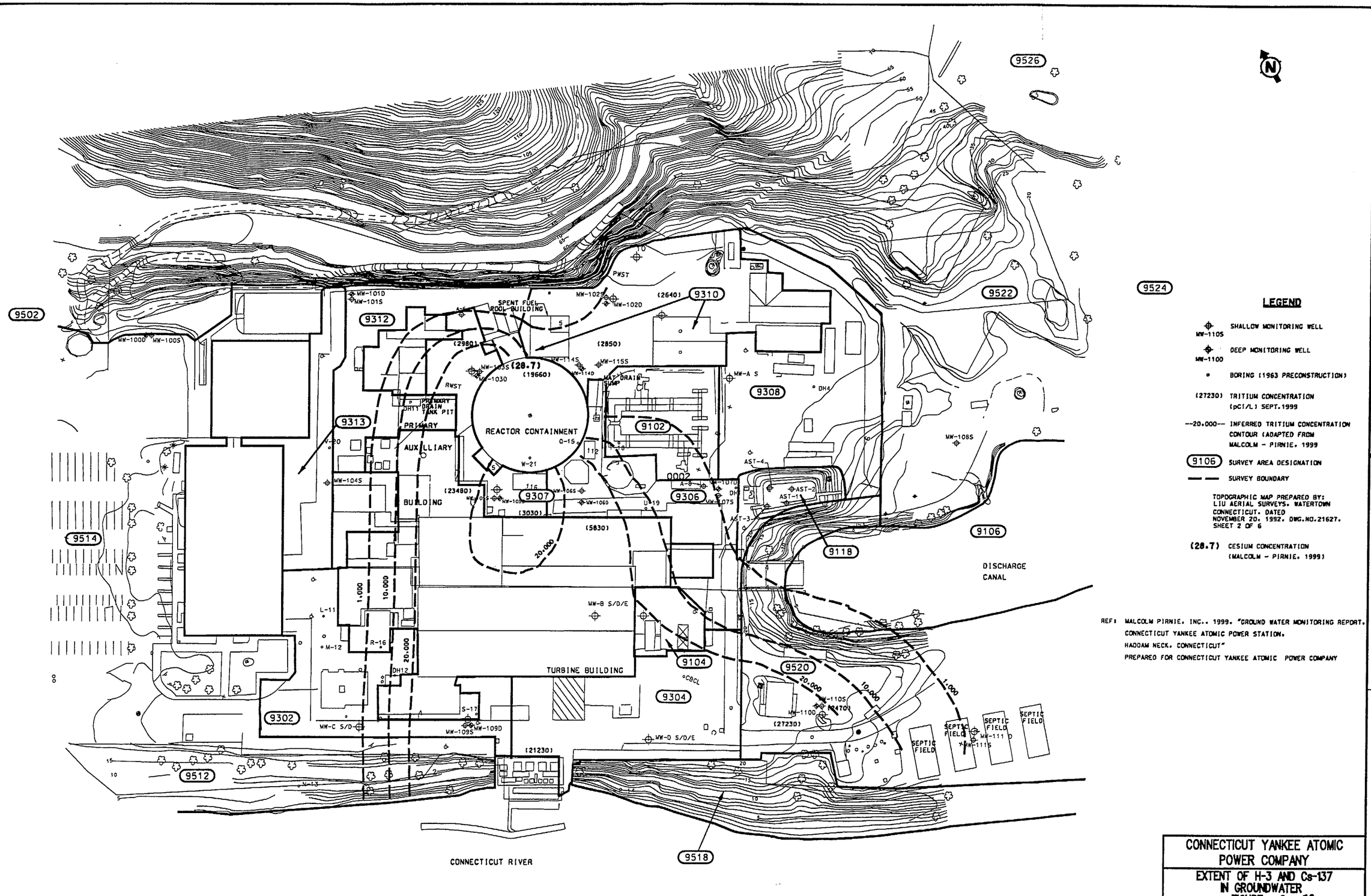

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3 IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

3.1 Introduction

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

CYAPCO's primary goals are to decommission the HNP safely and to maintain the continued safe storage of spent fuel. CYAPCO will decontaminate and dismantle the HNP in accordance with the DECON alternative, as described in the NRC's Final Generic Environmental Impact Statement. Completion of the DECON option is contingent upon continued access to one or more low level waste disposal sites. Currently, HNP has access to low-level waste disposal facilities located in Barnwell, South Carolina, and in South Clive, Utah.

CYAPCO is currently conducting active decontamination and dismantlement activities at the HNP site in accordance with the HNP PSDAR (Reference 3-2). Decommissioning activities are being coordinated with the appropriate Federal and State regulatory agencies in accordance with plant administrative procedures. In order to minimize the impact of ongoing decommissioning activities, a Spent Fuel Pool Island has been established to separate spent fuel storage functions from other plant functions and other decommissioning activities.

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

Table 3-1
Status of Major HNP Systems, Structures, and Components as of May 2000

SSC	Required for Wet Fuel Storage?	SSC Status
Reactor Coolant System	No	Partially Removed
Reactor Vessel Internals	No	Segmentation underway
Reactor Vessel	No	In place
Steam Generators	No	Removed
Reactor Coolant Pumps	No	In process of removal
Pressurizer	No	Removed
Chemical and Volume Control System	No	Partially removed
Safety Injection System	No	Partially removed
Residual Heat Removal System	No	Partially removed
Containment Spray System	No	Partially removed
Component Cooling Water System	No	Partially removed
Service Water System	No	Required path for processed water discharge
Spent Fuel Pool and Fuel Handling Equipment	Yes	In place
Spent Fuel Pool Cooling and Demineralizer System	Yes	In place
Condensate System	No	Partially removed
Feedwater System	No	Partially removed
Steam Generator Blowdown System	No	Partially removed
Primary Makeup Water System	No	Partially removed
Refueling Water Storage Tank	No	In place (unusable)
Plant Effluent Monitoring System	No	In place
Containment Ventilation System	No	In place
Fuel Building Ventilation System	Yes	In place
PAB Ventilation System	No	In place
Auxiliary Boiler	No	In place
Instrument and Service Air System	No	Partially removed
Gaseous Radioactive Waste System	No	In place
Solid Radioactive Waste System	No	In place

Table 3-1
Status of Major HNP Systems, Structures, and Components as of May 2000

SSC	Required for Wet Fuel Storage?	SSC Status
Liquid Radioactive Waste System	No	In place
Makeup Water System	Yes	In place
Radioactive Monitoring System	Yes	In place
Process Sampling System	No	Partially removed
Fire Protection System	Yes (portions)	Partially removed
Electrical Systems	Yes (portions)	In place
Containment Building	No	Some equipment removed. Decontamination activities are in progress.
Primary Auxiliary Building	No	Some equipment removed. Decontamination activities are in progress.
Fuel Building	Yes	In place
Turbine Building	No	Some equipment removed. Decontamination activities are in progress.
Service Building	No	Some equipment removed. Decontamination activities are in progress.

3.2 Spent Fuel Pool Island Activities

Plant closure activities were initiated following the decision to permanently cease HNP power operations in December 1996. At that time, CYAPCO performed evaluations of major plant systems, structures, and components (SSCs) to determine what function, if any, these SSCs would be expected to perform during the evolution to a decommissioned plant with a Spent Fuel Pool Island. Each major plant SSC was evaluated to determine if the SSC, in its entirety or any portion thereof, was required to support maintaining the spent fuel in a safe condition or was needed to perform a function during the decontamination and dismantlement of the plant.

Modifications were designed and installed to develop a Spent Fuel Pool Island. These modifications provide physical isolation between decommissioning activities and the Spent Fuel Pool. Additionally, they provide systems, independent from normal plant systems, for cooling, ventilation, and independent power supplies for the Spent Fuel Pool and its associated equipment and structure. A backup diesel generator has also been installed to maintain electrical capabilities during a loss of normal power supply. For planning purposes, it is anticipated that spent fuel will be stored in the Spent Fuel Pool until 2003, at which time it would be transferred to the onsite ISFSI pursuant to 10CFR72.210.

3.3 Completed and Ongoing Decommissioning Activities and Tasks

3.3.1 Overview

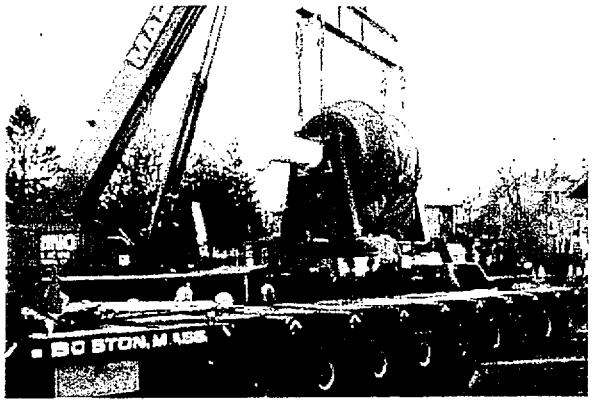
The major accomplishments described in the following sections are included in the LTP because they are similar to, and indicative of, the complexity of future activities to be performed. The successful completion of these activities demonstrates the project team's ability to safely and effectively decommission the HNP site. CYAPCO initiated decommissioning activities in 1997 with plant resources and subsequently decided that a Decommissioning Operations Contractor (DOC) would be used to complete the decommissioning, with CYAPCO personnel performing an oversight function. Bechtel began project DOC work in April, 1999, beginning with preparation and performance of Transition Period activities. A transition plan was developed to define infrastructure, programmatic, and procedural elements to be reviewed, modified as appropriate and approved for safe and effective transition to a DOC-performed project. During transition, the existing CYAPCO infrastructure was used as necessary to facilitate project performance. Transition was completed in November 1999.

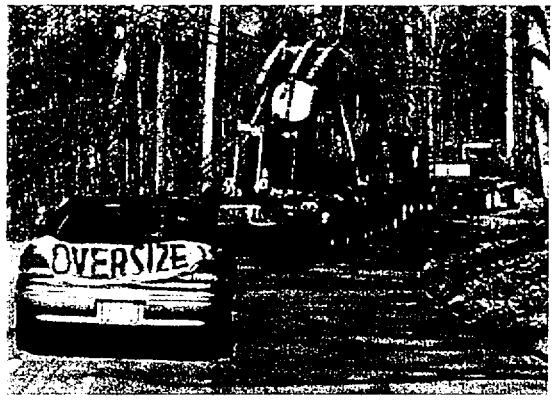
3.3.2 RCS Chemical Decontamination

One of the first major decommissioning activities at the HNP site was the chemical decontamination of the Reactor Coolant System, which was conducted as an ALARA initiative from July to August of 1998. The flush included the steam generators and pressurizer, as well as portions of the appended systems of letdown and charging, residual heat removal, loop fill and drains, and selected dead-leg piping. Approximately 131 Curies (not accounting for hard-to-measure radionuclides) were removed from the RCS during this process.

3.3.3 Turbine Rotors

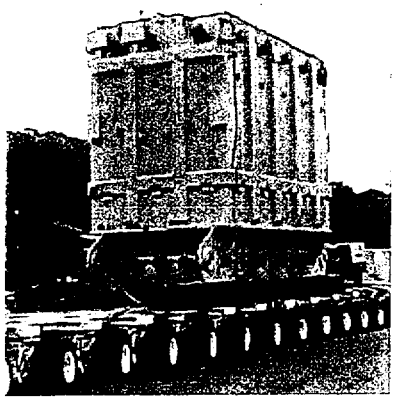
The Low Pressure Turbine Rotors were removed from the system and transported to the Palisades Nuclear Plant. These were the first large components removed from site. The High Pressure Turbine Rotor remains on site for eventual disposal.

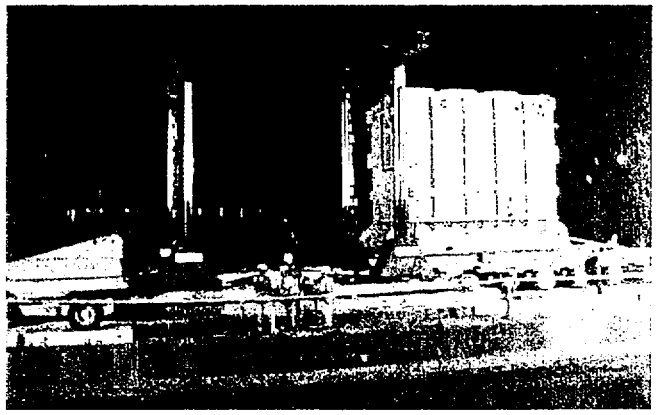




3.3.4 Removal of Spare Auxiliary Transformer

The spare auxiliary transformer has been removed from the HNP site. The transformer was stripped of the electrical components and cooling fans, for loading onto a multi-wheeled transporter and subsequent loading onto a barge for shipment on the Connecticut River. This activity proved to be a very valuable demonstration and test run for future activities involving onsite component handling and component removal by barge, such as those involving the steam generators, pressurizer, and eventually the reactor pressure vessel.

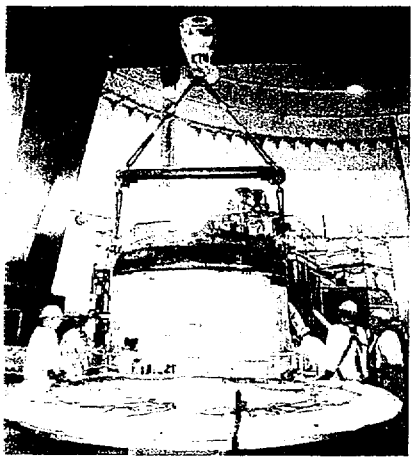


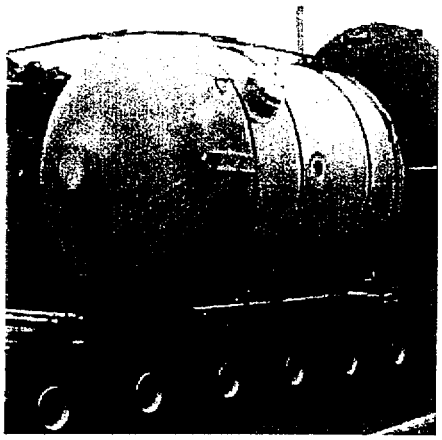


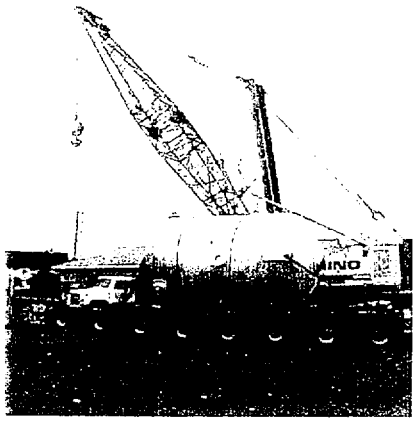
3.3.5 Removal of Steam Generator Steam Domes

The next major component removal task was the removal of the four steam generator steam domes. As the steam generators could not be removed in one piece without structural modification of the Containment Building, the generators were cut in the transition area at the feedwater ring. The steam domes were lifted and placed onto the Charging Floor where a cover plate was welded onto the bottom of the steam domes. Plates were also welded onto the top of the steam generator lower assembly. The nozzles were cut and capped. The domes were then rigged off of the Charging Floor, through the Containment Equipment Hatch, and onto a multi-wheeled transporter.

After final radiological survey, the domes were trucked approximately 13 miles to a local rail-head in Portland, Connecticut. The domes were then transported by rail to a licensed radioactive material handler in Tennessee.

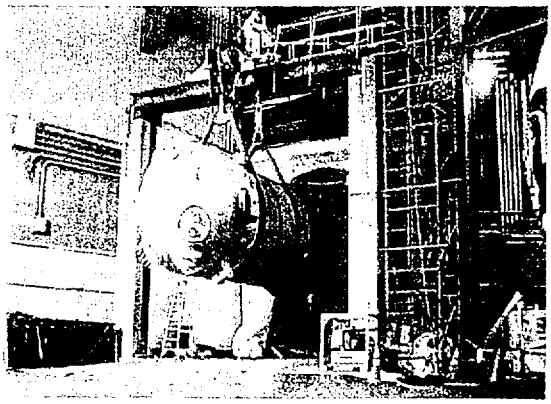


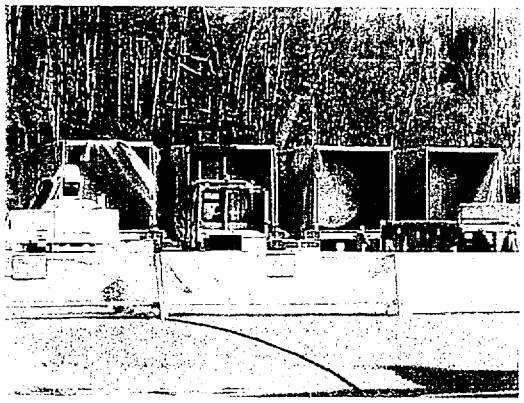




3.3.6 SGLA Removal from Containment

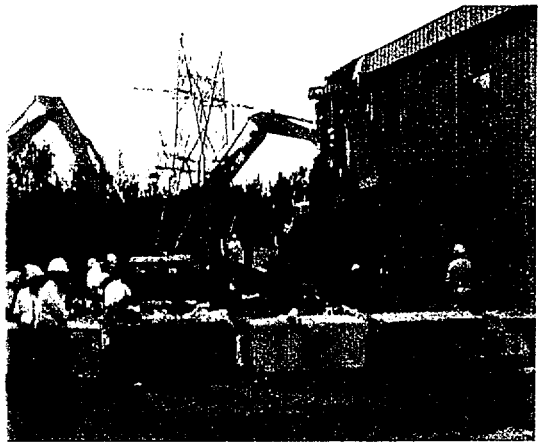
The Steam Generator Lower Assemblies (SGLAs) were removed with the intent of being placed on barges and shipped for disposal. However, due to low water levels on the Savannah River, barge transportation was not immediately achievable. Therefore a temporary laydown area was established. Potential dose to the on-site workers was reviewed and appropriate shielding was installed. Alternatives for shipment and disposal of the SGLAs are being evaluated.





3.3.7 Dismantlement of Outbuildings

Several maintenance and fabrication shops were demolished and removed from the site. Those for which no radioactivity was detected were released from the site as clean material. Material was released only after meeting approved survey and release processes or was sent to an offsite licensed radioactive material handler as appropriate. The removal of these buildings was to facilitate large component removal.



Security functions have been relocated, and the old Security and Fitness Center has been demolished and released, using currently approved survey and release process. The Engineering and Instrument and Controls Buildings have also been demolished.

3.3.8 Additional Activities

Additional activities that were completed or are ongoing include, but are not limited to the following:

- Continued assessment of the functional requirements for plant systems, structures, and components.

  Plant systems, structures, and components needed to support safe storage of the spent fuel, support spent fuel pool cooling, and facilitate ongoing plant activities have been identified.

- Isolation/de-energization and removal of plant systems, structures, and components.

  A comprehensive plant isolation/de-energization program was developed and is being implemented as described in Section 3.4.1.6. Systems, structures, and components not required to support decommissioning or spent fuel storage continue to be decontaminated and/or removed in accordance with the possession only license and approved plant procedures. Modifications were performed to provide for temporary systems and for construction power to support decommissioning activities, as needed.

- Planning for the Steam Generator lower assembly removal and transport off site.

  The Department of Transportation (DOT) exemption request was submitted in September 1998 and was revised in June 1999 (Reference 3-3). The DOT exemption was approved in July 1999. Concurrently, a request for receipt of the steam generator lower assemblies was submitted to the

South Carolina Department of Health and Environmental Control (DHEC) and was approved in September 1999.

- Planning for pressurizer removal and transport off site.

  The Department of Transportation (DOT) exemption request was submitted in January of 2000. (Reference 3-4) A subsequent submittal to the South Carolina DHEC will be made.

- Planning for the Reactor Vessel removal and transport to the Chem Nuclear facility at Barnwell, South Carolina.

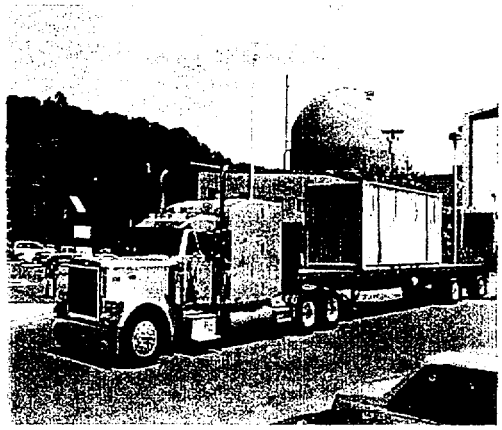
  Transportation, design, and licensing activities are being pursued to develop a transportation system under DOT rules to deliver the Reactor Pressure Vessel to Barnwell (Reference 3-5). The DOT exemption request was submitted to the DOT in March 2000. A subsequent submittal to the South Carolina DHEC will be made.

- Reactor Vessel Internals Segmentation

  The eventual disposition of the Reactor Pressure Vessel (RPV), as stated above, is in the Chem Nuclear facility in Barnwell, South Carolina. This facility limits a package radioactivity to 50,000 Curies. The activity of the RPV, including internals, is estimated to be greater than approximately 800,000 Curies. (Reference 3-6) Therefore, segmentation, removal, and onsite storage of the greater than Class C (GTCC) internal components (primarily the center section of core barrel, baffles, and lower core support plate) is being performed. This will reduce the RPV inventory to less than 50,000 Curies for shipment and disposal. The GTCC material will be stored on site until an approved facility is available for disposal of the GTCC waste.

- Continuing Removal of Wastes and Demolition Materials

  Plant activities working concurrently are radiological and non-radiological waste removal. These include removal of contaminated soils, asbestos abatement program, and removal of other hazardous and non-hazardous wastes.



3.4 Future Decommissioning Activities and Tasks

3.4.1 Overview

Decontamination and dismantlement activities are currently expected to continue until the spring of 2003, with exception of those activities related to the operation of the Spent Fuel Pool Island. As previously noted, the transfer of spent fuel from the Spent Fuel Building to dry storage canisters at the onsite ISFSI is anticipated to begin in 2003 and to be complete in 2004.

The remaining dismantlement and decontamination activities can be grouped into several classifications, the implementation of which may overlap. The current phase includes major component removal. The subsequent phases include contaminated system removal, clean systems removal, decontamination of site buildings and site remediation. The phases may be implemented on an area-by-area basis. Under this approach, often only a part of a system will be removed, while the remaining portions await removal in a subsequent phase. The remaining contaminated systems and components will be decontaminated or removed, packaged, and either shipped to an offsite processing facility, shipped directly to a low-level radioactive waste disposal facility, or handled by alternate methods in accordance with applicable regulations.

Decontamination of plant structures may be completed concurrently with equipment removal. Decontamination of structures may include a variety of techniques ranging from water washing to surface material removal (see Section 4 for a discussion of remediation methods). Contaminated structural material may be packaged and either shipped to a processing facility, or shipped directly to a low level radioactive waste disposal facility. Alternative disposal methods, in accordance with applicable regulations, may also be used.

The following sections provide a general description of the remaining decommissioning activities for the HNP site. These activities involve the reduction of radioactivity to as low as reasonably achievable (ALARA) levels, not to exceed 25 mrem/yr TEDE, allowing for release of the site for unrestricted use. This information provides the basis for development of programs and procedures for ensuring safe decommissioning and a basis for detailed planning and preparation of decontamination and dismantlement activities.

Following the removal or decontamination of contaminated systems, components, and structures, a comprehensive final radiation survey will be completed as described in Section 5. This survey will verify that residual radioactivity has been reduced to sufficiently low levels, as stipulated in 10CFR20.1402 (Reference 3-7), to allow the release of the site for unrestricted use. Upon completion of the final survey for a plant area, CYAPCO will document the results of that survey and will make them available for NRC inspection. Following completion of the final status survey and in the absence of any NRC inspection action finding the report deficient, CYAPCO may release buildings for demolition.

3.4.1.1 Detailed Planning and Engineering Activities

Detailed project plans will continue to be developed in accordance with design control procedures to support the decontamination and dismantlement activities. These plans are used to develop work packages, support ALARA reviews, aid in estimating labor and resource requirements, and track decommissioning costs and schedule.

Work packages are used to implement the detailed plans and provide instructions for actual field implementation. The work packages address discrete units of work and include appropriate hold and

inspection points. Administrative procedures control work package format and content, as well as the review and approval process.

3.4.1.2 General Decontamination and Dismantlement Considerations

As has been the current practice and in accordance with the HNP PSDAR, the following general decontamination and dismantlement considerations, as applicable, will continue to be incorporated into decommissioning work packages during the decontamination and dismantlement period. Specific considerations are described herein.

Dismantlement activities are reviewed to ensure that they do not impact the safe storage of spent fuel in the Spent Fuel Pool. Configuration control packages are implemented in accordance with administrative controls and require evaluations in accordance with the requirements of 10CFR50.59.

Temporary shielding and other standard Health Physics/Radiation Protection practices will be implemented to address ALARA considerations during decommissioning activities. These are described in further detail in subsequent paragraphs. Some dismantlement activities may be performed under water for shielding purposes, as well as for contamination control purposes.

As currently practiced at HNP and in accordance with the HNP PSDAR, the capability to isolate or to mitigate the consequences of a radioactive release will continue to be maintained during decontamination and dismantlement activities. Isolation is the closure or control of penetrations and openings to restrict transport of radioactivity to the environment. However, this consideration does not preclude the removal of penetrations and attachments to Containment, provided that effluents will be controlled or monitored.

Airborne radioactivity will be controlled considering the following:

- Operation of the appropriate portions of the containment ventilation and purge system, or an approved alternate system, during decontamination and dismantlement activities in the Containment Building;

- Operation of the appropriate portions of the Primary Auxiliary Building, Service Building, and Spent Fuel Building ventilation systems, or an approved alternate system, during decontamination and dismantlement activities in these buildings;

- Use of local high-efficiency particulate air (HEPA) filtration systems for activities expected to generate airborne radioactive particulates (e.g., grinding, chemical decontamination, or thermal cutting of contaminated components).

Work activities are planned to minimize the spread of contamination. Contaminated liquids are contained within existing or supplemental barriers and may be processed by a liquid waste processing system prior to release, if necessary. To minimize the potential for spread of contamination, the following considerations will continue to be evaluated for incorporation into the planning of decommissioning work activities:

- Covering of openings in internally contaminated components to confine internal contamination;

- Use of contamination control barriers as appropriate around activities that may result in airborne contamination during cutting and removal processes;

- Decontamination and dismantlement of contaminated systems, structures, and components by decontamination in place, removal and decontamination, or removal and disposal;
- Removal of contaminated supports in conjunction with equipment removal or decontamination of supports in conjunction with building decontamination;
- Removal of contaminated systems and components from areas and buildings prior to structural decontamination (block shield walls, or portions of other walls, ceilings, or floors may be removed to permit removal of systems and components.);
- Removal or decontamination of embedded contaminated piping, conduit, ducts, plates, channels, anchors, sumps, and sleeves during area and building structural decontamination activities;
- Use of local or centralized processing and cutting stations to facilitate packaging of components removed in large pieces; and
- Removal of small or compact plant components and parts intact, where feasible. (This includes most valves, smaller pumps, some small tanks, and heat exchangers. These components could then be decontaminated in whole or part, and disassembled or segmented in preparation for disposal or release.)

3.4.1.3 Decontamination Methods

Contaminated systems and components are typically removed and sent to an offsite processing facility, sent to a low-level radioactive waste disposal facility, or decontaminated onsite and released. Other decontamination methods typically include wiping, washing, vacuuming, scabbling, spalling, and abrasive blasting. Selection of the preferred method is based on the specific situation. Other decontamination technologies may be considered and used as appropriate.

Hand wiping may be used to remove loose surface contamination. Airborne contamination control and waste processing systems are used as necessary to control and monitor releases. If structural surfaces are washed to remove contamination, controls are implemented in accordance with approved procedures to ensure that wastewater is collected for processing by liquid waste processing systems.

Tanks and vessels are evaluated and, if ALARA, are flushed or cleaned to reduce contamination and to remove sediments prior to sectioning and/or removal. In cases where tanks, vessels, and piping are major contributors to dose rates, the ALARA principle will be applied to determine removal sequence. Precautions are taken to ensure that liquid inadvertently discharged from the tank is captured for processing by a liquid waste processing system. Sediment removed from the tank may be stabilized prior to shipment or may be shipped without stabilization in an approved condition. Wastewater is processed and/or sampled and analyzed in accordance with the National Pollutant Discharge Elimination System (NPDES) permit and Offsite Dose Calculation Manual (ODCM) before being discharged.

Concrete that has surface or near-surface contamination may be cleaned, if necessary, to meet applicable derived concentration guideline limits (DCGLs) and ALARA. Activated concrete may be removed as necessary to meet DCGLs (and ALARA considerations) or may be sent to a low-level waste (LLW) disposal facility, or handled by other methods in accordance with applicable regulations. The removal of concrete may be performed using methods that control the removal depth to minimize the waste volume produced. Vacuum removal of the dust and debris with HEPA filtration of the effluent may be used to minimize the spread of contamination and reliance on respiratory protection measures. Section 4 also discusses remediation methods.

3.4.1.4 Contaminated System Dismantlement

Dismantlement methods for contaminated systems can be divided into two basic types: disassembly and cutting. Disassembly generally means removing fasteners and components in an orderly non-destructive manner (i.e., the reverse of the original assembly). Cutting methods include flame cutting, abrasive cutting, and cold cutting. Abrasive water-jet cutting is being used to segment the Reactor Pressure Vessel internals.

Flame cutting includes the use of oxyacetylene and other gas torches, carbon arc torches, air or oxy arc torches, plasma arc torches, cutting electrodes, or combinations of these. Most of the torches can either be handheld or operated remotely. Abrasive cutting includes the use of grinders, abrasive saw blades, most wire saws, water lasers, grit blast, and other techniques that wear away metal. Cold cutting includes the use of bandsaws, bladesaws, mechanical disintegration methods, drilling, machining, shears, and bolt/pipe/tubing cutters. Selection of the preferred method depends on the specific situation. Other dismantlement technologies may be considered and used if appropriate.

Dismantling of systems includes the removal of valves and piping for disposal. Most valves can be removed with the piping. Larger valves and valves with actuators may be removed separately for handling purposes.

3.4.1.5 Removal Sequence and Material Handling

Removal sequences may be dictated by access and material handling requirements or by personnel exposure considerations. In some cases, a top-down approach may be used. Using this approach, materials and structures at the highest elevations are removed first to allow access to components in lower levels. In other cases, different approaches may prove more efficient.

In some cases, the first items removed are those that are not contaminated, or are only slightly contaminated, to preclude contamination by other equipment removal activities. However, personnel exposure considerations may not always allow this option. The ALARA principle will be applied by removal of hot spot items (e.g., piping with high dose rates) prior to other work. Where non-contaminated equipment or piping is not removed first, covers or other protection methods may be used to prevent cross-contamination.

Where rapid cutting techniques are available, pipes and equipment can be sectioned into pieces that are manageable using light rigging or by manual lifting. Where slow cutting techniques are used, the largest manageable pieces will typically be freed and further reduced in size at a more convenient location.

The plant is equipped with multiple cranes, hoists, and lifting and transport systems. These systems can be used to lift and transport components and equipment to support plant decommissioning activities. Forklifts, mobile cranes, front-end loaders, and other lifting and transport devices can also be used for plant decommissioning activities. The major installed plant cranes, hoists, and lifting and transport devices that may be available to support decommissioning include:

- Containment Building Polar Crane;
- Turbine Building Crane; and
- Yard Crane

Inspection requirements for these cranes meet the specific requirements of plant procedures. Other rigging equipment will be inspected and verified in accordance with procedures. Smaller rigging equipment (chain hoists, etc.) will be inspected and verified to be in good working condition prior to use.

The Containment Building Polar Crane is capable of reaching most locations inside the Containment Building and handling large, heavy loads. This crane has already been used for the Steam Generator Domes, Steam Generator Lower Assemblies and other smaller loads. After being modified, this crane may also be used for the Reactor Pressure Vessel removal. The Yard Crane has been used for material handling.

Installed cranes and hoists may be used in conjunction with temporary or mobile lifting and transport devices to support decommissioning. The installed plant cranes, hoists, and other lifting devices may be dismantled when they no longer are required to support decommissioning activities.

3.4.1.6 System Isolation/De-energization

Systems or components will continue to be deactivated prior to decontamination and dismantlement. In general, isolation/de-energization is implemented by mechanical isolation of interfaces with operating plant systems, draining piping/components, and de-energizing electrical supplies. Combustible materials (e.g., charcoal from filters, lube oil) are removed from the deactivated components, where practical. Chemicals used in, or resulting from, decommissioning activities are controlled in accordance with the plant chemical safety program. Plant critical drawings are updated to indicate deactivated portions of systems, and plant procedures are modified accordingly to reflect the changes.

Isolation/de-energization of plant systems is administratively controlled by approved procedures. Isolation/de-energization plans are established to implement the desired system valve lineup changes and electrical isolations. The design change process is used to remove components, lift electrical leads, install electrical jumpers, cut and cap piping systems, or install blank flanges.

Plant procedures also provide controls over the operation of deactivated system boundary valves. As additional systems are deactivated, existing isolation boundaries are re-evaluated and changed, as necessary, to reflect the new plant condition. Boundary valves are tagged for identification.

3.4.1.7 Temporary Systems Required to Support Decommissioning

Decontamination and dismantlement of systems, structures, and components often require the removal of interferences. Removal of some of these interferences may eliminate power, service air, and other services needed to support decommissioning. Also, use of installed plant systems for decommissioning support may become impractical, due to the risk of encountering energized systems or circuits. Temporary services and systems are being provided to support decommissioning activities. Temporary modifications to plant structures, systems, and components are controlled by design control procedures.

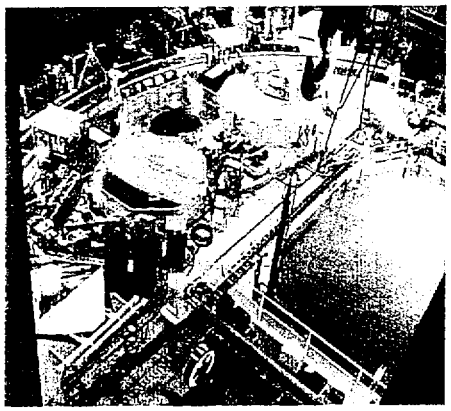
Portable load centers are powered from motor control centers, plant load centers, or the yard loop. These portable load centers can supply cutting and hoisting equipment, temporary lighting, or other power needs. Service air can be provided by portable air compressors using hoses or temporary air manifolds. Demineralized water is available from portable demineralizer skids or portable tankers brought onsite. Portable hydraulic power centers can be used to power hydraulic equipment.

Temporary liquid and solid waste processing systems are being used during decommissioning for processing plant waste. These systems include filters and/or demineralizers, and may be used at one or more locations in the waste-processing path.

Portable radiation monitors and air monitoring equipment provide local radiation monitoring. Localized temporary ventilation equipment and HEPA filtration is being used to supplement building ventilation and minimize the spread of radioactive particulate contamination.

3.4.1.8 Specific Decommissioning and Dismantlement Activities

3.4.1.8.1 Reactor Pressure Vessel Internal Segmentation



As stated earlier, segmentation of the internals is required so that the RPV package may be disposed of at Barnwell. The RPV internals are being segmented using abrasive water jet technology and mechanical disintegration methods (MDM). Once the GTCC is removed, the vessel internals will be loaded into the reactor vessel. The RPV will be prepared in accordance with a DOT-approved packaging plan and transported to a disposal facility.

3.4.1.8.2 Reactor Coolant Piping/Pressurizer

The Reactor Coolant Pump (RCP) motors have been separated from the pumps at the yoke and rigged out of the containment using the Polar Crane. The motors have been packaged for transportation and disposal. The primary coolant loop piping at the pump nozzles has been cut. The pumps are being removed from their foundation, rigged out of the containment, and packaged for ultimate disposal.

Reactor coolant pipes were cut at the pressurizer nozzles and covered. Due to contamination of the pressurizer, a separate DOT exemption request was developed and submitted. The pressurizer was rigged out using the Polar Crane and prepared for transportation and ultimate disposal.

The remaining Reactor Coolant piping spools, after being cut away from the major components, were segmented for ease of removal. This piping was rigged out of the containment using the Polar Crane and packaged for transportation and ultimate disposal.

3.4.1.8.3 Major Component Removal

Major Component Removal is being conducted with safety, ALARA, transportation, and disposal considerations in mind. The approach to work for each major component was evaluated from the point of removal through ultimate disposal. Many technologies and processes were evaluated considering safety, dose, and cost.

After the major components from the steam generator cubicles are removed, removal of ancillary systems, structures, and components will be initiated. The reactor internals are being segmented, and GTCC portions will be packaged for transfer to the SFP for storage. Next, clean-up of the refueling cavity will be performed in preparation for RPV removal. The RPV will be placed into a container, grouted and sealed. The RPV will then be moved to the equipment hatch area for final transportation preparation. Once the Containment is cleared of components and bulk commodities (conduit, small bore piping, pipe hangers, etc.), building decontamination and preparation for structure demolition will begin.

The neutron shield tank will be segmented into manageable pieces and packaged for transportation and ultimate disposal.

The Low Pressure Turbine internals were previously removed from the plant site. The associated turbine casings have been segmented to facilitate removal. Segments were rigged from the turbine pedestal using the Turbine Building Crane to the rail bay and packaged for transportation and disposal. The generator set (consisting of the rotor and stator) will be rigged and set for transport. The exciter module will be rigged in one piece from the pedestal and placed on a vehicle for transport and disposal.

3.4.1.9 Decontamination and Disposition of Site Buildings

CYAPCO has elected decontamination technologies that provide a high confidence of achieving site release criteria safely and cost-effectively while minimizing the potential for re-work.

The approach for the decontamination of the site buildings is to remediate the buildings before their demolition. The goals of the decontamination effort are to:

1. Minimize the spread of contamination within the building during component removal.
2. Clean building surfaces to minimize the amount of radioactive waste disposal.
3. Identify and remove areas of structure which cannot be economically decontaminated prior to demolition.

Review of the site indicates that many horizontal surface areas within the radiologically controlled area (RCA) have elevated surface contamination levels. In the Containment, essentially all the surfaces show measurable levels of contamination. Prior experience with similar facilities indicates that after the contaminated surface layer is removed, the remaining materials will meet the DCGLs. Therefore, to reduce worker exposure and to avoid generation of new LLW through cross-contamination of the clean base materials from the contaminated surfaces, structural concrete surfaces will be decontaminated before a structure is dismantled.

Exposed faces of buildings and foundations will be surveyed and decontaminated until the surface meets or is below the release criteria. The subsurface foundations (i.e., those more than three feet below ground level) will be decontaminated, as necessary to meet the release criteria, and left in place. Once the

surfaces have been confirmed to meet the criteria for release, the results of the final status survey will be documented and provided to the NRC for its inspection. Following completion of the final status survey and in the absence of any NRC inspection action finding the report deficient, buildings may be demolished and the concrete debris used on site as backfill.

3.4.2 General Description of and Remediation Considerations for Remaining Systems, Structures, and Components as of May 2000

This section presents a summary description of the remaining HNP systems, components, and structures that are known to be or are considered to be internally contaminated or that may be used to support decommissioning activities. This discussion includes general activities and remediation considerations associated with decommissioning these systems, structures, and components.

Because external contamination is generally considered to exist on systems, structures and components located in the RCA, it is not specifically discussed in the following system discussions. Systems, components, and structures that are externally contaminated will be decontaminated, as necessary, and released or disposed of as radioactive waste.

3.4.2.1 Reactor Pressure Vessel and Associated Internals

As stated previously, the Reactor Pressure Vessel Internals will be segmented and portions stored onsite as GTCC waste, until such time that a DOE repository is available. When the ISFSI is implemented, the GTCC will be moved with the spent fuel to the ISFSI in a shielded container. The remaining internals, and the Reactor Pressure Vessel itself, will be packaged as radioactive waste and sent to a licensed facility for final disposal.

3.4.2.2 Chemical and Volume Control System (CVCS)

The CVCS no longer supports purification of the RCS, and resin is being removed from plant ion exchangers. The CVCS has been isolated from the flooded refueling cavity inside the Containment. Remaining isolated portions of the CVCS located in containment and the Primary Auxiliary Building, including letdown piping, regenerative and non-regenerative heat exchangers, charging pumps, metering pump, RCP seal water heat exchanger, chemical addition tank, boric acid mixing subsystem, volume control tank, and associated piping have been or will be drained and will be prepared for dismantlement and removal. The CVCS is internally contaminated.

3.4.2.3 Component Cooling Water (CCW) System

Operation of the CCW system is no longer required. The system has been drained and prepared for dismantlement and removal. Portions of the CCW system are internally contaminated.

3.4.2.4 Service Water (SW) System

In support of existing NPDES permit requirements for liquid discharges to the environment, selected components of the service water system are maintained in an operational status. These components include two service water pumps (with at least one pump in operation during liquid discharges) and selected process radiation monitors. As decommissioning continues, it is anticipated that changes in NPDES permit requirements should allow the removal of additional service water system components.

Service water system components are used to support decommissioning activities that are independent of activities in support of the Spent Fuel Pool Island. Portions of the service water system may be internally contaminated.

3.4.2.5 Spent Fuel Pool and Fuel Handling Equipment

The spent fuel storage and handling SSCs are located in the Spent Fuel Building. These SSCs consist of: the Spent Fuel Pool, the spent fuel storage racks, the cask loading pit, and the area housing components of the modular systems for cooling and purification of the Spent Fuel Pool. The Spent Fuel Pool provides for storage of irradiated fuel until it is transferred to an alternative, licensed storage facility. The Spent Fuel Pool will also provide storage of GTCC radioactive waste transferred from the Containment Building, until that waste is transferred to an alternative, licensed storage facility.

The GTCC waste will be transferred from the Containment Building to the Spent Fuel Pool via the transfer canal in the Containment Building and the transfer tube between the Containment Building and the Spent Fuel Building. Following the transfer of the GTCC waste from Containment to the Spent Fuel Pool, the Spent Fuel Pool will be isolated from the transfer tube via a blind flange that will be installed in place of the existing sluice gate. Existing fuel handling equipment in the Spent Fuel Building will be used to transfer irradiated fuel or GTCC waste from the Spent Fuel Pool to approved dry-storage canisters.

Following the transfer of irradiated fuel and GTCC waste to an alternative, approved storage facility, there remains the potential that the remaining storage and handling components have high levels of contamination. The spent fuel storage racks may be accessed using the Fuel Building crane, and it is anticipated that these racks can be removed intact for sectioning and packaging at a different location. The liner for the spent fuel pool and cask loading pit may be sectioned for removal. To facilitate removal, the fuel handling cranes, the fuel transfer cart, associated tracks, and upender frames may be sectioned into pieces. The spent fuel pool bridge crane may also be used to support the decommissioning of the Spent Fuel Pool, following transfer of irradiated fuel and GTCC waste.

3.4.2.6 Spent Fuel Pool Purification

Spent fuel pool purification is performed using an independent demineralizer system located in the Spent Fuel Building. This system takes a slipstream off the pool via the Spent Fuel Pool Skimmer system. This system is an operable system and will be maintained until plans have been made to remove the spent fuel from the pool and transferred to dry cask storage.

3.4.2.7 Spent Fuel Pool Transfer Tube

The Spent Fuel Pool transfer tube connects the pool with the fuel transfer canal within the reactor refueling cavity. The transfer tube must be maintained functional to allow for the transporting of the RPV GTCC materials from the cavity to the pool. Upon completion of the RPV internal segmentation, the transfer tube will be sealed from the spent fuel pool. This will allow for the removal of the fuel transfer canal inside containment.

3.4.2.8 Makeup Water (MW)

The Spent Fuel Building Makeup Water (MW) system provides the required makeup water needed for the operation of the Spent Fuel Pool (SFP), Intermediate Cooling (IC) system, and the Spray Cooling system. These systems are required to support the heat removal from the SFP. The MW is stored in existing tanks (formerly the PWST and RPWST) and are now called "A & B" MWSTs. Piping and components

supporting this system will be maintained until the spent fuel is removed from the pool and transferred to an alternative, approved storage facility. This system is internally contaminated.

3.4.2.9 Main Steam and Feedwater (MS & FW) Systems

The Main Steam System is no longer required. Portions of the system have been removed to support steam generator removal and turbine rotor removal. The Feedwater System has been drained and is being dismantled and removed. Components include feedwater heaters, Condenser, Turbine Generator, associated piping and components. The feedwater pumps located in the Turbine Building have been removed and disposed. Portions of the MS and FW Systems may be internally contaminated.

3.4.2.10 Reactor Coolant System (RCS)

The reactor cavity maintains a full inventory of water to allow for the ongoing RPV internal segmentation. The cavity, RPV, and associated RCS loop piping are isolated from the rest of the system via closure of RCS loop stop isolation valves and sealing of other cavity-related piping.

Cavity isolation has allowed for the removal of the steam generators, Reactor Coolant Pumps, Pressurizer, and connecting piping.

Completion of the RPV internal segmentation will allow for the cavity to be drained, the reactor vessel packaged, and remaining piping and valves to be removed. A portion of the concrete cavity structure will be cut and removed to allow for the removal of the reactor vessel package through the containment equipment hatchway. The RCS is internally contaminated.

3.4.2.11 Residual Heat Removal (RHR) System

The RHR System is no longer required to perform its intended function. The system has been isolated from the flooded refueling cavity inside the Containment.

Remaining portions of the system have been isolated from operable systems, drained and are prepared for dismantlement and removal. Components include RHR pumps, heat exchangers, containment spray, containment charcoal filters, and associated piping. The RHR System is internally contaminated.

3.4.2.12 Safety Injection (SI) System

The Safety Injection System is no longer required to perform an operating function. The system has been isolated from the flooded reactor cavity.

The high-pressure Safety Injection System was used for a flow path to flush portions of the system containing spent resin deposited during chemical decontamination activities. With the completion of resin cleanup, the SI system upstream of the isolated cavity will be drained, dismantled, and removed. System components include high- and low-pressure SI pumps, RWST, and associated piping. The Safety Injection System is internally contaminated.

3.4.2.13 Service Air System

The plant Service Air System is no longer required. The compressors have ceased operation and will be removed. Construction air required for decommissioning activities will be provided via local industrial compressors. Portions of the Service Air System are internally contaminated.

3.4.2.14 Control Air System

Based on the significantly decreased control air requirements, the plant control air compressors have been permanently shut down. Smaller local air compressors have been located throughout the Turbine Building and the Primary Auxiliary Building. These temporary compressors are tied into portions of the old plant control air system to make smaller closed loop systems. These loops supply tank level indicators, flow control and air operated valves for various systems (Well Water, Turbine Building, Waste Processing, Tank Farm, Service Water, and Ventilation). Temporary compressors may be removed, as systems become abandoned and they are no longer required. The retired Control Air System and associated compressors will be dismantled and removed. Portions of the Control Air System may be internally contaminated.

3.4.2.15 Primary Water (PW) System

Demineralized water required for decommissioning activities will be provided from the existing Demineralized Water Storage Tank (DWST), a component of the Primary Water (PW) System. Rather than using other components of the PW System, temporary pumps and hoses are used for distributing water from the DWST.

Current decommissioning activities include emptying resin from the ion exchangers of the PW system. Following the completion of this activity, the remaining water in the PW system will be drained; and the PW system may be dismantled and removed. Portions of the PW System are internally contaminated.

3.4.2.16 Primary Ventilation System

Selected components of the Plant Primary Ventilation System are operated to support ventilation in the Primary Auxiliary Building (PAB) and purge ventilation for the Containment Building. These ventilation paths exhaust to the environment via the primary ventilation stack. The primary ventilation stack is monitored for radioactive particulates. With the permanent absence of nuclear fuel from the Containment Building, monitoring and filtration for fission by-product radioactive gasses in the primary ventilation stack are no longer necessary.

The Spent Fuel Building (SFB) ventilation system shares no components with the primary ventilation system. The Spent Fuel Building ventilation system includes filters and monitors for radioactive particulates and any release of radioactive gasses from the stored nuclear fuel. This system will be operated while fuel is stored in the spent fuel pool; and it will be operated to support activities for the decommissioning of the Spent Fuel Pool and the Spent Fuel Building. The Plant Primary Ventilation System and the Spent Fuel Building Ventilation System includes equipment and ducting that are internally contaminated.

3.4.2.17 Liquid Waste System

Currently, liquid radioactive wastes from decommissioning activities and Spent Fuel Pool Island operation are processed by a combination of a temporary, mobile water treatment demineralizer skid and selected components of the pre-existing liquid radioactive waste processing system (including Aerated Drains Tank (ADT), Aerated Drains Holdup Tanks, Waste Test Tanks, Recirculating Test Tanks, and associated pumps, components, and piping). The use of a temporary water-processing skid results in the release of other liquid waste system components for isolation and removal.

During future decommissioning activities, including operation of the Spent Fuel Pool, alterations in the configuration of liquid waste processing components may be implemented. Following the processing of liquid wastes, controlled effluent releases are performed in compliance with existing NPDES permit

requirements and applicable requirements of 10CFR20. These effluent discharges are monitored in accordance with approved procedures.

During future decommissioning of the Spent Fuel Pool and the Spent Fuel Building, an appropriate, temporary waste water treatment system may be used for processing of spent fuel water and other decommissioning process waste water. Appropriate monitoring systems will be in place to ensure compliance with applicable NPDES permit requirements and applicable requirements of 10CFR20. Existing and temporary liquid waste systems are internally contaminated.

3.4.2.18 Gaseous Waste System

The Gaseous Waste System is no longer required to perform its original design function. The components that are currently utilized for decommissioning activities include the equipment and floor drains, drain tanks, primary drains tank, and associated piping. Remaining equipment has been isolated from other plant systems and may be removed. The Gaseous Waste System is internally contaminated.

3.4.2.19 Turbine Building Waste Water Treatment System

The Turbine Building Wastewater Treatment System will remain in operation until the completion of the Turbine Building dismantlement. The system is required to process Turbine Building residual water and sediment. Components remaining include the waste neutralization and storage tanks, gravity separator, NPDES filters, sediment removal package, and associated piping. Support systems consist of construction power and local compressed air. Portions of the system may be contaminated.

3.4.2.20 Well Water and Water Treatment System

The Well Water and Water Treatment System supply water for both plant and personnel activities other than personnel consumption. The system uses only one or both of well "A" and well "B." Support systems consist of construction power and local compressed air.

As decommissioning continues, prior to the dismantlement of the Turbine Building, the system will be modified to continue to provide water throughout dismantlement and removal of the Turbine Building. The well water system is not internally contaminated.

3.4.2.21 Circulating Water and Vacuum Priming Systems

During water discharges, in conformance with existing NPDES permit requirements and REMODCM criteria, the Circulating Water System is utilized, including operation of one or more circulating water pumps. CYAPCO has submitted a request for a modification of the NPDES permit which would add an alternative discharge path that does not require the operation of the Circulating Water System. The Circulating Water System will be maintained until dismantlement of the Turbine Building. The Circulating Water System is not contaminated.

3.4.2.22 Closed Cooling System

The Closed Cooling Water system is no longer required. The system may be dismantled and removed. Portions of the system are internally contaminated.

3.4.2.23 Turbine Lube Oil System

The Turbine Oil Tank Room was used for the storage and transport of waste oil. The Turbine Lube Oil System is not internally contaminated.

3.4.2.24 Boron Recovery System

The Boron Recovery System has not been used since 1991, and is no longer required but can be used for water processing. The system has been drained. The boron recovery evaporators, coolers, reboilers, and associated pumps and interconnecting piping have been isolated and may be dismantled and removed. The Boron Recovery System is internally contaminated. The Borated Waste Storage Tanks (BWST) and the Recycle Test Tanks (RTT) are currently used as part of water storage and process and will be utilized for storage during the refueling cavity dewatering after completion of the RPV internal segmentation. The BWSTs and RTTs will be desludged, as necessary, prior to removal.

3.4.2.25 Leak Monitoring and Miscellaneous Systems

The Reactor Cavity Seal Telltale Drain System will be maintained to allow for prompt detection of any reactor cavity seal leakage. This system will remain intact until the refueling cavity water has been transferred to holdup tanks for processing. The reactor core sumps and the in-core instrumentation sumps collect miscellaneous leakage, drainage, or unplanned releases in Containment. These sumps, associated piping and equipment will remain in use to collect and transfer this leakage to the PAB. These systems are internally contaminated.

3.4.2.26 Site Electrical Distribution

The existing plant electrical distribution system is being modified to institute the site decommissioning power distribution system. This new system provides the Spent Fuel Pool Island power and power required to support plant decommissioning and demolition activities from 115 kV systems through Station Service Transformer 12R-22S (399) via 4160V Bus 12. This allows the existing plant power distribution system and associated equipment, not re-aligned to the new decommissioning power distribution system, to be de-energized and removed from service.

Equipment and cables used for the site decommissioning power are identified with unique indicators to help ensure worker safety.

3.4.2.27 Fire Protection System

The Fire Protection System provides manual and automatic fire suppression and automatic detection capabilities for plant areas. The Fire Protection System includes the following: portable fire extinguishers, water supply and distribution systems, fire suppression system, emergency lighting, and the fire detection and alarm system.

The Fire Water Distribution System is maintained operable by the use of the electric and diesel fire pumps located in the Screenwell Building, as well as the associated test headers, underground distribution system, fire hydrants, sprinklers and hose stations.

The fire protection supply to the Turbine Building, Maintenance Hot and Clean Shop, and the Boiler Room have been isolated in order to support removal activities in these areas. Other portions of the system may be de-activated to support future building decommissioning and removal activities. Such changes may require updates to the Technical Requirements Manual (TRM). The Fire Protection Systems are not internally contaminated.

3.4.2.28 Heating Steam and Condensate System

The Heating Steam and Condensate System were used through the 1999/2000 winter and are being drained, dismantled and removed. Future heating and freeze protection of operable systems and buildings

may be through the use of local heaters and/or independent heating systems. Portions of this system are internally contaminated.

3.4.2.29 Floor, Roof and Equipment Drains

The Floor, Roof, and Equipment Drain System will be maintained in operation. These drains are required to provide building drainage throughout the decommissioning effort and will be one of the last systems in each building to be removed.

Portions of these drain systems are internally contaminated.

3.4.2.30 Buildings

3.4.2.30.1 Spent Fuel Building

The Spent Fuel Building consists of a concrete structure with a superstructure on steel framing above the new fuel storage area and the spent fuel pool area. The Spent Fuel Pool is constructed of reinforced 6-foot thick concrete walls lined with a ¼-inch thick stainless steel liner and is 35 feet deep. The roof is an 8-inch thick reinforced concrete slab. The building also contains the spent fuel pool independent demineralizer cleanup system and the spent fuel pool independent cooling system and associated equipment.

The Spent Fuel Pool will remain in use until the spent fuel has been transferred to the DOE or to dry cask storage at an onsite ISFSI. Upon completion of this effort, the Spent Fuel Building will be decontaminated and dismantled. Contaminated concrete will be scabbled to remove surface contaminated areas. Portions of the Spent Fuel Cooling System are internally contaminated.

3.4.2.30.2 Containment Building

The Containment Building consists of two structures on a common foundation. One is the containment itself, and the other is the internal biological shield structure. Supports for equipment, operating decks, access stairways, and platforms are included in the containment internals.

The inside of the containment concrete shell is steel lined. The steel liner is 3/8-inch thick at the sidewall, ½-inch thick at the spherical dome, and ¼-inch thick at the bottom. Penetrations in the Containment Building include the equipment hatch. This hatch has been permanently dismantled and removed. Since decommissioning activities began, an alternate personnel access opening has been created in the East side of the Containment Building.

Portions of the concrete surfaces inside the Containment Building are contaminated. These surfaces may require decontamination using methods such as scabbling or grinding down the surface to remove contaminated layers. Plate and structural steel, grating, ladders, and platforms will be decontaminated as required for disposal.

The refueling cavity will remain flooded until segmentation of the Reactor Pressure Vessel Internals has been completed. The refueling cavity is lined with steel. The liner will be cut and removed to allow access to the cavity concrete walls for radiological surveys and any required decontamination activities.

The Containment Polar Crane will be used throughout decommissioning for removal of large components and structures. The Polar Crane will be one of the last items to be removed from Containment.

3.4.2.30.3 Primary Auxiliary Building (PAB)

The two-story Primary Auxiliary Building is primarily a reinforced-concrete structure with a one-story, braced steel frame superstructure. This building houses many of the reactor auxiliary systems. Shielding is provided by a combination of below grade construction and concrete walls and slabs on the first floor. The second floor consists of structural steel framing for the Containment and PAB Ventilation System.

The pipe trench is located at or below ground elevation and extends the length of the building with laterals connecting to individual equipment cubicles. The pipe trench is covered by removable concrete slabs.

Selected ventilation equipment in the PAB supports radiological controls during decommissioning activities in the PAB, the Containment Building, and other structures supported by PAB ventilation. Selected water processing equipment in the PAB supports the processing of wastewater from decommissioning activities. These selected components will support the corresponding activities until they are either replaced by appropriate, alternative systems or the associated supported decommissioning activities are verified to be complete. The concrete walls, floors, and pipe trench slabs may be decontaminated by scabbling methods to remove contaminated layers. Floor and equipment drains are potentially contaminated and flow to the Aerated Drains Tanks. These equipment drains will most likely be the last components to be removed.

3.4.2.30.4 Other Buildings

The Spent Resin Building is still functional and will be maintained until the existing plant ion exchangers and spent resin tank have been sluiced and moved to the shipping containers for offsite disposal. The building is contaminated and will be decontaminated using scabbling and concrete removal methods.

The Service Building will be maintained until the radiation control point and associated personnel are relocated. Subsequently the building will be dismantled and disposed. Portions of the building may be slightly contaminated.

The Turbine Building houses the plant's secondary side components, including the turbine generator, condenser and feedwater equipment. Portions of the Turbine Building systems are internally contaminated.

The Screenwell House located on the banks of the Connecticut River will be maintained until there is no requirement for Service Water or the Fire Pumps. These structure and components are not considered to be internally contaminated.

With the exception of the building sump pump and equipment drain pump, the Waste Disposal Building no longer contains any operable systems, as these systems have been drained and are available for removal. The building sump pump and equipment drain pump are required until the Aerated Drain Tank is no longer needed. Systems are internally contaminated, and some concrete surface may require decontamination.

3.5 Radiological Impacts of Decommissioning Activities

The decommissioning activities described herein are conducted under the provisions of the approved HNP Radiation Protection Program and Radioactive Waste Management Program. These programs continue to be implemented as described in the HNP UFSAR. The Radiation Protection Program implements the regulatory requirements of 10CFR20 (Reference 3-7) through approved plant procedures

established to maintain radiation exposures ALARA. The Radioactive Waste Management Program controls generation, characterization, processing, handling, shipping, and disposal of radioactive waste per approved CYAPCO Radiation Protection Program, Process Control Program, and plant procedures.

3.5.1 Occupational Exposure

Detailed exposure estimates and exposure controls for specific activities are developed during detailed planning per Radiation Protection Program procedures. Table 3-2 provides estimated personnel exposures for various decommissioning and fuel storage activities. The total radiation exposure impact for decommissioning and spent fuel management is estimated in Table 3-2 to total approximately 935 person-rem, as given in the PSDAR.

Table 3-2
Radiation Exposure Projections for Decommissioning and Fuel Storage Activities

Activity	Exposure (person-rem)
Dismantlement Activities	
RCS Decon.	32
Asbestos Abatement (includes scaffolding)	136
Steam Generators and Pressurizer	88
Reactor Vessel Internals	90
Reactor Vessel & Head Prep (CRDM/ICI Structure)	72
Main Coolant System	17
Plant Systems	248
Structures	74
Miscellaneous	72
Waste Processing (includes shipping and prep)	12
Subtotal Dismantlement Activities	**841**
Operational Activities	
1996 Operations--post certifications	1
1997 Operations	17
Spent Fuel Pool Isolation Modifications	10
Spent Fuel Storage	24
Fuel Transfer/Cask Loading	42
Subtotal Operational Activities	**94**
Total for Decommissioning and Fuel Storage Activities	**935**
Transportation (occupational and to general public)	**72**

3.5.2 Radioactive Waste Projections

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

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

Table 3-3 provides projections of waste quantities for decommissioning. These waste quantities are those reflected in the PSDAR. The total volume of HNP low-level radioactive waste for disposal has been estimated at 283,117 cubic feet. Actual waste volumes and classifications may vary, but the total quantity is not expected to exceed 283,117 cubic feet.

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

Table 3-3
Projected Waste Quantities

Item Description	Waste Classification per 10CFR61	Volume (cu. ft.)
Reactor Vessel	A and B	7892
Reactor Vessel Internals	A, B, C and GTCC	2444
Pressurizer	A	2083
Reactor Coolant Pumps and Motors	A	5592
Steam Generators (four)	A	20772
Balance of NSSS Equipment	A	6441
Primary System Decon. Resin	C	1000
Fuel Racks	A	17398
Balance of Non-NSSS Equipment	A	75534
Concrete and Structural Steel	A	7344
Soil and Pavement	A	93749
Dry Active Waste	A	12696
Contaminated Tools	A	10000
Processed Liquids	B (some may be classified as A)	6406
Contaminated Asbestos	A	21266
Estimated Total Volume Generated		**290617**
Estimated Volume Reduction		**-7500**
Projected Burial Volume:		**283117**

3.6 References

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

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

3-3 Letter CY-98-144 from CYAPCO to the Department of Transportation, "Request for Exempting the Shipment of the Haddam Neck Plant Steam Generator Subassemblies," dated November 5, 1998 as revised by CY-99-073, dated June 3, 1999.

3-4 Letter CY-00-015 from CYAPCO to the Department of Transportation, "Exemption Request from the Surface Contaminated Object Demonstration Requirements of 49CFR173.403 and the Packaging Requirements of 49CFR173.427(b)(1) for the Shipment of a Pressurizer from the Haddam Neck Nuclear Plant," dated January 18, 2000.

3-5 Letter CY-00-016 from CYAPCO to the Department of Transportation, " Request for Exemption for Shipment of the Haddam Neck Nuclear Plant Reactor Vessel Submitted to US DOT, " dated March 30, 2000.

3-6 "Haddam Neck Reactor Vessel Internals Characterization," WMG-9913-9007, Revision 1, dated March 2000.

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

4 SITE REMEDIATION PLANS

4.1 Introduction

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

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

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

4.2 Remediation Levels and ALARA Evaluations

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

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

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

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

4.2.1 Generic ALARA Screening Levels

As discussed in DG-4006, soil remediation beyond the DCGLs is not likely to be cost-beneficial due to the high costs of waste disposal. This will be confirmed in a generic ALARA evaluation for soils. It is expected that soil will be at ALARA levels when it meets the site-specific DCGLs discussed in Sections 5 and 6.

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

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

4.2.2 Survey-Unit Specific ALARA Evaluation

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

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

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

The remediation levels represent the radioactivity concentrations at which a remediation action is cost beneficial and, therefore, do not represent maximum, or "not-to-exceed" concentrations. The ALARA criteria is met by performing the remediation action and not necessarily by achieving results below a specific remediation level. An ALARA analysis ensures that the efforts to remove residual contamination are commensurate with the risk that exists with leaving the residual contamination in place, even if the target remediation levels are not achieved. However the residual contamination must be low enough to assure the annual dose to the average member of the critical group does not exceed 25 mrem/yr TEDE.

4.3 Remediation Actions

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

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

2. Soils; and

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

4.3.1 Structures

Concrete from contaminated structures will be remediated to a level meeting the radiological criteria for unrestricted release of the site, as discussed in Section 6. After completion of final status surveys and absent any findings during NRC inspections, concrete building debris from decontaminated structures may be used as backfill and placed into the remaining subsurface building foundations. Nonstructural materials will be assessed using the process in Section 5.6.

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

Remediation activities for an area may include wiping, vacuuming, and washing with low-or high-pressure applications. Surfaces may also be remediated using surface removal techniques such as scabbling or grinding. Use of surface removal techniques controls the removal depth, minimizing the waste volume produced.

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

4.3.2 Soils

Soil contamination above the remediation level will be removed and disposed of as radioactive waste. Offsite fill may used to replace the excavated materials. As discussed previously in Section 2, the site characterization process will establish the location, depth and extent of soil contamination. As needed, additional investigations will be performed to ensure that any soil contamination profiles that may change during the remediation actions are adequately identified and characterized.

4.3.3 Nonstructural Systems

The systems remaining to be decontaminated are described in Section 3. Contaminated plant systems and components may be sent to an offsite processing facility or to a low-level radioactive waste disposal facility. Slightly contaminated systems may be decontaminated onsite and released. Following completion of the final status survey and in the absence of any NRC inspection action finding the report deficient, nonstructural components within buildings meeting the criteria in Regulatory Guide 1.86 (Reference 4-4) may be released using the process discussed in Section 5.6. Components and buildings, removed as part of the decommissioning process, may be surveyed and released using existing plant procedures and processes.

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

4.4 References

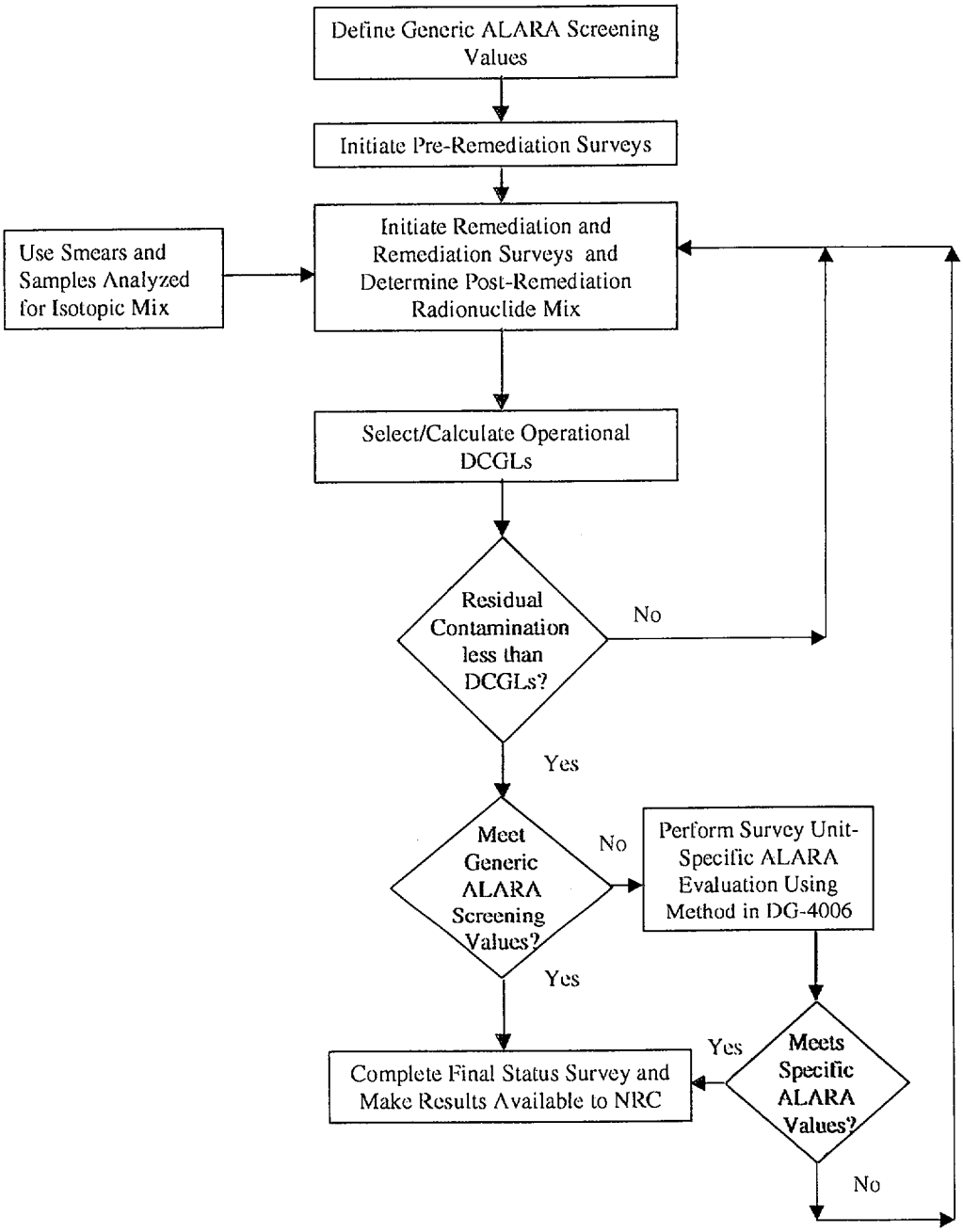
4-1 Code of Federal Regulations, Title 10, Part 50.82, "Termination of License."

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

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

4-4 Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," dated June 1974.

Figure 4-1
Survey Unit ALARA Evaluation Process



5 FINAL STATUS SURVEY PLAN

5.1 Introduction

The purpose of the Final Status Survey Plan is to describe the methods to be used in planning, designing, conducting, and evaluating final status surveys at the HNP site. These surveys serve as key elements to demonstrate that the dose from residual radioactivity is less than the maximum annual dose criterion for license termination for unrestricted use specified in 10CFR20.1402 (Reference 5-1). The additional requirement of 10CFR20.1402 that all residual radioactivity at the site be reduced to levels that are as low as reasonable achievable (ALARA) is addressed in Section 4. The Final Status Survey Plan was developed using the guidance of Draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination" (Reference 5-2); NUREG-1575, "The Multi-Agency Radiological Site Survey and Investigation Manual (MARSSIM)" (Reference 5-3); and Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" (Reference 5-4).

The final status survey process described in this plan adheres to the guidance of MARSSIM for the design of final status surveys. However, advanced survey technologies may be used to conduct radiological surveys that exceed MARSSIM requirements for quantity of data. This survey plan allows for the use of these advanced technologies, where survey quality and efficiency can be increased, as long as certain criteria are met. These criteria ensure that the survey results are at least equivalent to those that would have been obtained using the non-parametric sampling methods of MARSSIM in terms of their statistical confidence.

5.2 Scope

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

5.3 Summary of the Final Status Survey Process

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

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

The final status survey process consists of four principal elements:

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

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

Before the survey process can proceed to the design phase, concentration levels that represent the maximum annual dose criterion of 10CFR20.1402 must be established. These concentrations are established for either surface contamination or volumetric contamination. They are used in the survey design process to establish the minimum sensitivities required for the available survey instruments and techniques, and in some cases, the spacing of fixed measurements or samples to be made within a survey unit. Surface or volumetric concentrations that correspond to the maximum annual dose criterion are referred to as Derived Concentration Guideline Levels, or DCGLs. A DCGL established for the average residual radioactivity in a survey unit is called a $DCGL_W$. Values of the $DCGL_W$ may then be increased through the use of area factors to obtain a DCGL that represents the same dose to an individual for residual radioactivity over a smaller area within a survey unit. The scaled value is called the $DCGL_{EMC}$, where EMC stands for elevated measurement comparison.

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

Survey implementation is the process of carrying out the survey plan (package) for a given survey unit. This consists of scan measurements, fixed measurements, and collection and analysis of samples. Scan measurements will always be made, while fixed measurements and sampling may not be necessary. Data are collected and stored using a data management system.

Data assessment includes data verification and validation (V&V), review of survey design bases, and data analysis. For a given survey unit, the survey data are evaluated to determine if the residual activity levels in the unit meet the applicable release criterion and if any areas of elevated activity exist. In some cases, data evaluation will simply serve to show that all of the measurements made in a given survey unit were below the applicable $DCGL_W$. If so, demonstrating compliance with the release criterion is a simple matter and requires little in the way of analysis. In other cases, residual radioactivity may exist where measurement results both above and below the $DCGL_W$ are observed. In these cases, statistical tests must be performed to make a decision as to whether the unit meets the release criterion. The statistical tests that

might be required to make decisions regarding the residual activity levels in a survey unit relative to the applicable $DCGL_W$ must be considered in the survey design to ensure that a sufficient number of measurements are collected.

MARSSIM specifies two non-parametric statistical tests to be applied to final status survey data to evaluate whether a set of a few (typically 30 or less) measurement results demonstrate compliance with the release criterion for a given survey unit. If the data are radionuclide-specific and the radionuclide of interest is not present in background, then a one-sample statistical test (Sign test) is applied. If the radionuclide does appear in background (for radionuclide specific measurements) or if gross activity measurements are being considered, a two-sample statistical test (Wilcoxon Rank Sum test), may be used; or the data may be background subtracted and the Sign test applied. The latter approach provides a convenient way to evaluate gross activity measurements from survey units having multiple materials with different backgrounds.

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

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

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

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

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

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

5.4 Survey Planning

5.4.1 Classification of Survey Areas and Units

The adequacy of the final status survey process rests upon partitioning the site into properly classified survey units of appropriate physical area. Section 2 of this document discusses in detail the HSA for the HNP site and the classifications assigned to all of the site structures and grounds. Characterization is an ongoing effort throughout the decommissioning process, and survey unit classifications may be modified on the basis of new characterization information or impacts from decommissioning activities. The process described in Section 1.5 will be used to evaluate these changes. Survey areas have been determined as described in Section 2.3.3.2.

5.4.2 Survey Units

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

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

As reflected in Table 2-6 and Figures 2-1 through 2-19, areas of HNP and the associated site classified as impacted have been divided into survey units to facilitate survey design. Each survey unit has been assigned an initial classification based on the site characterization.

Table 5-1
HNP Survey Unit Surface Area Limits
(Reference 5-3, Roadmap Table 1)

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

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

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

5.4.3 Reference Coordinate Systems

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

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

5.4.4 Reference Areas and Materials

If any of the radionuclides of concern are present in background, establishing reference background concentrations may be necessary to identify and evaluate contributions attributable to site operations. Background concentrations will be determined via measurements made in one or more reference areas selected to represent the baseline radiological conditions for the site. In addition, measurements will be required to establish background levels for various construction materials to accommodate gross activity (non-radionuclide-specific) measurements.

For purposes of the HNP final status survey, the only radionuclide for which a background concentration is expected to be required is Cs-137. This radionuclide is present in soil and exterior building materials as a result of historical atmospheric nuclear weapons testing. The concentration of Cs-137 in the soil in the vicinity of the HNP site varies within normal environmental levels, and is typically above the detection limits of conventional soil activity measurements for gamma-emitting radionuclides. Thus, a reference concentration and its associated standard deviation will be required to distinguish Cs-137 in the soil as a

result of weapons testing from any that might be there from plant operations. The need for making this distinction will depend on the DCGL for Cs-137 relative to the baseline Cs-137 soil concentration.

A study to characterize background Cs-137 soil concentrations in locations in the vicinity of the HNP was conducted in May of 1998. The results of this study are documented in Reference 5-5. Twenty-two locations were chosen at random for sample collection. The locations were selected so they were no more than 10 miles from the site, but no closer than 2 miles. The locations were therefore representative of the areas surrounding the HNP while having little chance of having been affected by plant operations. Two surface soil samples were collected at each location. The samples consisted of soil cut from a 15 cm by 15 cm area down to a depth of 15 cm. The background levels of Cs-137 range from 0.00 to 1.80 pCi/g. These values are a small fraction of the associated DCGL. It is expected that the background Cs-137 concentrations established in the 22 offsite locations are representative of those for the wooded and open areas on the HNP property in terms of both the magnitude and variability.

In addition to a specific reference concentration for Cs-137, backgrounds will also be required for gross activity measurements. Surface activity measurements in particular are generally radionuclide independent, and thus appropriate backgrounds will be required for the various materials and surfaces that will be encountered in the final status survey effort to facilitate meaningful assessments of residual activity levels. Such background measurements will be made using either unaffected materials at the site or representative materials obtained from elsewhere.

Table 5-2 gives typical ranges for backgrounds expected to be encountered at the HNP during final survey activities. Ranges are given for several detector types (gross counters) and encompass the variability expected for different materials. The data in Table 5-2 are derived from both NUREG-1507 (Reference 5-6) and from experience at the HNP. The actual background values for the reference materials of interest will be determined as a part of final status survey. Any background readings encountered during final survey activities that exceed the upper limit of the ranges in Table 5-2 will be evaluated to ensure that the required measurement sensitivity can be achieved.

Table 5-2
Typical Backgrounds

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

Whether or not they are radionuclide specific, all background measurements should account for both spatial variability over the area being assessed and the precision of the instrument or method being used to make the measurements. Thus, the same materials or areas may require more than one background

assessment to provide the requisite background information for the various survey instruments or methods expected to be used for final status surveys. The result of these background assessments provide the basis for determining the mean and its associated standard deviation.

5.4.5 Area Preparation: Isolation and Control

Before final status survey activities can begin in an area, a transition must occur where planned decommissioning activities are completed and the area is subsequently assessed to scope the required isolation and control measures. This includes establishing if the area is ready for final survey activities and identifying any work practice issues that must be addressed in survey planning and design. Determination of readiness for final status survey will be based on characterization and/or remediation surveys indicating that the residual radioactive material is likely to comply with the DCGLs and the removable contamination is below station criteria (< 1000 dpm/100 cm^2). Following this assessment, isolation and control measures will be implemented to ensure that no plant-related radioactive material is introduced into the area, prior to, during or after final survey activities. Isolation and control measures are implemented for areas such as an entire building or large, open areas, for which there should not be any impact from on-going decommissioning activities. In the event that additional remediation is required in an area following the implementation of isolation and control measures, local contamination control measures such as tents, HEPA filters, or vacuums will be employed as appropriate.

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

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

- all planned decommissioning activities in the area are complete, including removal, as necessary, of items (e.g., equipment mounts, wall hangers, and exposed studs) that could interfere with final survey activities, but not including components addressed under Section 5.4.6.5 of this plan that are to be left in place and released with the associated structure;

- all planned decommissioning activities in areas either adjacent to the area to be isolated or that could otherwise affect it are either complete or are deemed not to have any reasonable potential to spread plant-related radioactive material to the area;

- all tools and equipment not needed for final survey activities are removed;

- any equipment to be used for final survey activities is evaluated to ensure it does not pose the potential for introducing plant-related radioactive material into the area; and

- all transit paths to or through the area, except those required to support final survey activities, are eliminated or re-routed.

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

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

Measures to prevent against the introduction of plant-related radioactive material by persons entering an isolated area may include personnel frisking stations at the entry point, the use of "sticky pads", or other such routine methods. Isolation from airborne material may include sealing off openings, including doors and ventilation ducts. Though not likely to be encountered, if a potential for waterborne material is deemed to exist (e.g., floor drains or penetrations left by decommissioning activities), similar measures will be taken to be sure such sources are sealed off from the isolated area.

Isolation and control measures will be implemented through approved plant procedures and will remain in force throughout final survey activities and until there is no risk of recontamination from decommissioning and fuel storage activities. In the event that isolation and control measures established for a given survey unit are compromised, evaluations will be performed and documented to confirm that no radioactive material was introduced into the area. In the unlikely event that radioactive material above previously detected levels is found, appropriate actions will be taken including additional remediation and resurvey of the affected area. Re-surveys will be scoped and performed in accordance with Section 5.5.3.5 of this plan.

5.4.6 Selection of DCGLs

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

Section 6 of this plan describes in detail the modeling performed to develop the radionuclide-specific DCGLs for soil, surfaces and activated concrete. These values will be used to establish operational DCGLs for survey units in cases where measurements are made that are not radionuclide specific or when difficult-to-measure radionuclides are present that necessitate the need for a surrogate radionuclide. In such cases, operational DCGLs will be established based on a representative radionuclide mix established for each survey unit. In cases where measurable activity still exists, it is expected that the radionuclide mix will be established based on gamma-ray spectroscopy or equivalent analyses on representative samples, with scaling factors used to establish the activity contribution for any difficult-to-measure radionuclides that might be present. Scaling factors will be selected from available composite waste stream analyses or similar assays. Such analyses are performed periodically and documented in support of waste characterization needs.

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

Section 6 of this plan establishes the basis for the DCGLs in soil and sediments to be the resident farmer scenario, and the DCGLs for structures to be the building occupancy scenario. Additionally, for structures, checks were made using two additional scenarios to determine the most limiting DCGL for each radionuclide. Table 6-23 presents the resultant radionuclide specific DCGLs, the bounding (minimum) value and the controlling scenario (building occupancy or concrete debris).

For a few radionuclides, the scenario involving the demolition of the concrete structure and use of the concrete as fill material in the building subsurface volume was determined to be the most limiting DCGL. The limiting scenario in this case is the resident farmer scenario with subsurface concrete debris, and a well sunk into the debris. To model such a possibility, the affected area is a 10,000 m^2 area surrounding the buried concrete debris. The 10,000 m^2 area is consistent with the guidance of MARSSIM for land area surveys and the dose modeling approach for the resident farmer. The significance of choosing a region immediately surrounding the area of the buried concrete debris is that this represents a limiting case. This is the only area of the HNP where one could reasonably anticipate encountering residual soil activity concentrations comparable to the soil DCGLs.

To show compliance with 25 mRem/yr and ALARA, the unity rule will be applied in those areas in which the dose can be a result of both surface radioactivity bounded by the resident farmer - concrete debris scenario and resident farmer – soil scenario. Use of the unity rule, as discussed in Section 5.8.4, will result in the development of operational DCGLs on an radionuclide specific basis.

5.4.6.1 Operational DCGLs

Operational DCGLs will be developed based on the contribution to the total dose from radionuclides in soil, dose from radionuclides in buried concrete debris through the water pathway (if that pathway is controlling for the specific nuclide), and dose due to existing radioactivity in the ground water. To demonstrate this process, the following example assumes that for a given survey unit three exposure pathways apply:

- Residual radioactivity in soil
- Existing radioactivity in water (from Cs-137 and H-3), and
- Radioactivity in well water from concrete debris.

These operational DCGLs will be selected to control the dose from the most restrictive of the following cases:

- For an occupant working in a building containing residual radioactivity,
- For the resident farmer living above buried concrete debris, and
- For a resident farmer on soil outside of the concrete area where a well is supplying water from the buried concrete debris area.

The most complex of these situations is the latter case where existing soil activity may be present and exposure may be from water supplied from a well in the concrete debris. For this case, the following relationship controls the total dose to 25 mrem:

$$25mrem = H_{soil} + H_w^{existing} + H_w^{concrete} \qquad \text{(Equation 5-1)}$$

where: H_{soil} is the dose contribution from soil.

$H_w^{existing}$ is the dose contribution from the existing radioactivity in the ground water, and

$H_w^{concrete}$ is the dose contribution from the water pathway from water obtained from a well located in the concrete debris.

The following example illustrates this case.

Example 1:

In this example, assume that the dose from the existing water activity, $H_w^{existing}$, is evaluated to be 3 mrem, therefore,

$$22mrem = H_{soil} + H_w^{concrete}$$

The following relates the dose, H, to the operational and base-case DCGL values.

$$H_{soil} = 25 * \frac{DCGL_{soil}^{operational}}{DCGL_{soil}^{base-case}} \quad \text{and}$$

$$H_w^{concrete} = 25 * \frac{DCGL_{concrete}^{operational}}{DCGL_{w,concrete}^{base-case}}$$

The value of $DCGL_{w,concrete}^{base-case}$ represents the adjusted surface activity DCGL for only the water dependent dose contributions for the concrete debris as follows:

$$DCGL_{w,concrete}^{base-case} = \frac{DCGL_{s,concrete}^{base-case}}{f_w}$$

Where: $DCGL_{s,concrete}^{base-case}$ is concrete debris surface DCGL value from Table 6-20, and

$$f_w = \frac{Water\ Dependent\ Dose}{Total\ Dose}$$

and the value f_w represents the fraction of the total dose from the water pathway where these dose values are taken from Table 6-20.

Therefore, by substitution, the following compliance formulation applies:

$$\frac{22}{25} = \frac{DCGL_{soil}^{operational}}{DCGL_{soil}^{base-case}} + f_w \frac{DCGL_{s,concrete}^{operational}}{DCGL_{s,concrete}^{base-case}}$$

Using the above relationship requires that the soil or concrete operational DCGL be chosen at a value lower than the corresponding base-case DCGL. In the above case, either the soil or surface DCGL is chosen to determine the appropriate value for the other. The choice will be based on knowledge of the soil and surface radioactivity determined throughout the remediation process. Implementation of this method, which will be controlled through site procedures, ensures that the total dose from all combinations of pathways is maintained ALARA and less than 25 mrem per year for each survey unit.

Once operational DCGLs are established for a given survey unit, field application of those values may include gross activity DCGLs, surrogate ratio DCGLs, and elevated measurement comparison DCGLs as described below.

5.4.6.2 Gross Activity DCGLs

Surface DCGLs

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

$$DCGL_{GA} = \frac{\sum_{1}^{n} f_i}{\sum_{1}^{n} \frac{f_i}{DCGL_i}} \qquad \text{(Equation 5-2)}$$

where:

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

Gross activity DCGLs can be developed for gross beta measurements, or a gross beta DCGL can be scaled so that it acts as a surrogate for gross alpha (see Section 5.4.6.3).

Equation 5-2 will be applied for radionuclides that are present in a survey unit in concentrations greater than 5% of their respective DCGL. The aggregate of all radionuclides not included in the gross activity DCGL, based on the percentage of their respective DCGL, will not exceed 10%. This practice is conservative relative to the process presented in 10CFR20 in which radionuclides that contribute less than 10% to dose, provided the aggregate does not exceed 30%, are not required to be included in the dose assessment.

The following example illustrates the determination of a gross activity DCGL.

Example 2:

As an example, consider the following radionuclide mix and associated DCGLs for a representative gross beta sample for a given survey unit where no soil contamination is present and where a total activity of 10,000 dpm/100 cm^2 is assumed.

Table 5-3
Example Gross Beta Radionuclide Mix

Radionuclide	Activity Fraction	Surface DCGL (dpm/100 cm^2)	Fraction of DCGL
H-3	0.002	7.23E+06	0.00
Fe-55	0.26	5.94E+07	0.00
Co-60	0.35	1.17E+04	0.30
Ni-63	0.10	2.55E+07	0.00
Sr-90	0.002	1.31E+04	0.002
Cs-134	0.02	1.51E+04	0.013
Cs-137	0.23	2.13E+04	0.11
Pu-241	0.04	2.47E+04	0.016

The example mixture fractions in the table above are from the 1999 HNP Composite Dry Active Waste (DAW) analysis used to establish the scaling factors used for waste characterizations. The mix therefore represents a recent composite mix for the Radiologically Controlled Areas of the HNP. The example of total activity (10,000 dpm/100 cm^2) was an arbitrary choice to demonstrate how gross activity DCGLs will be computed using the radionuclide-specific DCGL values developed in Section 6 for the most restrictive case between the building occupancy and resident farmer scenarios. It should be noted that the radionuclide mix used in this example is for illustrative purposes only. The appropriate representative mix data for a survey unit of interest will be used to compute gross activity DCGLs at the time of final status survey.

To compute the gross beta DCGL using the above example radionuclide mix, radionuclides that are not present in excess of 5% of their DCGL are eliminated. The aggregate of all radionuclides eliminated from the gross DCGL is 3.09% of their respective DCGLs. This results in the elimination of all radionuclides except Co-60 and Cs-137. The gross activity DCGL for this example is therefore:

$$\text{Gross Beta Activity DCGL} = \frac{0.58}{\frac{0.35}{1.17E+04} + \frac{0.23}{2.13E+04}} = 14{,}426\ \text{dpm/100 cm}^2$$

Using the above relationship, Table 5-4 illustrates the dependence of the gross DCGL on this ratio as it is varied from 0.1/0.9 to 0.9/0.1. In this evaluation, both the building occupancy gross DCGL and the resident farmer gross DCGL are computed using Equation 5-2 and the nuclide specific values in Section 6. The applicable gross DCGL is then selected as the most limiting of the two cases. As indicated in this data, the gross DCGL ranges from approximately 12,000 to 20,500 dpm/100cm^2 where applicable controlling scenario is shown.

Table 5-4
Gross Surface DCGLs for Various Cs-137 to Co-60 Ratios

Cs-137 Fraction	Co-60 Fraction	Ratio Cs-137/Co-60	Gross DCGL (in dpm/100 cm^2)
0.1	0.9	0.1	12300
0.2	0.8	0.3	12900
0.3	0.7	0.4	13500
0.4	0.6	0.7	14300
0.5	0.5	1.0	15100
0.6	0.4	1.5	16000
0.7	0.3	2.3	17100
0.8	0.2	4.0	18300
0.9	0.1	9.0	19700
0.95	0.05	19.0	20500

Soil/Concrete Debris DCGLs

In the case of contaminated soil, it is necessary to adjust the gross DCGL for surfaces to ensure that the dose limit of 25 mrem/yr will not be exceeded from the combination of all potential exposure pathways. In the case where this soil contamination is adjusted, the gross surface DCGL for concrete is determined as the most limiting from the following exposure scenarios:

- Building occupancy,
- Resident farmer on concrete debris, where water is obtained from a well placed within the concrete debris, or
- Resident farmer on soil, where water is obtained from a well placed within the concrete debris.

The most limiting of the first two scenarios is used to determine the concrete surface DCGL from activities associated directly with the concrete as provided in the above case example. To determine the gross surface DCGL from the last exposure scenario, the DCGL for the water pathway of the concrete debris case is determined and combined as follows:

$$DCGL_{s,gross}^{base-case} = \frac{1}{\sum_i \frac{f_{w,i} f_i}{DCGL_{s,i}}}$$

(Equation 5-3)

where: f_i is the fraction of the total activity contributed by radionuclide i, and
$f_{w,i}$ is the fraction of the dose from the water pathway for nuclide i.

Once calculated for a mixture of nuclides, this gross surface DCGL is compared to the most limiting of the DCGL applicable to the first two scenarios. The most limiting of this final comparison is used as the base-case gross surface DCGL for use in the compliance equation discussed above.

An example of this application is where the soil contains radioactivity at a level corresponding to 20% of the DCGL and the dose from the existing ground water is estimated at 3 mrem. In this case, the gross DCGL is computed for various Cs-137/Co-60 ratios as in the prior example. Using the input values provided in Table 5-5, Table 5-6 provides values of the most restrictive gross surface DCGLs for various Cs-137 and Co-60 mixes and for the initial conditions.

Adjustment is necessary to ensure dose from all pathways does not exceed 25 mrem/yr.

Table 5-5
Input Parameters for Example with Soil Considerations

Radionuclide	Fraction of Dose from Water Pathway from Concrete Debris	Concrete Debris Surface DCGL, dpm/100cm^2
Co-60	3.66E-02	4.80E+04
Cs-137	9.07E-01	2.13E+04

Table 5-6
Example Gross Surface DCGL Analysis for Soil at 20% DCGL and Water Dose Contribution = 3 mrem/yr

Cs-137 Fraction	Co-60 Fraction	Ratio Cs-137/ Co-60	Concrete Surface Gross DCGL Based on Concrete Debris Water Pathway (in dpm/100 cm^2)	DCGL from Table 5-4 (in dpm/100 cm^2)
0.1	0.9	0.1	142216	12300
0.2	0.8	0.3	77095	12900
0.3	0.7	0.4	52881	13500
0.4	0.6	0.7	40242	14300
0.5	0.5	1.0	32479	15100
0.6	0.4	1.5	27227	16000
0.7	0.3	2.3	23437	17100
0.8	0.2	4.0	20573	18300
0.9	0.1	9.0	18333	19700
0.95	0.05	19.0	17386	20500

The values for the Concrete Surface Gross DCGL are obtained using Equation 5-3. The values for f_w are the fraction of dose from water pathway values from Table 5-5, and the $DCGL_s$ is the concrete debris surface DCGL from the same table. The DCGL values are from Table 5-4. The results show that for an initial ground water contribution of 3 mrem/yr, and soil limits set at 20% of the soil DCGLs, the limiting DCGLs are based on the bounding scenario for Cs-137 to Co-60 ratios up to approximately 0.8. Above 0.8, the limiting scenario for building surface gross DCGL is the resident farmer on soil where water is obtained from a well placed within the concrete debris. As the fraction of dose from the soil is increased, the limiting concrete surface DCGL basis would shift such that the soil-water from concrete debris scenario would control at lower Cs-137 to Co-60 ratios.

As indicated in this data, the gross surface DCGL ranges from approximately 12,000 to 20,500 dpm/100cm^2. This example illustrates the method that will be applied to the variations of radionuclide mixes, and the various scenarios, using the unity rule to assure compliance with the criteria of ALARA and less than 25 mrem/yr.

5.4.6.3 Surrogate Ratio DCGLs

It is acceptable practice to make measurements to assay a difficult-to-detect radionuclide by using a surrogate measurement for an easier-to-detect radionuclide. A common example would be to assay an alpha emitting radionuclide by making a beta measurement. In such cases, in order to demonstrate compliance with the release criterion for a given survey unit, the DCGL for the surrogate radionuclide or mix of radionuclides must be scaled to account for the fact it is being used as an indicator for an additional radionuclide or mix of radionuclides. The result is referred to as a surrogate DCGL.

The surrogate DCGL is computed based on the activity ratio between a difficult-to-detect radionuclide(s) and the easy-to-measure radionuclide(s). When needed, such ratios will be established as described above, i.e., based on available waste stream or other appropriate characterization data.

The surrogate DCGL is computed as:

$$DCGL_{surrogate} = DCGL_{ETD} \times \frac{DCGL_{DTD}}{\left(f_{DTD:ETD} \times DCGL_{ETD}\right) + DCGL_{DTD}} \qquad \text{(Equation 5-4)}$$

where $DCGL_{ETD}$ = the DCGL for the easy-to-detect radionuclide;
$DCGL_{DTD}$ = the DCGL for the difficult-to-detect radionuclide; and
$f_{DTD:ETD}$ = the activity ratio of the difficult-to-detect radionuclide to the easy-to-detect radionuclide.

Consider the example from Section 5.4.6.2, where the 1999 DAW radionuclide mix was used to demonstrate the calculation of a gross activity DCGL for measurable gross beta. If the same method is applied to the gross alpha data (not shown), one finds the gross alpha DCGL to be 482 dpm/100 cm^2 (assuming all alpha-emitting radionuclides are present in excess of 5% of their DCGLs). The 1999 DAW radionuclide mix determination showed the composite gross alpha to gross beta (Co-60 and Cs-137 only) ratio to be 0.01. The gross beta DCGL can then be used as a surrogate for gross alpha by substituting these values into Equation 5-4, i.e.:

$$DCGL_{surrogate} = 14{,}400\text{dpm/100cm}^2 \times \frac{482\ \text{dpm/100cm}^2}{\left(0.01 \times 14{,}400\ \text{dpm/100cm}^2\right) + 482\ \text{dpm/100cm}^2}$$
$$= 11{,}100\ \text{dpm/100cm}^2$$

Thus, in this example, a DCGL of 11,100 dpm/100 cm^2 for directly measurable gross beta can be applied to accommodate both gross beta and gross alpha surface activity.

5.4.6.4 Elevated Measurement Comparison (EMC) DCGLs

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

Area factors are required for both the resident farmer and the building occupancy scenarios. However, since for either scenario, dose decreases less rapidly with decreasing area for the direct exposure pathway than for other pathways (e.g., inhalation, ingestion), direct exposure represents the limiting pathway for computing area factors. Of the radionuclides of concern for the Haddam Neck plant site, Co-60 represents the limiting radionuclide in terms of direct exposure per unit activity, and thus it was used for computing area factors for both scenarios.

Area factors for the resident farmer scenario were computed by running the RESRAD computer model with its default parameters and a unit activity of 1 pCi/g. Default parameters were used since the site specific parameters employed in Section 6 for computing $DCGL_W$'s do not affect the dose modeling results for Co-60 in terms of annual dose rate per unit area. Dose from Co-60 in the resident farmer scenario is dominated by direct exposure, and the difference between the site-specific contaminated zone thickness parameter and the default does not affect the direct exposure dose. The area factors for the

resident farmer scenario are provided in Table 5-7. Linear interpolation will be used when necessary for areas in between those given in the table.

For the building occupancy scenario, a somewhat different approach was used to compute the area factors than what was used to establish the $DCGL_W$'s (see Section 6). While the $DCGL_W$ is the average concentration over the entire survey unit (floor, ceiling and four walls), the $DCGL_{EMC}$ should reflect the exposure an occupant would receive from an area of elevated activity having dimensions that are much smaller than the total interior area of the room. For areas that are comparable to that for the room as a whole, evaluation against the $DCGL_W$ is appropriate since the survey unit likely would not meet the release criteria otherwise. The average concentration in the survey unit must be less than the $DCGL_W$ regardless of whether the concentration in a smaller area meets the $DCGL_{EMC}$. Thus, area factors for computing the $DCGL_{EMC}$ for the building occupancy case were computed for areas less than 100 m^2. Areas larger than 100 m^2 will be evaluated using the appropriate $DCGL_W$.

Given the 100 m^2 limit adopted for area factors and the fact that, just as for the resident farmer scenario, dose in the building occupancy scenario falls off less rapidly from direct exposure than it does from the inhalation pathway; the area factors for the occupancy scenario were computed based on direct exposure from Co-60. For convenience, point kernel calculations were used instead of RESRAD-BUILD. Area factors were determined based on the dose rate to a point receiver at a distance of one meter from a disk source of Co-60. The radius of the disk was varied to achieve the desired area while keeping the activity concentration constant. As for the resident farmer scenario given in Table 5-7, area factors for areas in between those given in the table will be determined via linear interpolation, and the area factors for the building occupancy scenario are given in Table 5-8 (Reference 5-7).

Area factors will not be computed for areas smaller than 1 m^2 for either the resident farmer or the building occupancy scenarios.

Table 5-7
Area Factors for the Resident Farmer Scenario

Area (sq. m)	Area Factor
10000	1.0
7500	1.0
5000	1.0
2500	1.0
1000	1.1
750	1.1
500	1.1
250	1.1
100	1.2
75	1.3
50	1.4
25	1.6
10	2.1
8	2.4
6	2.9
4	3.7
2	5.8
1	9.8

Table 5-8
Area Factors for the Building Occupancy Scenario

Area (sq. m)	Area Factor
100	1.0
75	1.1
50	1.2
25	1.6
10	2.4
8	2.7
6	3.3
4	4.2
2	7.1
1	12.7

5.4.6.5 Release Limits for Non-Structural Components and Systems

Residual activity limits for surfaces of non-structural components and systems will be taken from Table 1 of Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors" (Reference 5-8). Quantity of samples and surface surveys will be determined using engineering experience and controlled with plant procedures. Section 5.6 describes the survey protocol to be used with these limits.

Residual activity limits for radionuclides that cannot be practically detected in the field will be taken from NRC Policy Issue Paper SECY 94-145, "Increase of Tritium and Iron-55 Unrestricted Use Limits for Surface Contamination at Shoreham and Fort St. Vrain," (Reference 5-9). In this document, the Commission concludes that a surface residual activity limit of 2,000,000 dpm/100 cm^2 would result in a risk from Fe-55 that was equivalent to the "average risk from the other radionuclides in RG 1.86," and that the risk from tritium at this level "would be less than the average." However, for ALARA considerations, the Commission chose to reduce this value by an order of magnitude, i.e., from 2,000,000 dpm/100 cm^2 to 200,000 dpm/100 cm^2 for average residual surface activity. For the maximum residual surface activity, the Commission raised the average value by a factor of three, consistent with Regulatory Guide 1.86, to 600,000 dpm/100 cm^2. For removable activity, the Commission concluded that raising the limit for tritium and Fe-55 was not ALARA, and thus the Regulatory Guide 1.86 removable activity limit of 1,000 dpm/100 cm^2 should remain unchanged. As for all of the activity limits in Regulatory Guide 1.86, the limits for average residual activity apply to surface areas no larger than one square meter and those for maximum and removable activity apply to areas no larger than 100 cm^2.

The residual activity limits for tritium and Fe-55 given in SECY 94-145 were established using the dose conversion factors provided in NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning: Technical Basis for Translating Contamination Levels to Annual Dose" (Reference 5-10); and the RESRAD computer model. Given that the differences between the NUREG/CR-5512 dose conversion factors for other difficult-to-detect radionuclides (Ni-63, for example) and those for Fe-55 are small relative to the order of magnitude ALARA allowance used in SECY-94-145, the residual activity limits of SECY-94-145 will be extended to other difficult-to-detect radionuclides that might be encountered in final status surveys for non-structural systems and components at the HNP.

The release limits to be applied for final status surveys for non-structural systems and components at the HNP are summarized in Table 5-9.

Table 5-9
Release Limits For Non-Structural Systems And Components

Radionuclide	Average (dpm/100 cm^2)	Maximum (dpm/100 cm^2)	Removable (dpm/100 cm^2)
alpha-emitting transuranics	100	300	20
Sr-90	1,000	3,000	200
difficult-to-detect radionuclides (e.g., H-3, Fe-55, Ni-63)	200,000	600,000	1,000
beta-gamma emitters other than above	5,000	15,000	1,000

Radiological evaluations for pipes, drain lines, duct work, etc., that cannot be accessed directly will be performed via measurements made at traps and other appropriate access points where the activity levels are deemed to either bound or be representative of the interior surface activity levels.

5.5 Final Status Survey Design Elements—Grounds and Structures

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

Final status surveys for the HNP site will be designed, following MARSSIM guidance, using combinations of fixed measurements, traditional scanning surveys, and other advanced survey methods, as appropriate, to evaluate survey units relative to their applicable release criteria (see Section 5.7.1).

Under MARSSIM, the level of survey effort required for a given survey unit is determined by the potential for contamination as indicated by its classification. Class 3 survey units receive judgmental scanning and randomly located measurements or samples. Class 2 survey units receive scanning over a portion of the survey unit based on the potential for contamination, combined with fixed measurements or sampling performed on a systematic grid. Class 1 survey units receive scanning over 100% of the survey unit combined with fixed measurements or sampling performed on a systematic grid. Depending on the

sensitivity of the scanning method, the grid spacing may need to be adjusted to ensure that small areas of elevated activity are detected.

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

- For Class 1 survey units, 100% of the surface will be scanned;
- For Class 2 survey units, between 10% and 100% of the surface will be scanned in a combination of systematic and judgmental measurements for outdoor units and for floor and lower walls of structures; and 10% to 50% of the surface will be covered for upper walls and ceilings;
- Scanning will be done on a judgmental basis for Class 3 survey units.

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

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

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

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

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

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

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

5.5.1 Selecting the Number of Fixed Measurements and Locations

The MARSSIM methodology for evaluating whether a survey unit meets its applicable release criterion using fixed measurements plus scans is based on using non-parametric statistical tests for data assessment. Specifically, the methods of MARSSIM are based on two non-parametric tests: the Wilcoxon Rank Sum (WRS) test and the Sign test. The WRS test is a two sample test, making it applicable in cases where data are being compared against an appropriate reference. The Sign test is a one sample test, making it appropriate for radionuclide-specific measurements where the contaminant of interest does not appear in background or for gross activity measurements where background has been subtracted. Selection of the required minimum number of data points depends on which statistical test is going to be used to evaluate the data, and thus depends on what type of measurements are to be made (gross measurement, net measurement or radionuclide specific) and if the radionuclide(s) of interest appear(s) in background.

5.5.1.1 Establishing Acceptable Decision Error Rates

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

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

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

5.5.1.2 Determining the Relative Shift

Another input to the process of selecting the required number of measurements that is somewhat independent of the statistical test to be employed is the determination of what is called the relative shift. The relative shift is a parameter that quantifies the concentrations to be measured in a survey unit relative to the variability in these measurements. The relative shift is a function of the $DCGL_W$, a parameter called the "lower bound of the gray region" (LBGR), and either the expected standard deviation of the measurements to be made in the survey unit (σ_s) or the standard deviation established for the corresponding reference area (σ_r). The choice of σ_s or σ_r depends on whether the survey data are to be evaluated against a reference area(s). Reference areas are used if the WRS test is being applied or if gross measurements are to be background subtracted so the Sign test may be used. The σ_s values will be selected by:

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

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

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

The LBGR represents the concentration to which the survey unit must be cleaned (decontaminated) in order to have an acceptable probability of passing the statistical test. The difference between the $DCGL_W$ and the LBGR, known as the shift, can be thought of as a measure of the resolution of the measurements that will be made in a survey unit. If the LBGR is near the $DCGL_W$, the shift will be small, and thus a strong potential for Type I errors will exist. Likewise, if the shift is large, the probability of Type II errors increases. The shift is denoted as Δ.

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

To compute the relative shift, the appropriate sigma value and an initial LBGR are selected. Per MARSSIM and DG-4006, the initial value for the LBGR will be set to one-half of the $DCGL_W$. If the resulting relative shift is not between 1.0 and 3.0, the LBGR is adjusted until it is. If the relative shift is too low, the LBGR is decreased; and if the relative shift is too high, the LBGR is increased.

5.5.1.3 Selecting the Required Number of Measurements for the WRS Test

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

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

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

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

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

5.5.1.4 Selecting the Required Number of Measurements for the Sign Test

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

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{4\left(Sign\ p - 0.5\right)^2} \qquad \text{(Equation 5-6)}$$

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

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

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

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

Given the potential for small areas of elevated activity in Class 1 survey units, evaluations must be performed to assess the potential for missing such areas while scanning in locations not covered by fixed measurements. This evaluation, referred to as the Elevated Measurement Comparison (EMC), is performed by comparing the MDC of the scanning technique to be employed with the $DCGL_W$ for the

survey unit of interest. If the scanning MDC is larger than the $DCGL_W$, additional measurements are required beyond the minimum number computed via Equation 5-5 or 5-6. The effect of these additional measurement points is to tighten the grid spacing for the fixed measurements, thus reducing the probability of missing a small area of elevated activity to an acceptable level.

The number of measurements required for a Class 1 survey unit where the scan MDC is greater than the $DCGL_W$ is determined by first computing the ratio of the scan MDC to the $DCGL_W$. This ratio provides an area factor that represents the fraction of the $DCGL_W$ that can be detected via scanning, i.e.,

$$area\ factor = \frac{MDC_{scan}}{DCGL_W} \qquad \text{(Equation 5-7)}$$

The area that corresponds to the area factor is then determined by referring to the appropriate area factor table from Section 5.4.6.4. This area is denoted A_{EMC}.

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

$$N_{EMC} = \frac{A}{A_{EMC}} \qquad \text{(Equation 5-8)}$$

where A is the total area of the survey unit. N_{EMC} (computed via Equation 5-8) is then compared to N, the number of fixed measurement points computed via Equation 5-5 or 5-6. The larger of N_{EMC} or N is then used as the requisite number of fixed measurement locations and to compute the grid spacing (via Equations 5-9 or 5-10).

Additional fixed measurements for small areas of elevated activity are not considered for other than Class 1 survey units because Class 2 and Class 3 units should not have any such areas. If scanning does reveal elevated activity in Class 2 or Class 3 areas, an investigation would be performed, followed by possibly re-classifying the survey unit or subdividing it (see Section 5.5.3).

5.5.1.6 Determining Measurement Locations

For Class 1 and Class 2 survey units, fixed measurements will be performed over a systematic measurement pattern consisting of a grid having either a triangular or a square pitch. The pitch (grid spacing) will be determined based on the number of measurement required and whether the desired grid is triangular or square. Given that a triangular grid in general is more efficient than a square grid for detecting small areas of elevated activity, triangular grids should be employed for final status surveys involving fixed measurements in Class 1 and Class 2 survey units when practical.

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

also be generated in cases where a measurement cannot be made at a specific location because of an obstruction, inaccessibility, etc.

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

$$L = \sqrt{\frac{A}{0.866N}} \quad \text{for a triangular grid, or} \qquad \text{(Equation 5-9)}$$

$$L = \sqrt{\frac{A}{N}} \quad \text{for a square grid} \qquad \text{(Equation 5-10)}$$

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

In the case of Class 1 units, the value used for N in Equations 5-9 and 5-10 should be the larger of that from Equations 5-5 or 5-6 (if the scan MDC is sufficient to see small areas of elevated activity) or Equation 5-8. In all cases, the value of N should include additional measurements required to ensure against losses or unusable data.

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

5.5.2 Judgmental Assessments

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

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

The basis for judgmental assessments will be documented and will receive a technical review in accordance with plant procedures.

5.5.3 Data Investigations

5.5.3.1 Investigation Levels

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

When an investigation level is exceeded, the first step is to confirm that the initial measurement/sample actually exceeds the particular investigation level. Depending on the results of the investigation actions, the survey unit may subsequently require reclassification, remediation, and/or resurvey. Investigation levels are established for each class of survey unit. The investigation levels (criteria), to be employed for the HNP final status survey effort, are given in Table 5-10.

Table 5-10
Investigation Levels

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

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

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

In cases where an advanced survey method is used instead of fixed measurements or samples, the investigation levels given in Table 5-10 for fixed measurements or samples will be applied with the

exception of the statistical outlier test for measurements in Class 1 survey units. Since advanced survey methods record data over 100% of the area scanned, the elevated measurements comparison can be performed directly and there is no need for an investigation criterion relative to the $DCGL_W$. In cases where advanced survey methods are used as a means of traditional scanning, the investigation levels for scan measurements in Table 5-10 will be used.

5.5.3.2 Investigations

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

5.5.3.3 Remediation

If during the time of Final Status Survey, any areas of residual activity found to be in excess of the $DCGL_{EMC}$, they will be remediated to reduce the activity to acceptable levels. Any areas of residual activity may be remediated to meet the ALARA criterion. Remediation actions are discussed in Section 4 and documented as described in Section 5.9.

5.5.3.4 Re-classification

If survey results from a Class 2 unit indicate residual activity in excess of the $DCGL_W$, all or part of the survey unit will be re-classified as Class 1. The decision to re-classify all of the survey unit or sub-divide it will depend on the areal extent of the activity found to exceed the $DCGL_W$ (see Section 5.5.3.1). If a survey unit is sub-divided, the portion of the unit containing the elevated activity will be assigned the more restrictive classification, while the remainder of the unit will retain its original classification.

If survey results for a Class 3 survey unit indicate unexpected residual activity, all or part of the unit containing the activity will be re-classified. Since Class 3 survey units tend to be physically large, it is not anticipated that an entire unit would require re-classification in the event unexpected residual activity is found. However, if the unit is small enough, it is possible an entire Class 3 unit could be re-classified.

If residual activity is found in a Class 3 survey unit at levels below the $DCGL_W$, the affected portion of the unit will be re-classified as Class 2. In the unlikely event that residual activity is found in excess of the $DCGL_W$, the affected area will be re-classified as Class 1.

Re-classification of areas from a less to a more restrictive classification may be done without prior NRC approval; however, re-classification to a less restrictive classification would require prior NRC approval.

5.5.3.5 Re-survey

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

If a survey unit is sub-divided, the survey design for the remaining area of the original survey unit may or may not be affected depending on the remaining surface area of the unit and its classification. If the original survey unit was Class 3, then the only impact on the survey design (in the case of fixed measurements or sampling) is to perform additional measurements at randomly selected locations until the required total number of measurements is met (see Section 5.5.1). If the original survey unit was Class 2, the spacing of the measurement locations (from Equations 5-9 or 5-10) may need to be adjusted depending on the remaining surface area of the survey unit relative to its original area. If there is a large change in the surface area, then a new survey design will be necessary in order to accommodate the required number of measurements in the smaller area. If there is not a large change in area, the impact on the grid spacing is minimal (with respect to areal coverage), and additional measurement locations need only be selected at random in order to obtain the required number of measurements. Thus, for the purpose of final status surveys at the HNP, a change in the surface area of a Class 2 survey unit that changes the grid spacing by 25% or less will not require fixed measurements or sampling to be repeated using a tighter spacing. If the change in surface area is such that the grid spacing is changed by more than 25%, a new survey design will be required. Assessments against the 25% criterion, if applicable, will be an element of the investigation process and documented in the final status survey report for the affected survey unit.

If remediation is required in only a small area of a Class 1 survey unit, any replacement measurements or samples required will be made within the remediated area at randomly selected locations following verification that the remediation activities did not affect the remainder of the unit. Re-survey will be required in any area of a survey unit affected by subsequent remediation activities.

5.6 Survey Protocol for Non-structural Systems and Components

The guidance provided in MARSSIM and DG-4006 for conducting final status surveys does not include guidance for conducting final status surveys for non-structural system or components. Per DG-4006, non-structural systems and components refers to anything not attached to or not an integral part of a building or structure. Given that the methods of the MARSSIM do not apply to non-structural systems and components, an alternative set of release criteria must be chosen to facilitate site remediation for license termination purposes. The residual surface activity limits specified in RG 1.86 for release for unrestricted use were therefore chosen as the basis for release limits to be applied for non-structural system and components at the HNP. The limits to be applied are discussed in Section 5.4.6.4 of this plan. Survey plans will be developed for non-structural systems and components.

In accordance with Regulatory Guide 1.86, surface activity assessments for non-structural systems and components can be made by making measurements using HNP radiation protection procedures at traps and other appropriate access points where activity levels should be representative of those on the interior surfaces. Assessments may also be made via in-situ gamma-spectroscopy, provided adequate instrument efficiencies and detection limits can be achieved. Detection limits for surface activity assessments should be at least equivalent to the release limits given in Table 5-9 at the 95% confidence level. Detection limits will be computed using the methods described in Section 5.7.2.4 of this plan. If necessary, scaling factors may be applied to establish gross activity levels via radionuclide-specific measurements or other assessments, as appropriate.

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

5.7 Survey Implementation and Data Collection

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

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

An important element of the survey design process is establishing the DCGLs for the measurements to be made. The DCGLs will be determined as described in Section 5.4.6 based on characterization data for the survey unit(s) being considered. Isotopic mix, material backgrounds, and the variability of these will all be considered. The detection limit requirements dictated by the DCGLs affect the selection of both the instrumentation to be used for a given survey and the survey method(s) to be employed (advanced survey methods, fixed measurements, sampling; or combinations thereof).

5.7.1 Survey Methods

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

5.7.1.1 Scanning

Scanning is the process by which the operator uses portable radiation detection instruments to detect the presence of radionuclides on a specific surface (i.e., ground, wall, floor, equipment). The term scanning survey is used to describe the process of moving portable radiation detectors across a surface with the intent of locating residual radioactivity. Investigation levels for scanning surveys are determined during

survey planning to identify areas of elevated activity. Scanning surveys are performed to locate radiation anomalies indicating residual gross activity that may require further investigation or action. These investigation levels may be based on the $DCGL_W$ or the $DCGL_{EMC}$.

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

Table 5-11 gives the areal coverage requirements when scanning is used with fixed measurements.

Table 5-11
Traditional Scanning Coverage Requirements

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

5.7.1.2 Fixed Measurements

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

5.7.1.3 Advanced Technologies

In the context of this Plan, advanced technologies refer to survey instruments or methods that create a spatially-correlated log of the measurements made as the detector is passed over an area. This logging of all of the measurements allows quantitative assessments of activity levels to be made, thus serving the same role as fixed measurements. Having all of the measurements logged allows statistical analyses to be made using a large number of samples, which provides for enhanced detection sensitivity relative to traditional scanning. The sensitivity achieved using advanced survey methods may, in some cases, be small enough relative to the $DCGL_W$ that the advanced method alone will allow a decision to be made as

to whether a survey unit meets the release criterion without the need for additional fixed measurements. The fact that the instrument records every measurement made over the entire area it covers inherently addresses the issue of small areas of elevated activity. Average and maximum residual activity concentrations can be quantified over any area desired, allowing one to assess compliance with the applicable criteria ($DCGL_W$ or $DCGL_{EMC}$) by inspection.

5.7.1.4 Other Advanced Survey Technologies

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

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

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

5.7.1.5 Samples

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

Samples will be collected and analyzed by qualified individuals using the appropriate equipment and approved procedures. Samples will either be analyzed on-site or off-site, depending on the analyses required, the required detection limits, turn-around times, etc. Both on-site and off-site analyses will be subject to equivalent QA/QC requirements.

5.7.2 Survey Instrumentation

5.7.2.1 Instrument Selection

The selection and proper use of appropriate instruments for both fixed measurements and laboratory analyses is one of the most important factors in assuring that a survey accurately determines the radiological status of a survey unit and meets the survey objectives. The particular capabilities of a radiation detector establish its potential for being used in conducting a specific type of survey. Radiation survey parameters that will be needed for final survey purposes include surface activities and radionuclide concentrations in soil. To determine these parameters, both field measurements and laboratory analyses will be necessary. For certain radionuclides or radionuclide mixtures, both alpha and beta radiation may have to be measured. In addition to assessing average radiological conditions, the survey objectives must address identifying small areas of elevated activity.

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

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

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

Instruments must be stable and reliable under the environmental and physical conditions where they will be used, and their physical characteristics (size and weight) should be compatible with the intended application. The instrument must be able to detect the type of radiation of interest, and, depending on the application, the measurement system should be capable of measuring levels that are less than the DCGL. However, in some cases instruments used for scanning may have detection limits that are greater than the $DCGL_W$. This is allowed by MARSSIM and is acceptable as long as the grid spacing (for Class 1 survey units) and investigation levels used are in accordance with Sections 5.5.1.5 and 5.5.3.1, respectively, of this plan.

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

In general, instruments used for measurements to demonstrate that the average concentration in a survey unit is less than the $DCGL_W$ will have an MDC that is no greater than 50% of the $DCGL_W$. However, though it is not anticipated, there may be special circumstances where the best available technology cannot meet this goal. In such a case, measurements will be made at the best MDC that can be achieved, with this higher MDC accounted for in the data assessment phase. Calculation of MDCs is discussed in Section 5.7.2.4 of this plan. Section 5.7.2.5 gives nominal MDCs for the types of survey instruments and measurements anticipated for the final survey effort for the Haddam Neck Plant.

5.7.2.2 Calibration and Maintenance

All instrumentation used for measurements to demonstrate compliance with the radiological criterion for license termination at the Haddam Neck Plant will be calibrated and maintained under approved plant procedures and the project QA plan or vendor QA plan that satisfies the requirement of the project QA plan. Instruments will be calibrated for normal use under typical field conditions. Calibration standards will be traceable to the National Institute of Standards and Technology (NIST). If external vendors are used for instrument calibration or maintenance, these services must be approved and conducted under the project QA plan. Calibration records will be maintained as required by plant procedures and the project QA plan.

Instruments used to measure gross beta surface activity will be calibrated to Tc-99 or Co-60 so as to bound the beta energies for the beta-emitting radionuclides that will be encountered during final survey activities. Likewise, Pu-239 will likely be used to calibrate instruments used to assess alpha surface activity so the alpha energies of the TRU radionuclides that may be encountered are adequately bounded. However, Th-230 may also be used to calibrate instruments for alpha surface activity measurements. The alpha energy of Th-230 is (in terms of measurement efficiency) is appreciably lower than that for the

TRU radionuclides present at the HNP, meaning instruments calibrated to Th-230 will overestimate alpha surface activity levels.

Instrument efficiencies may require modifications to account for surface conditions or coverings. Such modifications, if necessary, will be established using the information in Section 5 of NUREG-1507 and pertinent site characterization data.

5.7.2.3 Response Checks

Instrumentation will be response checked in accordance with plant procedures. If the instrument response does not fall within the established range, the instrument will be removed from use until the reason for the deviation can be resolved and acceptable response again demonstrated. If the instrument fails a post-survey source check, all data collected during that time period with the instrument will be carefully reviewed and possibly adjusted or discarded, depending on the cause of the failure.

5.7.2.4 MDC calculations

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

5.7.2.4.1 MDCs for Fixed Measurements

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

$$MDC_{fixed} = \frac{3 + 4.65\sqrt{B}}{Kt} \qquad \text{(Equation 5-11)}$$

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

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

5.7.2.4.2 MDCs for Beta-Gamma Scan Surveys for Structure Surfaces

As recommended in Draft Guide-4006, MDCs for surface scans for structure surfaces for beta and gamma emitters will be computed via

$$MDC_{structure,scan} = \frac{1.38\sqrt{B}}{\sqrt{p}\varepsilon_i\varepsilon_s\left(\frac{A}{100}\right)t} \text{ dpm/100cm}^2 \qquad \text{(Equation 5-12)}$$

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

The value of 1.38 used for the sensitivity index corresponds to a 95% confidence level for detection of a concentration at the scanning MDC with a false positive rate of 60%. The numerator in Equation 5-12 represents the minimum detectable count rate that the observer would "see" at the performance level represented by the sensitivity index. The surveyor efficiency (p) will be taken to be 0.5, as recommended by DG-4006. The factor of 100 corrects for probe areas that are not 100 cm^2. In the case of a scan measurement, the counting interval is the time the probe is actually over the source of radioactivity. This time depends on scan speed, the size of the source, and the fraction of the detector's sensitive area that passes over the source; with the latter depending on the direction of probe travel. The source efficiency term (ε_s) in Equation 5-12 may be adjusted to account for effects such as self-absorption, as appropriate.

5.7.2.4.3 MDCs for Alpha Scan Surveys for Structure Surfaces

In cases where alpha scan surveys may be required, MDCs must be quantified differently than those for beta-gamma surveys because the background count rate from a typical alpha survey instrument is nearly zero (1 to 3 counts per minute typically). Since the time that an area of alpha activity is under the probe varies and the background count rates of alpha survey instruments is so low, it is not practical to determine a fixed MDC for scanning. Instead, it is more useful to determine the probability of detecting an area of contamination at a predetermined DCGL for given scan rates. However, in general, it is expected that separate alpha and beta surface activity measurements will not be necessary at the HNP and that surrogate measurements will instead be used for alpha surface activity assessments (see Section 5.4.6.3).

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

5.7.2.4.4 MDCs for Gamma Scans of Land Areas

As recommended in DG-4006, the values given in Table 6.7 of the MARSSIM will be adopted for gamma scans of land areas if NaI detectors of the dimensions considered in the table are used. If larger NaI detectors (e.g., 3 inch by 3 inch) or other detector types (e.g., plastic scintillator) are used, then the scan MDC will be computed using the methods of Section 6.7.2.1 of the MARSSIM. This is the same method as was used to derive the values given in MARSSIM Table 6.7.

The radionuclides represented in MARSSIM Table 6.7 encompass those expected to be encountered in gamma scans for land areas at the HNP. If desired, the methods of Sections 5.4.6.1 and 5.4.6.2 of this plan may be used to establish scan MDCs based on radionuclide mix ratios. Alternatively, the most limiting value for the radionuclide mix may be used, with most limiting in this case meaning the radionuclide for which the MDC is the largest fraction of its $DCGL_W$ for soil.

5.7.2.5 Typical instrumentation and MDCs

Table 5-12 provides nominal data for the types of field instrumentation anticipated for use in the final survey efforts for the Haddam Neck Plant.

Table 5-12
Available Instruments and Associated MDCs

Instrument	Application	Nominal Efficiency	Nominal Background	Nominal MDC (fixed measurement)	Nominal Scan MDC
pancake GM probe (20 cm^2)	beta-gamma scans or fixed measurements for structure surfaces	17% (Tc-99)	50 cpm	1,050 dpm/100 cm^2 (1 minute count)	3140 dpm/100 cm^2
gas proportional counter (100 cm^2)	alpha or beta scans or fixed measurements for structure surfaces	β plateau: 16% (Tc-99); α plateau: 23% (Am-241)	350 cpm (β plateau); 15 cpm (α plateau)	560 dpm/100 cm^2 (β plateau) 90 dpm/100 cm^2 (α plateau); 1 minute counts	1770 dpm/100 cm^2 (β plateau); 400 dpm/100 cm^2 (α plateau)
plastic scintillator (100 cm^2)	beta-gamma scans or fixed measurements for structure surfaces	30% (Co-60)	600 cpm	390 dpm/100 cm^2 (1 minute count)	1230 dpm/100 cm^2
dual-phosphor scintillator (100 cm^2)	scans or fixed measurements; α and β, independently or simultaneously	20% (Co-60) 18% (Am-241)	300 cpm (β mode); 6 cpm (α mode)	420 dpm/100 cm^2 (β mode); 80 dpm/100 cm^2 (α mode)	1300 dpm/100 cm^2 (β mode); 400 dpm/100 cm^2 (α mode)
ZnS scintillator (100 cm^2)	alpha scans or fixed measurements on structure surfaces	19% (Pu-239)	2 cpm	50 dpm/100 cm^2 (1 minute count time)	400 dpm/100 cm^2
1.25-inch by 1.5-inch NaI	gamma scans for soil	Varies with energy	Varies with energy	N/A	6 pCi/g Co-60 11 pCi/g Cs-137
HPGe	in-situ gamma spectroscopy - soil	Varies with energy and geometry	Varies with energy and geometry	0.05 pCi/g Co-60 0.05 pCi/g Cs-137 (10 minute counts)	N/A
3-inch by 3-inch NaI	in-situ gamma spectroscopy - soil	Varies with energy and geometry	Varies with energy and geometry	0.1 pCi/g Co-60 0.2 pCi/g Cs-137 (10 minute counts)	N/A
position-sensitive proportional counter	scan-and-record surveys	Co-60 (β): 18% Am-241 (α): 23%	350 cpm/100 cm^2 beta 15 cpm/100 cm^2 alpha	Typical values are 1,925 dpm/100 cm^2 β and 200 dpm/100 cm^2 α	

5.7.3 Survey Considerations

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

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

5.7.3.1 Survey Considerations for Buildings, Structures and Equipment

The condition of surfaces following decontamination activities can affect the choice of survey instruments and techniques. Removing contamination that has penetrated a surface usually involves removing the surface material. As a result, the floors and walls of decontaminated facilities can be scarred or broken up and uneven. Such surfaces are more difficult to survey because it is not possible to maintain a fixed distance between the detector and the surface. In addition, scabbled or porous surfaces may attenuate radiation - particularly alpha and low-energy beta particles, and pose an increased risk of damage to detector probe faces. Surface irregularities may also cause difficulty in rolling or maneuvering detector systems on wheels. In such cases, depending on the survey instrumentation to be used and the radiations of interest, evaluations of instrument efficiencies may be made to assess the need for corrections for variables such as source-to-detector distance or surface condition. Surface efficiency corrections, if necessary, may be established using the information given in Section 5 of NUREG-1507 or, if practical, via fixed measurements. If any corrections to measurement efficiencies are required, the impact of these corrections on instrument MDCs will be assessed to ensure that measurements can still be performed with the required sensitivity relative to the applicable DCGLs.

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

5.7.3.1.1 Activity Beneath Surfaces

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

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

5.7.3.1.2 Sewer Systems, Plumbing and Floor Drains

Residual radioactivity in piping or floor drains will be evaluated in the same manner as for systems and components, as discussed in Sections 5.4.6.5 and 5.6. Assessment of residual activity levels in piping or floor drains will be via sampling of sediments, fixed measurements, or both, as appropriate.

All non-RCA sanitary plumbing systems at the Haddam Neck Plant drain to on-site leach fields. Thus, if any residual activity is suspected in these systems, evaluations for both the leach fields and the associated system piping may be required. Radiological assessments for piping will be made via the guidance of Regulatory Guide 1.86 as described in Section 5.6 of this plan, i.e., by making measurements at traps and other appropriate access points where activity levels should be representative of those on the interior surfaces. Assessments may also be made via in-situ gamma-spectroscopy provided adequate instrument efficiencies and detection limits can be achieved. Evaluations required for any affected leach fields will be made as described in Section 5.7.3.2 of this plan for sub-surface activity.

All RCA plumbing systems drained to the aerated drains system and was part of the normal plant effluent. Thus, there are no leach fields associated with these systems. Any piping remaining following decommissioning will be evaluated as described above.

5.7.3.1.3 Ventilation Ducts – Interiors

Ventilation system interiors will be addressed in the same manner as for plumbing and floor drains, as discussed above. Exterior surfaces of such systems will be evaluated as part of the building or structure in cases where the system is attached to it or is otherwise an integral component. Otherwise, as with the interior surfaces, the exterior surfaces will be evaluated against the criteria of Regulatory Guide 1.86 and Policy Issue Paper SECY-94-145 as discussed in Sections 5.4.6.5 and 5.6 of this plan.

5.7.3.1.4 Piping and Embedded Piping

The construction of the Haddam Neck plant was such that there is not expected to be a significant amount of embedded piping to consider in the final survey effort. Most of the radiologically affected piping was in pipe trenches, and thus can be accessed and removed as necessary. Any affected embedded piping remaining after decommissioning is expected to be in wall penetrations between areas. Sections of such piping are not expected to be very long (no longer than the wall thickness) and thus should be able to be sampled or surveyed as appropriate to evaluate residual activity levels against the applicable release criteria. In cases where piping diameters and other variables are such that there is easy access to the

interior, surveys can be conducted via fixed measurements. Indirect methods, such as those described in Sections 5.6 and 5.7.3.1 of this plan, may be required in cases where there is not easy access to piping interior surfaces.

5.7.3.1.5 Activated Concrete

In-situ gamma spectroscopy will typically be used to survey activated concrete, after any required remediation, to demonstrate that it meets the applicable volumetric DCGLs (see Table 6-20). Such surveys would be conducted so that 100% of the affected volume was covered in overlapping measurements. Embedded materials (such as rebar) will be treated as concrete for purposes of assigning DCGLs. Assessments for any difficult-to-detect radionuclides that might be present in activated concrete will be by either direct measurements (core-bores or equivalent) or by establishing surrogate DCGLs for these radionuclides relative to some radionuclide easily measured via gamma-spectroscopy (Co-60, for example). Surrogate ratios will be established using pertinent characterization data for the survey unit of interest.

5.7.3.1.6 Systems and Equipment Interiors and Exteriors

In accordance with Regulatory Guide 1.86, surface activity assessments for non-structural systems and components will be made by making measurements at traps and other appropriate access points where activity levels should be representative of those on the interior surfaces. Assessments may also be made via in-situ gamma-spectroscopy, provided adequate instrument efficiencies and detection limits can be achieved. If necessary, scaling factors may be applied to establish gross activity levels via radionuclide-specific measurements or other assessments, as appropriate.

5.7.3.2 Survey Considerations for Outdoor Areas

5.7.3.2.1 Residual Radioactivity in Surface and Subsurface Soils

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

Sub-surface radioactivity refers to residual radioactivity that is underneath structures such as building floors/foundations or material that is covered with clean soil or some other unaffected layer(s).

The historical site assessment will be consulted to identify those survey areas where the potential exists for sub-surface radioactivity. Such areas include, but are not limited to, areas under buildings, building floors/foundations, or components where leakage was known or suspected to have occurred in the past; on-site storage areas where radioactive materials have been identified; and areas containing spoils from past dredging of the discharge canal. Data from both the historical site assessment and any pertinent characterization data will be used to establish a bounding depth profile for any potential sub-surface radioactivity. All survey units within the area will then be "surveyed" by taking core samples to the required depth, homogenizing over each one meter of depth. The number of samples required is determined via the methods of Section 5.5 of this plan. In accordance with DG-4006, scanning is not applicable to sub-surface activity assessments.

The dose modeling performed to develop the DCGLs for the Haddam Neck Plant is expected to be valid for any sub-surface radioactivity that may be encountered. Thus, additional, site-specific dose modeling to address sub-surface radioactivity is not expected to be necessary.

5.7.3.2.2 Paved Areas

Paved areas that remain at the HNP following decommissioning activities may require surveys for residual radioactivity on the surface, beneath the surface, or both. Surveys for pavement surfaces will be performed using the techniques similar to those for structure surfaces (i.e., appropriate combinations of fixed measurements and scanning). In cases where surveys for residual radioactivity beneath a paved surface are required, these will be performed using the same techniques as for surface soils. This includes areas with multiple layers of paving material.

5.7.3.2.3 Ground Water

Assessments of any residual activity in ground water at the Haddam Neck Plant will be via ground water monitoring wells. The monitoring wells installed at the site will monitor ground water at both deep and shallow depths.

5.7.3.2.4 Surface Water and Sediments

Based on the historical site assessment, there are no surface water bodies at the HNP site that contain sediments thought to have the potential to contain residual radioactivity at levels near the DCGLs. Sediments will be assessed by collecting samples within locations of surface water ingress or by collecting composite samples of bottom sediments, as appropriate. Such samples will be collected using approved procedures based on accepted methods for sampling of this nature. Sample locations will be established using the methods of Section 5.5 of this plan. Scanning in such areas is not applicable.

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

5.7.3.2.5 Storm Drains and Buried Piping

Any buried piping that remains following decommissioning activities that has a potential to contain residual activity will be addressed in the same manner as for plumbing and floor drains, as discussed in Section 5.7.3.1.2 above. Buried piping in this context includes storm drain lines.

5.8 Survey Data Assessment

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

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

methods, as appropriate, to determine if they exceed the release criterion. If the release criterion has been exceeded or if results indicate the need for additional data points, appropriate further actions will then be determined.

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

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

If the survey data are in the form of gross (non-radionuclide-specific) measurements or if the radionuclide of interest is present in background in a concentration that is a relevant fraction of the $DCGL_W$, then the initial data evaluation will be as described in Table 5-13.

Table 5-13
Initial Evaluation of Survey Results
(Background Reference Area Used)

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

If the survey data are in the form of radionuclide-specific measurements and the radionuclide(s) of interest is not present in background in a concentration that is a relevant fraction of the $DCGL_W$, then the initial data evaluation will be as described in Table 5-14.

Table 5-14
Initial Evaluation of Survey Results
(Background Reference Area Not Used)

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

5.8.1 Wilcoxon Rank Sum Test

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

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

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

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

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

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

5.8.2 Sign Test

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

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

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

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

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

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

Though it is not anticipated, if any of the data collected from a final status survey are reported as "less than MDC" or as background, these data will be assigned a value of the MDC for the measurement technique employed for purposes of applying the Sign test.

5.8.3 Elevated Measurement Comparison

The Elevated Measurement Comparison (EMC) consists of comparing each measurement from the survey unit with the investigation levels discussed in Section 5.5.3. The EMC is performed for both measurements obtained on the systematic-sampling grid and for locations flagged by scanning measurements. Any measurement from the survey unit that is equal to or greater than an investigation level indicates an area of relatively high concentrations that should be investigated, regardless of the

outcome of the nonparametric statistical tests. Thus, the use of the EMC against the investigation levels may be viewed as assurance that unusually large measurements will receive proper attention regardless of the outcome of those tests and that any area having the potential for significant dose contributions will be identified. The EMC is intended to flag potential failures in the remediation process. It should not be used as the primary means to identify whether or not a unit meets the release criterion.

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

$$\frac{\delta}{DCGL_W} + \frac{\overline{C}_{elevated} - \delta}{(AreaFactor) \times DCGL_W} < 1 \qquad \text{(Equation 5-13)}$$

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

A separate term will be used in Equation 5-13 above for each elevated area identified in a survey unit.

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

5.8.4 Unity Rule

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

$$\frac{C_1}{DCGL_1} + \frac{C_2}{DCGL_2} + \ldots + \frac{C_n}{DCGL_n} \leq 1 \qquad \text{(Equation 5-14)}$$

where:

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

5.8.5 Data Assessment Conclusions

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

Retrospective power analyses, if necessary, will be performed following the methods of MARSSIM Sections I.9 and I.10 for the Sign test and WRS test, respectively. If the analysis finds that an inadequate number of samples were collected to support the data assessment for a given survey unit, additional samples may be collected and the analysis repeated. Increasing the number of samples (measurements) acquired within a given survey unit increases the probability of rejecting the null hypothesis when it is indeed false. Likewise, if the analysis with the additional samples still concludes that the residual concentration in the survey unit exceeds the $DCGL_W$, then the unit must be remediated and resurveyed.

5.9 Final Status Survey Reports

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

- a physical description of the survey area which encompasses the unit (in many cases, areas and units will be the same);
- a summary of the characterization data associated with the area including any required investigations, re-classifications or subdivisions;
- the classification history of the unit;
- a description of remediation activities (if any) performed during final status survey;
- results and discussion of any ALARA evaluations performed;
- a discussion of the survey design (dose modeling validation; combination of scans, fixed measurements, and samples used; number of measurements; grid spacing; etc.);
- tabular and graphical depictions of survey results including quality control results;
- discussions of data assessments (see Section 5.8), including graphical depictions; and
- a statement that the survey unit meets the applicable release criteria.

In most cases, final status survey results will be made available to the NRC for survey areas rather than for individual survey units. Reports will be compiled after final status survey activities for all of the survey units within a given area are completed. Where appropriate FSS reports may address multiple survey areas. This approach should minimize the incorporation of redundant historical assessment information and provide for a logical approach to review and independent verification in that a more complete description of the final radiological status of an area will be provided.

5.10 Quality Assurance and Quality Control Measures

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

The CYQAP complies with the requirements set forth in Appendix B, of 10 CFR Part 50, along with applicable sections of the Updated Final Safety Analysis Report (UFSAR) for the license application, and

is responsive to Regulatory Guide 1.70, which describes the information presented in the Quality Assurance Section of the UFSAR for nuclear power plants.

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

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

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

Organization

The Vice President – Operations and Decommissioning is responsible for ensuring the implementation of site programs and processes. The Manager(s) of CY Construction and Safety Oversight review and monitor decommissioning activities at HNP. The Manager – Nuclear Safety directs and administers independent audits, surveillances, and inspections for the Final Status Survey. Both CY Oversight and Nuclear Safety have the authority to stop unsatisfactory Final Status Survey activities.

Quality Assurance Program

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

Design Control

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

Procurement Document Control

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

Procedures, Instructions and Drawings

The performance of the FSS will require procedures for personnel training, survey implementation, data collection, chain of custody, verification and record storage. These procedures will be developed to ensure compliance with the License Termination Plan and will meet applicable quality requirements. These requirements include that the procedures be developed utilizing the guidance of an approved procedure and will receive the appropriate review and approval.

Document Control

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

Control of Purchased Material, Equipment, and Services

Vendors may be used for the performance of the FSS and laboratory activities. The procurement of these services will be governed by an approved process as defined in implementing procedures. Additionally, audits and surveillance of these contractors will be performed to provide an adequate level of assurance that the quality activities are being effectively performed.

Inspection

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

Control of Measuring and Test Equipment

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

Handling, Storage, and Shipping

Some of the material samples will be transported to off-site laboratories for analysis. The process for controlling this material will be sufficient to ensure that a chain of custody is maintained. Additionally, protocols must be established to ensure that there is no cross-contamination

between samples and sample packaging. Appropriate controls will be defined in administrative procedures to ensure that sample integrity is maintained.

Nonconforming Materials, Parts, Components, or Services

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

Corrective Action

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

Quality Assurance Records

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

Controls of Special Processes

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

Quality Assurance Audits

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

Inspection, Test, and Operating Status

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

5.11 References

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

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

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

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

5-5 Technical Support Document CY-HP-0025, Revision 2, dated February 12, 1999.

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

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

5-8 Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," June 1974.

5-9 SECY 94-145, "Increase of Tritium and Iron-55 Unrestricted Use Limits for Surface Contamination at Shoreham and Fort St. Vrain," dated May 27, 1994.

5-10 NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning: Technical Basis for Translating Contamination Levels to Annual Dose," dated May 1999.

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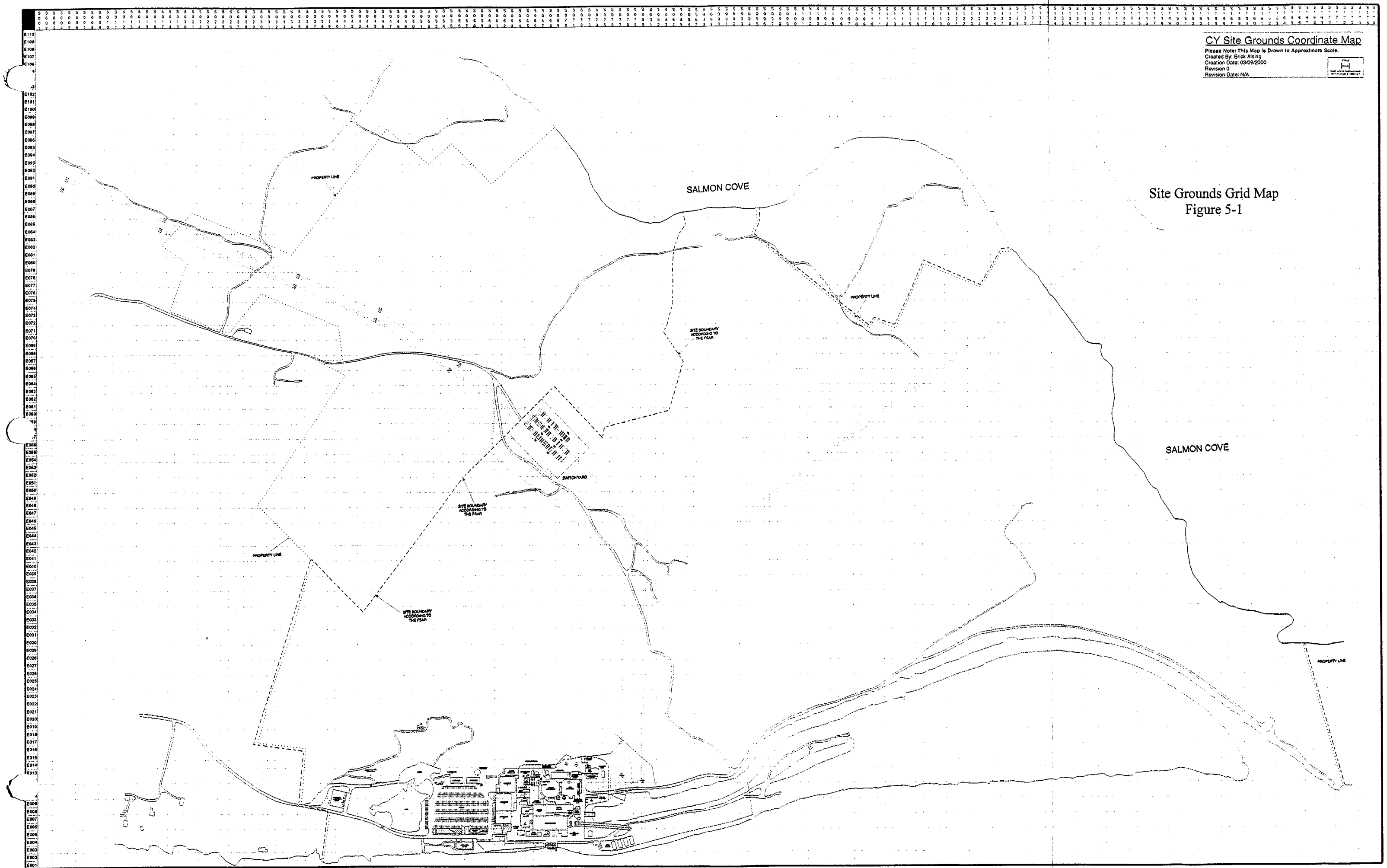
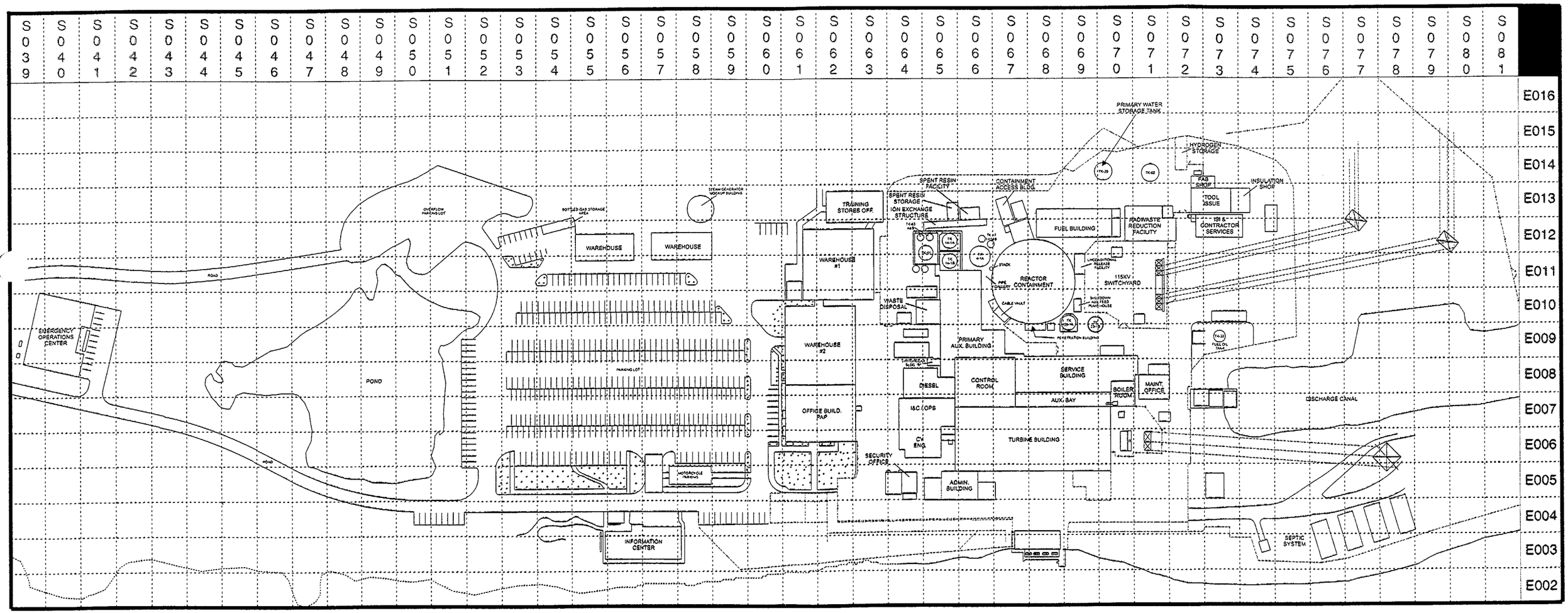


Site Grounds Grid Map
Figure 5-1

Site Grid Map
(Site View)

Site Area Grid Map
Figure 5-2



CONNECTICUT RIVER

Please Note: This Map is Drawn to Approximate Scale

Created By Erick Alsing
Date: 03/09/2000
Revision 0
Revision Date: N/A

SCALE
Each Grid is Approximately 60 ft in length & 3600 sq ft

6 COMPLIANCE WITH THE RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

6.1 Site Release Criteria

6.1.1 Radiological Criteria for Unrestricted Use

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

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

- ALARA Criterion: The residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA).

In addition, the criteria of RG 1.86 (Reference 6-2) will be applied to non-structural components and systems as discussed in Section 5.6.

6.1.2 Conditions Satisfying the Site Release Criteria

Levels of residual radioactivity that correspond to the allowable radiation dose and ALARA levels described above are calculated by analysis of various scenarios and pathways (e.g., direct radiation, inhalation, ingestion) through which exposures could be reasonably expected to occur. These derived levels, or derived concentration guideline levels (DCGLs), form the basis for the following conditions which, when met, satisfy the site release criteria as prescribed in 10 CFR 20.1402:

- The average residual radioactivity above background is less than or equal to the DCGL.

- Individual measurements, representing small areas of residual radioactivity which exceed the DCGL, do not exceed the elevated measurement comparison DCGL. The elevated measurement comparison DCGL ($DCGL_{EMC}$) is described in Section 5.4.6.4.

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

- Remediation is performed where it is ALARA to reduce the levels of residual radioactivity to below those concentrations necessary to meet the DCGLs. (See Section 4 for a detailed discussion of ALARA considerations).

The methods in MARSSIM and the DCGLs may not be appropriate for complex non-structural components. For those non-structural components that are considered to be complex to survey (e.g., pumps, heat exchangers, and diesel generators), the criteria of RG 1.86 will be applied during the final status survey. The component or system satisfying these criteria may be released, when the survey area

containing the non-structural component is confirmed to meet the radiological criteria for unrestricted release.

6.2 Dose Modeling Scenarios

Dose models, appropriate to the HNP site, are based on the guidance found in DG-4006 (Reference 6-3), NUREG-1549 (Reference 6-4), and NUREG/CR-5512, Volume 1 (Reference 6-5). These models are used to calculate the required DCGLs referred to in Section 6.1. Each dose model translates residual radioactivity levels into potential radiation doses to the public and is defined by three factors: 1) the scenario, 2) the exposure pathways, and 3) the critical group. The scenarios described in NUREG/CR-5512, Volume 1, address the major exposure pathways of direct exposure to penetrating radiation and inhalation and ingestion of radioactive materials. The scenarios also identify the critical group. The critical group is the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity within the assumptions of the particular scenario. The scenarios and their modeling are specifically designed to be reasonably conservative by generally overestimating rather than underestimating potential dose.

Two primary scenarios were considered for HNP. They are:

- Resident Farmer
- Building Occupancy

The resident farmer scenario was chosen to develop soil DCGLs for impacted areas at the HNP site. The applicability of this scenario and dose modeling approach used to develop DCGLs are described in Section 6.4.

The building occupancy scenario was used to develop DCGLs for the buildings. Section 6.5 describes the applicability of this scenario and the associated dose model used to develop DCGLs for building surfaces. This section also describes two additional scenarios that were evaluated to ensure that the DCGLs developed for building surfaces are conservative and bounding if building concrete is used as on-site backfill. These include a resident farmer scenario for concrete debris and an excavator scenario. DCGLs have been adjusted where ground water in contact with the dispositioned concrete debris impacts the allowable levels of residual activity on concrete surfaces.

6.3 Operational DCGLs

Since additional scenarios may be created by combining the two primary scenarios described above, a method to combine the dose contribution from each scenario is necessary. For example, a resident farmer may locate his residence and raise crops on soil containing residual radioactivity and use ground water that is in contact with the dispositioned concrete debris, which may also contain residual radioactivity. Soil and building surface DCGLs for these combined scenarios will be determined on an operational basis. Section 5.4.6.1 describes the methodology in detail.

6.4 Development of DCGLs for Soil

Soil at the HNP site may have been contaminated from licensed operations by unplanned liquid releases, long-term accumulation of material in the soil from effluent releases, or by material with surface and/or volumetric radioactive contamination. Dose modeling was used to develop DCGLs for these soils (i.e., the residual radioactivity that, if left in HNP site soils, would result in a TEDE of 25 mrem/year). The

approach generally follows that recommended in NUREG-1549. Calculation of radionuclide-specific DCGLs included the following steps:

1. identifying the critical group;
2. developing a conceptual model;
3. defining the scenario;
4. developing "Base Case DCGLs"; and
5. developing "Special Case DCGLs".

An overview of the dose modeling approach is provided below.

First, the critical group is defined. Based on demographic data for Middlesex and nearby counties and the characteristics of the HNP site, the resident farmer was selected as the average member of the critical group. Section 6.4.1 provides the basis for this selection.

A conceptual model of the system is then developed based on physical, geological, hydrogeological, and contaminant characteristics specific to the site. Section 6.4.2 describes the conceptual model and identifies the radionuclides attributed to licensed operations that could reasonably be expected to be found in site soils.

Based on the selection of the resident farmer as the average member of the critical group and the conceptual understanding of the site, a resident farmer scenario is chosen for dose modeling. Section 6.4.3 describes this scenario and identifies the associated pathways.

Next, "Base Case DCGLs" are determined for each radionuclide using a dose model representing the resident farmer scenario. These DCGLs assume that there is no radioactivity currently in the ground water. Base Case DCGLs are calculated using the characteristics of the industrial area as well as behavioral and metabolic characteristics recommended in NUREG/CR-5512, Volume 2. Significant exposure pathways are identified along with the parameters associated with these pathways. A quantitative sensitivity analysis is performed to determine the parameters that have the greatest influence on the calculated DCGLs. The values assigned to the most sensitive parameters are justified. Details of the dose modeling used to develop the Base Case DCGLs are provided in Section 6.4.4.

Section 6.4.5 discusses the applicability of the Base Case DCGLs, which are based on the characteristics of the industrial area, to other impacted areas of the HNP site. These other areas have some characteristics that differ from the industrial area. The results of the preceding sensitivity analysis are used to demonstrate that the Base Case DCGLs determined for the industrial area bound all other impacted areas.

Finally, "Special Case DCGLs" are presented in Section 6.4.6. Ground water underlying portions of the industrial area currently contains radioactivity related to licensed operations. The dose from this existing radioactivity was calculated for currently observed levels. DCGLs were then determined for soil, termed Special Case DCGLs, that account for the current dose contribution from ground water. The survey areas to which these Special Case DCGLs apply are also identified. Section 6.4.6 presents the methodology and DCGLs resulting from this calculation. If ongoing monitoring shows that ground water concentrations have changed significantly at the time of the final status survey (FSS), the Special Case DCGLs will be revised as appropriate.

Note that soil DCGLs have been developed on a site-specific basis in lieu of adopting the interim screening values published by the NRC in the Federal Register, December 7, 1999. The interim screening values are not applicable to sites with subsurface and/or with ground water contamination. As discussed

in Section 2, the HNP site has both subsurface soil contamination and ground water contamination. These features require that the dose assessment be conducted on a site-specific basis.

Lastly, it should be recognized that soil DCGLs have been developed based on the preliminary site characterization data. Site characterization will continue as part of the decommissioning process. If future site characterization data indicate contaminant characteristics to be (non-conservatively) different from those stated in this LTP, soil DCGLs will be revised as appropriate using the change criteria process presented in Section 1.5.

6.4.1 Critical Group

Demographic and economic data relevant to farming for Haddam, East Haddam, and Middlesex County, Connecticut were used to identify a critical group. These data are summarized in Table 6-1. Although a relatively small percentage of land (8%) and workforce (less than 3%) in Middlesex County and adjoining towns are directly related to agricultural activities, these data demonstrate that agriculture is a feasible lifestyle and means of employment in Middlesex County, and will likely continue to be for the foreseeable future. In addition, the flat topography of the industrial area at the HNP site, which is adjacent to the Connecticut River, would be conducive to agricultural activities. Therefore, the average member of the critical group was determined to be the resident farmer who lives on the plant site following site remediation, grows all of their diet on the site, and drinks water from a ground water source at the site (NUREG-1549 and NUREG/CR-5512, Volume 1).

It is unlikely that any other set of plausible human activities could occur onsite that would result in a dose exceeding that calculated for the hypothetical resident farmer. It is more likely that the behavior of future occupants would result in a lower dose. For example, it is more likely that the HNP site will be repowered using fossil fuel technology to take advantage of existing infrastructure. In this case, an industrial worker, as opposed to a resident farmer, would occupy the site. The hypothetical dose from the soil to an industrial worker would be less than a resident farmer since the industrial worker would not ingest food derived from onsite. Therefore, the use of the resident farmer as the critical group is both conservative and bounding for the calculation of soil DCGLs.

Table 6-1
Agricultural Statistics for Middlesex, East Haddam and Haddam Counties

Statistic	Value
Middlesex County[1]	
- Number of farms	288
- Total county farm acreage	18,682 acres
- Percent of total acreage used for farming	8 %
- Average farm size	65 acres
- Percentage of farms raising beef cattle	23 %
- Percentage of farms raising milk cows	3.5 %
- Percentage of farms harvesting Hay	4.3 %
- Percentage of farms harvesting vegetables	16 %
- Percentage of farms harvesting silage	43%
- Percentage of Farms with orchards	5.2 %
East Haddam[2]	
- Percentage of labor force involved in agriculture	2.4 %
Haddam[2]	
- Percentage of labor force involved in agriculture	1.3 %

[1] *1997 Census of Agriculture*, Volume 1 Geographic Area Series, "Table 1. County Summary Highlights: 1997" (Reference 6-6)
[2] *Connecticut Town Profiles 1998-1999*. Connecticut Department of Economic and Community Development; Research Section, Public and Government Relations Division (Reference 6-7)

6.4.2 Conceptual Model

A conceptual model was developed based on the site characteristics expected at the time of unrestricted release. This model is intended to represent the industrial area, which is shown in Figure 6-1. Developing a conceptual model requires an understanding of the physical, geologic, and hydrogeologic characteristics of the area and the relationship of these characteristics to contaminant source areas and potential pathways. A discussion of these conceptual model elements is provided below.

Physical Characteristics

The industrial area of the HNP site is located on the east bank of the Connecticut River on a level, 600 ft wide terrace at an elevation of 21 ft mean sea level (msl). A parking lot occupies the area to the north of the industrial area. The area north of the parking lot is occupied by a pond. To the south, a 5500 foot-long cooling water discharge canal leads to the river from the southern edge of the industrial area. It is separated from the Connecticut River by a 200 to 1,000 ft wide peninsula flood plain that ranges in elevation from about 5 to 15 ft msl. A steep wooded hillslope rises immediately east of the industrial area to elevations over 300 ft msl. The lowermost 30 to 40 ft of the hillside adjacent to the plant consists of nearly vertical rock cut.

Geologic and Hydrogeologic Characteristics

The geology and hydrogeology of the industrial area is documented in Reference 6-8. Cross sections depicting geologic and hydrogeologic characteristics are given in Figure 6-2. A brief discussion of the site characteristics is provided below.

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

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

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

The shallow ground water flow beneath the industrial area occurs within the unconsolidated sediments and bedrock. The depth to the water table averages about 10 ft below ground surface (bgs) in this area. Ground water generally flows southwest and downward near the hillslope, and upwards near the discharge canal and the Connecticut River. Locally, the Containment Building and mat drain sump are important hydrogeologic features. The ground water flow pattern around the impermeable Containment Building is distorted with a component of flow toward the drainage system under the Containment Building. The mat drain sump, located on the southern side of the Containment Building, currently removes ground water and depresses the water table around it. During decommissioning, the pump will be shut off, and the mat drain sump will no longer influence ground water flow patterns. The cooling water discharge tunnels divert the shallow ground water flowing around the southwestern side of the Containment Building farther to the south. Southwest of tunnels, the shallow ground water appears to flow southwesterly and upwards directly toward the river.

Contaminant Characteristics

Soil within the industrial area may be contaminated from licensed operations by unplanned liquid releases or long-term accumulation of material in the soil via effluent releases. The impacted soil includes that in current open areas as well as that which will be exposed in the future following demolition of overlying buildings and structures. The areas wherein soil could potentially contain residual radioactivity are identified and described in Section 2. Based on the documented release mechanisms, the residual radioactivity is generally confined to the surface soil layer. The surface soils in the industrial area are

composed of a silty sand that was imported as artificial fill. Site survey results indicate that there may be localized areas where the soil contamination is deeper, but still restricted to the unsaturated zone.

Site surveys have identified radionuclides that may be present in measurable quantities in site soils and that are likely associated with licensed plant operations. Table 6-2 summarizes these radionuclides and their half-lives and distribution coefficients. The distribution coefficient, represented with the symbol K_d, is the ratio of the mass of solute adsorbed or precipitated on the solids per unit of dry mass of soil to the solute concentration in the water. The distribution coefficient determines the mobility of a dissolved constituent in ground water, i.e., the larger the distribution coefficient the less mobile the radionuclide.

Table 6-2
Radionuclides Potentially Present in HNP Site Soils and Associated Properties

Radionuclide	Half-life[a] (years)	Distribution Coefficient (cm^3/g)
H-3	12.33	0[b]
C-14	5,730	4.34[b]
Mn-54	0.8561	84.1[b]
Fe-55	2.685	535[b]
Co-60	5.271	1,510[b]
Ni-59	7.5×10^4	37[b]
Ni-63	100	37[b]
Sr-90	28.8	31.4[b]
Nb-94	2.0×10^4	0.826[b]
Tc-99	2.14×10^5	7.37[b]
Cs-134	2.062	10.5[b]
Cs-137	30.17	10.5[b]
Eu-152	13.3	824.9[c]
Eu-154	8.5	824.9[c]
Eu-155	4.96	824.9[c]
Pu-238	87.74	13.6[b]
Pu-239	2.41×10^4	13.6[b]
Pu-241	14.4	13.6[b]
Am-241	432.2	1,430[b]
Cm-243	28.5	1,378[c]

[a] RESRAD Users Manual, Table 3.1 (Reference 6-9)
[b] NUREG/CR-5512, Volume 2, Table D.22 (Reference 6-10)
[c] RESRAD default value (Reference 6-11)

6.4.3 Scenario Definition

The resident farmer was identified as the critical group in Section 6.4.1. The characteristics of the industrial area indicate that the resident farmer scenario, as defined in NUREG-1549 and NUREG/CR-5512, Volume 1, is applicable. The resident farmer scenario defines the potential pathways that can be used to estimate human radiation exposure resulting from residual radioactivity in the soil. For this scenario, the residual radioactivity is assumed to be contained in a surface-soil layer on property that can be used for residential and light farming activities.

The potential pathways that apply to the resident farmer at HNP include the following:

1. Direct exposure to external radiation from the contaminated soil material;
2. Internal dose from inhalation of airborne radionuclides; and
3. Internal dose from ingestion of
    a) Plant foods grown in the contaminated soil and irrigated with contaminated water,
    b) Meat and milk from livestock fed with contaminated fodder and water,
    c) Drinking water from a contaminated well,
    d) Fish from a contaminated pond, and
    e) Contaminated soil.

Figure 6-3 schematically depicts the relevant exposure pathways.

6.4.4 Base Case DCGLs

Base Case DCGLs were calculated for the industrial area for the resident farmer scenario using the RESRAD (Version 5.91) computer code (Reference 6-9). RESRAD was developed by Argonne National Laboratory to calculate site-specific residual radiation guidelines and dose to future hypothetical on-site individuals at sites contaminated with residual radioactive material.

The implementation of RESRAD for the HNP site considers the guidance provided in NUREG-1549 and "Preliminary Guidelines for Evaluating Dose Assessments in Support of Decommissioning" (Reference 6-12). The latter document recommends an approach that includes the following steps:

1. Run RESRAD using as much site or regional data as possible to define physical parameters, along with the behavioral and metabolic parameters recommended in NUREG/CR-5512, Volume 2;
2. Identify significant exposure pathways (i.e., those contributing the greatest percentages to dose);
3. Identify the parameters associated with significant exposure pathways;
4. Perform sensitivity analysis on those parameters to determine which ones have the greatest influence on the calculated dose; and
5. Justify the values of the most sensitive parameters using site-specific information.

The methodology implemented to complete each of these steps along with the associated results are described in this section.

<u>RESRAD Analysis</u>

RESRAD input parameters were derived from a variety of sources, including site-specific data, NUREG/CR-5512, Volume 2 recommendations, literature data, RESRAD default values, and, in the absence of any site- or region-specific data, conservative assumptions. For ease of presentation, input parameters have been grouped into the categories of contamination, soil, water, ingestion (dietary/non-

dietary), and occupancy. These categories correspond to the RESRAD modules that allow input of these data. The assignment of parameters is discussed below.

- **Contamination:** Table 6-3 summarizes the bases for assigning the contamination group of parameters. Where applicable, the values recommended in NUREG/CR-5512, Volume 2, have been used. An important assumption is that the residual radioactivity is uniformly distributed over the source volume, taking no credit for any cover material. Other important assumptions were made regarding the area of the contaminated zone (10,000 m^2), shape of the contaminated zone area (circular), and the thickness of the contaminated zone (1 m). Historical site assessments and site characterization activities indicate that the actual contaminated zone areas and thicknesses are likely less than have been assumed. These assumptions result in conservative DCGLs for soil. The hydraulic properties of the soils comprising the contaminated zone were assigned using the RESRAD Data Collection Handbook (Reference 6-11) based on their silty-sand lithology (Reference 6-8).

- **Soil:** Table 6-4 summarizes how the soil group of parameters were assigned. All soil parameters were assigned using the RESRAD Data Collection Handbook based on the silty sand lithology of the industrial area soils. The average depth to the water table has been characterized to be about 3 meters (Reference 6-8). Considering that the contaminated zone occupies the upper 1 m of this interval, the unsaturated zone thickness was set at 2 m.

- **Water:** Table 6-5 summarizes the assignment of the water group parameters. Hydrologic modeling was completed external to RESRAD using the Hydrological Evaluation of Landfill Performance (HELP) Model (Reference 6-13) to determine site-specific values for the runoff coefficient (parameter #1) and the evapotranspiration coefficient (parameter #5). It has been assumed that the resident farmer will elect to locate and complete his well in the unconsolidated sediments rather than in the bedrock, as is common practice along the Connecticut River floodplain (Reference 6-14). The non-dispersion (versus mass balance) RESRAD approach to ground water modeling was chosen, based on the size of the contaminated zone area (> 1,000 m^2). A well intake depth of 10 m was assigned – the geologic sections shown in Figure 6-2 indicate that this is a reasonable depth for accessing the transmissive unconsolidated sediments underlying the river's floodplain. Hydraulic properties representative of the unconsolidated sediments were assigned to the saturated zone.

- **Ingestion:** Table 6-6 summarizes the bases for assigning the ingestion group of parameters. Where applicable, the default ingestion parameters from NUREG/CR-5512, Volume 2, have been used as recommended in Reference 6-12 for use with the RESRAD code. Note that all drinking water, livestock water, irrigation water, aquatic food, plant food, meat, and milk are assumed to be derived from onsite. This assumption is conservative in that the resident farmer will likely obtain a portion of his or her drinking water and food from offsite sources.

- **Occupancy:** Table 6-7 summarizes how the occupancy group of parameters were assigned. Where applicable, the default occupancy parameters from NUREG/CR-5512, Volume 2 have been used as recommended in Reference 6-12 for use with the RESRAD code. Note that the NUREG/CR-5512, Volume 2 values for fraction of time spent indoors onsite (65.71%) and fraction of time spent outdoors onsite (11.01%) are based on the largest and most current human activity pattern survey available. These values, which are the averages from this survey, are used to represent the average member of the critical group. Given the 180 day growing season for this region (Reference 6-12), spending 11.01% of the year outdoors onsite is equivalent to spending about 5 hours/day outdoors every day of the week during the growing season. If only the workweek is considered, this is equivalent to spending about 8 hours/day outdoors, five days a week during the growing season. These values are reasonable for the resident farmer scenario as defined in NUREG/CR-5512, Volume 1.

DCGLs for the radionuclides listed in Table 6-2 were calculated using RESRAD and the input parameters from Tables 6-3 through 6-7. The resulting Base Case DCGLs are summarized in Table 6-8 along with times to peak dose.

Significant Exposure Pathways

Significant exposure pathways were identified from the RESRAD output by tabulating the dose contribution from each pathway for each radionuclide. Table 6-8 summarizes these results, which have been broadly categorized into water independent pathways and water dependent pathways. The primary distinction between these two pathways is that the doses from the water dependent pathways are a function of the ground water contaminant levels calculated by RESRAD, while the doses from the water independent pathways do not depend on ground water conditions. The pathway making the greatest contribution to dose has been bolded for each radionuclide.

Review of the results included in Table 6-8 yields the following observations regarding the water independent pathways:

- The dose for 17 of the 20 radionuclides being considered is controlled by water independent pathways.
- Direct exposure to external radiation from the soil determines the dose for eight of the radionuclides, and the two most abundant radionuclides (i.e., Co-60 and Cs-137).
- Ingestion of plants grown in contaminated soil determines the dose for six of the radionuclides.
- Ingestion of meat and milk derived from animals fed contaminated fodder determines the dose for only three radionuclides (Fe-55, Ni-59, Ni-63).

Regarding the water dependent pathways, the results provided in Table 6-8 show the following:

- The dose for 3 of the 20 radionuclides is controlled by water dependent pathways.
- Ingestion of drinking water derived from a contaminated well determines the dose for two radionuclides (Pu-238, Pu-239).
- Fish ingestion is the primary pathway for C-14.

Therefore the significant exposure pathways are primarily water independent. These pathways determine the dose for 17 of the 20 radionuclides for which soil DCGLs are being developed, including Co-60 and Cs-137, the two most abundant radionuclides.

Table 6-3
RESRAD Contamination Input Parameters

Parameter	Description	Value	Bases
1. Area of contaminated zone	Area (m^2) that contains location of soil exceeding background radionuclide concentrations	10,000 m^2	Conservative assumption based on characterization data.
2. Thickness of contaminated zone	Distance (m) between the uppermost and lowermost soil samples that have radionuclide concentrations above background	1 m	Conservative assumption based on characterization data.
3. Length parallel to aquifer flow	Maximum horizontal distance (m) measured in the contaminated zone, from its upgradient edge to the downgradient edge, along the direction of the ground water flow in the underlying aquifer	112.84m	Diameter of circle with area of 10,000 m^2
4. Radiation dose limit	Effective dose equivalent from external radiation plus the effective dose equivalent from internal radiation	25 mrem/year	10 CFR 20.1402
5. Time since placement of material	Duration between the placement of radioactive materials onsite and the performance of radiological survey	0 year	RESRAD default value (Reference 6-11)
6. Times for calculations	Time(s) (years) at which DCGLs are calculated	1, 3, 10, 30, 100, 300, 1000 years	RESRAD default value (Reference 6-11) (does not affect calculation of DCGLs)
7. Initial concentration of principal radionuclides in soil	Initial concentration (pCi/g)of select radionuclides in contaminated zone	100 pCi/g	Assumed (does not affect calculation of DCGLs)
8. Initial concentration of radionuclides present in ground water	Initial concentrations (pCi/l) of radionuclide ground water contamination	NA	Not applicable to Base Case DCGLs
9. Radionuclide sorption coefficients	Ratio of mass of solute species adsorbed or precipitated on the solids per unit of dry mass of soil, to the solute concentration in the liquids	See Table 6-2	Table D.22 from NUREG/CR-5512, Volume 2

Table 6-3
RESRAD Contamination Input Parameters

Parameter	Description	Value	Bases
10. Cover depth	Distance (m) from the ground surface to the location of the uppermost soil sample with radionuclide concentrations above background concentrations	0 m	Assumed worst case (no cover material)
11. Density of cover material	Ratio of mass/volume (g/cm^3)	NA	Not applicable, no cover is assumed
12. Cover depth erosion rate	Average volume of soil material removed from one place to another by water, wind, or moving ice per unit of ground surface area per unit time	NA	Not applicable, no cover is assumed
13. Density of contaminated zone	Ratio of mass/volume (g/cm^3)	1.52 g/cm^3	Reference 6-11, Table 2.1 based on silty sand lithology (Reference 6-8)
14. Contaminated zone erosion rate	Average volume of soil material removed from one place to another by water, wind, or moving ice per unit of ground surface area per unit time	0.001 m/yr	RESRAD default value (Reference 6-11)
15. Contaminated zone total porosity	Ratio of the pore volume to the total volume of a representative sample of the medium	0.41	Reference 6-11, Table 3.2, based on silty sand lithology (Reference 6-8)
16. Contaminated zone effective porosity	Ratio if the part of the pore volume where the water can circulate to the total volume of a representative sample of the medium	0.33	Reference 6-11, Table 3.2, based on silty sand lithology (Reference 6-8)
17. Unsaturated zone field capacity	Ratio of volume of water retained in the soil sample, after gravity drainage, to total volume of sample	0.08	Equation 4.4 of Reference 6-11
18. Contaminated zone hydraulic conductivity	Measure of soils ability to transmit water (m/year) when submitted to a hydraulic gradient	4,930 m/yr	Reference 6-11, Table 5.2, based on silty sand lithology (Reference 6-8)
19. Contaminated zone b parameter	Empirical and dimensionless parameter used to evaluate the saturation ratio	4.38	Reference 6-11, Table 13.1, based on silty sand lithology (Reference 6-8)

Table 6-4
RESRAD Soil Input Parameters

Parameter	Description	Value	Bases
1. Number of unsaturated zone strata	Number of distinct unsaturated zones	1	Geologic site characteristics (Reference 6-8)
2. Unsaturated zone thickness	Portion of the uncontaminated zone that lies below the bottom of the contaminated zone and above the water table	2 m	Hydrogeologic site characteristics (Reference 6-8)
3. Unsaturated zone density	Ratio of mass/volume (g/cm^3)	1.52 g/cm^3	Reference 6-11, Table 2.1, based on silty sand lithology (Reference 6-8)
4. Unsaturated zone total porosity	Ratio of the pore volume to the total volume of a representative sample of the medium	0.41	Reference 6-11, Table 3.2, based on silty sand lithology (Reference 6-8)
5. Unsaturated zone effective porosity	Ratio if the part of the pore volume where the water can circulate to the total volume of a representative sample of the medium	0.33	Reference 6-11, Table 3.2, based on silty sand lithology (Reference 6-8)
6. Unsaturated zone field capacity	Ratio of volume of water retained in the soil sample, after gravity drainage, to total volume of sample	0.08	Equation 4.4 of Reference 6-11
7. Unsaturated zone soil-specific parameter	Empirical and dimensionless parameter used to evaluate the saturation ratio	4.38	Reference 6-11, Table 13.1, based on silty sand lithology (Reference 6-8)
8. Unsaturated zone hydraulic conductivity	Measure of soils ability to transmit water when submitted to a hydraulic gradient	4,930 m/yr	Reference 6-11,Table 5.2, based on silty sand lithology (Reference 6-8)

Table 6-5
RESRAD Water Input Parameters

Parameter	Description	Value	Bases
1. Runoff coefficient	Fraction of annual precipitation that does not infiltrate into the soil and is not transferred back to the atmosphere through evapotranspiration	0.11	HELP Model (Reference 6-13) and Equation 10.1 of Reference 6-11
2. Precipitation	Average volume of water in the form of rain, snow, hail or sleet that falls per unit area and per unit time at the site	1.1 m/yr	Site specific precipitation data (Reference 6-15)
3. Irrigation rate	Average volume of water added to soil at the site (m/year), per unit of surface area and per unit time	0.2 m/yr	RESRAD default value (Reference 6-11)
4. Irrigation mode	Method of irrigation (overhead or ditch)	Overhead	Appropriate for region
5. Evapotranspiration coefficient	Total volume of water that is transferred from the solid or liquid state to the gaseous state during a fixed period of time	0.50	HELP Model (Reference 6-13) and Equation 12.1 of Reference 6-11
6. Watershed area for nearby stream or pond	Region (m^2) contoured by an imaginary line connecting ridges of summits or high land and drained by or draining into a river, river system, or body of water	5.93×10^5 m^2	Conservatively taken to be watershed area for pond north of industrial area. Watershed areas for other potential receiving water bodies are larger.
7. Accuracy for water/soil computations	Number of significant digits in calculations	0.001	RESRAD default value (Reference 6-11)
8. Density of saturated zone	Ratio of mass/volume (g/cm^3)	1.52 g/cm^3	Reference 6-11,Table 2.1, based on silty sand lithology (Reference 6-8)
9. Saturated zone total porosity	Ratio of the pore volume to the total volume of a representative sample of the medium	0.41	Reference 6-11, Table 3.2, based on silty sand lithology (Reference 6-8)

Table 6-5
RESRAD Water Input Parameters

Parameter	Description	Value	Bases
10. Saturated zone effective porosity	Ratio if the part of the pore volume where the water can circulate to the total volume of a representative sample of the medium	0.33	Reference 6-11, Table 3.2, based on silty sand lithology (Reference 6-8)
11. Saturated zone hydraulic conductivity	Measure of soils ability to transmit water when submitted to a hydraulic gradient	4,930 m/yr	Reference 6-11, Table 5.2, based on silty sand lithology (Reference 6-8)
12. Saturated zone hydraulic gradient	Change of hydraulic head per unit of distance of the ground water flow in a given direction	0.017	Observed April 1999 (Reference 6-8)
13. Saturated zone b parameter	Empirical and dimensionless parameter used to evaluate the saturation ratio	4.38	Reference 6-11, Table 13.1, based on silty sand lithology (Reference 6-8)
14. Water table drop rate	Rate (m/year) at which the depth of the water table is lowered	0.001 m	RESRAD default value (Reference 6-11)
15. Wind Speed	Average velocity of wind	1.15 m/s	Site specific conditions (Reference 6-15)
16. Well pump intake depth (below water table)	Screened depth of a well within the aquifer	10 m	Anticipated depth of well completed in unconsolidated sediments comprising Connecticut River floodplain.
17. Model: non-dispersion or mass balance	Method by which ground water flow is considered	Non-dispersion	Applicable for contaminant areas greater than 1,000 m^2 (Reference 6-11)

Table 6-6
RESRAD Ingestion (Dietary/Non-Dietary) Input Parameters

Parameter	Description	Value	Bases
1. Fruits, vegetables, grain	Consumption rate (kg/yr) of fruits, vegetables, and grains	112 kg/yr	Reference 6-12
2. Leafy Vegetable	Consumption rate (kg/yr) for vegetables such as spinach and lettuce	21.4 kg/yr	Reference 6-12
3. Meat and Poultry	Consumption rate (kg/yr) for meat and poultry	65.1 kg/yr	Reference 6-12
4. Milk	Consumption rate (l/yr) for milk	233 l/yr	Reference 6-12
5. Fish	Consumption rate (kg/yr) for fish	20.6 kg/yr	Reference 6-12
6. Other Seafood	Consumption rate (kg/yr) for other types of seafood	0.9 kg/yr	RESRAD default value (Reference 6-11)
7. Drinking water	Average amount of water consumed (l/yr) by an adult	478.5 l/yr	Reference 6-12
8. Contaminated fraction of drinking water, livestock water, irrigation water, aquatic food, plant food, meat, and milk	Amount of dietary intake that is from contaminated sources	100%	Reference 6-12
9. Soil Ingestion	Accidental ingestion rate (g/yr) of soil material or soil dust	18.26 g/yr	Reference 6-12
10. Livestock fodder intake for meat	Amount of fodder consumed (kg/d) for beef cattle	27.1 kg/d	Reference 6-12
11. Livestock fodder intake for milk	Amount of fodder consumed (kg/d) for milk cows	63.25 kg/d	Reference 6-12
12. Livestock water intake for milk	Amount of water consumed (l/d) for milk cows	60 l/d	Reference 6-12
13. Livestock water intake for meat	Amount of water consumed (l/d) for beef cows	50 l/d	Reference 6-12
14. Livestock intake of soil	Daily intake of soil by livestock kept for meat or milk	0.5 kg/d	RESRAD default value (Reference 6-11)
15. Mass loading for foliar deposition	Concentration (g/m^3) of soil particles in air	$1x10^{-4}$ g/cm^3	RESRAD default value (Reference 6-11)

Table 6-6
RESRAD Ingestion (Dietary/Non-Dietary) Input Parameters

Parameter	Description	Value	Bases
16. Depth of soil mixing layer	Thickness at which soil mixing will occur	0.15 m	RESRAD default value (Reference 6-11)
17. Depth of roots	Average root depth of various plants grown in contaminated zone	0.9 m	RESRAD default value (Reference 6-11)
18. Storage time for fruits, non-leafy vegetables, and grain	Time item is stored prior to use	14 d	Reference 6-12
19. Storage time for leafy vegetables	Time item is stored prior to use	1 d	Reference 6-12
20. Storage time for milk	Time item is stored prior to use	1 d	Reference 6-12
21. Storage time for meat and poultry	Time item is stored prior to use	20 d	Reference 6-12
22. Storage time for livestock fodder	Time item is stored prior to use	0 d	Reference 6-12
23. Wet weight crop yield	Crop yield (kg) per area (m^2)	Non leafy: 0.7 kg/m^2 Leafy: 1.5 kg/m^2 Fodder: 1.1 kg/m^2	RESRAD default value (Reference 6-11)
24. Length of growing season	Minimum growing periods for non leafy vegetables, leafy vegetables, and fodder	Non leafy: 0.25 yr Leafy: 0.123 yr Fodder: 0.15 yr	Reference 6-12
25. Translocation factor	Fraction of activity deposited on plant surface that reaches edible portion of plant	Non leafy: 0.1 Leafy: 1 Fodder: 1	Table D.11 of NUREG/CR-5512, Volume 2
26. Weathering removal constant	Constant that accounts for removal of contaminants from plants	20 1/yr	RESRAD default value (Reference 6-11)
27. Dry foliar interception fraction	Fraction of Deposited contamination retained on various plants grown for food and animal feed as a result of weathering of the contaminated zone	Non leafy: 0.25 Leafy: 0.25 Fodder: 0.25	RESRAD default value (Reference 6-11)
28. Wet foliar interception fraction	Fraction of Deposited contamination retained on various plants grown for food and animal feed after above-ground irrigation with contaminated water	Non leafy: 0.35 Leafy: 0.35 Fodder: 0.35	Table D.10 of NUREG/CR-5512, Volume 2

Table 6-7
RESRAD Occupancy Input Parameters

Parameter	Description	Value	Bases
1. Inhalation rate	Volume of air inhaled by a member of critical group in one year – average value that accounts for multiple activity levels	1.169×10^4 m^3/yr	Reference 6-12
2. Mass loading for inhalation	Concentration of soil particles in air	3.14×10^{-6} g/m^3	Reference 6-12
3. Exposure duration	Time duration for critical group exposure	30 yr	RESRAD default value (Reference 6-11)
4. Shielding factor, inhalation	Ratio of indoor/outdoor onsite dust concentrations	0.4	RESRAD default value (Reference 6-11)
5. Shielding factor, external gamma	Ratio of indoor/outdoor onsite gamma radiation levels	0.5512	Reference 6-12
6. Fraction of time spent indoors onsite	Fraction of time individual spends inside a house or building at the site	0.6571	Reference 6-12
7. Fraction of time spent outdoors onsite	Fraction of time individual spends outdoors at the site	0.1101	Reference 6-12
8. Shape factor, external gamma	Used for to account for non-circular area	1	Circular contaminated zone conservatively assumed

Table 6-8
Base Case DCGLs for Soil

Radio-nuclide	DCGL (pCi/g)	Time to Peak Dose (years)	Dose Fraction From Water Independent Pathways (%)						Dose Fraction From Water Dependent Pathways (%)				
			Ground	Inhalation	Plant	Meat	Milk	Soil	Water	Fish	Plant	Meat	Milk
H-3	9.90E+02	0	-	2.3	**41.3**	5.2	30.7	-	12.1	0.1	2.8	0.8	4.8
C-14	4.25E+00	25.9	-	-	-	-	-	-	0.2	**99.1**	0.3	0.2	0.1
Mn-54	1.52E+01	0	**95.2**	-	4.6	-	0.2	-	-	-	-	-	-
Fe-55	5.35E+04	0	-	-	15.3	**78.6**	4.5	1.6	-	-	-	-	-
Co-60	3.49E+00	0	**94.1**	-	3.8	1.2	0.9	-	-	-	-	-	-
Ni-59	4.91E+03	0	-	-	27.4	2.5	**70.1**	0.1	-	-	-	-	-
Ni-63	1.79E+03	0	-	-	27.4	2.5	**70.1**	0.1	-	-	-	-	-
Sr-90	3.10E+00	0	0.1	-	**74.5**	8.4	16.9	-	-	-	-	-	-
Nb-94	7.10E+00	0	**99.8**	-	0.2	-	-	-	-	-	-	-	-
Tc-99	2.37E+01	0	-	-	**89.9**	0.1	9.9	-	-	-	-	-	-
Cs-134	5.74E+00	0	**80.2**	-	7.5	4.4	7.9	-	-	-	-	-	-
Cs-137	1.16E+01	0	**68.1**	-	12.1	7.0	12.8	-	-	-	-	-	-
Eu-152	8.24E+00	0	**99.9**	-	0.1	-	-	-	-	-	-	-	-
Eu-154	7.61E+00	0	**99.9**	-	0.1	-	-	-	-	-	-	-	-
Eu-155	3.21E+02	0	**99.2**	-	0.6	0.1	-	-	-	-	-	-	-
Pu-238	4.86E+00	81.5	-	-	0.4	-	-	-	**65.1**	20.5	13.8	0.1	-
Pu-239	2.30E+00	81.7	-	-	0.4	-	-	-	**65.1**	20.5	13.8	0.1	-
Pu-241	1.60E+03	73.6	2.5	0.3	**60.7**	0.8	0.1	6.4	19.1	6.1	4.1	-	-
Am-241	4.41E+01	0	3.5	0.4	**85.8**	1.1	0.2	9.0	-	-	-	-	-
Cm-243	3.97E+01	0	41.2	0.2	**52.7**	0.3	0.1	5.5	-	-	-	-	-

Bold represents the dominate dose pathway. "-" indicates a value of 0.0

Parameters Affecting Dose

To help identify the parameters associated with the significant exposure pathways, a screening level sensitivity analysis was performed using RESRAD's sensitivity graphic utility on input parameters expected to be relevant to the significant exposure pathways. This analysis was completed for selected radionuclides based primarily on their relative abundance. Some of the less abundant radionuclides were also selected to ensure that all significant exposure pathways were represented. The radionuclides selected include C-14, Fe-55, Co-60, Ni-63, Sr-90, Tc-99, Eu-152, Cs-137, Pu-238, and Am-241.

The RESRAD sensitivity utility operates by both reducing and increasing the selected input parameter by a common factor. The dose is then calculated for each perturbed parameter value. The output, including dose with the parameter unperturbed, dose with parameter reduced, and dose with parameter increased, is graphically displayed with time as the independent variable. A qualitative analysis (i.e., by inspection) was performed on the graphical output to determine the relative sensitivity of the dose (and the DCGL) and to identify the relevant input parameters.

Results are summarized in Table 6-9. The first column identifies the parameter evaluated. A check mark in the remaining columns indicates if a reduction or increase of that parameter had any discernable effect on the calculated dose for a given radionuclide (i.e., the dose calculated when a parameter is perturbed differs from that when unperturbed). The parameters identified as affecting the dose were then retained for further, quantitative sensitivity analysis.

Table 6-9
Parameters Impacting Dose

Parameter	Sensitivity									
	C-14	Fe-55	Co-60	Ni-63	Sr-90	Tc-99	Cs-137	Eu-152	Pu-238	Am-241
Contamination										
Contaminated zone thickness	✓	✓	✓	✓	✓	✓	✓		✓	✓
Contaminated zone area	✓		✓				✓	✓	✓	
Contaminated zone density	✓			✓	✓	✓	✓		✓	
Contaminated zone field capacity										
Contaminated zone hydraulic conductivity										
Soil										
Unsaturated zone thickness	✓								✓	
Unsaturated zone hydraulic conductivity										
Unsaturated zone total porosity	✓								✓	
Water										
Runoff coefficient	✓				✓	✓	✓		✓	
Precipitation rate	✓			✓	✓	✓	✓		✓	
Irrigation rate	✓				✓	✓	✓		✓	
Evapotranspiration coefficient	✓			✓	✓	✓	✓		✓	
Watershed area	✓								✓	
Saturated zone density	✓								✓	
Saturated zone total porosity	✓								✓	
Saturated zone effective porosity	✓								✓	
Saturated zone hydraulic conductivity	✓								✓	
Saturated zone hydraulic gradient	✓								✓	
Well pump intake depth									✓	
Well pumping rate										
Ingestion										
Fruit, vegetables, grain consumption		✓	✓	✓	✓	✓	✓		✓	✓

Table 6-9
Parameters Impacting Dose

Parameter	Sensitivity									
	C-14	Fe-55	Co-60	Ni-63	Sr-90	Tc-99	Cs-137	Eu-152	Pu-238	Am-241
Leafy vegetable consumption		✓		✓	✓	✓	✓		✓	✓
Meat and poultry consumption		✓	✓	✓	✓		✓			✓
Milk consumption		✓	✓	✓	✓	✓	✓			
Drinking water consumption									✓	
Soil ingestion										✓
Livestock fodder intake for meat		✓	✓	✓	✓		✓			
Livestock fodder intake for milk			✓	✓	✓	✓	✓			
Livestock soil ingestion		✓		✓	✓		✓			✓
Mass loading for foliar deposition										
Dry foliar interception fraction										
Wet foliar interception fraction									✓	
Occupancy										
Inhalation rate										
Mass loading for inhalation										
External gamma shielding factor			✓				✓	✓		✓
Fraction of time spent indoors onsite			✓				✓	✓		✓
Fraction of time spend outdoors onsite			✓				✓	✓		✓

Sensitivity Analysis

A quantitative sensitivity analysis was performed on the parameters associated with the significant exposure pathways to determine which ones have the greatest influence on the calculated DCGL. The parameters retained for this analysis are identified in Table 6-9. A sensitivity analysis involves the process in which one input parameter is changed, or perturbed, and the resultant change in the DCGL, if any, is compared against the DCGL calculated using the unperturbed parameter. The relative impact of perturbing the input parameter is then related to potential variability of the parameter and associated uncertainty in the calculated DCGL.

The methodology used to complete this sensitivity is described in Reference 6-16. Calculation of sensitivities requires that a performance measure be defined. In the current application, the DCGL calculated by the RESRAD model is used as this performance measure. This is functionally represented as

$$P = f(\{\alpha\}, t) \qquad \text{(Equation 6-1)}$$

where: P = system performance measure (DCGL); $\{\alpha\}$ = vector of system parameters (e.g., distribution coefficient, precipitation, etc.); and t = time. The system sensitivity may be developed in terms of normalized sensitivity coefficients S_k defined for parameter α_k as

$$S_k = \frac{dP_k / P}{d\alpha_k / \alpha_k} \qquad \text{(Equation 6-2)}$$

where $dP_k/d\alpha_k$ is the marginal sensitivity of P to α_k. S_k describes the percentage change of performance measure P to a 1 change of parameter α_k. The derivative that defines the marginal sensitivity can calculate to second order accuracy using a central difference scheme, allowing S_k to be represented as

$$S_k = \frac{(P_{+1} - P_{-1}) / P_0}{2\Delta\alpha_k / \alpha_{k0}} \qquad \text{(Equation 6-3)}$$

where the variables in the above equation are defined in Figure 6-4. The calculation of each individual S_k is then obtained by performing two RESRAD model runs: one with $\alpha_k = \alpha_{k+1}$ to determine P_{+1}, and one with $\alpha_k = \alpha_{k-1}$ to determine P_{-1}. All other input parameters are held constant except the parameter of interest during the sensitivity analysis.

The results are summarized in Table 6-10. A positive value of S_k indicates the DCGL is directly proportional to the parameter of interest (i.e., curve on Figure 6-4 has a positive slope), whereas a negative value indicates the DCGL is inversely proportional to the parameter of interest (i.e., curve on Figure 6-4 has a negative slope). A value of $S_k = 0$ indicates the DCGL is independent of the parameter (i.e., curve on Figure 6-4 is a horizontal line). To facilitate the identification of the most sensitive parameters, values in Table 6-10 have been bolded where $|S_k| > 0.5$.

Table 6-10
Normalized Sensitivity Coefficients, S_k, for Soil DCGLs

Parameter	C-14	Fe-55	Co-60	Ni-63	Sr-90	Tc-99	Cs-137	Eu-152	Pu-238	Am-241
Contamination										
Contaminated zone thickness	-0.31	-0.04	-0.01	-0.19	-0.19	-0.23	-0.09	-0.00	-0.32	-0.17
Contaminated zone area	**-1.06**	0.00	-0.02	0.00	0.00	0.00	-0.02	-0.02	-0.21	0.00
Contaminated zone density	-0.38	0.00	0.00	-0.01	-0.01	-0.02	-0.01	0.00	-0.35	0.00
Distribution coefficient	**0.95**	0.00	0.00	0.45	-0.01	**-0.60**	0.49	NA	**1.68**	0.00
Soil										
Unsaturated zone thickness	0.00	-	-	-	-	-	-	-	0.46	-
Unsaturated zone total porosity	0.00	-	-	-	-	-	-	-	-0.46	-
Water										
Runoff coefficient	-0.03	0.00	0.00	-0.01	0.00	0.00	-0.01	0.00	0.10	0.00
Precipitation	0.25	0.00	0.00	0.03	0.00	0.02	0.05	0.00	**-0.83**	0.00
Irrigation	0.04	0.00	0.00	0.01	0.00	0.00	0.01	0.00	-0.31	0.00
Evapotranspiration coefficient	-0.29	0.00	0.00	-0.03	0.00	-0.02	-0.04	0.00	**1.00**	0.00
Watershed area	**0.99**	-	-	-	-	-	-	-	0.21	-
Saturated zone density	0.26	-	-	-	-	-	-	-	0.48	-
Saturated zone effective porosity	0.27	-	-	-	-	-	-	-	-0.49	-
Saturated zone total porosity	-0.27	-	-	-	-	-	-	-	**-0.50**	
Saturated zone hyd. conductivity	-0.34	-	-	-	-	-	-	-	0.29	-
Saturated zone hydraulic gradient	-0.28	-	-	-	-	-	-	-	0.31	-
Well pump intake depth	0.01	-	-	-	-	-	-	-	**0.80**	-
Ingestion										
Fruit, vegetable, grain consumption	0.00	-0.13	-0.03	-0.23	**-0.68**	**-0.85**	-0.11	0.00	-0.08	**-0.80**
Leafy vegetable consumption	0.00	-0.03	-0.01	-0.04	-0.12	-0.14	-0.02	0.00	-0.06	-0.14

Table 6-10
Normalized Sensitivity Coefficients, S_k, for Soil DCGLs

Parameter	C-14	Fe-55	Co-60	Ni-63	Sr-90	Tc-99	Cs-137	Eu-152	Pu-238	Am-241
Meat and poultry consumption	0.00	**-1.24**	-0.01	-0.03	-0.08	0.00	-0.07	0.00	0.00	-0.01
Milk consumption	0.00	-0.05	-0.01	**-0.77**	-0.17	-0.10	-0.13	0.00	0.00	0.00
Fish consumption	**-0.99**	-	-	-	-	-	-	-	-0.18	-
Drinking water consumption	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	**-0.71**	0.00
Soil ingestion	0.00	-0.02	0.00	-0.01	0.00	0.00	0.00	0.00	0.00	-0.09
Livestock fodder intake for meat	0.00	-0.04	-0.01	-0.02	-0.08	-0.01	-0.06	0.00	0.00	0.00
Livestock fodder intake for milk	0.00	0.00	-0.01	**-0.72**	-0.17	-0.10	-0.11	0.00	0.00	0.00
Livestock soil ingestion	0.00	**-0.93**	0.00	-0.10	-0.01	0.00	-0.04	0.00	0.00	-0.01
Occupancy										
External gamma shielding	0.00	0.00	**-0.85**	0.00	0.00	0.00	**-0.57**	**-0.92**	0.00	-0.03
Fraction time spent indoors onsite	0.00	-0.01	**-0.81**	-0.01	0.00	0.00	**-0.56**	**-0.88**	0.00	-0.11
Fraction time spent outdoors onsite	0.00	0.00	-0.23	0.00	0.00	0.00	-0.16	-0.25	0.00	-0.02

A positive value indicates direct proportionality, whereas a negative value indicates an inverse proportionality. Bolded values indicate high sensitivity defined as $|S_k| > 0.5$. In some instances, sensitivity coefficients were not calculated when the qualitative analysis indicated no sensitivity. "-" indicates a value of zero.

Parameter Justification

The results presented in Table 6-10 show that Base Case DCGLs exhibit sensitivity to contamination, water, ingestion and occupancy parameters. That is, any variability or uncertainty in assigning this set of parameter values will be propagated to the calculated DCGLs. Therefore, the values assigned for the parameters exhibiting high sensitivity, defined as $|S_k| \geq 0.50$ in this case, require justification.

Contamination parameters requiring justification include contaminated zone area and distribution coefficient. Justifications for the values assigned to each of these parameters is provided below.

- **Contaminated zone area:** Site characterization data indicates that the contaminated zone area is less than the 10,000 m^2 area assumed. The use of the 10,000 m^2 will therefore result in a conservative DCGL calculation. Note that C-14 is the only radionuclide that is sensitive to this parameter.

- **Distribution coefficient:** Several of the radionuclides are sensitive to this parameter. Distribution coefficients were assigned based on recommendations given in NUREG/CR-5512, Volume 2, and are consistent with the default values implemented in NRC's DandD code. The NUREG/CR-5512, Volume 2, recommended values were chosen such that their use would lead to conservative estimates of the dose or the DCGL, based on literature data.

The calculated DCGLs for water-mobile radionuclides (i.e., those with relatively small distribution coefficients) are sensitive to several of the water parameters. Justifications for assigning these parameters are provided below.

- **Precipitation:** The average annual precipitation depth was obtained from the HNP UFSAR (Reference 6-15). This value is based on the long-term precipitation record at a regional airport. The value is therefore justified.

- **Evapotranspiration coefficient:** This parameter was derived using the HELP model (Reference 6-13). The HELP model uses soil and weather data and uses solution techniques that account for effects of surface storage, snowmelt, runoff, infiltration, evapotranspiration, vegetative growth, soil moisture storage, and unsaturated vertical drainage. Data inputs to this model were developed from site- or region-specific sources. The soil water balance was then simulated for a 30 year period on a daily basis. The evapotranspiration coefficient was then determined from the evapotranspiration depth calculated for simulation period. Because this analysis is based on site- and region-specific data and further considers the likely variability in weather conditions, the derived values are considered justified.

- **Watershed area:** The watershed area is used by RESRAD to calculate radionuclide concentrations in surface water bodies that are sustained by potentially contaminated ground water. As documented by Reference 6-9, the radionuclide concentrations in surface water (and fish) increases as the ratio of contaminated zone area to watershed area increases. Therefore, conservative DCGLs result when the watershed area of the smallest surface water body capable of sustaining a fish population is used. In this case, the watershed area of the pond north of the industrial area was selected. As the watershed areas of all other surface water bodies capable of sustaining fish are greater (i.e., the Connecticut River and the discharge canal), the use of the pond watershed area is bounding and conservative in this DCGL calculation.

- **Saturated zone total porosity:** The value assigned to this parameters is based on the silty-sand nature of the industrial area sediments, (Reference 6-8). Literature data show that this parameter

exhibits little variation for unconsolidated sediments. The assigned value is therefore considered justified since it is based on site-specific lithology.

- **Well pump intake depth:** The well pump intake depth of 10 m below the water table assumes that the well is completed in the unconsolidated sediments on the downgradient edge of the contaminated zone and adjacent to the Connecticut River. This location results in a conservative calculation of DCGLs as all the contaminated zone area would be contributing radionuclides to the ground water (Reference 6-9). Also, installation of wells in the alluvial deposits adjacent to the Connecticut River (as opposed to bedrock) is the practice in the area, as is evident from the existing water-supply wells that have been installed at the HNP site. Therefore, the assumed well location and well pump intake depth appears reasonable.

The sensitivity analysis indicates that DCGLs are sensitive to a number of ingestion and occupancy parameters. These include the following:

- Fruit, vegetable and grain consumption
- Meat and poultry consumption
- Milk consumption
- Fish consumption
- Drinking water consumption
- Livestock fodder intake for milk
- Livestock soil ingestion
- External gamma shielding
- Fraction of time spent indoors onsite

Values were assigned to these parameters using recommendations given in NUREG/CR-5512, Volume 2 and restated in Reference 6-12. These values also represent the defaults used in NRC's DandD code. According to NUREG/CR-5512, Volume 2, these default values have been defined through a systematic process of assessing the variability of each parameter across the U. S. and then defining default values that are unlikely to be exceeded at any real site (the basis for assigning default values will be published in NUREG/CR-5512, Volume 3). Use of the default parameter values will estimate the upper range of the dose that the average member of the critical group could receive at any site given the contaminant level at that site.

6.4.5 Applicability of the Base Case DCGLs to Other Impacted Areas

The Base Case DCGLs presented in Table 6-8 were developed based on the characteristics of the industrial area. As discussed in Section 2, there are other areas where soils may have been impacted by licensed operations. These other impacted areas fall into two distinct geographic regions: (1) Connecticut River floodplain areas primarily southeast of the industrial area; and (2) hillslope areas north and east of the industrial area. These impacted areas have some characteristics that differ from the industrial area. The applicability of the parameters used to develop the Base Case DCGLs to the other impacted areas is

discussed below. The discussion is limited to those parameters to which the DCGLs exhibit high sensitivity ($|S_k| \geq 0.5$).

- **Contamination:** Contamination parameters adopted for the industrial area are expected to be representative or bounding for other impacted areas. In particular, the contaminated zone area for these other impacted areas is expected to be less than for the industrial area, since historical releases have generally been confined to the industrial area. The distribution coefficients, which are based on NUREG/CR-5512 recommendations, are considered to be generically applicable; they should also be applicable to all impacted areas within the HNP site.

- **Water:** Table 6-10 shows that the DCGLs for water-mobile radionuclides are sensitive to a variety of water parameters. Due to the physical similarities between the industrial area and other impacted areas along the Connecticut River, the water parameters adopted for the industrial area should also be applicable to these other areas. For impacted areas on the hillslope, the water parameters adopted for the industrial area are expected to be representative or bounding. The steep terrain and shallow bedrock characteristic of hillslope areas will, for example, result in lower infiltration and radionuclide leaching rates from hillslope soils.

- **Ingestion and Occupancy:** Parameter values recommended in NUREG/CR-5512, Volume 2, and adopted in the calculation of the Base Case DCGLs are considered to be generically applicable. Because they are generically applicable to a variety of sites, they should also be applicable to all impacted areas within the HNP site.

It should be noted that farming of the hillslope area may not be practical, given the steep terrain and shallow bedrock. Pathways that consider ingestion of foods grown onsite, accounted for in the Base Case DCGLs, may not be applicable impacted areas on the hillslope. The application of Base Case DCGLs to these areas is therefore conservative.

The above demonstrates that the Base Case DCGLs developed for the industrial area bound those for other impacted areas and may therefore be conservatively applied to all impacted areas of the site, except as noted in the following section.

6.4.6 Special Case DCGLs

Radioactivity is present in the ground water underlying portions of the HNP site at levels above background, as discussed in Section 2. The affected areas are generally confined to the industrial area of the site. The radionuclides present that can be attributed to licensed operations include H-3 and Cs-137. The sources of radioactivity contributing to this existing ground water contamination have been eliminated.

If a well is installed in an affected portion of the aquifer, the well water may contain residual H-3 or Cs-137 associated with the current contamination. The levels of radioactivity that may be encountered depend on when the well is installed relative to release of the site. With the pre-decommissioning sources eliminated and with no other sources in place, concentrations of H-3 and Cs-137 in the ground water will decrease with time due to advection, dispersion, and radioactive decay.

The radioactivity currently in the ground water will contribute to the dose received by the critical group via the water dependent pathways defined for the resident farmer scenario. The presence of this radioactivity must therefore be accounted in the DCGLs for the soils overlying the contaminated regions of the aquifer. These soil DCGLs, which consider the dose component from existing ground water

contamination, are survey area-specific and have been termed "Special Case DCGLs." If ground water residual radioactivity is present, then the Special Case DCGLs must be less than the Base Case DCGLs to meet the overall 25 mrem/year TEDE limit.

The development of Special Case DCGLs includes identifying the affected survey areas and dose modeling. The methodology adopted to develop the Special Case DCGLs and the results of the dose modeling are described in the following sections.

6.4.6.1 Affected Survey Areas

Currently, ground water underlying portions of the industrial area contains H-3 and Cs-137. The nature and extent of this ground water radioactivity are discussed in Section 2.3.3.1.5. The survey areas affected by the presence of H-3 and Cs-137 in ground water are shown on Figure 6-5. Survey areas where all or a portion of the underlying ground water contains H-3 or Cs-137 were defined as being affected. To account for the possibility that the resident farmer may install a well in a contaminated portion of the aquifer but reside and farm on adjacent land, survey areas within a 100 m distance of the existing contaminated plume were defined as being affected, as well. This is consistent with the 10,000 m^2 land use assumption in the resident farmer scenario. Table 6-11 summarizes the affected survey areas.

Table 6-11
Survey Areas Affected by Existing Radioactivity in Ground Water

Survey Area
9102
9104
9302
9304
9306
9307
9310
9512
9518
9520
9118
9308
9312
9313
9502
9514
9522

6.4.6.2 Dose Modeling Approach

Dose modeling was conducted to calculate the Special Case DCGLs. The steps in the involved in calculation approach are as follows:

1. Determine the ground water (water dependent) dose contribution from H-3 and Cs-137, based on the ground water concentrations currently observed.
2. Adjust the dose limits for the affected soil considering the dose contribution from radioactivity currently in the ground water.
3. Calculate the Special Case DCGLs for soil in the affected areas by scaling the Base Case DCGLs.

Each step is described in more detail below.

Step 1. Ground Water Dose Contribution

The water dependent dose contributions for H-3 and Cs-137 were calculated using the RESRAD code. Assumptions significant to this analysis include the following:

- The conceptual model developed for the industrial area and previously presented in Section 6.4.2 is applicable to all affected survey areas. Since the affected areas are primarily in the industrial area, this assumption is valid.
- The ingestion and occupancy input parameters recommended in NUREG/CR-5512, Volume 2, and summarized in Table 6-6 and 6-7 are applicable to this calculation.
- The maximum H-3 concentration observed in ground water in September 1999 (27,230 pCi/l) is applicable. This assumption will result in a conservative (high) water dependent pathway dose because the average H-3 concentration within these areas is lower and the use of the maximum represents the worst case.
- The maximum Cs-137 concentration observed in ground water in September 1999 (28.7 pCi/l) is applicable. Cs-137 has not been observed at any other ground water monitoring locations. It has been conservatively assumed that Cs-137 is present over the same areal extent as H-3.
- The water dependent pathways utilized include the drinking water, plant, meat and milk ingestion pathways. Based on the plume characteristics and ground water flow patterns, radionuclides in the affected portions of the aquifer eventually discharge to the Connecticut River. Because the ratio of the contaminated area to the Connecticut River watershed area is very small, potential radionuclide concentrations in the surface water and fish would be correspondingly small.

Doses from H-3 and Cs-137 are calculated by using the ground water concentrations identified above directly in the RESRAD model. This is accomplished using the "Well Concentration Option" in the "Concentration Report" module in RESRAD. All water independent pathways are turned off for this calculation. With well water concentrations prescribed, the only RESRAD features being exercised are those that convert exposure concentrations in well water to dose for the applicable pathways.

Adjustment of some parameters was required to ensure that the values in the "Well Water Concentration Report" equal or exceed the H-3 and Cs-137 ground water concentrations used in the model. The following parameters were adjusted as described below:

- Contaminated zone thickness = 0.05 m
- Unsaturated zone thickness = 0.01 m
- Time since material placement = 1 year

Additional parameters require adjustment to ensure that the water independent dose contributions from the plant, meat, and milk are zero. The following adjustments were made:

- Cover depth = 0.5 m
- Depth of roots = 0.009 m

Table 6-12 summarizes the dose contributions from each of the applicable water dependent pathways and the associated totals for H-3 and Cs-137. These results show that H-3 and Cs-137 would contribute about 1 and 2 mrem/year, respectively, if the ground water were used for drinking, irrigation, and watering of livestock.

Table 6-12
Ground Water Contributions to Dose From H-3 and Cs-137

Radio-nuclide	Observed Concentration (pCi/l)	Doses From Water Dependent Pathways (mrem/year)				
		Water	Plant	Meat	Milk	Total
H-3	28,140	0.836	0.017	0.049	0.206	1.11
Cs-137	30	0.726	0.154	0.391	0.711	1.98

Step 2. Dose Limits for Affected Soil

The dose limit for the affected soils ($Dose_{soil}$) is obtained by subtracting the current ground water contribution ($Dose_{gw}$) from the 25 mrem/yr TEDE limit, i.e.,

$$Dose_{soil} = 25 - Dose_{gw} \qquad \text{(Equation 6-4)}$$

Note that $Dose_{gw}$ considers contributions from both radionuclides.

Step 3. Determination of Special Case DCGLs

Special Case DCGLs ($DCGL_{SC}$) are calculated for individual radionuclides by scaling the radionuclide's Base Case DCGL ($DCGL_{BC}$) to account for that portion of the 25 mrem/yr TEDE limit "consumed" by the existing radioactivity in the ground water. This calculation is made in accordance with the following equation:

$$DCGL_{SC} = \frac{Dose_{soil}}{25} DCGL_{BC} \qquad \text{(Equation 6-5)}$$

This approach is based on the premise that the dose contribution from residual radioactivity in soils (all pathways) may be superimposed on the dose contribution from existing radioactivity in ground water (water dependent pathways only).

6.4.6.3 Dose Modeling Results

Table 6-13 summarizes results of the dose modeling calculations used to determine the Special Case DCGLs that are applicable to soils in the areas where ground water has been impacted by

operations-related radioactivity. Special case DCGLs are listed for individual radionuclides. The survey areas to which these DCGLs apply are also identified.

Note that these doses are presented in Table 6-12 and DCGLs given in Table 6-13 area based upon September 1999 ground water conditions. As discussed earlier, the sources of these radionuclides have been eliminated and ground water concentrations are expected to decrease with time. Sampling of the ground water monitoring well system has verified this decreasing trend. Ground water sampling will continue as required to support the Final Status Survey. If ground water sampling at the time of Final Status Survey indicates conditions significantly different from those identified to date, ground water contributions to dose and soil DCGLs will be re-evaluated on a case by case basis.

Table 6-13
DCGLs for Soil Adjusted by Current Ground Water Concentration

Radio-nuclide	Base Case DCGLs (pCi/g)	Special Case DCGLs[1] (pCi/g)
H-3	9.90E+02	8.68E+02
C-14	4.25E+00	3.73E+00
Mn-54	1.52E+01	1.33E+01
Fe-55	5.35E+04	4.69E+04
Co-60	3.49E+00	3.06E+00
Ni-59	4.91E+03	4.31E+03
Ni-63	1.79E+03	1.57E+03
Sr-90	3.10E+00	2.72E+00
Nb-94	7.10E+00	6.22E+00
Tc-99	2.37E+01	2.07E+01
Cs-134	5.74E+00	5.03E+00
Cs-137	1.16E+01	1.02E+01
Eu-152	8.24E+00	7.23E+00
Eu-154	7.61E+00	6.67E+00
Eu-155	3.21E+02	2.81E+02
Pu-238	4.86E+00	4.26E+00
Pu-239	2.30E+00	2.02E+00
Pu-241	1.66E+03	1.46E+03
Am-241	4.41E+01	3.87E+01
Cm-243	3.97E+01	3.48E+01

[1] Special case DCGLs apply to survey areas 9102, 9104, 9302, 9304, 9306, 9310, 9512, 9518, 9520, 9118, 9307, 9308, 9312, 9313, 9502, 9514, and 9522 based on September 1999 ground water conditions (i.e., $Dose_{gw}$ = 3.09 mrem/year).

6.5 Development of DCGLs for Building Surfaces

The DCGLs for unrestricted release of buildings were developed from the building occupancy scenario using the RESRAD-BUILD computer code. This scenario has been selected as a bounding condition for dose to the public after release. The concrete building debris has been further assessed using the RESRAD farmer scenario and RESRAD-BUILD for an excavator scenario (see Figure 6-6). This ensures that the resultant dose to the public does not exceed 25 mrem/yr TEDE if the concrete is used as backfill.

DCGLs have been adjusted where ground water in contact with the concrete debris impacts the allowable level of residual activity. This section describes the process used to develop DCGLs for building surfaces, including above-grade and basement foundations, walls, floors, and ceilings.

The building occupancy scenario (as depicted on Figure 6-7) is described in detail in NUREG/CR-5512, Volume 1. The critical group for the this scenario, as defined in NUREG-1549, consists of adults who work in light industry. They are assumed to occupy a commercial facility in a normal manner during a typical 40 hour work week. Specific values of other building occupancy model parameters are discussed in Section 6.5.1.2.

In addition to the assessment of the building occupancy scenario, two other scenarios were chosen as checks on the dose from other future uses of the site:

- The Residential Farmer Scenario and
- The Excavator Scenario.

6.5.1 Base Case DCGL Development—Building Occupancy Scenario

The building occupancy scenario was chosen as the primary dose model because the basic premise of the model for buildings decontaminated to an unrestricted release is that this model will yield DCGLs that meet all the requirements for 10CFR20.1402. That is, any building meeting these requirements can be safely released for occupancy by the general public without further need for an NRC license. For HNP, this is a bounding assumption, since all concrete structures on the site will be demolished as described previously and the dose received by an individual will be no more than it would be if that person occupied the building.

The proposed building occupancy scenario assumes that the critical group consists of office workers, rather than industrial workers, who might work in a room of the building following license termination. The exposure duration is assumed to be a typical full work year (2080 hours).

Applicable pathways for the building occupancy scenario include:

1. External exposure to penetrating radiation from surface sources;
2. Inhalation of resuspended surface contamination; and
3. Inadvertent ingestion of surface contamination

As outlined in the flow diagram shown in Figure 6-6, the building occupancy scenario will be used to develop a DCGL for each radionuclide using the RESRAD-BUILD code. A description of the RESRAD-BUILD code and the model parameters is given in Sections 6.5.1.1 and 6.5.1.2.

The dose point location for the model is taken at the center of the floor of the room and 1 meter off the floor. Figures 6-7 through 6-9 illustrate the location and the manner in which RESRAD-BUILD models the geometry of the room for building occupancy scenario. This establishes a maximum dose point location for the modeled geometry, i.e., if an individual moves toward a specific wall, the contribution from others is reduced accordingly.

6.5.1.1 Code Selection (RESRAD-BUILD)

The RESRAD-BUILD code (Version 2.37) was chosen to assess the various exposure pathways for the building occupancy scenario. These pathways include ingestion and inhalation as well as direct exposure from contaminated surfaces.

RESRAD-BUILD is a computer code developed at Argonne National Laboratory for the U.S. Department of Energy to evaluate the potential radiological dose incurred by an individual who works or lives in a building contaminated with radioactive material. The radioactive material in the building structure can be released into the indoor air by mechanisms such as mechanical removal (decontamination activities) or erosion (removable surface contamination). The transport of radioactive material inside the building from one compartment to another is calculated with an indoor air quality model. The air quality model evaluates the transport of radioactive dust particles due to:

- air exchange between compartments and with outdoor air
- deposition and resuspension of particulates
- radioactive decay and ingrowth.

The RESRAD-BUILD radionuclide library consists of 67 radionuclides having half-lives of six months or greater; these are referred to as principal radionuclides. It is assumed that the short-lived decay products with half-lives less than six months, referred to as associated radionuclides, are in secular equilibrium with their parent (principal) radionuclides

RESRAD-BUILD can model multiple exposure pathways. The specific pathways used for the HNP building occupancy scenario are:

- external exposure directly from the source,
- external exposure to materials deposited on the floor,
- external exposure due to air submersion,
- inhalation of airborne radioactive particulates, and
- inadvertent ingestion of radioactive material.

6.5.1.2 Assignment of Code Input Parameters

The RESRAD-BUILD model for the building occupancy scenario uses the model parameters shown in Table 6-14. The justification of the parameters follows Table 6-14. A sensitivity study on these parameters was also performed and was used to arrive at the final parameter selection. The results of that sensitivity study are shown in Table 6-16.

Table 6-14
Parameters for RESRAD-BUILD Building Occupancy Scenario

Parameter	Description	Value	Bases
Time Parameters			
1. Exposure Duration	Amount of time that exposure occurs	365 days	Annual doses calculated per 10 CFR 20.1402
2. Indoor Fraction*	Fraction of the exposure duration that is spent inside the building	0.25 (40 hr/wk, 52 wk/yr)	Based on typical work schedule; consistent with NUREG/CR-5512, Volume 1, Table 6.21
3. Evaluation Time	Times at which doses are calculated	0 year; 1 year (other times if necessary to find peak dose; Pu-241 peak dose occurs at t=100 years)	Doses are calculated with the least credit for decay; consistent with NUREG/CR-5512, Volume 1, Section 3.2.1
Building Parameters			
4. Number of Rooms	Number of separate compartments in model of building	1	Model of building occupancy consistent with NUREG/CR-5512, Volume 1, Section 6.2.1
5. Deposition Velocity*	Velocity at which airborne particles are deposited onto surfaces	0.01 m/sec	RESRAD-BUILD Code; See Section 6.5.1.2 for justification.
6. Resuspension Rate*	Rate at which deposited material is resuspended into the air	5.0E-7 sec^{-1}	RESRAD-BUILD Code; See Section 6.5.1.2 for justification.
7. Building Exchange Rate*	Total volume of air going out of the building per unit time divided by the total volume of the building	0.8 hr^{-1}	Consistent with value of 0.75 hr^{-1} cited by American Society of Heating, Refrigerating, and Air-Conditioning Engineers, Inc. for conditioned spaces; See Section 6.5.1.2 for justification.
8. Room Area*	Floor area of the room	100 m^2	Size of typical room in plant; based on optimal geometry demonstrated in NUREG/CR-5512, Volume 1, Section 6.2.1; See Section 6.5.1.2 for justification.

Table 6-14
Parameters for RESRAD-BUILD Building Occupancy Scenario

Parameter	Description	Value	Bases
9. Room Height	Height of the room	2.5m	Height of typical room in plant; based on optimal geometry demonstrated in NUREG/CR-5512, Volume 1, Section 6.2.1
10. Room Exchange Rate	Total volume of air going out of the room per unit time divided by the volume of the room	0.8 hr^{-1}	Consistent with value of 0.75 hr^{-1} cited by American Society of Heating, Refrigerating, and Air-Conditioning Engineers, Inc. for conditioned spaces
11. In/Out Flow Rate	Flow rates of air into and out of the room	200 m^3/hr	Based on room volume of 250 m^3 and an air exchange rate of 0.8 hr^{-1}
Receptor Parameters			
12. Number of Receptors	Number of locations where doses are calculated	1	One representative location is chosen consistent with NUREG/CR-5512, Volume 1, Section 6.2.1
13. Room # Location	Room in which the receptor is located	1	Only one room is modeled consistent with NUREG/R-5512, Volume 1, Section 6.2.1
14. Time Fraction	Fraction of time within the building that the exposed individual spends at this receptor location.	1.0	Only one dose point is modeled consistent with NUREG/CR-5512, Volume 1, Section 6.2.1
15. Breathing Rate*	Inhalation rate of airborne material for this location	18 m^3/day	RESRAD-BUILD Code and NCRP 129, Table 4.12; realistic breathing rate for light office worker; See Section 6.5.1.2 for justification.
16. Ingestion Rate*	Ingestion rate of deposited material (dust) for this location	1.0E-4 m^2/hr	NUREG/CR-5512, Volume 1, Table 6.21; See Section 6.5.1.2 for justification.
17. Receptor Location*	Coordinates of the receptor (x,y,z) with respect to the sources for external exposure	5m, 5m, 1m	Located in center of room at height of 1m; representative of normally encountered conditions; consistent with NUREG/CR-5512, Volume 1, p. 6.4; ; See Section 6.5.1.2 for justification.

Table 6-14
Parameters for RESRAD-BUILD Building Occupancy Scenario

Parameter	Description	Value	Bases
Shielding Parameters			
18. Thickness	Thickness of the shielding between the contamination source and the receptor location	0	No intervening shielding materials between sources and building occupant
19. Density	Bulk density of the shielding material	Not Applicable	
20. Material	Identification of shielding material	Not Applicable	
Source Parameters			
21. Number of Sources	Number of sources in model	6	Places contamination on all room surfaces: floor, ceiling, four walls
22. Room # Location	All sources are located in Room #1	1	
23. Source Type	All sources are area sources (modeled as circular by RESRAD-BUILD)	Area	Places all contamination on surfaces; consistent with NUREG/CR-5512, Volume 1, Section 6.2.1 for building occupancy
24. Direction	Axis perpendicular to the exposed area	Floor (z), Ceiling (z), 4 Walls (x,y,x,y)	Orientation of sources within room consistent with NUREG/CR-5512, Volume 1, Section 6.2.1
25. Location	Center point of the source in x,y,z directions	Floor at 5m, 5m, 0m; Ceiling at 5m, 5m, 2.5m; Walls at 10m, 5, 1.25m; 5m, 10m, 1.25m; 0m, 5m, 1.25; 5m, 0m, 1.25m	Orientation of sources within room; contamination is uniformly distributed over all room surfaces; consistent with NUREG/CR-5512, Volume 1, Section 6.2.1
26. Geometry: Area	Area of the exposed surface over which the contamination is evenly distributed	100, 100, 25, 25, 25, 25 m^2	Representative of typical room in plant; based on optimal geometry demonstrated in NUREG/CR-5512, Volume 1, Section 6.2.1
27. Air Fraction*	Fraction of the eroded material that is released into the air	0.1	RESRAD-BUILD Code; See Section 6.5.1.2 for justification.

Table 6-14
Parameters for RESRAD-BUILD Building Occupancy Scenario

Parameter	Description	Value	Bases
28. Direct Ingestion	Direct ingestion rate of the source by any receptor in the room	0 g/hr	Direct ingestion of source material is not modeled; indirect ingestion of dust is modeled above (#16.)
29. Removable Fraction*	Fraction of the source that can be linearly removed between time zero and lifetime	0.1	10% of the contamination on the entire surface area of the room (floor, ceiling, four walls) is eroded over the course of one year; surfaces are assumed to be cleaned of easily removable contamination prior to license termination and building occupancy; See Section 6.5.1.2 for justification.
30. Lifetime*	Amount of time in which all of the removable fraction of the source is linearly eroded	365 days	RESRAD-BUILD Code; See Section 6.5.1.2 for justification.
31. Contamination: Radionuclide	Separate RESRAD-BUILD runs are made for each radionuclide of interest	See Table 6-2: Radionuclides Potentially Present in HNP	Radionuclides selected based on preliminary site surveys
32. Concentration	Unit concentration is initially run; results are normalized to 25 mrem per year to determine each isotopic DCGL	1.0 pCi/m^2	Initial concentration; DCGLs are obtained by multiplying by the ratio of 25 mrem to the resultant dose

*Parameters considered in Sensitivity Analysis; see additional discussion in Section 6.5.1.2, Parameter Justification.

Parameter Justification

Sensitivity studies were performed for major RESRAD-BUILD input parameters. The results presented in Table 6-16 show that the DCGLs exhibit strong sensitivity to indoor fraction, removable fraction, lifetime, air fraction, breathing rate, and building exchange rate. The DCGLs exhibit weak sensitivity to receptor height, room dimension, deposition velocity, resuspension rate, and ingestion rate.

Further discussion of the choice of parameter values is presented below.

Parameters exhibiting strong sensitivity:

Indoor Fraction:

An indoor fraction (occupancy factor) of 0.25 is assumed. This is based on a person working 40 hours/week for 52 weeks per year in the contaminated room (2080 hrs/8760 hrs = 0.24). The sensitivity analysis shows that DCGLs are directly dependent on this parameter.

The choice of indoor fraction of 0.25 is consistent with DandD, which uses a value of 97.46 days/year. This corresponds to an occupancy factor of 0.27. NUREG/CR-0672, Volume 2 (Reference 6-17), Section F.4.1 assumes an occupancy based on 40 hours worked each week for 50 weeks per year. NUREG/CR-5512, Volume 1, Table 6.21 considers an exposure duration of 2000 hours per year.

Removable Fraction, Lifetime, and Air Fraction:

RESRAD-BUILD considers two pathways to generate the airborne concentration arising from contaminated area sources. The primary source of airborne contamination is direct removal from the surface source, described by a removable fraction, time of source removal (called lifetime), and air fraction. For this analysis, a removable fraction of 0.1 was used for all surfaces of the room. This means that the floor, ceiling, and four walls all erode at the same rate, which is conservative. The time of source removal is 365 days, meaning that 10% of the source is linearly eroded during a year. This maximizes the dose during the first year of exposure. The fraction of removed source released into the air is assumed to be 0.1. (A secondary airborne source arises from the resuspension of loose material deposited on the floor area of the room. See discussion below of parameters exhibiting weak sensitivity regarding this parameter.)

The sensitivity analysis indicates that for those key nuclides with dominant dose contributions from the inhalation pathway (Fe-55, Ni-63, Sr-90, Tc-99, Pu-238, and Am-241) or the ingestion pathway (C-14), the DCGLs are directly affected by the choice of removable fraction, lifetime, and air fraction. For nuclides whose DCGLs are dominated by the direct exposure pathway (Co-60, Cs-137, Eu-152), the DCGLs are only minimally affected by the choice of removable fraction, lifetime, and air fraction.

For comparison to other references, the removable fraction and time of source removal can be expressed as a removal (or resuspension) rate:

$$Removal\ Rate = \frac{Removable\ Fraction}{Time\ of\ Source\ Removal} = \frac{(0.1)}{(365days)} * \frac{(day)}{(24hours)} = 1.14E-5hr^{-1}$$

or

$$Removal\ Rate = \left(1.14E-5hr^{-1}\right) * \left(\frac{hr}{3600\sec}\right) = 3.17E-9\sec^{-1}$$

Another method of comparison to other references is the calculation of a resuspension factor. As shown in the Draft Letter Report on DandD (Reference 6-18) :

$$Resuspension\ Factor = \frac{\lambda_{Resuspension}}{\lambda_{Ventilation}} * \left(\frac{Source\ Area}{Volume}\right)$$

or

$$Resuspension\ Factor = \frac{\left(1.14E-5hr^{-1}\right)}{\left(0.8hr^{-1}\right)} * \left(\frac{300m^2}{250m^3}\right) = 1.7E-5m^{-1}$$

The RESRAD model of the generation of the airborne source is consistent with other reported building occupancy models.

- Air release rates and respirable fractions for contamination on solid, noncombustible unyielding surfaces are reported in DOE Handbook 3010-94 (Reference 6-19), page 5-7. The bounding air release rate for aerodynamic entrainment and resuspension is 4E-5 hr^{-1}; the bounding respirable fraction is 1.0. These values apply indoors or outdoors with exposure to ambient conditions (normal process facility ventilation flow or less). However, the values are intended for immediate post-accident conditions and freshly deposited material. They would overestimate releases from long-term contamination (i.e., months to years).

- Per NUREG/CR-5512, Volume 2, DandD uses a single resuspension factor to generate an airborne concentration from contaminated sources with a given removable activity per unit area. The DandD resuspension factor value is 1.42E-5 m^{-1}.

- For a room occupancy scenario, the Health Physics Society uses a resuspension factor of 1E-6 m^{-1} along with a removable fraction of 1.0 (ANSI/HPS N13.12-1999, Reference 6-20).

- NUREG/CR-5512, Volume 1, Section 6.3.1 uses a resuspension factor of 1E-6 m^{-1} for building occupancy, because surfaces are assumed to be cleaned of easily removable contamination at the time of license termination.

- NUREG/CR-0672, Volume 2, Section F.3.2 uses a constant resuspension factor of 5E-6 m^{-1}. This value is one tenth of the value suggested for surfaces with removable contamination. This lower value is assumed since unrestricted release of nuclear facilities should not be contemplated with readily removable contamination remaining on accessible surfaces.

The data from these other references are summarized in Table 6-15 and demonstrate that the CY Model is well within those used by other studies.

Table 6-15
Comparison of Resuspension Rate and Resuspension Factor Data

Resuspension Rate: CY Model = 1.14E-5 hr^{-1}	
DOE Handbook 3010-94	4E-5 hr^{-1}
Resuspension Factor: CY Model = 1.7E-5 m^{-1}	
DandD	1.42E-5 m^{-1}
ANSI/HPS N13.12-1999	1E-6 m^{-1}
NUREG/CR-5512	1E-6 m^{-1}
NUREG/CR-0672	5E-6 m^{-1}

Breathing Rate:

The breathing rate is assumed to be 18 m^3/day. This value is considered to be a realistic long-term breathing rate for a building occupant who is performing light office work, as opposed to building renovation or demolition scenarios which would use a higher breathing rate.

The sensitivity analysis indicates that for those key nuclides with dominant dose contributions from the inhalation pathway (Fe-55, Ni-63, Sr-90, Tc-99, Pu-238, Am-241), the DCGLs are directly affected by the choice of breathing rate. For nuclides whose DCGLs are dominated by the direct exposure pathway (Co-60, Cs-137, and Eu-152) and the ingestion pathway (C-14), the DCGLs are minimally affected by the choice of breathing rate.

This breathing rate is consistent with NCRP 129 (Reference 6-21), which presents average breathing rates for various types of activity. Table 4.12 lists a range of 16 – 20 m^3/day for all activities. It also is within the range of breathing rates cited by EPA in the RESRAD Data Collection Handbook (ANL/EAIS-8, Reference 6-22), Table 43.1. The breathing rate for light activity level for average adults is 0.6 m^3/hr, equal to 14 m^3/day.

Building Exchange Rate:

The building exchange rate with the outside is assumed to be 0.8 hr^{-1}. Since only one room is modeled, this is also the room exchange rate. Based on the room size of 10.0m by 10.0m by 2.5m, the volume of the room is 250 m^3. An air exchange rate of 0.8 hr^{-1} equals 200 m^3/hr, or 118 cfm.

The sensitivity analysis indicates that for those key nuclides with dominant dose contributions from the inhalation pathway (Fe-55, Ni-63, Sr-90, Tc-99, Pu-238, Am-241) and the ingestion pathway (C-14), the DCGLs are directly affected by the choice of air exchange rate. For nuclides whose DCGLs are dominated by the direct exposure pathway (Co-60, Cs-137, Eu-152), the DCGLs are only minimally affected by the choice of air exchange rate.

The building exchange rate is consistent with the typical value cited by the American Society of Heating, Refrigerating and Air-Conditioning Engineers, Inc. (Reference 6-23) for conditioned

spaces in a house of 0.75 air exchanges per hour. Typical commercial air exchange rates are much higher, ranging from 1-4 hr^{-1} for a warehouse to 4-10 hr^{-1} for offices.

Parameters exhibiting weak sensitivity:

Receptor Height:

The receptor (building occupant) is assumed to be located in the center of the room at a height of 1.0 m. The choice of receptor location is conservatively selected to yield a representative dose and is what one could expect under normally encountered conditions.

As shown by the sensitivity analysis, the DCGLs for nuclides with a dominant external exposure pathway (Co-60, Cs-137, Eu-152) are only minimally affected by the height of the receptor. A receptor located at mid-height of the room (1.25 m above the floor) would receive a slightly lower dose. Consequently, DCGLs would be higher (non-conservative). Other nuclides with dominant inhalation and ingestion pathways are unaffected by the choice of receptor height.

The choice of receptor location is consistent with NUREG/CR-5512, Volume 1, Section 6.2.1 and ANL/EAD/LD-3 (Reference 6-24), Section 2.3, which states that "the receptor location should be the midpoint of the person. For example, if the receptor is standing on a contaminated floor, the receptor location should be 1 m above the floor." Also, NUREG/CR-0672, Volume 2, Section F.3.1 calculates dose rates at a point 1m above the floor.

Room Dimension:

The size of the room is chosen as 10.0 m by 10.0 m by 2.5 m in height. This is representative of a typical room in the plant, as indicated on plant drawings (Reference 6-25). The area (100 m^2) is also easily scalable to other room sizes. The room dimensions were selected to approximate the optimal geometry for this case as demonstrated in NUREG/CR-5512, Volume 1, Figure 6.1.

The sensitivity analysis shows that DCGLs are mildly dependent upon room size. For nuclides with dominant external exposure pathways (Co-60, Cs-137, and Eu-152), DCGLs vary inversely with room size up to a point. Large rooms begin to approximate infinite sources and DGCLs are no longer affected by further increases in room dimensions. For the other key nuclides with dominant inhalation and ingestion pathways, DCGLs increase with increasing room size. Larger volumes decrease the airborne concentration.

The size of the room is consistent with NUREG/CR-0672, Volume 2, Section F.3.1, which establishes the dimensions of a reference room as having a floor area of 154 m^2 and a height of 3m.

Deposition Velocity and Resuspension Rate:

In RESRAD-BUILD, a secondary airborne source arises from the resuspension of loose material deposited on the floor area of the room. This is modeled with a deposition velocity of 0.01 m/sec and a resuspension rate of 5.0E-7 sec^{-1}. (The primary airborne source is specified by a removable fraction, lifetime, and air fraction. See discussion above for parameters exhibiting strong sensitivity for these parameters).

The sensitivity analysis indicates that there is only one key nuclide with dominant dose contribution from the ingestion pathway (C-14). The DCGL for C-14 is directly affected by the

choice of deposition velocity and resuspension rate. For those key nuclides with dominant dose contributions from the inhalation pathway (Fe-55, Ni-63, Sr-90, Tc-99, Pu-238, and Am-241) or the direct exposure pathway (Co-60, Cs-137, Eu-152), the DCGLs are only minimally affected the choice of deposition velocity and resuspension rate.

The choice of deposition velocity and resuspension rate are based on ranges of values reported in Parameter Distributions for Use in RESRAD and RESRAD-BUILD Computer Codes (Reference 6-26).

Ingestion Rate:

RESRAD-BUILD uses two parameters to model ingestion of radioactive contamination. The primary source is the ingestion of loose contamination which has deposited on the horizontal surfaces (floor area) of the room. In the model, a value of 1.0E-4 m^2/hr for ingestion of loose contamination is assumed. The secondary source is direct ingestion of the volumetric source itself. Direct ingestion is not modeled, so the value of this parameter is set equal to zero.

The sensitivity analysis shows that for the key nuclides, only C-14 has a dominant ingestion pathway. Its DCGL is strongly sensitive to the choice of ingestion rate. Other key nuclide DCGLs are weakly affected.

The choice of ingestion rate is consistent with NUREG/CR-5512, Volume 1, Table 6.21, which lists an ingestion rate of 1.0E-4 m^2/hr from secondary ingestion of removable surface contamination. DandD models only one ingestion pathway, which is ingestion of contamination directly from the source. The DandD default value for ingestion rate, also called effective transfer rate, is 1.11E-5 m^2/hr.

Table 6-16
Normalized Sensitivity Coefficients, S_k for Building Occupancy DCGLs

Parameter	Co-60	Cs-137	C-14	Fe-55	Ni-63	Sr-90	Tc-99	Eu-152	Pu-238	Am-241
Receptor Height	0.09	0.08	-	-	-	0.01	-	0.09	-	-
Indoor Fraction	**-1.19**	**-1.19**	**-1.20**	**-1.19**	**-1.19**	**-1.19**	**-1.19**	**-1.18**	**-1.19**	**-1.18**
Room Dimension	-0.41	-0.39	0.37	0.38	0.37	0.35	0.37	-0.40	0.38	0.38
Removable Fraction	-0.03	-0.05	**-1.33**	**-1.33**	**-1.34**	**-1.21**	**-1.33**	-0.04	**-1.34**	**-1.33**
Lifetime	0.04	0.07	**1.00**	**1.00**	**0.99**	**0.94**	**1.00**	0.06	**1.00**	**1.00**
Air Fraction	-0.03	-0.05	**-1.33**	**-1.33**	**-1.34**	**-1.21**	**-1.33**	-0.04	**-1.34**	**-1.33**
Deposition Velocity	-0.01	-0.05	**-0.84**	-0.15	-0.19	-0.20	-0.33	-0.01	-0.01	-0.02
Resuspension Rate	0.02	0.06	**0.74**	0.16	0.22	0.23	0.36	0.02	0.02	0.02
Breathing Rate	-0.01	-	-0.28	**-0.71**	**-1.00**	**-0.85**	**-0.79**	-0.03	**-1.39**	**-1.38**
Ingestion Rate	-	-0.03	**-0.84**	-0.39	-0.20	-0.23	-0.33	-	-0.03	-0.02
HVAC Removal Rate	0.03	0.05	**1.01**	**0.78**	**0.99**	**0.92**	**1.00**	0.04	**0.99**	**1.00**

A positive value indicates direct proportionality, whereas a negative value indicates an inverse proportionality. Bolded text denotes areas of high sensitivity, which is defined to be $|S_k| > 0.5$. "-" indicates a value of 0.0.

6.5.1.3 Dose Modeling Results

Table 6-17 summarizes the results of the dose modeling calculations used to determine the DCGLs for the building occupancy scenario.

Table 6-17
Derived Concentration Guideline Levels for Building Occupancy

Nuclide	DCGL (pCi/m^2)	DCGL ($dpm/100cm^2$)
H-3	3.56E+10	7.89E+08
C-14	1.16E+09	2.57E+07
Mn-54	1.53E+06	3.40E+04
Fe-55	2.67E+09	5.94E+07
Co-60	5.27E+05	1.17E+04
Ni-59	2.73E+09	6.05E+07
Ni-63	1.15E+09	2.55E+07
Sr-90	5.00E+06	1.11E+05
Nb-94	7.84E+05	1.74E+04
Tc-99	7.27E+08	1.61E+07
Cs-134	8.17E+05	1.81E+04
Cs-137	2.17E+06	4.83E+04
Eu-152	1.09E+06	2.42E+04
Eu-154	1.02E+06	2.27E+04
Eu-155	1.95E+07	4.34E+05
Pu-238	2.23E+04	4.96E+02
Pu-239	2.03E+04	4.51E+02
Pu-241	1.11E+06	2.47E+04
Am-241	1.97E+04	4.37E+02
Cm-243	2.90E+04	6.45E+02

6.5.2 Checks of Base Case Scenario Model

As part of HNP decommissioning, the debris from the demolition of each decontaminated concrete structure may be placed in the basements of these structures. These concrete debris may contain reinforcing steel. When the building is demolished, it will be reduced to an elevation down to 3 feet below grade and the area regraded after backfill is complete. Basement floors will be perforated, as necessary, prior to the filling of the basements with concrete debris, thereby minimizing hydrostatic uplift forces on the subsurface structures.

Commercial facilities have been released under the building occupancy scenario historically by the NRC and, in fact, these buildings have been demolished and the debris left onsite. For example, at the Shoreham Nuclear Power Station, whose license was terminated and the facility released for unrestricted use in May 1995, the licensee left several large concrete blocks weighing between 4 and 7 tons on the

reactor floor. At the Fort St. Vrain Nuclear Generating Station, whose license was terminated and the site released for unrestricted use in June 1997, the licensee demolished the fuel building after completing the final survey. In this case, the survey report was approved by the NRC, and the debris remained onsite until after the license was terminated and the site released. (Reference 6-27)

Because the building debris may be used as backfill, two additional scenarios have been considered for site structures. The first is similar to the resident farmer scenarios but considers concrete in lieu of soil, and the second considers excavation and reuse of the concrete debris. If either of these additional scenarios results in a more limiting radionuclide-specific DCGL, the DCGL associated with that scenario will be used as input to the operational DCGL.

In this evaluation of building surface DCGLs, the resident farmer scenario for concrete debris uses the same basic approach as applied in the development of the soil DCGLs. In this case, the radioactivity concentration of the concrete debris from the building demolition is used to calculate the dose from farming activities in the land area adjacent to the demolished building. This includes water supplied from a well placed in the concrete debris within the demolished building basement. The excavator scenario evaluates the dose from the unlikely future excavation of potentially contaminated concrete debris. These various exposure scenarios (building occupancy, resident farmer, and excavator) are used to establish the most limiting surface contamination DCGL for survey units regardless of the final building configuration. A pictorial of this model checking process for demolished buildings is illustrated in Figure 6-10.

6.5.2.1 Check #1—Resident Farmer (Concrete Debris)

Concrete debris may contain residual levels of radioactivity. Disposition of the debris in the manner described above creates potential exposure pathways similar to those evaluated in the resident farmer scenario, which was used to develop DCGLs for soils as described in Section 6.4. A resident farmer scenario was therefore selected to test whether the DCGLs developed for the building occupancy scenario using the RESRAD-BUILD model are bounding. Figure 6-6 illustrates the logic behind this test.

This section describes the significant assumptions, conceptual model, dose modeling methodology, assignment of model parameters, and modeling results. The modeling assumptions and approach are identified and discussed as they are introduced.

6.5.2.1.1 Assumptions

Assumptions relevant to this analysis are summarized below:

1. The ultimate disposition of concrete debris resulting from the demolition of the Containment Building and the PAB will be in their respective basements. Concrete debris generated from the demolition of other decontaminated concrete buildings may placed in the basements of these two buildings as well.
2. All of the concrete is considered as potentially containing residual radioactivity.
3. The concrete debris fills each basement from its floor to an elevation 3 ft below the existing site grade. No credit is taken for any cover soil that will be placed over the debris-filled basements.
4. A resident farmer constructs his/her home over a debris-filled basement.
5. The well that supplies water for drinking, crop irrigation, and livestock is drilled and completed within the debris-filled basement.

Note that the third assumption of no cover material leads to conservative dose estimates. Upon completion of all decommissioning activities, cover soil will be placed over the debris-filled basements. The presence of this cover material will cause the direct exposure from the ground to be less than is

estimated here. Also, note that locating the well within the debris-filled basement, the fourth assumption, maximizes the levels of radioactivity in the well water, leading to conservative dose estimates. Due to the difficulty associated with drilling through concrete debris, it is unlikely that the resident farmer would elect to construct a well in this location, given the relative ease of installing a well elsewhere. The levels of radioactivity in a well constructed elsewhere, even immediately down-gradient of a debris-filled basement, will be less than those in the basement itself.

6.5.2.1.2 Conceptual Model

The conceptual model for this dose assessment is based on the planned disposition of concrete debris in the basements of the Containment Building and PAB. Plant drawings show that the floor elevation of the Containment Building basement is about 1.5 ft msl excluding the area under the reactor vessel, which is about -19.5 ft msl. The floor elevation of the PAB basement is about -19 ft msl. The existing site grade is about 21 ft msl. Upon completion of decommissioning activities, the basement walls will be demolished to a depth of 3 ft below the existing site grade at an elevation of about 18 ft msl. The water table elevation in the Containment Building and the PAB areas is approximately 10 ft msl (Reference 6-8). It is assumed that ground water within the debris-filled basements to rise to the ambient level of 10 ft msl.

The elevations at which concrete debris will be placed indicate that a significant portion of the concrete debris will be below the water table and in a saturated state. The concrete debris will extend about 8.5 ft below the water table for the Containment Building and about 29 ft below the water table in the case of the PAB. Debris will extend approximately 8 ft above the water table in both cases. The debris above the water table will be in an unsaturated state. Figure 6-11 illustrates the conceptual model that results from the assumptions and description provided above.

6.5.2.1.3 Applicable Pathways

Potential exposure pathways for the resident farmer scenario are defined in NUREG/CR-5512, Volume 1, and summarized in Section 6.4.3. These pathways may be categorized as water independent pathways and water dependent pathways. Table 6-18 summarizes the generally applicable pathways for residual radioactivity in soil. Note that these pathways were developed for residual radioactivity in soil rather than for residual radioactivity in concrete debris.

Table 6-18 also identifies the pathways considered applicable to concrete debris used as backfill at the HNP. The water independent pathways excluded (plant, meat, milk and soil ingestion) are not considered applicable to concrete debris. Cultivating and growing plants in concrete debris is not feasible; therefore, ingestion of plants and ingestion of products from livestock fed plants are not viable pathways. Ingestion of concrete dust by humans or livestock was excluded because this type of ingestion is usually associated with the ingestion of food crops by humans or fodder by livestock. Because it is unlikely that food crops and fodder will be grown in concrete debris, the soil (concrete dust) ingestion pathway was excluded as well. The water dependent fish ingestion pathway was also determined to be not applicable. Ground water flow directions identified in Reference 6-8 show that any radionuclides migrate from either the Containment Building or the PAB basements to the Connecticut River. The flow rate of potentially contaminated ground water relative to the flow rate of the Connecticut River is very small – any radionuclides that have migrated to the river would be reduced to very low levels.

Table 6-18
Applicable Pathways for Concrete Debris Under the Resident Farmer Scenario

Pathway	Applicable Pathways for Soil	Applicable Pathways for Concrete Debris
Water Independent		
Direct Exposure	✓	✓
Inhalation	✓	✓
Plant Ingestion	✓	
Meat Ingestion	✓	
Milk Ingestion	✓	
Soil Ingestion	✓	
Water Dependent		
Water Ingestion	✓	✓
Fish Ingestion	✓	
Plant Ingestion	✓	✓
Meat Ingestion	✓	✓
Milk Ingestion	✓	✓

6.5.2.1.4 Dose Modeling Approach

The conceptual model for concrete debris, used as backfill, includes sources of radioactivity both above and below the water table. The RESRAD code cannot be used to calculate doses or DCGLs directly for this scenario, because it allows only for a source above the water table. It was therefore necessary to develop a dose modeling approach consistent with the conceptual model. The steps involved in the calculation approach are summarized below:

1. Assume an initial radionuclide concentration for the concrete debris, S_0
2. Calculate using RESRAD the dose from water independent pathways, $Dose_1$
3. Determine the radionuclide concentration in ground water that is in contact with the concrete debris at the assumed concentration, C
4. Input C to RESRAD and calculate the dose from water dependent pathways, $Dose_2$
5. Sum $Dose_1$ and $Dose_2$ to obtain the total dose from all pathways for the assumed S_0, $Dose$
6. Scale S_0 to determine the concrete radionuclide concentration that gives a total effective dose equivalent of 25 mrem/yr, $DCGL_{Conc}$

Details of the dose modeling approach are described in this section along with bases for assigning the relevant model parameters.

Step 1. Initial Radionuclide Concentration

An initial concentration of S_0 = 100 pCi/g is assumed for each individual radionuclide. This represents the activity per unit mass of dry concrete. The magnitude of the assumed value is arbitrary. The assumed value is later scaled in Step 6 to determine the radionuclide concentration that gives a 25 mrem/year TEDE.

Step 2. Dose From Water Independent Pathways

The dose from the water independent pathways ($Dose_1$) is calculated using RESRAD, Version 5.91, for each radionuclide for the assumed S_0. The water independent pathways considered include direct radiation from the concrete debris assumed to be at the ground surface and inhalation of concrete dust. All other pathways in RESRAD were turned off for this step of the calculation. The model parameters required to complete this part of the dose calculation were assigned as follows:

- The contaminated zone area was taken to be 1533 m^2 and circular in shape. This area represents the footprint of the Containment Building basement. This basement, once filled with concrete debris, represents the bounding scenario for doses calculated for the water independent pathways.
- A contaminated zone thickness of 3 m was assigned. This conservatively represents the thickness of the concrete debris lying above the water table.
- Behavioral and metabolic parameters relevant to the water independent pathways were obtained from NUREG/CR-5512, Volume 2, and are unchanged from the values previously presented in Table 6-7.

Step 3. Initial Ground Water Concentrations

Radionuclide concentrations in ground water that is in contact with concrete debris are determined for the assumed S_0. These concentrations are calculated assuming the desorption of radionuclides from the concrete debris occurs instantaneously once saturated and that the distribution of the radionuclides in this solid-water system conforms to linear sorption theory. The sorption isotherm is expressed as

$$S = K_d C \qquad \text{(Equation 6-8)}$$

where: S = radionuclide concentration in concrete; K_d = distribution coefficient; and C = radionuclide concentration in ground water. Note that the desorption of radionuclides from concrete into ground water is a time-dependent, diffusion-limited process. Adopting an equilibrium approach is conservative in determining the ground water concentrations because no credit is taken for the radioactive decay that would occur during desorption. It is further assumed that the debris-filled basement behaves as a batch system (i.e., no inflow or outflow of water from the basement).

Based on the assumptions stated above, the ground water concentration associated with concrete debris containing a radionuclide at S_0 can be calculated from

$$C = \frac{1000 S_0 \rho_b}{\left(1 + \frac{K_d \rho_b}{n}\right) n} \qquad \text{(Equation 6-9)}$$

where: ρ_b = bulk density of concrete; and n = porosity of the concrete debris. This equation assumes that C is expressed in units of pCi/l and S_0 is expressed in units of pCi/g (hence, the conversion factor of 1000).

The calculation of C requires values of ρ_b and n for the concrete debris as well as K_d for each radionuclide. Values of ρ_b = 1.68 g/cm^3 and n = 0.3 have been assigned. Typically, these parameters exhibit little variability. K_d values for cement-water systems were obtained from a literature search. Table 6-19 summarizes the observed K_d range as well as the value adopted for dose modeling. (Note that the K_d for a given radionuclide in a cement-water system is expected to differ in magnitude from that in a soil-water system, because of dissimilarities in the two geochemical environments. Therefore, the K_d values in

Tables 6-2 and 6-19 differ in magnitude.) As Table 6-19 illustrates, this parameter exhibits significant variability. Most of these values were obtained from the preferred K_d database for water-cement mixtures published in NUREG/CR-6377 (Reference 6-28). It should be noted that the adopted values represent the low end of the observed range, which leads to conservative (high) estimates of ground water concentrations and therefore dose.

Currently, a program is being implemented to determine site-specific K_d values for selected radionuclides. This involves obtaining concrete cores from representative floors and walls, and measuring K_d values through laboratory analyses. If the laboratory-derived distribution coefficients cause the DCGLs for concrete to change significantly, the K_d values in Table 6-19 and associated $DCGL_{Conc}$ (DCGL for concrete) values will be revised as appropriate.

Table 6-19
K_d Values for Concrete Debris (cm^3/g)

Radio-nuclide	Minimum	Maximum	Adopted	Source
H-3	-	-	0[c]	-
C-14	100	10,000	100	Reference 6-28
Mn-54	NA	NA	100	Behavior similar to Ni assumed
Fe-55	NA	NA	100	Behavior similar to Ni assumed
Co-60	NA	NA	100	Behavior similar to Ni assumed
Ni-59	500	3,000	100	Reference 6-28
Ni-63	500	3,000	100	Reference 6-28
Sr-90	1	5	3	Reference 6-28
Nb-94	11	69,000	1000	Reference 6-28
Tc-99	-	-	0[b]	Reference 6-28
Cs-134	1	115	3	Reference 6-29 Reference 6-30
Cs-137	1	115	3	Reference 6-29 Reference 6-30
Eu-152	NA	NA	5,000[a]	Reference 6-28
Eu-154	NA	NA	5,000[a]	Reference 6-28
Eu-155	NA	NA	5,000[a]	Reference 6-28
Pu-238	390	17,000	5,000	Reference 6-28
Pu-239	390	17,000	5,000	Reference 6-28
Pu-241	390	17,000	5,000	Reference 6-28
Am-241	2000	35,000	5,000	Reference 6-28
Cm-243	NA	NA	5,000	Behavior similar to Am assumed

[a] Lanthanide elements expected to have adsorption properties analogous to Am-241.
[b] Zero value anticipated under oxidizing conditions; non-zero values observed under reducing conditions.
[c] Tritium, as a component of water, does not partition in the same sense as other radionuclides.
NA – Not available.

Step 4. Dose From Water Dependent Pathways

The dose from the water dependent pathways ($Dose_2$) is calculated for individual radionuclides for the assumed S_0 of 100 pCi/g and the corresponding ground water concentration determined in Step 3. This calculation is performed by inputting the ground water concentrations determined in Step 3 directly into the RESRAD model. This is accomplished through the use of the "Well-Water Concentration Option" in the "Concentration Report" module in RESRAD. All water independent pathways, already accounted for in Step 2, are turned off for this calculation. With well water concentrations prescribed, the only RESRAD features being exercised are those that convert exposure concentrations in well water to dose for the applicable pathways.

Adjustment of some parameters is required to ensure that the values in the "Concentration Report" equal or exceed the ground water concentrations calculated in Step 3 and input directly to RESRAD. The following parameters were adjusted as follows:

- Contaminated zone thickness = 0.05 m
- Unsaturated zone thickness = 0.01 m
- Time since material placement = 1 year

Additional parameters require adjustment to ensure that the water independent dose contributions from the plant, meat, and milk are zero. The following adjustments were made:

- Cover depth = 0.5 m
- Depth of roots = 0.009 m

Because ground water concentrations are input directly to RESRAD, there is zero sensitivity to the hydrologic parameters, unsaturated zone properties, and saturated zone properties that are normally used by RESRAD to calculate ground water concentrations. These parameters therefore play no role in calculation of dose from the water dependent pathways. Behavioral and metabolic parameters relevant to the water dependent pathways were obtained from NUREG/CR-5512 and are unchanged from the values previously presented in Tables 6-6 and 6-7.

Step 5. Total Dose

The total dose for the assumed S_0 = 100 pCi/g is calculated by summing the doses from the water independent and the water independent pathways, i.e.,

$$Dose = Dose_1 + Dose_2 \qquad \text{(Equation 6-10)}$$

Step 6. DCGL Determination for Concrete Debris

Radionuclide-specific DCGLs are calculated for concrete debris by scaling the assumed S_0 = 100 pCi/g by the ratio of the 25 mrem/yr TEDE limit to the total dose associated with S_0 and calculated in Step 5. This calculation is made in accordance with the following equation:

$$DCGL_{Conc} = \frac{25S_0}{Dose} \qquad \text{(Equation 6-11)}$$

As a check on the validity of the approach, several of the calculated $DCGL_{Conc}$ were used as the initial S_0 and the dose recalculated per Steps 2 through 5. The calculated doses agreed with the 25 mrem/yr TEDE limit.

6.5.2.1.5 Dose Modeling Results for Concrete Debris

Table 6-20 summarizes the results of the dose modeling calculations used to determine the DCGLs for concrete debris placed in a basement at the HNP site. For a given radionuclide, the larger of the water dependent and water independent doses have been bolded to help identify the pathway that contributes most to the dose. These results show that the dose and corresponding DCGL for concrete debris, used as backfill, is controlled by the water dependent pathways for two-thirds of the radionuclides.

The volumetric DCGL values ($DCGL_{conc}$) from Table 6-20 are converted to surface contamination DCGL values (DCGLs) such that surface contamination measurements may be evaluated. This conversion is performed by assuming that the entire quantity of radioactivity within the volume occupied by the available fill area up to 3 feet below grade, V, is distributed on the internal surface area of the building, A_b. The value of A_b is determined based on the surface area of concrete prior to demolition. This conversion is performed as follows.

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

(Equation 6-12)

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

Substituting these values and combining terms provides the following relationship:

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

(Equation 6-13)

The minimum value of V/A_b has been evaluated and documented in Reference 6-31. The minimum value of V/A_b was determined to be for the containment building where

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

The value of A_b represents the surface area of concrete within the containment building. The values presented in Table 2-6 represent all structural surfaces within the containment, including steel surfaces, that will be included in the final status survey.

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

$$DCGL_s(dpm/100cm^2) = 12{,}881 * DCGL_{conc}(pCi/g) \qquad \text{(Equation 6-14)}$$

The volumetric concrete debris DCGL values are converted to surface activity DCGL values using the above relationship and are provided in Table 6-20 for the resident farmer concrete debris case. These surface activity DCGL values are conservative since all of the concrete debris within the backfill volume is assumed contaminated at a level corresponding to the volumetric DCGL. This is unlikely since a large fraction of internal surfaces of the plant structures contain very low levels of contamination. This total activity is then distributed uniformly over all internal surfaces of the building. The resulting surface DCGL is conservative considering that no allowances are made for the large area occupied by the non-contaminated surface areas.

Table 6-20
Concrete Debris Dose Modeling Results and DCGL Summary

Radio-nuclide	Initial Conc, pCi/g	Water independent dose, mrem	Ground Water Conc. pCi/l	Water Dependent Dose, mrem	Total Dose, mrem	Concrete Debris DCGL* (pCi/g)	Extrapolated Pre-Demolition Concentration ($dpm/100cm^2$)
H-3	100	1.01E-01	560,000	**4.36E+00**	4.46E+00	5.61E+02	7.23E+06
C-14	100	4.87E-02	998	**3.39E+00**	3.44E+00	7.28E+02	9.38E+06
Mn-54	100	**1.50E+02**	998	1.27E+00	1.51E+02	1.65E+01	2.13E+05
Fe-55	100	8.95E-07	998	**4.59E-01**	4.59E-01	5.45E+03	7.02E+07
Co-60	100	**6.46E+02**	998	2.45E+01	6.70E+02	3.73E+00	4.80E+04
Ni-59	100	1.02E-06	998	**4.22E-01**	4.22E-01	5.92E+03	7.63E+07
Ni-63	100	2.36E-06	998	**1.16E+00**	1.16E+00	2.16E+03	2.78E+07
Sr-90	100	1.13E+00	31,461	**2.46E+03**	2.46E+03	1.02E+00	1.31E+04
Nb-94	100	**4.09E+02**	100	5.54E-01	4.09E+02	6.11E+00	7.87E+04
Tc-99	100	2.95E-03	560,000	**1.09E+02**	1.09E+02	2.29E+01	2.95E+05
Cs-134	100	3.35E+02	31,461	**1.81E+03**	2.14E+03	1.17E+00	1.51E+04
Cs-137	100	1.40E+02	31,461	**1.37E+03**	1.51E+03	1.65E+00	2.13E+04
Eu-152	100	**2.90E+02**	20	1.18E-01	2.90E+02	8.61E+00	1.11E+05
Eu-154	100	**3.15E+02**	20	1.71E-01	3.15E+02	7.94E+00	1.02E+05
Eu-155	100	**7.42E+00**	20	2.65E-02	7.45E+00	3.36E+02	4.33E+06
Pu-238	100	1.54E-01	20	**5.81E+01**	5.83E+01	4.29E+01	5.53E+05
Pu-239	100	1.75E-01	20	**6.46E+01**	6.47E+01	3.86E+01	4.97E+05
Pu-241	100	5.49E-03	20	**1.38E+00**	1.39E+00	1.80E+03	2.32E+07
Am-241	100	2.08E+00	20	**6.63E+01**	6.84E+01	3.66E+01	4.71E+05
Cm-243	100	2.51E+01	20	**4.51E+01**	7.03E+01	3.56E+01	4.59E+05

*Volumetric DCGLs, for relevant radionuclides, will be used to address activated concrete.

6.5.2.2 Check #2—Excavator Scenario

The second check, shown in Figure 6-12, is to test whether the DCGL developed by the RESRAD-BUILD model is also bounded by the DCGL that would be associated with an excavator scenario, i.e. an event when the critical group is represented by a person who excavates the concrete debris. For this case, the model, depicted in Figure 6-12, postulates the very conservative situation where the debris is reconfigured into an infinite plane source. The conservatism is used only to demonstrate that even in the most restrictive and unlikely situation described above, doses are in fact bounded by the building occupancy scenario when compared to excavation and reuse of the material by an excavator.

6.5.2.2.1 Conceptual Dose Model Approach

The excavator scenario considers the reuse of concrete debris. A person who excavates the concrete debris would be exposed to external radiation. The following assumptions have been made for the excavator dose model:

1. The concrete is assumed to be reconfigured into an infinite planar source with the radioactive source facing the worker. The infinite planar source is approximated in RESRAD-BUILD by inputting an area of 1000 m^2 for the source.

2. The residual surface radioactivity is characterized as an area source.

3. The receptor is assumed to be located in the center of the area source at a height of 1m. The choice of receptor location is conservatively selected to yield a maximum dose.

4. An indoor fraction (occupancy factor) of 0.25 is assumed. This is based on the person working 40 hours/week for 52 weeks per year on the slab.

5. Only the direct external dose pathway is considered. Parameters for all other pathways are set equal to zero.

6.5.2.2.2 Applicable Pathways

Only the direct external dose pathway is considered for the excavator scenario.

6.5.2.2.3 Assignment of Input Parameters

Table 6-21 summarizes the RESRAD-BUILD input parameters used in this scenario. Figure 6-13 illustrates the source and receptor locations.

Table 6-21
Parameters for RESRAD-BUILD Excavator Scenario

Parameter	Description	Value	Source
Time Parameters			
1. Exposure Duration	Amount of time that exposure occurs	365 days	Annual doses calculated per 10 CFR 20.1402
2. Indoor Fraction	Fraction of the exposure duration that is spent on source	0.25 (40 hr/wk, 52 wk/yr)	Based on typical work schedule; consistent with NUREG/CR-5512, Vol. 1, Table 6.21
3. Evaluation Time	Times at which doses are calculated	0 year; 1 year (other times if necessary to find peak dose; Pu-241 peak dose occurs at t=100 years)	Doses are conservatively calculated with the least credit for decay; consistent with NUREG/CR-5512, Vol. 1, Section 3.2.1
Building Parameters			
4. Number of Rooms	Number of separate compartments in model	1	Simple model of single slab source
5. Deposition Velocity	Velocity at which airborne particles are deposited onto surfaces	NA	Parameter not used because inhalation and ingestion pathways are not modeled in the excavator scenario
6. Resuspension Rate	Rate at which deposited material is resuspended into the air	NA	Parameter not used because inhalation and ingestion pathways are not modeled in the excavator scenario
7. Building Exchange Rate	Total volume of air going out of the building per unit time divided by the total volume of the building	NA	Parameter not used because inhalation and ingestion pathways are not modeled in the excavator scenario
8. Room Area	Floor area of the slab	$1000m^2$	Approximates infinite source
9. Height	Height of the room	NA	Parameter not used because inhalation and ingestion pathways are not modeled in the excavator scenario

Table 6-21
Parameters for RESRAD-BUILD Excavator Scenario

Parameter	Description	Value	Source
10. Room Exchange Rate	Total volume of air going out of the room per unit time divided by the volume of the room	NA	Parameter not used because inhalation and ingestion pathways are not modeled in the excavator scenario
11. In/Out Flow Rate	Flow rates of air into and out of the room	NA	Parameter not used because inhalation and ingestion pathways are not modeled in the excavator scenario
Receptor Parameters			
12. Number of Receptors	Number of locations where doses are calculated	1	One worst case receptor location is specified with regard to the slab source
13. Room # Location	Room in which the receptor is located	1	Simple model of single slab source
14. Time Fraction	Fraction of time that the exposed individual spends at this receptor location.	1	One worst case receptor location is specified with regard to the slab source
15. Breathing Rate	Inhalation rate of airborne material for this location	0	Parameter set to zero because inhalation pathway is not modeled in the excavator scenario
16. Ingestion Rate	Ingestion rate of deposited material (dust) for this location	0	Parameter set to zero because ingestion pathway is not modeled in the excavator scenario
17. Receptor Location	Coordinates of the receptor (x,y,z) with respect to the sources for external exposure	0,0,1	Located at center of slab at a height of 1m; representative of normally encountered conditions; consistent with NUREG/CR, Vol. 1, p. 6.4

Table 6-21
Parameters for RESRAD-BUILD Excavator Scenario

Parameter	Description	Value	Source
Shielding Parameters			
18. Thickness	Thickness of the shielding between the contamination source and the receptor location	0	No intervening shielding materials between source and excavator
19. Density	Bulk density of the shielding material	NA	
20. Material	Identification of shielding material	NA	
Source Parameters			
21. Number of Sources	Number of sources in model	1	Simple model of single source
22. Room # Location	Sources is located in Room #1	1	Simple model of single source
23. Source Type	Sources is an area source (modeled as circular by RESRAD-BUILD)	Area	Places all contamination on the surface of the reconfigured concrete slab
24. Direction	Axis perpendicular to the exposed area	Z	Orientation of reconfigured concrete slab
25. Location	Center point of the source in x,y,z directions	0m,0m,0m	Orientation of reconfigured concrete slab
26. Geometry: Area	Area of the exposed surface over which the contamination is evenly distributed	$1000m^2$	Orientation of reconfigured concrete slab; approximates infinite source
27. Air Fraction	Fraction of the eroded material that is released into the air	NA	Parameter not used because inhalation and ingestion pathways are not modeled in the excavator scenario
28. Direct Ingestion	Direct ingestion rate of the source by any receptor in	NA	Parameter not used because ingestion pathway is not modeled in the excavator scenario

Table 6-21
Parameters for RESRAD-BUILD Excavator Scenario

Parameter	Description	Value	Source
	the room		
29. Removable Fraction	Fraction of the source that can be linearly removed between time zero and lifetime	NA	Parameter not used because inhalation and ingestion pathways are not modeled in the excavator scenario
30. Lifetime	Amount of time in which all of the removable fraction of the source is linearly eroded	NA	Parameter not used because inhalation and ingestion pathways are not modeled in the excavator scenario
31. Contamination: Radionuclide	Separate RESRAD-BUILD runs are made for each radionuclide of interest	See Table 6-2: Radionuclides Potentially Present in HNP	Radionuclides selected based on preliminary site surveys
32. Concentration	Unit concentration is initially run; results are normalized to 25 mrem per year to determine each isotopic DCGL	1.0 pCi/m^2	Initial concentration; DCGLs are obtained by multiplying by the ratio of 25 mrem to the resultant dose

6.5.2.2.4 Excavator Scenario DCGL Results

Table 6-22 summarizes the results of the dose modeling calculations used to determine the DCGLs for the excavator scenario.

Table 6-22
DCGLs for Excavator Scenario

Nuclide	DCGL (pCi/m^2)	DCGL ($dpm/100cm^2$)
H-3	NA	NA
C-14	NA	NA
Mn-54	1.97E+06	4.37E+04
Fe-55	NA	NA
Co-60	6.91E+05	1.53E+04
Ni-59	NA	NA
Ni-63	NA	NA
Sr-90	1.41E+08	3.14E+06
Nb-94	1.05E+06	2.33E+04
Tc-99	NA	NA
Cs-134	1.07E+06	2.37E+04
Cs-137	2.92E+06	6.48E+04
Eu-152	1.44E+06	3.19E+04
Eu-154	1.36E+06	3.02E+04
Eu-155	2.58E+07	5.73E+05
Pu-238	1.01E+09	2.24E+07
Pu-239	2.34E+09	5.19E+07
Pu-241*	1.42E+09	3.15E+07
Am-241	4.13E+07	9.17E+05
Cm-243	1.43E+07	3.17E+05

*Pu-241 peak dose occurs at time = 100 yrs; all other radionuclides peak at time = 0 yrs.

6.5.3 Determination of Bounding Building Surface DCGL for Each Radionuclide

Based on the parameters shown in Table 6-14, individual RESRAD-BUILD calculations have been performed for each radionuclide to determine the DCGL in dpm/100cm^2. The bounding value from the comparison process shown in Figure 6-6 among the building occupancy scenario and the tests for the resident farmer scenario for concrete debris and the excavator scenario are used as the final DCGL for that radionuclide. These final values along with the values from the building occupancy scenario, the resident farmer (concrete debris) test and the excavator test are shown in Table 6-23. The next to the last column in Table 6-23 list the base case surface contamination DCGLs in units of dpm per 100 cm^2. The last column indicates which scenario dictated the bounding DCGL. Table 6-24 reprises the bounding DCGLs for each radionuclide in dpm/100cm^2.

6.6 Demonstration of Compliance With Site Release Criteria

In order to demonstrate compliance with the 25 mrem/yr criteria, operational DCGLs will be developed as described in Section 5.4.6. Actual assayed relative concentrations from surveys will be folded with these DCGLs using the unity rule as shown in Equation 5-19 to assure compliance. In order to fully meet the requirements of 10 CFR 20.1402, ALARA considerations must also be addressed. The details for how ALARA will be considered are described in Section 4.

Table 6-23
Comparison of Building Surface DCGLs from All Postulated Scenarios

	DCGL (dpm/100cm^2)				
Radio-nuclide	Building Occupancy	Check 1: Resident Farmer (Concrete Debris)	Check 2: Excavator	Bounding (Minimum)	Controlling Scenario
H-3	7.89E+08	7.23E+06	NA	7.23E+06	Resident Farmer (Concrete Debris)
C-14	2.57E+07	9.38E+06	NA	9.38E+06	Resident Farmer (Concrete Debris)
Mn-54	3.40E+04	2.13E+05	4.37E+04	3.40E+04	Building Occupancy
Fe-55	5.94E+07	7.02E+07	NA	5.94E+07	Building Occupancy
Co-60	1.17E+04	4.80E+04	1.53E+04	1.17E+04	Building Occupancy
Ni-59	6.05E+07	7.63E+07	NA	6.05E+07	Building Occupancy
Ni-63	2.55E+07	2.78E+07	NA	2.55E+07	Building Occupancy
Sr-90	1.11E+05	1.31E+04	3.14E+06	1.31E+04	Resident Farmer (Concrete Debris)
Nb-94	1.74E+04	7.87E+04	2.33E+04	1.74E+04	Building Occupancy
Tc-99	1.61E+07	2.95E+05	NA	2.95E+05	Resident Farmer (Concrete Debris)
Cs-134	1.81E+04	1.51E+04	2.37E+04	1.51E+04	Resident Farmer (Concrete Debris)
Cs-137	4.83E+04	2.13E+04	6.48E+04	2.13E+04	Resident Farmer (Concrete Debris)
Eu-152	2.42E+04	1.11E+05	3.19E+04	2.42E+04	Building Occupancy
Eu-154	2.27E+04	1.02E+05	3.02E+04	2.27E+04	Building Occupancy
Eu-155	4.34E+05	4.33E+06	5.73E+05	4.34E+05	Building Occupancy
Pu-238	4.96E+02	5.53E+05	2.24E+07	4.96E+02	Building Occupancy
Pu-239	4.51E+02	4.97E+05	5.19E+07	4.51E+02	Building Occupancy
Pu-241	2.47E+04	2.32E+07	3.15E+07	2.47E+04	Building Occupancy
Am-241	4.37E+02	4.71E+05	9.17E+05	4.37E+02	Building Occupancy
Cm-243	6.45E+02	4.59E+05	3.17E+05	6.45E+02	Building Occupancy

Table 6-24
Bounding Building Surface DCGLs

Radio-nuclide	Surface (dpm/100cm2)	Controlling Scenario
H-3	7.23E+06	Resident Farmer (Concrete Debris)
C-14	9.38E+06	Resident Farmer (Concrete Debris)
Mn-54	3.40E+04	Building Occupancy
Fe-55	5.94E+07	Building Occupancy
Co-60	1.17E+04	Building Occupancy
Ni-59	6.05E+07	Building Occupancy
Ni-63	2.55E+07	Building Occupancy
Sr-90	1.31E+04	Resident Farmer (Concrete Debris)
Nb-94	1.74E+04	Building Occupancy
Tc-99	2.95E+05	Resident Farmer (Concrete Debris)
Cs-134	1.51E+04	Resident Farmer (Concrete Debris)
Cs-137	2.13E+04	Resident Farmer (Concrete Debris)
Eu-152	2.42E+04	Building Occupancy
Eu-154	2.27E+04	Building Occupancy
Eu-155	4.34E+05	Building Occupancy
Pu-238	4.96E+02	Building Occupancy
Pu-239	4.51E+02	Building Occupancy
Pu-241	2.47E+04	Building Occupancy
Am-241	4.37E+02	Building Occupancy
Cm-243	6.45E+02	Building Occupancy

* Operational DCGLs may be lower than values presented above.

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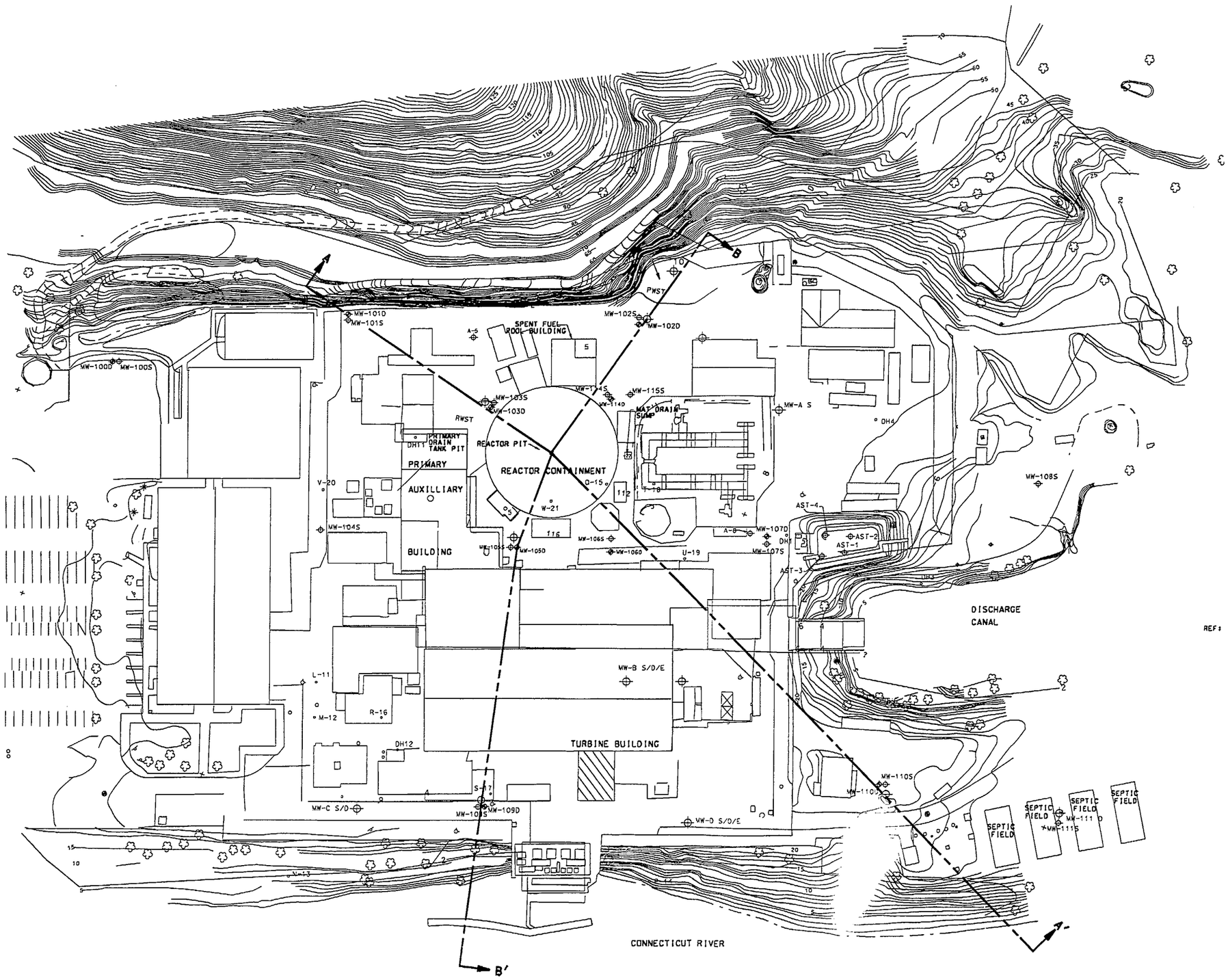


LEGEND

- MW-110S — SHALLOW MONITORING WELL
- MW-110D — DEEP MONITORING WELL
- BORING (1963 PRECONSTRUCTION)

TOPOGRAPHIC MAP PREPARED BY:
LIU AERIAL SURVEYS, WATERTOWN
CONNECTICUT, DATED
NOVEMBER 20, 1992, DWG.NO.21627,
SHEET 2 OF 6

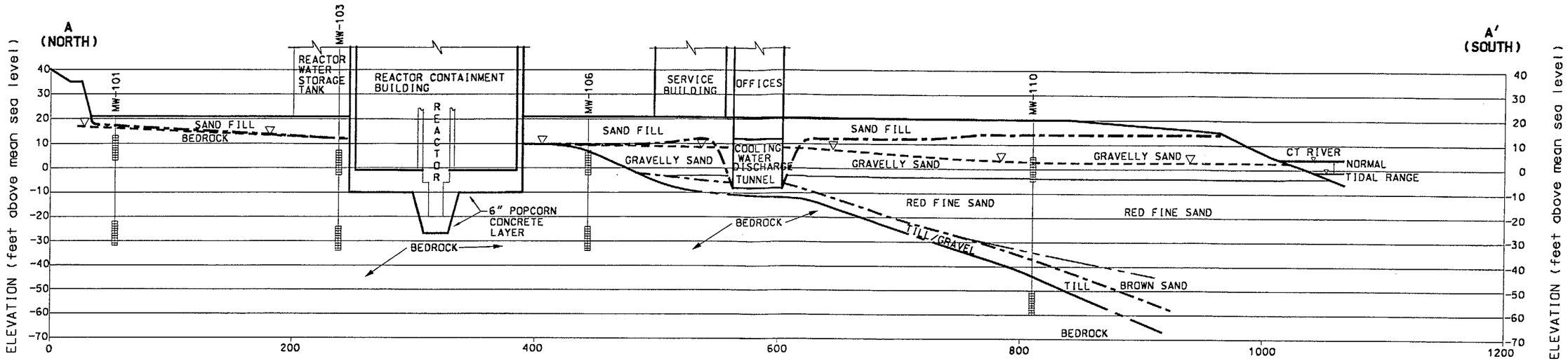
REF: MALCOLM PIRNIE, INC., 1999, "GROUND WATER MONITORING REPORT,
CONNECTICUT YANKEE ATOMIC POWER STATION,
HADDAM NECK, CONNECTICUT"
PREPARED FOR CONNECTICUT YANKEE ATOMIC POWER COMPANY

CONNECTICUT YANKEE ATOMIC POWER COMPANY

SITE LAYOUT
FIGURE 6-1

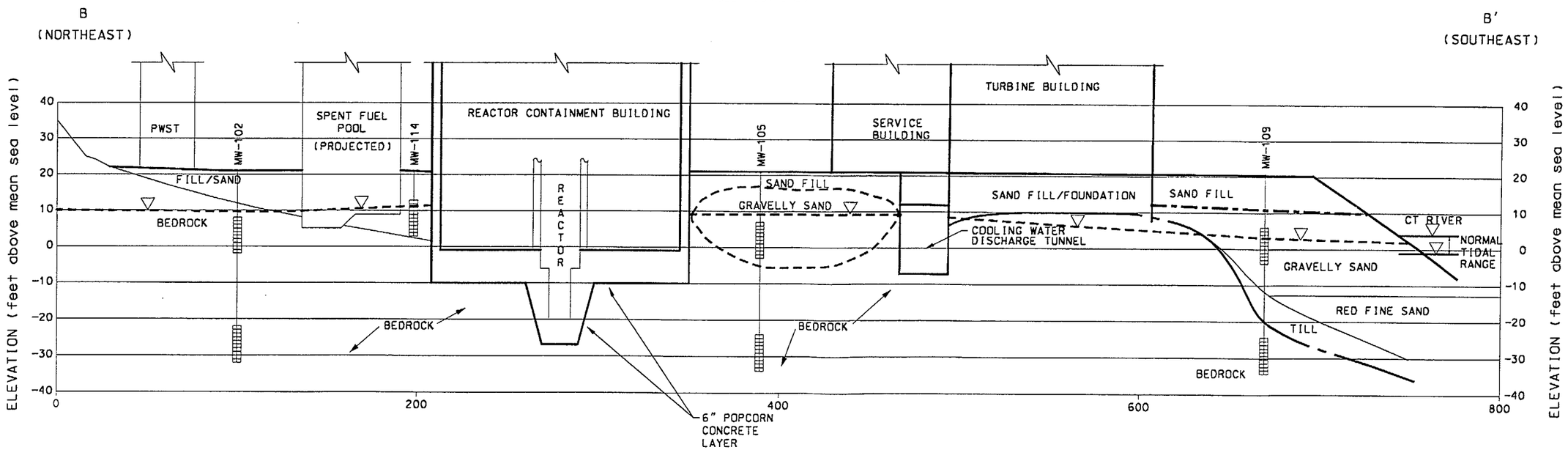
22 x 34 "D" SIZE

c:\dgn\site2d.dgn May. 17, 2000 09:31:15



SECTION "A-A'

(SEE FIG. 6-2)



SECTION "B-B'

(SEE FIG. 6-2)

REF: MALCOLM PIRNIE, INC., 1999, "GROUND WATER MONITORING REPORT, CONNECTICUT YANKEE ATOMIC POWER STATION, HADDAM NECK, CONNECTICUT" PREPARED FOR CONNECTICUT YANKEE ATOMIC POWER COMPANY

CONNECTICUT YANKEE ATOMIC POWER COMPANY

INDUSTRIAL AND PENINSULA AREA CROSS SECTION FIGURE 6-2

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

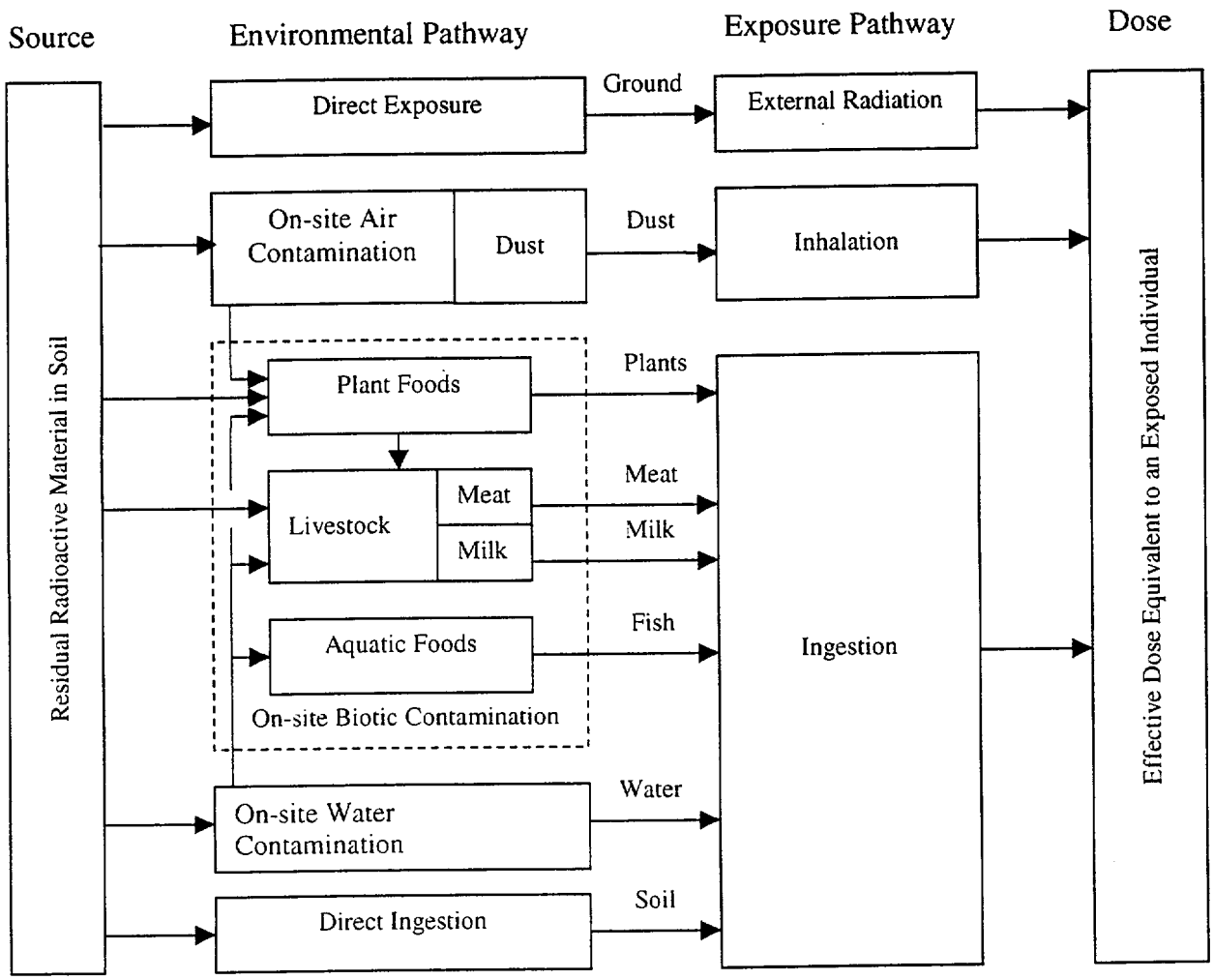
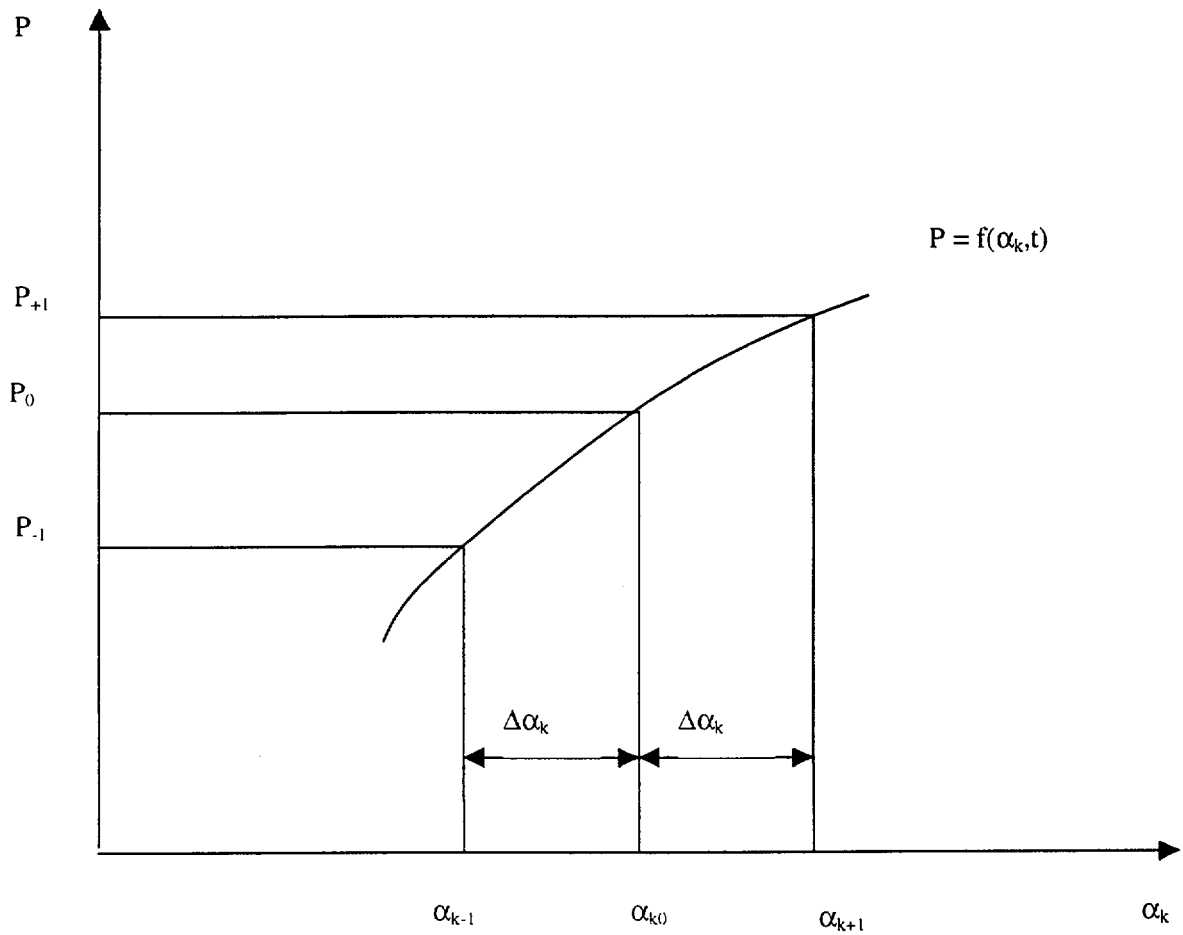


Figure 6-4
Illustration of Normalized Sensitivity Coefficients



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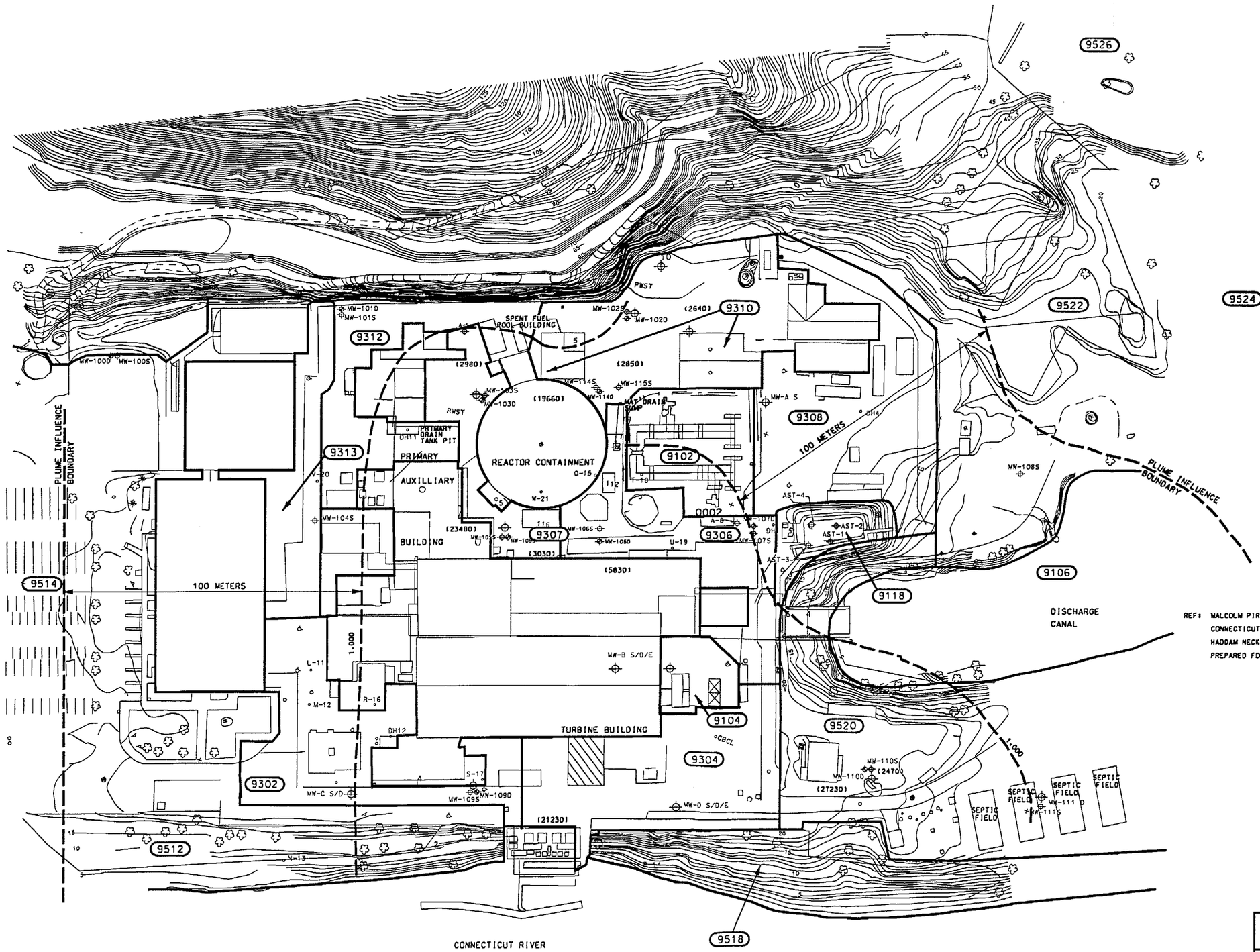


LEGEND

- MW-110S — SHALLOW MONITORING WELL
- MW-110D — DEEP MONITORING WELL
- BORING (1963 PRECONSTRUCTION)
- (27230) TRITIUM CONCENTRATION (pCi/L) SEPT.1999
- --20,000-- INFERRED TRITIUM CONCENTRATION CONTOUR (ADAPTED FROM MALCOLM - PIRNIE, 1999
- 9106 SURVEY AREA DESIGNATION
- SURVEY BOUNDARY

TOPOGRAPHIC MAP PREPARED BY: LIU AERIAL SURVEYS, WATERTOWN CONNECTICUT, DATED NOVEMBER 20, 1992, DWG.NO.21627, SHEET 2 OF 6

REF: MALCOLM PIRNIE, INC., 1999, "GROUND WATER MONITORING REPORT. CONNECTICUT YANKEE ATOMIC POWER STATION. HADDAM NECK, CONNECTICUT" PREPARED FOR CONNECTICUT YANKEE ATOMIC POWER COMPANY

CONNECTICUT YANKEE ATOMIC POWER COMPANY

GROUND WATER PLUME INFLUENCE BOUNDARY FIGURE 6-5

REF DFN:(XXXXXXX.CVL) DFN:(CYISFSI,EL)XXXXXXX.DGN 22 x 34 "D" SIZE
LEVELS 1234567890123456789012345678901234567890123456789012345678901234567890123 PSC-

...\dgn\groundwtrplumeboundary.dgn May. 26, 2000 09:57:56

Figure 6-6
Process for Determining Building Surface DCGLs

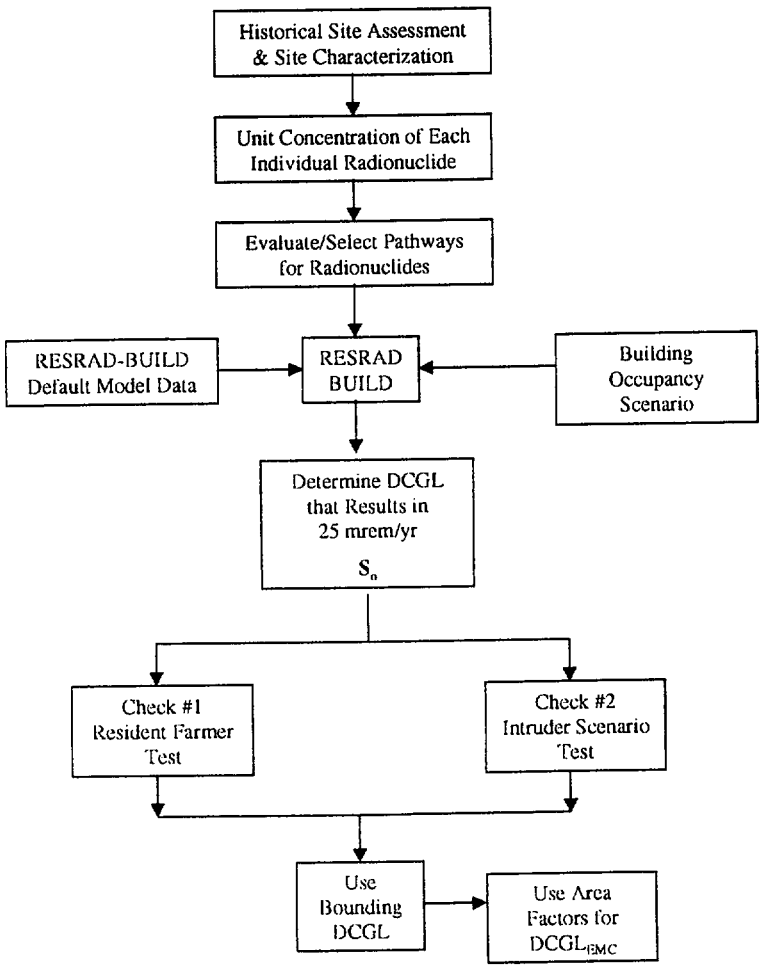


Figure 6-7
Building Occupancy Scenario

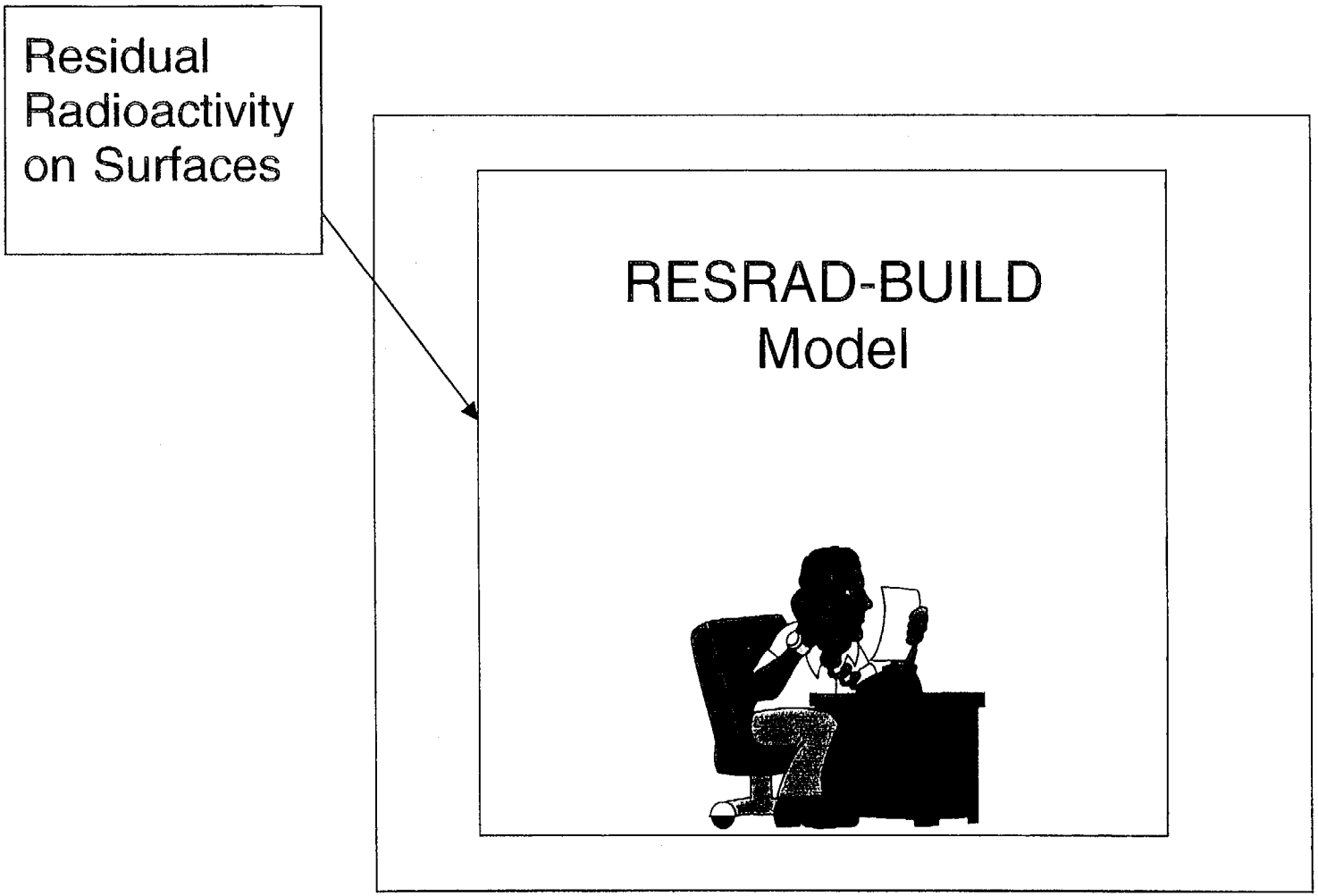
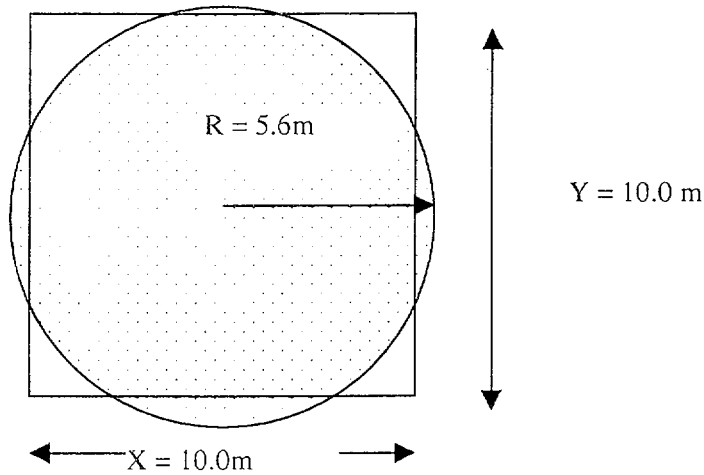


Figure 6-8
RESRAD-BUILD Models of Floor, Walls, and Ceiling

Floor and Ceiling: Each Square Area = (10.0m)(10.0m) = $100m^2$
NOTE: Treated as two circular areas in RESRAD-BUILD



Four Walls: Each Rectangular Area = (10.0m)(2.5m) = $25m^2$
NOTE:Treated as four circular areas in RESRAD-BUILD with R = 2.8m

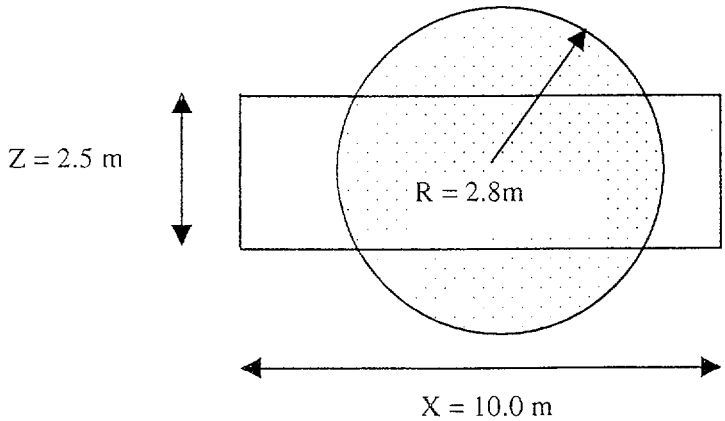
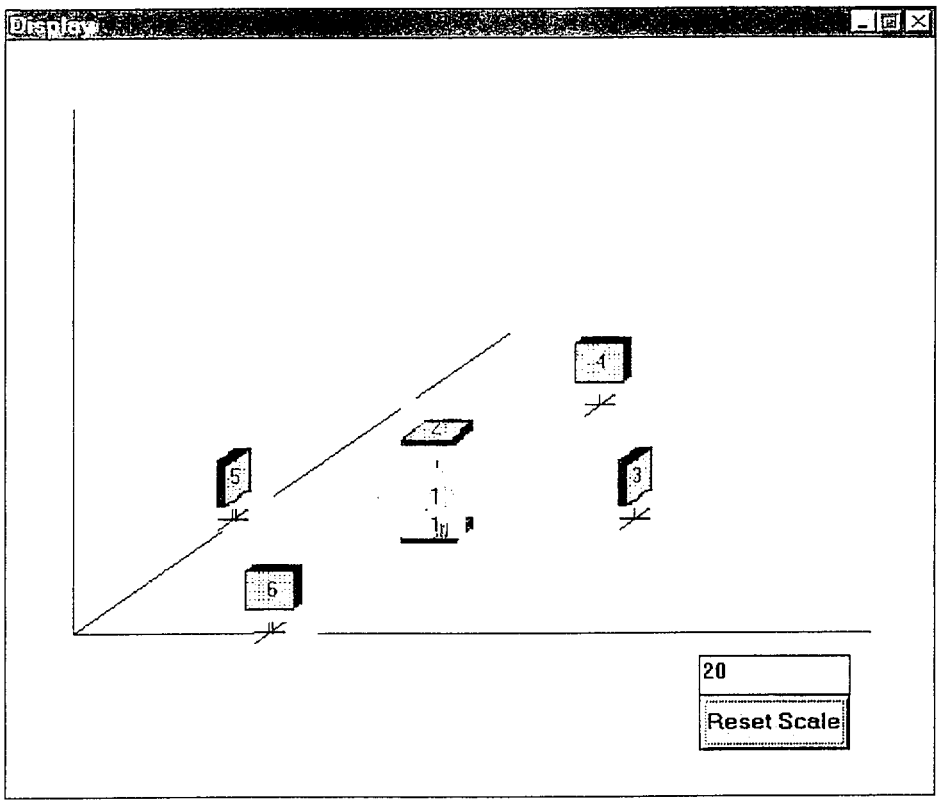


Figure 6-9
Display of Source and Receptor Locations in RESRAD-BUILD Model



Source	Description
1	Floor
2	Ceiling
3	East Wall
4	North Wall
5	West Wall
6	South Wall

Figure 6-10

Checks for Demolished Buildings

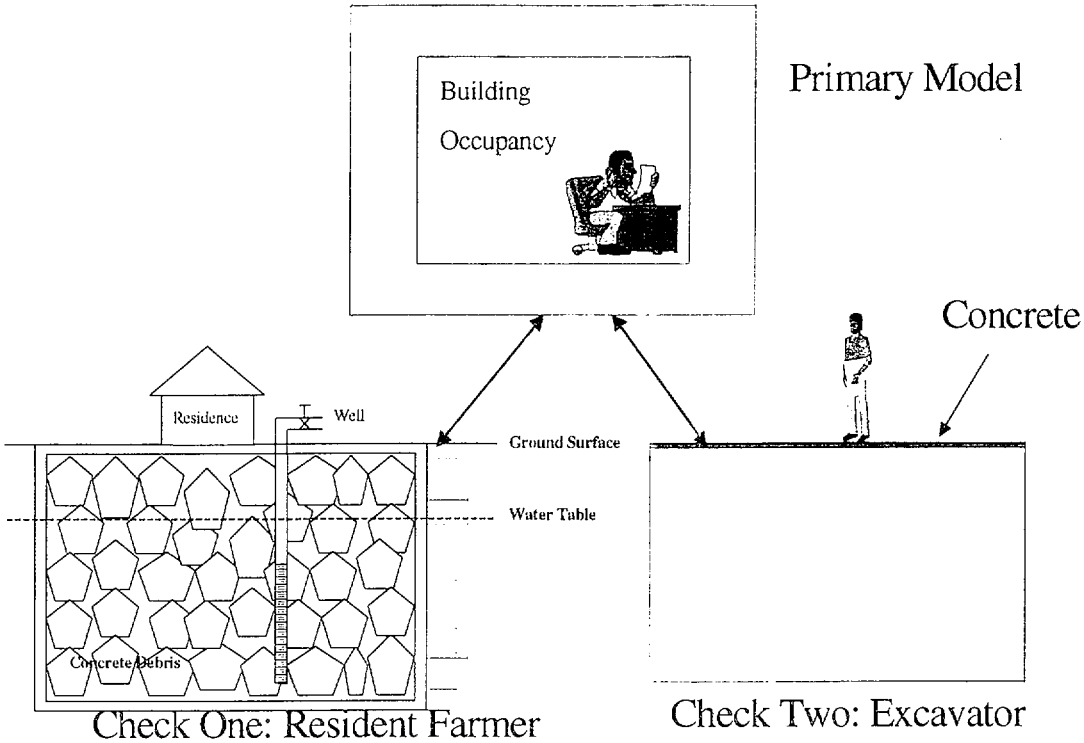


Figure 6-11
Conceptual Model for Concrete Debris Used as Backfill

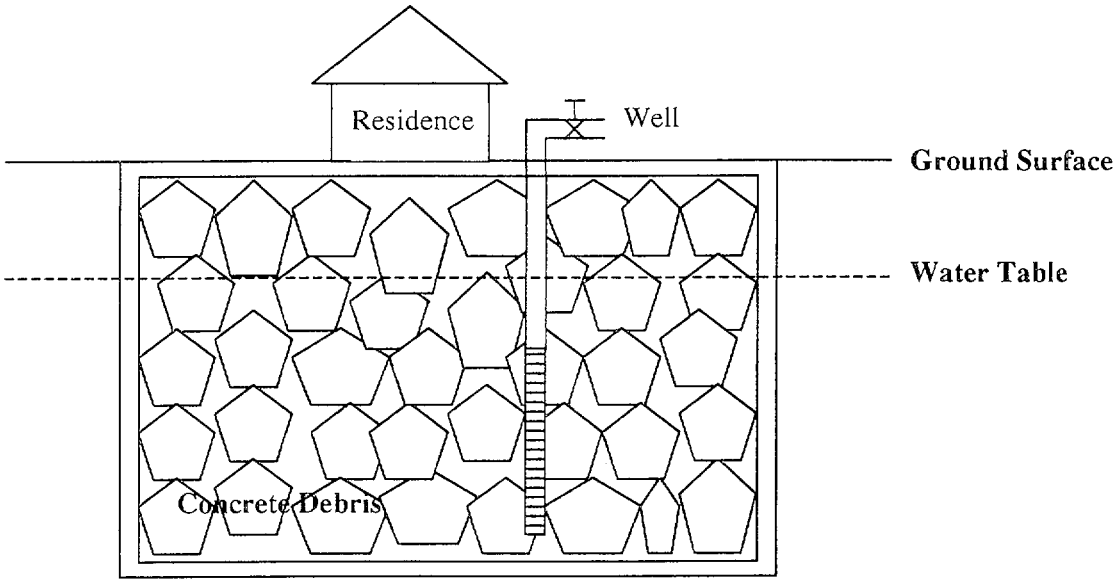


Figure 6-12
Excavator Scenario

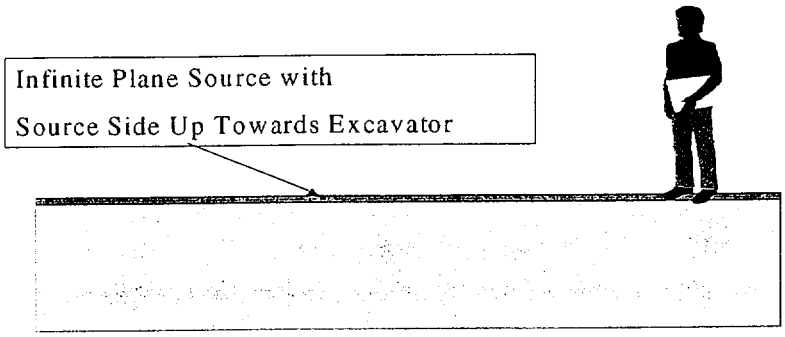
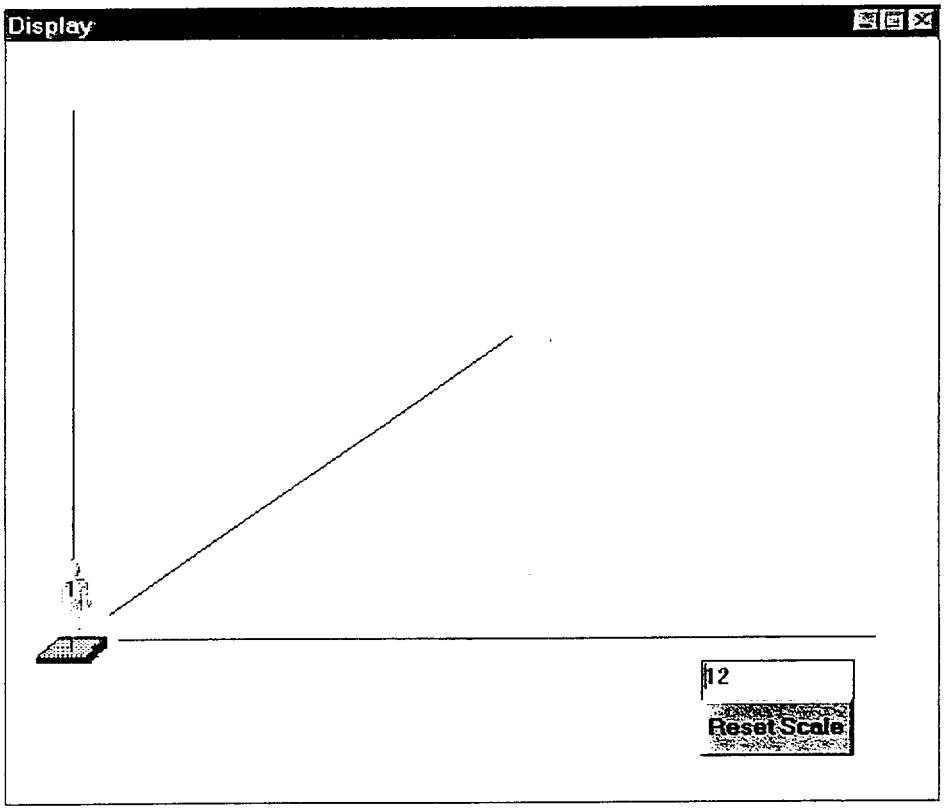


Figure 6-13
Display of Source and Receptor Locations in RESRAD-BUILD Model for Excavator Scenario



7 UPDATE OF SITE-SPECIFIC DECOMMISSIONING COSTS

7.1 Introduction

In accordance with 10CFR50.82(a)(9)(ii)(F) and Regulatory Guide 1.179, the site specific cost estimates and funding plans are provided. Regulatory Guide 1.179 discusses the details of the information to be presented.

The License Termination Plan (LTP) must:

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

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

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

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

During plant operations, CYAPCO sold the entire electrical output of the Haddam Neck Plant (HNP) to wholesale power purchase contracts (i.e., Power Contracts) with the ten New England utilities that collectively own 100% of the common equity of CYAPCO (the "Customers"). Over the HNP's operating life, CYAPCO recovered, and since the shutdown continues to recover, its costs of providing service (including the estimated costs of decommissioning HNP) through a formula rate set forth in its Power Contracts. Collections for decommissioning have been placed in a trust established under Connecticut law, with two funds—the Qualified Fund and the Non-Qualified Fund (the "Decommissioning Trust").

On December 26, 1996, CYAPCO submitted amendments to its Power Contracts (the "Amendatory Agreements") for filing (Reference 7-2). These Amendatory Agreements were executed to implement the decision to cease operations permanently. Also in that filing, CYAPCO proposed to increase its annual decommissioning charges in order to allow CYAPCO to recover the Company's estimate at that time of the cost to decommission HNP, $426.7 million (in 1996 dollars).

The cost estimate was prepared by Northeast Utilities Service Company (NUSCo) using the TLG Services, Inc., estimating model in accordance with 10 CFR 50.82(a)(8)(iii). The assumed method of decommissioning anticipated a prompt decommissioning technique commonly referred to as DECON. The Federal Energy Regulatory Commission (FERC) accepted the filing by its February 27, 1997, order

and set a hearing for the fall of 1997. In April of 1999, CYAPCO entered into a contract with Bechtel for decommissioning operation contractor services on a lump sump, turn key basis. Given the nature of this contract and the proprietary nature of the information providing the basis for the costs associated with the contract, detailed cost information is not provided within this License Termination Plan.

7.2 Decommissioning Cost Estimate

7.2.1 Cost Estimate Previously Docketed in Accordance with 10 CFR 50.82 and 10 CFR 50.75 Post Shutdown

A letter from CYAPCO to the USNRC dated August 22, 1997 (Reference 7-3) was docketed detailing the PSDAR. This submittal contained a site-specific decommissioning cost estimate. In accordance with 10CFR50.82(a)(8)(iii), CYAPCO submitted the site-specific decommissioning cost estimate to the USNRC in a letter dated August 25, 1998 (Reference 7-7). The estimate included with the PSDAR had been filed with the FERC prior to docketing. This estimate was the basis for which CYAPCO collected its rates and its anticipated decommissioning expenses. This cost estimate was prepared by NUSCo using the TLG model for estimating decommissioning costs. It was prepared in sufficient detail to identify an activity by activity work breakdown complete with costs for radioactive waste, utility labor, contractor labor, energy, materials and equipment.

In August of 1998, the presiding Administrative Law Judge (ALJ) in the FERC case issued an opinion to the FERC Commissioners that the estimate was "unreliable for rate-making" and recommendations were made identifying appropriate remedies based on the factual data submitted from CYAPCO and intervening parties. CYAPCO chose to prepare an updated estimate along with an Offer of Settlement with the intervening parties. These documents were submitted to the FERC for review and approval (see Appendix C).

With this settlement, CYAPCO has now agreed to accept most of the Department of Public Utility Control and the Office of Consumer Counsel's recommendations in the FERC proceeding, providing a number of benefits including:

- Reducing annual collections for decommissioning from $24.8 million to $16.7 million—a total reduction in the collections from ratepayers of $57 million through 2007.

- Replacing 1996 decommissioning costs estimates with fixed-price competitive bids for most of the decommissioning work, providing detailed, reliable estimates that form the basis for the settlement.

- Agreeing that CYAPCO will earn only a 6% return on its investment—a $29 million reduction from its earlier rates—effectively giving ratepayers the use of $110 million at an exceptionally low rate of interest.

- Reevaluating its decision to continue storing spent nuclear fuel in a wet pool. CYAPCO has now conducted more rigorous analyses and has concluded that dry storage is preferable.

- Agreeing that ratepayers should not pay spent fuel storage costs that are properly the U.S. Department of Energy's (DOE's) responsibility. CYAPCO has now brought and aggressively pursued litigation against the DOE to require the DOE to assume financial responsibility for spent fuel storage and has agreed to pass through any recovery after litigation expenses—as much as $130 million—to ratepayers. In addition, CYAPCO has agreed to use a portion of the previously collected spent fuel trust funds to pay storage costs instead of collecting more from ratepayers.

- Agreeing that its owners should bear the remedial costs for contaminated materials that were improperly released–$10 million to date.
- Conducting a comprehensive review of options to repower the site. CYAPCO has already realized $2 million from an option to repower the site and has agreed to flow this and any future benefits (as much as $10 million) through to ratepayers.
- Agreeing in the settlement that it will not charge ratepayers (an estimated $17 million) for disposing of high-level radioactive waste.
- Using an inflation rate of 3.35%, producing a savings for ratepayers of $12 million.
- Applying the decommissioning cost credits for the salvage value of plant equipment—$5 million to date—to reduce the decommissioning costs that ratepayers would otherwise pay.
- Including innovative incentive provisions that will encourage CYAPCO to complete decommissioning in the safest, most efficient manner and incentive provisions to reduce decommissioning costs.

7.2.2 Summary of the Site Specific Decommissioning Cost Estimate

Table 7-1 provides actual and projected decommissioning expenditure. With a pending FERC decision and numerous unresolved issues arising from the December 1996 decommissioning cost estimate, CYAPCO explored the concept of a Decommissioning Operation Contractor (DOC). In the Fall of 1998, CYAPCO produced a request for proposal for a DOC to perform a turnkey decommissioning on a firm fixed price contract. This contract was executed in April of 1999. The DOC contract, among other things, provides for completion of major decommissioning activities, construction of an Independent Spent Fuel Storage Installation, and site restoration by June 30, 2004.

The total cost of the remaining decommissioning work is the sum of the DOC contract and the costs for an oversight organization. The CYAPCO oversight organization mission is to oversee the work of the DOC; to safely store the spent nuclear fuel during decommissioning; to assure that all other Federal, State and Local regulatory requirements are met; and to act as an environmental steward for the Haddam Neck Plant site. The oversight costs include: oversight labor and benefits, insurance, spent fuel storage O&M costs, regulatory fees, legal fees, miscellaneous materials and other administrative and general costs.

Table 7-1
CYAPCO
Actual and Projected Decommissioning Expenditures
(Million $'s)

Cost Categories	Total
1997 Dismantlement and Decontamination	6.7
1998 Dismantlement and Decontamination	74.0
1999 Dismantlement and Decontamination	88.3
Subtotal 1997, 1998, and 1999	**169.0**
Remaining Decommissioning Costs (2000 $)	
Dismantlement and Decontamination	153.7
Radioactive Wastes Costs	78.6
Long Term Spent Fuel Storage	102.4
Site Restoration and License Termination	19.3
Subtotal 2000 through 2023	**354.0**
Total Decommissioning	**523.0**

Notes:

1. 1997, 1998, and 1999 represent nominal year dollars
2. Remaining Decommissioning Costs are in year 2000 dollars
3. Contingency has not been applied to "Remaining Costs" but is captured in the Offer of Settlement filed with the FERC.
4. 1998 and 1999 costs include spent fuel storage costs.

7.2.3 Dismantlement and Decontamination

The costs for the remaining dismantlement and decontamination activities include: utility oversight and DOC project management, labor to remove contaminated and non-contaminated systems, structures and components, disposal of non-radiological waste, soil and asphalt disposal, materials, equipment, fees and permits, salvage credits, and spent fuel island transition.

7.2.4 Radiological Waste Disposal

Radiological waste disposal includes: preparation, packaging, transportation and disposal of all forms of low level radioactive wastes. The large components from the nuclear steam supply system include: the four steam generators, pressurizer, reactor coolant pumps and piping, and the reactor vessel.

7.2.5 Long Term Spent Fuel Storage

In parallel with the final phase of decommissioning, an Independent Spent Fuel Storage Installation (ISFSI) will be constructed for the long-term storage of the spent nuclear fuel. Long term fuel storage encompasses the construction and capital cost of an ISFSI, and ISFSI operational and maintenance expenditures from mid-year 2004 through 2023 when the Department of Energy (DOE) is assumed to honor its contract obligations for the spent fuel. ISFSI operations and maintenance include: CYAPCO oversight labor and benefits, insurance, regulatory fees, legal fees, maintenance materials, and other administrative and general expenditures.

7.2.6 Site Restoration and License Termination

The current estimate for site restoration includes the cost to demolish the remaining buildings and structures, perform the final site survey and terminate the Part 50 license. This cost includes the costs associated with non-radiological remediation required by Federal and State agencies for such items as RCRA and TSCA closure, asbestos disposal, etc.

7.3 Decommissioning Funding

On July 18, 1990, CYAPCO submitted to the NRC a report as required by 10CFR50.75, indicating how reasonable assurance will be provided for funds to decommission the facility (Reference 7-4). The report described how CYAPCO has established an external sinking fund in 1984 to accumulate decommissioning funds. CYAPCO certified that each owner agreed to be financially responsible for its share of the decommissioning costs pursuant to the terms of the Power Contracts and Amendatory Agreements in accordance with the FERC regulations. These contracts have been filed with and approved by FERC. The Power Contracts and Amendatory Agreements were attached to the report.

On March 31, 2000, CYAPCO provided the most recent status report on the decommissioning fund to the NRC in accordance with 10CFR50.75 (Reference 7-5). This report restated the obligation that each wholesale power purchaser is responsible for its share of the facility decommissioning costs pursuant to the Power Contracts regardless of when the costs occur.

As a result of an interim FERC rate order associated with the December 24, 1996 rate filing dated, February 7, 1997, and effective March 1, 1997, the current collection is $24.8 million annually. In the current proposed Offer of Settlement (Reference 7-6), the collections would become $16.7 million annually.

Under the proposed Offer of Settlement agreement, CYAPCO is required to file with the FERC no later than July 1, 2004, for the purpose of examining any further rate adjustments and is not limited to the future cost of spent fuel storage.

All decommissioning activities are scheduled for completion by year end 2004. Assuming a SAFSTOR condition, decommissioning trust fund expenditures would be minimized to fuel storage activities only. Therefore, CYAPCO forecasts sufficient funding will exist should a SAFSTOR condition occur during the period between 2000 and 2004.

All spent fuel is expected to be transferred to the completed ISFSI and the existing Spent Fuel Pool Building is expected to be decommissioned by the end of 2004. The long term spent fuel storage costs after 2004 consist of the operation, and maintenance of the ISFSI, as well as decommissioning of the ISFSI. As shown in Column 5 of Table 7-2, sufficient funding will exist, based on CYAPCO's assumption that the DOE will assume responsibility to complete spent fuel storage and removal by 2023.

Finally, as demonstrated in Table 7-2, CYAPCO maintains a positive decommissioning trust fund balance for all $523M of decommissioning expenditures of Table 7-1, including spending the full contingency. Pursuant to 10CFR50.75 and 10CFR50.82 regulations, CYAPCO has demonstrated a financial plan which includes adequate reserves for the entire decommissioning and ISFSI-related costs, which therefore meet the requirements for decommissioning costs associated with decommissioning and dismantlement as defined by these regulations.

Table 7-2
Decommissioning/Spent Fuel Trust Analyses[1]
(in Thousands of Dollars)

	(Column 1)	(Column 2)	(Column 3)	(Column 4)	(Column 5)
Period Ending	**Decomm. Contributions**	**SFT Contributions**	**Decomm. Expenses**	**After Tax/Earnings and Adjust**	**Total Decomm Trust**
31-Dec-98	0		0	0	0
31-Dec-99	0		0	0	$335,949
31-Dec-00	$20,771	$14,882	($131,671)	$14,653	254,583
31-Dec-01	16,742		(91,719)	22,893	202,499
31-Dec-02	16,742		(61,761)	42,449	199,929
31-Dec-03	16,742		(28,993)	14,870	202,548
31-Dec-04	16,742		(22,806)[2]	14,740	211,224
31-Dec-05	16,742		(10,365)	15,771	233,372
31-Dec-06	16,742		(5,617)	17,417	261,914
31-Dec-07	8,371		(7,032)	19,014	282,267
31-Dec-08			(7,452)	20,138	294,952
31-Dec-09			(7,015)	20,917	308,854
31-Dec-10			(7,154)	21,989	323,689
31-Dec-11			(7,421)	23,123	339,570
31-Dec-12			(7,376)	24,273	356,468
31-Dec-13			(7,508)	25,496	374,456
31-Dec-14			(9,269)	26,745	391,931
31-Dec-15			(7,831)	27,815	411,914
31-Dec-16			(7,956)	26,823	430,781
31-Dec-17			(8,177)	25,027	447,630
31-Dec-18			(8,313)	25,102	464,419
31-Dec-19			(8,860)	26,134	481,693
31-Dec-20			(8,648)	27,223	500,269
31-Dec-21			(492,624)[3]	14,380	22,025
31-Dec-22			(12,779)	529	9,776
31-Dec-23			(9,941)	166	0
31-Dec-24				0	
Totals	$129,594	$14,882	($978,111)	497,686	
				Total Funds Available	**$0**

[1]Merger of the DOE Spent Fuel Trust and Nuclear Decommissioning Trust
[2]Decommissioning completed except for ISFSI
[3]Assumed DOE obligation paid to DOE in 2021 for spent fuel removal obligation, with fuel removal from ISFSI completed by 2023

7.4 References

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

7-2 Letter CYAPCO to FERC, "Amendatory Agreements," dated December 26, 1996.

7-3 Letter CYAPCO to USNRC, "Post Shutdown Decommissioning Activities Report," dated August 22, 1997.

7-4 Letter CYAPCO to USNRC, "Decommissioning Financial Assurance Certification Report," dated July 18, 1990.

7-5 Letter CYAPCO to USNRC, "Decommissioning Funding Assurance," dated March 31, 2000.

7-6 Letter CYAPCO to FERC, "Offer of Settlement," dated April 7, 2000, as supplemented by letter dated April 27, 2000.

7-7 Letter CYAPCO to USNRC, "Site-Specific Decommissioning Cost Estimate," dated August 25, 1998.

8 SUPPLEMENT TO THE ENVIRONMENTAL REPORT

8.1 Introduction

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

This conclusion was based on the following considerations:

1. The postulated impacts associated with the method chosen, DECON, have already been considered in the FGEIS.

2. There are no unique aspects of the plant or decommissioning techniques to be utilized that would invalidate the conclusions reached in the FGEIS.

3. The methods to be employed to dismantle and decontaminate the site are standard construction based techniques fully considered in the FGEIS.

4. The site-specific person-rem estimate for all decommissioning activities has been calculated using methods similar to and consistent with the FGEIS.

Specifically, this review concludes that the HNP decommissioning will result in generally positive environmental effects, in that:

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

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

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

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

- Hazardous materials and chemicals will be removed.

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

- Decommissioning will not adversely affect any endangered or threatened species on the site.

- Historic remains of the Venture Smith homesite have not been disturbed.

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

- No site specific factors pertaining to the HNP would alter the conclusions of the FGEIS.
- Radiation dose to the public will be minimal.
- Decommissioning is not an imminent health or safety problem and will generally have a positive environmental impact.
- The total occupational radiation exposure (excluding public and transportation dose) impact for the proposed decommissioning activities has been estimated in the PSDAR at approximately 935 person-rem, which is less than the 1,115 person-rem exposure estimate of the FGEIS for a PWR. This estimate is based primarily on January 1997 plant dose rate surveys with no credit for (1) decay in place of radionuclides (such as Co-60), (2) sequenced removal of higher dose rate components first, (3) aggressive ALARA program initiatives, (4) increased worker efficiency with experience, or (5) smaller scale decontamination initiatives.
- The release of soil and facility structures will be performed in accordance with MARSSIM. The methods and procedures described in MARSSIM have been determined by the NRC to be acceptable and are not inimical to the health and safety of the public. Therefore, no significant environmental impacts are anticipated from the release of site soils and structures.
- The release of facility non-structural components will be performed in accordance with Regulatory Guide 1.86. The methods and procedures described in this document have been determined by the NRC to be acceptable and are not inimical to the health and safety of the public. Therefore, no significant environmental impacts are anticipated from the release of non-structural components.
- PSDAR radiation exposure due to transportation of radioactive waste (includes both occupational and offsite radiation exposures) has been estimated. The occupational exposure due to transportation is approximately 61 person-rem. The cumulative radiation exposure to on-lookers and the general public due to transportation is approximately 11 person-rem. These values are bounded by the FGEIS values of 100 person-rem for transportation occupational exposure and 21 person-rem for the general public exposure.
- Radiation exposure to offsite individuals for expected conditions, or from postulated accidents, is bounded by the Environmental Protection Agency's Protective Action Guides and NRC regulations. Doses due to the release of radionuclides in effluents will be a small fraction of the allowable limits.
- No significant impacts are expected from the disposal of low level radioactive waste. The total volume of HNP low level radioactive waste for disposal has been estimated at 283,117 cubic feet, which is well bounded by the FGEIS volume of 647,600 cubic feet. The actual HNP volume may be further reduced by additional utilization of volume reduction techniques.
- The non-radiological environmental impacts from decommissioning are temporary and are not significant. The largest occupational risk associated with decommissioning HNP is related to the risk of industrial accidents. The primary environmental effects are short-term, small increases in noise levels and dust in the immediate vicinity of the site, and truck traffic to and from the site for hauling

equipment and waste. No significant socioeconomic impacts, other than those associated with cessation of operation (loss of jobs and taxes), or impacts to local culture, terrestrial or aquatic resources have been identified.

- Given the low level of contamination and the expected volume of waste, disposal of low level radioactive waste offsite in a timely manner should be possible. If for any reason some portion of these waste needs to be stored temporarily onsite, adequate space exists. No significant environmental impacts are anticipated from temporary onsite storage, because CYAPCO will ensure compliance with all applicable federal, state, and local regulations.

8.2 Environmental Impacts of Termination Activities Not Previously Addressed

The total occupational radiation exposure (excluding public and transportation doses) as given in the PSDAR was given as approximately 935 person-rem, which was stated to be less than the 1,115 person-rem exposure estimated in Section 4.3.1 of the FGEIS for a PWR. The estimate in the PSDAR was based upon January 1997 plant dose rate surveys with no credit for: decay in place of radionuclides, sequenced removal of higher dose rate components first, aggressive ALARA program initiatives, increases in worker efficiencies with experience, and smaller scale decontamination initiatives.

The public exposure due to radiological effluents continue to remain well below the limits of 10CFR20 and the ALARA dose objectives of 10CFR50, Appendix I. This conclusion is supported by the data submitted to the NRC in the 1999 Annual Radioactive Effluent Report for HNP in which individual doses to members of the public are calculated for liquid and gaseous effluents.

The PSDAR estimated the amount of low level radioactive waste to be approximately 283,117 ft^3, well within the FGEIS estimated volume of 647,600 ft^3. Table 3-3 estimates the total low level radioactive waste burial quantity as a result of HNP decommissioning and currently projects volumes less than the PSDAR estimate. Thus, these values and the waste volume that remains at the HNP are bounded by the amount assumed in the FGEIS.

The PSDAR assumed that spent fuel would continue to be stored in the spent fuel pool (wet storage). As previously noted, at this time it is anticipated that spent fuel will be moved to an ISFSI constructed onsite. The environmental impacts of transporting spent fuel to and storing the spent fuel in the ISFSI will be addressed in activities associated with licensing the ISFSI.

As previously discussed in Section 6, the DCGLs for site buildings are calculated using the building occupancy scenario as the primary modeling scenario. Because use of the demolished, decontaminated buildings as backfill is being considered, additional modeling scenarios have been used, as discussed in Section 6.5 (i.e., resident farmer and excavator for concrete debris). Buildings, decontaminated to or below the DCGLs (and considering ALARA) could be allowed to remain standing at the time of license termination. After decontamination, these buildings could be demolished and the debris dispositioned in a number of different manners. Consideration of the building occupancy scenario (as well as other scenarios) in determining the DCGL is compatible with the information in SECY 00-41 (Reference 8-5). SECY 00-41 concludes that, although the GEIS analyses do not specifically address the use of demolition debris as backfill, the building occupancy and resident farmer scenarios, as well as assumptions used in the GEIS to estimate public dose are sufficiently conservative to bound such a condition.

Abandoning concrete foundations in place, and filling holes and basements with concrete debris fill will not cause any significant adverse impacts to ground water. The CT DEP recognizes concrete debris as "clean fill" and, as such, does not pose a pollution threat to ground or surface waters (Reference 8-6).

Ground water monitoring data acquired from onsite wells give no indication that the existing concrete foundations have adversely impacted ground water quality.

Another important aspect of HNP site cleanup involves the assessment and remediation, as appropriate, of non-radiological hazards identified at the site. Planning and implementation of these activities will be performed in accordance with applicable Federal, state and local environmental regulations. An Environmental Closure Plan (ECP) is being developed to describe the processes and methodologies to be used for this cleanup. The ECP will identify non-radiological sampling and analysis plan elements, proposed cleanup criteria, detailed work package development and implementation steps, and the reporting requirements to be met in obtaining DEP release of the site for non-radiological purposes. CYAPCO will work with the Connecticut Department of Environmental Protection (CT DEP) and the US EPA during ECP development to ensure that the approach and strategy for the non-radiological site closure is acceptable. Certain elements of the non-radiological cleanup will be integrated with similar activities for the radiological cleanup of the site. Examples are site survey area designation, sample collection and control processes, and development of selected procedures that serve common purposes.

CYAPCO has reviewed the Natural Diversity Data Base (Reference 8-7) compiled by the CT DEP to ensure that ongoing decommissioning activities do not impact any critical biological resources that are located on the site. Field walkdowns are performed in areas of the site undergoing decommissioning or remediation to verify that endangered or threatened species are not present.

CYAPCO is also working with the Connecticut Historical Commission, the Thomas J. Dodd Research Center of the University of Connecticut, and the National Park Service to develop documentation on the historic and technological significance of the HNP. Documentation is being prepared to the standards of the Historic American Engineering Record and will be provided to the State Historical Commission for future use. Archeologists from the State Historical Commission are also working with CYAPCO to preserve the Venture Smith home site located on the property. This site is being considered for the National Register of Historical Places (Reference 8-8).

8.3 Conclusions

As evaluated above, there is no new information or significant environmental change associated with license termination activities with respect to the decommissioning activities previously evaluated. This License Termination Plan does not describe any different or additional plant activities beyond those that already may be conducted as described in the HNP PSDAR and the UFSAR. Therefore, the environmental impacts associated with the license termination activities described herein are bounded by the previously approved environmental assessment as referenced above.

8.4 References

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

8-2 NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," August 1988.

8-3 USNRC, Final Environmental Statement, Haddam Neck (Connecticut Yankee) Nuclear Power Plant, Docket No. 50-213, October 1973.

8-4 Letter, USNRC to CYAPCO, "Environmental Assessment for Proposed License Extension," dated November 23, 1987.

8-5 SECY-00-41, "Use of Rubblized Concrete Dismantlement to Address 10 CFR Part 20, Subpart E, Radiological Criteria for License Termination, " dated February 14, 2000.

8-6 Connecticut General Statutes, 22a, "Environmental Protection", Section 209, "Definitions."

8-7 CTDEP to CYAPCO letter, "Natural Diversity Data Base," April 24, 2000.

8-8 Letter, CT Historical Commission to NUSCO, "Connecticut Yankee Nuclear Facility," dated June 1, 1998,

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Appendix A
Acronym List

ALARA	As Low As Reasonably Achievable
AO	Abnormal Occurrence
bgs	Below Ground Surface
CT DEP	Connecticut Department of Environmental Protection
CYAPCO	Connecticut Yankee Atomic Power Company
DCE	Decommissioning Cost Estimate
DCGL	Derived Concentration Guideline Level
DEP	[Connecticut] Department of Environmental Protection
DHEC	Department of Health and Environmental Control
DOC	Decommissioning Operations Contractor
DOE	Department of Energy
DOT	Department of Transportation
EMC	Elevated Measurement Comparison
EPA	Environmental Protection Agency
FERC	Federal Energy Regulatory Commission
FGEIS	Final Generic Environmental Impact Statement
FSS	Final Status Survey
GTCC	Greater than Class C [Waste]
HEPA	High Efficiency Particulate Air
HNP	Haddam Neck Plant
HSA	Historical Site Assessment
ISFSI	Independent Spent Fuel Storage Installation
LBGR	Lower Bound Grey Region
LLW	Low Level Waste
LTP	License Termination Plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
msl	Mean Sea Level
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PAB	Plant Auxiliary Building
PIR	Problem Investigation Report
PSDAR	Post-Shutdown Decommissioning Activities Report
QA	Quality Assurance
QAP	Quality Assurance Program
QAPP	Quality Assurance Program Plan
QC	Quality Control
RCA	Radiologically Controlled Area
RCP	Reactor Coolant Pump
RCRA	Resource Conservation and Recovery Act
RCS	Reactor Coolant System
REMODCM	Radioactive Effluents Monitoring Offsite Dose Calculation Manual
RESRAD	RESidual RADioactivity [Computer Code]
RPV	Reactor Pressure Vessel
SG	Steam Generator
SGLA	Steam Generator Lower Assembly
SSCs	Structures, Systems, and Components
TEDE	Total Effective Dose Equivalent
TSCA	Toxic Substances Control Act
UFSAR	Updated Final Safety Analysis Report
WRS	Wilcoxon Rank Sum [test]

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Appendix B
ALARA Evaluations

B.1 Determining Remediation Levels

Per Draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination," the residual radioactivity level that is ALARA is the concentration, Conc, at which the benefit from remediation equals the cost of remediation. If the total remediation cost, $Cost_T$, is set equal to the present worth of the collective doses averted, the ratio of the concentration (Conc) to the $DCGL_W$ is as follows:

$$RL = \frac{Conc}{DCGL_w} = \frac{Cost_T}{\$2000 \times P_D \times 0.025 \times F \times A} \times \frac{r \times \lambda}{1 - e^{-(r+\lambda)N}} \qquad \text{(Equation B-1)}$$

Where:

RL	=	remediation level, as a fraction of $DCGL_W$
Conc	=	average concentration of residual activity in the area being evaluated
$DCGL_w$	=	derived concentration guideline equivalent to the average concentration of residual radioactivity that would give a dose of 25 mrem/yr to the average member of the critical group
$Cost_T$	=	total cost of remediation action, in dollars
\$2000	=	monetary value of one person-rem averted (Table 3.1, DG-4006)
P_D	=	population density for critical group scenario, in people/m^2
0.025	=	annual dose to average member of critical group from residual radioactivity at $DGCL_W$ concentration, in rem/yr
A	=	area being evaluated, in m^2
F	=	removable fraction for remediation action evaluated
r	=	monetary discount rate, in yr^{-1}
λ	=	radiological decay constant, in yr^{-1}
N	=	number of years over which the collective dose is calculated

Acceptable values for population density (P_D), monetary discount rate (r), and the number of years over which the collective dose is calculated (N) are given in Table 3.1 of DG-4006 and are given below:

Parameter	Acceptable Value	
	Building	Land
P_D	0.09 person/m^2	0.0004 person/m^2
r	0.07 per year	0.03 per year
N	70 years	1000 years

The development of values for the equation parameters of total Cost ($Cost_T$), and removable fraction for remediation action being evaluated, F, are described in Sections B.1.1 and B.1.2. Where values other than those in the table above or in Section 4.2.3 are used, justification is provided.

B.1.1 Calculation of Total Cost

Calculations of total cost generally include the monetary costs of:

(1) The remediation action being evaluated ($Cost_R$)
(2) Transportation and disposal of wastes generated ($Cost_{WD}$)
(3) Workplace accidents that occur because of the remediation action ($Cost_{ACC}$)
(4) Traffic fatalities resulting from transporting the waste generated by the action ($Cost_{TF}$)
(5) Doses received by workers performing the remediation action ($Cost_{WDose}$)
(6) Doses to the public from excavation, transportation, and disposal of the waste ($Cost_{PDose}$)

Thus,

$$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + Cost_{TF} + Cost_{WDose} + Cost_{PDose} \qquad \text{(Equation B-2)}$$

Other monetary costs may be included as appropriate for the specific situation.

The cost of waste transport and disposal, $Cost_{WD}$, is calculated using the following equation:

$$Cost_{WD} = V_A \times Cost_V \qquad \text{(Equation B-3)}$$

Where:

V_A = volume of remediation waste produced (in m^3)
$Cost_V$ = cost of waste disposal, in $\$/m^3$

The cost of workplace accidents, $Cost_{ACC}$, is calculated using the following equation:

$$Cost_{ACC} = \$3{,}000{,}000 \times F_W \times T_A \qquad \text{(Equation B-4)}$$

Where:

$3,000,000 = monetary value of a fatality equivalent to $2000 per person-rem (DG-4006, p. 31)
F_W = workplace fatality rate, in fatalities per hour worked, or 4.2×10^{-8}/hr (Table 3.1, DG-4006)
T_A = worker time required for remediation, in worker-hours

The cost of traffic fatalities incurred during the shipment of waste, $Cost_{TF}$, is calculated using the following equation:

$$Cost_{TF} = \frac{\$3{,}000{,}000 \times V_A \times F_T \times D_T}{V_{ship}} \qquad \text{(Equation B-5)}$$

Where,

$3,000,000 = monetary value of a fatality equivalent to $2000 per person-rem (DG-4006, p. 31)
V_A = Volume of waste produced, in m^3

F_T	=	fatality rate per kilometer traveled, in fatalities/km, or 3.8 x 10^{-8}/km (Table 3.1, DG-4006)
D_T	=	distance traveled, in km
V_{ship}	=	volume of a truck shipment, in m^3, or 13.6 m^3 (Table 3.1, DG-4006)

The cost of remediation worker dose, $Cost_{WDose}$, is calculated using the following equation:

$$Cost_{WDose} = \$2000 \times D_R \times T_R \qquad \text{(Equation B-6)}$$

Where,

$2,000	=	monetary value of person-rem averted (Table 3.1, DG-4006)
D_R	=	total effective dose equivalent to remediation workers, in rem/hr
T_R	=	time worked to remediate area, in person-hours

B.1.2 Determination Of Remediation Action Effectiveness

The remediation action effectiveness, F, is the fraction of the residual radioactivity removed by the remediation action. It is determined by collecting and analyzing pre- and post-remediation measurements in the area in which the remediation action is performed. A sufficient number of measurements are made to establish a consistent value.

B.2 ALARA Evaluation

When dismantlement actions are completed, residual radioactivity may remain. 10CFR20.1402 requires assurance that residual radioactivity has been reduced to levels that are ALARA. For evaluations prior to additional remediation actions, the ALARA analysis for data evaluation will be performed using data from operational Radiation Protection surveys in accordance with Regulatory Guide DG-4006 and will take into account:

- Radiation doses and environmental impacts for the decommissioning process and from the residual radiation remaining on site after the completion of decommissioning.
- Other costs and risks associated with the decontamination and decommissioning of the site.

Once the total cost, $Cost_T$, for a remediation action has been calculated, a remediation level, expressed as a fraction of a $DCGL_W$, can be determined and the ALARA evaluation can be performed using the previously presented equations.

As discussed above this evaluation determines the point at which remediation is cost beneficial and then compares existing residual radioactivity levels to that remediation level. When the residual radioactivity is in excess of the calculated remediation level, the remediation action is considered to be cost beneficial and the action will be taken. If residual activity is below the remediation level, the ALARA criterion is considered to be met already and no additional remedial action is required to be performed.

ALARA evaluations will be performed when justification is needed for not applying a remediation action to an area. This is consistent with the recommendations provided in Regulatory Guide DG-4006. As appropriate, the radiation survey report will appropriately document that all concentrations in the survey

unit are below the concentration remediation level. As previously discussed, if the decision to perform a given remediation action has been made, then the activity does not require an ALARA justification.

The remediation levels calculated do not represent activity values that cannot be exceeded, as they only represent the threshold at which a remediation action is cost beneficial. As previously noted, the ALARA criteria is met by performing the remediation action, not by achieving results below a specific remediation level. An ALARA analysis ensures that the efforts to remove residual contamination are commensurate with the risk that exists with leaving the residual contamination in place.

B.3 Example

The following example is one that was provided in DG-4006. The values for the cost of the remediation activity and the remediation action effectiveness were those presented in DG-4006. At the time that an actual ALARA evaluation is performed, site specific costs and remediation action effectiveness will be used.

The following example considers a building with residual Cs-137 radioactivity (λ=0.023/yr). The remediation being evaluated is the use of a scabbling tool that removes the top 1/8 inch of concrete. The estimated total cost of the scabbling is $5000 for the 100 m^2 floor, and the estimated remediation action effectiveness, F = 1 (i.e., the remediation action is assumed to remove all of the residual radioactivity). Using these values in Equation B-1 gives:

$$RL = \frac{\$5000}{\$2000 \times 1 \times 0.025 \times 0.09 \times 100} \times \frac{0.07 + 0.023}{1 + e^{-(0.07+0.023)70}}$$

$$RL = 0.97$$

Thus, the determination to take the additional remediation would be based on an RL of 0.97. If the residual radioactivity on the building floor could be demonstrated to be less than 0.97 DCGL, then scabbling would not be necessary.

UNITED STATES OF AMERICA
FEDERAL ENERGY REGULATORY COMMISSION

CONNECTICUT YANKEE ATOMIC POWER COMPANY	))	DOCKET NO. ER97-913-000

APRIL 7, 2000

OFFER OF SETTLEMENT

This Offer of Settlement ("Offer") is submitted, pursuant to Rule 602 of the Commission's Rules of Practice and Procedure, jointly by Connecticut Yankee Atomic Power Company ("CY", "Connecticut Yankee" or the "Company") and the following intervenors in this proceeding: the Connecticut Department of Public Utility Control ("DPUC") and the Connecticut Office of Consumer Counsel ("OCC")(collectively, the "Settling Parties").

If accepted and approved by the Commission, this Offer of Settlement will resolve, as among CY and the other Settling Parties, all issues that were or could have been raised in CY's contract amendment filing in Docket No. ER97-913-000, with the single exception noted in Part II.A.(8) below.

The signatories to this Offer of Settlement are:

1. Connecticut Yankee Atomic Power Co.
2. Connecticut Department of Public Utility Control.
3. Connecticut Office of Consumer Counsel

All issues that the Commission may set for hearing will remain unresolved for any parties contesting this Offer of Settlement. Accordingly, CY and the Settling Parties request that the

Commission approve the Offer of Settlement for all non-contesting parties and sever contesting parties, if any, to pursue their claims separately.

I. BACKGROUND

CY owns and operated a single 583 MWe nuclear power generating station located in Haddam Neck, Connecticut (the "Plant"). On December 4, 1996, CY's Board of Directors unanimously voted to cease operations at the plant permanently.

CY sold the entire electrical output of the Plant pursuant to wholesale power purchase contracts[1] with the ten New England utilities that collectively own 100% of the common equity of CY (the "Customers"). Over the Plant's operating life, CY recovered, and since the shutdown continues to recover, its costs of providing service (including the estimated costs of decommissioning the Plant) through a formula rate set forth in its Power Contracts. Collections for decommissioning have been placed in a trust established under Connecticut law, with two funds–the Qualified Fund and the Non-Qualified Fund (the "Decommissioning Trust") On December 26, 1996, CY submitted amendments to its Power Contracts (the "Amendatory Agreements") for filing. These Amendatory Agreements were executed to implement the decision to cease operations permanently. Also in that filing, CY proposed to increase its annual decommissioning charges in order to allow CY to recover the Company's estimate at that time of the cost to decommission the Plant, $426.7 million (1996 Dollars). The proposed rates included decommissioning charges of $24.8 million per year through June 29, 2007 (compared to 1995 Commission-approved rates of $15.2 million for 1997). By its February 27, 1997 Order, the

[1] These power purchase contracts (the "Power Contracts"), as supplemented and amended, are on file with the Commission.

Commission accepted the filing, suspended rates for a nominal period. and set them for a hearing.

After two weeks of contested hearings in October and November 1997. Administrative Law Judge William J. Cowan issued his Initial Decision on August 31. 1998. The Commission has not yet issued a final decision.

The decommissioning cost estimate in CY's 1996 filing contemplated that the Company would manage and/or conduct the decommissioning operations and activities itself. However, in late 1998 and early 1999, CY undertook a competitive bid process to engage a Decommissioning Operations Contractor ("DOC") to perform the major decommissioning and dismantlement operations at the Plant on a fixed-price, turnkey basis. As a result of that process, CY selected Bechtel Power Corporation ("Bechtel") as the DOC and executed a contract with Bechtel on April 3, 1999 (the "DOC Contract"). The DOC Contract. among other things, provides for construction of an Independent Spent Fuel Storage Installation ("ISFSI") on-site and completion of major decommissioning activities and site restoration by June 30. 2004. In addition, the DOC Contract fixes certain costs that had been estimated in the prior decommissioning cost estimate. In agreeing to the terms of this Offer of Settlement, the Settling Parties have taken into account the DOC process and costs, as well as CY's most recent estimate of DOC oversight and other decommissioning costs to be incurred by CY.

II. TERMS OF SETTLEMENT

Connecticut Yankee and the Settling Parties have agreed upon, or do not contest, the following terms and conditions, in full and final settlement of all issues raised in Docket No. ER97-913-000, with the single exception noted in Part II.A(8).

A. Non-Decommissioning Issues.

(1) Return on Equity. Commencing as of the Effective Date hereof (defined in Part II.D below), Connecticut Yankee shall change the methodology employed to calculate return on equity. The net unit investment methodology currently employed by CY will be abandoned and monthly collections for return on equity shall be calculated by multiplying the remaining equity balance by the return on equity (converted to a monthly rate) allowed herein. As of the Effective Date, the allowed return on equity to be used in this calculation shall be 6.0%. There shall be no change in the filed rate of return on equity for periods prior to the Effective Date or in the methodology for calculating the collections prior to the Effective date. The prospective return on equity incorporates the effect of rates collected through the Effective Date, and, accordingly, it is agreed there shall be no refunds of any amounts collected prior to the Effective Date. The bills under the Power Contracts shall be modified to reflect the change in methodology as of the Effective Date. Accordingly, all interest income earned on cash balances as well as any interest expense incurred will become part of the formula rate and become part of the charges and/or credits rendered on the monthly power bill. With respect to Settling Parties other than DPUC, the provisions of Part II.A(8) shall apply.

(2) Gain on Sale, Lease or Other Disposal of Land. The Settling Parties acknowledge that, as part of Connecticut Yankee's efforts to mitigate shutdown and decommissioning costs, Connecticut Yankee is seeking to sell, lease or otherwise dispose of all or a portion of its land located in Haddam Neck, Connecticut. Connecticut Yankee and the Settling Parties agree that, if at any time between the Effective Date and the later of (1) June 29, 2007 or (2) the dissolution of Connecticut Yankee, Connecticut Yankee shall sell, lease or

otherwise dispose of all or a portion of its land in Haddam Neck, Connecticut. Connecticut Yankee shall flow through to its Customers, in accordance with its formula rate, any net-of-tax gains in excess of escalated book value for such land, for the benefit of its Customers, and that none of the gain in excess of escalated book value (future escalation to be calculated on the basis of annual increases in the Gross Domestic Product) from such sale, lease or other disposition shall inure to the benefit of the shareholders of CY. Such flow-through of gain from the sale, lease or other disposition of land shall be effective whether the sale, lease or other disposition is for the purpose of re-powering at the Plant in Haddam Neck or for any other purpose. For purposes of determining gain on sale, lease or disposition of land, the Settling Parties acknowledge and agree that the escalated book value of said land as of January 1, 2000, is $2,530,830.

(3) Recovery of Unamortized Investment. The Settling Parties do not contest CY's ability to recover, pursuant to its revised rate schedules, all unamortized investment (including fuel) in the Plant. With respect to Settling Parties other than DPUC, however, the provisions of Part II.A(8) shall apply.

(4) Amendatory Agreements.

(A) Except as noted in Part II.A(8), the Settling Parties do not contest the effectiveness of the Amendatory Agreements submitted for approval in Docket No. ER97-913-000; provided that Connecticut Yankee shall make an informational filing with the Commission in advance of any acceleration of recovery of unamortized investment pursuant to the terms and conditions of Section 3, Part D(iii) of the Amendatory Agreements. The Settling Parties reserve the right to commence a proceeding pursuant to Section 206 of the Federal Power Act to challenge such acceleration.

(B) If the Commission issues an Order approving this Offer in its entirety without change, Connecticut Yankee will execute and file further amendments to its Power Contracts as necessary to effectuate the terms and conditions of the Offer. The Settling Parties agree not to contest such new Amendatory Agreements.

(5) DOE Litigation and Other Spent Fuel Storage Issues.

(A) The Settling Parties agree that the storage and disposal of spent nuclear fuel and high-level waste, including Greater-Than-Class-C ("GTCC") waste, are properly the responsibility of the U.S. Department of Energy ("DOE"). Connecticut Yankee shall use its best efforts to secure funding or reimbursement for storage and disposal of spent nuclear fuel and high-level radioactive waste, including GTCC waste, from the DOE. Connecticut Yankee is currently involved in litigation with the DOE to recover costs associated with the DOE's delay in taking possession of Connecticut Yankee's spent nuclear fuel. Connecticut Yankee agrees to continue to pursue its available remedies against the DOE, including, without limitation, orders requiring the DOE to assume its obligations and take prompt possession of Connecticut Yankee's spent nuclear fuel and high-level waste, including GTCC waste; recovery of damages caused by the DOE's delay; other remedy or remedies, legal, equitable, or administrative; or reasonable settlement of such claims. Connecticut Yankee further agrees that any funds recovered from the DOE or any savings resulting from the DOE's performance of its obligations, net of reasonable litigation costs, shall flow through fully to the benefit of Connecticut Yankee's Customers. The Settling Parties agree that the litigation and other claim-related expenses incurred to date by CY in pursuit of claims against the DOE are and have been reasonable. Connecticut Yankee agrees to advise the Settling Parties routinely on the status of pending litigation or potential settlement with the DOE, and to meet with such parties for this purpose at

their request, provided, however, that Connecticut Yankee shall have no obligation to disclose privileged or confidential information to any Settling Party.

(B) Connecticut Yankee and the Settling Parties agree that any monetary damages awarded as a consequence of the DOE litigation, or any funds received in settlement thereof, shall be applied as follows: (1) to pay taxes on such damages or funds; (2) to reimburse CY for reasonable claim-related expenses and legal fees not otherwise paid out of the decommissioning fund; (3) to reduce subsequent decommissioning collections; and (4) to reimburse Customers to the extent no further decommissioning collections are required. (The order of application in any case shall be the most tax efficient method.)

(C) The Settling Parties recognize that legislation, directing the DOE to grant spent fuel removal priority to permanently shutdown plants may resolve some of the spent nuclear fuel storage issues. In the event that such legislation produces decommissioning cost savings for Connecticut Yankee, such savings, net of any expenses, shall be passed through to the Decommissioning Trust for the benefit of Customers.

(6) Filing by July 1, 2004. Connecticut Yankee agrees to commence a proceeding with the Commission no later than July 1, 2004, for purposes of examining: (1) the status of the decommissioning expenditures and the expected remaining decommissioning costs and collections (2) the status of DOE litigation and cost recovery as described in Part II.A(5); (3) the status of legislation affecting the spent fuel storage costs; (4) the status of the pre-1983 spent fuel obligation to DOE as described in Part II.C; and (5) any other such decommissioning or non-decommissioning rate issues as Connecticut Yankee may consider appropriate for filing with and reporting to the Commission.

(7) **Formula Rate Specificity.** The specification of the manner in which Connecticut Yankee will calculate its formula rates to its Customers is attached as Attachment A.

(8) **Decommissioning Only Argument.** Various intervenors have contended in this proceeding that, under the Power Contracts in effect when the Plant permanently ceased operations in 1996, after January 1, 1998, CY is entitled to recover only the costs directly related to decommissioning the Plant and may not recover any of the remaining unamortized investment or any return on equity (the "Decommissioning Only Argument"). CY has disputed and continues to dispute this contention. Nothing herein shall constrain the Settling Parties, other than the DPUC, from continuing to assert the Decommissioning Only Argument before the Commission and on appeal therefrom, subject to any applicable laws or rules of practice and procedure. The DPUC, however, expressly waives and relinquishes its right to pursue the Decommissioning Only Argument. Accordingly, CY and the Settling Parties agree that the Decommissioning Only Argument shall be severed.

B. Decommissioning Issues.

(1) **Decommissioning Cost Collection.** Connecticut Yankee shall bill and collect $16.742 million annually for decommissioning costs, commencing on the Effective Date and extending until the cost of decommissioning the Plant is fully recovered (but no later than June 29, 2007), subject to subsequent adjustment as a result of Connecticut Yankee's agreement to submit a filing with the Commission pursuant to Part II-A(6) of this Offer of Settlement. The decommissioning collections set forth herein incorporate collections prior to the Effective Date and there shall be no refunds of any amounts collected prior to the Effective Date. The decommissioning collections set forth herein also assume merger of the assets of the Pre-1983

Spent Fuel Trust and the Decommissioning Trust, as set forth in Part II.C below. Although these collections are anticipated to stop no later than June 29, 2007, nothing in this Offer shall waive, discharge or otherwise relieve the obligation of the Customers to pay the total decommissioning costs, and spent fuel storage costs not otherwise paid by the DOE, as set forth in the Power Contracts.

(2) Estimates. For purposes of obtaining schedules of ruling amounts under Section 468A of the Internal Revenue Code, the decommissioning estimate for Connecticut Yankee assumed in this Offer of Settlement, as of January 1, 2000, in January 1, 2000 dollars, is $393.3 million for all remaining decommissioning costs, with Connecticut Yankee's collections based on the following economic assumptions: (1) after tax earned interest on accumulated fund amounts of 5.10%; (2) average annual inflation adjustment of 3.35% for costs that are subject to escalation (i.e., excluding the DOC contract, already incurred costs, and other costs that are not subject to inflation); (3) the assumed method of decommissioning is the DECON method; and (4) the assumed year in which substantial decommissioning costs were first incurred is 1997.

(3) Contaminated Soil and Asphalt.

(A) In order to achieve the site release criteria specified in the DOC Contract, Connecticut Yankee shall not collect from its Customers any costs for removal or disposal of contaminated soil and asphalt within the Connecticut Yankee Plant site boundary in excess of $12.6 million (in 1996 dollars), as escalated.

(B) Connecticut Yankee has not collected and shall not collect any funds from its Customers for the costs of removal or disposal of contaminated materials that were improperly removed from the Connecticut Yankee Plant site. CY's 2000 decommissioning cost estimate and the collection schedule set forth in Part II.B(1) do not contain any amounts for removal or

disposal of contaminated materials located in areas outside the Connecticut Yankee Plant site boundary.

(4) Safety and Budget Incentives. Connecticut Yankee and the Settling Parties agree that monetary incentives, properly structured, can play an effective role in enhancing safety and limiting decommissioning expenditures. To provide such incentives to Connecticut Yankee shareholders, Connecticut Yankee and the Settling Parties agree to the following incentive terms and conditions:

(A) The Settling Parties do not contest Connecticut Yankee's representations that it is committed to conducting decommissioning in a manner that protects employees and contractor personnel and has negotiated the DOC Contract such that there is zero tolerance for safety violations. Thus, the DOC Contract contains safety incentives for Bechtel that, unless reasonably waived, immediately decrease the DOC contract price for <u>any</u> Lost Time Accidents or OSHA Reportable Incidents (as defined in the DOC Contract). The amounts of these price decreases are set forth in the DOC Contract. The Settling Parties agree that, in the event there are any such price decreases, they shall be passed through, in their entirety, to the Decommissioning Trust for the benefit of CY's Customers. Because the standard provided for in the DOC Contract for Lost Time Accidents and OSHA Reportable Incidents is zero, there are no bonus provisions associated with this subject.

(B) The Settling Parties do not contest Connecticut Yankee's representation that it is also committed to conducting decommissioning in a manner that minimizes radiation exposure to employees and contractor personnel. Accordingly, the Settling Parties agree that in the event that decommissioning of the Plant is completed at a cost below the Incentive Budget minus $10

million savings target in Part II.B(4)(D), below, but the Generic Environmental Impact Statement ("GEIS") total site dose estimate for the project is exceeded. Connecticut Yankee's net incentive provided in Part II.B(4)(D) shall be reduced by 10%. In the event decommissioning is completed at a cost above the Incentive Budget plus $10 million overage target in Part II.B(4)(D) below, and the GEIS site dose is exceeded, Connecticut Yankee shall be required to pay to the Decommissioning Trust for the benefit of its Customers as set forth in Part II.B(4)(F) below, $10,000 per person/rem over the GEIS dose estimate.

The reduction in the incentive and the payment for the benefit of Customers, as the case may be, shall be decreased by the amount Connecticut Yankee is able to recover in any damages for such excess site dose from third parties (including price decreases from Bechtel). In no event shall Connecticut Yankee's payments for the benefit of its Customers under this paragraph II.B(4)(B), taken together with any overages funded by Connecticut Yankee under Part II.B(4)(D), exceed a total of $10 million.

(C) In order to implement a budget incentive for the decommissioning, Connecticut Yankee and the Settling Parties agree that Connecticut Yankee's revised decommissioning budget from January 1, 2000, through June 30, 2004 (the expected Final Acceptance Date under the DOC Contract) is $332,969,433 (Incentive Budget), including contingency and escalation. This amount includes the Bechtel DOC contract, including ISFSI-related costs, but excludes any ISFSI-related operating and maintenance costs incurred after July 1, 2004, the ISFSI dismantlement and decommissioning expenditures, the non-DOC cask handling costs related to the ISFSI, and the license termination costs related to the ISFSI. The Incentive Budget also assumes that such amount will be paid for work conducted prior to June 30, 2004, and that all

major decommissioning activities will have been completed by that date, but excludes costs associated with terminating any Nuclear Regulatory Commission license required for spent fuel.

(D) The Settling Parties agree that, subject to the terms and conditions set forth in Part II.B(4)(B), if Connecticut Yankee is able to complete all major decommissioning and other dismantlement of the plant for less than the Incentive Budget minus $10 million, Connecticut Yankee shall be entitled to retain 10% of any net savings (after payment of all other incentive fees and bonuses), with Connecticut Yankee's share of such savings capped at $10 million, and with the balance of such net savings flowing through to the benefit of CY's Customers. The Settling Parties further agree that if Connecticut Yankee's actual decommissioning costs exceed the Incentive Budget plus $10 million, subject to the terms and conditions of Part II.B(4)(B), Connecticut Yankee shall be required to fund 10% of any such net overage (calculated after recovery by Connecticut Yankee of all other damage or penalty payments from third parties, including Bechtel), and 90% of such net overage shall be recovered from the Customers (provided such overages are prudently incurred), with Connecticut Yankee's share of such overage capped at $10 million, and any additional prudently incurred overages fully paid by CY's Customers. The Settling Parties agree that any bonuses paid to CY under this Part II.B(4)(D), shall not be subject to or affected by the Decommissioning Only Argument.

(E) Connecticut Yankee and the Settling Parties agree that Connecticut Yankee savings and Connecticut Yankee overages, as the case may be, will not be subject to windfalls or force majeure events. That is, any windfalls that lower the costs of decommissioning below the Incentive Budget and that are beyond the control or direction of Connecticut Yankee shall be deducted from the Incentive Budget, and the savings flow through to the benefit of CY's Customers, without Connecticut Yankee sharing in any such savings. For the purposes of this

Part II.B(4)(E), any DOE recoveries that reduce the Incentive Budget shall be considered windfalls and shall not entitle Connecticut Yankee to earn any incentive payment. Similarly, the Settling Parties agree that any force majeure events (i.e., events that are beyond the control or direction of Connecticut Yankee such as a change in the site release criteria below 10 millirem) which result in unexpected and unanticipated increases in the Incentive Budget shall be flowed through for reimbursement by Customers, without Connecticut Yankee's share in the cost of any such overages. Connecticut Yankee and the Settling Parties agree that any payments or sums received for re-powering the site or selling the land shall not be treated as windfalls for purposes of this Part II.B(4)(E), but shall be flowed through for the benefit of CY's Customers subject to the provisions of Part II.A(2) above.

(F) Connecticut Yankee shall conduct a reconciliation of any adjustments for incentive bonuses or penalties, within sixty (60) days of the Final Acceptance Date (defined in the DOC Contract). For purposes of calculating the budget incentive amounts of Part II.B(4)(D), the actual costs incurred from January 1, 2000, to the Final Acceptance Date, shall be compared to the Incentive Budget set forth in Part II.B(4)(C). Any adjustment amounts arising as a consequence of such reconciliation, shall be adjusted for taxes and applied evenly to collections from the effective date of the reconciliation, such that the full amount of the adjustment shall have been accommodated by June 29, 2007. If the reconciliation shows a bonus to be paid to CY, the collections for this bonus, plus any applicable taxes, shall be separately identified and paid directly to CY as a monthly addition to net income. These collections shall not be paid to the decommissioning trust. If the reconciliation shows any penalty amount to be paid by CY, such penalty amount shall be amortized over the remaining period until June 29, 2007, and paid

to the decommissioning trust on a monthly schedule. CY will submit an informational filing to the Commission documenting this reconciliation.

C. Disposition of the Pre-1983 Spent Fuel Trust Fund.

(1) Consolidation of Trusts.

The Commission order approving this Offer shall constitute appropriate authority to obtain access to funds presently held in trust for the disposal of spent nuclear fuel. These funds have been collected previously from Customers pursuant to the Power Contracts and are held in trust for the express purpose of meeting Connecticut Yankee's pre-1983 spent fuel disposal obligations to the DOE (the "Pre-1983 Spent Fuel Trust"). The Settling Parties agree that there are significant financial benefits to Customers arising from dissolving the Pre-1983 Spent Fuel Trust and merging its assets into the assets of the Decommissioning Trust. The Indenture of Trust for the Pre-1983 Spent Fuel Trust provides that it may be merged into another trust if permitted by a governmental authority having jurisdiction. The Commission Order approving this settlement shall constitute appropriate authority to effect such consolidation and merger of assets. Connecticut Yankee will effect such consolidation when it can be achieved in the most tax-efficient manner. Connecticut Yankee will submit an informational filing to the Commission within thirty (30) days of the merger of the Pre-1983 Spent Fuel Trust and the Decommissioning Trust.

(2) Earlier Occurrence of the Pre-1983 Spent Fuel Obligation to DOE.

The decommissioning collection schedule set forth in Part II.B(1) above assumes that Connecticut Yankee will not be liable to the DOE for the payment on account of spent

nuclear fuel generated prior to 1983 until after June 29, 2007. In the event that, after CY has prudently pursued its legal and equitable remedies against the DOE, CY's liability to the DOE for pre-1983 spent nuclear fuel becomes due and payable from the Decommissioning Trust prior to June 29, 2007, the Settling Parties agree that Connecticut Yankee may file a proceeding with this Commission to increase collections as necessary to compensate for the premature payment of this obligation. The other Settling Parties agree that, in the event of such filing, they will not object to increases in collections expressly for such purpose.

D. Effective Date; Refunds.

This Offer of Settlement shall become effective the first day of the calendar month immediately following Commission approval of the Offer of Settlement by final and non-appealable order (the "Effective Date"). The Settling Parties have agreed that any amounts that otherwise would be refunded in Docket ER97-913-000 are reflected in reductions to the ongoing charges specified in this Offer.

III. GENERAL PROVISIONS

A. By entering into this Offer of Settlement, Connecticut Yankee and the Settling Parties shall not be deemed in any respect to make any admission that any allegation or contention in the above-captioned proceedings is true or valid.

B. This Offer of Settlement establishes no principles or precedents on any issue, including the appropriateness of the calculations or assumptions utilized in reaching this Offer of Settlement, and acceptance of this Offer of Settlement by any party or by the Commission shall

not in any respect constitute a determination as to the merits of any allegations or contentions made in this proceeding.

C. The discussions which have produced this Offer of Settlement have been conducted with the explicit understanding and pursuant to Rule 602(e) of the Commission's Rules of Practice and Procedure, that all offers of settlement. settlement agreements and discussions relating thereto are and shall be privileged. shall be without prejudice to the position of any party or participant presenting such offer, entering into such agreement, or participating in such discussions, and are not to be used in any manner in connection with these or any other proceedings involving one or all of the parties to this proceeding or otherwise, except to interpret or enforce this settlement.

D. In negotiating this Offer of Settlement, CY and the Settling Parties have considered the various separate promises and obligations contained herein, whether related to decommissioning, non-decommissioning, or other issues, as fundamental to a comprehensive agreement of all issues in Docket No. ER97-913-000. This Offer of Settlement is thus submitted under the condition that, in the event the Commission does not by order approve this Offer of Settlement in its entirety, without change or condition, Connecticut Yankee and the Settling Parties each reserve the right to withdraw this Offer of Settlement in its entirety. The Settling Parties agree that they shall each take such action as is reasonable, including filing statements, pleadings and briefs with the Commission, to facilitate the acceptance by the Commission of the this Offer and to defend the Offer against challenge by any non-settling parties, intervenors or other third parties, including participating in any appeals.

E. Nothing in this Offer of Settlement shall be construed to prevent CY from filing a proceeding under Section 205 of the Federal Power Act or prevent the other Settling Parties from

filing a proceeding under Section 206 of the Federal Power Act, or to prevent any party taking any position in response, except that, should the Commission accept this Offer as set forth above, neither CY nor any other Settling Party will, for the remaining term of the Power Contracts: (1) challenge in any way CY's recovery of its unamortized investment in accordance with the terms of this Offer; (2) seek to adjust the return on equity set forth in this Offer; or (3) seek to adjust the decommissioning charges in any respect prior to July 1, 2004, except as authorized in this Offer.

F. Nothing in this Offer of Settlement shall be construed to prevent the Settling Parties from challenging or contesting the prudence of any action taken by Connecticut Yankee after the date of this Offer of Settlement.

G. This Offer of Settlement does not resolve the Decommissioning Only Argument set forth in Part II.A.(8) above, except as to DPUC.

H. The DPUC and OCC agree between themselves that the DPUC shall open the appropriate proceedings to determine the means of returning or crediting the savings accruing under this Offer to Customers' Connecticut ratepayers.

I. This Offer of Settlement may be executed in one or more counterparts.

IN WITNESS WHEREOF, CY and the other Settling Parties have caused this Offer of Settlement to be executed by their respective officers, attorneys or agents, each having full authority to enter into this Offer of Settlement.

Dated: April 7, 2000

Respectfully submitted,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

By _____
Thomas W. Bennet, Jr
Its Vice President and Chief Financial Officer

CONNECTICUT DEPARTMENT OF PUBLIC UTILITY CONTROL

By _____

Its

CONNECTICUT OFFICE OF CONSUMER COUNSEL

By_____

Its

IN WITNESS WHEREOF, CY and the other Settling Parties have caused this Offer of Settlement to be executed by their respective officers, attorneys or agents, each having full authority to enter into this Offer of Settlement.

Dated: April 7, 2000

Respectfully submitted,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

By _____

Its

CONNECTICUT DEPARTMENT OF PUBLIC UTILITY CONTROL

By _____

Its Chairman

CONNECTICUT OFFICE OF CONSUMER COUNSEL

By _____

Its Consumer Counsel

IN WITNESS WHEREOF, CY and the other Settling Parties have caused this Offer of Settlement to be executed by their respective officers, attorneys or agents, each having full authority to enter into this Offer of Settlement.

Dated: April 7, 2000

Respectfully submitted,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

By _____

Its

CONNECTICUT DEPARTMENT OF PUBLIC UTILITY CONTROL

By _____

Its

CONNECTICUT OFFICE OF CONSUMER COUNSEL

By Guy R. Mazza

~~Its~~ CONSUMER COUNSEL

FORMULA RATES

In general, costs subject to formula rate treatment include amortization of unrecovered assets, fuel expense, decommissioning collections, income taxes, interest expense/(income) and operating income. The attached detailed schedules support the proposed billing method. The major components of the formula rate treatment, which will be provided for through the Cost of Service rates within the power contracts for Connecticut Yankee Atomic Power Company are:

(a) Fuel

Fuel expense includes the Department of Energy (DOE) Decontamination and Decommissioning Fund special assessment for domestic utilities assessed over a 15-year period beginning in 1992. This assessment is scheduled to be paid annually each October, when invoiced by the DOE, and is amortized over the 12 month period including the month of payment. This assessment obligation is expected to continue through the remaining license life of Connecticut Yankee.

(b) Amortization of Unrecovered Assets

The remaining net investment in utility plant (excluding the book value of land), fuel, and materials and supplies is classified as Net Unrecovered Assets. Any salvage of unrecovered assets, less cost of removal, is credited to this account. The amortization of net unrecovered assets is calculated on a straight-line basis over the remaining period ending June, 2007 and credited to this account.

(c) Decommissioning Collections

Decommissioning collections represent billings to customers to fund the Decommissioning Trust for the remaining estimated costs of decommissioning the Connecticut Yankee plant. The decommissioning collections are billed on a straight –line basis over the remaining period through June, 2007.

(d) Income Taxes

Income taxes are accounted for in accordance with federal, state and regulatory regulations as it relates to the taxable income of the Company, including the Decommissioning Trust, which determine the charges and credits to customers. Investment tax credits, which were previously deferred, are amortized and credited to customers through the period June, 2007. The tax effect of temporary differences (differences between the periods in which transactions affect revenues and the periods in which they affect the determination of income subject to tax) is accounted for in accordance with current regulations. The Company recognizes adjustments to deferred income taxes and a corresponding regulatory asset or liability to customers to reflect the future revenues or reduction in revenues that will be required when the temporary differences reverse and are recovered/(credited) in rates.

(e) Operating Income

Return on Equity (ROE) will be calculated by using the approved ROE rate times the remaining balance of equity as of the effective date of the new billing method.

(f) Interest Expense/(Income)

Interest expense or income, other than Spent Fuel Obligation /Trust interest expense/income, incurred/ accrued after the effective date of the new billing method will be charged/(credited) to customers.

Attachment A
Schedule A
Page 2 0f 2

Connecticut Yankee Atomic Power Company
Cost Of Service (*)
Current Month

				Reference Schedule
1	Operating Costs			
2	Operation Expenses:			
3	Fuel Expense		$	B, Page 1 0f 3
4	Other Operation Expenses			B, Page 1 0f 3
5	Maintenance Expenses			B, Page 2 0f 3
6	Amortization of Unrecovered Assets			B, Page 2 0f 3
7	Decommissioning Collections			B, Page 2 0f 3
8	Spent Fuel Storage Costs			B, Page 2 0f 3
9	Taxes Other Than Income Taxes			B, Page 2 0f 3
10	Taxes - Income			C, Page 1 of 1
11	Total Operating Expenses	Add Lines 7 thru 15		
	Adjustment of Prior Month Billing			
12	Operating Expenses - Actual			
13	Operating Expenses As - Billed			B, Page 3 0f 3
14	Total Adj for Prior Month Billing	Line 17 less Line 18		B, Page 3 0f 3
15	Return on Investment			
16	Current Month Return on Equity			D, Page 1 of 2
17	Other - Net			D, page 2 of 2
18	Total Return On Investment	Add Lines 21 and 22		
19	TOTAL COST OF SERVICE	Add Lines 16, 19 and 23	$	

Note

Accounts listed on this schedule and all supporting schedules are subject to change, without the need to file pursuant to Section 205 of the Federal Power Act, due to renumbering or redesignation by FERC; and provided that the description of the costs included under any new account number is is not materially different from the description of the costs which would have been included under the account numbers listed here.

1 CONNECTICUT YANKEE ATOMIC POWER COMPANY
2 OPERATING COSTS BY
3 FERC ACCOUNT SERIES

	Operating Costs	Ferc Account	Title	Current Month
4				
5	Fuel Expense			
6		518	Nuclear Fuel Expense	$ -
7		51801	DOE D&D Expense	
8	**Total Fuel Expense**			$
9	Other Operation Expenses			
10		517	Supervision/Engineering	$
11		519	Coolants and Water	
12		520	Steam Expenses	
13		521	Steam From Other Sources	
14		522	Steam Transfer	
15		523	Electric Expense	
16		524	Misc. Power Expenses	
17		525	Rents	
18		560 thru 567	Operation Expenses for	
19			Transmission	
20		922	Admin Exp. Transfer-Credit	
21		923	Outside Services Employed	
22		924	Property Insurance	
23		925	Injuries and Damages	
24		926	Employee Pensions & Benefits	
25		928	Regulatory Commission Exp.	
26		929	Duplicate Charges-Credit	
27		930.1	Gen. Advertising Expenses	
28		930.2	Misc General Expenses	
29		931	Rents	
30		933	Transportation Expenses	
31	**Total Other Operating Expenses**			$

Note

Accounts listed on this schedule and all supporting schedules are subject to change, without the need to file pursuant to Section 205 of the Federal Power Act, due to renumbering or redesignation by FERC; and provided that the description of the costs included under any new account number is is not materially different from the description of the costs which would have been included under the account numbers listed here.

In general, cost incurred relating to the pre-decommissioining liability continue to be charged to operations and included in Shedule B. Examples of these costs include: DOE D&D Assessment, Amortization of Unrecovered Assets, Income Taxes, etc.

Attachment A
Schedule B
Page 2 of 3

CONNECTICUT YANKEE ATOMIC POWER COMPANY
OPERATING COSTS BY
FERC ACCOUNT SERIES

	Operating Costs	Ferc Account	Title		Current Month
1	Maintenance Expense				
2		528	Supervision and Engineering	$	
3		529	Structures		
4		530	Reactor Plant		
5		531	Electric Plant		
6		532	Misc Nuclear Plant		
7		568 thru 574	Transmission		
8		935	General Plant		
9	**Total Maintenance Expense**			$	
10	Amortization of Unrecovered Assets				
11		403	Depreciation Expense-Plant	$	
12		407	Amortization of Property Losses Unrecovered Plant and Regulatory Study Costs		
13	**Total Amortization of Unrecovered Assets**			$	
14	Decommissioning Collections				
15		403	Depreciation Expense-Decommissioning	$	
16	**Total Decommissioning Collections**				
17	Spent Fuel Costs				
18		403	Spent Fuel Costs Not Included in Decommissioning	$	
19	Taxes Other Than Income	408.1	Taxes Other Than Income Taxes Which RelateTo Utility Operating Income. Such as, Assessments by Federal State, County, Municipal or Gov't Authorities.		
20	**Total Taxes Other Than Income**			$	
21	Taxes-Income				
22		409.1	Local, State and Federal Income Taxes Related to Operating Income	$	
23		410.1 and 411.1	Deferred Local. State and Federal Inc. Taxes Related to Oper. Income		
24		411.4	Investment Tax Credit Adjustments Related to Operating Income		
25	**Total Taxes-Income**			$	

1 Connecticut Yankee Atomic Power Company
2 Cost Of Service (*)
3 Adjustment For Prior Month Billing

			Prior Month Billing	Prior Month Actual
4				
5				
6	Operating Costs			
7	Operation Expenses:			
8	Fuel Expense		$	$
9	Other Operation Expenses			
10	Maintenance Expenses			
11	Amortization of Unrecovered Assets			
12	Decommissioning Collections			
13	Spent Fuel Storage Costs			
14	Taxes Other Than Income Taxes			
15	Taxes - Income		_____	_____
16	Total Operating Expenses	Add Lines 7 thru 15	_____	_____
	Adjustment of Prior Month Billing			
17	Operating Expenses - Actual			
18	Operating Expenses As - Billed			
19	Total Adj for Prior Month Billing	Line 17 less Line 18	_____	_____
			_____	_____
20	Return on Investment			
21	Current Month Return on Equity			
22	Other - Net		_____	_____
23	Total Return On Investment	Add Lines 21 and 22	_____	_____
24	TOTAL COST OF SERVICE	Add Lines 16, 19 and 23	$ _____	$ _____

Attachment A
Schedule C
Page 1 of 1

1 CONNECTICUT YANKEE ATOMIC POWER COMPANY
2 BILLING METHOD FOR
3 INCOME TAXES

4 Calculation of Composite State and Federal Income Tax Rate

5	Start with Pretax at 100%			%	100.00
6	Current State Rate		(1)		7.50
7	State Taxes as A Percentage of Pretax Income				7.50
8	Income Subject to Federal Income Tax	Line 5 - Line 8			92.50
9	Current Federal Rate		(1)		35.00
10	Federal Taxes as A Percentage of Pretax Income				32.38
11	Combined Federal and State Income Tax Rate	Line 7 + Line 10		%	39.88
12	Start with Pretax at 100%			%	100.00
13	Less the Combined Federal and State Income Tax Rate				39.88
14	Rate for Determining Required Net Income			%	60.12

15 Required Net Income Calculation , Example

16	Net Income Required		$	100.00
17	Divided by One Minus the Combined Rate	Line 14	%	60.12
18	Equals Grossed Up Income		$	166.33
19	Combined Federal and State Income Tax Rate	Line 11	%	39.88
20	Combined Federal and State Income Tax Dollars		$	66.33
21	Net Income Required	Line 18 - Line 20	$	100.00

22 Calculation of State and Federal Income Taxes for Operating Costs

23	Required Net Income		$	
24	Less: ITC Flow Through			
25	Plus or Minus:			
25	Prior Period Adjustments			
26	SFAS 109 (Adjustment of Deferred Taxes)			
27	Permanent Differences (Such as Non-deductible Meals)			
28	Income Subject to Gross Up	Lines 23 thru 27		
29	Divided by One Minus the Combined Rate	Line 14		
30	Equals Grossed Up Income			
31	Less Required Net Income			
32	Gross Up of Taxes to be Billed	Line 30 - Line 31		
33	Less: ITC Flow Through			
34	Plus or Minus:			
35	Prior Period Adjustments			
36	SFAS 109 (Adjustment of Deferred Taxes)			
37	Permanent Differences (Such as Non-deductible Meals)			
	Total Income Taxes - Operating Costs	Lines 32 thru Line 37	$	

Note:
1 Changes in the applicable tax rates are to be reflected in the actual calculations.

Attachment A
Schedule D
Page 1 0f 2

1 CONNECTICUT YANKEE ATOMIC POWER COMPANY
2 RETURN ON INVESTMENT
3 CALCULATED

4	RETURN ON				
5	INVESTMENT	FERC Account	Title	(1)	Prior Month
6	Equity	201 thru 218	Proprietary Capital Accounts, As Applicable		
7	**Total Equity Investment**			$	
8	Authorized Rate of				
9	Return on Equity	Not Applicable	Per FERC Order	%	
10	**Total Current Month Return On Equity**		Line 14 X Line 16 / 12	$	

Note

1 Prior month equity balances are used to determine the current month return.

2 Accounts listed on this schedule and all supporting schedules are subject to change, without the need to file pursuant to Section 205 of the Federal Power Act, due to renumbering or redesignation by FERC; and provided that the description of the costs included under any new account number is is not materially different from the description of the costs which would have been included under the account numbers listed here.

Attachment A
Schedule D
Page 2 0f 2

CONNECTICUT YANKEE ATOMIC POWER COMPANY
RETURN ON INVESTMENT
OTHER

1					
2					
3					
4	Return on Investment				
5	Other - Net	FERC Account	Title		Current Month
6	Other Income				
7		419	(1) Interest & Dividend Income	$	
8	**Total Other Income**			$	
9	Other Expense				
10		427 - 431	(1) Interest Expense and	$	
11			Related Costs		
12		408.2	(2) Taxes Other Than Income Taxes		
13			Which RelateTo Non - Utility		
14			Operating Income		
15		409.2	(2) Local, State and Federal Income		
16			Taxes Related to Non-Operating		
17			Income		
18		410.2 and 411.2	(2) Deferred Local, State and Federal		
19			Income Taxes Related to		
20	**Total Other Expense**			$	
21	**Total Return on Investment - Other, Net**		Line 8 + Line 20	$	

Notes

1 Spent Fuel Trust Interest income and expense accrued / incurred after the effective date of this new cost of service billing method will be credited / charged to the decommissioning trust fund and will not appear on this cost of service billing.

Other income and expense, other than Spent Fuel Trust Interest income and expense, accrued / incurred after the effective date of this new cost of service billing method will be included in this cost of service service billing method and will be credited / charged to customers.

2 To the extent that the tax relates to Other Income and Expense included in the cost of service.

FILED
OFFICE OF THE SECRETARY
00 APR 27 PM 4:3
FEDERAL ENERGY
REGULATORY COMMISSION

UNITED STATES OF AMERICA
FEDERAL ENERGY REGULATORY COMMISSION

CONNECTICUT YANKEE ATOMIC POWER COMPANY	))	DOCKET NO. ER97-913-001

APRIL 27, 2000

SUPPLEMENT TO
OFFER OF SETTLEMENT

This Supplement to Offer of Settlement ("Supplement") is submitted, pursuant to Rule 602 of the Commission's Rules of Practice and Procedure, jointly by Connecticut Yankee Atomic Power Company ("CY", "Connecticut Yankee" or the "Company") and the Connecticut Municipal Electric Energy Cooperative ("CMEEC"). The purpose of this Supplement is to add CMEEC as a party to the Offer of Settlement filed on April 7, 2000 by CY, the Connecticut Department of Public Utility Control ("DPUC") and the Connecticut Office of Consumer Counsel ("OCC") (the "Offer of Settlement").

Pursuant to this Supplement, CY and CMEEC agree that CMEEC shall become a signatory to the Offer of Settlement and shall be designated to be one of the Settling Parties as defined therein. CMEEC shall be a party to the Offer of Settlement in the same manner and to the same extent as the DPUC. Accordingly, it is agreed by CY and CMEEC that, except for Section III.H, wherever the Offer of Settlement uses the term "DPUC", it is to be modified to read "DPUC and CMEEC" and that appropriate changes in number shall be made to the wording where necessary (e.g. "are" vs. "is").

CY and CMEEC further agree that, if accepted and approved by the Commission, the Offer of Settlement will resolve, as between CY and CMEEC, all issues that were or could have been raised in CY's filing in Docket No. ER97-913-000.

IN WITNESS WHEREOF, CY and CMEEC have caused this Offer of Settlement to be executed by their respective officers, attorneys or agents, each having full authority to enter into this Offer of Settlement.

Dated: April 27, 2000

Respectfully submitted,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

By _____
Kathleen J. Jewell-Kelleher
Its: Treasurer

CONNECTICUT MUNICIPAL ELECTRIC ENERGY COOPERATIVE

By _____
Maurice R. Scully
Its Executive Director

CY and CMEEC further agree that, if accepted and approved by the Commission, the Offer of Settlement will resolve, as between CY and CMEEC, all issues that were or could have been raised in CY's filing in Docket No. ER97-913-000.

IN WITNESS WHEREOF, CY and CMEEC have caused this Offer of Settlement to be executed by their respective officers, attorneys or agents, each having full authority to enter into this Offer of Settlement.

Dated: April 27, 2000

Respectfully submitted,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

By _____
Kathleen J. Jewell-Kelleher
Its: Treasurer

CONNECTICUT MUNICIPAL ELECTRIC ENERGY COOPERATIVE

By _____
Its

UNITED STATES OF AMERICA
FEDERAL ENERGY REGULATORY COMMISSION

CONNECTICUT YANKEE ATOMIC POWER COMPANY	))	**DOCKET NO. ER97-913-000**

CERTIFICATE OF SERVICE

This is to certify that on April 27, 2000, I caused to be served, via first class mail, postage prepaid, a copy of the Supplement to Offer of Settlement upon the attached Service List.

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