



**UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
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June 20, 2000

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

SUBJECT: NRC INTEGRATED INSPECTION REPORT 05000213/2000-001

Dear Mr. Heider:

On May 19, 2000, the NRC completed an inspection of decommissioning activities at the Haddam Neck Plant. The Region I Office of Investigation assisted on a selected area that was inspected. The enclosed report presents the results of the inspection.

During the thirteen-week period covered by this inspection, your conduct of activities at the Haddam Neck facility was characterized by the safe wet storage of spent nuclear fuel in your spent fuel pool and the scheduling and completion of preventive maintenance surveillances for systems, structures, and components important to the safe decommissioning of the facility. The demolition of multiple site buildings and the removal of large segments of the reactor coolant system piping were appropriately focused on safety.

We noted that aggressive ALARA measures were in place to limit exposures and to reduce personnel contamination. Further, no regulatory exposure limits have been exceeded and your overall decommissioning dose goal was not in jeopardy. However, we are concerned that the reactor pressure vessel segmentation project has been a significant radiological challenge, with a job completion dose projection estimated to be approximately five times your original goal. As discussed between George Pangburn, Director, DNMS, and NRC staff with Russ Mellor, Chairman, CEO, and President of Connecticut Yankee and members of your staff, on May 23, 2000, the NRC staff will increase its onsite inspection effort to ensure that reasonable efforts continue to control personnel exposures ALARA. We also anticipate a thorough discussion of the results of the modifications to the project during our management meeting you requested on June 27, 2000 in the NRC Region I office.

Based on the results of this inspection, we plan to take no further action regarding the inaccurate information which was submitted in January 1998, to support a control room modification and change to your Updated Final Safety Analysis Report. Your self-identification and correction of the calculation errors along with the low safety significance of the potential impact to control room personnel at the time of the modification were considered in this determination.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR).

Sincerely,

/RA/

Mr. K Heider

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Ronald R. Bellamy, Chief
Decommissioning and Laboratory Branch
Division of Nuclear Material Safety

Docket No. 05000213
License No. DPR-61

Enclosure:
NRC Inspection Report No. 05000213/2000-001

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U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.: 05000213

License No.: DPR-61

Report No.: 05000213/2000-001

Licensee: Connecticut Yankee Atomic Power Company (CYAPCO)
P. O. Box 270
Hartford, CT 06141-0270

Facility: Haddam Neck Station

Location: Haddam, Connecticut

Dates: February 22, 2000 to May 19, 2000

Inspectors: Marie Miller, Senior Health Physicist
Joseph Nick, Enforcement Specialist
John Wray, Health Physicist

Approved by: Ronald Bellamy, Chief, Decommissioning and Laboratory Branch
Division of Nuclear Materials Safety

EXECUTIVE SUMMARY

Haddam Neck Station
NRC Inspection Report No. 05000213/2000-001

This routine inspection included aspects of licensee activities regarding dismantlement and decommissioning of the facility. The report covers a thirteen-week period of inspection by regional NRC personnel, and includes reviews and assessments of spent fuel safety, decommissioning performance, plant support activities, maintenance, surveillance and engineering.

Decommissioning Operations:

The staff was knowledgeable of monitoring requirements, alarms, and required actions with respect to spent fuel safety. Personnel promoted into site management positions meet the qualifications for these positions required by ANSI 18.1. The licensee's contractor workforce is staffed with qualified personnel and the organization is designed to contribute to health and safety of the public through management of decommissioning activities on-site.

The licensee has an adequate program to ensure safe wet storage of spent fuel. The visual inspection of spent fuel assemblies in the spent fuel pool was completed in a safe manner with good contamination controls and managerial oversight. Although the total exposure was fifty percent greater than originally estimated, adequate ALARA controls were followed.

Decommissioning Status:

The licensee continued safe dismantlement of radiologically contaminated major components. Two shifts continued to segment reactor pressure vessel internals. Follow-up to the reactor cavity filtration system water overflow event on March 16, 2000 was thorough and prompt. The demolition and removal of the engineering, maintenance, and alternate security access buildings were well planned, coordinated, and implemented. Survey results indicated that no detectable radioactive material was released to the environment in debris from the dismantled building.

Use of a management quality review board to resolve self-identified issues is a reasonable approach. The NRC will continue to monitor the licensee's actions to close out survey and dose projections as part of the Offsite Material Recovery Program. With respect to the on-site self-assessment activities, the licensee implemented an effective program for oversight of its Decommissioning Operations Contractor.

Plant Support and Radiological Controls:

Challenges with new techniques required better planning and more frequent assessment of high person-rem activities. The reactor pressure vessel segmentation project has been a significant radiological challenge, resulting in greater personnel exposures than originally estimated. ALARA planning for removal of the reactor coolant system piping indicated a weakness in coordination with other departments which resulted in an increase in re-work. However, for both of these major component removal activities, no exposure limits have been exceeded.

The new electronic dosimetry and radiological controlled areas access control system was adequately tested and installed. The system is capable of monitoring personnel exposures to ensure NRC dose limits of 10 CFR 20 Subpart C are not exceeded and that NRC dose

monitoring requirements of 10 CFR 20 Subpart F are met.

The waste-water processing streams from the reactor pressure vessel segmentation project were adequately characterized.

Maintenance Surveillance and Engineering:

Required surveillances important to the safe storage of spent fuel are being effectively scheduled and completed in a timely manner. Surveillances are being adequately performed on structures, systems and components important to the safe decommissioning of the plant.

The licensee implemented a safety review of major and minor decommissioning activities to determine if any involved an unreviewed safety question or a change in technical specification. As a result, there were two changes, specifically movement of heavy loads and reactor vessel segmentation, that were re-evaluated under 10 CFR 50.59 to ensure current TS and design basis accidents were considered.

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REPORT DETAILS

Summary of Facility Activities

The plant was maintained in a permanently shutdown condition during this inspection period. Dismantlement and removal of major plant equipment and structures continued with removal of the reactor coolant pumps, turbine cover and blades, reactor coolant system (RCS) piping, and demolition of the engineering, maintenance and alternate security access buildings.

I. Decommissioning Operations

O1 Conduct of Operations

O1.1 CYAPCO Organization

a. Inspection Scope (71801)

The inspectors observed the interactions of the licensee's staff, including back-shift staff, with respect to its operation of the Spent Fuel Pool Nuclear Island. The licensee's program for oversight of its Decommissioning Operations Contractor (DOC) was also reviewed. Specific details regarding the results of the licensee's focus self assessments are discussed in Section O7.1 of this report.

Organizational changes announced near the end of this inspection period were reviewed for compliance to position qualification requirements described in Technical Specifications (TSs), the Quality Assurance (QA) Manual, and ANSI 18.1.

b. Observations

For all periods inspected, adequate staffing was maintained in the control room. During a back-shift weekend tour of the control room, the inspectors observed that the control room was staffed with a certified fuel handler (CFH), a control room equipment operator and a field equipment operator, which was above the minimum shift crew composition for staff.

The licensee had documented a formal risk-based oversight program in December 1999. Fifteen program areas are evaluated using focused team assessments and standard weekly oversight assessment. The inspectors observed several licensee oversight personnel in containment providing coaching feedback to the DOC.

Near the end of this inspection period, several senior management changes to the licensee's organization became effective. These included promotions of site personnel to the following positions: Chairman, President and Chief Executive Officer; Vice President - Operations and Decommissioning; and Site Decommissioning Manager. The inspector reviewed these changes with regard to their position qualification requirements described in TSs, the QA Manual, and ANSI 18.1. All education and experience requirements for the persons promoted into these positions were satisfied. However, the inspector noted that the new Site Decommissioning Manager is not a licensed CFH. The licensee stated that the Operations and Maintenance Manager is CFH qualified and is designated a principal alternate for this position which satisfies condition 4.2.1 of ANSI 18.1. The inspector had no further questions regarding qualifications of personnel in new positions.

c. Conclusions

The staff was knowledgeable of monitoring requirements, alarms, and required actions with respect to spent fuel safety. Personnel promoted into site management positions meet the qualifications for these positions required by ANSI 18.1.

O1.2 DOC Organization

a. Inspection Scope (71801)

The inspector evaluated the decommissioning contractor organization, staffing, and qualifications to ascertain whether the DOC contributed to public health and safety through proper control of decommissioning activities.

b. Observations

The licensee's DOC announced a number of significant personnel and organizational changes during this inspection period. The inspector reviewed these changes with cognizant licensee and contractor personnel and interviewed affected management. The personnel changes included naming a new Project Manager responsible for all DOC decommissioning activities on-site and a new Site Manager responsible for the area managers as well as field engineering. The individuals selected for these positions meet the experience requirements expected of managers in comparable positions. The organizational change resulted in the Health Physics and Chemistry Manager reporting directly to the Project Manager. The inspector stated that the newly assigned personnel appeared to be well qualified and that the realignment of the reporting chain for the health physics and chemistry department should provide increased focus on radiological safety during the high dose decommissioning tasks presently being conducted at the facility. The inspector had no further questions.

c. Conclusions

The licensee's contractor workforce is staffed with qualified personnel and the organization is designed to contribute to health and safety of the public through management of decommissioning activities on-site.

O1.3 Spent Fuel Pool Safety

a. Inspection Scope (60705)

The inspection consisted of a review of the licensee's Spent Fuel Pool (SFP) cooling and makeup monitoring program and SFP leakage monitoring program procedures (ACP 1.2-2.88, Original and Revision 1, and ENG 1.7-175, Revision 1). This included a review of the SFP makeup and liner leakage trend plots, and the daily surveillance logs for the SFP temperature and level alarms.

b. Observations

Haddam Neck TS 6.6.6 requires that the primary method for SFP cooling and the primary method for spent fuel pool water makeup capability be monitored and maintained. Further, TS Limiting Conditions of Operation specify the water level and maximum temperature whenever fuel assemblies are stored in the SFP. The inspector noted that the licensee had procedures for evaluation of the SFP and connecting systems. The results of the licensee's annual SFP integrity evaluation noted no increasing trend for the average daily SFP makeup rate and the void space leakage collected has been consistently zero. The inspectors also discussed and reviewed the shift turnover logs and check-off lists for March 18 -19, 2000. SFP water level and temperature met required conditions. On-line display of these parameters were consistent with the recorded values.

c. Conclusions

The licensee has an adequate program to ensure safe wet storage of spent fuel.

O1.4 Spent Fuel Inspections

a. Inspection Scope (60705)

The licensee completed visual inspection of spent fuel assemblies in the SFP in preparation of loading the assemblies into dry cask storage containers.

b. Observations

The inspector observed the completion of the visual inspection of the 1016 spent fuel assemblies in the SFP (see Inspection Report 05000213/1999-004). Three additional assemblies had previously been determined to be damaged and therefore were not visually inspected during this operation. A final report of these activities dated March 13, 2000, was reviewed. Of the 1016 assemblies inspected, 39 were determined to be damaged. This appears to meet the design limits for damaged fuel of the NAC International (NAC) dry cask storage system.

The inspector verified that proper health physics practices were followed for this project by observing work activities in the Spent Fuel Pool Building (SFPB) and review of selected surveys and procedures. There were no personnel contaminations on the job and no individual exposures in excess of regulatory limits. The total exposure for the job was initially estimated to be approximately 600 person-millirem. The final dose for the project was approximately 900 person-millirem. The inspector discussed with cognizant licensee personnel the reasons why the job expended fifty percent more dose than originally estimated. The main cause of the overage was the decision to not operate a demineralizer made available for the project to reduce the cesium levels in the pool water. This decision was a concerted effort to keep the dose rates low in the occupied area where personnel controlled the operation of the equipment. Expected dose rates from the demineralizer in this area prevented the use of the demineralizer. The inspector acknowledged that the total dose was less than 1 person-rem and therefore the decision to not use the demineralizer appeared to be ALARA (As Low As Reasonable Achievable). However, the inspector stated that the increase by fifty percent in total dose was a trend observed in other site decommissioning projects and will be closely monitored during future inspections. (See Section R1.1 of this report.)

c. Conclusions

The visual inspection of spent fuel assemblies in the SFP was completed in a safe manner with good radiological controls and managerial oversight. Although the total exposure was fifty percent greater than originally estimated, adequate ALARA controls were followed.

II. Decommissioning Status

O2 Decommissioning Status of Facilities and Equipment

O2.1 Major Equipment Dismantlement

a. Inspection Scope (71801)

The inspector evaluated the licensee's status of decommissioning work through discussions with cognizant licensee personnel and observations of major equipment dismantlement activities.

b. Observations

The internals of the reactor pressure vessel (RPV) are being segmented into Class C or below and Greater-Than-Class C (GTCC). GTCC material will be placed into fuel assembly sized cans that will be transferred into the SFP for storage in the spent fuel racks. Class C or below material will be loaded back into the RPV and then prepared for disposal as low level radioactive waste. Also, sections of highly contaminated RCS piping were removed and staged for transport. During this inspection, the inspectors verified the removal and storage of equipment including RCP motors and pumps, and High Pressure Turbine Blades. Preparations for removal of the two Emergency Diesel Generators were verified also.

With respect to the RPV activities, there was generally good worker performance and radiological controls. However, the unique underwater Abrasive Water Jet (AWJ) cutting and Mechanical Disintegrating Machining (MDM) cutting had been very time consuming. Approximately 30,000 hours have been logged as of the end of this inspection period, and system improvements, including ALARA based changes have been required to improve system performance and manage radiological exposures. (See Section R1.1 for details.)

In addition, the inspector also reviewed the March 16, 2000, event follow-up and corrective actions, following an overflow of approximately one hundred gallons of reactor cavity water onto the charging floor inside containment and down to the mid and lower elevations of two loop areas. The cavity water filtration vent drum was modified to prevent backwash tank overflow. There were no personnel contaminations, no internal doses, and no radioactive releases to the environment as a result of this event. The inspector observed the follow-up discussions between the RPV contractor, DOC and CYAPCO, which included re-creation scenarios and a summary of the installed modifications. There was a good questioning attitude by the DOC and licensee with respect to lessons learned and preventative measures.

c. Conclusions

The licensee continued safe dismantlement of radiologically contaminated major components. Follow-up to the reactor cavity filtration system water overflow was thorough and prompt.

O2.2 Building Dismantlement

a. Inspection Scope (71801)

The inspector observed the dismantlement of the engineering, maintenance, and alternate security access buildings and reviewed survey records for compliance to release limits for unrestricted use.

b. Observations

The engineering modular building was surveyed, dismantled, and disposed of as non-radioactive waste during this inspection period. The inspector noted that the licensee used procedure 24265-000-GPP-GGGR-R2206-000, "Vehicle and Material Release from Radiologically Controlled Area", dated 11/4/99 (same as RPM 2.2-22 referenced in inspection report 05000213/1999-004). Direct frisk surveys included one hundred percent of all floors and walls up to two meters, ten percent of the walls above two meters, and ten percent of the ceilings. In addition, ten percent of all surveyed surface areas were surveyed for alpha contamination. A representative number of smear tests were also taken. Miscellaneous equipment from the building, such as light fixtures, air conditioning units, and desks, were direct frisked by hand or surveyed using a small article monitor (SAM) with appropriate alarm set points.

A one hundred percent direct frisk of the roof and exterior surfaces up to two meters was performed for beta contamination. Ten percent of the roof and exterior surfaces above two meters were surveyed for alpha contamination. A representative number of smear tests were also taken. Composite samples of the roof were obtained and analyzed in a spectroscopy laboratory. No radioactive material above background was identified in any of these surveys. The inspector concluded that the surveys of the engineering building performed before demolition were comprehensive and thorough.

The inspector also observed the survey and disposal of the maintenance modular building during this inspection period. The radiological surveys for fixed contamination were performed using a unique Surface Contamination Monitor (SCM)/Survey Information Management System (SIMS) developed by a licensee contractor. Detailed information describing the operational aspects of the SCM is documented in NUREG/CR-6450. The inspector reviewed the sensitivity of the instrumentation and determined that the equipment was capable of surveying surfaces and identifying radioactive material slightly above background. The software used to analyze the sample data points provides adequate evaluation to determine acceptance for release of the building from the radiologically controlled area (RCA). The inspector reviewed the licensee survey document dated March 14, 2000, entitled "Release Survey of the Maintenance Module" and discussed its results with licensee representatives. No radioactive material above background was identified. The inspector performed an independent survey of debris from the two buildings in dumpsters at different times during this inspection period and did not identify any radioactive material above background.

The alternate security access building was also dismantled during this inspection period. Survey techniques and equipment identical to those described above were used. The inspector reviewed survey results and determined that the alternate security access building met the criteria for release from the RCA prior to its dismantlement and disposal. No violations were identified and the inspector had no further questions.

c. Conclusions

The demolition and removal of the engineering, maintenance, and alternate security access

buildings were well planned, coordinated, and implemented. Survey results indicated that no detectable radioactive material above background was identified in debris from the dismantled buildings.

O2.3 Off-Site Remediation Activities

a. Inspection Scope (71801)

This inspection activity determined the status of the licensee's Offsite Material Recovery Program. This program has been active following the identification of soils and materials with low levels of residual radioactivity identified during the licensee's historical site characterization program in 1997. The inspection reviewed the summary of information that was provided to the licensee's Nuclear Safety Advisory Board on March 14, 2000 and the inspector discussed the conclusions with licensee management representatives.

b. Observations

The Offsite Material Recovery Program is projected for completion this summer. However, as part of the documentation process to close-out several properties, concerns were raised regarding the adequacy of the records to ensure that the Offsite Material Recovery Program was followed at all sites. The licensee plans to bring all closure packages to a Management Review Board. This Board, comprised of licensee senior managers, was chartered to resolve policy issues and confirm the adequacy of corrective actions for identified concerns. Additional measurements may be required at a few of the locations.

c. Conclusions

Use of a management quality review board to resolve self-identified issues is a reasonable approach. The NRC will continue to monitor the licensee's actions to close-out survey and dose projections as part of the Offsite Material Recovery Program.

O7 Self-Assessments

O7.1 CYAPCO Oversight Focus Reviews

a. Inspection Scope (40801)

The inspection reviewed the licensee's process to assess and evaluate the performance of their DOC. The DOC Oversight Manual approved December 6, 1999 describes the performance areas and indices. The inspector reviewed several selected standard evaluations and three focused assessments (worker training, fire protection and radiological effluents) that were conducted since the beginning of this year. The first quarterly report prepared by the DOC Oversight Organization to the licensee and DOC senior management, which summarized key performance indices was also reviewed.

b. Observations

The inspector determined that experienced licensee staff who formerly were responsible for the performance area conducted the evaluations. The inspector noted that a stop work order was issued on January 26, 2000, because of a system design change using a jumper to inject diatomaceous earth onto a waste processing filter had not been approved, which was identified during a standard evaluation. The majority of the evaluations did not identify concerns regarding the performance areas of occupational safety, radiological safety, environmental safety, and regulatory compliance. The quarterly report identified among other DOC feedback, concerns regarding worker compliance to an established standards and DOC identification of adverse conditions as areas for improvement.

The inspector also observed the discussions held between the licensee and the DOC during the weekly Management Review Team meetings that assessed the adequacy of the corrective actions planned or taken in response to deficiencies identified in the condition reporting system. These discussions including the assessment of issues for possible trending concerns. The inspectors observed a questioning-attitude during these meetings. In addition, the licensee provided weekly summaries of significant condition reports. Based on a review of a sampling of condition report records, the inspectors found the summaries were comprehensive.

c. Conclusions

The licensee implemented an effective program for oversight of its DOC.

III. Plant Support and Radiological Controls

R1 Radiological Protection Controls

R1.1 ALARA Controls

a. Inspection Scope (83750)

The inspection included the review of the ALARA measures taken in preparation and in follow-up to the segmentation of the RPV and removal of the RCS piping.

b. Observations

The inspector reviewed the radiological controls for RPV segmentation and discussed with

cognizant licensee representatives the exposure goals established for the project. The original dose goal for the project was 24.5 person-rem. This estimate was based on lessons learned from the Yankee Rowe reactor segmentation project which used plasma arc cutting and other industry experience in this work scope. Because of problems with plasma arc cutting, which caused significant schedule delays and increased personnel exposures, the licensee decided to use AWJ cutting technology. Based on input from its contractor, the licensee expected a total of approximately 9000 man-hours to be expended on this job. Dose reduction was expected to come from the remote set-up and use of cutting tools whereby personnel would spend most of their time in low dose areas. An improved cavity water filtration system was designed to provide better water clarity than at Yankee Rowe. Following a successful mock-up demonstration at the contractor's facility, the segmentation of the RPV internals began in March.

By May 2000 the project had encountered problems with equipment reliability, extensive health physics support coverage, and increased dose rates in the personnel working areas. Although the project was less than twenty-five percent complete, the personnel exposure exceeded the 24.5 person-rem estimate for the entire job. Radiation Safety Committee (RSC) meetings were held approximately every two weeks to ensure appropriate ALARA measures were being applied. The inspector attended these meetings during the weeks of April 3 and May 17, 2000. The most significant reason for the increased personnel exposures was identified as the increased man-hours required to perform the cutting operation. More time was being spent on the cavity bridge than expected and workers were spending much more time in 20 millirem per hour (mr/hr) to 40 mr/hr dose fields repairing cutting tools than expected. As of the RSC meeting held on May 17, 2000, the dose required to complete reactor segmentation was estimated at 127 person-rem and requiring approximately 80,000 man-hours.

The inspector reviewed with cognizant licensee representatives the reasons for the increases in man-hours and work area dose rates. The inspector verified that the application of aggressive ALARA engineering methods reduced dose rates in the area where personnel worked such that the effective dose rate was less than 2 mr per Radiation Work Permit (RWP)-hour. The original project effective dose rate used to calculate the original exposure goal was approximately 2.5 mr/RWP-hour. The ALARA controls included underwater pressure wash of tools before handling, extensive/frequent wash downs of the surfaces where plate out of radioactive material occurred, replacement of highly contaminated equipment, improvements to the collection head, and improved handling of highly radioactive trash. Although these ALARA measures were effective in reducing dose rates on the project, increased man-hours to perform functions such as loading the canisters with GTCC material and repairing cutting tools resulted in increased personnel exposure.

On May 23, 2000, the Director, Division of Nuclear Materials Safety, and NRC staff conducted a teleconference with licensee management to discuss concerns with the increasing trend in personnel exposures for the RPV segmentation project. The licensee stated that additional ALARA measures were planned to improve control of doses on the job. These ALARA measures include a reactor cavity bridge with increased shielding to reduce the dose rates where personnel work and a more efficient plan for loading canisters with GTCC material. The Director stated that the NRC staff will increase onsite inspections during the remainder of the segmentation project to ensure that all reasonable efforts continue to control personnel exposures.

Regarding the RCS piping removal project, the inspector noted that the licensee had to revise the radiation safety review five times. The inspector discussed with the licensee that basic physical parameters, such as access path, container size, and weight limitations should had

been better planned. Personnel with engineering and planning disciplines were added to the RSC meetings. The licensee also revised the qualification requirements for welders for future welding of rigging attachments to reduce the amount of re-work.

c. Conclusions

Challenges with new techniques required better planning and more frequent assessment of high person-rem activities. The RPV segmentation project has been a significant radiological challenge, resulting in greater personnel exposures than originally estimated. ALARA planning for removal of the reactor coolant system piping indicated a weakness in coordination with other departments to reduce re-work. However, for both of these major component removal activities, no exposure limits have been exceeded.

R1.2 External Exposure Monitoring System

a. Inspection Scope (83750)

The inspector evaluated the licensee's newly installed external electronic dosimetry and electronic RCA access control system. Information was gathered by a documentation, direct observation, and interviews with cognizant personnel.

b. Observations

The licensee determined in 1999 that their existing electronic dosimetry/RCA access control system was becoming obsolete and would be unable to continue efficient external exposure monitoring and RCA access control as decommissioning activities increased. See Inspection Report 50-213/99-03. During this inspection period, the licensee installed a new electronic dosimetry and access control system using the Merlin Gerin (MG) electronic dosimeters (model DMC 2000), an LDM 301 reader, a central control computer, and a video monitor. The central control computer is a server-class computer on the CY network and is capable of managing exposure data, RWP data, and personnel qualifications.

The inspector examined records of dosimeter calibrations, functional tests performed on the system, acceptance test results, and comparisons to the Eberline electronic dosimetry system being phased out. Licensee document "Health Physics Department Technical Support Document BCY-HP-0020 Revision 0" was reviewed with licensee representatives. The acceptance tests were thorough and comprehensive.

c. Conclusions

The new electronic dosimetry/RCA access control system was adequately tested and installed. The inspector concluded that the MG system is capable of monitoring personnel exposures to radiation at energies expected to be encountered at Haddam Neck will ensure NRC dose limits of 10 CFR 20 Subpart C are not exceeded and that NRC dose monitoring requirements of 10 CFR 20 Subpart F are met.

R1.3 Radioactive Waste Characterization

a. Inspection Scope (71801)

The inspector reviewed the technical basis for the characterization of waste-water processing streams generated during the RPV segmentation. Technical Support Document BCY-HP-005, dated January 18, 2000 was reviewed and discussed with the primary author and reviewer.

b. Observations

The inspector noted that the wastes generated from the RPV segmentation consists of ion exchange resin, charcoal vessels, the back-washed filters, and the cutting debris and sediments that will be collected in the A-43 High Integrity Container. Because of the high level of activity in the reactor internal components, the licensee implemented a dose-rate cut-off value to ensure that GTCC waste is not generated from the above waste streams. Based on RCS decontamination resin samples and RPV sediment samples, the licensee estimated that approximately 5500 Curies will be generated from water processing these waste streams.

The inspector compared the licensee assumptions to the safety evaluation calculation that was completed to evaluate the resin container accident scenario for the Update Final Safety Analysis Report where the licensee determined that the resin fire is the new bounding accident. No concerns were identified. The inspector also noted that additional sampling may be necessary if the expected source term were to change based on changes to the segmentation process.

c. Conclusions

The waste-water processing streams from the RPV segmentation project were adequately characterized.

IV Maintenance Surveillance and Engineering

M1 Conduct of Maintenance

a. Inspection Scope (62801)

A review was performed of the licensee's program to ensure that the maintenance and surveillance of structures, systems and components (SSCs) important to safety, and proper operation of radiation monitoring and effluent control equipment, are being effectively conducted.

b. Observations

The inspector reviewed the following surveillance packages for effectiveness and detail:

MET Tower	Quarterly PM package
R-18 (River Effluent Monitor)	Quarterly PM and 18 month calibration packages
R-1 (SFB Vent Monitor)	Quarterly PM and 18 month calibration packages
SFB Cooling Pumps	6 month PM

These systems and components are the direct responsibility of the CY organization. The inspector verified that an adequate program is in place to plan and schedule surveillances and work orders are generated to ensure timely completion of required surveillances and tests. No violations of NRC requirements were identified.

The inspector reviewed the program established by the DOC to ensure SSCs not related to the SFPB and scheduled for demolition but still functional are maintained and surveillance is completed at an appropriate frequency. The DOC has established a preventive maintenance program in procedure 24265-000-GPP-GGGM-00003 entitled "Maintenance Procedure". It provides the scheduling and tracking requirements for DOC controlled SSCs. Maintenance procedure 24265-000-GGP-GGGM-00002 entitled "Corrective Maintenance Program" describes the requirements of the DOC's corrective maintenance program and the Work Plan and Inspection Records procedure. This is the DOC equivalent to the licensee's work order program. The inspector determined that the DOC has established an adequate system for maintaining equipment important to the safe decommissioning of the plant.

The inspector reviewed the following DOC surveillance packages for effectiveness and detail:

Main Stack Vent Flow	18 month surveillance
Containment Polar Crane (CR-1-1A)	Quarterly PM
Ion Exchange Crane (CR-17-1A)	Annual PM

The Ion Exchange Crane Annual PM identified a problem with the system's braking mechanism. The inspector verified that the Ion Exchange Crane was blue tagged out of service and had no further questions.

c. Conclusions

Required surveillances important to the safe storage of spent fuel are being effectively scheduled and completed in a timely manner. Surveillances are being adequately performed on SSC's important to the safe decommissioning of the plant.

E.8 Miscellaneous Engineering Issues

E8.1 Application of 10 CFR 50.59

a. Inspection Scope (37801)

The inspection reviewed the licensee's safety review program to verify that major and minor decommissioning activities do not involve an unreviewed safety question or change to TSs in conformance with 10 CFR 50.59. Implementation of ACP1.2-2-42, "10 CFR 50.59, Applicability Reviews and Safety Evaluations" was reviewed with respect to heavy load drops on the reactor cavity seal, RPV internals segmentation, and a vendor waste processing tie-in.

b. Observations

ACP 1.2-2.42 provides the requirements, guidance, and examples for preparing applicability reviews and safety evaluations for the current defueled condition. It specifies that changes are evaluated if the change directly or indirectly affects the spent fuel cladding performance, or the change involves changes in the movement, treatment or storage of radioactive fluids, radioactive gases or other wastes, or radioactivity of equipment being decontaminated or dismantled that affect the UFSAR accident analyses. The inspector noted that application of the 10 CFR 50.59 process to major dismantlement activities is often limited because these systems are considered abandoned and no longer included in the Final Safety Analysis Report.

CY Safety Evaluation SY-EV-98-0037, dated June 9, 1998, evaluated the consequences of heavy load drops on the reactor cavity seal. It concluded that no TS change was required and no USQ existed. CY's 10 CFR 50.59 Applicability Review for the RPV internals segmentation (CY Doc. No. 24265-000-DCP-000028-000) concluded that the activity was bounded by previously performed safety evaluations for the resin fire and heavy load drops and also based on direct dose rate calculations that were evaluated for a beyond design basis of a SFP drain-down. Based on NRC comments regarding the completeness of these reviews, the licensee completed SY-EV-00-0019, dated June 1, 2000 for the RPV segmentation, which included a re-assessment of heavy loads using the current Technical Specification Basis and a consideration of the amount of direct radiation resulting from uncovering the internals due to draining the refueling cavity. No further actions were required.

SY-EV-99-0033 GTS Wastewater Processing Tie-In dated December 16, 1999 properly evaluated the proposed processing of contaminated water from the station tanks during reactor vessel segmentation. Procedures require that curie loading on the filter and the demineralizer will not exceed that assumed in the resin fire so that there is no increase in the probability of a resin fire during the de-watering. Also, the evaluation found that any spill from this tie-in is bounded by the liquid radioactive waste accident which assumes spill of a full liquid radioactive waste tank (250,000 gallons) to the environment. The inspector agreed with the planning assumptions used for this evaluation.

c. Conclusions

The licensee implemented a safety review of major and minor decommissioning activities to determine if they involved an unreviewed safety question or a change in TS. However, there were two changes, specifically movement of heavy loads and RPV segmentation, that were re-evaluated under 10 CFR 50.59 to ensure current TS and design basis accidents were considered.

E8.2 Status of Previous Inspection Items

(Closed) Unresolved Item 99-02-01: Incorrect Control Room Habitability Information Submitted for Updated Final Safety Analysis Report (UFSAR) . On March 2, 1999, in accordance with the requirements of 10 CFR 50.72(b)(1)(ii)(B), the licensee notified the NRC of a potential error in the calculation of the dose to control room personnel during a postulated resin fire accident as described in the UFSAR. The details regarding the technical issues and the resolution of those details was previously documented in NRC Inspection Report 05000213/1999-002, Section O8.1, dated August 20, 1999. The control room habitability information submitted on December 1997 for the UFSAR change for a modification to the control room based on a dose calculation to the control room personnel was incorrect. This unresolved item involved a determination of whether there was wrongdoing on the part of the licensee with respect to this error, since the calculation was not performed as a design controlled calculation. The status of the licensee's corrective actions recommended by its Root Cause Investigation report dated May 1999 was also inspected.

During the week of February 7, 2000 and on February 22, 2000, an NRC investigator with assistance from an NRC inspector conducted interviews with individuals responsible for requesting, preparing, reviewing, approving and processing the calculation in question. Pertinent documentation was acquired from the licensee and was also reviewed by the NRC. After careful consideration of the evidence developed during this investigation, the NRC concluded that there was not sufficient evidence of wrongdoing to warrant further investigative efforts in this matter. The inspector determined that the licensee's corrective actions to preclude a reoccurrence were completed.

The inspector also found that the changes to the UFSAR had no effect on the dose consequences to a member of the public as a result of a resin accident. In addition, the changes did not result in a significant increase in potential occupational dose to personnel in the control room during and following a resin accident.

Based on these findings, although the licensee failed to provide accurate information to the NRC in December 1997 regarding the basis to change the licensee's UFSAR, this failure to comply with 10 CFR 50.9 constitutes a violation of minor significance and is not subject to formal enforcement action. This item is closed.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management periodically during the inspection, and during an exit meeting held on-site with Mr. K. Heider and others on May 18, 2000. The licensee acknowledged the findings presented by the inspector. The inspector reviewed with the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

X2 Management Meetings

On March 16, 2000, the NRC Regional Administrator and the Chief, Decommissioning and Laboratory Branch, toured the site and met with licensee management personnel.

NRC staff attended Community Decommissioning Advisory Committee (CDAC) meetings on

March 21, 2000 and May 16, 2000.

PARTIAL LIST OF PERSONS CONTACTED

G. Bouchard, (Former) Unit Director (retired from CYAPCo)
*J. Bourassa, Nuclear Safety Manager
M. Cavanaugh, Communications Manager
H. Farr, Radiological Engineer, Bechtel
*N. Fetherston, Site Manager
P. Hollenbeck, Site Characterization Supervisor
*K. Heider, Vice President Operations and Decommissioning
*P. K. Jackson, Assistant Project Manager, Bechtel
S. Kumar, Regulatory Affairs
*R. McGrath, Radiological Engineering Supervisor
*R. Daly, Project Manager, Bechtel
*P. Labasta, Engineering, Bechtel
*C. D. Melin, Contracts Manager
*R. Mitchell, Unit Manager
*G. van Noordennen, Regulatory Affairs Manager
F. Perdomo, Regulatory Affairs
*M. Powers, Construction Oversight
*R. Prunty, Licensing, Bechtel
*B. Reilly, Vice-President, Bechtel
*R. Sexton, Health Physics and Safety Oversight Manager
*J. Tarzia, Radiation Protection and Chemistry Manager, Bechtel
T. Troutman, Transition Manger, Bechtel

* Denotes attendance at the exit meeting held on May 18, 2000.

INSPECTION PROCEDURES USED

IP 37801: Safety Reviews, Design Changes, and Modifications
IP 40801: Self Assessment, Audits and Corrective Actions
IP 60705: Spent Fuel Inspections
IP 62801: Maintenance and Surveillance
IP 71801: Decommissioning Performance and Status Review
IP 83750: Occupation Radiation Exposure Controls

ITEMS OPEN, CLOSED, AND DISCUSSED

Open

None

Closed

URI 99-02-01 Incorrect Control Room Habitability Information

LIST OF ACRONYMS USED

ALARAAs	Low As Reasonably Achievable
AWJ	Abrasive Water Jet
CDAC	Community Decommissioning Advisory Committee
CFH	Certified Fuel Handler
CFR	Code of Federal Regulations
CR	Condition Report
CYAPCo	Connecticut Yankee Atomic Power Company
DOC	Decommissioning Operations Contractor
GTCC	Greater than Class C
IR	Inspection Report
MDM	Mechanical Disintegrations Machining
MG	Merlin Gerlin
MMOD	Minor Modification
NAC	NAC International
NRC	Nuclear Regulatory Commission
PDR	Public Document Room
QA	Quality Assurance
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RWPs	Radiation Work Permits
SAM	small article monitor
SCM	Surface Contamination Monitor
SFP	Spent Fuel Pool
SFPB	Spent Fuel Pool Building
SIMS	Survey Information Management System
SSC	Systems, Structures, and Components
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item