U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-29
License No.	DPR-03
Report No.	50-29/96-04
Licensee:	Yankee Atomic Electric Company 580 Main Street Bolton, Massachusetts 01740-1398
Facility Name:	Yankee Nuclear Power Station
Location:	Rowe, Massachusetts
Dates:	November 19 - 21, 1996
Inspectors:	J. Nick, Region I Radiation Specialist M. Fairtile, NRR Project Manager
Approved By:	J. White, Chief, Radiation Safety Branch Division of Reactor Safety

Report Details

O1 SUMMARY OF FACILITY OPERATIONS

On October 12, 1995, the Commission served Memorandum and Order CLI-95-14 to Yankee Atomic Electric Company (YAEC) concerning activities at the Yankee Nuclear Power Station (NPS) in Rowe, Massachusetts. The Order states, in part, that "the NRC's approval of the Yankee NPS Decommissioning Plan cannot be accorded further legal effect, pending an [adjudicatory] hearing opportunity," and, that in accordance with the pre-1993 interpretation of the decommissioning regulations, "the Commission expects YAEC not to conduct any further 'major' dismantling or decommissioning activities until final approval of its [decommissioning] plan after completion of the hearing process." Subsequently, the Commission issued an Order (CLI-96-9), dated October 18, 1996, which granted YAEC's Motion for Summary Disposition in a hearing convened to determine whether the decommissioning plan should be approved. Since YAEC had originally submitted the decommissioning plan before the Commission amended its decommissioning regulations, and the decommissioning plan was approved by the NRC in February 1995, YAEC had been given approval to conduct decommissioning activities at the Yankee site per a letter from the NRC (reference correspondence, dated October 28, 1996, from Mr. Morton Fairtile to Mr. James Kay).

Based upon issuance of the Order and correspondence, the inspectors observed YAEC's activities during routine inspections on November 19 through 21, 1996. The inspectors verified by observation, documentation review, and/or discussions with responsible or involved plant staff, the activities on the current work schedule and the actual activities being conducted at the site. No major safety concerns were identified by the inspectors and appropriate radiological and industrial safety practices were observed for those jobs in-progress.

O2 Operations

O2.1 Facility Tours

The inspectors toured radiological controlled areas (RCAs) within the vapor containment. The inspectors noted operating air sampling equipment in various areas. Personnel dosimetry was worn by all workers in the area. Workers had removed most of the mechanical and electrical components (duct-work, conduit, cables, and fan units) outside the bioshield, in the vapor containment. The inspectors observed that high radiation area (HRA) and locked high radiation area (LHRA) controls were satisfactory. All areas were posted, barricaded, and locked as required by NRC regulations and Technical Specifications. The posting and labelling of radioactive material were satisfactory. Very good radiological controls were provided by health physics technician coverage for jobs/activities in RCAs.

The inspectors toured most of the RCAs outside the vapor containment including the primary auxiliary building (PAB), the service building, the radioactive waste processing (compactor) building, the potentially contaminated area (PCA) storage building (a storage/staging area for potentially contaminated equipment and materials), and the PCA warehouse attached to the radwaste processing building.

Workers had previously assembled a temporary waste processing system near the ion exchange pit. The temporary waste processing system included holding tanks and an evaporator which were planned to replace the existing waste evaporator system. The licensee had processed the first batch of contaminated water through the processing system without any major problems.

Other electrical and mechanical systems removal was continuing in the Primary Auxiliary Building (PAB). All radiation areas (RAs) and HRAs were posted and barricaded as required. Locked HRAs were maintained locked with appropriate warning signs. Housekeeping in contaminated areas was good, and contamination control was evident by the use of "step-off pads", personnel monitoring equipment (friskers), and contaminated area postings at the boundaries. No safety or NRC regulatory concerns were noted by the inspectors.

O2.2 Current Activities

The removal of the reactor vessel was the main work activity during the period of this inspection. The main coolant piping had previously been cut to remove small sections where the piping was attached to the reactor vessel. Other preparations had been performed including the consolidation and solidification of dross into a liner within the reactor vessel. The vessel was to be removed in one large piece. Accordingly, the reactor vessel was prepared for lifting by placing a temporary cover and lead shielding on the top of the vessel. A lifting device was attached to provide two lift points. The vessel was raised out of the reactor cavity, moved over the refueling floor, and lowered down through the equipment hatch to a specially-designed cask. The cask had been manufactured and brought to the site earlier in the year. The NRC has approved the cask as a shipping container. Shipment of the reactor vessel to the disposal site near Barnwell, South Carolina was planned for Spring 1997. Until shipment, the vessel will be stored in the sealed cask at the Yankee site.

Asbestos abatement work was continuing on the secondary side in the turbine building and the old radwaste evaporator system. Secondary and support systems removal was continuing in the turbine building. Renovations of the main gatehouse were continuing and near completion. The licensee planned to move the control room functions to the gatehouse to consolidate activities in one building with the security functions in early 1997.

Other work planned for early 1997 included completion of the items mentioned above, removal of the main coolant piping, removal of the upper neutron shield tank, removal of structural steel outside the auxiliary building, decontamination of the ion exchange pit (including removal of contaminated concrete), removal of the old radwaste evaporator system, and starting the final site survey project.

E1 Inspection of Licensee's 10 CFR 50.59 Safety Evaluations

E1.1 General Comments (37001)

Using Inspection Procedure 37001, and additional NRC guidance on environmental impacts and decommissioning costs, the inspectors determined that the three plant modifications, discussed herein, and performed by the licensee under the 1996 Engineering Design Change Requests (EDCRs), issued through November 1996, were conducted under 10 CFR 50.59 and other pertinent NRC requirements. The licensee properly included a 10 CFR 50.59 safety evaluation in each EDCR. The NRC has required shut down plants to also assess plant modifications for any environmental impact not previously evaluated and to verify that the modification would not cause an unanticipated increase in decommissioning costs.

E1.2 Purpose of Inspection

The inspection was performed to verify that the licensee properly followed the safety criteria of 10 CFR 50.59 and the additional NRC requirements outlined above.

The three subject 1996 EDCRs inspected by the NRC were:

EDCR 95-302, Revision 2 - Yankee Reactor Pressure Vessel Removal (completed)
EDCR 96-302 - Control Room Relocation (ongoing)
EDCR 96-303 - Temporary Waste Water Processing Island (completed)

E1.3 Overall Findings on These Three Plant Modifications

Based on the inspection of these three EDCRs and of the actual modifications, the inspectors determined that these changes do not involve an unreviewed safety question as defined in 10 CFR 50.59(a)(2). The inspectors also found that these changes met all other NRC guidance on decommissioning costs and environmental impacts. The inspectors observed the reactor pressure vessel removal on November 20, 1996. This evolution was carried out in a very safe manner and proceeded according to plan. the licensee informed the inspectors that the vessel in its storage/shipping canister will not be shipped offsite until early spring 1997. The NRC plans to inspect this shipment.

R1 Plant Support - Radiological Controls Program

R1.1 External Exposure Control

a. Inspection Scope (83100)

The inspectors reviewed the controls for external radiation exposure through observation of work activities, tours of the facility, interviews with personnel, and a review of licensee documents.

b. Observations and Findings

The inspectors observed the reactor vessel removal from the vapor containment and various other work activities during the period of this inspection. Personnel in the RCA were observed wearing their assigned thermoluminescent dosimeter (TLD) and pocket ion chamber (PIC) dosimeter. The dose totals for each individual were tallied on each workday and reports were available for review by personnel. Plant management periodically reviewed the status of workers in the respective departments.

As stated in Section O2.1 of this Report, controls for radiation areas and high radiation areas were appropriate throughout the facility. In addition, the temporary controls for the reactor vessel removal project were very good. During the lift of the vessel, technicians were stationed in various areas to monitor radiation dose rates and restrict access as necessary. Actual dose rates were not as high as expected from the reactor vessel, but barriers and postings were used appropriately to prevent inadvertent entry into areas with elevated dose rates. The inspectors verified that the proper controls were placed on all sides of the reactor vessel cask.

Radiation work permits (RWPs) and a computerized access control system were also used to control workers' radiation exposure. The inspector reviewed selected RWPs written for various work activities and concluded that they contained appropriate requirements including administrative dose limits, protective clothing, and special monitoring or dosimetry.

c. Conclusions

Controls for external radiation were very good including temporary controls used during the reactor vessel removal from the vapor containment. No violations of NRC regulations or safety concerns were identified.

R1.2 Internal Exposure Control

a. Inspection Scope (83100)

The inspectors reviewed the controls for internal radiation exposure through observation of work activities, tours of the facility, interviews with personnel and a review of licensee documents.

b. Observations and Findings

The inspectors observed air sampling equipment in the vapor containment during the reactor vessel removal. The equipment was positioned to provide representative sampling of the breathing air in areas occupied by workers. Air sampling was provided for workers in the reactor cavity preparing for lifting the reactor vessel and workers on the refueling floor. Although the airborne radioactivity was expected to be negligible, all workers in the vapor container were respirator qualified, and ready to don respirators if needed, due to asbestos concerns. In addition, air handling and filtration equipment was used in areas with potential airborne radioactivity.

The licensee had recently upgraded the software and hardware used to evaluate internal bioassay results. A new computer system had been installed and tested prior to the period of this inspection. The inspectors noted the improvements to the bioassay system.

The inspectors reviewed the results from internal dose assignments and determined that the dose assigned through air sampling and bioassay were very small when compared to the total dose assignment.

c. Conclusions

The licensee had provided good controls for internal radiation exposure including air sampling and bioassay for dose assessment. No violations of NRC regulations and no safety concerns were noted.

R1.3 Control of Radioactive Materials and Contamination, Surveys and Monitoring

a. Inspection Scope (83100)

The inspectors reviewed the controls for radioactive materials and contamination, surveys and monitoring through observation of work activities, tours of the facility, interviews with personnel and a review of licensee documents.

b. Observations and Findings

The inspectors verified that there was an adequate supply of radiation survey and monitoring equipment. All equipment checked by the inspector was operable and within the current calibration period. Portal monitors and frisking instruments were located throughout the facility for use by workers as they left radioactive materials areas or contaminated areas. Current radiological surveys of various work locations were reviewed by the inspector. The surveys contained detailed information regarding dose rates and hazards in the work areas. Surveys were posted at the main control point for the RCA and at the vapor containment. Appropriate licensee management personnel had reviewed the radiological surveys.

The inspectors observed a technician obtaining dose rate measurements on the outside of the reactor vessel cask after loading. The technician performed the survey appropriately with the proper survey instrument. Dose rates were taken on contact with the cask and in the general areas around the cask.

Radiological housekeeping was good throughout the plant with appropriate controls established to minimize the spread of contamination. Posting of radioactive material areas and labelling of radioactive materials was appropriate. Very good controls were established to prevent the spread of contamination during the reactor vessel removal. Plastic sheets were taped between the bottom of the equipment hatch and the top of the cask to enclose the area. Unnecessary material and items were minimized in the vapor container and other contaminated areas.

c. Conclusions

The licensee provided very good controls for radioactive materials and contamination, surveys and monitoring during decommissioning work activities. No violations or significant safety concerns were identified.

R1.4 Maintaining Occupational Radiation Exposures ALARA

a. Inspection Scope (83100)

Through interviews with personnel and review of several documents, the inspectors examined the program to maintain personnel exposures ALARA.

b. Observations and Findings

The personnel working at the Rowe site received a total radiation exposure of approximately 85 person-rem during the period from January 1, 1996 through November 19, 1996, as measured by thermoluminescent dosimeters (TLDs) and pocket ion chambers (PICs). The highest total dose assignment to an individual at the site was 1.564 rem though November 19, 1996. This is much below the NRC total annual dose limit to an occupational worker of 5.0 rem. The projected total exposure during 1996 was 91 person-rem for routine work and support of decontamination and decommissioning activities. Additional exposure was projected for dismantlement of the reactor vessel, the main coolant system, various other systems in the vapor containment, balance of plant systems, and other structures. Some additional exposure was projected for asbestos abatement and radioactive waste shipments. The total dose to personnel working on all activities at the Rowe site for the period from 1993 through August 1996 was approximately 482 person-rem. The total dose for decommissioning and dismantlement activities, including support work, was approximately 440 person-rem.

The inspector reviewed the planned ALARA initiatives for the reactor vessel removal. These initiatives included limiting the number of personnel in the vapor containment by establishing remote video viewing locations, erecting a lead curtain to shield the crane operator in the vapor containment, placing lead shielding on top of the reactor vessel, erecting water shield at the entrance through the bioshield wall for the crane load director, filling the auxiliary boiler tank with water to provide shielding in the boiler room, performing a dry run to identify potential problems, and using cameras for required surveys of the reactor vessel condition. The initiatives were well-planned and contributed to lower total dose assignments than expected. The entire reactor vessel removal (excluding some preparations) was expected to be nearly 6 person-rem. The actual dose was approximately 4 person-rem. This low dose assignment is significant due to the high total activity contained in the reactor vessel (approximately 4500 curies) and high expected dose rates (1 rem per hour at 5 feet from the unshielded vessel) during the reactor vessel removal.

c. Conclusions

The licensee continued to maintain an very good program for maintaining occupational radiation exposures ALARA. Effective ALARA initiatives were implemented for the reactor vessel removal project and other activities. Total occupational radiation exposure to workers at the Yankee site for the year to date was less than the projected total for the year.

R7 Quality Assurance in Radiation Protection Activities

R7.1 Audits and Appraisals

a. Inspection Scope (83100)

The inspectors reviewed the licensee's programs and systems for auditing and appraising the program for control of radiation and radioactive materials through examination of records and interviews with licensee personnel.

b. Observations and Findings

There had been no new audits of the radiation protection program since the last inspection; however, the inspectors noted recent audits of the special nuclear materials control program, the training program, the industrial safety program, the environmental program, the REMP/RETS/ODCM programs, and the corrective action system. Appropriate and timely corrective actions had been taken by the licensee's staff for the minor deficiencies and weaknesses that were identified during the audits performed by the quality assurance group.

The quality assurance group was continuing to perform surveillance of radiological work activities and waste shipments. The inspectors reviewed surveillance report regarding various radiological work activities. The surveillance reports indicated that the implementation of the radiation protection program was satisfactory.

c. Conclusions

The inspectors concluded that the audit, self-assessment, and corrective action programs were continuing to identify problem areas and improve the quality of the radiation protection program. No violations were identified in this area.

X1 Exit Meeting Summary

The inspectors met with the licensee representatives denoted below at the conclusion of the on-site inspection on November 21, 1996. The inspectors summarized the purpose, scope, and findings of the inspection. The licensee representatives acknowledged the inspection findings.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

*G. Babineau, Radiation Protection and Chemistry Manager
*W. Blackadar, Radiation Protection Engineer
H. Breite, YNSD, Lead Engineer
*W. Cox, YNSD, Radiation Protection Engineer
*R. Durfey, Senior Engineer/Maintenance and Construction
*N. Fetherston, Maintenance and Construction Manager
*R. Greenfeld, Radiation Protection Engineer/ALARA Program
R. Grippardi, YAEC, Quality Assurance Supervisor
*K. Heider, Site Manager
*S. Litchfield, Health and Safety Supervisor
*R. Mellor, YNSD Decommissioning Manager (via telephone)
S. Mullet, Radiation Protection Technician
*A. Trudeau, Quality Services Group Senior Engineer
*M. Vandale, Radiation Protection Senior Engineer
*B. Wood, Assistant Site Manger
F. Williams, Operations Manager

* Denotes those individuals participating in the exit briefing held on November 21, 1996.

INSPECTION PROCEDURE USED

IP 83100: Occupational Radiation Exposure During Decommissioning
IP 37001: 10 CFR 50.59 Safety Evaluation Program

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NONE

Closed

NONE

Discussed

NONE