CONNECTICUT YANKEE ATOMIC POWER COMPANY

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

June 11, 1997
Docket No. 50-213
CY-97-059

Re: 10CFR2.201

Director, Office of Enforcement
U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Haddam Neck Plant
Reply to a Notice of Violation (NOV)
Inspections 50-213/95-27, 96-06, 96-07, 96-08, 96-11, 96-80 & 96-201

In a letter dated May 12, 1997,(1) the NRC identified specific violations resulting from numerous inspections conducted at the Haddam Neck Plant from November 21, 1995 through November 22, 1996. These inspections included a special team inspection focused principally on engineering performance, a special Augmented Inspection Team inspection of a reactor vessel nitrogen intrusion event, an emergency preparedness inspection, and four resident inspections. The violations identified were grouped into several broad categories, namely, long-standing deficiencies in engineering programs and practices, including plant design, design control, and engineering support; operational deficiencies, including inadequate procedures, failure to follow procedures, and inadequate corrective actions; and inadequate implementation of the emergency preparedness program. The purpose of this letter is to provide CYAPCO's response to the violations.

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(1) Hubert J. Miller to Bruce D. Kenyon, "Notice of Violation (NRC Inspection Reports Nos. 50-213/95-27, 96-06, 96-07, 96-08, 96-11, 96-80, 96-201)," dated May 12, 1997.

On December 4, 1996,[2] an enforcement conference was held at which time the noted violations were discussed. At that time, the results of our common cause investigations were presented. These studies identified corrective actions specific to the violations as well as corrective actions for the programmatic and organizational common cause issues. Appropriate corrective actions have been initiated and continue to be implemented.

Since the December 4, 1996 enforcement conference, CYAPCO submitted a letter dated December 5, 1996,[3] that informed the NRC that the Board of Directors of CYAPCO had decided to permanently cease operations at the HNP and that fuel had been permanently removed from the reactor. Due to this change in the plant status, CYAPCO has focused it's efforts to define the plant licensing and design basis for the defueled condition, as described in our 10CFR50.54(f) letter dated February 6, 1997.[4] Our response to the violations reflects this focus and demonstrates our resolve to strengthen plant programs and organization to safely store spent nuclear fuel and decommission the facility.

Attachment 1 provides CYAPCO's overall reply to the Notice of Violation and summarizes the current implementation status of corrective actions. Attachment 2 provides our response to the individual violations pursuant to the provisions of 10CFR2.201. Attachment 2 also identifies where systems associated with identified violations are not required and associated specific corrective actions are no longer applicable due to the current defueled and permanently shutdown plant configuration.

In summary, CYAPCO takes these violations very seriously and is committed to implement and complete the broad scope corrective actions to improve station performance. We will continue to keep the NRC Staff informed on our progress in these areas and are committed to demonstrating improving trends in areas of past weakness.

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(2) John F. Rogge to Ted C. Feigenbaum, "Haddam Neck Enforcement Conference [docketed information]," dated December 20, 1996.

(3) T. C. Feigenbaum letter to the U.S. Nuclear Regulatory Commission, "Certifications of Permanent Cessation of Power Operation And That Fuel Has Been Permanently Removed From the Reactor," dated December 5, 1996.

(4) T. C. Feigenbaum letter to U. S. Nuclear Regulatory Commission, NRC Request for Information Pursuant to 10CFR50.54(f)," dated February 6, 1997.

On June 10, 1997, CYAPCO paid the full amount of $650,000.00 of the proposed civil penalty by electronic transfer to the Treasurer of the United States.

Should you have any questions regarding this submittal, please contact Mr. G. P. van Noordennen at (860) 267-3938.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

T C Feigenbaum

T. C. Feigenbaum
Executive Vice President and
Chief Nuclear Officer

Attachments

cc: H. J. Miller, NRC Region I Administrator
M. B. Fairtile, NRC Project Manager, Haddam Neck Plant
W. J. Raymond, NRC Senior Resident Inspector, Haddam Neck Plant
K. T. A. McCarthy, Director, CT DEP Monitoring and Radiation Division

Subscribed and sworn before me

this 11 day of June, 1997

Susan Allen

Date Commission Expires: 12/22/98

Docket No. 50-213
CY-97-059

Attachment 1

Haddam Neck Plant
Overall Reply to Notice of Violation (NOV)
Inspections 50-213/95-27, 96-06, 96-07, 96-08, 96-11, 96-80 & 96-201

June 1997

Overall Response

This attachment presents CYAPCO's overall response to the Notice of Violation (NOV) and summarizes the results of investigations to identify the common causes that underlay the conditions that contributed to the identified violations. Each of the four major sections of the NOV are addressed.

I. Violations Related to Inadequate Engineering

A common cause study was conducted to evaluate the apparent violations identified in NRC Inspection Reports 96-201, 96-06, and 96-08 and to identify common causes associated with these apparent violations. The results of this study were presented at an enforcement conference on December 4, 1996.

Both managerial and programmatic common causes were identified. Also, common to most problems were weaknesses in the implementation of configuration management and corrective action processes. Recommendations for corrective action were then developed.

Managerial deficiencies were identified as follows:

- Failure to establish and promulgate adequate standards and expectations for plant staff consistent with the continually rising standards for excellent performance in the nuclear industry, and

- Inadequate resource planning and management, especially in the areas of outage planning, maintenance of plant programs, and in the implementation of new programs and processes.

Programmatic weaknesses were identified in several areas. Lack of an effective configuration management program led to poorly identified linkages and interface requirements between various plant programs and processes. This affected the design control process and various plant procedures (e.g., Safety Evaluations, UFSAR Update, Safety Classification, Instrument Setpoint, and Procurement Engineering).

The documents presenting the design and licensing bases of plant structures, systems and components (SSCs) were not accurate and not readily available. The importance of maintaining and using accurate licensing and design basis information was not recognized by all personnel. Ineffective training may have contributed to this problem.

The corrective action program was not effective in the identification and resolution of plant problems. The lack of an effective program to consistently identify problems at a low threshold, to evaluate these problems, identify their

causes, and implement effective corrective actions to prevent recurrence was a major contributor to the degraded condition of plant design and licensing basis information. The lack of a good corrective action program also contributed to management's failure to detect and correct these deficiencies in a timely basis.

Corrective actions have been developed in the following common cause areas:

- Programmatic
  - Licensing/Design Basis Information
  - Configuration Management
  - Corrective Action Program
  - Procedural Deficiencies
  - Design Control Process

- Organization
  - Management Standards and Expectations
  - Resource Management
  - Training Programs

The following describes the corrective actions for each of these common cause areas and summarizes the current implementation status.

Licensing/Design Basis Information

This common cause area can be generally characterized as accurate licensing/design basis information not being readily available to support engineering activities. As a result, incorrect assumptions and errors were made in design changes, safety evaluations and operability determinations. Based on the magnitude and seriousness of this cause, a comprehensive Configuration Management Program was initiated to identify and verify the licensing and design bases. This effort is focused on the defueled configuration of the plant with an emphasis on maintaining the integrity of the spent fuel and spent fuel pool cooling. The following approach has been established:

- Review the basic calculations for the hydraulics, structures, electrical distribution and instrumentation.
- Review docketed correspondence to establish the licensing and design basis for each defueled system.
- Identify important attributes including flow, pressure, voltage, level, etc.
- Reconcile the design basis and attributes with the UFSAR, operating and surveillance procedures, and Technical Specifications.
- Review nonconformance reports and design changes to verify the design basis was not inadvertently changed.

- Perform walkdowns to verify expected configurations.
- Document the licensing and design basis.

All licensing commitments (over 13,000) from docketed correspondence have been identified and are currently being reviewed for applicability to the defueled condition. An upgraded procedure to capture future commitments has been drafted and is under review. A procedure to screen and prioritize past commitments has also been drafted.

The Chapter 15 accident analyses are being updated for the defueled configuration. Using safety evaluations, the UFSAR is updated and all associated Technical Specifications, procedures and programs are being changed.

The accident reanalyses will be reflected in an interim internal UFSAR revision in July, 1997. The licensing and design basis document for the major defueled systems, structures, and components will be developed in June. The reconciliation with UFSAR, operating and surveillance procedures, and Technical Specifications will be completed in the third quarter.

An UFSAR submittal reflecting a comprehensive description of the defueled condition will be submitted by December 1997. This revision will reflect those systems required and any systems and components no longer required for defueled conditions as well as correct known deficiencies associated with the defueled condition.

Configuration Management

The deficiencies identified affected our ability to maintain compliance with the plant design and licensing basis. Linkages were not adequate to assure that as one document changed, all other corresponding documents were known and subsequently changed or updated. As corrective action, the following procedures, programs and processes are being reviewed and revised to ensure the proper linkages are maintained:

- Accident analyses
- Plant design change process
- 10CFR50.59 safety evaluation process
- UFSAR updating process
- Design calculation process
- Licensing basis document change process

Programmatic changes and responsibility assignments to maintain linkages are being made. Configuration control will be maintained by the Engineering Department.

The plant safety evaluation procedure has been revised and issued. Similarly, the plant procedure for control of UFSAR revisions according to 10CFR50.71(e) has also been revised and issued. Training on both of these procedure revisions has been completed.

The revised processes for design changes and calculations will be developed in the second quarter of 1997 and implemented by the end of the third quarter of 1997. The procedure for preparation of the licensing and design bases is complete and the procedure for maintaining licensing and design bases documents will be completed by the third quarter.

Corrective Action Program

Weaknesses in the corrective action program allowed adverse conditions to exist without resolution and in some cases without detection. Weaknesses were identified in the following areas:

- Timeliness in problem identification
- Tracking of corrective actions
- Evaluation of corrective action effectiveness
- Guidance on initiation of adverse condition reports
- Root causes
- Key performance indicators

The following corrective actions have been or are going to be implemented:

- A dedicated individual has been assigned as the Corrective Action Manager. A dedicated staff has also been assigned to the program.
- A consistent approach to root cause evaluation is being monitored.
- A trending and monitoring program is being implemented.
- Benchmarking against plants known to have excellent programs will be conducted after the new process is developed and implemented in the third quarter.
- A new tracking system that is more user friendly and simpler will be completed in June and implemented in the third quarter following completion of training.

A new simpler Adverse Condition Report (ACR) process and new ACR database are under development and use a simpler action tracking system to replace the current system for tracking corrective actions.

The procedure for adverse condition reporting will implement improvements in the following areas:

- Tracking of corrective actions
- Evaluating effectiveness of corrective actions
- Providing guidance and training on initiation of adverse condition reports
- Developing more consistent root causes
- Using key performance indicators to monitor and trend corrective action effectiveness.

New procedures will include standardized causal factors coding to provide the necessary data to facilitate recognition of programmatic or recurring causes and evaluate the effectiveness of corrective actions (using the newly developed key performance indicators and trending program).

The process and procedure revisions are on schedule for implementation by the end of the second quarter 1997. Effectiveness of the revised corrective action program will be measured by the onsite Nuclear Oversight Group and by an independent third-party review by the fourth quarter.

Procedural Deficiencies

Procedure deficiencies have resulted in inconsistent implementation of processes resulting in errors and incomplete work. Procedure deficiencies were found in:

- Non-conformance reports
- Safety classification
- Safety evaluations
- Procurement process
- Instrument setpoint calculations

As corrective action, the above procedures are being revised to assure a technically correct and consistent approach is taken for the defueled condition. The safety evaluation procedure is complete and revision of the remaining procedures will be completed in the third quarter.

Design Control Process

Design control deficiencies resulted in errors being transferred from one document to another without discovery. Design control errors were found in plant modifications and calculations associated with setpoints, electrical loads, and hydraulics. The following corrective actions are being implemented:

- Conduct reviews of defueled setpoints.

- Conduct reviews of defueled hydraulic calculations.
- Conduct reviews of defueled electrical load calculations.
- Conduct reviews of defueled structural load calculations.

The defueled setpoints, if any, as well as defueled hydraulic calculations, electrical calculations and structural calculations will be included in the defueled systems licensing and design basis documents.

The existing Design Control Process is being revised to provide a plant-specific document, structured and formatted to more appropriately control design changes associated with the safe storage of spent fuel, as well as decommissioning activities.

A family of design control procedures is being developed to comprise a Design Control Manual. This manual will control and document changes made to the plant including documentation associated with:

- Physical modifications
- Revisions to Calculations and Specifications
- As-Built Configuration Changes to Plant Drawings
- Administrative and Editorial Changes to Plant Documents

These procedures will resolve the design control deficiencies and will be simplified and streamlined to facilitate the decommissioning process. Compatibility will be ensured with the:

- Design Input and Independent Design Verification Requirements
- Material and Equipment/Component Documentation Requirements
- Safety Evaluation Process
- UFSAR Maintenance/Update Process
- Decommissioning Work Control Process
- Document Control Process

This revised process will efficiently control design activities associated with the safe storage of spent fuel as well as decommissioning.

The updating of processes and procedures, with the exception of the Decommissioning Work Control Process, will be developed in the second quarter of 1997. All design control processes and procedures will be implemented by the end of the third quarter of 1997 following completion of training.

Management Standards and Expectations

The lack of management standards can generally be characterized as "living with" known problems. Weaknesses in specific standards and expectations were found in:

- Design input verification
- Rigor and precision in analyses
- Root cause analyses and engineering evaluations
- UFSAR maintenance
- Commitment follow through

As corrective actions, the following standards, policies, and expectations were established and are continuously being reinforced:

- Identifying and correcting problems in a timely manner
- The need for precision and rigor in engineering analyses
- The need for broad based root cause analyses
- The need for precision and rigor in verifying design inputs
- The need for prompt, thorough updating of ancillary documentation
- Tracking and monitoring workloads

Written standards have been implemented for engineering and will be benchmarked against industry leaders in the third quarter.

Resource Management

Two weaknesses were identified: (1) management failing to recognize the resources needed for emerging work and inconsistent prioritization of assignments, and (2) management failure to recognize the magnitude and scope of the resources needed for program implementation. Inconsistent resource management was found in the following areas:

- Outage planning and management
- Ownership of ancillary programs
- Setpoint uncertainty calculation program
- Adverse trend report process
- Action item tracking process

The following corrective actions are being implemented:

- Assess and provide sufficient resources for:
  - Spent fuel storage
  - Updating of ancillary tasks
  - Implementing new plant programs and processes

  - Adverse condition report processing
- Establish ownership for ancillary tasks
- Implement new programs with proper planning

Sufficient resources are available for defueled tasks. As old tasks are completed and new tasks are added, planning of resources is taken into account. Ownership for ancillary tasks is being established. Key performance indicators have been established to monitor and trend backlogs to assure they are not increased due to the lack of resources.

Training Programs

Inadequate training can be characterized as a lack of understanding of some day-to-day activities. Inadequate training was seen in connection with :

- Performing Safety Evaluation
- Updating the UFSAR
- Verifing design control inputs
- Updating Material Equipment and Parts List (MEPL)
- Understanding of the licensing/design basis
- Maintaining licensing commitments
- Performing root causes
- Achieving consistent corrective actions

As corrective action, the training program will be implemented and effectiveness improved by:

- Providing licensing and design basis training
- Providing safety evaluation training with emphasis on defueled accident analysis
- Providing UFSAR and licensing commitment training to assure understanding and availability of commitments
- Providing training for processing MEPL classifications
- Providing training for consistent and broad based corrective actions and root causes

The training for safety evaluations and UFSAR is complete. The root cause training is ongoing and we have committed to increase the number of trained personnel and departments with root cause trained personnel. Training in the remaining areas will be completed after the upgrades by the fourth quarter.

Independent Oversight

Enhancements and improvements are being made to build an internal oversight organization that provides intrusive, value added oversight, with special

emphasis in radiological controls. Nuclear Oversight's structure, processes, and staffing have been reviewed and revised to support decommissioning. Expertise has been added to the organization consistent with decommissioning.

Nuclear Oversight has instituted a comprehensive assessment and auditing program of the line organization. They have, and continue to, identify issues for improving plant performance. Oversight has performed in an intrusive, value added manner.

II. Violations Associated with Nitrogen Intrusion Event

The root cause assessment from the Independent Review Team (IRT) was presented in CYAPCO letter to USNRC dated October 23, 1996. This assessment identified four broad areas (material condition, procedural adequacy and usage, operator performance, and management performance) as needing improvement and described appropriate corrective actions.

Additionally, a common cause assessment was performed, utilizing the event timelines, in order to assess organizational, programmatic and management issues. In this process, all inappropriate actions by individuals were identified from the detailed timelines, sorted by the appropriate failure mode, and tallied to determine the most frequent occurring failure modes. Failure modes in the general categories of Human Error, Organizational and Programmatic Deficiencies, and Management Failure were considered.

Conclusions from the common cause analysis include:

- Lack of, or ineffective, information qualification, validation and verification, which resulted in inappropriate actions based on wrong or incomplete information.
- Incomplete or non-existent procedures for performing certain actions which caused workers to take action, relying on their own knowledge and skills, without the benefit of multi-discipline reviews.
- Lack of effective pre-job briefs.
- Lack of documented, communicated and reinforced management expectations, which resulted in a wide variation in work practices.

Three management issues were identified during the course of the investigation:

- The decision to stop the refueling activities for several days, with the reactor plant in a condition with limited vessel level and temperature indication and with the RCS loop stop valves closed, did not adequately consider the high risk.
- Management standards accepted procedures for Mode 5 operation that lacked sufficient detail.

- Management response to the initial nitrogen intrusion event on August 28, 1996 and the subsequent discovery of the continued leakage on September 1, 1996 was not sufficiently strong, given the safety significance.

The corrective actions taken in response to the nitrogen intrusion event are described in our October 23, 1996 letter and are summarized in Attachment 2. These actions were also presented at the enforcement conference on December 4, 1996. Those actions identified in our common cause study pertaining to organizational and programmatic changes were described previously in Section I of this attachment.

III. Violations Not Assessed a Civil Penalty Associated with Emergency Planning Deficiencies

A description of corrective measures taken or planned to be taken related to the conduct of the August 14, 1996 emergency preparedness exercise was provided in CYAPCO letter to USNRC dated November 25, 1996. This letter was in response to the weaknesses identified in NRC Inspection Report 50-213/96-07. A CYAPCO letter dated April 18, 1997 updated the status of corrective actions and commitments.

Since that time, all corrective actions have been implemented and table-top exercises were conducted in conjunction with an NRC inspection to demonstrate compliance at the Haddam Neck Plant. Attachment 2 contains the response to the specific violations.

IV. Other Violations Not Assessed a Civil Penalty

Attachment 2 contains the response to the specific violations.

Docket No. 50-213
CY-97-059

Attachment 2

Haddam Neck Plant
Reply to Notice of Violation (NOV)
Inspections 50-213/95-27, 96-06, 96-07, 96-08, 96-11, 96-80 & 96-201

June 1997

I. Violations Related to Inadequate Engineering

The violations in Sections I.A-I.C represent a Severity Level II problem (Supplement I). The violations in Section I.D represent a Severity Level II Problem (Supplement I).

Restatement of the Violation

A. Errors in Design Basis Documents or Errors Introduced by Design Changes

10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis for structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. These measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Design changes shall be subject to design control measures commensurate with those applied to the original design.

10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Contrary to the above, the licensee did not assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures and instructions, or did not assure that activities affecting quality were correctly prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances, as evidenced by the following examples, each of which constitutes an individual violation:

1. Calculation No. PA91-LOE-1171-GE, Revision 2, dated February 21, 1992, which determined the design duty cycle of the 125 Vdc station batteries, did not assume the design basis event specified in the Updated Final Safety Analysis Report (UFSAR) Section 8.3.2, namely, a simultaneous accident and loss of off-site power. Therefore, the design basis calculation did not account for all of the loads that would be powered during the event. Further, the calculation did not determine the battery voltage profile, and,

therefore, did not demonstrate that the battery voltage would remain above the minimum required voltage level. (01012)

2. Calculation No. 86-060-580GM, Revision O, dated September 2, 1986, which determined the minimum required net positive suction head for the high pressure safety injection pumps, used, as part of the calculation, an incorrect elevation for the top of the suction nozzle, and also did not account for the expected instrument uncertainty. As a result, Emergency Operating Procedure (EOP) ES-1.3, "Transfer to Sump Recirculation," Revision 13, dated March , 1995, which provides cautions to the operators to transfer high pressure safety injection (HPSI) pump suction to the containment sump before the refueling water storage tank (RWST) reaches a minimum level to prevent pump damage from cavitation, was inadequate in that the actual minimum level to preclude pump damage should have been 55,000 gallons instead of the 43,000 gallons as stated in the procedure. (01022)

3. Calculation IC-CY-1134GE, "Uncertainties for RWST Level Instrument Channels L-1806A and L-1806B," Revision 0, dated November 5, 1990, did not account for the errors associated with the change in hydrostatic head resulting from changes in tank water temperature and post-seismic effect, and resulted in inaccurate level indication. As a result, EOP ES-1.3, "Transfer to Sump Recirculation," Revision 13, dated March, 1995, was inadequate in that decision points for transfer to sump recirculation based on RWST indicated level did not account for these errors. (01032)

4. Proto-Power Calculation 95-MDE-01252MY, "CY SW System - Analysis of Design Basis Hydraulic Conditions," Revision 0, dated May 19, 1995, which was used to predict temperatures at the discharge of the CAR fan coils, did not accurately model system flow in that it did not model the transient two-phase flow conditions that were previously calculated to exist at this location, and which could adversely affect containment heat removal capability. (01042)

5. Calculations PA78-741-01-GE, "Diesel Generator Automatic Loading Analysis," Revision 3, dated January 21, 1991 (including CCNs 1-5) and PA90-LOE-1167-GE, "Diesel Generator Manual Loading Analysis," Revision O, dated February , 1991 (including CCNs 1-7) did not account for all of the applicable electrical loads. Specifically, the calculations did not incorporate cable power losses and power distribution transformer losses, and used a

non-conservative diversity factor. As a result, the margin of safety to emergency diesel generator (EDG) overload was less than identified in the licensee design analyses. (01052)

6. Calculation EQE-494-C-009, "Diesel Generator Starting Air," Revision 0, dated March 29, 1996, which established the seismic qualification for the diesel air start piping, incorrectly assumed that a qualified hose was installed. Although the calculation used a stiffness value of 10 pounds-per-inch to model the flexible hoses attached to the diesel, the installed flexible hoses had a less conservative stiffness value of 0.75 pounds-per-inch. A subsequent calculation determined that the air start piping displacement would exceed the vendor's acceptance limit during a design basis seismic event. (01062)

7. Prior to July 24, 1996, calculation C2-517-567-RE, "Uncontrolled Rod Withdrawal Transient Analysis" (which established the trip setpoints for the wide range nuclear instrumentation) did not account for channel uncertainties. As a result the channel uncertainties did not account for rack calibration accuracy, rack drift, rack temperature allowance and overall indicator accuracy. (01072)

8. Calculation CY-LPSI-89-RPS-700, "CY-LPSI Flow for Core Deluge Testing", Revision 0, dated August 29, 1989 determined the LPSI pump flow and system flow for one LPSI pump injecting into a vented reactor through the core deluge valves. This calculation concluded that the low pressure safety injection (LPSI) system would deliver 3810 gpm to the reactor based on a pump flow of 4000 gpm. The calculation did not adequately account for system flow resistances and the pump curve was not conservative. On December 15, 1995, the licensee documented its discovery (ACR 95-578) that LPSI injection flows could be as low as 3540 gpm, which was significantly less than the LPSI flow assumed in the safety analysis performed per 10 CFR Part 50, Appendix K to demonstrate satisfactory emergency core cooling system (ECCS) performance for Cycle 19 operations. Thus, the licensee failed to adequately translate the design basis into the testing program intended to demonstrate safety system performance in accordance with the accident analysis assumptions. (01082)

9. Calculation 95-EWA-01-01323-DY, Revision 0, dated February 2, 1996, which was used to support a technical specification (TS) amendment request, dated March 7, 1996, to the CAR fan surveillance testing acceptance criteria, did not appropriately

consider the fan performance capability. Specifically, the amendment requested a minimum acceptable flow of 40,000 cfm, which was not supported by the vendor's fan performance curve, did not provide sufficient margin for potential filter fouling, and could have resulted in flows less than designed. (01092)

10. Calculation 95-LKSL-1296-MY, Revision 0, dated May 19, 1995, which was used in the analysis of the impact of leak sealant injection into the four feedwater regulating valves, used a design pressure of 1000 psi in the calculation even though the actual design pressure of the system was 1210 psi. This calculation had been reviewed by a second engineer, the engineering supervisor, and the Plant On-site Review Committee (PORC). (01102)

11. Engineering evaluation CY-CD-1970, dated July 29, 1992, which was prepared to support the replacement of EDG annunciator panel relays, indicated that the safety classification of the relays was "non-QA" and concluded that the alarm and control circuits would not be adversely affected by the change. The evaluation was incorrect in that failure of the devices could have resulted in an inadvertent shutdown of the EDGs. (01112)

12. Calculation Change Notices 1 through 5 were initiated for Calculation PA78-741-01-GE, "Diesel Generator Automatic Loading Analysis," Revision 3, dated January 21, 1991 to revise the EDG loading tabulation in Attachment 4. However, the worst-case loading profile for EDG EG2B in Attachment 4 was not updated to be consistent with the revised loading tabulation; as a result, the margin of safety to EDG overload was less than identified in the licensee design analyses. (01122)

13. Calculation Change Notice 6, dated June 19, 1995, for Calculation PA90-LOE-1167-GE, "Diesel Generator Manual Loading Analysis," Revision O, dated February , 1991, changed the LPSI pump load from 874 kW to 945.85 kW when a design modification changed a piping orifice size; however, a similar Change Notice was not initiated for related Calculation PA78-741-01-GE, "Diesel Generator Automatic Loading Analysis," Revision 3, dated January 21, 1991; as a result, the margin of safety to EDG overload was less than identified in the licensee design analyses. (01132)

14. A licensee letter submitted to the NRC on July 7, 1993, stated that containment isolation valves were not needed for the service water return from containment for the CAR system because the flow indication provided early assurance to the operators of detection of

a service water line break. However, when the licensee issued TS Clarification C-TSC-059, dated January 1, 1996, to plant operators to address the impact of the loss of service water flow indication to the containment air recirculation coolers, the clarification did not address the containment isolation function and incorrectly concluded that the flow indication was not necessary for system operability. (01142)

15. NE-95-SAB-293, "CY Design Basis Containment Analysis For Large Break LOCA," dated July 21, 1995, calculated a new design basis maximum containment temperature which was not correctly translated into the design basis for all of the affected components inside containment. Although the shell side of the residual heat removal (RHR) heat exchangers had a design basis temperature of 200 degrees F, the revised calculation determined temperatures could reach 252 degrees F under certain conditions, and as of April 26, 1996, the effect of the containment temperature changes on the operability of the RHR heat exchangers had not been determined. (01152)

16. Calculation 93-00099-1064-DY, Revision 0, dated June 22, 1994, which was prepared to support the upgrading in safety classification of 120 VAC lighting panels, did not adequately address the seismic qualification of the lighting panels in that:

    i. the as-built anchor bolt configuration had not been evaluated;

    ii. anchor bolt capacities had been assumed with no documented bases; and

    iii. the analyses had disregarded the centers-of-gravity for the attached components. (01162)

17. Integrated Safety Evaluation CY-95-013, dated March 15, 1995 was performed to evaluate a change to EOP ES-1.3, "Transfer to Sump Recirculation" (which moved a step that directs the operators to stop the LPSI pumps earlier in the procedure); however, an appropriate calculation was not used to provide the basis for the change. The evaluation used Calculation NE-93-SAB-017, "CY Sump Recirculation Times - Justification for Turning LPSI pumps off at 10 minutes post-LOCA," which assumed the pumps would be turned off 10 minutes after the initiation of a large break loss of coolant accident (LOCA). However, the procedure modification resulted in stopping the LPSI pumps in 7.8 minutes. (01172)

18. Stone and Webster Calculations 15198-001 and 15198-002, dated March 21, 1986, and April 29, 1986, respectively, determined that the reactor coolant system head vents were susceptible to adverse hydrodynamic loading, and based the calculation on the assumption that the downstream vent valve was opened first during the venting sequence to prevent the impact of any fluid on a partially opened downstream valve. However, this design assumption was not maintained in plant procedures, in that a revision made in 1990 to the original 1986 procedure failed to maintain the required valve operating sequence. Further, this design was not included in EOP FR-I.3, "Response to Voids in Reactor Vessel," Revision 10, dated March , 1995 which was inadequate in that it directed the operators to open both reactor coolant system head vents in the same step. (01182)

19. On August 17, 1973, as part of plant modification Plant Design Change Request (PDCR) 156, Flooding Protection of Safeguards Equipment, carbon steel barriers were installed around floor openings on both levels of the primary auxiliary building to preclude internal flood water from reaching the RHR pumps. TS amendment 27 was issued based on a calculation that the assumed operator response time for the worst case internal flood in the primary auxiliary building was approximately 12 minutes. However, this calculation was inadequate in that on October 23, 1996, the licensee identified approximately 35 additional floor penetrations in the primary auxiliary building that were not modified with barriers and were not considered in the calculation of flooding time. As a result, flood water could render the RHR pumps inoperable in less than 12 minutes (approximately 8 minutes). (01192)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. These violations were caused by personnel errors and programmatic deficiencies. Additional contributing factors included (1) inattention to detail and lack of rigor in verifying design inputs, (2) lack of plant process for identifying and correcting original design errors, and (3) lack of independent review of design input.

Affected Programs Include:

Configuration Management
Licensing/Design Basis

Procedure Deficiencies (Programmatic)
Training Programs

Corrective Steps that will be taken

The systems associated with the violations noted, with the exceptions of those identified as 4, 6, 11 and 16 are not required in the defueled mode. While specific actions to correct these individual violations are no longer appropriate, actions to correct programmatic issues were identified in our common cause study and are currently being implemented. The details of, including schedule and implementation status for, the programmatic changes related to the Configuration Management Program, Procedure Deficiencies, and Training Program are presented in Attachment 1.

From Violation 4, the generic concerns associated with water hammer apply to the Service Water System (SWS). The SWS has been reviewed and modified for water hammer concerns in the defueled mode. Our assessment was described in CYAPCO letter to NRC dated March 21, 1997 and reported in LER 97-007-00, dated April 9, 1997. The NRC discussed resolution of this concern in NRC Inspection Report 97-01 dated May 8, 1997.

Violations 6, 11 and 16 are associated with the Emergency Diesel Generators. To address the question regarding flex hose stiffness (6), an operability evaluation was performed. The operability evaluation assessed the maximum resultant hose displacement and determined the allowable number of cycles to prevent fatigue failure. Under original design basis SSE loading conditions it was concluded that the hose would not experience a fatigue failure. To address the incorrect safety classification of the relays (11), a MEPL determination was completed and reclassified the relays as safety related, QA Category I. To address the inadequate seismic qualification of the lighting/power panel (16), the support configuration was upgraded. The as-installed configuration did not provide for full bearing against the anchor bolt nut surface. The anchor bolt configuration has since been modified to ensure full bearing against the nut.

Full compliance will be achieved once the revised programs are implemented.

Restatement of the Violation

B. Inadequate, or Lack of Safety Evaluations and Failures to Update the Final Safety Analysis Report

1. 10 CFR 50.59, "Changes, tests and experiments," permits the licensee, in part, to make changes to the facility as described in the safety analysis report without prior Commission approval provided the change does not involve an unreviewed safety question (USQ). The licensee shall maintain records of changes in the facility and these records must include a written safety evaluation which provides the bases for the determination that the change does not involve a USQ.

   Contrary to the above, the licensee made the following changes to the facility as described in the safety analysis report without performing a written safety evaluation for any of these changes to provide the basis for the determination that the changes did not involve a USQ, as evidenced by the following examples, each of which constitutes an individual violation:

   a. During the week of March 25, 1996, in order to facilitate maintenance activities, the facility was changed by removing an internal flood protection floor block from the Primary Auxiliary Building pipe trench tunnel, and installing a temporary cover which could not provide adequate flooding protection, even though the flood protection floor blocks are shown in UFSAR Figure 3.8-15. (01202)

   b. PDCR 1411, completed on August 13, 1994, removed breathing air stations inside containment and changed the air connections to support the use of portable air units, even though these stations are explicitly described in UFSAR Section 9.3.1.2, "Compressed Air System - System Description." (01212)

   c. PDCR 1479, completed on March 22, 1994, removed two service water elbow tap flow indicators and the associated tubing and instrument valves for the measurement of flow to the RHR heat exchangers, even though UFSAR Section 9.2.1.4, "Service Water System - Instrumentation Requirement," specifically identifies the elbow tap arrangement in describing the instrumentation used to measure service water flow to the RHR heat exchangers. (01222)

d. PDCR 1520, completed on September 6, 1995, changed system controls, dryer purge air supplies, and replaced the activated alumina desiccant with a molecular sieve desiccant for the control air system, even though UFSAR Section 9.3.1.2, "Compressed Air System - System Description," states that the control air system uses three air dryers that are of the activated alumina, automatically regenerated type. (01232)

e. PDCR 1448, completed on April 6, 1996, added a tank, pump, and tubing to utilize ethanolamine (ETA) as well as hydrazine for the control of secondary water chemistry, even though UFSAR Section 10.3.5, "Water Chemistry," states that secondary water chemistry is controlled using chemical feed equipment that adds hydrazine to the condensate system, and does not mention ETA. (01242)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. These violations were caused by procedure and programmatic deficiencies. Additional contributing factors included not having a procedure for screening 10CFR50.59 applicability and poor knowledge of the 50.59 process by both engineering and management personnel.

Affected Programs Include:

Configuration Management
Licensing/Design Basis
Design Control Process
Training Programs

Corrective Steps that will be taken

Of the five identified violations (a) and (e) are associated with systems not required in the defueled mode. While specific actions to correct these violations are no longer appropriate, actions to correct programmatic issues were identified in our common cause study and are currently being implemented. The details of, including schedule and implementation status for, the programmatic changes related to the Configuration Management Program, Licensing Basis/Design Basis Program, Training Program and Design Control Process are presented in Attachment 1.

The remaining violations were input into the Unresolved Item Report (UIR) process and safety evaluations prepared to address the changes. The UIR process originated from the configuration management program effort in response to the NRC Staff 50.54(f) letter. Two of the three safety evaluations [(b) and (c)] have been approved by PORC. The third safety evaluation [(d)] will be processed by the third quarter 1997 once the licensing basis/design basis review is complete to determine if the system is still required.

The plant procedure for conduct of 50.59 safety evaluations has been revised and issued. A new screening process was included in this revision. Training on this procedure has been conducted and is complete.

Full compliance will be achieved once the revised programs are implemented.

Restatement of the Violation

2. 10 CFR 50.59, "Changes, tests and experiments," permits the licensee, in part, to make changes to the facility as described in the safety analysis report without prior Commission approval provided the change does not involve a USQ. The licensee shall maintain records of changes in the facility and these records must include a written safety evaluation which provides the bases for the determination that the change does not involve a USQ.

   10 CFR 50.71(e) requires the licensee to update the FSAR originally submitted as part of the application for the operating license to assure that the information included in the FSAR contains the latest material developed. The updated FSAR shall be revised to include the effects of, in part, all safety evaluations performed by the licensee in support of conclusions that changes did not involve a USQ.

   10 CFR 50.9(a) requires, in part, that information provided to the NRC by a licensee or information required by regulation to be maintained by a licensee shall be complete and accurate in all material respects.

   Contrary to the above, the description of the facility in the FSAR was not accurate in all material respects in that the FSAR did not match the facility, required safety evaluations were not performed, and the FSAR was not properly updated, as evidenced by the following examples, each of which constitutes an individual violation:

   a. Prior to April 1994, FSAR Section 3.8.1.3, "Design Loading Criteria," inaccurately described maximum snow loadings for safety-related structures as 60 lb/ft^2 although the original design specifications and as-built construction were for 40 lb/ft^2. As the safety related structures were not designed and constructed for 60 lb/ft^2 maximum snow loads, the inaccuracy was material in that no evaluation existed to determine that the inaccuracy did not constitute a USQ nor was the FSAR updated to correct the inaccuracy. Following recognition of the discrepancy, FSAR Change Request (FSARCR) 94-CY-1 in April 1994 changed the description of the maximum snow loadings for safety-related structures in FSAR Section 3.8.1.3 from 60 lb/ft^2 to 40 lb/ft^2, to agree with the original design specifications, without a safety evaluation

to determine that the as-found condition did not constitute a USQ. (01252)

b. Prior to June 1994, FSAR Section 8.3.1.1.5, "Emergency AC Power System Description," inaccurately described the non-emergency trips of the EDGs as including a "Generator - Loss of Field" trip. As the facility EDGs did not have Generator - Loss of Field trips, the inaccuracy was material in that no evaluation existed to determine that the inaccuracy did not constitute a USQ nor was the FSAR updated to correct the inaccuracy. Following the discovery that the existing circuit did not provide this feature, FSARCR 94-CY-7 in June 1994 changed the description of non-emergency trips of the EDGs in Section 8.3.1.1.5 to delete the "Generator - Loss of Field" trip without a safety evaluation to determine that the as-found condition did not constitute a USQ. (01262)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. These violations were caused by programmatic and training deficiencies. A contributing factor was the lack of accountability for correctness and thoroughness of the required reviews.

Affected Programs Include:

Configuration Management
Licensing/Design Basis
Design Control Process
Training Programs

Corrective Steps that will be taken

Actions to correct programmatic issues were identified in our common cause study and are currently being implemented. The details of, including schedule and implementation status for, the programmatic changes related to the Configuration Management Program, Licensing Basis/Design Basis, Design Control Process and Training program are presented in Attachment 1.

Both violations have been input into the Unresolved Item Report (UIR) process for updating the UFSAR as needed for defueled conditions and will be processed by the third quarter 1997

Updating the UFSAR to reflect the defueled configuration is ongoing and the updated UFSAR will be submitted in the 4th quarter 1997.

Procedures to conduct 50.59 safety evaluations and control revisions to the UFSAR per 10CFR50.71(e) have been revised and issued. Training on these procedures has been conducted and is complete.

Full compliance will be achieved once the revised programs are implemented.

Restatement of the Violation

3. 10 CFR 50.59, "Changes, tests and experiments," permits the licensee, in part, to make changes to the facility as described in the safety analysis report without prior Commission approval provided the change does not involve a USQ. A proposed change, test, or experiment shall be deemed to involve a USQ if, in part, a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created.

   Contrary to the above, the safety evaluation associated with PDCR 1435, "Replacing Battery Charger BC-1-1A," Revision 0, dated February 28, 1994, did not adequately assess the possibility of a malfunction of a different type than previously evaluated when replacing the battery charger with a new battery charger of a substantially different design. Specifically, the safety evaluation did not consider the potential for degrading effects and failure modes on the 125 Vdc system due to the installation of the new battery charger. (01272)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violation. This violation was caused by programmatic deficiencies. A contributing factor was the lack of review or lack of understanding of the required depth of a 10CFR50.59 evaluation.

<u>Affected Programs Include:</u>

Configuration Management
Licensing/Design Basis
Design Control Process
Training Programs

Corrective Steps that will be taken

Actions to correct programmatic issues were identified in our common cause study and are currently being implemented. The details of, including schedule and implementation status for, the programmatic changes related to the Configuration Management Program, Design/Licensing Basis, Training Program, and Design Control Process are presented in Attachment 1.

This violation has been input into the UIR process for updating the UFSAR. The UFSAR will be updated to reflect the defueled configuration in the 4th quarter 1997.

Procedures to conduct 50.59 safety evaluations and control revisions to the UFSAR per 10CFR50.71(e) have been revised. Training has been conducted and is complete.

Full compliance will be achieved once the revised programs are implemented.

Restatement of the Violation

4. 10 CFR 50.59, "Changes, tests and experiments," permits the licensee, in part, to make changes to the facility as described in the safety analysis report without prior Commission approval provided the change does not involve a USQ. The licensee shall maintain records of changes in the facility and these records must include a written safety evaluation which provides the bases for the determination that the change does not involve a USQ.

   Contrary to the above, on June 11, 1996, the licensee discovered that no records existed to assure that the main feedwater regulating valves could close against the differential pressure anticipated during a main steam line break inside containment. Licensee engineering calculation (FCV-1301-1449-DY) on July 1, 1996, concluded that the feedwater regulating valve could not close under main steam line break conditions, even though UFSAR Section 15.2.9., states that the following functions provide the protection for a steam line rupture, "isolation of the main feedwater lines by two valves in series will occur on a safety injection actuation. This is done via closure of the feedwater isolation valve and the feedwater regulating valve after a short time delay." Thus, the licensee safety evaluations previously completed for modifications to the feedwater line isolation system were inadequate in that the existence of a USQ was not identified; namely, the safety evaluations for plant modifications per PDCR 423 in October 1981; the safety evaluations to support TS License Amendment No. 125 in 1991; and the safety evaluations for PDCR 1533 in February 1995. (01282)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violation. This violation was caused by a programmatic deficiency. A contributing factor was the lack of understanding of the required depth of a 50.59 evaluation.

<u>Affected Programs Include:</u>

Corrective Action
Licensing/Design Basis
Procedure Deficiencies (Programmatic)
Design Control Process

Corrective Steps that will be taken

The system equipment is not required in the defueled mode. While specific actions to correct this violation are no longer appropriate, actions to correct programmatic issues were identified in our common cause study and are currently being implemented. The details of, including schedule and implementation status for, the programmatic changes related to the Corrective Action Program, Licensing/Design Basis, Procedure Deficiency, and Design Control Process are presented in Attachment 1.

The UFSAR Chapter 15 accident analyses for the steam line rupture has been removed. The UFSAR will be submitted to reflect the defueled configuration in the 4th quarter 1997.

Procedures to conduct 50.59 safety evaluations and control revisions to the UFSAR per 10CFR50.71(e) have been revised. Training has been conducted and is complete.

Full compliance will be achieved once the revised programs are implemented

Restatement of the Violation

5. 10 CFR 50.71(e) requires the licensee to update the FSAR to assure that the information included in the UFSAR contains the latest material developed. Updates must be filed annually or 6 months after each refueling outage. The updates must reflect all changes up to a maximum of 6 months prior to the date of filing.

Contrary to the above, as of April 26, 1996, the licensee failed to update the UFSAR to reflect plant conditions, which existed more than 6 months prior to the previous UFSAR update, as evidenced by the following examples, each of which constitutes an individual violation:

a. UFSAR Table 9.2-1, "Service Water Major Component Interface," lists the "approximate required" service water flow rate for the individual components such as the diesel generator, residual heat removal heat exchanger, and spent fuel pool plate heat exchanger 2E Although these flow rates differ by up to 50 percent from the actual functional requirement flow rates, as listed in the Design Basis Document Package, dated September 1, 1995, and the "CY Service Water System GL 89-13 Item IV, Design Basis Summary Report," dated July 15, 1994, the UFSAR had not been updated as of April 26, 1996. (01292)

b. UFSAR Section 9.2.1, "Service Water System," states that service water system valves SW-MOV-1 and SW-MOV-2 close on a high containment pressure signal. However, the Design Basis Document Package, dated September 1, 1995, correctly identified that the valves closed on a Safety Injection Actuation Signal (SIAS). As such, the UFSAR was incomplete in that although high containment pressure is an input that initiates a SIAS signal, it is not the sole input, yet the UFSAR had not been updated as of April 26, 1996. (01302)

c. UFSAR Section 6.3.2.1, "ECCS - Schematic Piping and Instrumentation Diagrams," lists four interlocks associated with ECCS operation. Although the list does not include the interlock associated with SI-MOV-901 and SI-MOV-902, RHR to HPSI Crosstie Isolation Valves, which were installed in 1988 in accordance with PDCR 854, the UFSAR had not been updated as of April 26, 1996. (01312)

d. UFSAR Section 6.3.2.2, "ECCS System Design - Equipment and Component Description," provides a description of the ECCS

design and did not list valves SI-V-905, 906, 907, and 908 associated with the HPSI discharge lines. Although these valves were installed in the HPSI discharge lines in 1988 in accordance with PDCR 854, the UFSAR had not been updated as of April 26, 1996. (01322)

e. UFSAR Section 6.3.2.8, "ECCS System Design - Manual Action," states that the ECCS will be realigned for long-term sump recirculation after 8 hours following a loss of coolant accident. Although EOP E-1, "Loss of Reactor Coolant or Secondary Coolant," Revision 13, dated March , 1995, specifies that long-term sump recirculation is to be established at 1.5 hours, the UFSAR had not been updated as of April 26, 1996. (01332)

f. UFSAR Section 6.3, "Emergency Core Cooling System," describes system valve lineups for injection and sump recirculation. Although the short-term and long-term recirculation valve lineups differ from EOP E-1, "Loss of Reactor Coolant or Secondary Coolant," Revision 13, dated March , 1995, the UFSAR had not been updated as of April 26, 1996. (01342)

g. UFSAR Section 8.3.2.1.2, "Battery Chargers," states that during normal operation the 125 Vdc safety-related train A and train B battery chargers are operated in a float condition to maintain charger output voltage at 130 Vdc. Although the float voltage setting was revised from 131.8 Vdc to 132 Vdc following setpoint change request No. 94-17, dated May 5, 1994, the UFSAR had not been updated as of April 26, 1996. (01352)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. These violations were caused by procedure and programmatic deficiencies. Additional contributing causes included inadequate guidance to ensure complete and accurate UFSAR and/or Design Basis Document updating, and lack of high standards for timeliness, accuracy and detail for review and updating of the UFSAR.

Affected Programs Include:

Configuration Management
Licensing/Design Basis
Design Control Process
Training Programs

Corrective Steps that have been taken

Of the seven identified violations, (a) and (b) are applicable to systems required in the defueled mode. UFSAR changes have been processed and are complete.

Corrective Steps that will be taken

Deficiencies associated with (g) will be corrected with a August 1997 interim update of the UFSAR. The remaining violations are associated with systems not required in the defueled mode. The UFSAR will be submitted to reflect the defueled configuration in the 4th quarter 1997.

Actions to correct programmatic issues were identified in our common cause study and are currently being implemented. The details of, including schedule and implementation status for, the programmatic changes related to the Configuration Management Program, Corrective Action Program, Design Control Process and Training Program are presented in Attachment 1.

Procedures to conduct 50.59 safety evaluations and control revisions to the UFSAR per 10CFR50.71(e) have been revised. Training has been conducted and is complete.

Full compliance will be achieved once the revised programs are implemented.

Restatement of the Violation

C. Inadequate Corrective Actions

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

1. Contrary to the above, the licensee did not assure that conditions adverse to quality were promptly corrected, as evidenced by the following examples, each of which constitutes an individual violation:

   a. Adverse Condition Report (ACR) 95-467, dated November 13, 1995, identified conditions adverse to quality, namely that instrument loop errors may exist in the EDG kilowatt meters used for TS surveillance tests, and that similar instrument loop errors may exist for other instruments used to meet TS surveillance requirements. However, as of April 26, 1996, an uncertainty calculation had not been prepared, and an investigation into the generic implications had not been completed. (01362)

   b. The licensee identified in 1989 that instrument uncertainties had not been incorporated into the establishment of the EOP decision point for initiating containment spray, and reported the issue in Licensee Event Report (LER) 89-005, dated May 11, 1989. The licensee determined that instrument uncertainties and the effects of adverse environments should be reviewed for the EOP decision points, and incorporated where appropriate. The licensee's review was completed in March 1994. However, as of April 26, 1996, the licensee had not accounted for instrument uncertainty in the RWST level instrument decision points in the EOPs. (01372)

   c. UFSAR Section 7.5.2 states that RWST level instruments meet the criteria specified in Regulatory Guide 1.97. Northeast Utilities memorandum No. NE-90-SAB-230, dated September 18, 1990, identified that RWST level instruments were not correctly classified in UFSAR Section 7.5.2. In addition, Material Equipment and Parts List (MEPL)

Determination CY-CD-2130, dated May 5, 1994, identified that the RWST level instruments were not correctly classified in accordance with Regulatory Guide 1.97. However, prior to April 12, 1996, action had not been taken to correct the RWST classification discrepancies. (01382)

d. Quality and Assessment Services Audit Report No. A25098, dated November 30, 1994, identified that plant information reports (PIRs) had remained open well beyond the procedurally-specified time limit without receiving an extension approval. This weakness was a repeat occurrence of a previous audit deficiency. A later audit, No2E A601, dated July 3, 1995, concluded that the corrective actions to improve procedural compliance were not effective. However, as of April 26, 1996, adequate actions had not been taken to correct this recurring program deficiency in that numerous ACRs (to which PIRs had been converted in 1995) remained open beyond specified time limits. (01392)

e. A third-party audit entitled, "Station Blackout Assessment," Report No. 24-00116, dated October 1994, was performed to review the licensee's implementation of actions taken in response to the station blackout requirements in 10 CFR 50.63. The third party audit identified deficiencies in the licensee's implementation of the station blackout rule, including inadequate calculations for alternate AC loading, voltage drop, and battery sizing. Other deficiencies involved the adequacy of EDG reliability programs and discrepancies in the analysis of safe shutdown scenarios. However, as of April 26, 1996, the licensee had not taken corrective actions to address all of the deficiencies identified in the audit. (01402)

f. The root cause investigation report for ACR 95-577, dated January 15, 1996, recommended 19 corrective action items to address the condition regarding actual LPSI flow rates following a LOCA being less than that assumed in the safety analysis. The event was the subject of an NRC enforcement conference on February 12, 1996. The recommended corrective actions included safety system impact evaluations, design document revisions, a self-assessment initiative, and numerous procedure reviews and revisions. However, as of April 26, 1996, the corrective actions had not been initiated. (01412)

g. Calculation Change Notice 2, dated November 23, 1992 for Calculation No. PA-76-633-0040-GE, Revision 5, revised the degraded voltage protection system calculations, in response to an NOV from NRC Electrical Distribution System Functional Inspection, Inspection Report 50-213/91-80. In addition, the licensee committed to revise the associated surveillance procedure and TS. However, as of April 26, 1996, appropriate corrective action had not been taken in that the licensee had not revised the surveillance procedure or TS, resulting in the design basis calculation inconsistent with the TS and surveillance procedure. (01422)

h. While performing an operability evaluation for the containment sump screen mesh size on February 26, 1996, as part of ACR 96-1, the licensee did not identify and correct a condition adverse to quality. Specifically, ACR 96-1 documented the potential for screen mesh size to be different than designed based on such a finding at Millstone 2; however, the licensee did not identify that the containment sump screen mesh holes at Haddam Neck were 0.5 inches rather than 0.375 inches assumed in licensee analyses, thereby potentially rendering downstream ECCS components inoperable during an accident. (01432)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. These violations were caused by organizational and programmatic deficiencies. Additional contributing factors included lack of engineering resources, lack of formal process for ensuring instrument uncertainty calculations are performed, and lack of self-monitoring capability in tracking and prioritizing issues.

Affected Programs Include:

Corrective Action
Licensing/Design Basis
Procedure Deficiencies
Design Control Process
Configuration Management
Training Programs

Corrective Steps that will be taken

Of the eight identified violations, (b), (c), (f), (g) and (h) applied to systems not required in the defueled mode. While specific actions to correct these violations are no longer appropriate, actions to correct programmatic issues were identified in our common cause study and are currently being implemented. A major initiative to revise the corrective action program is nearing completion. The details of the programmatic changes related to the Corrective Action Process including schedule and implementation status are presented in attachment 1.

The remaining items [except (e)] have been entered into the UIR process for tracking corrective actions and will be processed by the third quarter 1997. For violation (e), the issue of station blackout assessment will be addressed in the Licensing/Design Basis effort described in Attachment 1.

Full compliance will be achieved once the revised Corrective Action Program is implemented.

Restatement of the Violation

2. Contrary to the above, the licensee did not assure that the causes of conditions adverse to quality were determined, and corrective action taken to preclude repetition, as evidenced by the following examples, each of which constitutes an individual violation:

    a. Licensee investigations into CAR fan surveillance failures reported in PIR 95-042, dated February 1, 1995, and LER 95-04, Revision 1, dated July 31, 1995, did not assure the causes of the failures were determined. Specifically, test results were not consistent with the root causes stated in the LER, and additional deficiencies that could have contributed to the surveillance failures were identified by the NRC during the week of April 15, 1996. (01442)

    b. In March 1995, the licensee replaced battery charger BC-1-1A with a new solid state design. During testing and subsequent operation, the battery exhibited ammeter fluctuations as documented in trouble report 15-CY-14018 BC 1-1A, dated March 1, 1995. On November 2, 1995, ACR 95-433 was issued to enter the condition into the corrective action program. However, as of April 25, 1996, effective corrective action had not been taken to determine the cause of the fluctuations and the condition adverse to quality remained uncorrected. (01452)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. These violations were caused by organizational and programmatic deficiencies. Additional contributing factors included inadequate or inaccurate design documentation, and non-conservative management decision.

<u>Affected Programs Include:</u>

Corrective Action
Licensing/Design Basis
Training Programs

Corrective Steps that have been taken

A "battery service test" surveillance was performed on the battery in July 1996 and additional troubleshooting on the battery charger performed in August 1996. The successful completion of this surveillance, in conjunction with continued periodic inspections of the battery, demonstrates that there has not been any degradation to the battery caused by the current fluctuations. The results of the additional troubleshooting were still indeterminant on the cause of the current fluctuations, but it is believed to be a system harmonics/stability problem. A safety evaluation was prepared in December 1996 to evaluate the effects the replacement battery charger, including current fluctuations, has had on the DC system. This evaluation concluded that the replacement of the charger was not an unreviewed safety question and it was safe.

Further troubleshooting will not be performed due to the defueled configuration of the plant and reduced loads on the DC system and the inverters. ACR 95-433 was closed on 1/4/97. Compliance has been achieved.

Corrective Steps that will be taken

Of the two identified violations, (a) applies to a system not required in the defueled mode. While specific actions to correct this violation are no longer appropriate, actions to correct programmatic issues were identified in our common cause study and are currently being implemented. A major initiative to revise the corrective action program is nearing completion. The details of the programmatic changes related to the Corrective Action Process including schedule and implementation status are presented in Attachment 1.

Full compliance will be achieved once the revised program is implemented.

Restatement of the Violation

D. Violations of Technical Specifications Caused by Inadequate Engineering

1. TS 3.5.1.a.6 requires two ECCS subsystems to be operable during Modes 1, 2, and 3 with an operable flow path capable of taking suction from the containment sump during the recirculation phase of operation.

   Contrary to the above, prior to July 22, 1996 during operation in Modes 1, 2, and 3, under certain conditions, the long-term recirculation phase flowpath for ECCS systems needed to mitigate postulated loss of coolant accidents was inoperable. On August 1, 1996, the licensee determined that the original 10 inch and 8 inch diameter piping from the containment sump to suction for the RHR pump resulted in insufficient NPSH to support RHR pump operation without relying (inappropriately) on containment backpressure. Specifically, EOP ES-1.3 provided instructions to the operator to operate an RHR pump through a single sump suction valve (RHR-MOV-22 or RHR-V808A) when transferring the ECCS to sump recirculation mode of operation following a postulated accident. In the configurations allowed by ES-1.3, adequate NPSH could not be assured for the range of possible conditions as the containment cooled and depressurized following an accident. Inadequate NPSH would result in RHR pump cavitation, vapor binding and eventual pump failure. (02012)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by programmatic deficiencies. Additional contributing factors included failure to fully analyze the applicable flowpath conditions, and licensing design basis deficiency.

Affected Programs Include:

Configuration Management
Licensing/Design Basis
Design Control Process

Corrective Steps that will be taken

This Technical Specification is not applicable in the defueled mode. While specific actions to correct this violation are no longer appropriate, actions to correct programmatic issues were identified in our common cause study and are

currently being implemented. The details of the programmatic changes related to the Configuration Management Program, Corrective Action Program and Design Control Process including schedule and implementation status are presented in Attachment 1.

Full compliance will be achieved once the revised programs are implemented.

Restatement of the Violation

2. TS 3.5.1.a.6 requires two ECCS subsystems to be operable during Modes 1, 2, and 3 with an operable flow path capable of taking suction from the refueling water storage tank and manually transferring suction to the containment sump during the recirculation phase of operation.

   Contrary to the above, prior to July 22, 1996 during operation in Modes 1, 2, and 3, the recirculation phase flowpath required to mitigate postulated loss of coolant accidents was inoperable. Specifically, after the reactor was shutdown on July 22, 1996, the licensee identified on July 26, 1996, that the (1) the containment sump screen mesh holes were larger than the .375 inch value assumed in licensee analyses, and (2) there was a 3 inch by 2 foot hole in the screen, thereby rendering the downstream ECCS components potentially inoperable during an accident. (02022)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by programmatic deficiencies. Additional contributing factors include inadequate or inaccurate design documentation, and failure to verify/validate information.

<u>Affected Programs Include:</u>

Configuration Management
Licensing/Design Basis

Corrective Steps that will be taken

This Technical Specification is not applicable in the defueled mode. While specific actions to correct this violation are no longer appropriate, actions to correct programmatic issues were identified in our common cause study and are currently being implemented. The details of the programmatic changes related to the Configuration Management Program, and Design/Licensing Basis including schedule and implementation status are presented in Attachment 1.

Full compliance will be achieved once the revised programs are implemented.

Restatement of the Violation

3. TS 3.6.2 requires at least four CAR units to be operable in Modes 1, 2, 3, and 4. The TS action statement requires that with only three CAR units operable, the inoperable CAR unit be restored to an operable status within 72 hours or be in at least hot standby within the next six hours and in cold shutdown within the following 30 hours.

   Contrary to the above, during all times when the reactor was in Modes 1, 2, 3, and 4 prior to July 22, 1996, all four CAR units were inoperable in that they could not have performed their intended function during LOCAs. Specifically, engineering analysis concluded that the service water piping structural limits would be exceeded due to waterhammer loads. The service water system is a support system for the CAR units and a part of the primary containment boundary. (02032)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by programmatic deficiencies. A contributing factor was the lack of understanding of design basis information.

<u>Affected Programs Include:</u>

Configuration Management
Licensing/Design Basis

Corrective Steps that will be taken

This Technical Specification is not applicable in the defueled mode. While specific actions to correct this violation are no longer appropriate, actions to correct programmatic issues were identified in our common cause study and are currently being implemented. The details of the programmatic changes related to the Configuration Management Program, and Design/Licensing Basis including schedule and implementation status are presented in the previous section.

Full compliance will be achieved once the revised programs are implemented.

II. Violations Associated with the Nitrogen Intrusion Event

The violations in Section II represent a Severity Level II problem (Supplement I).

Restatement of the Violation

A. TS 6.8.1 requires, in part, that written procedures and/or administrative policies shall be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Appendix A requires procedures for filling and venting the reactor coolant system; and for the startup, operation, and shutdown of the shutdown cooling system and the chemical and volume control system (including the Letdown/Purification System). The instructions in these procedures should include changing the mode of operation in the reactor coolant system.

Contrary to the above, written procedures and/or administrative policies were either not established or not implemented, as evidenced by the following examples, each of which constitutes an individual violation:

1. On August 22, 1996, Normal Operating Procedure (NOP) 2.7-4, "RHR Purification System Operation," Attachment 4, Step 1.4, which requires that the RHR purification pump suction valve from the RWST, PU-V-261A, be closed, was not properly implemented in that the valve was not closed. As a result, there was a diversion of approximately 500 gallons of reactor coolant to the RWST. (03012)

2. On August 28, 1996, Surveillance procedure (SUR) 5.1-159B, "Boron Injection Flow Path Verification and Metering Pump Test," step 6.1.2, which requires the operator to verify each component in the flowpath is in its specified position on the valve lineup checklist, and step 5.1.1, which requires the operator to immediately notify the shift supervisor and not proceed if a component is not found in its specified position, were not properly implemented in that an operator repositioned valve BA-V-355 from closed to open upon determining that the valve was not in its specified position without notifying the shift supervisor. This action resulted in water and nitrogen addition into the reactor coolant system. (03022)

3. On August 29, 1996, NOP 2.4-3, "Shutdown of an Individual Loop," was inadequate in that no instructions existed to preserve overpressure protection of an isolated reactor coolant system loop

so as to preclude exceeding design stress levels in an isolated loop. (03032)

4. On August 31, 1996, NOP 2.9-1, "Placing the Residual Heat Removal System In Service," was inadequate in that it did not have instructions to shift RHR pumps, vent the RHR pumps, isolate RHR heat exchangers, and place limitations on maximum RHR flow through the heat exchangers. (03042)

5. On August 31, 1996, NOP 2.4-7, "Return of a Loop to Service with the Plant Shutdown," was not properly implemented in that isolated loop boron concentrations and loop temperatures were not determined as required prior to opening the loop stop isolation valves. (03052)

6. On September 3, 1996, NOP 2.9-6, "Primary Vent Header Operation," required the installation of a vacuum pump to vent the reactor coolant loops. This procedure was not properly implemented in that during the NRC walkdown of the system, no vacuum pump installation connections were available to support venting the reactor coolant loops, and the system configuration as depicted in NOP 2.9-6 did not match the field installation. In addition, this procedure was not adequately established in that no procedural controls existed to periodically verify the operation of the vent system. The procedural deficiencies contributed to ineffective venting of non-condensable gases within the reactor coolant system and the reactor vessel. (03062)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations.

The cause of these violations were described in the CYAPCO Letter to the USNRC dated October 23, 1996, and are summarized below:

1. Inadequate procedures.

2. Improper procedure implementation.

3. Lack of an effective pre-job briefing.

4. Failure to solicit assistance to resolve discrepancy.

5. Lack of a questioning attitude.

6. Inappropriate decision making.

7. Lack of Instrumentation.

8. Inadequate training.

9. Poor equipment condition.

10. Poor implementation of generic information.

11. Inappropriate planning and scheduling.

Corrective Steps that have been taken and the results achieved

Corrective actions that were taken to address the specific violations are described in the CYAPCO Letter to the USNRC dated October 23, 1996.

In mid-September 1996, the Operations Department formed a dedicated procedure group to upgrade operations procedures and ensure the level of detail was improved (e.g., proper wording, proper sequencing, adequate detailed guidance). In addition to full time CYAPCO employees, the group includes a formally licensed contracted Senior Reactor Operator (SRO) to ensure that the department obtained an outside perspective of the operating procedures. The upgrading of procedures focuses on procedures associated with (1) near term evolutions, (2) Temporary Procedure Changes, and (3) spent fuel building and associated systems procedures. The upgrading of the remaining procedures has been incorporated into the biennial review process.

Of the six procedures identified, one that was designated as inadequate, NOP 2.9-6, "Primary Vent Header Operation" has been revised to correct the deficiencies. Four have been or are scheduled to be designated "DO NOT USE" due to the impending decommissioning. The remaining procedure, NOP 2.7-4, "RHR Purification System Operation" did not require revision since implementation (failure to follow procedure) was the cited problem.

Corrective Steps that will be taken

The management team has initiated a concerted effort to raise the performance standards for all personnel on site and establish high standards for the decommissioning process. Included in this effort is reinforcing high standards for the core values of honesty, integrity and commitment to the job, including program and procedure ownership and adherence, safe working practices and a questioning attitude. These core values have been established in the management and supervisory team through group meetings and discussions.

Group meetings to discuss these core values are also being conducted between senior site management and non-management/supervisory personnel.

The management team has worked with their individual department staff to establish department level standards that reflect these higher core values. All departments have completed draft standards and expectation documents that have been reviewed by senior site management to assure consistency with the core values. Each of the department standards and expectations addresses the expectation of adherence to site procedures. Site-wide and department level standards and expectations will be published by June 30, 1997. This process of ensuring consideration of the input from all site staff will help to ensure site-wide ownership of these core values.

Full compliance will be achieved once the corrective actions have been completed.

Restatement of the Violation

B. TS 6.8.2 requires that each procedure required of TS 6.8.1 shall be reviewed by PORC and approved by the Vice President - Haddam Neck prior to implementation.

Contrary to the above:

1. On August 28, 1996, and September 1, 1996, the licensee vented and refilled portions of the charging system with instructions that were not reviewed by PORC and approved by the Vice President - Haddam Neck prior to implementation.

2. On August 29, 1996, the licensee drained the reactor coolant system with instructions that were not reviewed by PORC and approved by the Vice-President - Haddam Neck prior to implementation. (03072)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. These violations were caused by a programmatic deficiency which allowed procedures to be performed without proper review and approval.

Corrective Steps that have been taken and the results achieved

The two identified evolutions were conducted using ACP 1.2-5.3, "Evaluation of activities/Evolutions Not Controlled by Procedure" guidance that allowed operators to develop written guidelines and conduct operations that were not spelled out in a procedure or reviewed by PORC prior to implementation. The ACP was canceled by PORC on October 23, 1996 and detailed written procedures were developed and approved to cover the listed evolutions. As required, all procedures are prepared and approved in accordance with ACP 1.2-6.5A, B, and C, "Station Procedures."

NOP 2.6-2, "Chemical and Volume Control System Operation" Attachment 1 provides instructions for filling and venting the Charging System. NOP 2.6-12, "Draining the RCS in Mode 5 and 6" was developed to provide detailed instructions to draindown the RCS during Mode 5 and 6.

Compliance has been achieved.

Restatement of the Violation

C. 10 CFR Part 50, Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed by procedures appropriate to the circumstances, and shall be accomplished in accordance with the procedures. Procedures shall include appropriate qualitative acceptance criteria for determining that activities affecting quality have been satisfactorily accomplished.

Contrary to the above, normal operating procedure (NOP) 2.3-4, Shutdown from Hot Standby to Cold Shutdown and procedure steps developed under ACP 1.2-5.3, Evaluation of Activities/Evolutions Not Controlled by Procedures, used to drain the reactor on August 29, 1996 and to operate the reactor in a partially filled and vented condition from August 29 - September 3, 1996 was inadequate in that operators lacked the reactor vessel level and core exit thermocouple instrumentation used to verify that level was acceptable and that draining and fill evolutions are satisfactorily accomplished. The instrumentation had been disconnected during a period of extended operation in a partially filled and vented condition due to a change in the refueling plan and schedule that had not been thoroughly reviewed for impact on shutdown risk. This lack of review and planning resulted in the plant being placed in a vulnerable configuration, with only limited instrumentation and indications available to the operators. (03082)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by inadequate procedures and insufficient planning.

Corrective Steps that have been taken and the results achieved

Corrective actions that were taken are described in the CYAPCO letter to the USNRC dated October 23, 1996.

Work Control Manual procedure, WCM 1.2-9, "Outage Planning, Scheduling and Implementation," was revised to require significant delays and work stoppages to be processed as an outage schedule change which includes an assessment as to whether the change is significant to shutdown risk.

Plant operators were sensitized to the need to protect the key safety functions (decay heat removal, RCS inventory, electrical power, reactivity control, and containment) via the new Operations Department Instruction, ODI 191 "Shutdown Risk Assessment."

NOP 2.6-12, "Draining the RCS in Mode 5 and 6" was developed to provide detailed instructions to draindown the RCS during Mode 5 and 6. Jumpers were installed to provide one train of Reactor Vessel Level Indicating System (RVLIS) indications and to provide two core exit thermocouple (CET) readings to Control Room operators.

Corrective Steps that will be taken

The management team has initiated a concerted effort to raise the performance standards for all personnel on site and establish high standards for the decommissioning process. Included in this effort is reinforcing high standards for the core values of honesty, integrity and commitment to the job, including program and procedure ownership and adherence, safe working practices and a questioning attitude. These core values have been established in the management and supervisory team through group meetings and discussions. Group meetings to discuss these core values are also being conducted between senior site management and non-management/supervisory personnel.

The management team has worked with their individual department staff to establish department level standards that reflect these higher core values. All departments have completed draft standards and expectation documents that have been reviewed by senior site management to assure consistency with the core values. Each of the department standards and expectations addresses the expectation of adherence to site procedures. Site-wide and department level standards and expectations will be published by June 30, 1997. This process of ensuring consideration of the input from all site staff will help to ensure site-wide ownership of these core values.

Full compliance will be achieved once the corrective actions have been completed.

Restatement of the Violation

D. 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Action), requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, deficiencies, and deviations, are promptly identified and corrected.

Contrary to the above, measures had not been established to assure that conditions adverse to quality, such as failures, deficiencies, and deviations, are promptly identified and corrected, as evidenced by the following examples, each of which constitutes an individual violation.

1. Between August 28, 1996, and September 5, 1996, a condition adverse to quality existed, namely nitrogen gas from the volume control tank entering the reactor vessel as a result of a failure to adhere to a procedure. The gas leakage continued even after the licensee believed that the leak was isolated, resulting in a displacement of reactor coolant to a level approximately three feet below the vessel flange. During this time period, various unexplained indications existed, such as reactor coolant system level anomalies and unexplained increase in nitrogen use, that could have alerted the operators to this condition. However, licensee personnel were ineffective in identifying and correcting the full extent of the gas intrusion into the reactor coolant system, a condition adverse to quality, until September 5, 1996, when, the licensee isolated the nitrogen gas leak and restored reactor vessel level. (03092)

2. Between August 31 and September 6, 1996, the licensee management and technical support responses to the nitrogen bubble and degraded RHR subsystem events were fragmented and protracted, resulting in untimely corrective actions for significant conditions adverse to quality. The untimely responses were reflected in the failure to fully appreciate the significance of the event, resulting in delays in initiating an integrated event response; establishing actual reactor vessel level; reestablishing control room indications for reactor vessel level and temperature; aligning a reactor coolant pump for service; and establishing and implementing an independent review team. Also, the actions to monitor the operating A RHR pump, following the B RHR pump failure, were not comprehensive or timely. (03102)

3. Between September 1 and September 26, 1996, several avoidable delays were encountered in the licensee's corrective maintenance on the B RHR pump. These delays included a lack of quality

replacement parts, inadequate vendor supplied information, lack of technical evaluations for floor block removal, and absence of appropriate vent locations. However, this condition adverse to quality was not promptly corrected during this time. (03112)

4. ACR 96-1106, dated September 26, 1996, identified that approximately 600 gallons of unborated water was diverted to the reactor coolant system during a makeup to the refueling water storage tank. This event was caused by leak-through of a two inch chemical and volume control system manual globe valve (BA-V-367). Prior to this event, five additional ACRs were prepared in September 1996, identifying various chemical and volume control system valves with leak-through, however, the diversion of the unborated water was not identified. (03122)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. These violations were caused by programmatic deficiencies (i.e., adequacy and timelines of corrective action).

Corrective Steps taken and the results achieved

The Operations Department Instruction (ODI) for "Conduct of Operations" was revised to place an emphasis on having a questioning attitude. New expectations for off-normal conditions were outlined to improve management performance and lower the threshold for operating crews to seek assistance. Guidance was provided on when to call in assistance to ensure a multi-discipline approach for off-normal events. Roles and responsibilities between line and oversight organizations were clarified. Training was provided on applicable procedures and department instructions.

Compliance has been achieved.

1. Clarifications to the Emergency Action Level (EAL) documents were issued.

2. Training on the EAL clarifications was conducted.

3. Three table-top drills were conducted and observed by NRC.

Compliance was achieved based on the successful completion of the table-top drills. An inspection was conducted by the NRC to verify the ability of emergency response personnel to properly classify emergency conditions and should result in closure of this violation.

Restatement of the Violation

B. 10 CFR 50.54(q) states, in part, "A licensee authorized to possess and operate a nuclear power reactor shall follow and maintain in effect emergency plans which meet the standards in 10 CFR 50.47(b) and the requirements in Appendix E of this part."

EPIP 1.5-43, Personnel Radiation Control and Dosimetry Issue During Nuclear Emergencies, requires consistent with 10CFR 50.54(q), in part, that guidance is provided for on site radiation exposure control for nuclear incident/accident emergencies; and Emergency Preparedness Operating Procedure (EPOP) 4428G, Revision 0, Protective Action Recommendations (PARs), requires, in part, that areas beyond the 10 mile emergency planning zone can be addressed on an ad hoc basis if the area is threatened by the plume.

Contrary to the above, during the emergency exercise on August 14, 1996, the licensee, in responding to the exercise scenario, failed to implement protective actions based upon dose projections for the site emergency response organization at the emergency operations facility (EOF) and for personnel on site, and also failed to consider PARs beyond the 10 mile emergency planning zone which was threatened by the plume. Specifically, the licensee, in the exercise, did not make provisions for evacuating emergency operating facilities and site personnel due to potentially high projected dose rates. (Because dose projections based on the scenario exceeded the protective action guideline of 10 rem for residents beyond the 10 mile radius, the PAR should have been extended to include those residents projected to receive greater than 1.0 rem.) (04023)

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violation. This violation was caused by programmatic deficiencies; specifically a breakdown of the program intended to define roles and responsibilities, establish organizational structure, establish organizational interfaces, and provide training to maintain the knowledge and skills of personnel designated emergency response personnel.

Corrective Steps that have been taken and the results achieved

The following corrective action steps were taken:

1. The process for making protective action recommendations was changed and the procedure revised.

2. Training on the revised procedure was conducted.

3. Three table-top drills were conducted and observed by NRC.

The NRC conducted an inspection during the table-top drills and concluded that the new PAR process and procedure were much improved, effective and timely. Compliance has been achieved at the Haddam Neck Plant.

IV. Other Violations Not Assessed a Civil Penalty

Restatement of the Violation

A. TS 6.8.1 requires, in part, that written procedures be established, implemented and maintained in accordance with the provisions contained in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

Procurement Engineering Group (PEG) Departmental Instruction PEG 6.05, "Vendor Interface for Key Safety Related Components," provides the instruction to implement the Northeast Utilities commitment for vendor interface for key safety related components, as described in the licensee's response, dated April 19, 1991, to NRC Generic Letter (GL) 90-03, "Relaxation of Staff Position in Generic Letter 83-28, Item 2.2 Part 2, 'Vendor Interface for Safety-Related Components.'"

Section 5 to PEG 6.05 requires that once per calendar year, the individual assigned responsibility will develop a list of key safety related components for the Connecticut Yankee and Millstone Units 1, 2, and 3 (sub-section 5.1); identify the vendors for the key equipment (sub-section 5.1); review the file for each vendor, noting any audit findings, new design announcements, or related information (sub-section 5.4); develop a list of generic questions for each vendor (sub-section 5.5); and call each vendor and try to get the questions answered (sub-section 5.6).

Contrary to the above, the licensee did not execute PEG 6.05 during the calendar years 1994 and 1995 and therefore did not initiate vendor contacts during these years, consistent with the licensee response to GL 90-03 for key safety-related components. (05014)

This is a Severity Level IV violation (Supplement I).

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by personnel error. Contributing factors include lack of communication, inadequate commitment tracking, and low standards/management expectations.

Corrective Steps that have been taken and results achieved

Disciplinary action was taken. The commitment has been reinstituted.

Corrective Steps that will be taken

The management team has initiated a concerted effort to raise the performance standards for all personnel on site and establish high standards for the decommissioning process. Included in this effort is reinforcing high standards for the core values of honesty, integrity and commitment to the job, including program and procedure ownership and adherence, safe working practices and a questioning attitude. These core values have been established in the management and supervisory team through group meetings and discussions. Group meetings to discuss these core values are also being conducted between senior site management and non-management/supervisory personnel.

The management team has worked with their individual department staff to establish department level standards that reflect these higher core values. All departments have completed draft standards and expectation documents that have been reviewed by senior site management to assure consistency with the core values. Each of the department standards and expectations addresses the expectation of adherence to site procedures. Site-wide and department level standards and expectations will be published by June 30, 1997. This process of ensuring consideration of the input from all site staff will help to ensure site-wide ownership of these core values.

Full compliance will be achieved once the corrective actions have been completed.

Restatement of the Violation

B. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed by documented procedures.

10 CFR Part 50, Appendix B, Criterion XV, "Nonconforming Materials, Parts or Components," requires, in part, that measures be established for nonconforming materials, parts and components, which include procedures for disposition.

Contrary to the above, as of April 26, 1996, the licensee did not provide procedural guidance for dispositioning non-conformance reports pertaining to non-QA materials that had been installed in safety-related applications. (06014)

This is a Severity Level IV violation (Supplement I).

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by training and programmatic deficiencies.

1. No guidance provided in any plant procedure regarding equipment upgrade.

2. Lack of training for system engineers responsible for the equipment upgrade.

Corrective Steps that will be taken

In conjunction with the Configuration Management Program initiative described in Attachment 1, the NCR procedure will be reviewed in the defueled condition.

Compliance will be achieved upon completion of the Configuration Management Program initiative scheduled for implementation in the third quarter 1997.

Restatement of the Violation

C. 10 CFR Part 50, Appendix J, Section II.G, defines Type B tests, in part, as tests intended to measure leakage across leakage limiting boundary for primary reactor containment penetrations, including piping penetrations fitted with expansion bellows. TS 4.6.1.2.d states that containment leakage rates shall be demonstrated in conformance with the criteria in Appendix J of 10 CFR Part 50, and that Type B tests shall be conducted at intervals not greater than 24 months and at a pressure not less than Pa, 39.6 psig.

Contrary to the above:

1. Penetration P-50, fuel transfer tube bellows assembly, has never been tested in accordance with the requirements of 10 CFR Part 50, Appendix J.

2. Containment penetration CN-2, hydraulic tubing that is part of the air lock door operating mechanism, has never been tested in accordance with Appendix J. (07014)

This is a Severity Level IV violation (Supplement I).

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by programmatic deficiencies. Additional contributing factors were inattention to detail, and improper interpretation of NRC regulations.

Affected Programs Include:

Configuration Management
Licensing/Design Basis

Corrective Steps that will be taken

This violation applies to a system no longer required in the defueled mode. While specific actions to correct the individual violations are no longer appropriate, actions to correct issues were identified in our common cause study and are currently being implemented. The details of the programmatic changes related to the Configuration Management Program and Design/Licensing Basis including schedule and implementation status are presented in Attachment 1.

Full compliance will be achieved once the revised programs are implemented.

Restatement of the Violation

D. 10 CFR 50.72(b)(2)(iii)(B) requires the licensee to report within four hours of its occurrence an event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to remove residual heat.

Contrary to the above, the accumulation of nitrogen in the reactor vessel, which could have prevented the removal of residual heat, was discovered at 9:00 a.m. on September 1, 1996 but was not reported until September 11, 1996. (08014)

This is a Severity Level IV violation (Supplement I).

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by an improper reportability determination. A contributing factor was the lack of conservative reporting philosophy.

Corrective Steps that have been taken and the results achieved

On September 11, 1996 an Adverse Condition Report was initiated to document a recommendation to promptly report the accumulation of nitrogen in the reactor vessel. ACR 96-1016 was discussed with the Management Review Team on September 12, 1996 and a conservative reporting philosophy was emphasized.

On September 24, 1996 the Adverse Condition Reports for the Nitrogen Intrusion Event were reviewed again for reportability. It was determined that the loss of the "B" RHR pump on September 1, 1996 was reportable as a condition prohibited by the Technical Specifications.

Corrective Steps that will be taken

The reporting procedure and accompanying guidance document will be revised by September 30, 1997 to provide more examples of a conservative reporting philosophy in the defueled condition.

Full compliance will be achieved by September 30, 1997.

Restatement of the Violation

E. TS 3.1.2.1 requires during reactor operations in Mode 5 that at least one boron injection flow path be operable.

Contrary to the above, on August 28, 1996, with the reactor in Mode 5, at least one boron injection flow path was not operable. Specifically, while attempting to establish a boron injection flowpath from the boric acid mix tank through a charging pump to the reactor in accordance with SUR 5.1-159B, a valve lineup error allowed nitrogen gas to be introduced into the charging system, rendering the boron injection flow path inoperable. The licensee failed to declare the boration flow path inoperable as nitrogen continued to leak into the charging system and the reactor from August 28, 1996 until September 1, 1996. (09014)

This is a Severity Level IV violation (Supplement I).

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by operator misdiagnosis of the problem. A contributing factor was the lack of monitoring equipment, specifically RVLIS to monitor reactor vessel water level.

Corrective Steps that have been taken and the results achieved

The corrective actions associated with the nitrogen intrusion event are described in Attachment 1.

Once the nitrogen intrusion problem was identified, the boration flow path was immediately restored.

A new Operation's Department Instruction, ODI 190, "RCS Inventory in Modes 5 and 6," was issued on September 1, 1996 to record and trend RCS level readings and consumption of gasses on site. Additionally, procedures for performing a RCS drain down include steps for calculating an inventory balance and following through on comparing expected results with predicted results.

ODI 1, "Conduct of Operations," was revised to place an emphasis on having a questioning attitude. New expectations for off-normal conditions were outlined to improve management performance and lower the threshold for operating crews to seek assistance. Guidance was provided on when to call in assistance to ensure a multi-discipline approach for off-normal events. Roles and responsibilities between line and oversight organizations were clarified. Training was provided on applicable procedures and department instructions.

ODI 191, "Shutdown Risk Awareness," was generated to increase the awareness of the five key safety functions (decay heat removal, RCS inventory, electrical power, reactivity control, and containment), including associated procedural controls and operational philosophies, among plant operators.

Jumpers were installed subsequent to the event, to provide one train of reactor vessel level indicating system (RVLIS) indication and to provide two core exit thermocouples (CET) readings to control room operators.

Compliance has been achieved.

Restatement of the Violation

F. 10 CFR Part 50 Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis for structures, systems and components are correctly translated into specifications, drawings, procedures and instructions.

Contrary to the above, as of September 27, 1996, the results of design basis calculations for safety-related instrumentation setpoints were not translated into the plant TS and instrumentation calibration procedures as evidenced by the following examples:

1. Setpoint calculations did not assure that the design basis requirements were translated into the TS allowable values for safety-related instrumentation. Specifically, incorrect allowable values were calculated for calculations that were performed to support a 24-month fuel cycle operation.

2. The results of design basis calculations for instrumentation setpoints were not translated into instrumentation calibration surveillance procedure acceptance criteria. Specifically, the instrument uncertainty results of calculation PA 90-013-26EY Rev. 2, "Uncertainties and Setpoints for Steam Generator Narrow Range Level L-1301-1A/C/D, 2A/C/D, 3A/C/D , 4A/C/D," were not translated into appropriate calibration acceptance criteria in SUR 5.2-6.1, "Steam Generator #1 Narrow Range Level Channel Calibration," resulting in non-conservative acceptance criteria. (10014)

This is a Severity Level IV violation (Supplement I).

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by the failure to revise associated procedures when a setpoint calculation was revised and potential non-conservative assumptions in certain setpoint calculations.

Corrective Steps that have been taken and the results achieved

These setpoint calculations are not applicable in the defueled mode.

The Design Control Manual has been revised to require identification of procedures to be revised in the event of a calculation revision.

Corrective Steps that will be taken

Actions to correct programmatic issues were identified in our common cause study and are currently being implemented. The details of the programmatic changes related to the Design Control effort including schedule and implementation status are presented in Attachment 1.

Full compliance will be achieved once the revised programs are implemented.

Restatement of the Violation

G. 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

Contrary to the above, during instrument calibrations performed in February 1995, Instrument Calibration Review Forms (ICRs) 95-009, 95-011, 95-23, 95-24, and 95-025 documented instrumentation calibration acceptance criteria failures and the licensee did not identify the cause of the failures or implement corrective action to prevent repetition. (11014)

This is a Severity Level IV violation (Supplement I).

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by the failure of the Instrument Calibration Review (ICR) results to be tracked and trended properly.

Corrective Steps that have been taken and the results achieved

These ICRs are not applicable in the defueled mode.

The ICR procedure, WCM 2.3-7, has been revised and issued to require a determination of the cause and corrective action of unanticipated calibration results.

Corrective Steps that will be taken

Actions to correct programmatic issues were identified in our common cause study and are currently being implemented. The details of the programmatic changes related to the Design control effort including schedule and implementation status are presented in Attachment 1.

Full compliance will be achieved once the revised programs are implemented.

Restatement of the Violation

H. TS 3.1.2.1 and 3.1.2.2 require that boron injection flow paths be operable during operation in Mode 5 & 6 and Modes 1 through 4, respectively, including a flow path from the boric acid tank to the metering pump. TS 4.1.2.1.a and 4.1.2.2.a require the temperature of the heat traced portion of the flow path from the boric acid tank to be greater than 140 degrees F.

Contrary to the above, during plant operation in Modes 1 through 6 prior to October 10, 1996, certain locations in the boron injection flow path had temperatures which were below the required minimum of 140 degrees F, rendering the associated portions of the boration system inoperable. On October 10, temperatures were measured to be as low as 120 degrees F in the gravity feed line to the metering pump, and 90 degrees F at the suction of the charging pumps at the junction of the discharge from the boric acid pumps. (114)

This is a Severity Level IV violation (Supplement I).

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by inadequate design of the heat trace and controls.

Corrective Steps that have been taken and the results achieved

LER 96-027-00 (submitted November 7, 1996) provided the details on cause and corrective actions for this issue. The 9 watt per foot heat trace was replaced with 6 watt per foot heat trace. The RTD's were relocated to the coolest part of the associated piping, thereby ensuring that the entire piping run was heated above the Technical Specification minimum temperature.

For the defueled condition, a review of required freeze protection has been completed and changes implemented.

Corrective Steps that will be taken

Since this system is no longer required in the defueled mode, no additional corrective action will be taken. Compliance has been achieved.

Restatement of the Violation

I. TS 6.8.1 requires, in part, that written procedures and/or administrative policies shall be established, implemented, and maintained covering the activities as recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, Section F.25 requires procedures to be established to combat significant events such as irradiated fuel damage during refueling.

Contrary to the above, written procedures had not been established, implemented, and maintained in that prior to October 24, 1996, a procedure to combat a significant event such as irradiated fuel damage during refueling did not exist. (13014)

This is a Severity Level IV violation (Supplement I).

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by the failure to have a written procedure, specifically having incomplete or non-existent procedures for performing certain actions, which caused workers to take action, relying on their own knowledge and skills without the benefit of multi-discipline reviews.

Corrective Steps that have been taken and the results achieved

AOP 3.2-63, Rev. 0, "Fuel Handling Accident" was prepared and approved by PORC on 10/23/97.

Additional corrective actions related to our common cause assessment are provided in Attachment 1, Section II.

Full compliance will be achieved once the revised programs are implemented.

Restatement of the Violation

J. TS 3/4.9.12 requires the Fuel Storage Building Air Cleanup System to be operable and in operation during operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool with a flowrate of 4,000 +/- 10% cubic feet per minute (cfm). The TS action statement states with the Fuel Storage Building Air Cleanup System inoperable, or not operating, all operation with loads over the fuel storage pool are to be suspended.

Contrary to the above, between May 27 and June 14, 1993, and between February 6 and February 28, 1995, during fuel movement within the fuel storage pool, the Fuel Storage Building Air Cleanup System was inoperable, in that the system flowrate was less than 4,000 cfm +/-10%, and fuel movement operations were not suspended. (14014)

This is a Severity Level IV violation (Supplement I).

Reason for the Violation and Causes

Connecticut Yankee Atomic Power Company (CYAPCO) does not dispute the cited violations. This violation was caused by the failure to fully understand the impact of the coincident operation of the PAB and SFB air cleanup systems on the air flow test.

Corrective Steps that have been taken and the results achieved

LER 96-025-01 (submitted February 21, 1997) was prepared to describe the cause and corrective actions associated with this issue. Initial corrective action was to adjust a SFB air cleanup system damper to achieve the required air flow rate. Additionally, the operating procedures were revised to verify adequate air flow prior to moving fuel or operating the crane with a load over the spent fuel pool.

Compliance has been achieved.