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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket No. 50-271

JAN 23 1991

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19 cys

MEMORANDUM FOR: Chairman Carr
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick

FROM: Steven A. Varga, Director
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: UPDATED BOARD NOTIFICATION REGARDING ALLEGATIONS ON SPENT FUEL POOL COOLING SYSTEM AT VERMONT YANKEE (BN 91-02)

On August 23, 1990, the staff issued BN 90-09 which informed the Commissioners, cognizant Hearing and Appeal Boards, and other parties to the Vermont Yankee Spent Fuel Pool proceedings of the receipt of allegations related to the operation of the Spent Fuel Pool Cooling System.

The staff has resolved this allegation by issuance of Inspection Report 50-271/90-10 dated November 27, 1990 and, as explained below, we now consider this matter closed. A Notice of Violation (NOV) was issued with the Inspection Report. The NOV stated that a fuel pool cooling pump was inoperable for more than 30 days and that the pump motor was not environmentally qualified due to a short to ground in the motor windings. Contrary to plant procedures, the station was not placed in a cold shutdown condition after the 30-day period. In response, the licensee in a letter dated December 27, 1990, stated that the pump was capable of running. The licensee requested the staff to review the basis for the alleged violations and to rescind the violations, thus keeping this enforcement action open.

In regard to the safety consequences of operation with this degraded cooling system, Inspection Report 90-10 concluded, "... it is apparent that a number of installed design features provide appropriate means of mitigating the consequences of the licensee's actions." Therefore, while the allegation had merit, the probability of the fuel pool water exceeding the boiling point with both trains of the cooling system out of operation and a concurrent loss-of-coolant accident was found to be remote. The licensee has a procedure in place for this contingency. A copy of the Inspection Report has been previously sent to all Board Members and parties to the proceeding. We are enclosing a copy for each of the Commissioners.

Copies of this notification are also being provided to the Licensing Board in the Construction Permit Recapture Proceeding, for their information.

Edward J. [illegible] /for
Steven A. Varga, Director
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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PDR ADOCK 05000271
P PDR

Enclosure:
Inspection Report 50-271/90-10

per 1/29/91 telcon w/ Jim McKnight at ~3:05pm
N005

cc w/o enclosures: SECY (3)
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T. Murley
J. Sniezek
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Vermont Yankee Nuclear Power Station, Docket No. 50-271

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Atomic Safety and Licensing Board
Panel Docket
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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UNITED STATES
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Docket No. 50-271

MEMORANDUM FOR: Chairman Carr
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Commissioner Remick

FROM: Steven A. Varga, Director
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: UPDATED BOARD NOTIFICATION REGARDING ALLEGATIONS ON SPENT FUEL POOL COOLING SYSTEM AT VERMONT YANKEE (BN 91-02)

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The staff has resolved this allegation by issuance of Inspection Report 50-271/90-10 dated November 27, 1990 and, as explained below, we now consider this matter closed. A Notice of Violation (NOV) was issued with the Inspection Report. The NOV stated that a fuel pool cooling pump was inoperable for more than 30 days and that the pump motor was not environmentally qualified due to a short to ground in the motor windings. Contrary to plant procedures, the station was not placed in a cold shutdown condition after the 30-day period. In response, the licensee in a letter dated December 27, 1990, stated that the pump was capable of running. The licensee requested the staff to review the basis for the alleged violations and to rescind the violations, thus keeping this enforcement action open.

In regard to the safety consequences of operation with this degraded cooling system, Inspection Report 90-10 concluded, "... it is apparent that a number of installed design features provide appropriate means of mitigating the consequences of the licensee's actions." Therefore, while the allegation had merit, the probability of the fuel pool water exceeding the boiling point with both trains of the cooling system out of operation and a concurrent loss-of-coolant accident was found to be remote. The licensee has a procedure in place for this contingency. A copy of the Inspection Report has been previously sent to all Board Members and parties to the proceeding. We are enclosing a copy for each of the Commissioners.

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Edward G. Hummel /for
Steven A. Varga, Director
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Inspection Report 50-271/90-10

cc w/o enclosures: SECY (3)
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T. Murley
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Vermont Yankee Nuclear Power Station, Docket No. 50-271

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NOV 2 7 1990

Docket No. 50-271

Vermont Yankee Nuclear Power Corporation
ATTN: Mr. Warren P. Murphy
Senior Vice President, Operations
RD 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

Gentlemen:

Subject: NRC Region I Inspection Report No. 50-271/90-10

This refers to the routine NRC safety inspection conducted on August 13 - October 9, 1990, at the Vermont Yankee Nuclear Power Station, Vernon, Vermont. The results of the inspection were discussed with Mr. D. Reid at the conclusion of the inspection.

Based on the results of this inspection, it appears that certain of your activities associated with the Fuel Pool Cooling System were not conducted in full compliance with NRC requirements, as set forth in the Notice of Violation enclosed as Appendix A. These violations have been categorized by severity level in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions."

In addition, a deviation from your May 3, 1985 commitment to maintain operable equipment (which is addressed by the Vermont Yankee Environmental Qualification Program) was identified. This deviation is described in Appendix B.

During our evaluation of these issues, two weaknesses were identified. First, operators and some key supervisors were not fully aware of the administrative requirements of a management directive and the fuel pool cooling system operating procedure. Second, the independent review of plant equipment tagouts needed strengthening. You are required to respond to this letter and, in so doing you should follow the instructions in Appendices A and B. In your response, please address these two concerns stated above.

Your cooperation with us is appreciated.

Sincerely,

Original Signed By:
Jon R. Johnson

Jon R. Johnson, Chief
Reactor Projects Branch No. 3
Division of Reactor Projects

IE01

Enclosure:
1. Appendix A, Notice of Violation
2. Appendix B, Notice of Deviation
3. NRC Region I Inspection Report No. 50-271/90-10

cc w/encls:
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D. Reid, Plant Manager
J. DeVincentis, Vice President, Yankee Atomic Electric Company
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Washington, D.C. 20555

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King of Prussia, Pennsylvania 19406

APPENDIX A

NOTICE OF VIOLATION

Vermont Yankee Nuclear Power Corporation
Vermont Yankee Nuclear Power Station

Docket No. 50-271
License No. DPR-28

During a routine NRC inspection conducted on August 13 - October 9, 1990, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1990), the violations are listed below.

A. Technical Specification Section 6.5, Plant Operating Procedures, requires that detailed written procedures involving both nuclear and non-nuclear safety, covering operation of systems and components of the facility including applicable check-off lists and instructions shall be prepared, approved, and adhered to. Operating Procedure OP 2184, Fuel Pool Cooling System, requires that from and after the date that one of the fuel pool cooling subsystems is made or found inoperable (and the remaining subsystem is capable of maintaining the fuel pool temperature below 150 degrees F) then the reactor shall be in cold shutdown within thirty days unless such subsystem is sooner made operable.

   Contrary to the above, between August 4, 1989 and July 3, 1990 the reactor was not placed in a cold shutdown condition, when the "A" fuel pool cooling subsystem remained inoperable for more than thirty days with the "A" fuel pool cooling pump power supply breaker, P9-1A white tagged (Danger Tagged) in the open position.

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

B. 10 CFR 50, Appendix B, Criterion XVI, requires that conditions adverse to quality, such as defective equipment and nonconformances be promptly identified and corrected. Additionally, 10 CFR 50.49(f) requires that electrical equipment important to safety be qualified, in part, by testing or by analysis in combination with partial type test data. As stated in the licensee's Environmental Qualification Program Manual, the "A" Spent Fuel Pool cooling pump motor is environmentally qualified (electrical) equipment important to safety.

   Contrary to the above, the "A" Spent Fuel Pool cooling pump motor was not qualified, due to a lack of testing or analysis in the degraded condition. Between June 9, 1989 and July 27, 1990, the pump motor was in a degraded condition in that at least one phase of the motor winding shorted to ground following a brief period of operation. This condition adverse to quality represents a nonconformance that was not promptly identified and corrected.

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

Pursuant to 10 CFR 2.201, Vermont Yankee Nuclear Power Corporation is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C., 20555 with a copy to the Regional Administrator, Region I, and a copy to the NRC Resident Inspector, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reasons for the violation, or if contested, the basis for disputing the violation; (2) the corrective steps that have been taken and the results achieved; (3) corrective steps that will be taken to avoid further violations; and (4) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order may be issued to show cause why the license should not be modified, suspended or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending this response time.

Under the Paperwork Reduction Act of 1980, PL 96-511, the response directed above is not subject to clearance by the Office of Management and Budget.

APPENDIX B

NOTICE OF DEVIATION

Vermont Yankee Nuclear Power Corporation — Docket No. 50-271
Vermont Yankee Nuclear Power Station — License No. DPR-28

During a routine NRC inspection conducted on August 13 - October 9, 1990, a deviation of the licensee's written commitment of May 3, 1985 was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1990), the deviation is listed below.

> Vermont Yankee Nuclear Power Corporation letter to the NRC, dated May 3, 1985, stated that it is the policy of Vermont Yankee's corporate management that all equipment and components which are addressed by Vermont Yankee's Environmental Qualification (EQ) program shall be maintained operable and fully environmentally qualified at all times, commensurate with the status of the plant. In addition, the licensee committed that whenever safety class equipment or components which are EQ but are not covered by Vermont Yankee Technical Specifications fail (are not operable), a Nonconformance Report shall be generated with disposition of the discrepancy provided within 30 days.
>
> Contrary to the above, on July 5, 1989, the "A" Spent Fuel Pool level instrumentation channel equipment (safety class and addressed by Vermont Yankee's EQ program) was made inoperable by the removal of its power source. This condition remained until July 3, 1990, and a Nonconformance Report had not been generated to disposition the discrepancy.

Please provide to the U.S. Nuclear Regulatory Commission, ATTN: Document control Desk, Washington, D.C., 20555, with a copy to the Regional Administrator, Region I, and a copy to the NRC Resident Inspector, in writing within 30 days of the date of this Notice, the reason(s) for the deviation, the corrective steps which have been taken and the results achieved, the corrective steps which will be taken to avoid further deviations, and the date when your corrective action will be completed. Where good cause is shown, consideration will be given to extending the response time.

Under the Paperwork Reduction Act of 1980, PL 96-511, the response directed above is not subject to clearance by the Office of Management and Budget.

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-271/90-10

Docket No. 50-271 License No. DPR-28

Licensee: Vermont Yankee Nuclear Power Corporation
RD 5, Box 169
Brattleboro, Vermont 05301

Facility: Vermont Yankee Nuclear Power Station

Location: Vernon, Vermont

Dates: August 13 - October 9, 1990

Inspectors: Harold Eichenholz, Senior Resident Inspector
Thomas G. Hiltz, Resident Inspector
Morton B. Fairtile, Project Manager, NRR
Jason C. Jang, Sr. Radiation Specialist, DRSS
Leonard S. Cheung, Senior Reactor Engineer, DRS
Peter D. Drysdale, Senior Reactor Engineer, DRS
Carl H. Woodard, Reactor Engineer, DRS

Approved by: John F. Rogge 11/27/90
John F. Rogge, Chief, Reactor Projects Section 3A Date

Inspection Summary: Inspection on August 13 - October 9, 1990

Areas Inspected: Resident safety inspection of the following areas: plant operations, radiological controls, maintenance and surveillance, security, engineering and technical support, safety assessment and quality verification, and allegation followup.

Results: Inspection results are summarized in the attached Executive Summary.

EXECUTIVE SUMMARY

VERMONT YANKEE INSPECTION REPORT 50-271/90-10

AUGUST 13 - OCTOBER 9, 1990

Plant Operations

A full reactor protection system actuation was received due to a spike in a shared local power range monitoring instrument while shutdown. Corrective actions to address this event are not fully complete and the event report remains open. A proper safety perspective and an aggressive questioning attitude was exhibited during resolution of a refueling bridge interlock which inhibited scheduled operations. Vermont Yankee management demonstrated a balance between acceptable equipment performance and personnel ALARA considerations in resolving control rod drive (CRD) equipment problems. A violation involving failure to follow procedural requirements for operation of the spent fuel pool cooling system was identified during the followup of an allegation (VIO 90-10-04).

Radiological Protection

Licensee corrective actions described in LER 90-05 appear adequate to prevent a repeat violation of containment air sampling requirements (NCV 90-10-07). Cobalt-60 was detected during exit whole-body counts taken on four contract employees. Management provided timely assessment of the health effects of the cobalt-60 uptakes and demonstrated an appropriate response to address turbine floor contract workers' concerns. The ALARA program effectiveness benefitted from several changes implemented during the refueling outage. Licensee corrective actions early in the refueling outage corrected deteriorating control of Radiation Work Permit (RWP) activities. A non-cited violation (NCV 90-10-01) was caused by a contract worker who did not adhere to RWP procedures.

Maintenance and Surveillance

The effect of potential clutch gear failure on motor operated valves was evaluated by Vermont Yankee. Implementation of the licensee program to ensure updated vendor information requires further assessment and is unresolved (UNR 90-10-02). The trial LCO preventive maintenance program was reviewed and potential weaknesses were identified (UNR 90-10-03). LERs describing two missed surveillance events were evaluated. The inspector reviewed core verification and control rod drive friction testing. Loading requirements for the monthly diesel generator operability test surveillance were not in accordance with Technical Specifications (NRC review of this issue is discussed in Report No. 50-271/90-80). A deviation from licensee written commitments was identified that involved inadequate dispositioning of inoperable EQ equipment (DEV 90-10-06). This condition resulted from the inadequate manner in which the licensee evaluated and controlled maintenance activities on the "A" spent fuel pool cooling pump.

Executive Summary

Security

A random cocaine drug search was considered a positive licensee initiative. Three incidents involving improper access control/clearance and fitness-for-duty are discussed and were the subject of a special NRC Region I security inspection (NRC Inspection Report 50-271/90-11).

Engineering and Technical Support

Turbine control oil system responses are undergoing further evaluation during the post-outage turbine startup and an associated event report (LER) is considered open. The technical support and evaluation provided to assess Cycle 14 fuel failures demonstrated a strong technical competence. Due to the inability of the licensee's staff to properly identify and promptly correct degradation of the motor on the "A" spent fuel pool cooling pump from an EQ program perspective, a violation of 10 CFR 50, Appendix B and 10 CFR 50.49 was identified (VIO 90-10-05).

Safety Assessment and Quality Verification

A NUREG-0737 commitment (Clarification of TMI Action Plan Items) item was reviewed and closed. An unresolved item (UNR 50-271/86-22-04) regarding whether the reliability of the ATWS RPT System is consistent with Generic Letter 85-06, was closed. The inspectors evaluated the effectiveness of the Vermont Yankee quality assurance program and determined that in general it was effective.

TABLE OF CONTENTS

DETAILS

1. SUMMARY OF OPERATIONS

Vermont Yankee Nuclear Power Station entered the report period operating at approximately 90 percent of rated core thermal power in end-of-core life power coastdown. Two Limiting Condition for Operation (LCO) preventive maintenance periods were conducted between August 14 and August 17 and between August 21 and August 24. During these periods, selected Residual Heat Removal (RHR) pumps, RHR service water pumps, and Low Pressure Coolant Injection flow paths were taken out-of-service to perform preventive maintenance. On August 31 at 11:00 a.m., operators commenced a reactor shutdown from approximately 86 percent rated core thermal power. Approximately 10 hours later all control rods were full-in and the reactor mode switch was placed in the "Refuel" position. A cooldown was started and reactor vessel water temperature was brought to below 212 degrees F using RHR shutdown cooling. Refueling and maintenance activities commenced. A neutron monitoring system High-Flux scram signal was received on September 2. The scram was the result of spiking on Local Power Range Monitor LPRM 4B-32-25.

The major work completed during the refueling outage included fuel sipping, removal and replacement of lower support plate bypass flow hole plugs, feedwater check valve replacement, low pressure turbine refurbishment, control room human factors enhancements, replacement of control rod drive mechanisms, replacement of the uninterruptable power supply system and motor-operated valve maintenance and testing.

Most plant restoration was complete at the close of the inspection period with the exception of the main turbine.

2. PLANT OPERATIONS (71707, 86700, 93702, 92700)

2.1 Inspection Activities

The inspectors verified that the facility was operated safely and in conformance with regulatory requirements. Management control was evaluated by direct observation of activities, tours of the facility, interviews and discussions with personnel, independent verification of safety system status and Limiting Conditions for Operation, and review of facility records.

2.2 Inspection Findings and Significant Plant Events

This period included deep backshift and weekend inspections conducted on August 14, 21, 31, September 1, 2, 3, 4, 8, 9, 12, 18, 22, 26, and October 2, 5, and 8. Operators and shift supervisors were alert, attentive and responded appropriately to annunciators and plant conditions.

A. LER 90-11: Full Reactor Protection System Actuation from Spike in a Shared LPRM

On September 2, 1990, with the reactor shut down, a high-high flux signal was received from average power range monitors (APRMs) "C" and "F" resulting in a full scram signal. High spiking on shared local power range monitor (LPRM) 4B-32-25 provided a high-high flux input to both APRM "F" and APRM "C" which resulted in a reactor scram signal. At the time the scram signal was received, the control rods were already full-in and no change in reactor power was experienced.

A maintenance request (MR) was generated to investigate the cause of the event and make necessary repairs to the LPRM (MR-90-2244). This maintenance request was canceled and the work was placed under the scope of MR-90-3101. The root cause of the event, as initially assessed in LER 90-11, was believed to be a loose or broken connector resulting from under vessel repositioning of instrument monitoring cables. A pre-startup under vessel inspection, conducted after issuance of LER 90-11, identified no discrepancies with this instrument cable. The licensee has now focused investigation efforts on the LPRM detector.

The inspector concluded that further corrective actions are warranted and that the information contained in LER 90-11 needs updating. This LER remains open.

B. Refueling and Spent Fuel Pool Activities

The inspector reviewed a sampling of the licensee's refueling procedures and observed refueling operations in progress on the refueling floor in the reactor building. The major activities observed by the inspector were: (1) removal of Cycle 14 fuel from the core, transfer to the Spent Fuel Pool (SFP) and placement in the spent fuel racks; (2) the conduct of fuel sipping operations to detect failed fuel bundles; and (3) movement of fuel assemblies into the reactor from the SFP. Observations and discussions with personnel on the refueling floor indicated that personnel were knowledgeable of their job requirements and work progressed in an orderly, professional manner. A proper level of oversight and involvement by a Senior Control Room Operator (SRO licensed) on the refueling floor was noted by the inspector. Good communication was maintained between personnel in the control room and refueling floor.

C. Refueling Bridge Interlocks

The reactor Mode Switch was placed in the "Shutdown" position on September 8, 1990. This action was one of the conditions necessary to support the removal of the "B" Emergency Diesel Generator and the "B" Emergency Core Cooling Systems from service to support maintenance activities. During this period, the licensee's refueling outage plans called for conducting fuel sipping operations in the Spent Fuel Pool (SFP), reactor vessel inspections, and core plug replacements. All of these activities required the use of the refueling platform. New techniques were being employed this outage in the conduct of some of these activities. Specifically, the reactor vessel inspection of the jet pumps and the core plug replacement were to be done with a flooded-up cavity and use of the fuel grapple. In the past, the licensee used a service platform installed over the reactor vessel head flange with the cavity in a drained down condition. Scheduling and ALARA improvements were the intended benefits of the revised procedures.

When the operators attempted to move the refueling platform from the SFP area towards the core, the platform stopped prior to entering the core cavity area. The licensee investigated the inability to move the refuel bridge over the core, and discovered that the circuits controlling the bridge movement had a design feature that precluded the intended actions. This condition only occurred because the Mode Switch was in "Shutdown." If the Mode Switch was in the "Refuel" position, no prohibition would have be present. This design feature was not part of the licensed refueling interlocks and was essentially unknown to the plant staff because they had no prior experience with using the refuel bridge over the reactor core with the Mode Switch in "Shutdown."

The inspector observed the conduct of the technical reviews by the plant staff in response to the unexpected inhibit on the refuel bridge. The involvement of senior plant managers and the Senior Vice President - Operations was noted. Management sensitivity to this issue was due, in part, to a November 7, 1973 inadvertent criticality incident that occurred due to the defeating of refueling interlocks to support refueling activities. Temporary Modification (TM) No. 90-51 was developed to defeat the interlock that prevented the movement of the refuel bridge over the core with the Mode Switch in other than the "Refuel" Mode. The inspector attended the Plant Operations Review Committee Meeting convened to review the TM. By 1:30 p.m. on September 8, 1990, the TM was installed and the issue resolved.

The inspector verified that implementing instructions and post implementation testing were specified and conducted to assure that no unintended loss of design features occurred. In addition, proper controls were specified by the licensee to preclude the accidental lifting of a fuel bundle because of the use of the fuel grapple.

A proper safety perspective was evident in the manner in which the licensee responded to this unanticipated situation. A thorough and probing technical review of conditions was evident. The review of a prior major event of significance was conducted to ensure no concerns were identified due to the planned implementation of TM No. 90-51. A good level of licensee management involvement was noted, which provided additional assurance that the plant staff's resolution to the issue was appropriate.

D. Control Rod Drive Problems

During the refueling and maintenance outage, Vermont Yankee planned to change-out ten control rod drives (CRD) and 29 control rod blades. The CRDs to be changed-out would be disconnected from under the vessel and exchanged with rebuilt CRDs using a new drive change-out machine. By using this new change-out machine, Vermont Yankee anticipated faster CRD change-out, less under vessel time, and consequently less personnel radiation exposure.

Equipment problems caused the CRD change-out activities to become a critical path outage item. Management decided to reschedule replacement of one of the CRDs.

In total, 12 control rod drives were replaced during the refueling and maintenance outage. Nine of these drives were scheduled replacements and three drives were replaced due to equipment inadequacies.

The inspector concluded these activities associated with CRD replacement reflected a balance between acceptable equipment performance and personnel radiation exposure. The decision to replace the three degraded CRDs was made after a careful consideration of all corrective options. The inspector determined that additional reduction in personnel exposures could be gained through improved equipment reliability. The inspector could not assess the impact the new handling equipment had on reducing personnel exposures. The inspector noted excellent coordination between Reactor and Computer Engineering personnel, Operations personnel, ALARA and Radiation Protection personnel, and contractor personnel.

3. RADIOLOGICAL CONTROLS (71707, 92701)

3.1 Inspection Activities

Implementation of the radiological protection program was verified on a periodic basis.

3.2 Inspection Findings and Review of Events

A. (Closed) Unresolved Item 90-02-01: Review of LER 90-05, Incomplete Evaluation of Containment Air Sample

Licensee Event Report (LER) 90-05, in conjunction with the NRC Unresolved Item 90-02-01 was reviewed by a Region I based inspector. On March 15, the licensee analyzed a containment air sample before venting to the atmosphere, either through the stack directly or through the Standby Gas Treatment System, as required by Section 3/4.8.L.1 of the Technical Specifications (TS). On March 23, the licensee discovered that an incomplete evaluation of the containment air sample analytical result was performed prior to venting the containment. The analytical results of the containment air sample slightly exceeded TS limits, therefore, the containment air should have been vented through the Standby Gas Treatment System as required by the TS. However, the licensee vented the containment to the stack directly. The root cause of this event was failure to follow procedures. The procedure required an evaluation be performed to ensure that applicable limits were met. Failure to meet TS requirements and to follow procedures to provide an adequate review process during the event on March 15 constitute an apparent violation of the TS.

However, failure to comply with the requirements during this event is considered a licensee identified violation in that (1) it was identified by the licensee; (2) it fits into Severity Level IV or V; (3) it was reported; (4) the licensee took aggressive actions to correct the deficiency and to prevent future recurrence; and (5) this was the first occurrence of this type of event (NCV 50-271/90-10-07). The licensee's actions have been substantially completed in this area and appear to be adequate to prevent recurrence. The evaluation of the radiological consequences this event indicated that no significant potential for adverse offsite impact had occurred. Consequently, no notice of violation will be issued and this issue is considered closed.

B. Cobalt-60 Intake

On September 12 detectable levels of Cobalt-60 (Co-60) were found during exit whole body counts of four contract workers. The four contract workers were involved in maintenance activities on the turbine floor. Co-

60 is formed from neutron activation of Cobalt-59. Cobalt-59 is a constituent of stellite and stainless steel which through normal corrosion can become irradiated by neutrons in the reactor.

One worker was released af . an additional whole body count. The three other workers were deconta ted, recounted and required to return the next day for an additional whole body count. Subsequent whole body counts revealed that two workers showed no indication of internal contamination. The third individual indicated low levels of Co-60 and was appropriately released. On September 18, this individual returned to Vermont Yankee and received an entrance whole body count which showed no trace of activity.

Vermont Yankee management recognized the potential effects that lack of knowledge and misunderstanding of this incident could have on turbine floor workers. Management immediately responded to the concerns of turbine floor workers. Additional body counts of workers were scheduled and on September 13 the radiological protection supervisor, the maintenance supervisor, the ALARA engineer, and the senior dosimetry assistant met with turbine floor workers to discuss the incident, address individual concerns, and reemphasize each individuals rights to access their exposure history. The inspector monitored one of these meetings and determined that it was responsive, open, timely, and thorough. The workers appeared to have benefitted from these discussions.

The highest maximum permissible organ burden (MPOB) to any individual was approximately 4%. This individual was subsequently recounted and MPOB determined to be approximately 2%. The final whole body count of this individual showed no trace of activity. Total individual exposure due to Co-60 was less than 5 millirem. Swipe contamination surveys indicated measurable levels of Co-60. The licensee evaluated the work activities, experience of workers, and potential sources of contamination. The licensee determined that ingestion likely occurred as a result of poor individual radiological practices.

As a result of this incident the licensee increased radiological protection technician coverage on the turbine deck, increased survey frequency, and provided additional high efficiency particulate (HEPA) filters for use in enclosed turbine work areas. The inspector determined that licensee response to this event was acceptable.

C. Refueling Outage ALARA Program

Vermont Yankee implemented several changes with the 1990 refueling outage ALARA program. The most significant change required all initial radiation work permits (RWPs) to be drafted by designated ALARA coordinators. ALARA coordinated surveys which could be performed prior to commencement of the outage were completed. A daily ALARA report was published which contained actual versus estimated exposures for 31 work packages that could involve exposures greater than one man-rem.

At the end of the inspection period the inspector noted that several work packages had exceeded their man-hour, man-rem ALARA estimates. In almost all cases this appeared to be the result of increased job scope, not poor ALARA estimates.

The inspector observed ALARA coordinators actively involved in their assigned work packages. The ALARA program was widely published and fully supported by management. The inspector concluded, based on in-plant observation, ALARA report review and discussions with the ALARA engineer, that the ALARA program was effective during the 1990 refueling outage.

D. Routine Inspection and Radiological Protection Walkdown

On September 25 the inspector conducted a radiological protection (RP) walkdown with both RP shift supervisors. The inspector toured the turbine building and the reactor building, inspecting most RWP areas. The inspector found the RP shift supervisors to be knowledgeable. During the walkdown the inspector addressed several areas of interest including: (1) radiological housekeeping, (2) qualification and performance of contractor RP personnel, (3) licensee identified areas of concern, (4) frequency of radiological surveys, (5) management involvement in the outage radiological program, (6) number of RP incidents (specifically those involving personal contamination), and (7) radiological posting requirements.

In the area of radiological housekeeping the inspector questioned the RP shift supervisors on Vermont Yankee policies for the use of yellow herculite and personnel contamination (PC) clothing for general cleanliness purposes. The inspector noted PCs hanging from a valve handwheel inside a RWP area, and used respirators inside another RWP area. A posting for a high radiation area on the torus catwalk had fallen and some required information was missing. This was identified by the RP shift supervisor during the walkdown and immediately corrected. During subsequent tours the inspector noted improvement in housekeeping.

Vermont Yankee took aggressive corrective action early in the outage to correct deteriorating control of RWP activities. In particular, work activities in the turbine building appeared to be progressing at a pace inconsistent with appropriate radiological coverage. Management recognized the problem and withdrew all active RWPs. The licensee evaluated the RP manpower resources and experience levels, the scope and volume of work activities, and systematically re-issued the radiation work permits. Senior technicians were judiciously positioned consistent with the level of radiological concern. Access to the turbine building was limited, and workers were processed through an RP checkpoint. The inspector concluded that these corrective actions were adequate.

E. Contractor Worker Not Adhering to Radiological Posting

During a routine tour of the reactor building on October 2, the inspector determined that a contractor individual engaged in vacuuming activities inside a RWP posted area did not meet the minimum dress requirements for entrance into the area. The inspector also determined that the individual was not signed onto an RWP, and the individual was not fully aware of requirements for entering the posted area. The inspector notified the RP checkpoint and a technician was sent to investigate. As a result of this event, an RP incident report was generated, the individual's dosimetry was withdrawn, and remedial training was prescribed. The inspector spoke with the plant health physicists concerning this event and licensee corrective actions. In absence of similar previous incidents, the inspector determined that this was an isolated occurrence. The inspector determined that, due to the minor safety significance and the prompt corrective actions taken by the licensee, this finding met the criteria, specified in 10 CFR 2, Appendix C, Section V.A for a non-cited violation (NCV 50-271/90-10-01).

4. MAINTENANCE AND SURVEILLANCE

4.1 Maintenance Inspection Activity (62703)

The inspectors observed selected maintenance activities on safety-related equipment to ascertain that these activities were conducted in accordance with approved procedures, Technical Specifications, and appropriate industry codes and standards.

4.2 Maintenance Observations

A. Potential Clutch Gear Failure on Motor Operated Valves

On September 10 a through-wall crack was found in the worm shaft clutch gear on MOV-14-11A (one of the core spray system discharge isolation valves). The valve was still operable, but the potential failure of this gear would prevent motor operation. This concern was identified to the NRC in a letter from the valve actuator vendor, Limitorque. The letter, dated August 13, 1985, did not list Vermont Yankee as a candidate for further evaluation. Limitorque's search of historical records did not indicate that any safety related valves, prone to identified failure characteristics, had been shipped to the site. The valves subject to this failure were limited only to size 2 Limitorque actuators (Type SMB, SB, and SBD), and only when combined with a two-pole AC-motor (3600 RPM, 60 Hz or 3000 RPM, 50 Hz) or a DC motor using an actuator ratio less than 55.84:1. In addition, the failure is prevalent only when this type actuator is combined with repetitive transfer of the actuator clutch mechanism from the manual (handwheel) to the motor drive mode.

VY identified seven safety related valves, required to reposition during postulated accidents, which are subject to this potential failure: recirculation system discharge valves (MOV-2-53A, MOV-2-53B); RHR system shutdown cooling inboard suction valve MOV-10-18; and core spray pump discharge isolation valves MOV-14-11A, MOV-14-11B, MOV-14-12A, and MOV-14-12B. Two other safety class valves, not required to automatically reposition during postulated accidents, were identified as candidates for this failure: recirculation pump suction valves MOV-2-43A, MOV-2-43B.

The clutch trippers for MOV-10-18, MOV-14-11A, MOV-14-11B, MOV-14-12A, and MOV-14-12B were removed. This eliminated engagement of the clutch under high speed, high inertia conditions as the clutch is always in the motor drive mode except when the valve is being manually operated. With the clutch trippers removed it is necessary to hold the declutch lever in the depressed position while turning the handwheel. The work on MOV-10-18, MOV-14-11A, MOV-14-12A, was completed under engineering change notice (ECN) 2 and incorporated into engineering design change request (EDCR 87-409). Modifications to MOV-14-11B and MOV-14-12B were completed under a temporary modification (90-53) and are planned to be integrated into a permanent modification in accordance with VY procedures. The cracked worm shaft clutch gear on MOV-14-11A was replaced. No other defects in the worm shaft clutch gears were noted.

The recirculation system valve actuator modification has been scheduled for the next refueling outage in February 1992. This decision was based on the frequency that these valves are operated in the manual mode and the result that no other worm shaft clutch gear failures were noted.

A previous unresolved item 85-22-02, relating to the issue was closed in inspection report 50-271/89-21. The item addressed the lack of a continuing program to ensure updated vendor information for safety-related components is identified and incorporated into maintenance and surveillance programs. This item was initially identified during an inspection to review and assess the licensee response to generic letter (GL) 83-28, Generic Implications of Salem ATWAS Event. Additional generic guidance was made available to utilities in Generic Letter 90-03. By letter dated September 27 to the NRC, Vermont Yankee responded to GL 90-03. In their response the licensee indicated that AO 0312, Equipment Technical Information, and procedure OP 0027, Nuclear Network, provided methods which adequately met the requirements of GL 90-03. The inspector concluded that the mechanism prompting contact with vendors is largely dependent on external information received through vendor manual updates or nuclear network. The NRC in a letter dated October 18 found the response acceptable, however, this event may indicate an inchoative program of periodic contact with vendors of key safety-related components and is worthy of further review. The implementation of the licensee program to ensure updated vendor information for safety-related components requires further assessment. This item is unresolved (UNR 50-271/90-10-02).

B. Performing Preventive Maintenance During Power Operations

Vermont Yankee recently adopted a policy for performing limiting condition for operation (LCO) preventive maintenance during power operations. The fundamental philosophy of this policy is to effectively minimize total out-of-service time for safety related equipment and thus potentially maximize system availability.

Two LCO preventive maintenance periods were scheduled during this inspection period. The first period commenced August 14 and ended August 17, and the second period commenced August 21 and ended August 24. The scope of work for each period was similar and involved work on residual heat removal (RHR) service water pumps, low pressure coolant injection valves, reactor recirculation units, and selected motor and manually operated valves associated with these systems.

The process for conducting LCO preventive maintenance during power operations was described in a VY memorandum dated July 16, 1990. The process is planned to be used on a trial basis until December 31 1990 and then reviewed and incorporated into the maintenance program.

The inspector reviewed LCO preventive maintenance planning checklists for the maintenance activities. The worksheet requires the Planning Coordinator to provide justification for performing the maintenance, selecting from a list of seven pre-defined justifications. The inspector concluded that the "multiple choice" method of maintenance justification is too general. Any maintenance activity could be justified using these seven criteria. For example, future reduction in system out-of-service time and improved reliability are logical and expected benefits derived from performing any preventive maintenance. The additional safety benefit gained by performing the maintenance activity at power was not clear to the inspector.

During plant tours, the inspector noted high levels of attention in areas associated with the performance of LCO preventive maintenance. The activities appeared to be well coordinated and properly managed. During a tour on August 16, the inspector noted that the reactor recirculation unit (RRU)-7 was out-of-service for maintenance. The RRU ensures that the environment in the area of the RHR and "A" core spray pump remains a mild environment and thus these systems remain environmentally qualified. The inspector questioned the operability of the core spray pump with the RRU inoperable. Previously, the licensee has declared equipment in the vicinity of the RRU, which is required to be environmentally qualified (EQ), inoperable when the associated RRU became inoperable.

In response to this concern, plant management developed a bases for determining operability of associated EQ equipment when one of the corner room RRUs (RRU 5, 6, 7, 8) becomes inoperable. The bases addresses the ability of the EQ equipment to perform its intended function with the RRU inoperable and concludes that the RRU's should not be implicitly linked to operability determinations for associated EQ equipment.

The inspector concluded the adequacy of justification for performing LCO preventive maintenance during power operations and the consistency in identifying the impact of interrelated system and component operability warrants additional review. This item is unresolved (UNR 50-271/90-10-03).

4.3 Surveillance Inspection Activity (61726, 62703, 92700, 90712, 92701)

The inspectors performed detailed procedure reviews, witnessed in-progress surveillance testing, and reviewed completed surveillance packages. The inspectors verified that the surveillance tests were performed in accordance with Technical Specifications, approved procedures, and NRC regulations.

The surveillance testing activities inspected met the safety objectives of the surveillance testing program.

4.4 Surveillance Observations

A. LER 90-02: Missed Surveillance of a Key Fire Protection Valve Due to Procedural Deficiency

A triennial fire protection audit conducted in 1987 identified inside fire hose stations surveillances, as specified in Technical Specifications, that had not been completed. This was reported in LER 87-04. During the subsequent triennial fire protection audit, completed in early 1990, an isolation valve for the fire pump sprinkler was found missing from required surveillances. Immediate corrective actions were taken, and as a result of this LER, Vermont Yankee initiated a comprehensive review of fire protection procedures specifically focusing on identification of procedural inadequacies in vital fire protection water system surveillances. No additional discrepancies were found. On July 12, OP 4020, Fire Protection Equipment Surveillances, was canceled and replaced by separate functional procedures. These new procedures more accurately reflect department responsibility for performing fire protection equipment surveillances. This LER is closed.

B. LER 90-06, Rev. 1: Technical Specification Requirement Missed Due to a Failure to Include Technical Specification Basis in Tracking List

This LER reported missed surveillance calibrations on three advanced off-gas system flow instruments. The interval for surveillance of these instruments is included on the master surveillance list, however, the requirement to perform this surveillance was not identified as being TS. Department management has scheduling flexibility for administrative surveillances, but not for TS required surveillances. The inspector reviewed root cause analysis and corrective actions associated with this LER and found them to be appropriate. As a result of this event, the master surveillance list was updated to reflect specific instrument numbers and the TS requirements. No other procedures were changed as a result of this event. The program for implementation of technical specification changes currently requires the Technical Services Superintendent to verify that plant documentation is revised to reflect the change prior to issuance. This level of review was not in effect at the time of this event. This LER is closed.

C. Core Verification

Core verification is the method used to verify the location and proper loading of fuel assemblies in the reactor core. Core verification in accordance with OP 1411, Rev. 10, "Core Verification," was completed on September 27 for reload cycle 15. The fuel cell associated with control rod 30-19 was voided on September 28 to support control rod drive changeout. The fuel loading for this cell was subsequently verified on September 29. The procedure requires two independent verifications of proper fuel configuration and specifically requires the following to be determined: fuel serial numbers agree with specified fuel loading, fuel assemblies are properly oriented, channel fasteners appear intact, and fuel bundles appear free of debris.

During the first independent verification, two new fuel bundles were found to be misloaded. Bundle LYV 673 was found in reactor location 09-20 and LYV 676 was found in reactor location 25-18. The two bundles were transposed. There was no change in cold shutdown margin because the two bundles are identical in mechanical design and enrichment loading. In addition, a bent channel fastener on bundle LYV 010 in reactor location 23-36 was detected.

The two misloaded bundles were interchanged and the bent channel fastener was replaced. The inspector reviewed videotapes of core loading and assessed the adequacy of the second verification. From the videotape the inspector determined that all fuel assemblies in the core were properly oriented, channel fasteners appeared intact and fuel bundles appeared free of debris. The inspector randomly verified that fuel serial numbers agreed with specified loading. The inspector verified that the two misloaded bundles were properly repositioned and that the bent channel fastener was repaired. The inspector also determined that the person performing the second independent verification was qualified. The second verification was meticulously performed; however, the inspector noted the cycle load diagram used to verify the core was not a controlled document. The inspector questioned this practice and the method used to ensure that the persons performing the verification used a document which accurately reflected core loading. The cycle load diagram used during the second verification accurately described the core loading. No discrepancies were noted during the second verification. The inspector found this verification thorough and comprehensive.

Circumstances leading to the original misplacement of the fuel bundles have been evaluated by Vermont Yankee. The two bundles arrived on-site in the same fuel box, were inspected in the same period, and most likely were transposed during movement to the spent fuel pool storage racks. Development of long term corrective actions to preclude initial bundle misloading is appropriate. The inspector concluded that, despite the initial bundle misload, adequate controls are in place to ensure proper core loading and verification.

D. <u>Control Rod Friction Testing</u>

On September 27 the inspector observed friction testing for control rods 18-11 and 14-23 at the hydraulic control units (HCU), and on September 28 the inspector observed friction testing for control rods 34-15 and 26-11 at the control room panel. At the conclusion of control rod friction testing the inspector reviewed oscilloscope traces for all control rods. As noted on the traces, friction testing during continuous rod insertion should not exceed 15 psid in the range of control rod positions between 48 and 03. The testing was conducted in accordance with OP 4111, Rev. 22, "Control Rod Drive Systems Surveillance."

Friction testing is designed to check proper drive operation and may also be used as a drive troubleshooting aid. During friction testing, subcritical checks, coupling verification, and control rod functional tests are performed.

Prior to withdrawal of each individual control rod and with the mode switch in refuel, a two rod interlock functional test is required to be performed. The inspector observed this test for control rod 26-11. The inspector noted that the operator experienced difficulty in moving the rod from its 00 position. The operator sequentially raised control rod driving pressure to approximately 320 pounds before the control rod moved from the 00 position to the 04 position. The two rod interlock functional test requires the rod to be withdrawn to the 02. The operator drove the rod to the 02 position and successfully completed the two rod interlock test. The inspector questioned senior operations personnel to determine the operational switch requirements which resulted in overshooting the 02 position. Specifically, the inspector questioned why the operator was in "notch override," a position usually associated with continuous rod withdraw, not single notch withdraw.

Three weaknesses were noted during inspection activities. The first two are procedural weakness. During the observed venting operations, control rod 26-11 was fully withdrawn and vented for 15 seconds. Additional venting was necessary, but was not adequately addressed by the procedure. Second, if additional venting is required, the procedure does not adequately address the process. If additional venting is required and continued rod withdrawal is appropriate, then the procedural requirement to verify the reactor is subcritical should occur during the first full-out rod withdrawal.

The third weakness involved timely documentation and system restoration. The inspector observed that subcriticality was not documented upon verification and that the CRD accumulators were not immediately returned to service at the conclusion of individual control rod friction testing (the 113 valve was not opened prior to proceeding to the next CRD accumulator). The valve positions were tracked and all accumulators were returned to service.

The inspector determined that these weaknesses were of minor safety significance. The inspector found operations personnel and reactor and computer engineering personnel knowledgeable and well prepared, the proper procedural revision in use, and nteworthy coordination between these two departments as they performed this procedure.

E. LER 90-10: Failure to Meet Technical Specifications for Diesel Generator Operational Readiness Test

On August 16, 1990, with the reactor operating at approximately 89 percent of rated thermal power, it was identified that the required monthly operational readiness tests for the A and B emergency diesel generators (EDGs) had not been performed in accordance with Technical Specification (TS) Section 4.10.A.1a. This section states, in part, that the diesel will be tested at expected maximum emergency loading not to exceed the continuous rating. The expected maximum emergency load used in the surveillance procedure, OP 4126, Diesel Generators Surveillance, was less than the true maximum emergency load based upon the value stated in the Final Safety Analysis Report (FSAR).

The impetus for resolving the maximum emergency loading TS surveillance issue was provided by an on-site NRC Safety System Functional Inspection (SSFI) team. On April 9, 1990 a Yankee Nuclear Services Division (YNSD) calculation was approved revising the expected load on the diesel to a worse case of 2751.2 kw (at a 0.85 power factor for motors over 50 horsepower) for the EDGs. The change to the FSAR was still pending. Surveillance procedure OP 4126 was not identified as requiring revision. As a result of questions from NRC SSFI team members, the licensee expedited the procedural revision to reflect this maximum emergency loading in surveillance procedure OP 4126.

The FSAR provided a value of 2467.3 kw for the maximum emergency load for the diesel generators. The values in the FSAR are a summation of the kw ratings of the loads; not considering power factor. The value specified in the testing procedure was between 2500-2750 kw. After seeking additional engineering analysis from YNSD, the licensee revised the EDG surveillance procedure to incorporate the value of 3200 kw (at a 1.0 power factor) for the EDG operational testing. The licensee considered this value electrically equivalent to the calculated maximum emergency load of 2751.2 kw at a power factor of 0.85. After Plant Operations Review Committee (PORC) review and approval of the revised procedure, surveillance testing on both EDGs was completed. The A EDG was tested twice at the new maximum emergency loading. The B EDG was tested three times utilizing the new maximum emergency loading value. The EDGs were considered operable during resolution of this loading issue.

Further evaluation by YNSD concluded that the surveillance test required a minimum of 2751.2 kw and 3175 kVA. The conclusion effectively decoupled power factor from surveillance consideration and recommended that the EDG continuous rating of 3000 kw not be exceeded. The licensee requested guidance and clarification from YNSD and from the vendor, Fairbanks Morse. In correspondence dated September 12, 1990, the vendor responded by stating that the maximum load that the Vermont Yankee EDG can be run for one hour without adversely impacting the standard maintenance interval is 3025 kw. While approximate electrical equivalency was achieved using 3200 kw at unity power factor, the mechanical load of the EDG (real power in kw) was exceeded.

The licensee subsequently conducted maintenance inspections of the diesel generators. The inspection on the A EDG identified a crack on the No. 11 cylinder piston insert. The inspection on the B EDG revealed that one upper pin floating bushing was undersized. These discrepancies were corrected. These deficiencies did not affect the operability of the EDGs and appeared unrelated to the overload events. The surveillance procedure was also revised to perform the operability test at 2650-2750 kw for the first hour, and the remaining seven hours at 2500-2700 kw.

The NRC SSFI Team Report No. 50-271/90-80 further discusses this event.

The inspector expressed concern about several items. The first item is the interface between engineering support activities and plant operation activities. The amount of time required to incorporate the YNSD engineering analysis concerning FSAR maximum expected EDG loading in plant procedures appears excessive and the absence of OP 4126 from the list of procedures which ultimately require revision as a result of this engineering analysis requires further licensee investigation. Secondly, the adequacy of information received from the vendor during initial contact concerning this issue appeared to be incomplete. Finally, the technical review failed to identify overload concerns prior to procedural implementation.

Pending results of further licensee and NRC inspector evaluation of this event, LER 90-10 remains open.

5. SECURITY (71707, 93702)

5.1 Observations of Physical Security

Implementation of the security program was verified on a periodic basis, including the adequacy of staffing, entry control, alarm stations, and physical boundaries.

5.2 Fitness-for-Duty and Access Clearance Issues

A. On October 2, as part of VY's Fitness-for-Duty Program, three dogs specially trained to detect certain illegal drugs, were brought within the protected area to conduct a random, unannounced search. The dogs were handled and provided by the Vermont State Police. No illegal drugs or substances were found during the search.

B. On August 23, 1990, the licensee informed the inspectors about three incidents concerning access clearance and Fitness-for-Duty requirements. The first incident involved a union business agent signing a letter indicating that two individuals were members of the union for three years or more when, in fact, they had not been. One of these individuals was employed as a contractor at the site and had an outstanding arrest warrant for a probation violation for a previous felony conviction. This individual was arrested at the site by a local low enforcement agency on August 22, 1990.

The second incident involved two individuals who did not have fully completed background investigations in accordance with site access procedures. This incident reflected the failure of the licensee to properly scrutinize contractor provided information. The third incident involved the licensee being advised of a badged contractor employee who had terminated from previous employment for alcohol abuse. The licensee determined that this incident resulted from the failure of their contractor to conform with the VY Policy for Fitness-for-Duty.

These issues were the subject of a special NRC Region I Security Inspector, which is documented in Inspection Report 50-271/90-11.

6. ENGINEERING AND TECHNICAL SUPPORT (92700, 90712, 71707, 93702)

6.1 LER 87-15, Rev. 1: Reactor Scram Due to Transient in Turbine Control Oil System

The inspector reviewed LER 87-15, Rev. 1. This revision provided an update on Vermont Yankee's evaluation of a reactor scram due to transient flow imbalance in the turbine control oil system. The event was discussed in inspection report 50-271/87-16, Section 9.5.

A comprehensive review of the turbine startup procedure was performed by General Electric. As a result of this review, a new acceptable range on the observed bearing header oil pressure was established. The new pressure range was established at 30 psig $\pm$ 2 psig. The inspector determined that further evaluation of this event is warranted. This determination is based on the information contained in LER 90-04, which provides information on a recent, similar event.

6.2 LER 89-05, Rev. 1: Inadvertent Primary Containment Isolation System Activation Due to Inadequate Procedure

Details of the event were discussed in inspection report 50-271/89-02, section 6.4. This revision was submitted to reflect modifications to corrective actions contained in the original submittal. Specifically, more time was required to evaluate technical specification 3.2.B and plant design bases to determined if bypassing the refuel floor radiation monitors during dryer movement is appropriate. In this LER the licensee committed to complete this evaluation and any subsequent procedure changes before the February 1992 refueling outage. Based on inspector observations during the 1990 refueling outage, previous corrective actions resulting from LER 89-05 were effective. This LER is closed.

6.3 Fuel Failures

Vermont Yankee experienced fuel failures during the past operating cycle (Cycle XIV). The fuel failures were initially identified by an increase in Steam Jet Air Ejector (SJAE) off-gas activity levels. The increase in off-gas activity levels continued to rise throughout the operating cycle and peaked at approximately 60,000 uCi/second by the end of the operating cycle. The licensee developed the "Fuel Performance Monitoring Guidelines and Failed Fuel Action Plan" to address concerns with rising off-gas levels. Licensee efforts to deal with failed fuel were effective. More detail with regard to licensee performance and NRC assessment in this area is contained in NRC Inspection Reports 50-271/89-09 and 50-271/90-01.

Using data collected during the operating cycle, Vermont Yankee was able to predict the approximate number and locations of the failed fuel assemblies in the reactor. Initial vacuum sipping of fuel bundles from the suspected areas identified four leaking fuel assemblies. Three of these failed fuel assemblies (LYC 210, LYC 170, LYC 202) had been in the reactor core for three operating cycles and were scheduled for discharge to spent fuel pool. The fourth assembly (LYN 777) was new when installed during the February 1989 refueling outage.

A fuel rod in the newer assembly, LYN 777, experienced the most significant failure. Cracking and deterioration resulted in the release of approximately 4 inches of the fuel column from this fuel rod. At the top end of this rod, in the vicinity of the spring/end plug assembly, weld failure was evident.

During the second vacuum sipping period, the licensee sipped the remaining first and second cycle fuel bundles. This conservative action resulted in the identification of a fifth fuel assembly with leaking fuel (LYJ 040). The failure in this second cycle bundle was induced by fretting. The fretting resulted from a small piece of metal wire which had accidently fallen from the refuel floor during the previous cycle refueling activities. This fretting was not predicted by the licensee's lost part analysis.

The licensee requested General Electric Company, the fuel vendor, to provide a complete detailed analysis of the fuel failures. In a response dated October 5, 1990, General Electric responded to the licensee's request and provided documents which evaluated Cycle XIV fuel failures.

The inspector reviewed information contained in these documents and concluded that the information adequately addressed several concerns. The information postulated a "most probable" failure scenario for the failed fuel rod in fuel bundle LYN 777. The failure was most likely the result of a manufacturing defect. Moreover, most of the fuel material released from the failed fuel was deposited on the residing fuel assemblies. The remaining fuel material is most likely trapped in stagnation areas in the primary system or has been removed by the reactor water cleanup system.

The licensee continued to demonstrate strong technical competence and conservative safety attitude in response to failed fuel. As part of its efforts to improve fuel performance, the licensee placed four American Nuclear Fuel manufactured assemblies in the core. The trial performance of these lead test assemblies is planned to be closely monitored throughout the core operating cycle and compared with previous fuel performance.

The inspector concluded that management response to failed fuel was excellent. Vermont Yankee's actions in response to the failed fuel were well executed and appropriate. Management efforts to keep plant personnel informed of the fuel failures and of the sipping results aptly addressed individual safety concerns.

7. SAFETY ASSESSMENT AND QUALITY VERIFICATION (35501, 71707, 90713, 92701)

7.1 Review of NUREG-0737 Commitments

These items have been broken down into numbered descriptions (Enclosure 1 to NUREG-0737, "Clarification of TMI Action Plan Items"). Licensee letters containing commitments to the NRC were used as the basis for acceptability, along with the NRC clarification letters. The following item was reviewed.

A. Item II.F.2 Instrumentation for Detection of Inadequate Core Cooling

This item was last reviewed during inspection 86-22. The inspection report noted the licensee's commitments of December 6, 1984 and March 26, 1985 to replace the existing reactor vessel level measurement system (RVLMS) with one that conforms to Item II.F.2. The design of the new system was approved in an NRC letter dated May 24, 1985. The licensee requested and received a deferral of installation of the new system until the 1987 outage. The licensee's letter of July 18, 1985 requested the deferral, and it was accepted by the NRC in a letter dated September 6, 1985. These actions closed out all of Item II.F.2 except Requirement (4) - Installation of Additional Instrumentation.

The inspector reviewed licensee actions concerning installation of additional instrumentation, including the design change, as-built drawings, and procedures used. Based on this review, the inspector verified that the modification of the RVLMS, implemented in 1987, meets licensee commitments and NRC requirements. In addition, it was verified that the modifications were properly approved and controlled, that the procedures were revised to reflect the new design and that personnel were trained on the new system. No Technical Specification changes were needed. The preoperational testing was completed, and the system was calibrated and declared operable in 1987. The licensee has satisfied all the requirements for NUREG-0737; Item II.F.2, and this matter is closed.

7.2 (Closed) Unresolved Item 50-271/86-22-04: Review Licensee's Action to Establish Reliability of the ATWS RPT System Consistent with Generic Letter 85-06

Generic Letter (GL) 85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety Related," was issued to all power reactor licensees on April 16, 1985 in order to provide fundamental QA criteria applicable to such equipment. This unresolved item was identified during NRC inspection 86-22 after the licensee had responded to the GL. The inspectors had determined that the licensee did not apparently fully verify compliance with the QA guidelines promulgated in the letter for the anticipated transient without scram-alternate rod insertion/recirculation pump trip (ATWS-ARI/RPT) equipment at VY. All ATWS-ARI/RPT equipment at VY is safety related by design except the General Electric Type AK (AKF-2-25) field circuit breakers and the shunt trip coils for the recirculation pump MG sets.

10 CFR 50.62 requires that the ATWS equipment be designed to perform its function in a reliable manner. Although the licensee demonstrated (letter FVY-85-93, September 29, 1985) that these requirements had been satisfied for the ATWS RPT equipment, inspection 86-22 noted that the licensee had not "established the reliability" of the system with respect to the installed MG field breakers, and further noted that their reliability was still questionable. Due to previous multiple failures of AK type breakers at Vermont Yankee and a recent AK breaker failure on the main generator the inspectors determined that the reliability of the AKF-2-25 MG field breakers should be established. This was considered necessary despite newly implemented preventive maintenance (PM) requirements which appeared to have had a positive effect in reducing the number of AK breaker failures in the plant. Although GL 85-06 did not require additional reporting under the guidance provided, it did indicate that licensee's QA organizations were expected to verify compliance with the guidance provided in the letter.

During inspection 86-22, plant management stated that they had not yet compared the QA existing controls for the breakers to those contained in Generic Letter 85-06, although no special reliability measures were being applied to the recirculation pump MG set field breakers. The licensee subsequently issued letter FVY 87-41 (April 10, 1987) stating that the ARI equipment at VY was installed as Class 1E and that it therefore falls under the Operational Quality Assurance Program for safety class electrical equipment (YOQAP-1-A) which is "based upon" 10 CFR 50, Appendix B criteria, and ANSI 18.7-1976. The issue remained open because the licensee did not establish the reliability of the installed breakers and because no comparison with the GL guidelines had been documented.

On June 6, the licensee responded by written memorandum which specifically addressed all QA guidance outlined in the GL and provided information on AK type breaker failures over the life of the plant. It specifically noted the improvement in breaker performance after improved preventive maintenance requirements had been imposed upon these breakers in 1982. The memorandum also reported performance test results and generally addressed reliability-related QA criteria currently applied to these breakers. The description of QA activities in response to specific GL guidance focused around the performance of plant procedure OP 5221, "480 Volt AC Circuit Breakers Inspection, Calibration, and Testing." The procedure provides instructions for performing maintenance on all safety related and non-safety related General Electric Type AK switchgear and field breakers in the plant. This procedure is performed every refueling outage on the MG set AK field breakers. It directs critical QA actions to be performed during inspection, maintenance, and testing of the breakers to identify nonconformances, to assure breaker operability, to assure replacement parts meet ATWS guidelines, to document the level of quality applied to breaker repairs, and to provide periodic requalification of breaker quality and operability. The inspector reviewed OP 5221 and concluded that all QA actions directed by this procedure for safety related breakers also applied to, and have been performed on the non-safety MG set field breakers. The June 6 memorandum also described specific QA organization activities such as audits and inspections of maintenance department practices, receipt inspection bench testing of replacement parts, the application of identical controls to the refurbishment of all safety and non-safety AK breakers, and the application of uniform controls over all documentation associated with these breakers. In addition, NRC inspection report 89-05, which reviewed the licensee's conformance to the ATWS Rule 10 CFR 50.62, noted that a review of surveillance test data documented between 1988 and 1989 demonstrated that the ATWS system was functional.

Based upon the above, the inspector concluded that the licensee's actions adequately address NRC concerns in this area. The licensee's response to this item was determined to be thorough and complete, and demonstrated that they are fully applying the level of quality verification prescribed by the GL to the non-safety related ATWS equipment. This item is closed.

7.3 Evaluation of Licensee Quality Assurance Program Implementation

The inspectors evaluated the effectiveness of the licensee's quality assurance program. Quality assurance is defined in ANSI N45.2.10-1973 as "...all those planned and systematic actions necessary to provide adequate confidence that an item or a facility will perform satisfactory in service." This definition includes those who achieve quality (managers, supervisors, and workers) and those who verify that quality was achieved (QA organization, peer inspection).

The inspectors reviewed operational data, NRC inspection reports, previous NRC SALP reports, Licensee Event Reports for the past twelve months, licensee corrective actions for NRC inspection findings, outstanding unresolved items, and Quality Assurance Department (QAD)/Quality Services Group (QSG) audit/surveillance reports. In addition, the inspectors met with QAD managers on August 29 at the Yankee Atomic Energy Company Headquarters in Bolton, Massachusetts.

An in-office review of the above specified documents revealed no programmatic or repetitive weaknesses in identifying the root cause of failures or for providing proper corrective actions. The inspectors noted one area that has required additional licensee attention: missed surveillances. Corrective actions to preclude missing TS required surveillances will continue to be evaluated by the NRC

Based on discussions with the QAD managers, the inspectors concluded that the QA Program is dynamic, providing timely internal self-assessment and periodic upgrades in anticipation of changing NRC and industry expectations. The QAD aggressively seeks opportunities to expand the program perspective by exchanging technical specialists with utilities and outside organizations. The QAD has demonstrated the ability to perform thorough performance based audits.

The inspectors concluded that the on-site QSG is adequately staffed, competently managed, and able to meet its QA function. Vermont Yankee management maintained an active interest in the audit process, providing adequate personnel resources and ensuring timely disposition of audit discrepancies. The inspectors concluded that the Vermont Yankee QA Program was in general effectively implemented.

7.4 LERs

The inspector reviewed the licensee event reports listed below to determine that with respect to the general aspects of the events: (1) the report was submitted in a timely manner; (2) description of the events was accurate; (3) root cause analysis was performed; (4) safety implications were considered; and (5) corrective actions implemented or planned were sufficient to preclude recurrence of a similar event.

LER 87-15 (Rev. 1) March 7, 1990, Reactor Scram Due to Transients in Turbine Control Oil System (Section 6.1)

LER 89-05 (Rev. 1) September 21, 1990, Inadvertent Primary Containment Isolation System Actuation Due to Inadequate Procedure (Section 6.2)

LER 90-02	March 22, 1990, Missed Surveillance of a Key Fire Protection Valve Due to Procedural Deficiency (Section 4.4.A)
LER 90-05	April 18, 1990, Incomplete Evaluation of Containment Air Sample (Section 3.2.A)
LER 90-06 (Rev. 1)	May 3, 1990, Technical Specification Requirement Missed Due to a Failure to Include Technical Specification Basis in Tracking List (Section 4.4.B)
LER 90-10	September 14, 1990, Failure to Meet Technical Specifications for Diesel Generator Operational Readiness Test (Section 4.4.E)
LER 90-11	October 1, 1990, Full Reactor Protection System Actuation from Spike in a Shared LPRM (Section 2.2.A)

7.5 Periodic and Special Reports

The licensee submitted the following periodic and special reports which were reviewed for accuracy and the adequacy of the evaluation.

-- Monthly Statistical Report for plant operations for July and August 1990.

-- Feedwater leakage detection system monthly performance summary for July and August 1990.

7.6 Open Item Followup

The following previous inspection items were followed up during this inspection and are listed below for cross reference purposes.

90-02-01, Section 3.2.A.
86-22-04, Section 8.2

8. UNRESOLVED ITEMS

Unresolved items are items about which more information is required to ascertain whether they are acceptable, violations or deviations. Unresolved Items are discussed in Section 4.2.A (UNR 90-10-02), 4.2.B (UNR 90-10-03).

9. MANAGEMENT MEETINGS (30703)

9.1 Preliminary Inspection Findings

A summary of preliminary findings was provided to the Plant Manager at the conclusion of the inspection. During the inspection, licensee management was periodically notified of the preliminary findings by the resident inspectors. No written inspection material was provided to the licensee during the inspection. No proprietary information is included in this report.

9.2 Management Meetings Conducted by Region Based Inspectors

Date	Subject	Inspector Report No.	Reporting Inspector
8/6-17/90	Safety System Functional Inspection	90-80	S. Chaudhary
8/29-30/90	Special Security Inspection	90-11	G. Smith
10/1-5/90	MTI Follow-up	90-12	P. Drysdale
10/1-5/90	Radiation Protection	90-13	P. O'Connell
10/1-5/90	Allegation Follow-up	90-14	S. Chaudhary

9.3 Visit by NRC Commissioner

On September 19, 1990, NRC Commissioner James Curtiss and a technical assistant conducted a visit to the VYNPS. A meeting was held with the resident inspectors, and a site tour was conducted by the Plant Manager. Following the tour, discussions were held with corporate and site management on NRC and VYNPS issues. These activities also involved the Director, Division of Reactor Safety, who was representing the NRC Region I Office.

ATTACHMENT A

Inspection Report 50-271/90-10

INSPECTION FOLLOW-UP FOR PORTIONS OF ALLEGATION RI-90-A-100

TABLE OF CONTENTS

1. PURPOSE AND SCOPE OF INSPECTION

This documents inspection activities conducted at the Vermont Yankee Nuclear Power Station (VYNPS) as a result of allegations raised about the manner in which the plant was being operated. Information relevant to these concerns was provided to the NRC, and the licensee, by the Department of Public Service of the State of Vermont. The specific areas reviewed by the inspectors were: (1) the adequacy of the licensee's minimum shift staffing requirements to accomplish a safe shutdown of the plant during a fire emergency; and (2) the manner in which the plant's spent fuel pool (SFP) cooling equipment was operated and the ability of this equipment to perform its intended design functions.

Information used by the inspectors to obtain an understanding of the issues identified above, and to allow appropriate conclusions to be drawn, was obtained by reviewing licensee and NRC documents and conducting interviews with various licensee personnel. In addition, information and reviews, as appropriate, were provided by the NRC:NRR Project Manager and NRC:RI inspection personnel. Although the licensee conducted its own investigation of the matters discussed in this report, the NRC inspection effort and results were independent of the licensee's activities. The licensee's investigation report was made available to the inspectors.

2. SUMMARY

The inspectors reviewed the adequacy of the licensee's minimum shift staffing requirements to accomplish a safe shutdown of the plant during a fire emergency. No undermanned conditions were identified, and the licensee staffing levels are consistent with the capability to provide for the safe shutdown of the plant during a fire emergency.

The inspections associated with the manner in which the plant's SFP cooling equipment was operated and equipment was repaired, between the period of June 9, 1989 and July 27, 1990 identified a number of deficiencies involving the conduct of licensee activities. These included: (1) a violation involving the failure to follow the procedural requirements for operation of the SFP cooling system; (2) a deviation from licensee written commitments involving dispositioning of inoperable EQ equipment; and (3) a violation of 10 CFR 50, Appendix B and 10 CFR 50.49 requirements due to a failure to properly identify and assess degradation of safety-related equipment from an EQ program perspective. These deficiencies appear to be attributable to a number of weaknesses involving: (1) the lack of safety committee reviews to assess the impact of degraded equipment conditions and planned corrective actions on facility design features and licensee commitments; and (2) a less than adequate understanding of the licensee's established EQ program as it pertains to addressing inoperable safety class equipment.

3. REVIEW OF CONCERN

3.1 Adequacy of Minimum Shift Manning

A. Statement of Concern:

The licensee has undermanned operating crews, such that, they could not perform a safe shutdown of the plant during a fire emergency as required by Technical Specifications (TS).

B. Discussion of Issue and Assessment

The organizational requirements of Vermont Yankee (VY) are stipulated in the TS, the Security Plan, and station procedures. The current TS requirements for minimum shift manning are in excess of the requirements stipulated in NUREG-0737, Item I.A.1.3, Shift Manning, due to its specifying that a Shift Engineer is included in the minimum shift staffing. Table I represents a correlation of the various TS and staffing requirements that currently exist. The station procedures adequately reflect both TSs and 10 CFR 50, Appendix R, Section III.H requirements for the Fire Brigade. For example, the Shift Supervisor is not a member of the fire brigade, and the brigade staffing does not reduce licensed operator levels within the control room below TS requirements while at the same time providing for at least five members on each shift to be available for the Fire Brigade.

From a minimum staffing condition, the most limiting task would be the implementation of alternate shutdown capability at the same time that a fire emergency is declared. The emergency declaration would result in manning the Fire Brigade.

In February 1988 the NRC reviewed the licensee's compliance with 10 CFR 50, Appendix R. This review was documented in Inspection Report 50-271/88-04, which included an examination of the licensee's capability to achieve and maintain hot shutdown and the capability to bring the plant to cold shutdown conditions in the event of a fire in various areas of the plant. Station Procedure OP 3126, Shutdown Using Alternate Shutdown Methods, was reviewed to ascertain that the shutdown could be attained in a safe and orderly manner. No unacceptable conditions were identified. The review included a walk-through of selected portions of the procedure to determine by simulation that shutdown from outside the control room could be attained in an orderly and timely fashion. The procedure walk-through was accomplished by four members of the licensee's operations staff. The NRC review determined that the licensee did not have a time-line analysis to verify that all procedure OP 3126 requirements could be implemented with the minimum manpower available. At the time of the inspection, the licensee committed to perform the time-line check. In response to the inspector's questions on the status of this item, the licensee's staff provided an Operations Department Memorandum dated August 24, 1988. This memorandum provided the required time-line analysis of procedure OP 3126 using the available manpower associated with a declared fire emergency. The inspector determined that the licensee's analysis was responsive to the NRC's concerns.

The inspector determined that the licensee's minimum shift staffing requirements are consistent with the plant's licensing basis. The inspector identified no unacceptable conditions and based upon prior NRC inspections in this area, concluded that there is a proper level of assurance tha[illegible]censee has the capability to provide for the safe shutdown of the pl[illegible]g a fire emergency.

3.2 Spent Fuel Pool (SFP) Cooling System Operations

A. Statement of Concern:

The SFP cooling system relied upon defective equipment for an extended period of time. This condition was contrary to operability requirements for power operation of the plant. Because of an electrical fault on a SFP cooling pump the vital emergency core cooling system was threatened. Concern was expressed about the ability to cool the SFP if the reactor building became uninhabitable, because there would be a dependence on only one pump and power supply. A question was raised about intentional oversight in not repairing the pump motor because the licensee was to install a new SFP cooling system.

B. System Description

The SPF cooling and demineralizer system cools the fuel storage pool by transferring the spent fuel decay heat through a heat exchanger to the reactor building closed cooling water system. Water purity and clarity in the storage pool are maintained by filtering and demineralizing the pool water through a filter-demineralizer.

The system consists of two circulating SFP cooling pumps connected in parallel, two heat exchangers, two filter-demineralizers, and the required piping, valves and instrumentation. Each pump has a design capacity equal to the system design flow rate and is capable of simultaneous operation. Two filter-demineralizers are provided, each with a design capacity equal to the design flow rate. The pumps circulate the pool water in a closed loop, taking suction from the spent fuel storage pool, circulating the water through the heat exchangers and filters, and discharging it through diffusers at the bottom of the fuel pool and reactor well.

The SFP pumps and heat exchangers are located in the reactor building below the bottom of the fuel pool. The SFP filters, which collect radioactive corrosion products, are located in the radwaste building.

The SFP is filled and make-up is supplied from the condensate transfer system. Water is removed from the SFP via the fuel pool pumps through the filter-demineralizer units to the condensate storage tank.

The operating temperature of the SFP is permitted to rise approximately 25 degrees F above the normal operating temperature (125 degrees F) when circulation flow is temporarily interrupted or when larger than normal batches of fuel are stored. The heat exchangers in the residual heat removal system can be used in conjunction with the fuel pool cooling and demineralizer system to supplement pool cooling in the event that a larger than normal amount of fuel is stored in the pool.

The system instrumentation is provided for both automatic and remote manual operations. Instrumentation and controls are provided to detect, control and record pump operation, pool temperature and system flow. A pool leak detection system has been provided to monitor leakage and thus indicate pool integrity.

The pumps are controlled locally in the reactor building or at Panel 20-22 in the Radwaste Building control room. Pump low suction pressure automatically turns off the pumps. A pump low discharge pressure alarm indicates in the main control room and in the pump room.

The safety objective of the fuel pool cooling and demineralizer system as stated in Section 10.5 of the FSAR, is to maintain fuel pool water temperature at a level which will prevent damage to the fuel elements, and to maintain the Reactor Building environment at a level which will bound the qualification of electrical equipment.

C. <u>SFP Licensing Issues and Commitments</u>

On April 25, 1986, Vermont Yankee submitted its Proposed Change (PC) No. 133 amendment request to the NRC to allow the expansion of the capacity of the SFP and the increased storage of spent fuel in the pool.

If granted, the amendment to the facility operating license would authorize the licensee to increase the capacity from the current 2000 fuel assemblies to the proposed capacity of 2870 fuel assemblies in the pool. In the licensee's letter (FVY 87-65) to the NRC on June 11, 1987, they responded to (PC No. 133 related) NRC staff questions regarding operational controls associated with the SFP cooling system by committing to administratively implement proposed controls by startup from the 1987 refuel outage. These administrative controls were presented in the form of proposed limiting conditions for operation (LCO) and associated surveillance requirements. The licensee further committed that these proposed controls would be submitted to the NRC for approval as a separate TS amendment request.

Subsequently, on September 1, 1987, the licensee submitted its letter FVY 87-87 to the NRC that provided a summary of the administrative controls that they would procedurally implement prior to startup from the 1987 refuel outage. The administrative controls were intended by the licensee to provide assurance that adequate cooling was available for heat removal in the SFP by providing, in part, for fuel pool cooling equipment operability constraints and SFP and related equipment surveillances. However, based upon discussion with the NRC staff about VY's SFP expansion reports, the licensee determined that incorporation of the operational controls within the TSs was not necessary and therefore an amendment request would not be submitted.

On February 9, 1988, a public meeting was held between the NRC staff and VY to consider information needed to complete the staff's review of PC No. 133. In order to expedite the NRC staff review of the subject license amendment request, VY committed to design, install, test and make operational, a redundant seismically designed SFP cooling system prior to the time that they exceed the existing 2000 spent fuel assembly storage limit in the SFP. The licensee's letter FVY 88-17, which was submitted to the NRC on March 2, 1988, documented and expanded upon the information presented at the public meeting. In addition, each of the remaining open technical issues was addressed. Specifically, the licensee addressed the single failure issue by stating that VY is single active failure proof with one SFP cooling pump in standby and one pump operating with two heat exchangers operating in parallel.

As a result of the above licensing issues and commitments, the VY Manager of Operations (MOO) issued on March 3, 1987, MOO Directive 87-1. This directive required the VY Plant Manager to administratively implement the conditions specified in their letter FVY 87-87 to the NRC. Procedure OP 2184, Fuel Pool Cooling System; OP 4341, Fuel Pool Level Switch Calibration; and OP 0150, Responsibilities and Authorities of Operations Department Personnel, were revised accordingly.

The NRC issued Amendment No. 104 to the Facility Operating License on May 20, 1988. This amendment allowed for the re-racking of the SFP to accommodate 2870 assemblies. However, the present TS limit of 2000 assemblies in the pool was not changed. Consideration of storage of more than 2000 assemblies was determined by the NRC to await a determination of the adequacy of SFP cooling for more than 2000 assemblies, including the yet to be designed enhanced SFP cooling system.

According to the licensee, the commitment to design and install a new enhanced SFP cooling system of and by itself addressed all of the NRC staff concerns on PC No. 133. But, because the VY letter FVY 87-87 was incorporated by reference in Amendment No. 104, it was judged by the licensee not to be worth the effort and possible additional complexity to attempt removal of the administrative controls. The licensee has indicated that the controls will remain until after the new enhanced fuel pool cooling system is installed.

D. <u>SFP Pump A Operability</u>

An intermittent ground on the "A" SFP cooling pump motor was detected on June 9, 1989. The ground, originally thought to be on the standby liquid control (SLC) system's tank heater, appeared only after the SFP pump motor had been operating for several minutes. Additional investigation and data collection led VY personnel to the conclusion that the pump motor should be replaced. The decision to replace the pump motor vice rewinding the motor was based on a derived safety benefit from having a pump, albeit with a phase ground, in place in the unlikely event that the redundant "B" SFP cooling pump failed. On July 5, 1989 the breaker for "A" SFP cooling pump was white tagged out-of-service due to the motor ground.

Between July 5, 1989 and July 3, 1990 the "A" SFP cooling pump was white tagged out-of-service. A Maintenance Request (MR 89-2291) remained active during this period, but no maintenance was performed. Because the licensee thought that the motor would function if the white tag was removed and the breaker was closed, VY management considered the pump operable.

Following receipt inspection and disposition of dimensional deviations a new motor was installed by July 27, 1990. The "A" SFP cooling pump motor power supply breaker was white tagged open and the pump was declared inoperable during the new motor replacement.

The inspector concluded that the "A" SFP cooling pump was inoperable between July 5, 1989 and July 3, 1990. Station Procedure AP 0140, Vermont Yankee Local Control Switching Rules, states that white tags (Danger Tags) provide visual indication that operation is not allowed for the protection of personnel or equipment or necessary to maintain system integrity. Furthermore, AP 0140 Appendix B, Miscellaneous Switching and Tagging Rules, states that any component which is white tagged shall not be operated under any circumstances. Moreover, upon completion of an active corrective maintenance request the shift supervisor shall perform specified post maintenance testing (PMT), and based upon results of the PMT declare the equipment operable and close out the corrective MR. This information is contained in AP 0021, Rev. 17, Maintenance Requests. Based upon the previously discussed procedural guidance, the inspector concluded that equipment or components positioned and white tagged to prevent operation shall be considered inoperable. A white tag used to administratively restrict operation of a component or equipment renders that equipment or component inoperable. In some instances, where white tags are used only as a higher level of equipment control, the equipment may be made operable by removing the white tag and repositioning a breaker, switch, valve, or other tagged component.

VY committed to administratively implement certain controls prior to start-up from the 1987 refueling outage, as discussed in Section 3.2.C above. The administrative controls were procedurally implemented in procedure OP 2184 and administratively implemented in Manager of Operations (MOO) Directive 87-01. One of these controls stated that from and after the date that one of the fuel pool cooling subsystems is made or found inoperable and the remaining subsystem is capable of maintaining the fuel pool temperature below 150 degrees F, then the reactor shall be in cold shutdown condition within thirty days unless such subsystem is sooner made operable. The inspector concluded that from July 5, 1989 to July 3, 1990 the "A" SFP subsystem was made inoperable and that the procedural controls of procedure OP 2184 were not implemented. This is considered a failure to follow a procedural requirement and is a violation of Technical Specification, Section 6.5 (VIO 50-271/90-10-04).

The inspector concluded that this was an isolated event. However, the evaluation of the event identified two weaknesses which require additional licensee attention. First, operators and some key supervisors were not fully aware of the administrative requirements contained in the MOO Directive 87-01 and in the fuel pool cooling system operating procedure. The MOO Directive was not readily available to operators, consequently, the decisions regarding repair of the "A" SFP cooling pump did not benefit from guidance contained in these instruction. Second, the sequence of events identified the need for PORC to review plant tagouts to detect any potential safety hazards. The licensee has identified this concern and PORC now conducts periodic reviews of plant tagouts which are active for greater than 60 days.

The inspector concluded that a procedural requirement, formally committed to the NRC, was not effectively implemented, and that management review did not adequately address the event or document the acceptability of the condition.

E. Environmental Qualification of the "A" SFP Cooling Power Motor

Engineering Design Change Request (EDCR) 83-32, Fuel Pool Cooling EQ Modifications was implemented in 1984 by the licensee in order to meet the requirements of 10 CFR 50.49 and assure that safety-related electrical equipment in the Reactor Building would not be subjected to a post-LOCA harsh environment from the SFP. The design change would assure operation of the SFP cooling system long-term, post-LOCA with loss of off-site power, and provide controls to the system when the reactor building was not accessible. The electrical portion of the SFP cooling system was reclassified as safety class, and was required to be qualified to assure post-LOCA operation when off-site power is not available. The modified SFP cooling system would be capable of operating post-LOCA, prevent the SFP from boiling, and thereby preclude creating a harsh environment in the reactor building.

Since the SFP cooling pumps are safety-related and are required to be operable to maintain the ambient environment for which other safety-related components in the reactor building are qualified, the licensee included the pump motors into the EQ master list, and qualified them in accordance with 10 CFR 50.49 requirements. Normally only one pump is required to maintain the fuel pool temperature. The second pump is required to operate when the first one fails.

On June 9, 1990, the plant Maintenance Department determined that an intermittant ground existed in the "A" SFP cooling pump motor. With the pump stopped and cold, testing originally did not indicate a ground. Subsequent testing determined that the ground appeared after about 15 to 20 minutes of operation when the motor was hot and at operating temperatures. On July 5, 1989, the power to the pump motor was de-energnized and the feeder breaker white tagged in the open position. The details of the troubleshooting, repair, and procurement efforts related to the ground condition are contained in Table 2, Sequence of Events. The motor was replaced with a new motor on July 27, 1990.

During the period from June 9, 1989 to July 27, 1990, the "A" SFP cooling pump motor was in a degraded condition, in that at least one phase of the motor winding was shorted to ground. The licensee did not provide evidence to demonstrate that the motor, while in the degraded condition, was qualified for the post-LOCA environment. This is in violation of 10 CFR 50, Appendix B, Criterion XVI and 10 CFR 50.49, Paragraph f, which require nonconformances promptly corrected and electrical equipment important to safety to be qualified (VIO 50-271/90-10-05).

10 CFR 50, Appendix B, Criterion XVI requires that measures shall be established to assure that conditions adverse to quality, such as defective equipment and nonconformances, to be promptly identified and corrected. Although the licensee identified the deficiency of the pump motor on June 9, 1989, corrective action was not accomplished until July 27, 1990. In addition, NRC Generic Letter 86-15 regarding information relating to compliance with 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants," was issued to the licensee on September 22, 1986. The letter clearly stated that when a licensee discovers a potential deficiency in the environmental qualification of equipment (i.e., a licensee does not have an adequate basis to establish qualification), the licensee shall make a prompt determination of operability, shall take immediate steps to establish a plan with a reasonable schedule to correct the deficiencies, and shall have written justification for continued operation.

In July 1990, after the degraded motor was replaced, the licensee generated various documents to argue that the degraded pump motor was operable during a postulated post-LOCA condition. For an ungrounded electrical power system, the pump motor can be operated even with one phase shorted to ground. However, there was no analysis available to prove that when the pump motor temperature increased during the post-LOCA condition a second phase would not short to ground, since the pump motor was in a degraded condition. The licensee already identified that the first phase of the motor winding shorted to ground when the motor warms up to normal operating temperature.

F. SFP Level Instrumentation

In 1983, the licensee identified the need to replace the existing SFP level alarm instrumentation. This was due to insufficient test documentation for the existing instruments, which would prevent the installed instrumentation from being included in the Vermont Yankee upgrade program for EQ of safety related electrical equipment. The installed instrumentation included a single high and low level alarm function (LSH 19-60 and LSL 19-60).

In their June 29, 1984, letter to the NRC, FVY 84-74, which provided information on the Vermont Yankee upgraded EQ Program, the licensee stipulated that the existing SFP level alarm instruments would be replaced with redundant class 1E instruments.

The implementation of EDCR 83-32 during the 1984 refueling outage provided for the installation of redundant safety class level instrumentation. The instrumentation was EQ and provided conformance with 10 CFR 50.49. The replacement level instrumentation consisted of two major parts: the level sensor located in the SFP and the electronics. Each sensor (of which there are two) has two sensing elements, an upper and lower, to detect high and low water levels. The low level alarm condition would occur at 25 1/2" below the top of the SFP. This alarm condition would cause the annunciation of the "Fuel Pool Cooling System Trouble" alarm in the control room and the "Low Level Fuel Storage Pool" alarm in the Radwaste Building control room. The power for the level instruments was derived from the SFP cooling pump motor power supplies that are supplied from the emergency diesel generator. Essentially, the "A" SFP cooling pump power energized the "A" level instrument channel of the redundant level instrumentation system for the SFP. It was the design intent of the EDCR for the redundant level alarms to allow the plant operators to add water to the SFP as required. The post-LOCA makeup water operation can be performed manually in the Radwaste Building.

The issuance of MOO Directive No. 84-04 on August 3, 1984 provided guidance that was intended to ensure that the licensee will remain in full compliance with the EQ Program. Accordingly, the plant staff was directed, in part, that the SFP level alarm switches shall be operable. This guidance also clearly directed and limited the timeframes for corrective action if a deviation from the requirement occurs. It was the intent of the MOO Directive to ensure that the potential for post-accident environmentally induced problems are minimized. In April, 1985, the Vermont Yankee EQ Plan superseded the MOO Directive. The licensee stated in the Plan, Section V, Operability Requirements for Environmental Equipment and Components, that "...it is the policy of Vermont Yankee's corporate management that all equipment and components which are addressed by Vermont Yankee's EQ Program shall be maintained operable and fully environmentally qualified at all times, commensurate with the status of the plant." However, administrative controls are specified for actions the licensee will follow in the event the EQ status of a component becomes uncertain. In its letter FVY 85-40, dated May 3, 1985, the licensee provided the NRC with the in place administrative controls associated with operability requirements for EQ equipment and components. The stated administrative controls were identical to those contained in the EQ Plan, Section V.

The licensee committed to the NRC that whenever safety class equipment or components which are environmentally qualified but are not covered by the Vermont Yankee TS fail (are not operable), a Non Conformance Report shall be generated with disposition of the discrepancy provided within 30 days. Corrective actions will be completed within the time frame specified in the approved NCR disposition. The NCR shall include a justification for continued operation.

In a June 15, 1984, Yankee Nuclear Services Division memorandum, it was noted that the Vermont Yankee Operations Department must be able to determine if any plant conditions that may occur during normal operation could impact the Design Bases of the EQ Program. The memorandum summarized the equipment and conditions relied upon to control accident environments and ensure EQ is not jeopardized. This new redundant SFP level alarm switches were specified to be electrical components required to be qualified for single failure-proof availability because they are equipment relied upon to control accident environments and ensure EQ.

On July 5, 1989, the Operations Department, with the concurrence of the Maintenance Department, white tagged out of service the power supply for the "A" SFP cooling system pump. This action also placed out of service the "A" channel of the redundant hi/lo level alarm instrumentation for the SFP. The equipment remained in this condition until July 3, 1990.

The failure of the licensee to identify and disposition the loss of operability of the SFP level instrumentation with a Nonconformance Report is considered a deviation from their written commitment to the NRC (DEV 50-271/90-10-06).

G. Ground Detection

As a result of the recent ground problem identified with the SFP cooling pump motor, the NRC inspected and evaluated the 480 Vac ground detection system used on safety related Bus No. 9. Also the effect of the ground (for other than the EQ related issue) on one phase of the motor was evaluated.

The ground detection circuitry for this bus of the 480 Vac distribution system consists of a local ground detection voltmeter for each phase of the three phase bus. Operations Procedure AP 0150, Responsibilities and Authorities of Operations requires that the Auxiliary Operator (AO) take readings on the Bus No. 9 ground detection meters each shift. The readings are recorded on the AO round sheets (VYAPF 0150.05), which requires the voltage readings be within 15 Vac of each other. Out of specification notations on the round sheets are circled. A bases for the 15 Vac difference value could not be determined.

As indicated in Table 2, Sequence of Events, the events associated with the ground were initiated by a ground alarm condition on the Standby Liquid Control (SLC) system's tank heater. The SLC tank heater and the "A" SFP cooling pump motor are both connected to Bus No. 9 480 Vac distribution system at Motor Control Center 9B.

The SLC tank heater's 480 Vac power supply is equipped with a "heater short-out" detection circuit which provides local indications and a control room alarm for identifying this condition, which can occur on any of the three phases. The design of this detection circuit utilizes a high impedance to ground on each phase. This intentional ground can, and has, created an interaction with the Bus No. 9 ground detection because the ground is intermittent (i.e., the heater turns on and off). Further, if the magnitude of the ground differs between phases, it will produce a differential voltage on the ground detection meter which could cause confusion. Essentially, a ground anywhere in the Bus No. 9 480 Vac distribution system will be sensed by the SLC heater short-out alarm circuit.

A review of the licensee's actions taken following the receipt of the SLC tank heater ground alarm was determined by the inspector to have resulted in a prompt investigation and corrective actions. These actions led to locating the original source of the ground and the realization that the SLC tank heater short-out detection circuit also detected the ground condition on the pump motor. A calibration analysis for the circuit revealed that the alarm relay was set to pick up at 24 volts rather than the 42 volts as required. Appropriate adjustments were made to circuit components. Since this circuit also depends upon differences on phase voltage for an alarm condition, the 42 volt setting should make it less sensitive to high impedance grounds than the Bus No. 9 ground detection circuit.

Observations made by the inspector include the following:

-- A bases for the maximum voltage difference for the voltage to ground readings for the Bus No. 9 ground detections circuit was not established. The bases equate the minimum acceptable impedance to ground for each phase. The meter readings did not correlate to the impedance of the ground.

-- The effect of the SLC tank heater deliberate high impedance ground was not factored into the system. Interactions between the two systems need to be determined and evaluated. Operator training did not address these interactions.

-- Procedures did not include operator response and actions required when the voltage differences exceed the established maximums.

-- The licensee's maintenance personnel were responsive in locating the ground and in addressing SLC tank heater short-out alarm calibration problems.

-- Effective ground detection on an ungrounded electrical distribution system can be most beneficial by detecting equipment and circuit weaknesses before these weaknesses cause an equipment failure or an interruption in the power feed(s) associated with the system. Based upon this discussion, the inspector determined that VY maintenance personnel followed good practices for operation of an ungrounded distribution system by isolating the grounded motor. Licensee records and documents indicate that from the time the ground condition on the motor was diagnosed until the motor replacement was effected, the motor was only energized (and then only potentially challenged the security of the electrical system) to conduct further maintenance investigations. As a result of discussions with VY personnel, the inspector determined the licensee exhibits a proper regard for not using grounded equipment for routine operations.

H. Safety Assessment

The SFP cooling system was designed to provide the capability to remove decay heat from the pool and maintain the pool temperature below the TS temperature limit for 150 degrees F. In achieving this safety objective, the reactor building environment is maintained within the bounding limits of the environmental qualification of electrical equipment. Essentially, the SFP temperature must be maintained below boiling. A single train of the two train system is capable of performing this function. All electrical equipment of this system was designed to meet EQ requirements. Specifically, the maximum post-accident reactor building temperature is 115 degrees F and the radiation level within the building assumes a TID 14844 core damage source term. Because of the "beyond design bases accident" assumed source term, the manually started pumps can be controlled from the shielded environment of the radwaste building.

Due to the ground on the "A" SFP cooling pump, and because it was believed to be prudent to not operate the pump with this condition unless it was necessary, the feeder breaker supplying power to this pump was white tagged in the open position. This feeder breaker is on a motor control center located within the reactor building. Thus, the plant operators would be required to enter the reactor building to restore power to this pump prior to its use. The licensee's actions also resulted in de-activating one of the two redundant level alarm instrumentation channels provided to monitor the SFP level.

The following facts and conditions are relevant in assessing the impact of the licensee's actions on plant safety:

-- At least 36 feet of water is maintained in the SFP (a TS limit). The "A" SFP cooling pump was de-energized approximately 145 days following the last refueling. The licensee's calculations, assuming an initial SFP temperature of 100 degrees F, determined that a total loss of cooling would cause the pool temperature to reach 150°F within 40 hours and boilding would occur 30 hours from the initiation of the loss of cooling event. Thus, sufficient time for plant operator action exists.

-- According to the licensee's analysis, the reactor building is accessible following the design basis accident described in the FSAR. Thus, a plant operator could enter the reactor building, close the feeder breaker and manually start the subject pump well before boiling of the pool initiated. However, given the ground condition of the pump and considering the elevated temperature of the building, the ability of the pump to perform its function in an elevated temperature environment is questioned. For the case of post-accident reactor building access and a loss of the normal SFP cooling function, plant operators can manually initiate augmented SFP cooling using the Residual Heat Removal System.

-- The plant design assumes a loss of normal power as part of the design basis LOCA. Following the automatic loading of plant equipment on an emergency deisel generator, a SFP cooling pump is started. Therefore, the starting of the pump from the radwaste building is an envisioned manual action, and the only unanticipated action would be the need to enter the reactor building.

-- The NRC requested the licensee to conduct an evaluation of the consequences of their actions that could have resulted from de-energizing the "A" SFP cooling pump. They calculated the probability of a scenario leading to core damage, plus the need to energize the non-operating pump to maintain fuel temperature below boiling, as in the order of 10 E-6 to 10 E-7 per reactor year. Taking credit for the "A" SFP cooling pump reduces the probability of exceeding the SFP temperature limits to 10 E-7 to 10 E-8 per reactor year. However, assuming the worst case conditions, the licensee's calculation shows that the SFP can be maintained below boiling by feed and bleed using the condensate transfer system in the radwaste building. Although not currently proceduralized, this potential plant response is viable to address the case of a loss of habitability of the reactor building. The licensee has an off-normal response procedure ON 3157, Loss of Fuel Pool Level. For the case of an inaccessible reactor building, this procedure provides instructions for SFP makeup via the operation of the condensate transfer system equipment located in the radwaste building.

-- With regard to the loss of one of the two redundant SFP level instrumentation channels, and assuming single failure conditions, a number of additional alarm features pertaining to SFP temperature are available to the plant operators in the control room. Had temperature indications, either in the vicinity of the SFP or the pool itself, been indicative of a loss of SFP cooling condition, the re-energization of the "A" SFP cooling pump of an by itself would have returned the redundant level channel to service. It is the use of the high and low level alarms that aid the plant operators in maintaining the correct level within the pool where normal makeup operations are not performed within the immediate area of the SFP.

Although it is of concern to the NRC that the licensee de-energized and relied upon a degraded SFP cooling pump, it is apparent that a number of installed design features provide appropriate means of mitigating the consequences of the licensee's actions.

TABLE 1 TO ATTACHMENT A

VERMONT YANKEE MINIMUM SHIFT STAFFING REQUIREMENTS

Minimum TS Shift Staffing* Table 6.1.1 Required:	Minimum 5-Member Fire Brigade* (TS 6.1.E)**	Alternate Shutdown Fire Emergency Not Declared	Assignment*** Fire Emergency Declared
Shift Supervisor (SRO)			Operator #1
Supv. CR Operator (SRO)		Operator #1	Operator #2
CR Operator (RO)		Operator #2	Operator #3
CR Operator (Alternate CRO with RO License)	Member #1	Operator #3	Operator #4
	Or		Or
Auxiliary Operator	Member #1	Operator #4	Operator #4
Auxiliary Operator	Member #2		
Shift Engineer	Member #3 (Brigade Commander)		
RP Technician (TS 6.1.D.1)	Member #4		
Security Personnel	Member #5		

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* Procedure AP 0894, Shift Staffing/Overtime Limits, identifies the shift personnel requirements for plant operations, including Fire Brigade Duties. TSs require that a minimum of two operators shall be in the Control Room, during startup or operations, at least one of these operators must be a senior operator.

** Procedure OP 3020, Fire Brigade and Fire Fighting Procedure, designates personnel trained to be fire brigade members and the composition of the on-duty brigade. The Alternate CRO and one of the Auxiliary Operators are interchangeable in terms of brigade duties.

*** Procedure OP 3126, Shutdown Using Alternate Shutdown Methods, designates the personnel assignments, within the constraints of the minimum shift staffing, which are necessary to provide safe shutdown from outside the control room. The Alternate CRO and Auxiliary Operator are interchangeable in terms of fire brigade and alternate shutdown assignments.

TABLE 2 TO ATTACHMENT A

SEQUENCE OF EVENTS

DATE	EVENT
05/26/89	Initiated MR-2160 to investigate ground alarm on the Standby Liquid Control (SLC) system's tank heater.
06/09/89	SLC ground alarm discussed at weekly OPS meeting.
06/09/89	Investigation determines ground on Bus No. 9 is due to "A" SFP cooling pump (or P9-1A) and not SLC tank heater. Ground occurs after operating pump approximately 20 minutes, MR 89-2291 initiated.
06/12/89	MR 89-2291 assigned to Maintenance Department to investigate and repair cause of ground.
06/13/89	Operations releases pump P9-1A to maintenance to conduct investigation, electrical breaker opened, electrical meggar indicates no phase to ground short.
06/14/89	Pump P9-1A white tagged out of service for the day for maintenance to obtain installed equipment data (Tagging Order 89-1362)
06/15/89	Fuel pool level switch "A" calibration performed by I&C Department in accordance with procedure OP 4341. Qualification Documentation Review (QDR) package No. 9.5 specifies 18 month calibration requirement (next due 12/15/90).
06/15/89	Maintenance Department initiates a Requisition (No. 11050) to procure on a routine priority a replacement motor for pump P9-1A. Date needed is specified as 10/15/89. Purchased item is classified as Safety Class Electrical, Seismic required, and EQ required.
06/16/89	Weekly OPS meeting notes that Maintenance Department could not identify the source of the ground on P9-1A; may be temperature induced.
06/23/89	Weekly OPS meeting notes same status as 6/16/89 entry above and that Operations Department is considering restarting pump.
06/24/89	Pump P9-1A restarted, ground reappeared within approximately 1 hour.
06/29/89	Acting Maintenance Supervisor approves Requisition No. 11050. Date needed is changed to 08/01/89 and procurement priority is changed to an emergency status.
06/30/89	Weekly OPS meeting notes same status as 6/16/89. Operations Department restarted pump P9-1A, ground reappeared. Maintenance investigating.

07/05/89	Operations Supervisor (OS) agrees with Maintenance Department for SFP cooling pump P9-1A to be white tagged out of service due to motor ground. Opening circuit breaker for pump on MCC-9B also places out of service one of the two redundant Hi/Lo level alarm instrumentation systems for the SFP. Electrical meggar readings indicate a dead short phase to ground. Digital Ohmmeter indicates 1500 ohms phase to ground. Maintenance Department did not review equipment tag out to assess impact on the SFP level instrumentation system. Acting maintenance supervisor considers pump inoperable, but is not aware of existence of Manager of Operations (MOO) directive 87-01.
07/07/89	Weekly OPS meeting notes that the Maintenance Department has ordered a new motor for pump P9-1A. (NOTE: This is the last mention of this issue as an outstanding item in the meeting minutes.)
07/17/89	Pump motor vendor responds in writing to licensee verbal request for quotation - specified a 36 week delivery.
08/17/89	Following licensee's Procurement Engineering and Yankee Atomic Electric Company's Yankee Nuclear Services Division reviews for technical and quality requirements, Requisition No. 11050 is issued as Material and Service Purchase Request No. 89-179.
08/22/89	Purchase Order (PO) No. 39059 is issued to pump motor vendor for delivery of a new motor by 05/15/90. At about this time Maintenance Department instructs Purchasing Department to investigate purchase from an alternate vendor and the possibility of refurbishing the existing motor.
09/06/89 & 09/12/89	Pump motor vendor acknowledgement of receipt of PO No. 39059 specifies a shipping schedule of 05/10/90.
09/12/89	Alternate pump motor vendor submits a quotation on a replacement motor at almost six times the cost of PO 39059 and an estimated shipment of between 26-36 weeks.
03/25/90	Fuel pool level switch calibrations scheduled to be performed per procedure OP 4341.
03/30/90	Fuel pool level switch "B" calibration performed. OP 4341 lists as a discrepancy the inability to perform the "A" instrument channel calibration due to .... FP "A" pump motor burned up; breaker W/T open. Closed breaker provides power for a level instruments, cannot do "A".
04/02/90	Shift Supervisor (SS) acknowledges on VYOPF 43.41.01 the status of the subject instrument calibrations.

05/22/90	Purchasing Department contacted pump motor vendor on status of delivery. Licensee informed, due to problems with testing the new motor, that a new shipping date of 07/30/90 was established. PO turned over to licensee expeditor to follow item.
05/25/90	Pump motor vendor advised by expeditor that motor is required for fuel pool but, not a Technical Specification or LCO item. New scheduled shipping date of 06/22/90 established. New motor being fabricated had to be rewound and requalified.
05/29/90 thru 06/27/90	Licensee's Purchasing Department contacted pump motor vendor at least 14 times to expedite delivery and exploring alternatives for obtaining a new equivalent motor.
05/30/90 (approx)	I&C Engineer (who performs the duties of the I&C Department Surveillance Test Coordinator) aware that "A" fuel pool level instrumentation was not completed, reviewed MOO Directive 87-01, and informed Assistant Operations Supervisor (AOS) of issue. AOS did not know if directive was still in effect. He would review matter with OS and research issue.
06/15/90 (approx)	I&C Engineer contacted AOS, who had not researched issue as of this time. He then contacts the SS on duty to ascertain status of the MOO directive. The SS could not find any information on the MOO directive.
06/26/90	I&C Engineer requests the Operations Support Department (OSD) Liaison Engineer to determine the status of the MOO Directive.
06/27/90	Liaison Engineer reviews status of the SFP cooling system, procedures, licensing documents and delivery status of the new motor. He concludes that the administrative controls contained in MOO Directive 87-01 were still in effect and the commitment to the NRC that would implement those controls were still applicable.
06/27/90	Licensee hires a dedicated truck to pick up new motor at vendor facility.
06/28/90	New motor received at plant.
06/28/90	Plant Manager was briefed on the issue, directs that a review of NRC correspondence be conducted to ensure that the intent of the MOO Directive had not been withdrawn. Operations Supervisor (OS) considers pump operable with white tag and de-energized, in part because no maintenance work was performed.

06/29/90 During receipt inspection of the new motor, Material Disposition Request (MDR) No. 90-100 was issued to document a number of discrepancies between the PO requirements and the as received unit. These discrepancies include dimensional, ratings, characteristic and documentation deviations.

07/03/90 White Tag on "A" SFP cooling pump is cleared, breaker is closed making power available to pump, and a Caution Tag is issued for the pump that stipulates to leave the pump in the OFF position and for emergency use only. This action appears to have been taken at this time to resolve, for the time being, questions the licensee had about the ability of the tagged out of service pump to perform its EQ Program safety function. The issues raised were subsequently documented in a 7/20/90 licensee memorandum.

07/16/90 Licensee conducts meeting on Vermont Yankee (VY) EQ Program requirements associated with "A" SFP cooling pump. Questions were raised as to the intent of the 10 CFR 50.49 Rule and Section V.2.1 of the VY EQ Program Plan, which addresses the operability of EQ safety-related equipment not covered by VY TSs. OSD was to pursue interpretation of the rule to ascertain if reportability and/or corrective actions are warranted.

07/17/90 Pump motor vendor certifies dimensional deviation as acceptable to meet motor performance and seismic qualification.

07/20/90 Technical evaluation and justification is provided for MDR 90-100, and disposition is to use motor as is.

07/24/90 New motor for pump P9-1A released from stockroom.

07/24/90 Licensee contracts with consultant to provide engineering services to review the VY EQ Program Plan, specifically Section V.2-1 and to provide guidance with respect to EQ equipment operability and compliance to the EQ rule. The licensee's compliance and reportability of the SFP cooling pump case was to be specifically addressed as well as any generic implications.

07/25/90 Pump P9-1A released for work by the SS. A white tag is issued and the motor breaker is tagged open. Senior Control Room Operator lists the "A" SFP cooling pump as inoperable in the shift turnover log. A 30 day time limit is specified in accordance with MOO Directive 87-01. Operations Department considers the pump available but not operable.

07/27/90 Motor replacement complete, white tag is cleared and motor breaker is shut. A Caution Tag was issued to indicate that a Nonconformance Report (NCR) by the Maintenance Department is outstanding.

07/30/90	Maintenance Department Senior Engineer requests via OSD the services of YNSD engineering to prepare an EQ NCR due to incomplete EQ documentation and preparation of a QDR for the new motor. The request notes that the 30 day time limit specified in the EQ program to develop the NCR expires on 08/24/90.
08/01/90	PORC reviews "A" SFP cooling pump motor replacement. Notes problems with associated EQ documentation, that the issue will be resolved by the NCR process, and the motor will not be declared operable until this issue is resolved.
08/10/90	YNSD engineering responds to the service request, notes that an NCR is not required for the newly installed pump, and provides EQ documentation to meet EQ Program requirements.
08/10/90	Licensee's EQ consultant provides a summary report on the operability and qualification status of the SFP cooling pump motor, with a conclusion that the qualification of the motor with the ground was not compromised.
08/15/90	Caution Tag on "A" FPC cooling pump is cleared, inoperability in Shift Turnover Log is closed.