U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No.	93-33
Docket No.	50-271
Licensee No.	DPR-28
Licensee:	Vermont Yankee Nuclear Power Corporation RD 5, Box 169 Ferry Road Brattleboro, VT 05301
Facility:	Vermont Yankee Nuclear Power Station Vernon, Vermont
Inspection Period:	November 28, 1993 - January 15, 1994
Inspectors:	Harold Eichenholz, Senior Resident Inspector Paul W. Harris, Resident Inspector Daniel Dorman, Project Manager

Approved by: Gene Kelly — February 15, 1994
Eugene M. Kelly, Chief — Date
Reactor Projects Section 3A

Scope: Station activities inspected by the resident staff this period included Operations, Maintenance, Engineering, Plant Support, and Safety Assessment and Quality Verification. Review of recent industry event assessments was an initiative selected for inspection. Backshift and "deep" backshift including weekend activities amounting to 33 hours were performed on November 28, December 2, 17-19, 27, 1993 and January 11-13, 1994. Interviews and discussions were conducted with members of Vermont Yankee management and staff as necessary to support this inspection.

Findings: An overall assessment of performance during this period is summarized in the Executive Summary. Unresolved items were opened regarding the use of routine versus significant corrective action reports, specifically Corrective Action Report 93-59 for under-vessel work performed during the Fall 1993 outage (Section 3.1); and, the licensing basis, specifically the number of pumps and flow, for an operable service water subsystem (Section 3.2).

EXECUTIVE SUMMARY

Vermont Yankee Inspection Report 93-33

Operations

Well managed reactor operations established plant conditions necessary for the conduct of corrective maintenance. "Black board" (i.e., few or no annunciators lighted) conditions on control room panels contributed to safe plant operation. Appropriate operability determinations were made following review of surveillance results for a condensate storage tank level transmitter.

Maintenance

A corrective action report initiated to review inadequately controlled maintenance on a control rod drive during the Fall 1993 refueling outage lacked a critical evaluation of human factors, equipment deficiencies, and other issues. Independent safety, quality and management reviews and a formal root cause determination were also not performed. A review of service water (SW) pump preventive maintenance identified no concerns; however, a question regarding the number of pumps necessary to constitute an "operable" SW subsystem was identified.

Engineering

Debris shields on turbine casing over-pressure vent pipes for the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems were installed without an engineering evaluation during original plant construction; system operation was not adversely affected. Inservice testing data obtained from a HPCI pump surveillance were properly evaluated.

Plant Support

A radiological safety issue involving control of a high radiation area was effectively resolved. Trending of reactor vessel conductivity resulted in improvements in condensate demineralizer performance.

Safety Assessment and Quality Verification

Review of industry events contributed toward increased awareness of main turbine operations and good evaluation of a potentially degraded condition in the emergency diesel generator air start systems. Initiatives are underway to improve corrective action processes and implementation, including the use of third-party audits and insights gained from other nuclear utilities. Appropriate detail and assessment were provided in a report on fuel element performance.

TABLE OF CONTENTS

ATTACHMENTS

Note: Procedures from NRC Inspection Manual Chapter 2515, "Operating Reactor Inspection Program" which were used as inspection guidance are parenthetically listed for each applicable report section.

DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

Vermont Yankee Nuclear Power Station continued full power operations this period, although on three occasions the reactor was shutdown to repair non-safety related components (Section 2.2). The degraded conditions represented challenges to continued plant operation, however, did not preclude safe plant operation. In addition, low power reactor operation was necessary for a period of five days to identify the failure of the main condenser equalizing pipe. An enforcement conference (Section 8.1) was held at NRC:Region I (NRC:RI) to discuss causes and corrective actions for safety system issues identified during refueling/maintenance outage (RFO) XVII, and which were documented in NRC Inspection Reports 93-21 and 93-29.

2.0 OPERATIONS (71707, 93702, 92700)

2.1 Operational Safety Verification

Daily, the inspectors verified adequate staffing, adherence to procedures and Technical Specification (TS) limiting conditions for operation (LCO), operability of protective systems, status of control room annunciators, and availability of emergency core cooling systems. Plant tours confirmed that control panel indications accurately represented safety system line-ups. Safety tagouts properly isolated equipment for maintenance. A review of the reactor operator logs verified that out of specification indications were circled, reviewed, and causes understood by operators. Corrective actions for abnormal conditions were initiated. The inspectors observed that Shift Supervisors (SS) communicated accurate plant conditions and planned maintenance and surveillance to shift personnel during control room pre-shift briefs. During these briefs, chemistry technicians enhanced operator knowledge of condensate demineralizer performance by providing a trend of reactor water conductivity.

Prior to the reactor power changes described in Section 2.2, the control room operators (CROs) were observed to prepare for anticipated activities by reviewing procedures and receiving management instructions. A professional atmosphere was maintained in the control room and operators were competent in their duties. During the short duration shutdowns, safety systems required to support the current mode of reactor operation were operable in accordance with TS requirements. Control panel valve lineups and electrical power supplies supported this determination.

2.2 Plant Operations to Support Maintenance

On three occasions this period Vermont Yankee (VY) reduced reactor power to support corrective maintenance on non-safety related systems. The first power reduction was preplanned and occurred on December 6 and lasted for two days during which an emergency drain valve for the "A" moisture separator was repaired. The failure of this valve (LCV-103-23A) occurred subsequent to startup from RFO XVII and did not represent a safety concern (NRC Inspection Report 93-26). The valve was repaired, cause was attributed to foreign material, boroscopic and ultrasonic testing verified good piping integrity, and a plant startup was initiated. However, the power ascension was halted because main condenser air inleakage increased to approximately five times normal values (typically 10-15 scfm). This condition was evaluated as an unacceptable condition warranting repair, although it did not immediately challenge safe plant operation. As a result, VY placed the plant in cold shutdown on December 9 and replaced a turbine expansion boot on the north condenser that was thought to be the source of the air inleakage. On December 12, the boot replacement was completed and the reactor was again made critical, however, the high air inleakage still existed.

For the next five days, VY operated at less than 25 percent rated power to maintain condenser vacuum, which was necessary to investigate and identify the cause of the air inleakage. Work crews conducted pre-planned system inspections around the clock and verified system valve lineups. A multi-disciplined task team was chartered to investigate, stage, and repair the condition. Responsibilities for key tasks were assigned and communications with other nuclear facilities occurred to aid the investigation. Yankee Nuclear Services Division (YNSD) and VY engineering assessed long-term low power operations, and adjustments to average power range instrumentation were evaluated. Appropriate instructions were provided to control room operators regarding plant conditions and repair efforts.

On December 17, VY identified that a weld failure on the main condenser equalizing pipe was the cause of the air inleakage, and a shutdown to cold conditions was initiated. The function of the 72-inch diameter equalizing pipe, which connects the north and south condensers, is to equalize pressure resulting from normal and/or transient plant operation. The pipe is physically difficult to access and to conduct visual inspections and, therefore, was not subject to investigation prior to the first air inleakage repair on December 9. Metallurgical analyses of the pipe were conducted to assess pipe integrity and to support repairs. The welds that failed are circumferential and connect the pipe to both the north and south condensers. A YNSD materials specialist assessed that the failure was due to poor weld quality exacerbated by thermal and hydraulic stress. On December 22, VY returned to full power operation.

The shutdowns and power reductions were well planned and executed. Bar charts and man-loading schedules were developed to support maintenance. The actual activities performed paralleled the anticipated sequence of events. Vermont Yankee safely managed the periods of low power and shutdown operations to conduct other corrective maintenance and surveillance. Good management oversight and involvement were demonstrated during the planning resulting in effective inter-departmental coordination. Evaluation of low power operations and high offgas flow rates contributed toward an overall safety assessment of facility operation. Vermont Yankee's ability to safely change reactor modes of operation and maneuver at power without unnecessary plant transients, safety system actuation, or subsequent equipment failures represents excellent control of facility operations.

2.3 Control Room Panel Annunciators and Indications

Throughout this inspection period, VY prioritized maintenance to assure the timely repair of control room panel annunciators and indications. For example, an expedited repair was conducted to restore a group of panel annunciators following a premature failure of an annunciator power supply. This corrective maintenance received appropriate review and planning. On a daily basis, the inspectors performed control panel walkdowns and continually observed that system indications were within calibration and supported plant operation; inoperable or degraded annunciators are rarely observed. Equipment deficiencies that do exist are understood by control room operators and repair priority is based on the availability of redundant equipment, safety significance, and effect on plant control.

The prompt repair of panel deficiencies reduced unnecessary annunciations and panel indications. From a human factors standpoint, this contributed to timely assessments of alarmed conditions, aids in the manual operation of safety systems, and reduced the likelihood of inaccurate system assessments. The maintenance of blackboard conditions represents an appropriate operating philosophy.

2.4 Reportability Assessment

On December 21, during surveillance of the condensate storage tank (CST) water level instrumentation, VY identified that the as-found calibration of one of the two level transmitters was out of specification low. These instruments provide for an automatic high pressure coolant injection (HPCI) suction swap over from the CST to the torus on low CST level. The condition was deemed reportable by the control room shift supervisor (SS) based on TS Table 3.2.1, "Emergency Core Cooling System Actuation Instrumentation - High Pressure Coolant Injection System." This TS states that should the number of operable instruments be less than two for the trip system, HPCI system shall be considered inoperable and the requirements of TS 3.5.E apply. Based on this, the SS entered the 7-day LCO, initiated a potential reportable occurrence (PRO) report, and made a 10 CFR 50.72 4-hour notification at the same time as declaring HPCI inoperable. Later that day, the instrument was calibrated and the surveillance completed. Subsequently, following engineering assessment and management review of the PRO, the 4-hour notification was retracted because HPCI functionality was never lost.

Vermont Yankee appropriately concluded that the event was not reportable as a 10 CFR 50.72 4-hour notification or a 10 CFR 50.73 licensee event report (LER). In the condition described above, the HPCI would have performed its design function and the HPCI pump suction transfer would have occurred, because this automatic suction swap is based on one-out-of-two logic and the second instrument was within calibration and operable. A clarifying note in Table 3.2.1 for this instrumentation states that there is one trip system with initiating instrumentation arranged in a one-out-of-two logic. The engineering evaluation of the PRO was of adequate detail and clarity.

3.0 MAINTENANCE (62703, 61726, 40500)

The inspectors observed selected maintenance on safety-related equipment to determine whether these activities were effectively conducted in accordance with VY TS, and administrative controls (Procedure AP-0021 and AP-4000) using approved procedures, safe tagout practices and appropriate industry codes and standards. Interviews were conducted with the cognizant engineers and maintenance personnel and vendor equipment manuals were reviewed.

3.1 Maintenance on an Incorrect Control Rod Drive

During refueling outage (RFO) XVII, VY identified that maintenance was performed on an incorrect control rod drive (CRD 38-27). This was identified on October 17, 1993, during the reactor pressure vessel (RPV) hydrostatic test because reactor coolant was leaking from the flange connecting the CRD to the RPV lower head. The leakage was greater than the acceptance criteria of 20 drops per minute contained in plant procedure OP 4101, "RPV Operational System Leakage and Hydrostatic Test," although a quantitative leak assessment was not performed. The inspectors concluded that the safety significance of this leakage was relatively low for the plant conditions at the time of the hydrotest because: CRD flange leakage is procedurally recognized and evaluated; control room operators periodically monitor containment leakage and reactor vessel water level; and, CRD "shootout" steel was installed during the hydrostatic test. Vermont Yankee found that three of eight cap screws on the wrong CRD had been inadvertently untorqued on August 29, 1993, at the beginning of RFO XVII. The licensee stated that two of eight are necessary to maintain pressure boundary integrity at operating pressure.

Corrective action report (CAR) 93-59 was initiated on October 19, 1993 to assess this event and identify corrective action recommendations. The report was classified as routine and reviewed by the Maintenance Manager and Operations Superintendent. The apparent cause was determined to be the inadvertent detorquing of the wrong flange bolts due to loss of co-worker verification and the later switching of undervessel crews in the middle of a change out. Immediate corrective actions included retorquing the flange bolts and, for future work, the placement of identification tags on the CRDs. Procedure precautions are also planned to assure workers verify correct core location prior to detorquing and to require that two workers be under the vessel for all work activities. Vermont Yankee plans to discuss this event and the importance of job turnovers at the pre-job brief scheduled for the next refueling outage. The actions described above meet the intent of VY procedure AP 0007, "Corrective Action Reports."

Administrative Procedure AP 0007 states that CARs are prepared to investigate the causes of plant events which have the potential to result in conditions adverse to quality. Classification, as either routine or significant, is made by a Department Superintendent based on details of the event using criteria established in a separate VY Corrective Action Guideline. A significant CAR applies to a condition adverse to quality which could or has affected plant safety, or is a condition that represents a breakdown in the Quality Assurance Program. The management reviews for a significant CAR include PORC, Quality Assurance, Plant Manager, Technical Programs, YNSD engineering, and the Vice President, Operations. A formal root cause analysis is also performed to develop actions to preclude recurrence. A routine CAR is used for adverse conditions which appear to be isolated, not symptomatic of other conditions, will not lead to a significant item, or do not typically pose an immediate safety concern. An apparent cause is determined and management reviews include the Department Manager and Superintendent. It should be noted that of the 71 CAR's issued in 1993, 18 were classified as significant.

Based on the above, classification of CAR 93-59 as routine represented a missed opportunity for VY to more comprehensively investigate the significance of this issue. The problem remained undetected for approximately 50 days until discovery during the hydrotest. The CAR lacked a critical assessment of supervisory oversight, personnel safety issues, communications, and the acceptability of CRD tooling. Its unclear whether contract workers involved were actually interviewed. Specific complicating factors contributing to maintenance on the wrong CRD, which were ascertained by the NRC inspector from detailed maintenance logs but were not identified as part of routine CAR 93-59, included:

- concerns involving visual difficulties and insufficient air pressure to the air-fed personnel protection hood worn by the workers were dismissed following a perfunctory assessment of the air supply regulator, although a medical emergency involving worker fatigue subsequently occurred;

- ineffective communications between an assigned engineer (outside the drywell) and the work crews contributed to initiating maintenance on the wrong CRD flange;

- deviation from the pre-planned work sequence occurred for another CRD without either VY knowledge or consideration for radiation dose assessment; and,

- CRD maintenance continued despite frequent and recurring problems with tooling.

Another result of not treating CAR 93-59 as significant was that the YNSD QA Group does not necessarily monitor the responsible contractor, as part of additional surveillance, when performance problems may be suspect. The inspectors concluded that the lack of quality maintenance on a RPV component represented a safety significant condition adverse to quality, caused by breakdowns in multiple layers of work control, and warranting a more formal root cause assessment and independent evaluation (including trending) by the PORC, QA, engineering, and senior licensee management. The adequacy of the guidance provided for the classification of conditions adverse to quality within the corrective action process is unresolved pending expanded review of CAR's by the NRC and evaluation of the conclusions of a recently initiated VY Task Force (refer to Section 6.2) **(URI 93-33-01)**.

3.2 Review of the Service Water Pump Preventive Maintenance

The inspector reviewed the "C" service water (SW) pump maintenance package and concluded that VY appropriately justified the removal of this safety system pump from service based on preventive maintenance practices. These practices included: (1) trending of degrading conditions associated with pump and motor bearings to assure repair prior to component failure; (2) overhaul of the pump and motor to improve corrosion resistance and electrical efficiency; (3) surveillance of valve and motor heaters; and, (4) good performance resulting from previously conducted SW pump maintenance at power. Replacement parts were available and responsibilities were assigned. The maintenance was planned based on eight workers, five days a week, for eleven days. Review of the maintenance package was performed by Maintenance, Instrument & Control and Engineering Departments. No additional comments or considerations were documented. The status of this maintenance was reviewed at the daily Plant Manager's meetings.

The licensee concluded that this maintenance would be tracked by an administrative entry into a 30-day LCO, because the requirements of TS paragraph 3.5.D do not address the removal of one pump from operation. The TS and associated bases, as well as the Final Safety Analysis Report, are silent as to the number of SW pumps required for subsystem operability. Nonetheless, based on their assumptions, selection of the 30-day LCO was consistent with the VY LCO Maintenance Guideline, and voluntary removal of the pump for maintenance was justified based on the considerations of Generic Letter 91-18. The lack of a clear licensing basis for the service water subsystem's TS (either required flow or number of pumps) has been discussed with NRC Office of Nuclear Reactor Regulation representatives, warrants further VY and NRC review, and is therefore unresolved **(UNR 93-33-02)**.

4.0 ENGINEERING (71707, 92700, 92701)

4.1 Turbine Casing Vent Pipe Debris Shields (Closed URI 93-19-01)

Prior to plant startup from refueling outage (RFO) XVII, the inspector observed a pre-startup PORC meeting during which the committee concluded that the debris screens installed on the HPCI and reactor core isolation cooling (RCIC) systems did not represent a system operability concern. Further, the licensee concluded that the shields were installed without an engineering evaluation during original plant construction. Procedure changes to restrict access to the general area of the vents were necessary to assure personnel safety in the unlikely event that individuals are in the area of the vent exhaust during a rupture disk failure. The inspection was conducted, in part, in response to an event at the Quad Cities Nuclear Power Station in which personnel were injured because the HPCI turbine case vent pipe exhausted into the space occupied by plant personnel. During this event, concerns were identified involving preventive maintenance and system design (NRC Information Notice 93-67).

Vermont Yankee reviewed Information Notice 93-67 and responded to inspector questions regarding a turbine casing rupture disk failure, as documented in CAR 93-39. The licensee determined that no significant damage would result to surrounding equipment should a debris screen be ejected from the end of a vent pipe during an abnormal over-pressure condition, and screen installation did not affect system operability. Further, the licensee determined that maintenance performed to verify the functionality of the steam line condensate drains (a potential cause of turbine casing over-pressure) was adequate and periodically performed. The inspector independently confirmed this assessment and reviewed the results of the last surveillance performed during RFO XVII. The inspector also confirmed that pressure switches that indicate and provide alarm functions based on high exhaust pressures are similarly surveilled. In both cases, no deficiencies were identified.

4.2 Evaluation of HPCI Surveillance Results

On December 21, surveillance of the high pressure coolant injection (HPCI) system was performed to verify system operability in accordance with TS and inservice testing (IST) requirements. Results indicated that the HPCI pump developed flow and pressure within acceptance criteria, however, the stroke time for the turbine control valve was approximately one second too fast (9.2 to 9.4 seconds verses the ten second allowable). The lower limit on valve time is in recognition of the potential for turbine overspeed. This condition placed the valve in the IST "Alert" range, requiring a retest. On December 22, the retest was conducted at "cold iron" conditions and results were similar; the HPCI system was then briefly considered inoperable for a period lasting 45 minutes. Later that day, maintenance/engineering justified continued operability of the HPCI system based on discussions with the turbine vendor, evaluation of recent surveillance results, and discussions with the Operations Department regarding the ability of the system to perform its design function. Justification for continued operability with the slightly fast control valve was based on the integrated effect of: (1) auxiliary oil pump hydraulic pressure development; (2) stop valve opening; and (3) the dampening by the ramp generator speed controls. Also, the licensee has in the past experienced control valve opening times as fast as 6.3 seconds, without an overspeed trip of the turbine.

The evaluation of HPCI surveillance results were in accordance with plant procedure AP 0164, Rev. 3, "Operations Department Inservice Testing." The operability assessments by the SS and engineering were timely and based on good engineering practices, involving vendor discussions and sensitivity analysis of the effects of the fast stroke time on turbine operation. Good control room log keeping was observed that accurately described the current state of system operability. The Maintenance Department and IST coordinators were cognizant of the identified condition, and a corrective action report was generated to track resolution. Further evaluation of HPCI system performance will be conducted in March 1994, and vendor assistance is planned. A VY initiative to develop generic guidelines for the assessment of IST data and the implementation of the program continues. The IST deficiency was not reportable. Recent NRC reviews of the VY IST Program are documented in NRC Inspection Reports 93-21 and 94-04.

5.0 PLANT SUPPORT (71707, 93702, 92700)

5.1 Radiological Controls

Inspectors routinely observed and reviewed radiological controls and practices during plant tours. The inspectors observed that posting of contaminated, high airborne radiation, radiation and high radiation areas were in accordance with administrative controls (AP-0500 series procedures) and plant instructions. High radiation doors were properly maintained and equipment and personnel were properly surveyed prior to exit from the radiation control area (RCA). Plant workers were observed to be cognizant of posting requirements and maintained good housekeeping. An inspector review of VY's control of high radiation boundaries and locked door keys identified no concerns (Section 5.1.1).

5.1.1 Control of Locked High Radiation Boundaries

On November 17, a radiation protection (RP) supervisor questioned RP Department control of the CRD equipment hatch to the drywell. The supervisor observed that RP postings appropriately defined the access as a very high radiation area, however, the key for the lock securing the access hatch was controlled by the Security Department. This appeared to be different than required by TS 6.5.B.1, which requires that keys for locked doors shall be maintained under administrative control of the Shift Superintendent (SS) and/or the plant health physicist. A potential reportable occurrence (PRO) was written to document the observation and to initiate an engineering review. In addition, an RP lock was installed as a conservative measure to assure positive RP control and the applicable procedure was changed pending the PRO review. On December 7, the Technical Services Superintendent concurred with the completed PRO assessment which concluded that the identified condition did not represent a personnel safety concern nor TS noncompliance. This was based on NRC Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," and the Final Safety Analysis Report definition of "access."

The inspector discussed VY's assessment with an NRC:RI radiation specialist and concluded that the licensee's assessment was correct. The identification and evaluation of this RP concern demonstrated good attention to detail and timely disposition of a potential radiological safety issue.

5.2 Security

The inspector verified that security conditions met regulatory requirements and the VY Physical Security Plan. Physical security was inspected during regular and backshift hours to verify that controls were in accordance with the security plan and approved procedures. Following a walkdown of the security perimeter, the inspector discussed contingencies to assure that environmental obstacles caused by adverse weather did not reduce the effectiveness of Security

Officer rounds. In addition, the inspector reviewed the contingencies in place to support SW pump preventive maintenance (Section 3.1.2). In both cases, appropriate measures were implemented and officers were cognizant of Security Department instructions.

5.3 Chemistry

During this period, the Chemistry Department trended increasing reactor vessel conductivity and concluded that this condition did not represent an immediate safety issue. Chemistry personnel noted that vessel conductivity indicated a slow increase from historically low levels of approximately 0.07 umhos/cm to a current value of 0.081 umhos/cm. Technical Specification limits are approximately 100 times these values and procedural action limits are pre-established to initiate corrective actions prior to reaching TS limits. Based on the current rate of increase, the first procedural action level (0.2 umhos/cm) would not be exceeded until March 15, 1994. Daily, department managers discussed the increasing trend at the Plant Manager's meeting and plans were initiated to assess condensate demineralizer performance.

5.4 Housekeeping and Fire Protection

Plant housekeeping this period remained very good. Inspections conducted in the reactor building confirmed that transient materials were properly stored and did not affect the operability of safety systems. Lighting was adequate and accessibility to plant components was unrestricted. A walkdown of flammable material exclusion areas (fire separation zones between safety syst em trains) identified no transient combustibles. All exclusion areas were conspicuously identified.

6.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (40500, 90712, 90713, 92700)

6.1 Industry Event Assessment

Vermont Yankee continues their effort to inform department managers and control room operators (CROs) of industry events that represent potential safety issues or lessons learned that are applicable to VY. One issue reviewed this period involved a 10 CFR Part 21 concern regarding premature cracking of Fairbanks Morse emergency diesel generator (EDG) air start distributer cams. Vermont Yankee verified that inspection of the cam on the "A" EDG was performed during RFO XVII and no indications were identified. Further, inspection of the " B" EDG cam will be conducted in March 1994, during planned preventive maintenance. Parts are on order to support replacement, and communications with the diesel vendor occurred regarding this issue. The Plant Manager's Meeting facilitated a timely review of the distribut or cam issue and commitments were assigned to assess potential corrective actions to preclud e occurrence at VY. A second event assessed by VY involved a turbine trip and subsequent reactor scram at the Fermi Nuclear Power Station. This event occurred due to a loss of lubricating oil to the turbine bearings, causing high turbine vibration. Operator review of thi s event was required by the Operations Night Order book, and based on discussions, the inspectors found that CROs were knowledgeable of the Fermi event.

6.2 Corrective Action Audit

The licensee recently completed quality assurance (QA) Audit Number 93-17 of its corrective action processes in December 1993. The audit team found that corrective actions for identified concerns have not been effectively implemented to meet management's standards and expectations, and cited several examples. The team recommended that attention be given to process inadequacies as well as implementation, and that consideration should be given to oth er independent assessment, and third party reviews.

As noted by the licensee's audit team, this issue has been identified in several forums and various ways. The inspector noted that in January 1994, the licensee named a task force to determine the root cause of this problem and to recommend corrective action that will prevent problems from recurring. The conclusions and recommendations of this task force will be evaluated as part of routine NRC reviews of facility operations.

6.3 Self-Assessment

Vermont Yankee issued a new self-assessment policy in June 1993. This policy was briefly reviewed by the Operational Safety Team Inspection (NRC Inspection Report 93-80). The policy required that each department develop its own self-assessment program according to the guidance provided in the policy statement. Department self-assessment programs were developed late in the summer of 1993 and have been implemented. The inspector reviewed several department programs and found that they represented a good first effort. However, some of the programs did not clearly address the issues raised in the policy statement, including acceptance criteria and objective methodologies.

The inspector reviewed several recent departmental self-assessments conducted pursuant to the new departmental self-assessment programs. They were generally comprehensive, evaluating a good mix of compliance, practices and process issues. Several assessments of surveillance procedures for TS compliance provided recommendations for procedure clarification and use of administrative limits in the procedure acceptance criteria. These recommendations were forwarded to the individuals responsible for the associated procedures. No tracking of resolutions was provided for. The responsible individuals were expected to resolve the recommendations or, if they were not urgent, to hold the recommendations in the procedure file until the next routine review of that procedure (once every two years). The current process does not require feedback by the cognizant department on the nature of the observation and the manner in which resolution was provided.

A self-assessment of non-conformance reports (NCR) generated by the Mechanical Engineering and Construction Department focused on how the product met the guidance for the process as stated in the licensee's NCR procedure (AP-6021) and the Vermont Yankee Corrective Action

Guideline. The stated objectives and results of the self-assessment report did not address the validity of the process which has been the subject of concern in NRC inspection reports, licensee QA findings, and other independent assessments. The self-assessment report did, however, provide recommendations in process areas for: (1) improving the "paper trail" to facilitate verification of commitments arising from an NCR, and (2) broadening the scope of trending routine items. These recommendations are consistent with previous independent findings. The inspector considered the continuing identification of these process concerns to be appropriate.

Late in 1993, the licensee's Technical Programs Supervisor participated as a technical specialist in a quality assurance audit of the self-assessment programs at another nuclear power plant and returned with suggestions to further improve VY's self-assessment programs. He applied these in a candid and critical self-assessment of his own department's self-assessment program, recognizing several opportunities for potential improvement. The inspector found that this was typical of the current self-assessment culture at VY. While the licensee has achieved improvement in this area, it also recognizes that room for improvement remains and is looking outward, as well as inward, for ideas.

6.4 Third Party Audits

Vermont Yankee recently changed its membership for third party audits from the Combined Utility Assessment Group (CUAG) to Joint Utility Management Audits (JUMA). The CUAG included several utilities that routinely provided technical specialists for QA audits at VY. Additionally, VY provides specialists for QA audits at these utilities. Since VY was already drawing on the experience of these utilities through the QA program, VY elected to change their third party audit membership to JUMA. Through JUMA, VY envisions that it will draw on experience from other utilities with single unit BWR sites. Experience of several of the members of the CUAG will be retained through the licensee's QA programs. The inspector viewed this change as a positive initiative in independent assessment and use of outside industry experience.

6.5 Review of Written Reports

The inspectors reviewed the following Licensee Event Reports (LERs) to verify accuracy, description of cause, and adequacy of corrective action. The inspectors considered the need for further information, possible generic implications, and whether the event warranted further onsite followup. The LERs were also reviewed with respect to the requirements of 10 CFR 50.73, and the guidance provided in NUREG 1022.

- LER 93-07: Failure to Perform Daily Instrument Checks Due To Management and Human Factors Weaknesses

During the conduct of surveillance procedure reviews, VY identified that five TS-required daily instrument checks were not performed due to management and human factors weaknesses. Enhanced procedure reviews were a corrective action for identifying and correcting surveillance weaknesses. The instruments involved were the auxiliary power monitor and pump bus power monitors for the low pressure coolant injection and core spray systems, the high drywell pressure sensors for the emergency core cooling and primary containment isolation systems, and the RCIC bus power monitor. In all cases the licensee determined that, despite the failure to perform instrument checks, the events were of low safety significance because redundant alarms and indications existed and/or the systems were able to perform their safety function.

The immediate corrective actions to assure satisfactory performance of TS-required daily instrument checks were appropriate. These actions consisted of Standing Orders to CROs, changes to operating procedures, and assessments to determine the "operability" of subject systems. Long-term corrective actions focused on the review of TS surveillance requirements, procedures, and envelope corrective actions already identified and initiated due to previous concerns involving TS surveillances (NRC Inspection Reports 93-12 and -13). In VY letters dated July 2, 1993 and December 30, 1993, they documented their initiative to perform a more comprehensive review of surveillance requirements by May 1994. In addition, by letter dated January 14, the Vice President, Operations acknowledged that an adverse trend in the area of TS surveillance has occurred and programmatic improvements have been initiated. The inspector verified that the corrective actions as documented in LER 93-07 will be tracked and dispositioned by VY in this effort. A recent NRC special safety inspection (NRC Inspection Report 93-31) confirmed that the surveillance test program appropriately ensures safety-related equipment operability. The NRC will continue to review the long term effect of these program improvements.

- LER 93-18: Group 4 Primary Containment Isolation on Initiation of "A" Shutdown Cooling System Due to Pressure Spike

This LER, dated January 13 documented the December 17, 1993, actuation of a residual heat removal (RHR) system isolation in the shutdown cooling (SDC) mode of operation. This isolation, as controlled by the primary containment isolation system (PCIS), was determined to have been caused by a momentary pressure transient in the RHR discharge piping moments after the "A" RHR pump was started for shutdown cooling. Shutdown cooling was immediately established using the "A" RHR pump (of the "A" RHR subsystem) following reset of the PCIS isolation. NRC LERs 93-11 and 91-06 document similar actuations received on the "B" subsystem, however, the December 17 event was the first of this type when attempting to initiate SDC using an RHR pump in the "A" subsystem.

The inspector concluded that the PCIS actuation was valid, reportable in accordance with 10 CFR 50.73, and did not represent a significant condition adverse to system integrity. The pressure spike was of short duration and relatively low magnitude, and not recorded by the plant process computer. On the evening of the event, the inspector verified that the system operating procedure and alarm response for the PCIS actuation were followed. Auxiliary Operators, who vented the RHR system prior to pump start, verified piping integrity and observed no significant pipe motion following pump start. The inspector observed the subsequent restart of SDC. On December 18, during weekend inspection coverage, VY management was found to be cognizant of the isolation and familiar with long-term corrective actions already initiated due to the previous occurrences.

LER 93-18 contains some new information in regards to system response following pump start and previous procedure changes implemented to preclude recurrence. A significant corrective action report was initiated and VY conducted interviews to analyze operator actions and RHR system performance. The inspector interviewed the operating personnel who were on-shift during the event, verified that plant procedures were in use and adhered to in initiating SDC, and independently confirmed the licensee's assessment of the event. An increased priority was placed on the completion of a previous engineering service request made to YNSD to determine the root cause of the pressure spikes that have now occurred in both subsystems. A review of industry events evaluated similar group isolations at the Pilgrim and FitzPatrick nuclear power plants. NRC review of VY actions taken to correct this condition are documented in NRC Inspection Report 93-19. The LER adequately addressed the reportability criteria of 10 CFR 50.73, assessed previous occurrences, and documented appropriate corrective actions.

Periodic and Special Reports

Vermont Yankee submitted the following periodic and special reports which were reviewed for accuracy and found to be acceptable:

- Monthly Statistical Reports for November and December, 1993
- Report of Fuel Failure Status and Parameter Trends for December 1993. For this period, VY determined that no fuel failures have yet occurred based on the offgas release rate and percent recoil. Currently, VY estimates that offgas levels will decrease over the Cycle 17 due to depletion of tramp uranium. A review of reactor coolant isotopic concentrations indicated expected variations due to the power changes that occurred this period. No abnormal trends were identified. During the changes in reactor power to support maintenance (Section 2.2), rod pulls were conducted using pre-established guidelines to minimize fuel element stress. This report contained appropriate detail.

7.0 MANAGEMENT MEETINGS (30702)

7.1 Preliminary Inspection Findings

Meetings were held periodically with VY management during this inspection to discuss inspection findings. A summary of preliminary findings was also discussed at the conclusion of the inspection in an exit meeting held on January 21, 1994. No proprietary information was identified as being included in this report.

7.2 Enforcement Conference

On December 2, 1993, an enforcement conference was held at the NRC:RI office with VY representatives to discuss three issues: core spray suction strainer sizing, alternate cooling tower silting and surveillance testing; and seismic anchoring of Class 1E electrical switchgear (NRC Inspection Reports 93-21 and 93-29). A list of meeting attendees and copies of overhead slides used in the VY presentation are contained in Attachments A and B to this report, respectively.

ATTACHMENT A

LIST OF ATTENDEES

ENFORCEMENT CONFERENCE, DECEMBER 2, 1993

NRC Attendees

W. Lanning, Deputy Director, Division of Reactor Projects (DRP)
W. Hodges, Director, Division of Reactor Safety (DRS)
J. Linville, Chief, Projects Branch No. 3, DRP
D. Dorman, Project Manager, Project Directorate I-3, Nuclear Reactor Regulation
H. Eichenholz, Senior Resident Inspector, Vermont Yankee
S. Chaudhary, Senior Reactor Engineer, DRS
D. Holody, Enforcement Specialist, Office of Enforcement (OE)
K. Smith, Regional Counsel
E. Kelly, Section Chief, DRP
B. Whitacre, Reactor Engineer, DRP

Licensee Attendees

J. Pelletier, Vice President, Engineering
G. Cappuccio, Mechanical Engineering Supervisor
T. Watson, Maintenance Manager
J. Hoffman, Yankee Nuclear Services Division Engineer

ATTACHMENT B

ENFORCEMENT CONFERENCE SLIDES

DECEMBER 2, 1993

ENFORCEMENT CONFERENCE AGENDA

- Seismic Qualification of Electrical Switchgear
- Alternate Cooling Tower Deep Basin Silt
- Core Spray System Suction Strainers
- Focus On:
  - Causes and Full Impact on Plant Safety
  - Short and Long Term Corrective Actions
- Overall NRC Concerns:
  - Design Basis Documentation
  - Industry Experience Reviews
  - Depth and Quality of LER's
- Conclusion

VERMONT YANKEE
NUCLEAR POWER CORPORATION

SEISMIC QUALIFICATION OF ELECTRICAL SWITCHGEAR

During an engineering walkdown of out-of-service safety related electrical equipment, an apparent lack of adequate anchorage was found on 4160 volt Bus 4 in the Switchgear Room.

Traditional engineering analyses were unable to conclude that the component would be operable for a design basis seismic event.

IMMEDIATE ACTIONS

Notify plant management and develop a fix.

Inspect safety related Busses 3, 8 and 9 in the Switchgear Room.

SHORT TERM CORRECTIVE ACTION

Inspect all other safety related switchgear and busses for positive anchorage.

FINDINGS

Only Busses 3, 4, 8 and 9 were found to be without positive anchorage.

All were repaired before returning the equipment to service.

VERMONT YANKEE
NUCLEAR POWER CORPORATION

SAFETY SIGNIFICANCE

The industry seismic experience database was reviewed to determine if data existed relative to the performance of unanchored switchgear during seismic events.

Nine examples were available.

The equipment was operable and would have functioned properly in a design basis seismic event.

WHY WERE THE INSPECTIONS BEING PERFORMED?

Engineering has been walking down out-of-service electrical equipment for the past several outages as part of its preparation and familiarization of engineers for the A-46 program. Based on the issuance of the NRC's final SSER and the GIP, the A-46 project is underway and planned for completion by December 1995.

The walkdowns were part of an overall effort of preparation for SQUG, along with drawing reviews, file searches for old documentation and SQUG training for engineers.

These four busses are the first time any equipment has been found without positive anchorage.

VERMONT YANKEE
NUCLEAR POWER CORPORATION

WHY WERE THE BUSSES UNANCHORED?

Contacted three Ebasco personnel who were directly involved in Vermont Yankee construction activities.

They all concluded that the busses were intended to be anchored.

The process required the electrical installation contractor to develop an installation package based on vendor supplied information.

Original construction deficiency localized to the Switchgear Room.

COULD THE CONDITION HAVE BEEN FOUND SOONER?

NRC issued I&E Notice 80-21 in May 1980 alerting licensees to conditions found at some SEP plants.

Review of VY and YNSD records found no documentation discussing response to I&E Notice. Industry practice at the time did not formally document response to Notices.

In that time period, there were many seismic upgrades and installations at Vermont Yankee. No recollection or documents reporting finding of unanchored equipment.

Quite likely, Notice may have been informally dispositioned based on knowledge of personnel at the time regarding observed plant condition.

Switchgear was not identified in the Notice as equipment where problems were found at other plants.

A 100 percent inspection of equipment identified in the Notice would not have identified this condition.

VERMONT YANKEE
NUCLEAR POWER CORPORATION

OTHER CONSIDERATIONS

ORIGINAL CONSTRUCTION CONDITION

Associated with original plant construction.

Localized to the Switchgear Room.

No other electrical components with this installation detail.

Identified through Engineering activities.

Part of a proactive effort to develop plant familiarity to address A-46 issue.

RAPID RESPONSE

Reported immediately to NRC.

Followup briefing with Resident Inspector.

All fixes installed within three days.

Inspected all other safety related switchgear and busses.

Engineering evaluation conducted on as-found condition to assess operability.

CONCLUSIONS

The busses were operable and satisfied Technical Specifications requirements.

The equipment satisfied FSAR seismic design basis requirements.

Safe plant operation was assured, even though the design basis was not properly translated into installation requirements.

LER will be updated to report later information.

VERMONT YANKEE
NUCLEAR POWER CORPORATION

ALTERNATE COOLING TOWER
SILT ACCUMULATION

As a part of the cleaning of the Deep Basin, an as found silt survey was performed. At this time the level of silt accumulation in the Deep Basin Pit containing the 24" alternate cooling suction pipe was found to be above the top of the pipe.

The cause of the event was "a lack of understanding of the alternate cooling tower subsystem and the basis and scope of the once per cycle inspection". The lack of definitive inspection criteria failed to provide assurance that ongoing and continued operability of this system was maintained.

VERMONT YANKEE
NUCLEAR POWER CORPORATION

SAFETY SIGNIFICANCE

"The objective of the Alternate Cooling System is to provide an alternate means of shutdown cooling in the unlikely event of a failure of the Vernon Dam."

- Dam failures are predominantly a result of seismic events.
- Vernon Dam failure would not occur well beyond the design basis for the Plant.

IPE ASSESSMENT

- Core damage frequency from all causes is about 4.3 X 10E-6.
- Core damage frequency increases to about 6.5 X 10-6 if the Alternate Cooling System is not included.

CORRECTIVE ACTIONS

SHORT TERM

- Thoroughly cleaned the Deep Basin and Associated pit.
- Performed an extensive review of the Alternate Cooling System to determine if other problems existed.
    - Mechanically cleaned the 24" Alternate Cooling Main Supply Header from the Deep Basin to the Isolation Valve located in the Plant.
    - Changed valve lineup, vented, filled and chemically treated 24" diameter Alternate Cooling Header to prevent growth of MIC.
    - Repaired two of four Cooling Tower Distribution Manual Valves that were sticking open.

SHORT TERM

CONTINUED

- Underwater inspection in the Fall and Spring to determine the rate of silting that occurs over time.
- Definitive maintenance inspection criteria for silting of the Deep Basin and Suction Header and the development of a specific P. M. Program for inspection and cleaning by 12/30/93.

CORRECTIVE ACTIONS

LONG TERM

- A review of T.S. systems to verify that the inspection and testing performed meets the operability criteria as required by the license by 12/31/93.

- Engineering Study Re: Pipe Modification.

- Increased Housekeeping Requirements.

VERMONT YANKEE
NUCLEAR POWER CORPORATION

CORE SPRAY SYSTEM STRAINER FOUND UNDERSIZED

- Background Information
  - G.E. Analysis (1986)
  - VY RHR System Strainer Replacement (1986)
  - Cause of the Event

- 93 Outage - Condition Found
  - BWR Owners Group Request
  - Siesmic Calc vs NPSH Calc
  - VY Took Immediate Corrective Action

VERMONT YANKEE
NUCLEAR POWER CORPORATION

CORE SPRAY SYSTEM STRAINER FOUND UNDERSIZED

- Safety Significance
  - Original As Found CS Strainer 7.8sq ft
  - G.E. Design Analysis 9.0sq ft
  - Estimated CS Strainer size 9.8sq ft
- Design is Conservative
  - Post Accident Pressure
  - Transport
  - New LOCA - Less Core Cooling Required
- For the majority of accidents involving the Core Spray suction strainers are adequate. Only for the very unlikely DBA LOCA would the system not meet the design.
- Our opinion, the Core Spray System was operable due to the conservative design.

VERMONT YANKEE
NUCLEAR POWER CORPORATION

SHORT TERM CORRECTIVE ACTIONS

- Replaced strainers. Updated and reviewed applicable calculations/analysis.

- Measured and produced drawing for the HPCI and RCIC suction strainers. Verified they meet design requirements.

- Re-verified the RHR strainer size (replaced in 1986).

- Inspect Torus bottom for debris.

- Reviewed documents for similar events.

- Visual inspections were performed on all strainers.

LONG TERM CORRECTIVE ACTION

- Revise procedures for preparation of Engineering calculations by May of 1994.
- Review a sample of Engineering
- Training will be given to all Engineering personnel by 12/31/94.

VERMONT YANKEE
NUCLEAR POWER CORPORATION

FACTORS FOR CONSIDERATION

- Vermont Yankee identified as a result of formal effort to gather information for the BWROG.
- Immediate action was taken to correct the problem.
- Appropriate broadly focused long-term corrective actions are underway.
- This problem would not have been discovered by routine effort.

OVERALL NRC CONCERNS

- Potentially More Widespread Problems With Design Basis Documentation

- Weak Industry Experience Reviews, Particularly for Events Prior to 1991

- Depth and Quality of LER's

  - Insufficient Detail in Analysis of Safety Significance

  - Narrow Focus of Long Term Corrective Action

VERMONT YANKEE
NUCLEAR POWER CORPORATION