U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No.	94-11
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:	April 3 - May 14, 1994
Inspectors:	Harold Eichenholz, Senior Resident Inspector Paul W. Harris, Resident Inspector John T. Shedlosky, Project Engineer

Approved by: Richard J. Conte — 6/9/94
Richard J. Conte, Chief, Reactor Projects Section 3A — Date

Scope: Station activities inspected by the resident staff this period included Operations, Maintenance, Engineering, Plant Support, and Safety Assessment and Quality Verification. Backshift and "deep" backshift including weekend activities amounting to 26.5 hours were performed on April 6, 10, 20, 23, 30 and May 5, 6 and 12. 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. Enforcement discretion was exercised for the failure to properly conduct reactor coolant conductivity sampling (Section 6.1, LER 94-06).

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EXECUTIVE SUMMARY

Vermont Yankee Inspection Report 94-11

Operations

Vermont Yankee continues to monitor reactor coolant leakage (0.03 gph) past core spray isolation valves into low pressure core spray piping. Deficient lighting in the reactor building and a concern regarding the piping configuration of two air monitoring systems required resolution by Vermont Yankee. Very good control room operator performance was observed in response to two unrelated events: a reactor pressure spike on April 17, and a turbine trip/reactor scram on April 10. Vermont Yankee implemented procedure controls to administratively control the use of overtime by plant personnel.

Maintenance

Although a lack of programmatic depth in the lubrication oil program was identified, the replacement of unacceptable oil in the residual heat removal service water pumps was well controlled. Effective corrective maintenance to replace and adjust shaft packing was conducted on the service water pumps. Good staff performance was demonstrated during the investigation and repair of a failed motor operator in the core spray system; Vermont Yankee identified that additional operator training in the area of reportability determinations was necessary. Although administrative guidance regarding the planning and conduct of troubleshooting have not been implemented, other procedural controls are in place. Good attention to detail was demonstrated during surveillance testing. The surveillance procedure for the conduct of emergency diesel generator testing was not well developed from a human factors perspective.

Engineering

Corrective actions to improve the control of maintenance effecting primary containment integrity and Technical Specifications involving the hydrogen/oxygen monitoring system were considered effective. Licensee initiatives to effectively use information generated by the Individual Plant Examination are occurring. Inaccurate scram timing data was generated by the plant process computer during the April 10, reactor scram due to deficiencies in the software program. Vermont Yankee was effective in resolving questions related to equipment performance during degraded grid conditions by the installation of a modification to load shed the cooling towers during accident conditions assuring continuity of Class 1E power to safety systems.

Plant Support

A security exercise was conducted to demonstrate heightened facility security and to test the capabilities of the Security Department. The Fire Brigade was professional and calm during their effective response to a small fire on the "B" emergency diesel generator; the post-fire critique was not sufficiently thorough. Vermont Yankee continues to assess improvements in

(EXECUTIVE SUMMARY CONTINUED)

the diesel room smoke detection system due to the recurrence of spurious actuations during diesel operation. A fire protection audit was of very good detail. The VY Observation Program was reviewed and it appears to be a good licensee initiative. Vermont Yankee will consider installing enhanced communications circuits to better alert personnel in the diesel generator rooms during diesel operation.

Safety Assessment and Qualify Verification

Previously identified corrective actions were credited to preclude recurrence of a non-conservative setpoint range assigned to the rod block monitor system. The use of a Chemistry Department training activity to conduct a self-assessment of surveillance activities identified a missed Technical Specification surveillance test and reflected positively on the questioning attitude of the involved staff.

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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 (VY) continued full power operations for most of this inspection period. On April 10, during weekly turbine valve testing, a turbine trip and subsequent reactor scram occurred due to high water level in the "C" moisture separator. Safety and balance of plant systems operated as designed and all control rods inserted. A primary containment isolation system actuation occurred. Control room operators monitored and responded to the plant transient, reviewed and implemented appropriate procedures, and placed the plant in a safe condition (Section 2.2). The scram on April 10 was the first in 34 months. On April 12, VY returned to full power operations.

On April 11, Mr. Greg Maret assumed the position of Operations Superintendent responsible for Operations and Maintenance Departments. Mr. Maret came from Yankee Nuclear Power Station where he managed the component removal program. His previous experience at VY included participation on several task forces, quality assurance audits, and the VY Nuclear Safety Audit Review Committee.

2.0 OPERATIONS (71707, 92901, 93702, 90712, 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 for the fuel pool cooling, residual heat removal service water (RHRSW), and emergency diesel generator (EDG) systems properly isolated equipment for maintenance. Operator logs were reviewed. Work orders were generated to resolve equipment operating parameters that were out of specification in the stack flow, augmented offgas, and area radiation monitoring systems. A review of the administrative controls associated with the entry and exit from TS LCOs, equipment tagouts, and Standing Night Orders identified no concerns. The inspector verified the control rod pull sequence to the displayed rod pattern, operability of dc electrical sources, and that operators were aware of plant activities which could influence safe plant operation.

This period, VY continued to monitor and trend core spray (CS) low pressure piping pressure. As described in NRC Inspection Report 94-08, reactor coolant pressure boundary leakage past a check valve (V14-13A) within containment and a normally shut, remotely operated, gate valve (V14-12A) caused a slow pressurization of the low pressure CS piping downstream of the "A" CS pump discharge check valve. Plant operators closely monitored and vented the system as required to maintain system pressures below 300 psig. A relief valve set at 375 psig protects the low pressure piping from overpressure up to 100 gpm leakage from the main coolant system. A pressure switch in the low pressure piping actuates at 300 psig and alarms in the control room. A second normally open system, isolation valve, V14-11A, is available for isolation, if required. The design pressure of this piping is 475 psig at 175 degrees F. No increase in temperature has been detected. Inservice and leak rate testing data for the two system isolation valves (V14-12A and 11A) identified no degradation or unacceptable conditions. A trend of system pressure indicated a pressurization rate of approximately 10 psig/hour corresponding to a leakage rate of approximately 0.03 gallons/hour. Based on an engineering evaluation, VY determined that the current leakage rate was consistent with the leak rate obtained during testing conducted during the previous refueling outage.

2.1.1 Plant Tours

Air Monitoring Design Configuration

The inspector observed that rubber hoses were installed in the pump suction and discharge lines for the reactor building (RB) ventilation air radiation monitor. Similarly, a hose was installed in the process line for the pump suction for the containment atmospheric radiation monitor. These configurations appeared to differ from that represented in the vendor technical manual and plant drawing (G-191165) because hose was used instead of pipe as represented in the drawings.

Both systems are designed to monitor air for particulate and gaseous radioactivity. In accordance with TS 3.6.C.2, the drywell containment atmosphere monitor (CAM) is required to be operable during power operation to sample the drywell to detect reactor coolant pressure boundary leakage. The system is non-safety related and isolates during accident conditions. The RB ventilation monitor provides signals to isolate the RB on high radiation conditions and to start the standby gas treatment system. The RB ventilation monitor also initiates a Group III primary containment isolation system (PCIS) isolation. This safety-related system mitigates the consequences of a refueling accident. Operability of the RB ventilation monitor, as described in TS 3.2.C and Table 3.2.3, is required when secondary containment integrity is required to be in effect. The inspector was concerned that failure of the rubber hoses would result in non-conservative radiation indications due to sampling dilution caused by air inleakage.

Based on an initial review of information contained in the maintenance planning and work control (MPAC) system, the inspector determined that the hoses had been installed for a number of years. A review of the design change process documentation confirmed that the hoses were not installed as part of a design change. A system walkdown confirmed that the hoses were sound. Corrective actions initiated by VY included drawing revisions to properly represent the system configuration, and an Environmental Qualification (EQ) assessment that subsequently identified no concerns. The licensee plans to review the MPCA system to assess whether changes are necessary to clarify the safety and TS classification of the two air monitoring systems. At the end of this inspection period, VY was confirming an adequate basis for hose installation and related preventive maintenance.

Deficient Plant Lighting

The inspector observed deficient lighting in the inner annular area of the torus near the reactor pedestal wall. The area was completely dark and challenged the identification of personnel, fire prevention, and plant safety concerns without temporary light sources. The Electrical Department foreman was informed, and the condition was subsequently corrected. The inspector concluded that the deficient lighting was of low safety significance, because the affected area contains no equipment, no combustible materials, and is not routinely accessed. The inspector was informed that the cognizant electrical maintenance staff was aware of the deficiency but as yet had not corrected the condition.

The inspector reviewed the April 1994, VY Observation Program findings to assess whether the deficient lighting in the torus was identified by this program. Although, the findings reviewed did not document this particular deficiency, other lighting observations were identified and placed within the MPAC system for correction. An assessment by the inspector of the VY Observation Program is documented in Section 5.3.4.

2.1.2 Reactor Pressure Spike

On April 17, during performance testing of the electric pressure regulator (EPR), a 17 pound (psi) reactor pressure spike occurred when the EPR setpoint was raised by one pound in accordance with the surveillance procedure (a one second setpoint change is expected to produce a one pound change in reactor pressure). The increase in reactor pressure from 1010 to 1027 psig caused an increase in reactor power from 100 to 105 percent and an automatic switch over to the mechanical pressure regulator (MPR). Vermont Yankee determined that the EPR setpoint motor drive mechanism failed resulting in an increased system response time. On April 22, the drive mechanism was replaced and satisfactorily tested.

Control room operators responded properly to the increase in reactor pressure and power. Vermont Yankee management conservatively concluded that continued long term plant operation using the MPR was unacceptable because failure of the MPR would result in a subsequent pressure transient. A high priority maintenance work order was initiated. Operations Department Night Orders provided guidance to increase operator awareness of the potential for a reactor pressure transient. Daily, plant management reviewed the status of corrective maintenance during the Plant Manager Meeting. This operational issue was appropriately resolved.

2.1.3 Reactor Protection System Walkdown

The inspectors conducted a walkdown of the reactor protection system (RPS). Analog trip cabinets, cable runs, and alarm displays were in good material condition. Plant procedures were used as inspection guides. Support structures for cabling and cabinets were properly installed, cleanliness internal and external to the cabinets was good, and relays were clean and free of discoloration. No flammable materials were identified in the vicinity of the RPS equipment.

Plant operator logs appropriately documented system operating parameters. Based on circuit drawings and equipment isometrics, configurations were accurately represented. No conditions were identified adverse to system operability.

2.2 Turbine Trip and Reactor Scram

On April 10, at approximately 98.5 percent rated power and while performing weekly turbine valve surveillance, a turbine trip and subsequent reactor scram occurred due to high water level in the "C" moisture separator. Control room operators verified all rods were fully inserted and actuation of primary containment isolation systems (PCIS) Groups 2, 3, and 5 occurred on low reactor water level. Emergency Operating Procedure (EOP) 3100, Reactor Scram, and support procedures were entered. Electrical bus transfer to the startup transformers occurred, as designed, to maintain power to the balance of plant and safety buses. The reactor scram, turbine trip, and transient to hot standby conditions occurred, as designed and were well controlled. No abnormal thermal performance was identified and the scram was properly reset.

Vermont Yankee determined that turbine testing caused a normal perturbation on the moisture separator water level, however, the level increase was not mitigated by either the normal or emergency dump valves. Their investigation identified that the normal and emergency moisture separator drain tank level controllers experienced instrument drifts resulting in slow and incomplete dump valve response. The "C" emergency level controllers experienced an operating band shift while the "C" normal level transmitter experienced a zero shift. The level control components were sent to the vendor for failure analysis and root cause determination.

Prior to plant startup, the Plant Operating Review Committee (PORC) reviewed the scram precursors, the cause of the scram, and equipment and operator performance. Corrective actions focused on the initiation of a significant Corrective Action Report (CAR) to investigate the adequacy of level controller and transmitter maintenance and to determine root cause. A second significant CAR was assigned to the Reactor and Computer Engineering (R&CE) Department to assess inaccurate scram data generated by the plant process computer (Section 4.3). The PORC also recommended augmented monitoring of moisture separator levels during plant startup, a review of operator aids and alarm response procedures, and an investigation of the cause of spurious alarms on the event recorder. Next refueling outage, the Instrument and Controls (I&C) Department plans to replace the level positioners and rebuild the air filter supply regulators. In addition, the double-disk, emergency dump valves are planned to be replaced with a new single-disk design to improve operating characteristics. Yankee Nuclear Services Division (YNSD) and the vendor were tasked to evaluate instrument loop data, in part, to assess the adequacy of the current design and its effect on the installation of the new steam turbine system planned for the March 1995 refueling/maintenance outage. Based on a review of the post-trip reports, the inspectors determined that good operator and plant performance occurred during this plant transient. The PORC had reasonable confidence that the cause of the scram was known and corrected. Plant performance data, indicated that reactor water level and pressure control, PCIS Group isolations, standby gas treatment, and electrical systems operated properly. On April 12, a safe ascension to power was conducted. An accurate 4-hour ENS notification to the NRC Operations Center was performed.

2.3 Primary Containment Isolation Group III

On April 20, a PCIS Group III isolation occurred due to spiking of one of the two refuel floor radiation monitors (RM17-453B). The Group III signal went to completion resulting in the automatic closure of drywell and suppression chamber purge and vent valves, automatic start of the standby gas treatment (SBGT) system, and isolation of the reactor building ventilation (RBV) system. Control room operators followed the alarm response procedure, verified system responses, and confirmed by process and area radiation monitoring that no abnormally high radiation condition existed on the refuel floor. Instrument RM17-453B was bypassed and declared inoperable. Vermont Yankee determined that this signal was invalid and caused by oxidized contacts on the monitor's function switch contacts. The contacts in the other three radiation monitors associated with this PCIS sub-system were cleaned. No additional spiking was observed.

Their initial determination did not consider that the PCIS Group III actuation of containment isolation valves was reportable, despite the invalid signal. Vermont Yankee was originally uncertain whether this event met the reportability requirements of 10 CFR 50.73(a)(2)(iv). Initially, they concluded during the LER reportability review of the Potential Reportable Occurrence Report that the event was not reportable because the subject engineered safety feature (ESF) involved the RBV and SBGT systems, and the actuation was caused by an invalid signal. Vermont Yankee submitted the 30-day report after clarification was obtained from NRC staff (AEOD - Analysis and Evaluation of Operational Data).

Plant procedure AP 0156, Notification of Significant Events, was considered by the inspector to adequate. The Operations Department provided amplifying information in a Night Order to ensure that all control room personnel were aware of the need to report the PCIS Group III actuation.

2.4 Closed (URI 93-14-04): Controls for Outage Overtime

Previous NRC staff observations regarding a lack of administrative controls and conformance to NRC guidelines and policy statements involving the control of overtime were documented in NRC Inspection Report 93-14 (URI 93-14-04). Vermont Yankee responded in writing to the NRC on September 27, 1993, describing their actions to administratively control overtime during outages. Therein, VY committed to review lessons learned from the 1993 refueling/maintenance outage and implement procedure requirements assuring the consistent application of overtime for personnel conducting or responsible for the performance of maintenance, repair, modification, or calibration of safety-related structures, systems, or components.

Revision 3 to AP 0894, Shift Staffing/Overtime Limits, was implemented to assure that the use of overtime for plant employees contributes to safe plant operation and maintenance. The guidance contained in NRC Generic Letter 82-12, Nuclear Power Plant Staff Working Hours, was adequately incorporated into AP 0894, and the principles described in the NRC Standard Technical Specifications were used as a reference. The limits assigned are: (1) 16 hours straight; (2) 16 hours in any 24-hour period; (3) 24 hours in any 48-hour period; (4) 72 hours in any 7-day period; and, (5) a break of at least eight hours between work periods. Revision 3 also assigns program responsibility to individual department supervisors and requires approval by the Plant Manager prior to any situation exceeding the overtime limits. Documentation of a request for work in excess of established limits or the failure to meet requirements is required. Currently AP-0894, Rev. 3 does not define a normal working day or working week, however, the inspector verified that the plant operating and maintenance staffs work 8-hour days five days a week. Personnel on rotating shifts (Operations, RP, and Chemistry) work similar 8-hour days (not including turnover) up to seven days straight. Based on acceptable implementation of corrective actions to resolve this issue described in NRC Inspection Report 93-14, this item is closed.

3.0 MAINTENANCE (62703, 62702, 37551, 71707, 61726, 92902)

3.1 Maintenance

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.

3.1.1 Lubrication Oil Analysis Program

The inspector reviewed the implementation of preventive maintenance procedure DP-0213, Rev. 0, Lubrication Oil Analysis Program. Currently, VY has yet to accumulate a sufficient number of lubrication oil analysis results to effectively trend oil performance or assess pending degradation. Nonetheless, VY identified that the particulate concentrations in the four RHRSW pumps exceeded acceptance criteria and the oil was changed to re-baseline oil data. This maintenance was well-controlled and the maintenance personnel were knowledgeable of program responsibilities. The first round of sampling also identified other plant components in which particulate concentrations exceeded normal ranges; maintenance work requests were initiated.

The inspector discussed the program elements, acceptance criteria, and technical bases with the cognizant Maintenance Department engineer. Personnel qualification and training requirements are proceduralized, and national standards were used to establish acceptance criteria. Vermont Yankee performs particulate oil analysis onsite and uses an offsite vendor to analyze for metals, water, viscosity, and other physical data. The inspector observed that although corrective action alternatives are identified in the procedure, the scope and timeliness were dependent on the cognizant engineer. Further, the alternatives were not based on oil sample results or the safety classification of the equipment being evaluated. This was demonstrated when the RHRSW oil samples were obtained on March 28, 1994, and analyzed on March 31; however, the oil was not changed until May 1994. The lack of programmatic depth represents a challenge for the consistent implementation of corrective actions based on lubrication oil analysis results and safety significance.

The inspector identified no concerns regarding the removal of RHRSW pumps from service for corrective maintenance. The re-baselining of oil analysis and removal of oil that exceeded pre-established acceptance criteria justified the removal of the RHRSW pumps from service. System unavailability is trended by the Operations Department and the maintenance was discussed at the daily Plant Manager meetings. The pumps were sequentially removed from service for the corrective maintenance and each pump was returned within approximately 10 hours.

3.1.2 Service Water Pump Preventive Maintenance

During this inspection period, VY repacked and adjusted the packing for the four service water pumps (P-7-1A, B, C, and D). Reference to the vendor technical manual and guidance provided in OP 5201.03, Valve Packing Checklist, were used to supplement information required to be documented by the work order. This checklist, part of OP 5201, Safety System Valves, provided amplifying information regarding the number of rings, use of spacers, reference dimensions, and material condition of the valve stem and stuffing box. A revision to OP 5200, Safety System Rotating Equipment Maintenance, will be assessed by VY to include similar requirements assuring consistent documentation of as-found and as-left information.

Although a valve packing checklist was utilized for pump maintenance, the information recorded was accurate. The work instructions were also adequate to assure the successful completion of this service water pump corrective maintenance. Vermont Yankee considers packing replacement and adjustment within the skill of the trade. The inspector identified no specific personnel performance concerns during the maintenance and observed satisfactory pump performance following the maintenance.

3.1.3 Core Spray Motor Operator Valve Failure

During the conduct of routine quarterly in-service testing of the "A" CS sub-system at 8:38 a.m. on April 12, the minimum flow valve CS-5A would not operate when the control room operator placed the control switch to closed. Plant operators opened the electrical supply breaker for the valve when it was reported that smoke was coming from the motor. Control room operators declared the "A" CS sub-system inoperable (7-day allowed outage time per TS 3.5.A.2) at 8:40 a.m. and notified the NRC Operations Center at 9:03 a.m. that the event was a 4-hour non-emergency in accordance with 10 CFR 50.72(b)(2)(iii). Subsequently, at 10:17 a.m., the ENS notification was withdrawn when further evaluation with the Technical Superintendent determined that this was not a reportable event (i.e., only one train of a redundant two train system was affected). To address VY management and inspector concerns about this inadvertent

ENS notification, the Assistant Operations Manager issued Training Change Request No. 94-0156 for the Training Department to conduct additional reportability training in Licensed Operator Requal Cycle 94.2. Repair activities resulted in returning the "A" sub-system to operable status at 10:30 p.m. on April 14.

The CS-5A valve is a 3-inch normally open Walworth gate valve with a Limitorque SMB-000 size motor operator. The valve is located in the RB, at elevation 213" of the "A" ECCS corner room. The valve closes automatically after the pump starts and provides minimum flow. Plant design change activity (TM 92-56 and EDCR 92-402) replaced the original 2 ft-lb motor with a 5 ft-lb motor to ensure that the valve motor develops sufficient torque to achieve the higher thrust settings, as determined from review of the MOVATS database and NRC Generic Letter 89-10 issues.

Regarding the repair activities, the Maintenance Department issued Work Order No. 94-3239 to perform troubleshooting and repair of the valve. Maintenance procedures associated with motor control center (MCCs), motor operators, and diagnostic testing of motor operated valves were used. The investigation determined that the motor shaft was binding and therefore would not turn. Other than a motor problem, no valve or valve operator performance concerns were identified. Because the VY design uses 300 percent motor overload heaters, there is no effective motor protection for this type of electrical problem to de-energize the power to the valve. Electrical components within the MCC were replaced either as a precaution or due to damage from the overload. The issue of the size of the overload will be addressed in a VY "significant" Corrective Action Report (CAR No. 94-24) to be developed. Because it was not evident as to the cause of the motor binding, the motor was transported to its offsite repair vendor for analysis and determination of failure cause. In turn, VY learned that the probable cause of motor binding was due to the failure of one motor bearing that appeared to have been installed incorrectly during motor manufacturing. Recommendations for further corrective actions due to the bearing failure are to be included in the CAR. Both electrical and MOVATS testing were included in the post-maintenance testing. The inspector verified that predictive/preventive maintenance was performed on this valve during the 1993 Refueling Outage. Experience with motor operated valves at VY and the results and the investigation supports the initial Maintenance Department conclusion that the failure of the valve operator was an isolated case of equipment failure.

A Quality Assurance (QA) surveillance was conducted during the VY investigation and repair activities. This surveillance reviewed, in part, control of contractor services and test equipment, performance of work in accordance with approved procedures, replacement material quality, and the authorization and documentation of work. The inspector noted that the completed Surveillance Report (No. 94-20) was of sufficient detail to observe the very broad perspective that YNSD QA personnel apply to important safety related maintenance.

With the exception of the unwarranted 10 CFR 50.72 notification by VY, the performance of operating and maintenance personnel was good, in that, corrective actions were timely, methodical, and well thought out. Proper safety concern was demonstrated in ensuring that the valve and its operator were not removed so as to maintain containment integrity. Good management oversight and involvement was evident as the investigations and repairs were conducted within the allowed TS limits. Good QA involvement was evident.

3.1.4 Review of Troubleshooting Activities (RI TI 94-01)

The inspector reviewed the licensee's administrative controls of plant equipment during troubleshooting activities. This review was performed because several incidents at other reactor facilities resulted from inadequate control of troubleshooting. These events occurred when personnel failed to fully understand the impact of their actions, and in other cases poor communication and documentation were causal factors that hindered the identification of a failed component.

The inspector reviewed the licensee's administrative procedures for the control of plant equipment, and the control of maintenance, for maintenance work orders and for temporary modifications and equipment safety tagging. The inspector discussed the application of these requirements in the context of troubleshooting activities with licensee representatives and examined the plant records of completed and in-progress troubleshooting.

The licensee has not provided specific instructions to guide the development of troubleshooting activities. Troubleshooting is generally performed within the controls of the work order system. Administrative procedure AP 0021, Work Orders, was recently revised to require that specific bounds be established for troubleshooting and that work orders clearly distinguish between troubleshooting and repair activities. This change was made in response to a licensee audit finding when a repair crew failed to revise a work order that authorized troubleshooting for the repairs needed to correct the problem.

Most troubleshooting plans are developed by personnel from the Instrumentation and Control (I&C) or Maintenance Departments. Depending on the nature of the problem, a foreman or engineer, the two working together or in some cases a team will develop the troubleshooting activities. These are placed within a work order. Additional controls of activities such as lifting electrical leads, installing electrical or mechanical jumpers or removing circuit boards are provided by the administrative controls of AP 0020, Control of Temporary Modifications. The inspector found that the troubleshooting plans recognize general "skill of the trade" but can be enhanced at the discretion of the originator. For example, specific reference to exact location were data is taken may be included. The troubleshooting instructions are generally not detailed step-by-step procedure instructions, but provide meaningful instructions to gather appropriate information for corrective actions or root cause failure analysis. Two examples of work in progress concerned performance problems with the recirculation pump motor generator set voltage regulators and also a recent failure of a core spray valve operator motor.

Although the licensee has not provided specific guidance on developing a troubleshooting plan, both the Maintenance and I&C Departments have pre-printed troubleshooting work orders within the data base used to manage work orders and equipment history. These pre-prints may be further modified by the person developing the troubleshooting work order.

The Shift Supervisor (SS) is responsible for releasing equipment for troubleshooting as with any other work order. The SS provides the final check on the impact of the troubleshooting activity and may specify additional requirements. This has not always resulted in desired results. For example, during the investigation of a reactor relief valve bellows leakage alarm on March 17, an emergency work order (94-02263) was used to investigate the issue. A temporary modification was not used to connect pressure instruments to the bellows leak detection line because personnel reasoned that the valve (RV-2-71C) was declared to be inoperable. Personnel did not recognize the significance of the instrument line as a primary containment boundary, and the safety class of the pressure switch (PS-2-71C) was incorrectly stated in the MPAC system (NRC Inspection Report 94-09, Section 3.1.3). Additionally, test personnel failed to comply with the shift supervisors instruction that the instrumentation be removed before the end of the day shift (Reference Vermont Yankee Significant Corrective Action Report CAR 94-018). A second example of a weakness in procedural controls was during a surveillance test of the standby liquid control system. Personnel deviated from an approved procedure to investigate the potential for discharge check valve leakage without executing a procedure change to reposition valves (Reference NRC Inspection Report 94-01, Section 3.1.4).

In conclusion, although the licensee has not provided detailed guidance on planning the troubleshooting process, procedural controls are in place at the station to control activities involving safety related equipment and to protect personnel safety. The success that the licensee has experienced in the troubleshooting process is partly due to the experience level of the personnel involved. Instructions for troubleshooting are provided within work orders. These work orders are approved by the SS who maintains authority over the status of safety related equipment.

3.2 Surveillance (61726)

3.2.1 Surveillance Observations

The inspector reviewed procedures, witnessed testing in-progress, and reviewed completed surveillance record packages. The surveillances which follow were reviewed and were found effective with respect to meeting the safety objectives of the surveillance program. The inspector observed that all tests were performed by qualified and knowledgeable personnel, and in accordance with VY TS, and administrative controls (Procedure AP-4000), using TS approved procedures.

- OP 4337, Rev. 27, Reactor Water Level ECCS Initiation Functional Test

- OP 4127, Rev. 6, John Deere Diesel Generator Surveillance

On May 1, during the monthly operational surveillance of the "John Deere" diesel (JDD), the Auxiliary Operator (AO) performing the surveillance observed that the JDD day tank was not filling as required. Although the fuel oil transfer pump was operating, the make-up rate to the day tank was exceeded by the fuel oil consumption rate of the diesel. The control room was informed in a timely manner of the abnormal condition, and the diesel secured and declared inoperable. Vermont Yankee identified that the backflow prevention device in the fuel oil supply line failed preventing adequate fuel oil flow. The inspector concluded that the AO demonstrated good attention to detail during this surveillance.

- OP 4336, Rev. 16, Reactor Vessel Shroud Level/Containment Spray Low Water Level Interlock Functional Test

The inspector observed the conduct of this surveillance in the control room and reactor building. Accurate communications between the I&C specialists located in the control room and the field were demonstrated when relays and alarms were verified operational during the conduct of the surveillance. During the pre-system walkdown to confirm that the field conditions matched surveillance prerequisites, the I&C specialist questioned the environmental qualification (EQ) of thread sealant used on Rosemount transmitters LT-2-3-73A/B. The cognizant I&C, EQ, and electrical engineers reviewed this issue and verified that the configuration was acceptable based on their review of maintenance records, a previous non-conformance report (NCR 90-18), and NRC Inspection Report 90-15. The use of teflon for thread sealant was acceptable to prevent failure of Rosemount transmitters during a high energy line breaks; however, for each application, the EQ was dependent on cumulative radiation exposure. The identification of this potential EQ concern demonstrated good attention to detail by the I&C specialist during his pre-surveillance equipment walkdown.

3.2.2 Emergency Diesel Generator Surveillance

On April 25 and 26, VY conducted monthly surveillances of the "A" and "B" EDGs using a slow start sequence. This start technique allows the EDG to automatically ramp to an initial speed of approximately 400-500 rpm followed by operator action to slowly and deliberately raise EDG to the rated speed of 900 rpm. Prior to the April tests, fast starts were performed in which the EDG automatically ramped to rated speed, typically within 10 seconds. The slow start sequence was permitted following issuance of License Amendment No. 138 dated March 22, 1994. The Safety Evaluation issued with Amendment No. 138 documented previous regulatory guidance involving slow starts and cited industry experience that demonstrated reduced engine wear and improved EDG reliability and availability. Fast starts will be performed every six months.

Plant operators followed procedure OP 4126, Rev. 33, Diesel Generator Surveillance, performed required log keeping, and informed the control room of EDG operating conditions. The Operations Planning Group and the Senior Operations Engineer observed the start of the EDG and provided insight into expected performance. Good coordination and management oversight was observed during the diesel start sequence and when the speed switch 810 rpm setpoint was tested. The SS conducted a pre-job brief and discussed the slow start sequence, precautions, and prerequisites. Fire brigade response to an alarm and small fire on the "B" EDG during the surveillance is discussed in Section 5.3.1.

The inspector reviewed OP 4126 and observed that the sequence of procedure steps to test the 810 rpm centrifugal speed sensing switch (SSW) was not developed well from a human factors perspective. For example: (1) instructions for the SSW test are provided for the 810 rpm (breaker permissive) and 1035 rpm (overspeed) setpoints; however, instructions are not provided regarding the deletion of steps when the overspeed test is not performed; (2) the procedure subsection for the SSW test restores the EDG to standby readiness; however, at this point in the surveillance, eight more hours of diesel operation were required to satisfy operability and inservice testing requirements; (3) double verification of a valve lineup deviation is required in the SSW subsection; however, similar control is not required during the slow start sequence when the identical deviation is also implemented; and, (4) the surveillance data sheet assumes the completion of steps that were not performed. These observations were discussed with the SS prior to the conduct of "B" EDG surveillance test on April 26, and with the Operations Manager prior to the post-maintenance/operability test on April 27.

Although the procedure steps associated with the testing of the SSW switch were not well sequenced, plant operators safely performed the surveillances. This indicated that the pre-job brief was effective and that the operators were knowledgeable of the surveillance requirements. This represented another situation (see also section 3.1.2), in which, middle management placed high reliance on plant personnel to perform a procedure that was not easy to implement from a human factors perspective.

4.0 ENGINEERING (71707, 37551, 92901, 40500, 37700)

4.1 Closed (URI 93-02-02): Primary Containment Integrity Concerns

Previous NRC staff observations regarding primary containment integrity concerns were documented in NRC Inspection Report 93-02 (URI 93-02-02) involving the ambiguous wording associated with TS 4.7.A.3 and TS Table 4.7.2.b, and the lack of administrative controls to assure containment integrity during the conduct of maintenance. Vermont Yankee assigned a Commitment Tracking System item to resolve the concern.

In regards to the first concern, VY concluded that an immediate change to TS was not required. The licensee determined that valves VG-29A/B and VG-30A/B were adequate isolations to assure primary containment integrity and worker safety. This was also in conformance with TS requirements (TS Table 3.7.2.b) that requires one of two remote-operated solenoid valves (VG-109-75, A-D) to be operable to isolate the hydrogen/oxygen (H2/O2) monitors. Further, these valves are not required to be leak rate tested and are in 1-inch diameter sampling lines. The H2/O2 system is considered an extension of primary containment. The licensee acknowledged that TS clarifications and improvements in the area of leak rate testing may be warranted as they continue to disposition recommendations from the Appendix J Program assessment recently conducted (see below).

In regards to the second inspector item, VY plans to update the MPAC system and administrative procedure AP 0021, Work Orders, to provide further assurance that primary containment integrity would be maintained during maintenance. These actions will identify valves credited in the Appendix J Program thereby assuring that primary containment integrity is maintained. Other procedures have been identified by VY and will undergo revision to assure appropriate control.

The above actions and others were initiated to resolve both NRC (Inspection Reports 92-12 and 93-02) and VY self-identified program weaknesses (PROs 93-26 and 94-09, CAR 92-13, and LER 93-03). A licensee initiative to conduct an independent assessment of the leak rate testing program is expected to resolve the many outstanding issues associated with the Appendix J Program. This effort will be independently conducted by a contractor. The effort will also include a review of the design and licensing bases, testing requirements, and administrative procedures. The licensee expects that this review will result in potential changes to the TSs, Appendix J testing exemptions, and a revision to the Appendix J Leak Rate Testing Program. These actions are expected to be completed by the March 1995 refueling/maintenance outage.

As a result of the contractor review, VY identified primary containment penetrations sealed with Type B gasketed flanges that were not tested in accordance with Appendix J Program requirements. The as-found piping configurations precluded Type B testing and the current program does not acknowledge the subject flanges. The seven penetrations were found in the primary containment atmospheric control and transverse incore probe systems, and in the turbine exhaust lines for the high pressure coolant injection and reactor core isolation cooling systems. Vermont Yankee has preliminarily determined that, although local leakages are unknown, they have reasonable confidence based on previous Type A integrated leak rate tests that the penetrations will prevent unacceptable containment leakages during the bounding design basis accident. Further, maintenance has not been performed on these penetrations to disturb their previously demonstrated containment integrity capability. Vermont Yankee has currently concluded that this issue is not reportable, tracks this item by PRO 94-39, and continues to assess the findings from their independent assessment.

Vermont Yankee has assigned appropriate commitments to resolve and improve their implementation of the Appendix J Program. Integration of outstanding issues was also reviewed by a NRC: Region I (NRC:RI) specialist (NRC Inspection Report 94-03) and found acceptable. The identification of the potential programmatic weakness involving inadequate Type B testing demonstrated the thoroughness of the on-going Appendix J Program assessment. This item is closed.

4.2 Individual Plant Examination

On December 21, 1993, in response to Generic Letter 88-20: Report on the Individual Plant Examination for Severe Accident Vulnerabilities, VY submitted its completed Individual Plant Examination (IPE) Report to the NRC. A briefing by cognizant YNSD representatives was conducted for members of the plant and engineering staffs on April 6. At this meeting a summary of the completed IPE and suggestions on how to apply the report to daily activities was provided, as well as facilitating questions and answers on the subject. The YNSD presentation provided insights as to: (1) using the report in an off-the-shelf manner; (2) describing further uses; (3) classes of problems solved by the report, (4) and provided twelve case studies that demonstrated its use. The inspector learned that this meeting occurred as a result of the request of the Vice President, Operations.

Prior to this meeting, VY issued a Service Request (No. 94-24) on March 11 that directed YNSD to perform a risk importance ranking of plant systems using the IPE. Further, it was requested that the written report, due to be completed June 1, would include a definition of risk importance measures and modeling assumptions that affect the usefulness of the rankings. The Engineering Director, subsequent to attending the site briefing, issued a memorandum to the Vice President, Engineering that raised a number of policy issues associated with the IPE. Specifically, concerns about maintenance of the report as a living document, the need to designate the organization entity responsible for the application of the IPE, the development of training programs and their implementation, and how it will be used were documented. The inspector also reviewed an internal VY memorandum, dated April 7, which specified that it was the request of the Vice President, Operations that YNSD evaluate the need to update the IPE following the 1995 Refueling Outage. This update would reflect changes made to the facility since the initial IPE and industry developments. Vermont Yankee stated that it was their expectation that YNSD will maintain an awareness of industry and regulatory activity in this area and advise them as necessary.

The inspector concluded from review of available documentation and discussions with both VY and YNSD representatives that keen interest exists to use the IPE methodology and report as a valuable resource and tool in support of safe plant operations. It is notable that important policy questions about the future use of the IPE are being identified for resolution by both line and senior management.

4.3 Control Rod Scram Timing Analysis

Immediately following the April 10 reactor scram, control room operators and Reactor and Computer Engineering (R&CE) personnel determined that the control rod scram timing data was erroneous and contradictory, because negative accelerations, data gaps, and too many extrapolations existed. A significant CAR was assigned to resolve this issue and vendor communications occurred. In parallel with these activities, scram times were manually calculated verifying that TS requirements were met. No adverse trend was identified. The computer-generated data was evaluated by the vendor; the problem was reproducible; and the software was corrected. Root cause was that the control rod program did not meet design specifications for purging scram data from the previous day.

The scram time software at VY is unique and not installed at any other nuclear facility. Prior to installation, the vendor (Nuclear Utilities Services, Inc.) validated the code and performed site acceptance testing using a function generator output. Vermont Yankee procedure OP 0452, Process Computer Updating, was implemented to control the software changes. Surveillance procedure OP 4424, Control Rod Scram Timing and Data Reduction will be revised as necessary. Vermont Yankee will consider a Quality Assurance audit of the software vendor. The Emergency Response Facility Information System (ERFIS) process computer and the subject software are not safety-related. This event appears to have been beyond the reasonable ability of VY to identify the existence of the software coding error because acceptance testing efforts could not detect the very unique and low probability circumstances that need to occur to manifest the April 10 conditions. The inspector concluded that VY adequately verified and validated the software change prior to installation into the process computer.

4.4 Temporary Modification - Cooling Tower Load Shed

Temporary Modification (TM) No. 94-006, Cooling Tower Load Shed, was installed on May 9 following approval by the PORC and Plant Manager on April 27 and 30, respectively. This TM removes non-essential loads from startup transformer T-3-1B during accident conditions which result in ECCS initiation. This ensures that the voltage on the Class 1E Safety Bus No. 4 recovers to a sufficient level to prevent inadvertent separation of this power from the preferred power source (i.e., the startup transformer) in the event that the accident is coincident with the minimum expected grid voltage. This issue was documented in NRC Inspection Report 92-81, as Unresolved Item 92-81-04: Degraded Grid Undervoltage Relays. The NRC conducted further review on this item during the period April 18-22, which will be documented in NRC Inspection Report 94-05.

The installation of the TM causes residual heat removal control logic relays to trip open 4160 Vac breaker 53, which effectively sheds cooling tower loads from transformer T-3-1B. The inspector verified that the TM was properly reviewed and approved, contained appropriate installation and test instructions, and contained a safety evaluation as required by 10 CFR 50.59 (a)(2). The inspector identified no installation concerns and verified that operators received training prior to the installation of the TM. A good level of detail was contained in the TM and a comprehensive 10 CFR 50.59 review was performed, both of which reflected positively on the contribution of the engineering organization to improving safe plant operations.

5.0 PLANT SUPPORT (71707, 90712, 92700, 90713)

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. Plant workers were observed to be cognizant of posting requirements and maintained good housekeeping.

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.

5.2.1 Security Exercise

On April 30, VY conducted a security exercise demonstrating heightened facility security. The exercise scenario involved a non-violent demonstration at the plant main gate that prevented normal access to the VY facility. Challenges to the owner controlled and protected areas were planned and compensated. Response strategies were pre-planned, coordinated with the local law enforcement agencies, and based on previous experiences. The inspector reviewed the scenario and contingencies, discussed strategy and manning with the Security Operations Supervisor and Security Project Manager, and concluded that the drill accurately represented an actual demonstration and tested the capabilities of the VY Security Department.

5.3 Fire Protection and Housekeeping

5.3.1 Fire on the "B" Emergency Diesel Generator

On April 26, during monthly surveillance of the "B" EDG, the inspector observed VY respond to a small fire on the external surface of the opposite-control-side (OCS) exhaust plenum. The resident inspector was in the EDG room during the surveillance and the fire. The VY EDGs are Fairbanks-Morse, twelve cylinder, opposed piston with parallel exhaust plenums driving twin turbochargers. The EDG was electrically loaded four minutes into the 1-hour operability test.

At approximately 10:10 a.m., the fire detection system in the EDG alarmed and the fire brigade was dispatched. Auxiliary Operators in the EDG room commenced a walkdown of the EDG and identified the fire. Almost coincidentally, the fire self-extinguished (assisted by the quick actions of an AO) and a report to the control room communicated the scope of the fire and actions taken. The fire did not spread, and there was no significant flame or smoke. There was no visual EDG damage and the diesel was shutdown in a controlled manner. The EDG was declared inoperable to support the post-fire investigation. The post-fire investigation identified no significant charring, oil accumulation, or material damage. The exhaust manifolds, flanges, and gasket materials were inspected and verified in good condition; gaskets were replaced as a precautionary measure. The upper cylinder rings were inspected and found in good condition; no material was found in the ring catcher. The as-found conditions of the exhaust manifold were inspected by the inspector. Additional fire extinguishing and detection systems were installed during the post-fire/post-maintenance/operability test on April 27 as a precautionary measure. This re-test was conducted without incident.

During the fire, the inspector observed a small, 6-inch ball of flame situated between two non-combustible insulating blankets that covered the debris catcher portion of the exhaust plenum. This area is located just before the inlet to the OCS turbocharger and adjacent to the body of the EDG. There was some red flashing indicative of flame, however, this was not substantial. The fire appeared to be caused by hot temperatures (generated by the exhaust manifold and captivated by the insulating blankets); atomized fuel and lubricating oils (due to incomplete combustion and a probable microscopic leak in debris catcher flange gasket); and oxygen (fed from both the EDG room and the leak in the debris catcher flange gasket). These elements were confirmed by VY during their post-fire investigation. The inspector concluded that the insulating blankets created a local environment favorable for sustaining a controlled burn.

During the fire, plant personnel were calm and demonstrated professionalism. Communications to the control room were clear and detailed. The fire response was well controlled by the Fire Brigade Leader who assured personnel safety by requiring unnecessary personnel evacuated from the immediate area (a VY self-identified area for improvement) and equipment safety by recommending that the EDG be shutdown in a controlled manner. The Operations Manager observed the surveillance and fire. His actions contributed to the control room operator's knowledge of the magnitude of the fire and fire fighting actions, and to the prompt review of emergency operating entry conditions and reportability requirements.

The post-fire critique was observed and the documented evaluation was reviewed by the inspector. Although the Fire Incident Report (FIR) was accurate and generally descriptive, it was not sufficiently thorough. For example: (1) the initial actions to extinguish the Type B fire did not include the use of the dry chemical extinguisher mounted within the "B" EDG room; the fire fighting strategy for a small fire in the diesel room was not assessed; (2) on two occasions following the self-extinguishment of the fire, all persons temporarily evacuated the EDG room without stationing of a reflash firewatch; the appropriateness of this action while the EDG was still operating was not assessed; and, (3) the FIR lacked an assessment of the magnitude of oil leakage and smoke accumulation within the room prior to, during, and after the fire; this information is important for future comparisons and the initiation of maintenance work orders, where necessary. These observations were discussed with VY management who acknowledged and generally agreed with the inspector's comments.

5.3.2 Emergency Diesel Generator Room Fire Detection

The EDG operations on April 25-27, caused some smoke accumulation within the room. The smoke was caused by a combination of combustion gases leaking from flanges in the exhaust manifold and from the vaporization of fuel and lubricating oils on hot surfaces. A source of the vaporizing oil also appeared to be from the maintenance access plates located on the bottom surface of the exhaust plenum near the inlet to the turbocharger. Similar smoke conditions were previously observed and documented in NRC Inspection Reports 93-02 and 93-05. Therein, the NRC staff was concerned that the EDG room smoke detectors were not of appropriate sensitivity to assure the detection of a fire. This issue was reviewed by an NRC:RI EDG specialist in NRC Inspection Report 94-07, who found that VY actions to assure detector operability were adequate. However, since then, three spurious fire detector actuations have occurred in the "B" EDG room during surveillance apparently due to smoke accumulation; none occurred in the "A" EDG room.

By memorandum dated June 7, 1993, VY maintains that fire detection capability in the EDG rooms remains adequate and in accordance with license requirements. However, the licensee continues to re-assess the effectiveness of the fire detection system. This was most recently demonstrated when the EDG Performance Review Group was tasked to evaluate the recent fire and spurious smoke detector actuations. Recommendations from this multi-disciplined engineering group included installation of alternate/augmented fire detection capability to better discriminate for spurious detector actuations; and, implementation of an engine modification to allow air rolling of the EDGs to remove lubricating oils from the upper crank area (implementation of this modification is expected by September 1994). Discussions with Fairbanks-Morse also occurred regarding, in part, the scope of the post-fire investigation and EDG slow start operation.

5.3.3 Fire Protection Audit

During the first quarter of 1994, VY conducted an audit of fire protection and housekeeping to assure that appropriate programs and requirements are implemented to prevent and mitigate fires having potential impact on plant safety. Functional areas reviewed included maintenance of fire equipment and detection systems, fire brigade training, program self-assessments, and management oversight. The inspector concluded that this audit was of very good detail and broad in scope. Areas for improvement were identified in brigade training and program elements.

5.3.4 Vermont Yankee Observation Program

The inspector conducted a review of the VY Observation Program. This review involved: sampling of the April 1994 program observations, discussions held with plant staff to determined the adequacy of resolution, and independent inspection of similar plant areas reviewed by licensee representatives. As described in NRC Inspection Report 94-09, the VY Observation Program was a positive initiative for the timely detection and resolution of housekeeping deficiencies. Based on this, the following conclusions were made.

The identification threshold for deficiencies was sufficiently low. Observations regarding plant betterment, such as painting and beautification, to potential operational concerns identified during the conduct of maintenance and surveillance were identified. The threshold for VY Observation Program deficiencies is lower than that for the Potential Reportable Occurrence (PRO) corrective action process, which is appropriate.

Program participation by VY managers and supervisors is trended. Currently, VY has 71 percent participation by managers and supervisors indicating an improvement since February 1994. Although program participation does not necessarily assure program effectiveness, the inspector concluded that participation provides increased confidence that deficient plant conditions will be timely identified.

The VY Observation Program assigns equal priority to the deficiencies identified. Observations involving calibration, labeling, procedural difficulties, and instrument settings were equal priority as observations involving painting and housekeeping. In addition, the VY Observation Program does not assure that deficiencies adverse to safe plant operation are timely reviewed by cognizant individuals. Although plant deficiencies were corrected by the MPAC process, which also assigns a work order priority, the balance of observations were not prioritized. This potentially challenges timely management review and disposition of deficiencies. No operability concerns were identified based on the observations reviewed.

A relatively high percentage of observations were performance based. Plant operators, department supervisors, and VY management observed the conduct of maintenance and surveillance and evaluated the conditions observed. Deficiencies were documented and initial corrective actions were identified. A number of observations were outside the area of expertise of the observer. This demonstrated inter-departmental ownership and responsibility.

These conclusions were discussed with the Technical Programs Manager and Assistant to the Plant Manager who indicated they would evaluate the inspector comments and observations. For other than minor items, the program relies on existing corrective action processes to resolve identified deficiencies or concerns. Although the Technical Program Department performs a monthly assessment of Observation Program use, they do not include an evaluation of the use and effectiveness of the corrective action processes to resolve items identified by the program. The inspector concluded the program appears to be a good initiative but the above noted condition represents an area for potential enhancement.

5.4 Emergency Preparedness

During the "B" EDG surveillance and fire brigade response to the small fire on the diesel exhaust plenum (Section 5.3.1), the inspector observed that communication circuits, (fire alarms and operating panel alarms) are difficult to hear in the EDG room during diesel operation. The Director of External Affairs (responsible for emergency preparedness) stated that this issue would be reviewed to determine whether enhanced communication circuits within the EDG rooms are necessary.

6.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (40500, 90713)

6.1 Review of Written Reports

The inspectors reviewed Licensee Event Reports (LERs) submitted to the NRC 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 94-01 Non-Conservative Technical Specification Setpoint Range Applied to Calibration Procedure for Rod Block Monitor Auto-Bypass Feature Due to Personnel Error

Vermont Yankee identified that due to a cognitive error made during the preparation of a revision to plant procedure OP 4304, Rod Block Monitor Functional/Calibration Test, which changed the setpoint from 30% to $\geq$30% reactor power. This error was made in 1973 and a subsequent procedure revision implemented in 1978 assigned a calibration range of 30 to 32.5 percent. As such, the calibration range for the rod block monitor setpoint exceeded the TS-required trip setpoint of $\leq$30% reactor power.

The purpose of this rod block is to prevent localized fuel damage during an inadvertent rod withdrawal of a high worth rod. Vermont Yankee determined that although the setpoint was non-conservatively set, there was no safety significance. This was based on, in part, the large reactivity changes necessary to cause fuel damage, and other protective features such as average power range monitor rod blocks and high flux scrams.

The inspector reviewed the corrective actions and observed that VY took credit for a number of long-term initiatives already implemented to resolve previously listed surveillance issues. These involved the conduct of their TS review program which is nearing completion, and the performance of department self-assessments. The initial corrective actions were appropriate.

- LER 94-04 Reactor Scram From High Moisture Separator Level Due to an Unexpected Failure of the "C" Moisture Separator Normal Drain Valve Level Transmitter

The LER adequately described event and corrective actions taken. The LER did tend to interchange "turbine control valve fast closure scram" with "load reject scram;" however, functionally the scram did occur due to actuation of both trip signals, as designed. In addition to the corrective actions described in the LER, VY also plans to rebuild the positioners and install new air filter regulators in the emergency level control system, and plans to replace the emergency dump valves with a new single seat valve design during the next refueling outage.

- LER 94-06 Missed Reactor Coolant Conductivity Samples Due to a Misinterpretation of the Test Method Used to Meet Technical Specifications

On April 19, 1994 and during a chemistry training class, a Chemistry Supervisor questioned the validity of the 96-hour conductivity sample analysis required by TS 4.6 as being fulfilled by an on-line monitor. Licensee review of the associated Potential Reportable Occurrence (PRO) report led to their conclusion that the on-line monitor did not satisfy TS 3.6/4.6 and related basis for the TS for about a 2 year period. The on-line monitor was then supplemented by sampling and laboratory analysis. The root cause of the problem was a misinterpretation (apparent individual cognitive error) of the TS requirement on or about the time a procedure revision was issued on June 19, 1992, implementing the use of a new "state of the art" on-line conductivity monitor and the other daily sampling/water quality analyses.

The LER was of sufficient detail to understand and analyze the event. The NRC staff is exercising enforcement discretion because the violation of TS 4.6 meets the criteria of 10 CFR 2, Appendix C, Section VII.B(2) as being licensee-identified. The licensee individual who self-identified the problem exhibited good attention to detail. Licensee middle management showed an aggressive and conservative approach in resolving the problem along with specifying actions to prevent recurrence. The apparent individual cognitive error in processing the related procedure revision on this event are not considered to be repetitive or preventable from a previous occurrence or violation.

- LER 94-07 Inadvertent Primary Containment Isolation System Actuation Due to a Spurious Spike on a Refuel Floor Radiation Monitor

This event and VY corrective actions were evaluated in Section 2.4 of this report.

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 March and April 1994.

- 1993 Personnel Exposure Report

This report, submitted pursuant to TS 6.7.A.2, documented man-rem exposures by functional area, contract workers, and plant employees. Plant employees received a total of 47.8 man-rem for 1993 representing a 17 percent reduction from the 1992 total. A similar reduction was observed in the man-rem totals from contract workers. Refueling/maintenance outages were performed each year.

7.0 MANAGEMENT MEETINGS

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 and prior to report issuance. No proprietary information was identified as being included in this report.

7.2 Region Based Inspection Findings

Two Region based inspections were conducted during this inspection period. Inspection findings were discussed with senior plant management at the conclusion of the inspections.

Date	Subject	Rpt. #	Inspector
4/18-22/94	EDSFI Followup	94-05	L. Cheung
5/6-10/94	Security	94-12	R. Albert

7.3 Other Meetings

On April 25, a meeting was held at the NRC:RI office between members of the NRC and representatives of VY to discuss Requalification Program findings contained in NRC Inspection Report 94-02.