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REGION I

Report No. 95-23

Docket No. 50-271

Licensee No. DPR-28

Licensee: Vermont Yankee Nuclear Power Corporation
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Ferry Road
Brattleboro, VT 05301

Facility: Vermont Yankee Nuclear Power Station
Vernon, Vermont

Inspection Period: October 3 - November 6, 1995

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Approved by: Richard J Conte 11/29/95
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Scope: Station activities inspected by the resident staff this period included Operations, Maintenance, Engineering, Plant Support, and Safety Assessment and Quality Verification. Backshift inspections including weekend activities amounting to 15 hours were performed on October 7, 9, 18, 19, 20, 27 and November 1 and 3. 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. The failure to adequately test reactor core isolation cooling and high pressure coolant injection stop-check valves was identified as a violation of 10 CFR 50.55a.

EXECUTIVE SUMMARY

Vermont Yankee Inspection Report 95-23

Safety Assessment and Qualify Verification

VY staff (multiple departments) follow-up of two industry operating events (one involving a scram discharge instrument volume high level logic relay concern and the second involving a Part 21 on EDG synchro-check relays) was prompt and thorough.

A focused examination of the Event Reporting System identified a good systematic and comprehensive initial review of identified problems and deficiencies. Conservative Event Report decision making and sound initial corrective actions and operability determinations were routinely observed and were found to have contributed to overall safe day-to-day plant operations. The downgrading of a July 12, 1995 ER involving identified IST deficiencies was viewed as an exception to this generally conservative approach.

Corrective actions documented in LER 95-15, which addressed the VY staff's identification of improper containment local leak rate testing of a small set of flanged butterfly valves, were prompt and thorough. This licensee identified Technical Specification violation of minor safety consequence was not cited.

Operations

The plant staff's response to grass, leaves, and debris build-up on the intake structure trash racks following the October 21 and 22 heavy rainfall was timely and demonstrated sound operating staff understanding of plant systems' function and dynamics.

A review of the safety relief valve (SRV) leakage detection system determined that the plant staff's monitoring of SRV tailpiece temperature does not provide a conclusive assessment of relief valve leakage. In particular, the SRV temperature base-lining methodology employed by the plant staff does not compensate for external heating and cooling effects caused by changes in the drywell ventilation system. Notwithstanding, no leakage was evident and operators have monitored closely the available information for detecting SRV leakage.

Maintenance

Corrective maintenance performed on the "C" main feed pump and "A" rotating un-interruptible power supply and the replacement of the Stack Gas I and II radiation monitors were well coordinated and executed.

Maintenance staff completion of cooling tower fan No. 2-1 gear box lubricating oil change-out on November 4, within a 90 minutes time period, represented good planning and execution. The identification of the lubricating oil pour point concern demonstrated a thorough reliability based maintenance base-lining effort and a timely review, with respect to precluding a potential cold weather operating vulnerability.

Observation of the "A" emergency diesel generator (EDG) surveillance test identified improvement in the pre-test briefing and the overall conduct of this test. The pre-test brief was considered noteworthy, based upon the level of detail and the application of lessons learned from previous testing problems.

Engineering

The failure to properly test high pressure coolant injection and reactor core isolation cooling check valves V13-818 and V23-843 in the closed direction on reverse flow, represented additional examples of an Inservice Testing (IST) program deficiency already identified and in the process of being addressed. VY's handling of corrective actions generated from earlier identified IST problems, documented in IR 95-19, and in particular the decision to downgrade the associated ER from a Level 1 to Level 3 review, represented a missed opportunity for the VY staff to fully evaluate the IST program and to uncover similar testing problems. This example indicated a VY management insensitivity to correcting IST programmatic problems in a timely manner and was another example of inadequate corrective actions addressed in Violation C of IR 95-22. Further, this finding and others support the overall assessment of weak IST program management, as concluded in IR 95-22.

The failure to adequately test stop-check valves V13-817 and V23-842 represented a different IST program testing deficiencies than previously identified and docketed. Inspector review of the docketed corrective actions (reference LER 95-17) determined that it would be speculative to conclude that the corrective actions to resolve the IR 95-22 identified problems would capture this specific stop-check valve testing deficiency. In this regard, the failure to adequately test stop-check valves V13-817 and V23-842 was a violation of 10 CFR 50.55a and OM-10a (**VIO 95-23-01**). Once identified, the VY staff initiated appropriate testing to verify closure of these valves.

The VY staff was timely and thorough in evaluating a pin hole leak in the service water system and demonstrated a heightened sensitivity for the significance of this condition. Although SW pin hole leaks have been a recurrent problem at VY, the licensee's evaluation of each particular problem has been good and focused on safety. No significant structural degradations have been identified.

The approved minor modification package and supporting engineering justification for the modification of the service water effluent radiation monitor sampling system were not well founded based upon the lack of quantitative analysis and the lack of a clear understanding of system design and operation.

Review of the EDG air intake screen modification identified that both the design change package and field installation efforts were of high quality.

The engineering justifications and evaluation provided reasonable assurance that EDG operability would be maintained and this longstanding problem corrected.

Plant Support

Inspector review and verification of compensatory measures for the Appendix R deficiencies identified a heightened sensitivity of the plant staff with respect to good fire prevention/protection practices, minimization of combustible loading, improved control of ignition sources, and better housekeeping in the affected zones.

Implementation of selected elements of the radiological controls and security programs were found to be proper.

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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) operated at 100 percent rated reactor power throughout this inspection period. Minor reactor power changes were made to support surveillance testing. On November 2 and 3, 1995, VY sponsored a Turbine Retrofit Symposium at the Brattleboro Conference Center. The purpose of the symposium was to share with their industry counterparts the multiple lessons learned from the Spring 1995 refueling outage when both low pressure turbines were replaced.

During the weeks of October 23 and November 6, 1995, a region based specialist inspector, with contractor technical support, reviewed the 10 CFR 50, Appendix R Program. Inspection findings and observations will be documented in inspection report 95-25.

During the week of October 30, 1995, a region based specialist inspector conducted a routine inspection of the radiological effluent monitoring program and its implementation. Inspection findings and observations will be documented in inspection report 95-24.

2.0 OPERATIONS

2.1 Operational Safety Verification

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 tag-outs properly isolated equipment for maintenance.

2.2 Intake Structure Partial Blockage

On October 21 and 22, heavy rains hit the area resulting in a significant increase in level and flow in the Connecticut River. The increased flow resulted in a build-up of river grass and leaves on the intake structure trash racks. The rapid level rise and increased flow initially rendered the floating boom in-effective (designed to minimize the Fall season leaves entrainment in the intake). Plant staff manually removed the grass and leaves built-up on the racks before it had a significant plant operational impact. The control room operators modified the circulating water flow and were able to prevent further build-up of debris on the trash racks.

The inspector monitored the plant staff's efforts to keep the intake structure clear and discussed system design and operating configurations with operations and engineering representatives. By using intake bay recirculating flow (typically used in the winter months for tempering), operating two vice three circulating water pumps, and using the bubbler system (used to prevent ice build-up on the intakes) in combination with the boom, the operating staff was able to maintain the masses of grass, leaves, and other floating debris away from the intake structure. The inspector concluded that the plant staff's response to this operating challenge was timely and represented a sound understanding of systems function and dynamics.

2.3 Safety Relief Valve Leakage Monitoring and Drywell Ventilation Effects

The inspector reviewed VY's operation of the drywell ventilation system to assess the impact of drywell atmospheric conditions on safety relief valve (SRV) leakage monitoring system. This inspection was initiated, in part, because of a recent industry event in which an SRV had inadvertently opened (reference Information Notice 95-47), and because of an observation on October 23 at VY where all six SRV tailpiece temperatures increased 4-6 degrees when the #4 drywell air cooling (DAC) unit line-up was changed. During the VY occurrence, the "C" SRV tailpiece temperature approached the alarm setpoint of 195 degrees.

As described in VY FSAR Chapters 4.4 and 7.4 and General Electric Service Information Letter (GESIL) No. 196, "Summary of Recommendations for Target Rock Main Steam Safety/Relief Valves," increases in SRV tailpiece temperature indicate SRV leakage and a potential for a subsequent SRV problem. As discussed in Information Notice 95-47, the SRV that inadvertently opened was previously leaking and indicated an elevated tailpiece temperature. Although VY representatives stated that their SRVs were not currently leaking, the inspector noted that the current operation of the non-safety related drywell ventilation system (in particular the DAC units) has a direct effect on tailpiece temperature indication and has the potential to mask any temperature increase due to SRV leakage changes.

The inspector discussed this potential problem with the Operations Department staff. The staff presented trend information of SRV tailpiece temperatures and articulated that the information provides a baseline for a "normal" SRV tailpiece temperature. As described in GESIL No. 196, a 10 degree increase above a baseline could indicate a potential SRV problem (a 50 degree increase should be seen before the SRV fails open). The inspector acknowledged VY's trending initiative, but noted that changes in DAC cooling water flows and/or changes in DAC operation were not annotated on the trends. Consequently, these SRV tailpiece baseline trends were not normalized to compensate for temperature changes caused by external factors (i.e., DAC unit operation). In addition, the inspector noted that all six SRVs share a common alarm setpoint of 195 degrees and that the change in SRV tailpiece temperature to reach this setpoint varies from 7 degrees for the "B" SRV to approximately 50 degrees for the "A" and "D" SRVs. Therefore, the magnitude of SRV leakage required to reach the alarm setpoint differs between the SRVs. The Senior Operations Engineer acknowledged these observations and stated that SRV base-lining methodology and alarm setpoints would be reviewed for possible enhancements.

In summary, the inspector has verified that the drywell ventilation system was operated in accordance with VY procedure OP-2115, "Primary Containment," and that drywell temperatures have been maintained within established administrative limits (a daily drywell temperature surveillance is conducted). In addition, control room operators (CROs) effectively used the plant process computer to monitor SRV tailpiece temperatures. However, the current SRV tailpiece base-lining methodology does not normalize the measured temperatures to compensate for external heating or cooling effects, not associated with SRV leakage. As a result, any increases in the current VY SRV baseline information would not provide a conclusive assessment of SRV leakage or pending SRV problems, and an increase in SRV leakage could potentially be masked. VY representatives agreed to review this problem.

3.0 MAINTENANCE AND SURVEILLANCE

3.1 Maintenance

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

The inspectors reviewed corrective maintenance on the "C" main feed pump (MFP) and "A" rotating un-interruptible power supply (RUPS), and the replacement of the Stack Gas I and II radiation monitoring equipment. These activities were conducted with proper safety tag-outs to assure equipment and personnel protection. The corrective maintenance on the "C" MFP was required to resolve a small water leak on the inboard pump seal. Because this seal had recently been replaced, the pump seal vendor was onsite to assist VY in the resolution of this repetitive condition. The corrective maintenance on the "A" RUPS was to adjust the A2 brush holder spring which was identified as being skewed during the weekly maintenance check. This condition was corrected and the "A" RUPS was subsequently returned to service. An ER was initiated to evaluate the cause of this problem. The entry into the TS action statements for Stack Gas I and II was well understood by the CROs and routinely discussed at the Plant Manager morning meeting. This work was completed within a week and involved good coordination between the Operations, Instrument & Controls, and Radiation Protection staffs. An NRC radiological effluent specialist also reviewed this equipment upgrade (reference inspection report 95-24).

3.1.1 Cooling Tower Fan Maintenance

On November 4, VY made a one-hour non-emergency notification per 10 CFR 50.72 to identify entry into TS 3.5.D.4 which necessitated the reactor be in cold shutdown within 24 hours. Entry into this limiting condition for operation (LCO) was commenced at 10:23 a.m. when cooling tower fan (CTF) No. 2-1 was removed from service (scheduled) to replace the gear box lubricating oil. Removal of CTF 2-1 from service in conjunction with both service water subsystems having been declared inoperable (a 7-day LCO entered on November 2, reference Section 4.2) placed the unit in a more restrictive 24-hour action statement. The CTF 2-1 gear box lubricating oil change-out was based upon an earlier discovery by the VY staff that the lubricating oil installed had a pour point of 25 degrees F. The 25 degree F pour point was identified via the reliability based maintenance base-lining efforts being conducted by the maintenance department.

The inspector determined that the CTF 2-1 lubricating oil change-out and satisfactory post-maintenance testing was completed in approximately 1.5 hours and that CTF 2-1 was declared operable at 11:50 a.m. The inspector noted that the lubricating oil previously used in the CTF fan gear boxes was recommended by the vendor. The replacement lubricating oil has a pour point of minus 50 degrees F, which bounds any potential cold weather conditions in which the cooling tower fan may be called upon to operate. Lube oil change-out of the remaining (non-safety related) CTF gear boxes has not been scheduled and will likely not occur because the non-safety related cooling towers are not placed in operation (closed cycle cooling) during the colder winter months.

The inspector concluded that VY staff identification of the inadequate pour point of the CTF 2-1 gear box lubricating oil was noteworthy. The VY staff's planning and execution of this maintenance activity in 90 minutes was good, and that the prompt action taken to preclude a cold weather operating vulnerability was appropriate.

3.1.2 Review of the Forced-Outage Work List

The inspector reviewed the VY forced-outage work list (dated October 25, 1995) and identified no outstanding maintenance conditions adverse to continued power operation. The majority of the work items on the list were associated with balance of plant systems and associated instrumentation and controls. For example, calibrations were planned for feedwater level transmitters and control valves, turbine bypass valve and control intercept valve position indication, and turbine supervisory instrument panel. Inspector review of maintenance to be conducted on a drywell pressure transmitter, planned repairs to service water pin hole leaks, and the replacement and calibration of average power range monitor flow converters also identified no concerns.

3.2 Emergency Diesel Generator Surveillance Review

As described in inspection report 95-19, the "A" emergency diesel generator (EDG) tripped during surveillance testing because an operator failed to perform a particular step in the surveillance procedure. The inspector observed the conduct of this quarterly EDG surveillance with particular inspection emphasis focused on the assessment of operator performance in the field.

The surveillance was conducted in accordance with OP 4126, revision 37, "Diesel Generator Surveillance," section C.1. A pre-test control room brief was conducted that emphasized procedure adherence and communications. Special emphasis was placed on reviewing problems that occurred during previous EDG surveillance tests. These problems included the above mentioned procedural adherence concerns and fire protection considerations in light of the current Appendix R fire protection concerns (reference Section 5.3). In the field, operators were observed verifying prerequisites and conducting a walk-down of the EDG prior to the surveillance test. Immediately prior to EDG start, operators second checked the position of speed and load settings and the service water flow control valve position. A failed breaker position indication on the local EDG control panel was properly evaluated and resolved.

The inspector concluded that the operators correctly implomented the surveillance procedure. The pre-test brief was considered noteworthy, in that, the brief was detailed and involved lessons learned from previous EDG surveillances. Off-normal conditions identified during the surveillance were properly resolved.

4.0 ENGINEERING

4.1 (Open) VIO 95-23-01: Inservice Testing Deficiencies

During a review of reactor core isolation cooling (RCIC) surveillance testing results on October 7, the inspector identified that stop-check valve V13-817 was not properly tested to assure cessation of reverse flow. The inspector also identified that check valve V13-818 (located immediately upstream and in series with V13-817) was not tested per Inservice Testing (IST) requirements. Both safety related valves are located in the RCIC vacuum pump discharge piping and are required to promptly close during a design basis accident. The same configuration and testing deficiency was identified in the high pressure coolant injection (HPCI) system (valves V23-842 and V23-843, respectively).

The requirements for IST are contained in TS 4.6.E.2, which requires testing in accordance with the 10 CFR 50.55a and the American Standard of Mechanical Engineers (ASME) Code, Section XI. Section XI incorporates, by reference, Part 10 (OM-10) of ASME/ANSI OMa-1988, which establishes the requirements for the testing of "stop-check" valves. These requirements were documented in NUREG 1482, "Guidelines for Inservice Testing at Nuclear Power Plants," and further clarified in NRC Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Program," which states:

> "If a prompt closure of [stop-check valves] on cessation or reversal of flow is required to accomplish a safety-related function, closure must be verified by reverse flow testing or such other positive means as acoustic monitoring or radiography."

As described in VY procedure OP 4121, "Reactor Core Isolation Cooling System Surveillance," and OP 4120, "High Pressure Coolant Injection Surveillance" the closed position of "stop-check valves" (V13-817 and V23-842, respectively) were verified by use of the manual hand wheels. This test methodology did not provide a positive means of verifying the cessation of reverse flow because valve seat and disk leak integrity cannot be verified by stem travel alone. Contrary to the requirements stated above, the inspector concluded that Vermont Yankee did not perform reverse flow testing o; these stop-check valves.

In regards to check valves V13-818 and V23-843, OM-10, Section 1.1 of the ASME Code requires testing of valves installed to protect systems that perform a specific function in mitigating the consequences of an accident. NUREG 1482, Section 4.3, clarifies that requirement to test ch k valves based on Section III of the ASME Code or the applicable code of construction (ASME B31.1). Check valves V13-818 and V23-843 are in the VY IST Program and are identified for testing in the open direction only. However, these valves have a safety function to close, but were not tested in this direction, contrary to the above requirements.

The inspector discussed the above findings with the VY IST Coordinator, who initiated an Event Report that resulted in management reviews. Radiography was performed during the week of November 4, confirming that all four subject valves were operable. Radiography will also be performed quarterly thereafter to verify that the check and stop-check valves properly close.

Previous IST Program Deficiencies

As described in inspection report (IR) 95-19, VY failed to test check valves in the residual heat removal and core spray systems. This violation of TS requirements was identified by the licensee on July 12, and subject to enforcement discretion, in part, based on the Level 1 Event Report (No. 95-0436) being written to implement corrective actions and complete a root cause determination within 30 days. On August 3, this Event Report (ER) was downgraded to a Level 3 review, requiring an apparent cause determination and the completion of corrective actions within 120 days (e.g., November 8, 1995). At the end of this inspection period, the inspector reviewed the status of these corrective actions and found that (with the exception of radiography to verify valve operability) none have been dispositioned by VY management.

As documented in IR 95-22, the NRC staff identified additional IST program and testing methodology problems during an inspection conducted in September 1995. A number of power-operated valves, check valves, and relief valves were determined not to be included in the IST program and consequently not tested. LER 95-17 reported these deficiencies and corrective actions to the NRC. Vermont Yankee's docketed response to the IR 95-22 violations was pending at the conclusion of this inspection period.

Summary

The failure to properly test HPCI and RCIC check valves V13-818 and V23-843 was similar to the testing deficiency documented in ER No. 95-0436 and reviewed in IR 95-19. In particular, the check valves were included and tested as part of VY's IST program, however, the testing methodology to verify valve closure was not conclusive because the test verified that only one of the two in-series valves (i.e., the check valve or "stop-check" valve) was capable of reverse flow closure. In addition, Violation B of IR 95-22 described more problems associated with check valve testing. Consequently, the inadequate testing of check valve V13-818 and V23-843 represented additional examples of a program deficiency already identified and in the process of being corrected.

VY's handling of corrective actions generated from the earlier inservice testing problems, documented in IRs 95-19 and 95-22, and in particular the decision to downgrade the ER from a Level 1 to Level 3 review, represented a missed opportunity for the VY staff to fully evaluate the IST program and to uncover similar testing problems. Moreover, it highlighted VY management insensitivity to correcting IST programmatic problems in a timely manner.

This finding was another example of inadequate corrective actions addressed in Violation C of IR 95-22 and supports the overall assessment of less than effective IST program management that was a conclusion of IR 95-22.

The failure to adequately test stop-check valves V13-817 and V23-842 (in distinction to "regular" check valves to stop reverse flow) represents a different IST program testing deficiencies than previously identified and docketed. Inspector review of the docketed corrective actions (reference LER 95-17) determined that it would be speculative to conclude that the corrective actions to resolve the IR 95-22 identified problems would capture the specific stop-check valve testing problems identified above. In this regard, the failure to adequately test the stop-check valves V13-817 and V23-842 was a violation of 10 CFR 50.55a and OM-10a. **(VIO 95-23-01)**

4.2 Service Water Pin Hole Leak

On November 2, VY confirmed that a pin hole leak existed in the SW to fire main keep-fill line located in the SW intake structure. Both SW subsystems were declared inoperable and VY entered the 7-day LCO shutdown action statement. That morning, plant management was briefed on the problem and assessed the safety significance of this safety related, ASME Class 3 pipe, through-wall leak. Engineering expertise from VY and YAEC were assigned to evaluate this condition and discussions with NRC staff were initiated.

The pin hole is near the 12 o'clock position of the 1.5 inch seamless carbon steel SW pipe and outside any heat-effect weld area. The pin hole leak rate is very small and subject to evaporation. Nondestructive examination by ultrasonic testing (UT) and radiography confirmed adequate wall thickness for structural integrity and indicated that the leakage was caused by microbiological-induced corrosion (MIC). UT was also performed in five additional locations to evaluate whether a more wide spread wall thinning mechanism was prevalent. No other deficient conditions were identified. Surveillance of the pin hole is performed by auxiliary operators every eight hours and structural integrity is verified quarterly by UT.

Similar to previous SW pinhole leaks (reference inspection report 95-19), appropriate engineering rigor and nondestructive testing provided confidence that adequate structural integrity existed in the vicinity of the leak. The VY staff demonstrated a heightened sensitivity for the significance of this non-conforming condition and promptly assessed the condition using NRC guidance (Generic Letter 90-05). Appropriate consideration of spray and flooding protection and fire water system operation was made. A strong questioning attitude was exhibited when the pin hole was examined using radiography. Although a leak repair could not be accomplished because the 24-inch gate isolation valves leaked, the repair plan was aggressive and meticulous. The inspector noted that this observed gate valve leakage was considered normal for this size of valve and system pressure. System configuration prevented the establishment of a drain path to maintain the weld areas dry.

In summary, the VY staff was timely and thorough in evaluating this pin hole leak. Vermont Yankee was treating the pin hole leak as a housekeeping issue as per Generic Letter 90-05, Temporary Non-Code Repair of ASME Class 1, 2, and

3 Piping, and will docket a letter describing the issue and corrective actions. Although SW pin hole leaks have been a recurrent issue at VY, the licensee's resolution of each particular problem has been good and focused on safety. No significant structural degradations have been identified. The inspector determined that VY has initiated the development of a strategic plan for the long-term maintenance of the SW system using a combination of SW chemical treatment and inspection/monitoring programs.

4.3 Service Water Radiation Monitor Design Change

A service water (SW) radiation monitor system (RMS) continuously samples a slip-stream of water from the SW discharge header prior to discharge to the Connecticut River. Operability of the SW RMS is specified in TS 3.9.A.1. Daily SW grab samples and analysis are required with the SW RMS out-of-service.

Inspector review determined that since 1993 the SW RMS has been inoperable on a number of occasions due to low SW RMS flow rates caused by silt accumulation. During this current operating cycle, the SW RMS was inoperable three times because river water silt had accumulated to a point that diminished proper system operation. The corrective action this period, for each occurrence, was identical to that recommended in a 1993 engineering evaluation: flush the SW RMS slip-stream to clear the sample line of silt. In 1995, the plant staff recommended moving the SW RMS slip-stream booster pump from the downstream side (where it was originally installed in September 1974) to the upstream side of the RMS. The recommendation stated that this change would improve slip-stream flow and reduce silt deposition in the RMS process line.

The inspector reviewed the minor modification (MM) package prepared to relocate the booster pump in the sample line and noted that the engineering evaluation for this change was purely qualitative. In particular, the engineering justification for the proposed modification was subjectively based on assumptions of system performance and did not involve any disciplined hydraulic analysis. Notwithstanding, the inspector identified no immediate safety concern with the proposed modification. Follow-up questions with the responsible engineering staff identified that the design basis and requirements for the original configuration were not fully understood. Information such as design flow rates, pump head, and integrated system operation was not available. Silt deposition rate and causes of silt accumulation were speculative. In addition, the post-modification acceptance criterion was based on a subjective assessment of SW flow without consideration for a long-term evaluation of silt deposition or RMS system performance.

The above observations were discussed with the Mechanical Engineering and Construction (ME&C) Manager who acknowledged that further engineering evaluation would be necessary to justify the removal of this TS radiation monitoring system from service. The ME&C manager also stated that a proposal to re-pipe the SW RMS supply was being pursued to reduce system head loss due to suspected inner wall piping corrosion. In summary, the approved modification package and supporting engineering justification for the removal of this TS system from service were not well founded and were being re-evaluated at the close of the inspection period.

4.4 Emergency Diesel Generator Air Screens

Vermont Yankee modified the emergency diesel generator (EDG) combustion air intake by installing screening material over the air plenum inlet. This modification was required to keep birds from nesting in this area of the EDG air intake to the combustion air filters. These filters have been inspected weekly and periodically cleaned, as required, to remove fecal and nesting material from the filter surface. Inspector assessment of VY efforts to resolve this problem was described in inspection reports 95-04 and 94-20, and characterized as a condition potentially adverse to EDG operability.

The inspector reviewed the modification package and inspected the work area to evaluate this safety system design change. The modification package provided a sound justification for the design change. Elements included seismic and structural loading calculations, independent engineering reviews, and a conservative free-air space requirement. Worker safety, EDG availability, and design configuration control were also considered in the modification package and found to be appropriately evaluated. Prior to modification, other engineered options to remedy the problem were explored and evaluated. Lessens learned from of other nuclear facilities were evaluated and pest control experts were consulted prior to implementing the air screen modification.

Based on field inspections, the modification was properly installed in accordance with the modification package. Quality control elements included peer evaluations, supervisor oversight, and proper foreign material control and configuration control methodology. A periodic surveillance was also initiated via the modification process to detect the accumulation of deleterious materials on the air screens. The inspector noted, however, that acceptance criteria for this periodic inspection were not clearly defined.

In summary, the design change package was of high quality. Although its too early to tell whether the modification will preclude the entry of deleterious material into the EDG filters, the engineering justifications and evaluations provided reasonable assurance that EDG operability would be maintained. The modification was installed without an adverse impact on EDG availability or operability.

5.0 PLANT SUPPORT

5.1 Radiological Controls

Inspectors routinely observed and reviewed radiological controls and radiation protection 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. A walk-down of the high radiation access doors to the heater and condenser bays verified that the doors were properly maintained. Inspections conducted at the control point to the radiologically control area

(RCA) confirmed that personnel and equipment were properly surveyed prior to exit and that the radiation protection (RP) log accurately reflected the work in progress and surveys performed.

The inspector conducted inspections of work activities within the emergency core cooling system (ECCS) corner rooms and on the refuel floor and noted proper radiological protection practices. In particular, the radiological controls implemented for the workers refurbishing and implementing design changes to the refuel bridge provided positive control of contamination and segregated the work site from the spent fuel pool. Workers were observed to follow posted RP instructions and to implement contamination control practices. Within the ECCS corner rooms, the inspector verified that RP survey results were consistent with the actual radiological conditions and that the scope of the surveys enveloped the maintenance conducted.

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. A walk-down of the Protected Area (PA) fence verified that appropriate security compensatory measures were implemented while contractors were adjacent to the PA fence installing the vehicle barrier. Contractors within the Owner Controlled Area (OCA) were properly accounted for, as required by the VY Emergency Preparedness Plan. Inspections of the Secondary Alarm Station (SAS), verified that the SAS guard was attentive and cognizant of security barrier deficiencies and security compensatory measures in place. No concerns were identified with the security surveillance and alarm equipment.

5.3 Fire Protection

The inspectors conducted inspections of the reactor building (RB) with particular emphasis in the RB zones identified by VY as having potential Appendix R fire protection concerns. These areas included, in part, the RB 252' north, torus room, and RB 280' north. The inspectors verified that continuous and hourly firewatches adequately inspected the areas for fire and potential fire hazards. The firewatches interviewed understood their fire protection responsibilities and had an appreciation for the significance of the Appendix R fire concerns and of the particular areas and systems of importance within the RB.

The inspectors noted good housekeeping and low combustible loading. Vermont Yankee fire protection management re-emphasized to the plant staff the importance of good fire protection, combustible material and ignition source control, and the RB zones of concern. Department and control room operator training was conducted this inspection period to re-familiarize plant personnel with the specific RB locations of concern. The effectiveness of this effort was evidenced by fire protection observations made by the general plant staff and Fire Brigade Leaders (FBLs). For example, a radiation protection technician identified material deficiencies with fire exclusion zones and fire water stations, and FBLs identified potential concerns with the onsite control of combustible pressurized gases and a deficient fire penetration seal in the control room. Each observation was reviewed by plant management and properly resolved.

5.4 Control of Staging

A review and walk-down of staging erected in the reactor building confirmed that the engineered safety requirements, as described in plant procedure AP 0021, Control of Temporary Materials, were properly implemented. Temporary staging has been erected in many locations to support the radiography of valves per the Inservice Testing Program. Locations within the reactor building include both low pressure emergency core cooling rooms and the torus room. The inspector verified that staging erected in the torus room had sufficient clearance between the structural elements of the staging and the suppression chamber wall. This clearance is necessary to preclude the potential rupture of this primary containment boundary during a design basis event. In the ECCS corner rooms, the staging was erected appropriately to prevent an adverse impact on adjacent ECCS pumps and valves. Staging access ladders were properly secured and personnel safety rails and kick boards were installed. Materials used for staging were verified to be fire retardant. The inspector also verified that staging did not restrict access to plant equipment nor did it impact the operability of installed fire detection equipment.

6.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION

6.1 Operating Experience Reviews

During this inspection period VY initiated a timely evaluation of an industry operating experience involving a design problem with the scram discharge volume (SDV) high level reactor protection logic system. The subject event involved the discovery that a failure of a single sub-channel relay, concurrent with a SDV high level, would fail to generate a reactor scram. Within one week of receipt of this operating experience information, three different plant organizations initiated independent reviews of this potential reactor safety problem and confirmed that it was not applicable at VY.

A second operating experience review conducted this period involved synchro-check relays on the EDGs. This particular problem was described in a 10 CFR Part 21 notification and involved incorrect ratings applied to relays that are part of the electronic circuit that detects voltage and determines the phase angle for local-manual paralleling of the EDGs. The problem, as described in the Part 21 report could result in the undetectable failure of these relays during normal operation and surveillance testing. As a potential consequence, during local-manual operation an inaccurate phase angle relation between the EDG and the energized bus could exist prior to EDG generator output breaker closure. If this were the case, a significant equipment and personnel safety hazard may result from the EDG being paralleled out-of-phase.

The inspector determined that this particular relay failure is distinguishable by visual inspection. A weekly surveillance has been initiated by the VY staff. The inspector notes that although local-manual parallel operation is proceduralized, this mode of operation is very infrequent. EDG operation is asynchronous (loaded on a de-energized bus) during design basis accident response and during EDG surveillance testing the EDG is paralleled to the 480 vac bus from the control room. To preclude this potential problem VY plans to replace the subject relays during the next convenient EDG maintenance period.

6.2 Event Reporting System Assessment

As described in inspection report 95-06, VY implemented the Event Reporting System as a vehicle to identify and track to resolution plant problems, adverse trends, material deficiencies, and generally any other item requiring corrective action. The Event Report (ER) also contributes to the trending and integration of potentially adverse conditions originating from plant operation, maintenance, quality assurance, and self-assessments. Inspection 95-06 credited this licensee initiative as an enhancement to the VY corrective action program. During the period August 22 to November 6, the inspectors observed the ER screening meetings for 59 ERs to assess whether the screening meeting dispositioning of the ERs was appropriate. This sample size represented approximately 50 percent (59/116) of all ERs generated during that period.

Based on the inspector's assessment of the problems documented in these ERs, VY's initial management reviews have been timely and focused on safety. It has been a matter of routine that ERs have been screened by a multi-disciplinary group of department managers supervised by the Plant Manager and/or Superintendents within 24 hours of the occurrence. Recommendations and immediate actions were discussed, refined, and documented during the screening meeting. The inspector found that this structured forum was consistently implemented, as described in the governing plant procedure AP 0009, "Event Reports," and resulted in appropriate senior management oversight and involvement. The inspector identified no case where a particular ER languished prior to management review at the screening meeting.

Each ER reviewed included the assignment of responsibility, assessment of safety significance, and an initial management review of reportability and operability. VY's immediate decisions were conservative and reflected an appropriate focus on plant safety. For example, ERs involving service water pinhole leaks; the test failure of an EDG voltage over-current relay; IST program and testing problems; a fuse failure in a safety relief valve bellows failure alarm circuit; and, inadequate quality control applied to fuses staged for Appendix R considerations, were all critically evaluated by the management team. Particular emphasis was focused on generic implications, immediate and long term impact on plant safety, and whether the issue represented a recurrence of a past problem. The inspector routinely observed the escalation of the ER priority level for ERs of a recurring nature. This was evidenced in VY's handling of ERs concerning Appendix R and personnel safety issues. In spite of the generally good initial screening and corrective action development, the VY staff was less effective in dealing with a few issues of a recurring nature involving compensatory firewatch implementation (reference inspection report 95-21), the control of locked high radiation area doors (reference inspection report 95-08), and the resolution of IST program deficiencies discussed in Section 4.1 of this report.

The inspector observed that the plant staff involvement and contribution to the ER screening meetings has matured since the inception of the program in February 1995. This was reflected in generally more detailed technical discussions and an increased emphasis on improving performance. The multi-disciplined management review has had a positive impact on communications between plant organizations. This was particularly evident during the resolution of deficiencies involving Appendix R emergency lighting when priorities were appropriately modified to restore operability and short term corrective actions were implemented to provide confidence that the problem would not recur.

In summary, the initial management reviews of ERs has resulted in a good systematic and generally comprehensive review of problems and deficiencies. Conservative decision making has been routinely observed in the implementation of immediate corrective actions and operability determinations. With a few exceptions, the initial VY management review via the ER screening meeting has contributed to plant safety and enhanced VY's corrective action process.

6.3 Review of Written Reports

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

- LER 95-14 and LER 95-14 Supplement 1, Incomplete implementation of 10 CFR 50, Appendix R, based on identified deficiencies in the safe shutdown capability analysis, dated August 24, 1995 and September 20, 1995, respectively.

These LERs documented VY's discovery and ongoing review of multiple deficiencies in their Appendix R Safe Shutdown Program (reference inspection reports 95-19 and 95-21). As stated in Supplement I, the various deficiencies identified, to date, have been collected for disposition by a Project Team, established on September 10, 1995. The Project Team's charter, in addition to dispositioning the identified problems is: to rewrite the Safe Shutdown Capability Analysis; to identify any necessary design changes; and to ensure the Appendix R program becomes a well documented and comprehensive program.

NRC staff review of this issue was ongoing as of the close of this inspection period and was planned to be documented in inspection report 95-25. As stated in these LERs, the NRC resident inspectors and region based specialists have been kept well appraised of the VY staff findings and their efforts to resolve these issues via periodic briefings and telephone calls.

- LER 94-08 Supplement 2, HPCI/RCIC system inoperable due to low spiking of level transmitter LT-2-3-72B instrument loop, dated August 29, 1995.

Supplement 2 documents VY's inability to identify the root cause for down-spiking of level transmitter LT-2-3-72B. Vendor troubleshooting and destructive examination was inconclusive.

- LER 94-16 Supplement 1, Unisolable service water (SW) piping leaks resulting in inoperability of the SW subsystems and the alternate cooling subsystem, dated August 4, 1995.

Supplement 1 documents the results of VY's metallurgical analysis of the through-wall leaks in the service water discharge piping to the "A" reactor building closed cooling water heat exchanger. Analysis identified microbiological induced corrosion (MIC) as the cause. The MIC attack sites on the piping were the result of improper pipe fit-up and lack of fusion at a root of the weld.

- LER 95-15, Technical Specification 4.7.A.4 leakage rate exceeded due to leakage from the inboard flange of valve AC-8, dated August 18, 1995.

This LER documented VY staff discovery of primary containment isolation valve leakage in excess of TS limits. Valve AC-8, an 18-inch diameter butterfly valve (drywell purge and supply isolation) was identified during the April 1995 containment integrated leak rate test to have an inboard flange leak in excess of the TS limit of 0.0400 Wt%/day (as found leakage was 0.1316 Wt%/day). The bolted flange was tightened and the leakage path eliminated. Detailed follow-up by the VY staff identified that the local leak rate testing methodology for this valve and seven similarly configured valves was inadequate to verify their leak tight integrity per 10 CFR Part 50, Appendix J. However, none of the seven additional valves identified failed during the April 1995 integrated leak rate test.

The inspector concluded that the VY staff took appropriate action to remedy the valve AC-8 flange leak and to identify similarly deficient leak rate testing of seven other containment isolation valves. As stated in LER 95-15, the VY staff had earlier opportunities (in 1992 and 1994) to identify this type of testing inadequacy, but did not adequately follow-up on the industry operating experience and vendor recommendations, respectively. The inspector notes that VY follow-up of identified problems has been an area of historical regulatory concern. Improvement in this area of performance has been noted, as depicted by the corrective actions for this April 1995 event. In that, this TS violation was licensee identified, of minor safety significance, and the corrective action prompt and thorough, this violation was not cited, consistent with Section IV of the NRC's Enforcement Policy.

7.0 MANAGEMENT MEETINGS

7.1 VY Management Visit

Senior VY managers visited the NRC Region I office on October 24. The Vice Presidents of Operations and Engineering and the Director of Quality Assurance met with the Director of the Division of Reactor Safety and other members of the Region I staff to discuss VY performance. Included in the discussions were VY's revised planning process for 1996 operations and the status of efforts to complete the planned reorganization of the various engineering groups.

7.2 Regional Administrator's Tour of the Facility

On November 3, Thomas T. Martin, Regional Administrator, NRC Region I, toured VY and held discussions with plant management and the Vice President, Engineering. Discussions focused on Engineering Department performance, use of enforcement discretion, industry events, and VY maintenance practices.

7.3 Preliminary Inspection Findings

Meetings were held weekly 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. A final exit meeting was conducted on November 22, 1995. No proprietary information was identified as being included in this report.