U.S. NUCLEAR REGULATORY COMMISSION
REGION I

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Report No.	96-08
Licensee:	Vermont Yankee Nuclear Power Corporation
Facility:	Vermont Yankee Nuclear Power Station
Location:	Vernon, Vermont
Dates:	July 14 - August 31, 1996
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EXECUTIVE SUMMARY

Vermont Yankee Nuclear Power Station
NRC Inspection Report 50-271/96-08

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 7-week period of resident inspection; in addition, it includes the results of announced inspections by regional specialists.

Operations

The inspector concluded that the licensee's efforts to improve the overall adequacy of the corrective action processes, as defined in AP-0009, Event Reports, have been generally effective (exceptions noted in recent past inspection reports) and closed unresolved item URI 93-33-01. The adequacy of the VY root cause analysis in the area of human performance related events continues to be monitored by the inspectors.

Inspector review of the most recently published Institute of Nuclear Power Operations station evaluation (performed January 1996) identified no observations or assessments inconsistent with recent NRC observations or assessments of VY staff performance.

Maintenance

The reactor core isolation cooling (RCIC) system limiting condition for operation (LCO) maintenance had been thoroughly planned, and received a strong supporting engineering review and appropriate management review. Adequate support for the maintenance activities was available, as evidenced by completion of the system outage within the planned duration. The sequencing of the RCIC room re-vitalization per the security plan and the subsequent RCIC operability testing was adequate.

Operator performance during observed emergency diesel generator surveillance testing was adequate and the resolution of an ambiguity in the load limit settings identified by the inspector was appropriately resolved.

The East and West switchgear room enclosure integrity testing was appropriately developed and executed by the VY staff. The test results for both switchgear rooms demonstrated that the CO_2 suppression systems could not perform their intended design function. Design changes to address testing deficiencies have not yet been determined by the VY engineering staff and will be reviewed in a subsequent inspection period. This item is unresolved **(URI 96-08-01)**.

The new fuel inspections were adequately performed, in that no discrepancies were observed. Although the personnel involved in the process were very knowledgeable and experienced, the inspector noted that some aspects of the fuel assembly inspections were done more quickly than necessary. For example, performing the procedure from memory reduced the effectiveness of self-checking and placed a greater reliance on peer verification. Given the large number of individual components in a fuel assembly and the

observed inspection time per assembly, the inspector considered that the effectiveness of visual inspection could only have been enhanced by taking more time to inspect. Occasional channel impacts on the upper fuel pin spacer during channel installation suggested that closer attention to this operation was warranted.

The inspector's review of the numerous corrective actions to address the valve FCV-6-12A failure found those corrective actions to be appropriate. Similarly, the actions to clarify non-asbestos worker activities during asbestos abatement events were found to be appropriate. During this inspection period, the inspector monitored troubleshooting of observed oscillations of the "A" feedwater regulating valve (determined to be flow induced and characteristic of the feedwater control system response at that reactor power level) and concluded that the VY staff took proper precautionary measures while investigating the cause of the oscillations. The VY staff conducted a thorough and self-critical review of the December 8, 1995 reactor scram and their corrective actions were appropriate. Unresolved item 95-25-01 is closed.

Engineering

The installation of the new reactor water clean-up system bypass line, per engineering design change request 95-404, was well controlled and properly designed as noted in the thorough safety evaluation and detailed installation and test procedure.

Overall, the Appendix R design modifications were of good quality, well supported with sound technical design basis, and in conformance with the established design control procedures. The inspector noted that the licensee had made good progress in addressing Appendix R program-related issues. However, significant work remains for completion prior to restarting the unit from the Fall 1996 refuel outage.

VY had established a cohesive process for requiring qualitative reviews of OE information and tracking subsequent actions for satisfactory resolution.

Good efforts had been implemented to improve programs and program activities within the Performance and Project engineering departments. Based on review of selected self-assessments, Functional Area Assessments (FAAs) completed, and interviews with various staff level personnel, the inspector determined that self-critical evaluations had been performed and aggressive efforts had been displayed to improve engineering performance. The inspector determined that VY's self-assessment activities provided for effective evaluations and good performance monitoring.

VY's Event Report (ER) process effectively incorporated the use of root cause evaluations for identifying and resolving plant issues. Good training had been provided to those employees tasked with evaluating such issues. The inspector found that a reliable process existed to support consistent evaluations of issues and to provide useful information of VY performance.

The licensee properly accounted for the usable volume of the fuel oil day tank in determining the level of fuel oil (converted to gallons) available for diesel-driven fire pump operation. The calculation assumptions were reasonable and conservative, the

mathematical analysis precise, and the results were within TS requirements for the period in question. Unresolved item 94-08-02 is closed.

The licensee's response to the containment penetration piping over-pressurization event was appropriate. The immediate operability assessment was reasonably founded and was promptly reviewed by PORC. The BMO, although not quantitative for the specific instances in question, provided reasonable assurance that the applicable containment penetrations were operable. However, as the licensee noted in ER 96-0474, the original code of construction for piping, ANSI-B31.1 1967, requires that reliefs be provided to piping that could exceed its design limits if it becomes isolated. Pending VY staff resolution and evaluation by the NRC staff, this event remains unresolved **(URI 96-08-02)**.

VY had taken appropriate actions to heighten the awareness of plant personnel for identifying and addressing human performance issues. The inclusion of this focus into FAAs and self-assessments was considered by the inspector to be a generally good means for re-emphasis on this topic.

VY staff's root cause assessment and corrective actions, for a violation of TSs involving an unreviewed temporary plant modification, were found thorough and appropriate to prevent recurrence. The root cause was identified as a failure of management systems to promptly implement corrective actions for a previous temporary modification implementation problem. This root cause was symptomatic of a weak corrective action process and poor staff performance with respect to proper evaluations of temporary plant modifications. Violation 94-31-01 is closed.

The VY staff implemented appropriate interim administrative controls for the apparent lack of TS and licensing basis clarity for the service water and alternate cooling systems. Conversion to standard TS, targeted for implementation in mid-1998, was planned by the licensee to ultimately resolve this item. Unresolved item 93-33-02 is closed.

Plant Support

Vermont Yankee continued to maintain an overall effective program for radioactive material and radioactive waste management and transportation, including; audits and appraisals; program changes and facility condition versus the FSAR; personnel training and qualifications; the solid radioactive waste management program; and the shipping program including implementation of the revised Department of Transportation regulations. Review of radiological waste management and transportation activities indicated that the UFSAR had been appropriately maintained. A previously identified violation (VIO 95-18-01) regarding access controls to high radiation areas was closed due to the comprehensive and effective corrective actions taken by the licensee's staff.

Audits and appraisals by the licensee's staff continued to improve the quality of the radiological controls program. Some personnel changes had been made in the recent past and all personnel were determined to be trained and qualified as appropriate. Facility tours indicated good material conditions and housekeeping in areas used for processing, staging, and storing radioactive materials and radioactive waste. Radioactive waste and radioactive material shipment records were maintained in very good order. Program procedures

appropriately incorporated the changes to the DOT and NRC regulations, and training on the revised regulations was adequate.

The licensee maintained an effective security program. Management support is ongoing, as evidenced by the completion of the vehicle barrier system and the procurement of training aids for tactical weapons training. The central and secondary alarm station operators were knowledgeable of their duties and responsibilities and were not engaged in activities that would interfere with their response functions. Management controls for identifying, resolving, and preventing programmatic problems were effective. Maintenance of security equipment was being performed in a timely manner as indicated by minimal compensatory postings associated with security equipment repairs. The security force members were found to possess the requisite knowledge to carry out their assigned duties and the training program was found effective.

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DETAILS

Summary of Plant Status

Vermont Yankee (VY) operated at full power throughout this inspection and commenced reactor power coastdown beginning in early August. The refueling outage was scheduled to commence September 7, 1996.

The new Region I Regional Administrator, Hubert J. Miller, visited the VY facility on July 14 and 15 to tour the plant and interview plant management and staff.

On July 16, the NRC staff conducted a Plant Performance Review (PPR) in the Region I office covering the performance period of July 15, 1995 to July 16, 1996. The results of the PPR were documented by letter to VY dated August 6, 1996.

During the week of July 15, a region based specialist inspector conducted a review of the radioactive materials/waste management and transportation area. During the week of July 22, a region based specialist inspector conducted a review of the security and safeguards area. During the week of July 29, a region based specialist inspector conducted a review of engineering related activities. During the week of August 19, a region based specialist inspector conducted a follow-up inspection in the area of fire protection and reviewed proposed modifications developed to resolve Appendix R discrepancies. The results of these specialists inspections are integrated into this report.

I. Operations

O1 Conduct of Operations[1]

O1.1 10 CFR 50.72 Notification, August 19, 1996

At 1:20 p.m. on August 19, the VY control room operators notified the NRC staff (Event No. 30892) in accordance with 10 CFR 50.72, that the 10 CFR 50, Appendix R design requirements were not met by the East and West switchgear room CO_2 suppression systems to maintain a 50% concentration for 20 minutes (see additional inspection observations and findings in section M2). The control room staff's notification of the NRC was found to be in accordance with 10 CFR 50.72 reporting requirements.

O1.2 10 CFR 50.72 Notification, August 22, 1996

At 4:07 p.m. on August 22, the VY control room operators notified the NRC staff (Event No. 30912) in accordance with 10 CFR 50.72, that a 3-inch gap in the fire barrier wrapping (3M-Interam-E-54A material) of a reactor building cable tray was identified on August 15. The VY staff determined on August 22 that this deficiency may potentially have an adverse impact on the effected safety systems in the event of a fire. This condition, identified as being outside the plant design basis, will be further evaluated by the VY staff and reported, in writing, in accordance with 10 CFR 5[illegible].

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[1]Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

O1.3 10 CFR 50.72 Notification, August 27, 1996

At 1:12 a.m. on August 27, the VY control room operators notified the NRC staff (Event No. 30925) in accordance with 10 CFR 50.72, that an unplanned engineered safety feature (ESF) actuation occurred at 12:19 a.m. The ESF actuation involved a reactor protection system (RPS) half-scram and a group III (reactor building ventilation and drywell sampling systems) primary containment isolation system (PCIS) actuation. The cause of the RPS half-scram and PCIS isolation was the trip of the supply breaker 5ACB1A between the RPS bus "A" and control room panel 9-15 which caused a loss of the logic circuit power.

Investigation by the VY staff identified a loose connection on the load side lead to the breaker. The load side lead insulation was found brittle and discolored indicating the breaker tripped due to the excessive heating (poor electrical contact causing arcing). The damaged portion of the load side cable was removed, the terminal lug tightened (torqued), breaker cleaned and cycled satisfactorily, and the load side cable insulation was resistance checked. A visual inspection of the adjacent 5ACB1B supply breaker was performed and no discrepancies noted. Additional details of this event and licensee corrective actions will be documented in the Licensee Event Report (LER) to be submitted in accordance with 10 CFR 50.73.

O8 Miscellaneous Operations Issues

O8.1 Review of January 1996 INPO Evaluation

In accordance with the Inspection Procedure (IP) 71707 guidance, the inspector reviewed the most recently published station evaluation by the Institute of Nuclear Power Operations (INPO) conducted during January 1996. The inspector's review of INPO findings and performance assessments did not identify any items or assessments inconsistent with recent NRC observations or assessments of VY staff performance.

O8.2 (Closed) URI 93-33-01: Adequacy of the guidance provided for the classification and processing of conditions adverse to quality.

a. Inspection Scope

This unresolved item was initiated based upon the inspectors' observations, over a long period of time, of the adequacy of the licensee's corrective action program and of the licensee's poor review of a specific event involving the performance of corrective maintenance on the wrong control rod drive (CRD) unit. The actions taken by the licensee to address the specific CRD maintenance issues have been adequately implemented (reference corrective actions report (CAR) 93-59), but the actions to address the overall adequacy of the corrective action process have been the subject of NRC staff monitoring for some time.

b. Observations and Findings

To address this issue, the VY staff made some significant changes to the corrective action program following a review by the Corrective Action Task Force (reference inspection report 95-80). The Task Force made a number of recommendations which were captured in a new administrative procedure, AP-0009, Event Reports, which was implemented in early 1995. Since the implementation of the AP-0009, the resident inspectors and specialists have observed and assessed the VY staff's use of this corrective action system on a frequent basis (reference inspection report (IR) 94-08, section 5.1.2; IR 95-11, section 6.3; IR 95-17, section 2.3; IR 95-21, section 6.2; IR 95-23, section 6.2; and IR 96-02, section 6.1). Collectively, these observations have demonstrated that the licensee has improved the overall quality and timeliness of problem identification and resolution. Notwithstanding, a few isolated events (reference inspection reports 96-03, section 3.1 and 96-07, section E8.1) have occurred indicating the need for continued emphasis in this area to ensure the consistency of the event review and corrective action processes and to ensure the adequacy of root cause analyses involving performance related events.

c. Conclusions

The licensee's efforts to improve the overall adequacy of the corrective action processes, as defined in AP-0009, Events Reports, have been generally effective. The adequacy of the licensee's root cause analyses in the area of human performance related events continues to be monitored by the inspectors.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Observations

a. Inspection Scope

The inspectors observed portions of plant maintenance activities using inspection procedure 62707 to verify that the correct parts and tools were utilized, the applicable industry code and TS requirements were satisfied, adequate measures were in place to ensure personnel safety and prevent damage to plant structures, systems, and components, and to ensure that equipment structures, systems, and components, and to ensure that equipment operability was verified upon completion of post-maintenance testing.

b. Observations, Findings, and Conclusions

The inspectors observed portions of the following maintenance activity:

- Valve V13-39, reactor core isolation cooling (RCIC) system torus suction isolation motor operator replacement, observed on July 24, 1996

The inspector noted that the fasteners that attached the motor operator to the valve did not include lock washers, whereas lock washers had been used in this application on two similar motor operated valves in the torus suction line. The licensee determined that lock washers were shown in the manufacturer's design drawing. The licensee stated, however, that the lock washers were not required if the fasteners were properly tightened; in this case, the spring factor of the fastener serves to prevent loosening, and the lock washer acts only as a flat washer. The manufacturer included lock washers because they serve to prevent loosening if the fastener is not adequately torqued. Since the installation procedure (OP-5220) requires the fasteners to be tightened to a specific torque value, the licensee concluded that lock washers were not required. The licensee noted, however, that this condition represented a configuration control problem. Although the existing configuration was deemed acceptable, a non-conformance report was generated to document the condition. The inspector concluded the VY staff's actions to address this configuration control problem were appropriate. (Additional inspector observations made during RCIC system outage are discussed in section M1.2 of this report.)

M1.2 Reactor Core Isolation Cooling System Maintenance

a. Inspection Scope

The inspector reviewed the licensee's preparations for, and observed the conduct of, maintenance on the reactor core isolation cooling system that was performed during an elective entry into a limiting condition for operations (LCO) during the period July 22-26, 1996.

b. Observations and Findings

The licensee's procedure that governs elective entry into a limiting condition for operations (LCO) for the purpose of conducting planned maintenance is the "LCO-Maintenance Guideline." The inspector reviewed the RCIC system LCO maintenance plan that had been prepared pursuant to this guidance. The maintenance period was scheduled to last four days of the seven day RCIC system LCO. The plan had been reviewed by the operations superintendent, although such review was not required by the guideline for maintenance periods of less than 60 percent of the LCO period. The plan identified craft and engineering support requirements during this period, and specified that high risk surveillances would not be performed, nor would unnecessary work be performed, on any ECCS systems. Engineering review of the plan included a quantitative, IPE-based analysis which demonstrated that the change in yearly average risk due to the maintenance activity was less than the proposed industry threshold for risk significance. This review also identified systems that, from an IPE perspective, would become more important while the RCIC systems was not available; restrictions within the plan were consistent with maintaining maximum reliability of these systems.

During the RCIC system maintenance period, security requirements for the RCIC room were relaxed to facilitate access (room not required to be maintained a vital area while the RCIC system is inoperable). The LCO maintenance plan included the plan that was to be used for devitalizing and revitalizing the RCIC room. Although the plan was adequate in terms of physical security measures, the inspector noted that it did not require that the RCIC

room be re-established as a vital area prior to completion of RCIC post-maintenance acceptance testing. Due to ongoing maintenance on a ventilation damper, the licensee considered waiting to revitalize the area until after the completion of RCIC testing. However, the inspector verified that the RCIC enclosure was revitalized prior to the completion of surveillance testing.

c. Conclusions

The RCIC system LCO maintenance had been thoroughly planned, and received a strong supporting engineering review and appropriate management review. Adequate support for the maintenance activities was available, as evidenced by completion of the system outage within the planned duration. The sequencing of the RCIC room re-vitalization per the security plan and the subsequent RCIC operability testing was adequate.

M1.3 Surveillance Observations

a. Inspection Scope

The inspectors observed portions of surveillance tests using inspection procedure 61726 to verify proper calibration of test instrumentation, use of approved procedures, performance of work by qualified personnel, conformance to limiting conditions for operation (LCOs), and correct post-test system restoration.

b. Observations and Findings

The inspectors observed portions of the following surveillance test:

- OP-4126, Diesel Generators Surveillance, section C.1, Diesel Generator Readiness Demonstration Monthly, observed July 19, 1996

The inspector noted that adjustments are made to the governor speed droop and load limit settings prior to testing, and that the settings are returned to normal on completion of the test. The procedure indicates that the normal setting for the load limit is 10 (the scale is 0-10). On August 1, 1996, the inspector noted that the "A" emergency diesel generator (EDG) load limit was set to a value greater than 10 (approximately 11, if the scale was continued beyond 10), whereas the "B" EDG load limit was set at 10. The inspector informed operations personnel of this observation. Through discussions with the vendor, the licensee determined that the normal setting for load limit should actually be fully clockwise; depending on the particular governor, this may or may not correspond to the upscale value of 10. The licensee stated that the procedure would be revised to eliminate the ambiguity of the load limit settings.

c. Conclusions

Operator performance during observed emergency diesel generator surveillance testing was adequate and the resolution of an ambiguity in the load limit settings identified by the inspector was appropriately resolved.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Review of East Switchgear Room Enclosure Integrity Test

a. Inspection Scope

As one of a number of corrective actions for identified 10 CFR 50, Appendix R and Fire Protection Program deficiencies, the VY staff closely examined the adequacy of pre-operational testing of installed carbon dioxide (CO_2) fire suppression systems. Based upon this examination, no test data could be found for the East and West switchgear rooms' CO_2 suppression systems. The inspector reviewed the testing conducted by the VY staff and their vendor during the week of August 5, to demonstrate the design adequacy of these CO_2 suppression systems.

b. Observations and Findings

The inspector reviewed and discussed with the responsible test engineer special test procedure No. 96-009, East Switchgear Room Enclosure Integrity Test, dated 8/1/96. This test used methodologies endorsed by NFPA 12A-1992, and previously approved by the NRC staff, to demonstrate that a CO_2 discharge to the room would achieve and maintain the necessary CO_2 concentration to suppress a design deep-seated fire. One portion of the test involves the use of a tracer gas (sulfur hexaflouride - SF_6) released to attain a maximum concentration of four parts per million (ppm) in the room. This initial concentration is then monitored for approximately one hour to determine the decay and air exchange rate of the room. Another portion of the test involves a door fan test which, with the aid of a computer and associated data collection instrumentation, models the pressurization and depressurization of the room to determine the approximate leakage area and the calculated CO_2 retention time. These tests were completed during a two-day period on the mid-shift to minimize the impact on routine operations, maintenance, and surveillance testing.

The inspector determined that during the first tracer gas test run of the East switchgear room on August 8, the gas concentration decay was more rapid than expected. Per procedure, the West switchgear exhaust fan was secured and the tracer gas test rerun. The significance of this test configuration change was that the West switchgear exhaust fan does not automatically secure upon initiation of the East switchgear CO_2 suppression system and vice-versa. Thus, if the tracer gas test data demonstrates that this configuration is needed to assure adequate CO_2 concentration, than a design change to the initiation logic would be warranted.

Following data reduction by the vendor during the week of August 12, the VY staff determined that both East and West switchgear room enclosure integrity test results were unsatisfactory. These test failures and CO_2 system design inadequacies were reported in accordance with 10 CFR 50.72 on August 19. Consequently, CO_2 suppression system design changes appear warranted for both switchgear room CO_2 systems to ensure they meet the minimum design specifications.

c. Conclusion

The East and West switchgear room enclosure integrity testing was appropriately developed and executed by the VY staff. The test results for both switchgear rooms demonstrated that the CO_2 suppression systems could not perform their intended design function. Design changes to address testing deficiencies have not yet been determined by the VY engineering staff and will be reviewed in a subsequent inspection period. This item is unresolved **(UNR 96-08-01)**.

M7 Quality Assurance In Maintenance Activities

M7.1 Receipt Inspection and Fuel Channeling of Reactor Fuel Assemblies

a. Inspection Scope

The inspector observed new reactor fuel handling activities, from unloading from the shipping crates through placement into the spent fuel pool (SFP) using the guidance of IP 60705.

b. Observations and Findings

On July 23, 1996, the inspector observed the unloading, inspection, and storage of several new reactor fuel assemblies. This activity was performed in accordance with OP-1401, New Fuel Inspection and Channelling. Inspections were performed by two maintenance personnel who were specifically trained and qualified for new fuel inspection. For each fuel assembly, tasks prior to movement in the spent fuel pool (unloading from the shipping crates, removal of shipping supports, verification of proper rod gap, installation of the channel, and transfer to the SFP) were all performed from memory, and then recorded in the master procedure. This practice is allowed by the procedure, since OP-1401 is classified as "for reference use". The inspector noted no occurrences of missed inspection requirements. One reactor engineer was in overall control of the inspection activities and maintained the signature copy of the procedure. The work progressed rapidly, requiring approximately 20 minutes from the start of unloading until transfer into the SFP. The entire new fuel inspection was completed in seven days; historically, this operation has taken 10-11 days.

On several occasions, the inspector observed that the fuel channel impacted the first fuel pin spacer while it was being lowered onto the fuel bundle. Although no movement of the spacer was observed on these occurrences, the inspector noted, in several other cases, that contact was easily avoided by hand-guiding the channel until it passed over the first fuel pin spacer.

c. Conclusions

The new fuel inspections were adequately performed, in that no discrepancies were observed. Although the personnel involved in the process were very knowledgeable and experienced, the inspector noted that some aspects of the fuel assembly inspections were done more quickly than necessary. For example, performing the procedure from memory

reduced the effectiveness of self-checking and placed a greater reliance on peer verification. Given the large number of individual components in a fuel assembly and the observed inspection time per assembly, the inspector considered that the effectiveness of visual inspection could only have been enhanced by taking more time to inspect. Occasional channel impacts on the first fuel pin spacer during channel installation suggested that closer attention to this operation was warranted.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) URI 95-25-01: Review of licensee's root cause analysis for December 8, 1995 reactor scram.

a. Inspection Scope

This unresolved item involved the review of the pending root cause evaluation by the VY staff of the December 8, 1995 reactor scram. Preliminary review by the inspector identified that the automatic scram may have been preventable (reference IR 95-25, section 2.2). The licensee conducted two separate root cause evaluations to address difference aspects of this event. The inspector examined these evaluations to assess the licensee's thoroughness in evaluating staff performance and the adequacy of corrective actions.

b. Observations and Findings

One root cause evaluation (tracked by VY internal commitment number INS952501), addressed the adequacy of operations staff actions taken following the pinning of the "A" feedwater regulating valve. Of particular interest was whether or not the stationing of the auxiliary operator in the turbine lube oil room (adjacent to the feed pump room and out of direct line of sight with the pinned feedwater regulating valve) was an appropriate action given the potential for asbestos fiber exposure. The reactor scrammed due to the uncontrolled reactor water level transient following the locking pin vibrating loose.

Following the initial transient and valve pinning, the control room staff's judgement to station the AO in the turbine lube oil room was based upon a concern that the AO would be exposed to asbestos. Asbestos fibers had been shaken from the feedwater piping insulation during the earlier transient and had settled on the floor of the feedpump room. Maintenance workers (certified asbestos workers) were cleaning-up the asbestos in order to restore normal access to the feedpump room. The AO (as with all other AOs) was not a trained (certified) asbestos worker. The control room operators were aware of the fact that the locking pin was not fully engaged, however, it was concluded that the difficulty in engaging the pin partially gave some degree of confidence that it would not vibrate out.

Follow-up investigation and review by the VY staff of Vermont Occupational Safety and Health Agency and State Department of Health regulations determined that a person who enters an asbestos abatement area to perform non-asbestos work does not need to be a certified asbestos worker. That individual need only be aware of the hazards and wear protective clothing similar to the asbestos workers, including a respirator if warranted. Based upon this determination, AP 0509, Working with Asbestos, was revised (Revision 7,

dated 5/15/96) to address emergency type situations involving small quantity asbestos releases and worker protective clothing and activity limitations. This information was also incorporated into General Employee Training and operator awareness training.

Based upon the above, the inspector determined that, in hindsight, the AO charged with monitoring the "A" feedwater regulating valve while it was in manual control (pinned) could have been placed in protective clothing and a respirator in order to maintain visual contact with the pinned valve. This is not to say that the AO would have prevented the pin from vibrating loose, but he would have been afforded the opportunity to have potentially prevented the event.

The second root cause evaluation was conducted by the Instrumentation and Controls (I&C) department staff to assess the cause of the feedwater regulating valve (FCV-6-12A) piston rod extension disconnecting from the piston rod. The I&C staff determined from a review of completed work orders that both feedwater regulating valves were rebuilt during the 1995 refuel outage. During post-maintenance testing of FCV-6-12A, the air operator failed and had to be repaired. The I&C staff concluded that during the second reassembly of FCV-6-12A, proper re-torque of the valve piston rod assembly (operator shaft) was not conducted. Consequently the threaded fitting between the piston rod and piston rod extension disengaged, causing the valve disc to "flutter" in the feedwater flow stream. The licensee determined that factors contributing to this failure to properly re-torque the piston rod assembly were: valve operator design; ineffective training of the technicians involved; poor planning with respect to review of vendor advisory information; maintenance procedure (OP-5353) not updated with new torque requirements; and poor maintenance supervisory oversight.

The inspector learned from interviews that past troubleshooting activities to identify the cause for excessive feedwater regulating valve oscillations were similarly addressed, with respect to conducting a down power maneuver (to less than 50 percent reactor power to remove the valve from service) and observing valve response. Previous operating experience did not demonstrate the need for a more conservative approach (pinning the valve prior to commencement of the down power maneuver). The inspector learned that diagnostic performance data on valve FCV-6-12A gathered prior to December 8 was not analyzed prior to commencement of the down power maneuver to remove FCV-6-12A from service. Based upon the vendor's experience, the data taken prior to the event could not have conclusively demonstrated the type of problem which was subsequently identified by valve disassembly. Review of that data, subsequent to the reactor trip and identification of the root cause being valve operating piston separation, merely clarified what the data represented.

c. Conclusions

The inspector's review of the numerous corrective actions to address the valve FCV-6-12A failure found those corrective actions to be appropriate. Similarly, the actions to clarify non-asbestos worker activities during asbestos abatement events were found to be appropriate. During this inspection period, the inspector monitored troubleshooting of observed oscillations of the "A" feedwater regulating valve (determined to be flow induced and characteristic of the feedwater control system response at that reactor power level)

and concluded that the VY staff took proper precautionary measures while investigating the cause of the oscillations. The VY staff conducted a thorough and self-critical review of the December 8, 1995 reactor scram and their corrective actions were appropriate. Unresolved item 95-25-01 is closed.

III. Engineering

E1 Conduct of Engineering

E1.1 Reactor Water Clean-up Bypass Line Installation Review

a. Inspection Scope

During the weeks of August 5 and 12, the licensee commenced installation activities for a new reactor water clean-up (RWCU) system bypass line. The purpose of the new bypass line is to permit reactor water clean-up system operation while shutdown when the feedwater system (normal return to the reactor vessel) must be secured. During normal power operations this bypass line will be isolated via newly installed isolation valves. Using inspection procedure 37551, the inspector reviewed the engineering design change package, associated safety evaluation, and the installation and test (I&T) procedure, and discussed bypass line installation activities with the cognizant engineer while observing work in the field.

b. Observations and Findings

The inspector determined that the new 2-inch diameter stainless steel RWCU bypass line design change was part of Engineering Design Change Request (EDCR) 95-404, Feedwater Check Valve Replacement. As the title implies, the major portion of this design change package consists of the replacement of the installed Walworth Y-lift check valves with new Anchor/Darling swing check valves. The feedwater check valve replacement effort is scheduled to be conducted during the upcoming refuel outage.

The inspector examined the entire EDCR 95-404 package, focusing on the RWCU bypass line design changes, and identified an excellent description of the changes, including: scope, operational description, design basis impact, installation methods, and other operational and maintenance considerations. The safety evaluation (enclosure A to EDCR 95-404) was likewise sufficiently detailed and thorough. The inspector noted a number of EDCR package review comments were made prior to Plant Operations Review Committee approval and that all of the questions and comments were satisfactorily addressed or resolved. The inspector also reviewed the RWCU bypass line I&T procedure and followed its use during bypass line installation. The I&T procedure prerequisites and precautions were verified and found appropriate and properly adhered to.

The inspector verified proper isolation of the RWCU system to conduct the necessary field work and noted appropriate radiological controls being implemented during the breaching of the RWCU system at RWCU-Spool-1. Although the welding of the residual heat removal (RHR) line sockolet-type fitting was completed, the 8-inch RHR process line will not be breached until the refuel outage. The inspector verified that the new bypass line was

adequately supported (new pipe hangers installed and attached) prior to the RWCU system being returned to service.

c. Conclusions

The installation of the new reactor water clean-up system bypass line, per engineering design change request 95-404, was well controlled and properly designed as noted in the thorough safety evaluation and detailed installation and test procedure.

E1.2 Conduct of Engineering - Appendix R Program Review Follow-up

Background

During July 1995, the licensee identified a number of areas in which the fire protection program did not satisfy the requirements of the Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," (Appendix R). During October and November 1995, an inspection was conducted to review VY's short term and long term corrective actions to address the Appendix R deficiencies in their program, as documented in NRC inspection report No. 50-271/95-26, dated December 22, 1995. As a result of that inspection, an enforcement conference was held on January 11, 1996. Following the enforcement conference, a Notice of Violation (NOV) to the licensee was issued on February 13, 1996.

In the response to the NOV, dated March 12, 1996, the licensee agreed with the violation and identified the short term compensatory measures which they had instituted and stated that VY intended to achieve full compliance with 10 CFR 50, Appendix R, requirements prior to startup from the Fall 1996 refuel outage. The licensee's long term corrective actions identified in their response included the following:

1. Verify the safe shutdown capability analysis (SSCA) design and strengthen and clarify the Appendix R design basis documentation, as necessary. Issuance of the revised SSCA was targeted for prior to startup from the Fall 1996 refuel outage. In addition, provide Appendix R training by the end of 1996 to engineers responsible for designing, reviewing, or implementing modifications.

2. Achieve full compliance with Appendix R requirements, without the need of compensatory measures, by startup from the Fall 1996 refuel outage. This action includes design changes, analysis, exemption approvals, and enhanced operating procedures, as necessary.

3. Institute design changes to eliminate reliance on fuse replacement, for achieving and maintaining hot shutdown, prior to startup from the Fall 1996 refuel outage. After implementation of these design changes, VY intends to withdraw the existing exemption.

4. Licensing documentation supporting all Appendix R exemptions would be reviewed for accuracy and consistency prior to startup from the Fall 1996 outage by the Appendix R Project Team.

5. Line organization self assessments of Safety Evaluation Reports (SERs) in other areas (EQ, IST, and Appendix J) would be completed by the end of 1996, including Quality Assurance emphasis in this area via the planned program audits.

6. Review the process for submitting licensing requests and reviewing NRC SERs to identify any improvements that would ensure completeness of communications by the end of 1996.

At the time of this inspection, the licensee had not fully established and independently reviewed the SSCA and Fire Hazard Analysis (FHA) documents. Therefore, the primary focus of this inspection was to review the quality of the design modifications initiated to resolve deficiencies related to the Fire Protection and Appendix R programs.

a. Inspection Scope

Using inspection procedure 64150, the inspector reviewed selected design modification packages developed to comply with the Appendix R requirements and walked down those modifications partially implemented to assess their quality and conformance with the applicable installation and testing requirements. The following Engineering Design Change Requests (EDCRs) were reviewed:

- EDCR No. 96-401: Appendix R alternate shutdown system redundant fuses modification.

- EDCR NO. 96-402: Vernon tie upgrade for Appendix R and Loss of Offsite Power (LNP) circuitry modification.

- EDCR No. 96-403: Appendix R/Safety Relief Valve (SRV) hot short protection modification.

- EDCR NO. 96-404: Appendix R drywell temperature indication modification.

- EDCR No. 95-407: Motor Operated Valve (MOV) design improvements including Appendix R hot short modifications.

- EDCR NO. 96-411: Residual Heat Removal (RHR) minimum flow V10-16A/B and Air Recirculation Unit (RRU) 7/8 Appendix R circuit modification.

b. Observations and Findings

The inspector reviewed the above modifications and noted that these design modifications, initiated by the Yankee Nuclear Service Department (YNSD), were based on the new Appendix R adopted strategies to make the alternate shutdown safety systems available in the event of fires in various plant fire areas. EDCR 96-401, in particular, was issued to address the exemption-related problem (more than the allowed systems) discussed in the 1995 inspection report. EDCR 96-401 eliminates the need to replace the fuses in various alternate shutdown systems in the event of a fire in various plant areas. Per the new design, the existing system fuses and postulated damaged control circuits would be

isolated and the new set of redundant fuses and associated circuits would be connected to these systems manually by operating the switches on the Appendix R alternate shutdown system panels. The inspector determined that the licensee plans to withdraw the existing approved exemption (for fuse replacements) upon completion of this modification and achieving full compliance with Appendix R.

Inspector review of the above design documentation identified that the licensee had appropriately evaluated the impact of these design changes on existing design calculations, the FSAR, TS requirements, 10 CFR 50.59 safety evaluations, and other applicable design documents, where applicable. For the EDCRs reviewed, the inspector noted that where specialty group reviews were needed (i.e., seismic support analysis, equipment qualification reviews, and fire hazard analysis impact), applicable and appropriate analyses were completed to support the specific design change. The inspector, by system walk-down, verified that the partial installation of the new SRVs conduit routing and cables (EDCR 96-403) and alternate shutdown panel modifications were consistent with the design documentation, installation and test procedures, and interim completed Appendix R SSCA.

Based on the above review, the inspector determined that the licensee had made good progress in completing the non-outage work. However, a significant amount of work remained to be accomplished on these modifications during the outage, such as cable termination and installation of components in the operating panels, motor control centers, switchgear, and other normally energized components. Per discussion with the licensee, the inspector re-affirmed that the licensee intends to complete all fire protection-related modifications prior to restart from the Fall 1996 refuel outage.

c. Conclusions

Overall, the Appendix R design modifications were of good quality, well supported with sound technical design basis, and in conformance with the interim strategy basis of the new SSCA and the established design control procedures. As noted above, at the completion of the inspection, VY had not fully established and independently reviewed the SSCA and FHA, or issued Supplemental 3 of Licensee Event Report 95-14 which captures a number of fire protection-related issues identified since inspection 95-26. Accordingly, these items will be examined in subsequent NRC inspections in conjunction with follow-up of violations 95-268-01013 and 02014.

E2 Engineering Support of Facilities and Equipment

E2.1 Operating Experience Reviews

a. Inspection Scope

Using IP 37550 guidance, the inspector reviewed VY's internal responses to selected NRC Information Notices (INs) and a Generic Letter (GL) to assess the quality of their process for responding to technical issues and operating experience. The selected NRC documents were:

- GL 96-01 Testing Of Safety-Related Logic Circuits
- IN 96-24 Preconditioning Of Molded-Case Circuit Breakers Before Surveillance Testing
- IN 95-21 Unexpected Degradation Of Lead Storage Batteries
- IN 96-10 Potential Blockage By Debris Of Safety System Piping Which Is Not Used During Normal Operation Or Tested During Surveillances

VY's review process of operating experience and technical issues was prescribed in administrative procedure AP-0028, Revision 16, "Operating Experience Review And Assessment/Commitment Tracking."

b. Observations and Findings

The inspector interviewed the operating experience (OE) specialist responsible for screening the OE information and including it in the commitment tracking program, a database system. The inspector found the specialist to be knowledgeable and very familiar with procedures and requirements. The inspector found that all NRC OE had been appropriately assigned for further engineering review. However, none had been dispositioned within the originally assigned due dates. A noteworthy backlog of engineering work activities has been a topic of NRC staff concern. Due to VY's efforts to resolve emergent plant technical issues, a significant reduction in backlog has not been achieved. The inspector noted that several engineering self-assessment reports addressed the need to reduce backlog work and that increased emphasis was planned following completion of the scheduled refueling outage (scheduled for completion in early October 1996).

The inspector found that the OE screening process required reviewers of information to continue review efforts even if VY did not have the specific component type which was identified in the OE document. The OE procedure requires further review to determine whether similar plant components could be vulnerable to the type of failure discussed in the OE description. The inspector considered this approach to be noteworthy for assessing pertinent information.

The inspector found that the tracking of OE commitments sampled identified the responsible individual(s) and an appropriate status of the item. The inspector verified that internal commitments requiring additional staff action were also included in the review process and tracked for resolution.

VY initiated a multi-disciplinary task team in March 1996, per the plant manager's request, to re-evaluate the adequacy of the operating experience (OE) review process, provide a comparison of the OE program against industry leaders, implement improvements, and establish performance indicators for the program. At the time of this inspection, the task team had not completed their evaluation. The inspector noted the OE specialist was a member assigned to the task team.

c. Conclusions

VY had established a cohesive process for requiring qualitative reviews of OE information and tracking subsequent actions for satisfactory resolution.

E2.2 Self-Assessments

a. Inspection Scope

Using IP 37550 guidance, the inspector reviewed several self-assessments completed by the Performance and Project Engineering departments. This review was performed to assess the efforts by these departments for effectively identifying and correcting deficiencies associated with engineering performance.

b. Observations and Findings

The inspector reviewed several self-assessments that implement VY Policy 115, "Vermont Yankee Self-Assessment Policy." This policy established a tool for engineering and other plant departments to continuously improve performance by questioning practices, identifying problem areas, recommending areas for improvement, and aggressively pursuing resolution of issues.

Several engineering activities were evaluated by the VY engineering staff to verify the adequacy of work performed. These activities included system testing practices, limiting condition for operation performance problems, and modification development and implementation practices. Weaknesses, strengths, and planned actions were identified. This feedback was included in the engineering monthly report for discussion at routine department meetings, and used by department managers for assessing performance within their responsible areas.

In addition to departmental self-assessments that are conducted twice each month, the inspector found that an annual Functional Area Assessment (FAA) was required to be performed. The goals of FAA included the identification of those areas for improvement that could be monitored throughout the year and an action plan that prioritized such needs. The inspector also found that monitoring of engineering performance had been conducted and additional actions had been planned by the QA department. Additional QA surveillances with enhanced audit scopes had been completed by QA of numerous engineering programs including: motor-operated valves; inservice inspection; inservice testing; environmental qualification; Appendix R; and preventive maintenance.

c. Conclusions

Good efforts had been implemented to improve programs and program activities within the Performance and Project engineering departments. Based on review of selected self-assessments, FAAs completed, and interviews with various staff level personnel, the inspector determined that self-critical evaluations had been performed and aggressive efforts had been displayed to improve engineering performance. The inspector determined that VY's self-assessment activities provided for effective evaluations and good performance monitoring.

E2.3 Corrective Action Process

a. Inspection Scope

Using IP 37550 guidance, the inspector reviewed the corrective action process to assess VY's effectiveness for identifying, evaluating, and correcting plant issues. This review addressed the association between the event report (ER) process, with root cause evaluations, and the trending of data indicative of plant performance to ensure problems are identified, evaluated, and corrected properly and consistently.

b. Observations and Findings

The inspector reviewed administrative procedure AP-0009, Revision 3, "Event Reports," and observed three event screening panels held to evaluate recently initiated ERs. The inspector noted that conservative decisions had been made by the panel members (plant department managers) when categorizing ERs into one of the four significance level categories. Per procedure, levels 1 and 2 ERs were of the greatest potential/safety significance and required a root cause analysis. Level 3 ERs required either a root cause or apparent cause analysis.

Root cause analysis training had been provided to the engineering staff and a root cause analysis guideline established to supplement this training for those tasked with performing an analysis. The inspector found that the information provided to the engineering staff for performing a root cause analysis was well developed and contained objective evaluation criteria. A Root Cause Specialist had been assigned to independently evaluate each ER to verify an analysis was performed, when necessary.

The inspector found that the Root Cause Specialist had established performance indicators used to track and trend ER parameters for use by various VY departments. ER parameters included: cause; event type; and activity type. The inspector found that the ER parameter information enabled personnel to analyze events and determine what causes were most or least prevalent. VY department managers discussed their plans to develop and implement further actions to reduce the number of ERs and subsequently improve plant performance. Engineering initiatives planned by VY are discussed in section E8.1.

c. Conclusions

VY's ER process effectively incorporated the use of root cause evaluations for identifying and resolving plant issues. Good training had been provided to those employees tasked with evaluating such issues. The inspector found that a reliable process existed to support consistent evaluations of issues and to provide useful information of VY performance.

E2.4 (Closed) URI 94-08-02: Maintaining the diesel fire pump fuel oil day tank above the minimum Technical Specification (TS) level.

a. Inspection Scope and Findings

During an earlier review, the inspectors identified a longstanding condition (approximately six weeks) where the diesel fire pump fuel oil day tank level had been recorded out-of-specification low (below the 9/16 full log specification) without any definitive action to restore the fuel oil level to above this minimum specified level. The lowest recorded level during this six-week period was 1/2 full. Per the governing operating procedure (OP-4105.1, Monthly Operational Check of Fire Pumps) a tank level of 15/32 full satisfies the TS minimum of 150 gallons. The inspectors reviewed the calculation of record which converted tank level to gallons and questioned whether the determination of usable volume properly accounted for tank level instrumentation, the tank slope, and the unusable sump volume at the bottom of the fuel oil tank. During this inspection period, the inspector reviewed calculation No. VYPC 94-010, dated November 23, 1994, in support of the VY staff's determination that the diesel fire pump fuel oil day tank was properly maintained greater than the 150 gallons TS minimum.

b. Conclusions

The licensee had properly accounted for the usable volume of the fuel oil day tank in determining the level of fuel oil (converted to gallons) available for diesel-driven fire pump operation. The calculation assumptions were reasonable and conservative, the mathematical analysis precise and the results were within TS requirements for the period in question. Unresolved item 94-08-02 is closed.

E2.5 Design Deficiency in Containment Piping Penetrations

a. Inspection Background and Scope

Fluid systems that penetrate the primary containment are generally provided with two containment isolation valves. This arrangement ensures that failure of one isolation valve will not result in a loss of containment integrity. In most cases, the piping between the valves does not include a relief path, so fluid in this section of piping is trapped if both valves are shut. During this inspection period, the licensee determined that, under some accident conditions, heat from the primary containment would cause the isolated fluid to expand, thereby pressurizing the associated piping beyond its design limit. If the pressure increase was great enough to cause the piping to rupture, a potential loss of containment integrity could result.

The inspector examined the VY staff's investigation of this issue and their interim resolution of this problem.

b. Observations and Findings

On July 25, 1996, the VY performance and design engineering staffs identified the above scenario for potential loss of containment integrity as a result of similar problems that had been identified at other commercial nuclear power plants (i.e. Maine Yankee). The concern was specifically for water systems which normally operate at low to intermediate temperature. The primary containment isolation valves would be closed following a loss of coolant accident (LOCA). The peak temperature in containment would be 280°F for the design basis large break LOCA, and 325°F for the design basis small break LOCA. Heat transfer by conduction would cause the water trapped between the two closed isolation valves to expand, and thereby increase pressure in the piping. If pressure exceeded design limits, it could result in single or combined failure of the two containment isolation valves and associated piping.

The licensee identified six plant systems that met the necessary initial conditions to be susceptible to this problem. The systems, and associated containment penetrations, are:

System	Penetration
1. Reactor Building Closed Cooling Water (RBCCW)	X-23 and X-24
2. Residual Heat Removal (RHR)	X-12
3. Main Steam Line Drain Lines (MSD)	X-8
4. Liquid Radioactive Waste (LRW)	X-18
5. Liquid Radioactive Waste (LRW)	X-19
6. Recirculation System Sampling Line	X-41

The licensee discussed this problem with Yankee Atomic Energy Corporation (YAEC) and General Electric, Nuclear Energy Technical Services (GE) to determine if an immediate operability concern existed. Through this discussion, it was determined that several pressure relief mechanisms exist that could preclude catastrophic failure. These included: 1) isolation valve seat leakage; 2) isolation valve stem packing leakage; and 3) ductile expansion of the piping. It was also noted that none of the systems in question were required to be operable for the plant to achieve safe shutdown. Based on these considerations, the licensee concluded that the condition did not represent an immediate operability concern, with respect to primary containment.

The concern for potential over-pressurization of containment penetration piping was documented in Event Report (ER) 96-0474, dated July 25, 1996. The following day, the Plant Operations Review Committee (PORC) reviewed the condition and concurred with the immediate operability assessment, pending development of a Basis for Maintaining Operation (BMO).

On July 31, 1996, PORC met to review BMO 96-11, Potential for Piping Associated with Containment Integrity to Exceed Design Limits. The BMO included a qualitative evaluation, prepared by GE, for each of the six identified susceptible penetrations. In all cases, except the recirculation system sampling line, the evaluations concluded that valve leakage mechanisms and expansion of the carbon steel piping will preclude structural damage to the piping. The recirculation system sampling line was determined not to be susceptible to

over-pressurization because the isolation valves are designed to relieve excess pressure. The BMO concluded that the pressure relief mechanisms in each specific case made the loss of containment integrity due to over-pressurization a low probability event. PORC concurred with this assessment. At the close of the inspection period, the licensee was developing a modification to install thermal relief valves on the affected portions of piping.

c. Conclusions

The licensee's response to this event was appropriate. The immediate operability assessment was reasonably founded and was promptly reviewed by PORC. The BMO, although not quantitative for the specific instances in question, provided reasonable assurance that the applicable containment penetrations were operable. However, as the licensee noted in ER 96-0474, the original code of construction for piping, ANSI-B31.1 1967, requires that pressure reliefs be provided to piping that could exceed its design limits if it becomes isolated. Pending further VY staff evaluation and final resolution and review by the NRC staff, this event remains unresolved **(URI 96-08-02)**.

E8 Miscellaneous Engineering Issues

E8.1 Engineering Initiatives

a. Inspection Scope

Engineering staff support of plant operations has been identified as a concern by the NRC staff and addressed by VY. In an effort to improve engineering and overall plant performance, VY has planned and initiated several projects. In VY's reply letter to the NRC, dated July 19, 1996, VY presented current and planned engineering improvement initiatives. In addition to those initiatives, efforts have been taken by VY to address and minimize human performance problems. The inspector held discussions with engineering staff and department managers, the acting plant manager, and the operations support manager to better understand those efforts.

b. Observations and Findings

The inspector was told that additional evaluations and considerations had been made to identify and correct human performance issues. Human performance laboratories have been developed at the training facility for technicians and maintenance workers to practice tasks without consequence. In addition, evaluation of human performance and potential errors had become a focus area in the functional area assessments and departmental self-assessments performed quarterly, as well as, during daily ER screening panel meetings. In addition, management's expectations for thorough planning and turnovers have been communicated to the staff via site publications and meetings. The inspector found that the engineering staff was aware of the need to identify, as well as, prevent human performance related problems.

c. Conclusions

VY had taken appropriate actions to heighten the awareness of plant personnel for identifying and addressing human performance issues. The inclusion of this focus into functional area assessments and self-assessments was considered by the inspector to be a generally good means for reemphasis on this topic.

E8.2 (Closed) VIO 94-31-01: Un-evaluated temporary modification

a. Inspection Scope

During an earlier inspection period, the inspector identified a violation of TS 6.5 involving the temporary use of a multi-channel recorder to monitor the master trip circuit of level instrument 2-3-72B (reference inspection report 94-31, section 4.1). The inspector identified that a proper safety evaluation to assess the potential safety impact of this monitoring device on the level instrument had not been performed. This failure to appropriately invoke the administrative controls and 10 CFR 50.59 evaluation requirements of temporary plant modifications was determined to have been repetitive in nature. Using IP 92903, the inspector reviewed the VY staff's root cause analysis and associated corrective action for this event, as discussed in their March 15, 1995 response to the Notice of Violation and associated documentation for Significant Corrective Action Report (SCAR) 95-005.

b. Observations and Findings

The VY staff attributed this event to a failure of management systems, in that, the implementation of corrective action for a previous similar event was untimely. As mentioned above, a previously issued SCAR (94-018), initiated approximately five months earlier, had recommended actions to revise Administrative Procedure (AP) 0020, Control of Temporary and Minor Modifications, to more explicitly define what conditions warrant temporary modification considerations and to better define intrusive monitoring activities. This action had not been implemented and no short-term measures had been initiated at the time of the second event (SCAR 95-05). The inspector found this root cause reasonable, based upon the information presented in SCAR 95-005 and inspection report 94-31, section 4.1. The inspector noted that the I&C technicians responsible for monitoring instrument 2-3-72B did ask for engineering staff assistance in determining whether or not their work order scope should be expanded to a temporary modification (per AP-0020). However, the lessons learned from SCAR 94-018 had not been communicated to the engineering staff, and thus, this was a missed opportunity.

During periodic plant tours and recent accompaniments on operator rounds, the inspectors have not observed any unauthorized plant modifications or temporary monitoring equipment. A review of currently active temporary modifications identified that appropriate administrative controls were exercised and that the associated safety evaluations were properly documented. The inspector reviewed the post-implementation 10 CFR 50.59 evaluation for the instrument 2-3-72B monitoring equipment and confirmed that this temporary modification did not introduce an unreviewed safety question or result in a needed change to TSs. The inspector also verified that appropriate long-term corrective

actions (AP-0020 revisions and personnel training) were completed to provide reasonable confidence that this type of event would not recur.

c. <u>Conclusions</u>

VY staff's root cause assessment and corrective actions for the violation (VIO 94-31-01) of TSs involving an unreviewed temporary plant modification were found thorough and appropriate to prevent recurrence. The root cause was identified as a failure of management systems to promptly implement corrective actions for a previous temporary modification implementation problem. This root cause was symptomatic of a weak corrective action process and poor staff performance with respect to proper evaluations of temporary plant modifications. Violation 94-31-01 is closed.

E8.3 (Closed) URI 93-33-02: Lack of clear licensing basis for the service water subsystem Technical Specification

a. <u>Inspection Scope</u>

During an earlier inspection period, the performance of service water system limiting condition for operation (LCO) maintenance was examined by the inspectors. The inspectors identified that neither the TS nor Updated Final Safety Analysis Report (UFSAR) addressed the specific (required flow or number of operable pumps) service water subsystem minimum operability design requirements. Consequently, an unresolved item was initiated to ensure the minimum subsystem operability requirements were clarified for future reference by the VY and NRC staffs. Using IP 92903, the inspector conducted a followup of this issue to determine the NRC staff's response and the VY staff's resolution of this item.

b. <u>Observations and Findings</u>

Discussion with responsible licensee representatives identified that the VY staff addressed this item initially by clarifying the system operability definition and associated administrative controls via the system operating procedure OP-2181, Service Water and Alternate Cooling Operating Procedure. OP-2181 clearly defines the A and B service water subsystems and respective electrical power supplies. In addition, OP-2181 prohibits removal from service for maintenance any more than one service water pump at a time. During this inspection period, the licensee was completing final revisions and Plant Operations Review Committee (PORC) approval of a TS interpretation which further defines service water subsystem minimum operability requirements which have been extended to specific service water system manual and automatic valve functions.

The inspector also examined an NRC internal memorandum, McKee to Cooper, dated October 13, 1995, which responded to a request for technical assistance on VY alternate cooling system operability and lack of integrated performance testing. This NRC staff review specifically addressed the adequacy of current TS and the licensing basis for the alternate cooling and service water systems. Based upon this review, the NRC staff concluded that although there was room for improving the current systems' TS (such as adopting the standard TS), the imposition of a backfit to amend the service water and

alternate cooling systems TSs would not result in a substantial increase in the overall protection of public health and safety or bring the facility into compliance. Accordingly, an NRC staff imposed TS Amendment (backfit) could not be justified by current Commission regulations. The inspector notes that earlier this year the VY staff committed personnel and resources to convert the facility's current TS to the standard TS format for NRC staff review. VY plans to submit their new TS by September 1997 and implement the new TS within two to three months following approval by the NRC staff, which allows for VY staff training and familiarization.

c. Conclusions

The VY staff implemented appropriate interim administrative controls for the apparent lack of TS and licensing basis clarity for the service water and alternate cooling systems. Conversion to standard TS, targeted for implementation in mid-1998, was planned by the licensee to ultimately resolve this item. The NRC staff had concluded that a TS change would be a backfit that was unwarranted at this time. Unresolved item 93-33-02 is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

The licensee's program for radioactive materials and radioactive waste management and transportation was reviewed. Specific areas reviewed included: audits and appraisals of the program; changes in personnel, procedures, and equipment; facility condition versus the UFSAR; training and qualifications of personnel; the solid radioactive waste program; the radioactive materials and waste shipping program; and implementation of the new DOT regulations. The inspection also included a review of other items such as the implementation of corrective actions for a previously identified violation regarding access controls to high radiation areas (VIO 95-18-01).

R3 RP&C Procedures and Documentation

R3.1 Solid Radioactive Waste Program

a. Inspection Scope (86750)

The inspector reviewed the solid radioactive waste program through a review of licensee procedures, interviews with licensee personnel, and review of licensee records.

b. Observations and Findings

The licensee used a combination of direct isotopic sampling, scaling factors, and dose-to-curie conversions to determine the isotopic and curie content of radioactive waste containers. Waste streams were sampled and sent to an offsite laboratory on a periodic basis to determine the radioisotopic content. Hard to measure radionuclides (beta and alpha emitters) were related to the gamma emitting isotopes through scaling factors. Subsequent samples were not always sent off site, but were counted in the onsite detector

for gamma isotopic analysis. Routine packages were also analyzed and dose-to-curie conversions were developed based on the waste stream. The inspector reviewed the use of the scaling factors, dose-to-curie conversion factors, and laboratory sampling and determined that they were used appropriately and in accordance with NRC and industry guidance.

The licensee used a customized software program (RUNSHIP) to determine the radioactive waste classification, fissile class, and shipping categories. The inspectors reviewed various computer records and determined that the licensee had used the software appropriately.

The licensee maintained current copies of licenses for facilities that received radioactive materials and radioactive waste from their facility. Certificates of Compliance were maintained for high integrity containers (HICs).

The procedures for transferring and packaging radioactive waste and radioactive materials were assessed by the inspector to be very good. The Process Control Program (PCP) was maintained current; however, the licensee was not currently solidifying or encapsulating any liquid waste. However, the licensee was using a vendor supplied system to de-water liquid waste (spent resins), also approved through the PCP, before shipment to a disposal facility or onsite storage. Most radioactive waste was stored on the site in concrete storage modules. The storage pad had numerous containers filled with waste, but the storage pad was not near full capacity.

c. Conclusions

The licensee maintained a very good program for packaging, storing, and handling radioactive waste and radioactive materials including the appropriate use of waste stream analysis, isotopic composition, waste classification, and radioactivity calculations. Program procedures were very good including the Process Control Program.

R3.2 Radioactive Waste/Radioactive Material Shipping Program

a. Inspection Scope (86750)

The inspectors reviewed the radioactive waste and radioactive material shipping program through a review of licensee records, interviews with licensee personnel, and review of licensee procedures.

b. Observations and Findings

The licensee was using various vendors for processing of radioactive waste and had current access to the burial site in South Carolina. However, due to commitments to the Texas compact, the licensee was shipping very small amounts of radwaste to South Carolina. Some containers were staged in the warehouse or in cargo vans until they were completely filled or a container was ready, then the waste was transferred to the storage site adjacent to the plant's restricted area. Some waste was shipped for processing (volume reduction, incineration, or decontamination). The inspector toured all areas used to store radioactive waste and determined that the areas were in good physical condition.

The inspector reviewed completed shipping records and noted the records were complete with the appropriate information, reviewed and certified by qualified individuals, and were maintained in good condition. Attention to detail was very good. No violations of regulatory requirements were noted.

The inspector also reviewed the emergency response information provided as guidance to emergency responders in the case of a transportation accident during transit. The information was appropriate and was easy to locate. When a shipment was in transit, the licensee provided a copy of the emergency response information to the radiation protection technician on shift. The control room operators were listed as the emergency response contact after normal operating hours, and would contact the shift radiation protection technician to obtain shipment information. Because there was no shipment in transit during the period of this inspection, the emergency contact number and information were not directly verified by the inspector.

c. Conclusions

The licensee maintained storage areas for radioactive waste and radioactive materials in good condition. Shipping records were maintained in good condition and emergency response information was provided as required.

R3.3 Implementation of the Revised DOT Shipping Regulations

a. Inspection Scope (TI 2515/133)

The inspector reviewed the training of personnel and implementation of the revised regulations for radioactive materials and radioactive waste through a random selection of training records and interviews with various licensee personnel.

b. Observations and Findings

The licensee had provided training on the revised DOT regulations to appropriate personnel involved with the radioactive waste management program (see Section R5.1 of this Report). The training was very thorough and covered the new regulations in good detail.

Licensee procedures also incorporated the new requirements including the revised A_1/A_2 (allowed activities) values, the low specific activity (LSA) and surface contaminated object (SCO) definitions, and the classification of fissile materials. Also, the licensee implemented the requirement for SI units; however, this regulation is not a requirement until mid-year 1997.

c. Conclusions

The licensee appropriately implemented the revised DOT regulations for shipping radioactive waste and radioactive materials.

R5 Staff Training and Qualification in RP&C

R5.1 Training and Qualifications of Personnel

a. Inspection Scope (86750)

The inspector reviewed the training and qualifications of personnel involved with the radioactive waste management program through a random selection and review of training records and interviews with various licensee personnel.

b. Observations and Findings

Training records of attendance and copies of examinations showed that all selected individuals completed the required training for hazardous material handling, processing, and shipping. The individuals designated as certifiers of radioactive waste shipments completed vendor training on the revised DOT and NRC radwaste packaging and transportation regulations. The inspector noted that the appropriate individuals had been trained relative to the new DOT regulations, but none of the technicians involved with the radwaste shipping program had taken this course, as of the date of this inspection. The inspector agreed with the licensee's management that the persons certifying and reviewing the records required the training, while the technicians who prepared the paperwork could benefit from the training even though it was not required. The inspector did not identify the lack of training of the technicians on the revised regulations as an immediate safety concern; however, the inspector identified this as an area for program improvement.

The qualification records for radwaste personnel were very well documented. All records checked by the inspector indicated that personnel who performed radwaste processing or shipping activities maintained the appropriate qualifications and attended refresher training as required. A review of the course content and examinations for the courses revealed that they were technically accurate and contained very relevant information.

c. Conclusions

The licensee appropriately trained staff members on the procedures and regulations pertaining to radioactive waste handling, processing, packaging, and shipping. The staff was highly qualified for the positions and tasks associated with the radioactive waste management program. An area for program improvement was identified regarding training of technicians on the revised DOT and NRC regulations for radwaste packaging and shipment.

R6 RP&C Organization and Administration

R6.1 Changes in the Radiological Controls Program

a. Inspection Scope (86750)

Changes to the radioactive waste management program were reviewed by the inspector through interviews with licensee personnel.

b. Observations and Findings

The licensee implemented staffing changes in the radioactive waste management program since the last inspection. Specifically, the radwaste coordinator (supervisor) position had been vacated when an individual left the company and the licensee filled the position with another individual from the radiation protection staff. In addition, the position of principle training instructor for radiation protection/radwaste had been filled by a technician from the radiation protection staff. The inspector reviewed the qualifications for personnel in the new positions and determined that they were very well qualified for the positions. These organizational changes did not adversely affect the quality of the program.

c. Conclusions

The changes to the licensee's radioactive waste management program were determined to be acceptable.

R7 Quality Assurance in RP&C Activities

R7.1 Audits and Appraisals

a. Inspection Scope (86750)

Audits, surveillance reports, internal assessments, and deviation reports of the radiological controls program documented since the last NRC inspection were reviewed by the inspector.

b. Observations and Findings

The last quality assurance (QA) audit of the radioactive waste management program was conducted in August 1995. Members of the audit team included QA staff members, technical specialists from other nuclear power plants, and an outside consultant. The auditors reviewed the methods used by the plant staff for classification, packaging, and transportation of radioactive material and radioactive waste. The auditors reported that the program was satisfactory and met regulatory requirements. Program strengths were identified including good communication of waste reduction initiatives and recent procedure revisions for packaging materials. Some weaknesses and recommendations were identified regarding the reliance upon personnel knowledge in lieu of written procedure, and inaccuracies in the shipping manifest records. Licensee management reviewed the recommendations and assigned action items as appropriate. Areas of weakness were tracked via the Event Reporting system to ensure timely corrective actions were implemented. The inspector reviewed the corrective actions and determined they were timely and technically acceptable.

The QA staff had performed several surveillances of radioactive waste activities in the last 12 months. The activities were documented as surveillance reports and included surveys of radioactive waste storage facilities, packaging of radioactive material for shipment, and handling of casks used for storage and shipment of radioactive waste. The QA staff identified minor deficiencies. The inspector reviewed the surveillance reports and noted

the corrective actions to prevent recurrence of the deficiencies. The corrective actions implemented by the VY staff were appropriate and timely.

Reports of problems or deficiencies, written by the staff during 1995 and 1996, identified and documented minor deficiencies within the radioactive waste management program. The inspector reviewed the reports and noted that the VY staff implemented timely and technically acceptable corrective actions to prevent recurrence.

c. Conclusions

The inspector concluded that the licensee continued to improve the quality of the radioactive waste management program through the self-identification and correction of minor deficiencies.

R8 Miscellaneous RP&C Issues

R8.1 (Closed) Violation 95-18-01: Failure to properly control high radiation area access

a. Inspection Scope (92704)

The inspectors reviewed the corrective actions implemented as a result of a previously identified violation regarding access controls for high radiation areas.

b. Observations and Findings

The licensee responded to the Notice of Violation (NOV) in a letter dated September 5, 1995. The response outlined the licensee's corrective actions to prevent recurrence of the event, including; revised procedures to require a double verification that high radiation area doors were locked when dose rates exceeded 1000 mrem in 1 hour at a distance of 30 centimeters from the source; training of personnel on the revised procedure and responsibilities associated with locked doors; and establishment of a multi-disciplined team to evaluate and recommend changes to the radiation protection program. The team supplied many recommendations for improving the radiation protection program and several changes were implemented. The changes included revisions to the general employee training, purchase and installation of new high radiation area signs, and general awareness training to all employees regarding radiation protection policies.

The inspector verified that the licensee had implemented the corrective actions described in the response letter to the NOV. As a result of these corrective actions, the licensee has not had a recurrence of the violation in the past year. This item is closed.

c. Conclusions

The licensee's corrective actions were appropriate and timely to prevent recurrence of violations regarding access to high radiation areas.

S1 Conduct of Security and Safeguards Activities

S1.1 Inspection Scope

The inspector reviewed the security program using IP 81700 during the period of July 22-25, 1996. Areas inspected included: effectiveness of management control; management support and audits; protected area detection equipment; alarm stations and communication; testing, maintenance and compensatory measures; and training and qualification. The purpose of this inspection was to determine whether the licensee's security program, as implemented, met the licensee's commitments and NRC regulatory requirements.

S2 Status of Security Facilities and Equipment

S2.1 Protected Area Detection Aids

The inspector conducted a physical inspection of the protected area (PA) intrusion detection systems (IDSs) on July 22, 1996. The inspector determined by observation that the IDSs were installed and maintained as described in the NRC-approved security plan (the Plan).

S2.2 Alarm Stations and Communications

The inspector observed Central Alarm Station (CAS) and Secondary Alarm Station (SAS) operations, and verified that the alarm stations were equipped with the appropriate alarm, surveillance, and communication capabilities. Inspector interviews of CAS and SAS operators found them knowledgeable of their duties and responsibilities. The inspector also verified that the CAS and SAS operators were not required to engage in activities that would interfere with assessment and response functions, and that the licensee had exercised communications methods with the local law enforcement agencies as committed to in the Plan.

S2.3 Testing, Maintenance and Compensatory Measures

S2.3.1 Testing and Maintenance

The inspector's review of testing and maintenance records for security-related equipment confirmed that the records were on file and that the licensee was testing and maintaining systems and equipment as committed to in the Plan. A review of these records indicated that repairs were being completed in a timely manner and that a priority status was assigned to each work request.

S2.3.2 Compensatory Measures

The inspector's review of the use of compensatory measures found it to be appropriate and minimal. It was apparent that priority repair efforts were carried out by the maintenance group when problems required compensatory measures.

S5 Security and Safeguards Staff Training and Qualification

The inspector met with the security training supervisor and discussed training program enhancements made since the last inspection conducted in November 1995. These included firing range improvements to enhance the present weapons stress course and the procurement of training aids to add realism during contingency response training.

The inspector observed tactical range and tactical movement training on July 24, 1996. The training consisted of tactical weapons manipulation, target acquisition and tactical movement, stressing the use of cover and concealment. The instructors did an excellent job controlling the drills and the range was controlled in a safe manner. Additionally, the inspector interviewed several security force members (SFMs) and determined that, based on the SFMs responses to the inspector's questions, the training provided by the security training staff was effective.

S6 Security Organization and Administration

S6.1 Management Support

Management support for the physical security program was determined to be excellent. This determination was based on the inspector's review of various program enhancements made since the last inspection, which was conducted in November 1995. These included completion of the vehicle barrier system installation and the procurement of simulated weapons, to add realism during tactical response training.

S7 Quality Assurance in Security and Safeguards Activities

S7.1 Effectiveness of Management Controls

The inspector determined that the licensee had controls for identifying, resolving, and preventing security program problems. These controls included the performance of the required annual quality assurance (QA) audits, a formalized self-assessment program, and ongoing security shift supervision oversight. The licensee also utilized industry data, including adverse data, such as violations of regulatory requirements identified by the NRC at other facilities, as a basis for self-assessment to determine if similar conditions existed in its program. A review of documentation applicable to the programs indicated that initiatives to minimize security performance errors and identify and resolve potential weaknesses were being implemented and were effective.

S7.2 Audits

The inspector reviewed the 1995 Quality Assurance (QA) audit of the security program conducted October 16-20 and 27, 1995 (Audit No. VY-95-04), the combined 1995 audit of the access authorization (AA) and fitness for duty (FFD) programs conducted February 13-17, 1995, and February 21, 1995 (Audit No. VY 95-19), and the combined 1996 audit of the AA/FFD programs conducted February 12-16 and 23, 1996 (Audit No. 96-19). The inspector determined that the audits were conducted in accordance with the Plan. To enhance the effectiveness of the audits, the audit teams included an independent security specialist.

The 1995 security program audit identified no findings and two recommendations; the 1995 combined AA/FFD audit identified no findings and two recommendations; and the 1996 combined AA/FFD audit identified one finding, in the area of AA, concerning follow-up credit history checks by the contracted background investigation company. The inspector determined that the AA/FFD finding and recommendations were not indicative of programmatic weaknesses, but would enhance program effectiveness. The inspector also determined, based on discussions with security management and a review of the responses to the finding, that the corrective actions were effective.

The inspector's review concluded that the audits were comprehensive in scope and depth, that the finding was reported to the appropriate levels of management, and that the programs were being properly administered.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors met with licensee representatives periodically throughout the inspection and following the conclusion of the inspection on September 13, 1996. At that time, the purpose and scope of the inspection were reviewed, and the preliminary findings were presented. The licensee acknowledged the preliminary inspection findings.

X2 Pre-Decisional Enforcement Conference Summary

On July 23, 1996, a pre-decisional enforcement conference was held at the NRC Region I office to discuss potential enforcement issues identified in inspection report 96-07. These issues involved the VY staff's April 11, 1996 discovery of the residual heat removal system minimum flow valve single failure vulnerability as documented in Licensee Event Report 96-010, dated May 9, 1996. The handouts used in the VY staff's presentation at the conference have been included as Attachment A to this report.

X3 Review of Updated Final Safety Analysis Report (UFSAR)

A recent discovery of a licensee operating its facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compares plant practices, procedures, and parameters to the UFSAR description. While performing the inspections discussed in this report, the inspector reviewed the applicable portions of the UFSAR that related to the areas inspected. This included portions of sections 1.9 and Appendix F pertaining to the quality assurance program.

The inspector reviewed the processing (as appropriate) and storage of radioactive waste and material at the Vermont Yankee Nuclear Station relative to descriptions and commitments provided in the UFSAR. There were no inconsistencies identified between the UFSAR and current practices relative to processing or storage of radioactive waste. The overall adequacy of the program for onsite storage of radioactive material and updating of the UFSAR, in accordance with the 10 CFR 50.71(e) was considered very good.

Since the UFSAR does not specifically include security program requirements, the inspector compared licensee activities to the NRC-approved physical security plan, which is the applicable document. The inspector reviewed Section 4.2 of the Plan, Revision 27, dated February 23, 1996, titled, "Protected Area Barriers," and determined that the Plan was consistent with current security program procedures and practices. Overall, the inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters discussed in the preceding sections of this report.

PARTIAL LIST OF PERSONS CONTACTED

LICENSEE AND CONTRACTOR PERSONNEL

R. Wanczyk, Plant Manager
G. Maret, Operations Superintendent
E. Lindamood, Technical Services Superintendent
L. Doane, Operations Manager
M. Watson, I&C Manager
T. Watson, Maintenance Manager
F. Helin, Reactor Engineering Manager
M. Desilets, Radiation Protection Manager
S. Skibniowsky, Chemistry Manager
G. Morgan, Security Manager
J. Moriarty, Security Operations Specialist
D. Calsyn, Quality Assurance Supervisor
E. Harper, Project Manager, The Wackenhut Corporation (TWC)
E. Wright, Security Operations Supervisor, TWC
J. Jasinski, Security Training Supervisor, TWC
T. McCarthy, Radwaste Supervisor
E. Miller, Radwaste Technician
M. Pletcher, Training Instructor
J. Thayer, Engineering Vice President
P. Corbett, Project Engineering Manager
R. Sojka, Operations Support Manager
J. Boivin, Technical Support Manager
J. Duffy, Licensing Engineer
A. Parker, Fire Protection Program Manager
D. Yazi, Appendix R Task Team Leader, YNSD

INSPECTION PROCEDURES USED

86750	Radioactive Waste Management and Transportation
TI 2515/133	Implementation of the Revised 49 CFR 100-179 and 10 CFR 71
71707	Plant Operations
62707	Maintenance Observations
61726	Surveillance Observations
60705	Preparations for Refuel
37550	Engineering
37551	Onsite Engineering
71750	Plant Support Activities
81700	Physical Security Program

ITEMS OPENED, CLOSED, AND DISCUSSED

OPEN

URI 96-08-01	East switchgear room design corrective actions.
URI 96-08-02	Containment penetration thermal relief protection adequacy.

CLOSED

VIO 94-31-01	Unevaluated temporary modification.
URI 94-08-02	Maintaining the diesel fire pump fuel oil day tank above the minimum TS level.
URI 93-33-01	Adequacy of guidance to classify adverse conditions/events.
URI 93-33-02	Lack of clear licensing basis for the service water subsystem TS.
VIO 95-18-01	Access controls to high radiation areas.
URI 95-25-01	Root cause for December 8, 1995 scram.

LIST OF ACRONYMS USED

VY	Vermont Yankee
INPO	Institute of Nuclear Power Operations
NRC	Nuclear Regulatory Commission
CO_2	Carbon Dioxide
TS	Technical Specification
SCAR	Significant Corrective Action Report
LCO	Limiting Condition for Operation
FSAR	Final Safety Analysis Report
UFSAR	Updated Final Safety Analysis Report
QA	Quality Assurance
AA	Access Authorization
FFD	Fitness for Duty
SFMs	Security Force Members
CAS	Central Alarm Station
SAS	Secondary Alarm Station
IDSs	Intrusion Detection Systems
PA	Protected Area
Plan	Physical Security Plan
VIO	Violation
HICs	High Integrity Containers
PCP	Process Control Program
SCO	Surface Contaminated Object
ER	Event Report
EQ	Environmental Qualification
FAA	Functional Area Assessment
GL	Generic Letter
IN	Information Notice
ISI	Inservice Inspection
IST	Inservice Testing
MOV	Motor Operated Valve
OE	Operating Experience
SFP	Spent Fuel Pool
RCIC	Reactor Core Isolation Cooling
ECCS	Emergency Core Cooling System

ATTACHMENT A

[Predecisional Enforcement Conference]

[Licensee Slide Presentation]

Vermont Yankee Nuclear Power Corporation

RHR Minimum Flow Valve Enforcement Conference

July 23, 1996

Vermont Yankee
Nuclear Power Corporation

Attendees

Ross Barkhurst	President and CEO
Jay Thayer	Vice President Engineering
Bob Wanczyk	Plant Manager
Stan Miller	Design Engineering Manager
Jim Callaghan	Lead Fluid Systems Engineer
Michele Sironen	VY Nuclear Engineering Coordinator
Bruce Slifer	Senior Fluid Systems Engineer
Jim Duffy	Licensing Engineer

Enforcement Conference Agenda

• Introduction	Jay Thayer
• Background	Bruce Slifer
• Missed Opportunities	Stan Miller
• Corrective Action Process	
- Short-Term Corrective Actions	Jim Callaghan
- Root Cause Analysis	Bruce Slifer
- Single Failure Assessment	Jim Callaghan
• Safety Assessment	
- RHR Pump Performance	Stan Miller
- LOCA Analysis Impact	Michele Sironen
- IPE Impact	Michele Sironen
• Conclusion	Jay Thayer

Introduction

Apparent Violation

Narrow focus: past process insufficient to detect/correct problems

Mitigating Factor

Current Corrective Action Process broad based and comprehensive

Introduction

Apparent Violation

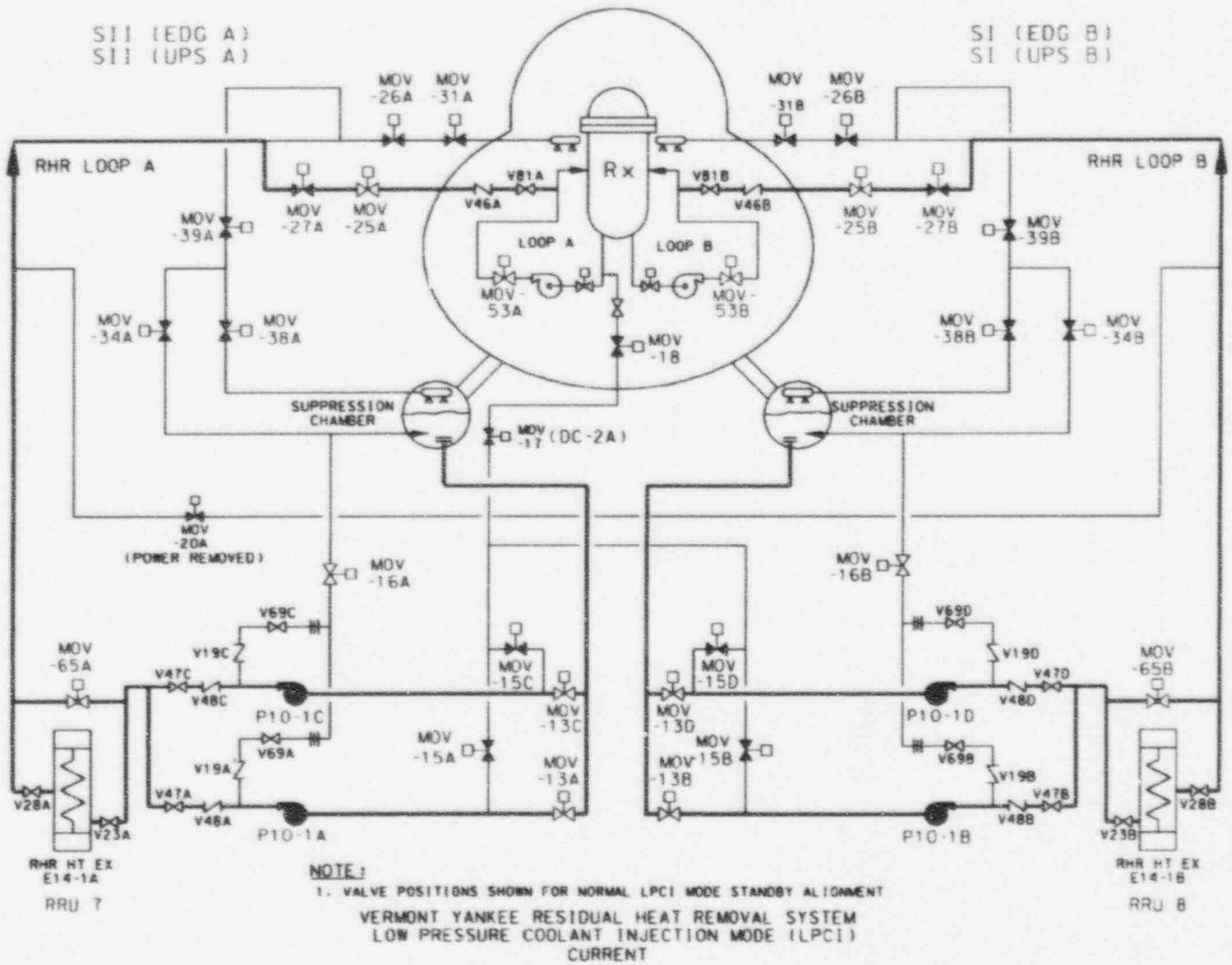
Failure to perform Appendix K, LOCA analysis assuming worst-case single failure

Mitigating Factors

- **Self-identified**
- **Prompt and comprehensive corrective actions**
- **LOCA analysis had been performed with worst-case single failure known at the time**
- **Existing beyond design basis LOCA analysis bounded the identified worst-case single failure scenario**
- **This analysis shows that 10CFR50.46 limits would not be exceeded**

Background

System Diagram



Background

Appendix R Reanalysis Finding

- Event discovered as result of comprehensive review by fire protection team
- Review conducted as corrective action in response to Appendix R violation
- Cable-by-cable verification was in progress for all systems
- Cross-connects between Bus 3 and 4 identified
- Appendix R team identified single failure concern and promptly initiated event report process

Missed Opportunities

To Prevent:

- Failure to address vessel draindown concern (1971)
- Failure to change P&ID (1971)
- Failure to perform thorough single failure evaluation (1974)

Missed Opportunities

To Discover:

- NEDO-20967 single failure evaluation (1975)
- Generic evaluation of dc power failure (1978)
- IEB 86-01 minimum flow logic problems (1986)
- GE safety evaluation regarding minimum flow design adequacy (1987)
- LER 89-09 single failure results in loss of both RHRSW loops (1989)

Missed Opportunities

To Discover:

- Plant Design Change Request (PDCR) 89-04
- Review of Cooper Significant Event Report (1993)
- New LOCA analysis with RELAP5YA (1993)
- Transfer IPE insights to design basis LOCA results (1993)
- P&ID corrective drawing update to show minimum flow valves closed (1995)

Missed Opportunities

Fundamental Question

- **With the opportunities presented over 22 years, how did we fail to identify this mistake earlier?**

Missed Opportunities

- Need for questioning attitude in all activities not consistently reinforced
- Methods for performing event evaluations not always broad enough in focus
- Design basis documentation not easily retrieved

Program Improvements

- Improved questioning attitude/safety culture/external perspective
- Improve review process and set higher management expectations for evaluating operating events
- Significant improvements in engineering training programs
- Accelerate DBDs for high and medium safety significant systems consistent with IPE and Maintenance Rule
- Complete FSAR upgrade in 1997
- Developing System Engineering Program

Missed Opportunities

Summary

- Continued emphasis on questioning attitude and safety culture supported with external perspective
- Problem evaluation program has been improved over the years and is in the process of additional improvements
- Design basis documentation and FSAR upgrades to be accelerated

Corrective Action Process

Short-Term Corrective Actions

- Prompt event report and immediate operability assessment
- Basis for Maintaining Operation (BMO)
- Timely 10CFR50.59 safety evaluation for valve position change

Short-Term Corrective Actions

Event Report 96-229

- Initiated April 11, 1996 on LOCA single failure analysis concerns (1615)
- Initial design engineering assessment on operability concern
  - Pump vendor information limited break size concerns
  - Industry events of RHR pumps in no-flow condition
  - Analysis for one core spray pump and ADS (1993)
  - Discussed evaluation with plant management and NRC Resident on April 11, 1996

Short-Term Corrective Actions

Event Report 96-229

- Discussed concern with shift supervisor on April 11, 1996 (1710)
  - Shift supervisor determined, based on engineering evaluation, no immediate operability concern
  - Shift supervisor determined VY was outside design basis per 10CFR50.72(b)(1)(ii)B
  - Shift supervisor initiated non-emergency, one-hour notification (1724)

Short-Term Corrective Actions

Event Report 96-229

- Event report discussed at event report screening meeting on April 12, 1996
  - ER was determined to be a Level 1 event, the highest level available
  - Basis for maintaining operation was requested within a seven day time frame

Short-Term Corrective Actions

Event Report 96-229

- Event report presented to Plant Operations Review Committee (PORC) immediately following screening
  - Significance of finding led to speciai PORC presentation
  - Event discussed in length, with emphasis on operability concerns
  - PORC determined initial assessment acceptable, re-emphasized BMO time frame
- Formal root cause analysis initiated based on Level 1 event report

Short-Term Corrective Actions

Basis for Maintaining Operation (BMO 96-07)

- Initiated on April 12, 1996 to support operability assessment of event report
- Identified factors which compensated for adverse condition
  - Discussions with pump vendor of potential damage
  - Industry experience of pumps operating in a no-flow condition
  - Analysis for one core spray and ADS LOCA case
  - IPE analysis conclusion of low risk significance

Short-Term Corrective Actions

Basis for Maintaining Operation (BMO 96-07)

- Recommendations for correcting the condition
  - 10CFR50.59 initiated to support line-up change for minimum flow valves
- BMO 96-07 presented to PORC April 19, 1996
  - PORC concluded basis for operability was sound but additional documentation was needed
- BMO 96-07, Revision 1, presented to PORC April 25, 1996 and approved by Plant Manager on May 1, 1996

Short-Term Corrective Actions

10CFR50.59 Safety Evaluation

- Safety evaluation of changing normal position of minimum flow valve initiated
- Comprehensive design basis evaluation considered:
  - Original system design and minimum flow valve position
  - Impact of position change on current calculations (LOCA, long-term containment cooling)
  - Impact of position change on Vermont Yankee programs (Appendix R, Appendix J, ISI, IST, MOV)
  - Impact on plant controlled documents (FSAR, procedures)
  - Discussions with other utilities

Short-Term Corrective Actions

10CFR50.59 Safety Evaluation

- Safety evaluation was presented to PORC on April 24, 1996
  - Evaluation accepted with commitment to address changes required
  - Evaluation approved by Plant Manager on April 25, 1996
- Minimum flow valve normal position changed to open on April 26, 1996

Short-Term Corrective Actions

10CFR50.59 Safety Evaluation

- Safety evaluation was presented to PORC on April 24, 1996
  - Evaluation accepted with commitment to address changes required
  - Evaluation approved by Plant Manager on April 25, 1996
- Minimum flow valve normal position changed to open on April 26, 1996

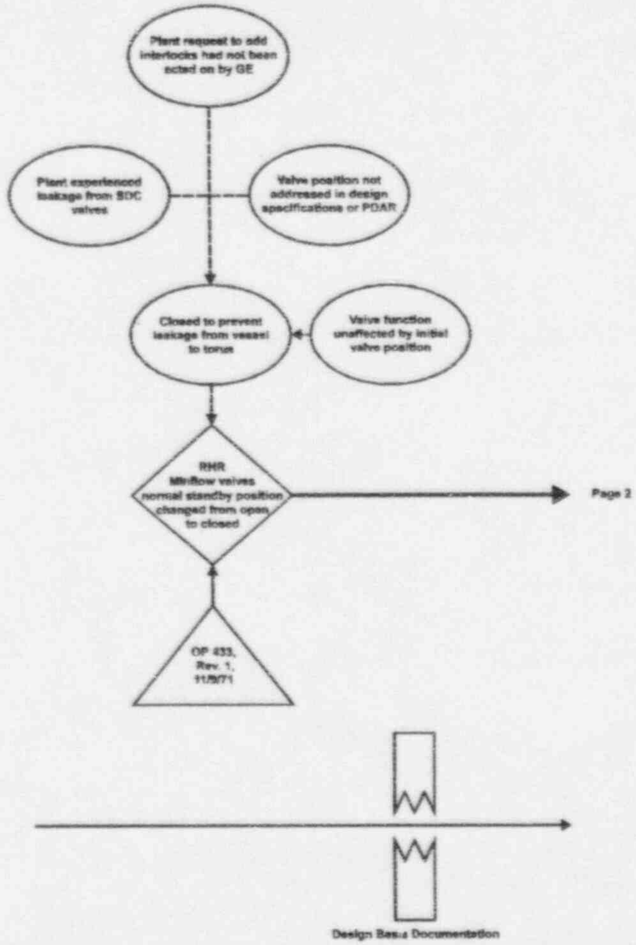
Short-Term Corrective Actions

Summary

- Prompt
- Immediate
- Comprehensive

Root Cause Analysis

Loss of RHR Minflow Protection



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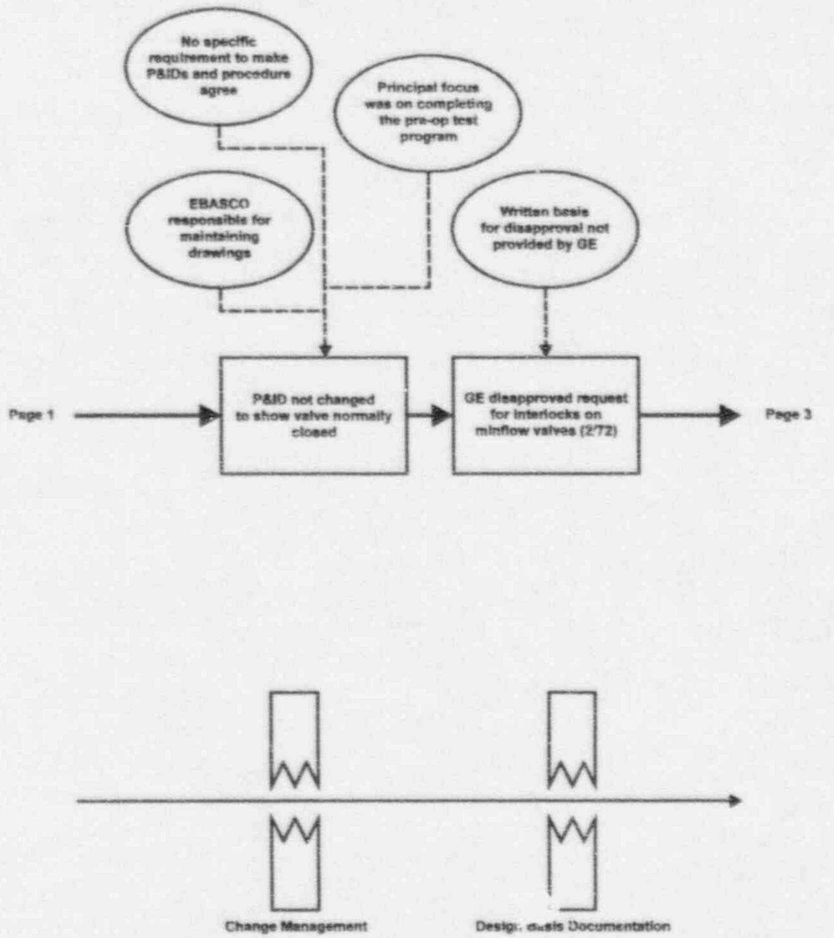
Root Cause Analysis

Loss of RHR Minflow Protection



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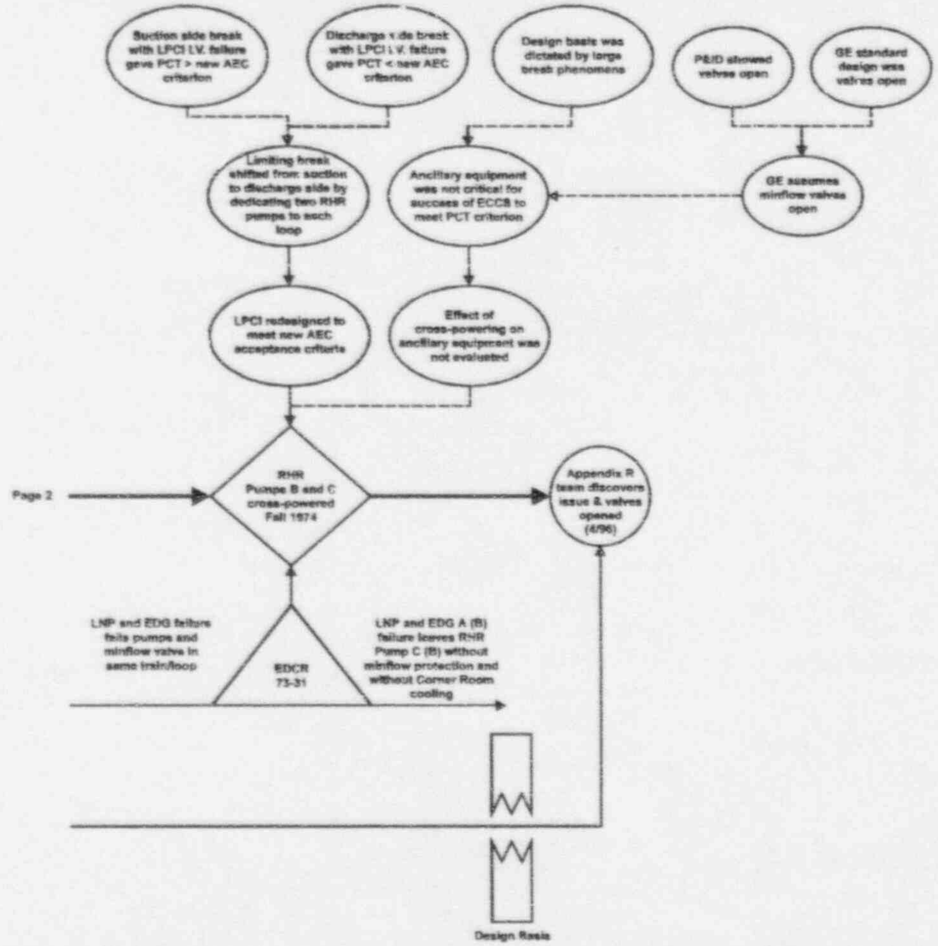
Root Cause Analysis

Loss of RHR Minflow Protection



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Root Cause Analysis

- Event Definition
  - Failure to provide minimum flow protection for cross-powered RHR pumps
- Primary Effects
  - Minimum flow valves closed by procedure, but P&ID unchanged
  - RHR Pumps B and C cross-powered but not minimum flow valves

Root Cause Analysis

- Root Cause
  - Inadequate design/single failure evaluation

- Contributing Causes
  - Inadequate documentation of minimum flow design basis
  - ECCS design focus on large break phenomena
  - Design engineers assumed minimum flow valves were open when they were closed

Root Cause Analysis

Corrective Actions - Completed

- Maintain minimum flow valves open during normal operation
- Revise LOCA analysis to account for failure of valve to close
- Review current design change process to ensure broad scope, comprehensive reviews

Root Cause Analysis

Corrective Actions - In Progress

- Complete single failure vulnerability review for all ECCS
- Revise FSAR description of minimum flow function
- Revise P&ID to show minimum flow valves open
- Develop policy on use of P&ID as design basis document

Root Cause Analysis

Corrective Actions - Planned

- Provide copies of RCA and missed opportunities to:
  - Task team working on OE assessment improvements (7/96)
  - Training Department for inclusion in ESP training (7/96)
  - Design engineering personnel (7/96)
- Self-assessment of process used to transfer analysis assumptions to operating procedures (12/96)
- Self-assessment of drawing revision process (12/96)

Root Cause Analysis

Summary

- Investigated event origins
- Identified root and contributing causes
- Corrective actions

ECCS Single Failure Assessment

- **Purpose**
  - **Identify all single active failures**
  - **Confirm ECCS design basis LOCA requirements are satisfied**

ECCS Single Failure Assessment

Scope

- Initial detailed assessment complete - undergoing independent review
- ECCS systems/components
  - HPCI
  - ADS
  - CS
  - LPCI
- Electrical systems
  - 125V/24 Vdc power
  - 4 kV/480 Vac power
  - 480 V/LPCI-UPS power

ECCS Single Failure Assessment

Scope

- Short-term ECCS injection (blowdown, refill, reflood)
- No credit for ECCS delivery to broken loop/pipe
- Coincident loss of off-site power

ECCS Single Failure Assessment

Bounding Single Active Failure Cases

- DC-1 bus failure
- DC-2 bus failure
- LPCI UPS failure

Example Case Study
Single Failure of DC Bus 1 with LNP

SINGLE ACTIVE FAILURE AVAILABLE ECCS SYSTEMS	No Break	Break Location: Recirc. Discharge Line, Loop A	Recirc. Discharge Line, Loop B	Recirc. Suction Line, Loop A	Recirc. Suction Line, Loop B	Core Spray Line	Feed Water Line	Main Steam Line
DC-1 Failure w/ Loss of Normal Power (LNP)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
INDEPENDENT SYSTEMS AVAILABLE FOR SHORT TERM[12] ECCS INJECTION MODE (with DC-1 failure)								
Division D2 DC Power via Battery B-1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Division S2 AC Power via EDG-1A/Bus 4	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
ECCS Logic Division A	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
RCIC via DC-2	Yes	Yes[5]	Yes[6]	Yes[5]	Yes[5]	Yes[5]	No[1]	No[6]
ADS:								
- Logic via Division A	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
- 4 SRVs powered from DC-2C	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes[7]
ADS Availability:	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
LPCI Loop A:								
- Pump P-10-1A via EDG-1A/Bus 4	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-16A Min. flow valve (N.O., A.C.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* RRU-7 NE Corner Room Cooler	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
- Loop A Valves w/Power from MCC-89A (LPCI MG-Set-1A) and ECCS Div. A Logic								
* Inj. Vlv MOV-25A (N.O., R.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* Inj. Vlv MOV-27A (N.C., A.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* Recirc. Discharge MOV-53A (N.O., A.C.)	Yes	No[2]	Yes	No[2]	Yes	Yes	Yes	Yes
* Recirc. Bypass MOV-54A (N.O., A.C.)	Yes	No[2]	Yes	No[2]	Yes	Yes	Yes	Yes
LPCI Loop A Availability:	Yes	No[1]	Yes	No[1]	Yes	Yes	Yes	Yes
LPCI Loop B:								
- Pump P-10-1B via EDG-1A/Bus 4	ST[8]	ST[8]	ST[8]	ST[8]	ST[8]	ST[8]	ST[8]	ST[8]
* MOV-16B Min. flow valve (N.O., F.O.)	Yes[8]	Yes[8]	Yes[8]	Yes[8]	Yes[8]	Yes[8]	Yes[8]	Yes[8]
* RRU-8, SE Corner Room Cooler	No	No	No	No	No	No	No	No
- Loop B Valves w/power from MCC 89B (LPCI MG-Set-1B) and ECCS Div. A Logic								
* Inj. Valve MOV-25B (N.O., R.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* Inj. Valve MOV-27B (N.C., A.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* Recirc. Discharge MOV-53B (N.O., A.C.)	Yes	Yes	No[2]	Yes	No[2]	Yes	Yes	Yes
* Recirc. Bypass MOV-54B (N.O., A.C.)	Yes	Yes	No[2]	Yes	No[2]	Yes	Yes	Yes
LPCI Loop B Availability:	ST[8]	ST[8]	No[1]	ST[8]	No[1]	ST[8]	ST[8]	ST[8]

Example Case Study
Single Failure of DC Bus 1 with LNP

SINGLE ACTIVE FAILURE AVAILABLE ECCS SYSTEMS	No Break	Break Location: Recirc. Discharge Line, Loop A	Recirc. Discharge Line, Loop B	Recirc. Suction Line, Loop A	Recirc. Suction Line, Loop B	Core Spray Line	Feed Water Line	Main Steam Line
CS Train A:								
- Pump P-46-1A (via EDG-1A/Bus 4)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-11A (N.O., R.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-12A (N.C., A.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-5A Min. flow valve (N.O., A.C.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* RRU-7, NE Corner Room Cooler	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
CS Train A Availability:	Yes	Yes	Yes	Yes	Yes	No(1)	Yes	Yes
INDEPENDENT SYSTEMS AVAILABLE FOR LONG TERM(13) TORUS COOLING MODE (with DC-1 failure)								
RHR/Torus Cooling Loop A via EOPs:								
- RHR/LPCI Pump P-10-1A	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-16A Min. flow valve (N.O., A.C.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* RRU-7, NE Corner Room Cooler	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-65A (N.O., R.M.C.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-34A (N.C., R.M.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
RHRSW Pumps P-8-1A and P-8-1C	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* MOV-89A (N.C., R.M.O.)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
* RRU-5, NE Corner Room Cooler	Yes(10)	Yes(10)	Yes(10)	Yes(10)	Yes(10)	Yes(10)	Yes(10)	Yes(10)
SW Pumps P-7-1A and P-7-1C, and SW MOV-20	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
RHR/Torus Cooling Loop A Long Term Availability:	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes

Notes:
(1) LOCA analysis assumes no credit for ECCS delivery to broken loop.
(2) LOCA analysis assumes no credit for closure of recirc. loop discharge valve in broken loop for discharge and suction line breaks.
(3) HPCI available, but not credited in LOCA analysis.
(4) HPCI assumed failed due to break location in steam line. Not credited in LOCA analysis.
(5) RCIC available, but not credited in LOCA analysis.
(6) RCIC assumed failed due to break location in steam line. Not credited in LOCA analysis.
(7) Performance of SRV on broken MSL may be degraded due to break location.
(8) Available for short term (ST) LPCI Injection Mode only. Injection from this path credited in LOCA analysis.
(9) Min. flow valve fails open, no auto closure on high flow. Analysis assumes valve remains opens and causes small flow diversion.
(10) Room cooler RRU-5 not needed when RRU-7 is operating.
(11) Room cooler RRU-6 not needed when RRU-8 is operating.
(12) Short term (ST) refers to post accident period including vessel blowdown, lower plenum refill, and core reflood.
(13) Long Term (LT) refers to post accident period including sustained core cooling w/ clad temperature near Tsat.

N.C./R.C. = Normally Closed/Remain Closed
N.O./R.O. = Normally Open/Remain Open
A.O./A.C. = Auto-Open/Auto-Close
F.O./F.C. = Fail Open/Fail Closed
R.M.O./R.M.C = Remote Manual Open/Remote Manual Close

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ECCS Single Failure Assessment

Summary

- Did not identify any new single active failures

- ECCS availability matched design basis LOCA analysis

Safety Assessment

RHR Pump Performance

- Immediate operability assessment
- A PWR operated an RHR pump 66 minutes in a no-flow condition without damage
- Another PWR tested a spare RHR pump in a no-flow condition for approximately one hour without damage
- A BWR operated an RHR pump in a no-flow condition for more than one hour without damage

Safety Assessment

RHR Pump Performance

- Sister utility ran RHR pump for five hours in no-flow condition (1981)
- Pump remained operable
- Pump tested three days later - no degradation in head/capacity
- Pump test showed low vibration levels giving assurance of no mechanical damage
- Pump was of same manufacturer and model as Vermont Yankee pumps

Safety Assessment

LOCA Analysis Impact

- Original ECCS design basis analysis relied only on CS and ADS
- SAFE/REFLOOD ECCS analysis results (intermediate-to-large breaks) unaffected by availability of RHR pump
- Realistic ECCS analyses using SAFE for small breaks with one CS System + two ADS valves: PCT < 2200°F
- RELAP5YA-BWR beyond design basis analysis - one core spray + ADS: maximum PCT = 1806°F
- RELAP5YA-BWR design basis analysis - current, RHR minflow valve open: maximum PCT = 1793°F
- With current analysis methods (RELAP5YA-BWR) 10CFR50.46 limits would not be exceeded due to loss of all LPCI flow

Safety Assessment

IPE Impact

- IPE correctly modeled RHR minimum flow valves closed
- IPE assumed closed valves would fail RHR pumps for small and medium LOCA events
- Overall CDF of 4E-6 not affected by discovery of design basis issue
- All LOCAs are 2% of CDF
- RHR pump failures due to minimum flow valves did not appear in any LOCA sequence >1E-9

Safety Assessment

Summary

- RHR pumps are expected to work

- ECCS criterion is met

- IPE CDF is not impacted

Enforcement Considerations

Identification

- Self-identified through comprehensive corrective actions from a previous enforcement action (Appendix R)

- Not identified through an event

- Involves an old design issue

Enforcement Considerations

Corrective Actions

- Prompt and comprehensive corrective actions
  - Immediate operability assessment
  - Basis for maintaining operation
  - 10CFR50.59 evaluation
  - Licensee event report
  - Comprehensive root cause analysis
  - Extensive management oversight

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Conclusion

Design Basis/Engineering Program Review

- Threshold for findings low

- Actions on findings more consistent

- Management reinforcement of questioning attitude