October 16, 1997

EA 97-453

Mr. Donald A. Reid
Senior Vice President, Operations
Vermont Yankee Nuclear Power Corporation
RD 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

SUBJECT: NRC INSPECTION REPORT 50-271/97-80 AND EXERCISE OF DISCRETION

Dear Mr. Reid:

This letter refers to the engineering inspection conducted from July 28, 1997, to August 22, 1997, at the Vermont Yankee Nuclear Power Station. The purpose of the inspection was to determine whether activities authorized by the license were conducted safely, and in accordance with NRC requirements. At the August 22, 1997, exit meeting, the findings were discussed with Mr. Greg Maret and other members of your staff.

The inspection was directed toward areas important to public health and safety. The areas examined during this inspection included: 1) your corrective actions in response to the identification of Appendix R program issues; 2) your corrective actions regarding fire barrier penetration seals; and 3) your evaluations of four previously identified NRC inspection items.

We reviewed the circumstances surrounding your identification and corrective actions regarding Appendix R issues found in 1995 and 1996. Issues were identified by your staff during a comprehensive Appendix R program review, started upon your discovery of similar problems in 1995.

The problems were violations of NRC requirements of 10 CFR Part 50.48, Appendix R, Section III.G, which would normally be considered for escalated enforcement and subject to a civil penalty. However, after consultation with the Director, Office of Enforcement, I have been authorized to not issue a Notice of Violation and not propose a civil penalty in this case in accordance with the provisions of Section VII.B.4 of the NRC's Enforcement Policy. This decision was made after consideration that: (1) the violations were identified by your staff as part of the corrective action for the previous Appendix R related issues with the RCIC, ADS and EDG systems; (2) they had the same root cause as the previous issue; (3) they did not substantially change the safety significance or the character of the regulatory concern arising out of that finding; and (4) corrective actions, both taken and planned, were comprehensive and reasonable. The exercise of discretion acknowledges your effort to identify and correct subtle violations, that would not be readily identified by routine efforts, before the degraded safety systems are called upon.

I-4

Based on the results of this inspection, your corrective actions for four violations regarding Appendix R program and fire barrier penetration seals were appropriately resolved and these items are closed. Your response to three related unresolved items were found satisfactorily resolved and these items were closed. In addition, three unresolved items associated with the electrical systems were found adequately resolved, and these items were also closed.

No reply to this report is necessary and your cooperation with us is appreciated.

Sincerely,
ORIGINAL SIGNED BY
W.L. AXELSON FOR

Hubert J. Miller
Regional Administrator

Enclosure: NRC Inspection Report 50-271/97-80

Docket No. 50-271

cc w/encl:
R. McCullough, Operating Experience Coordinator - Vermont Yankee
R. Wanczyk, Director, Safety and Regulatory Affairs
G. Maret, Plant Manager
J. Duffy, Licensing Engineer, Vermont Yankee Nuclear Power Corporation
J. Gilroy, Director, Vermont Public Interest Research Group, Inc.
D. Tefft, Administrator, Bureau of Radiological Health, State of New Hampshire
Chief, Safety Unit, Office of the Attorney General, Commonwealth of Massachusetts
D. Lewis, Esquire
G. Bisbee, Esquire
J. Block, Esquire
T. Rapone, Massachusetts Executive Office of Public Safety
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-271
Licensee No:	DPR-28
Report No:	50-271/97-80
Licensee:	Vermont Yankee Nuclear Power Corporation
Facility:	Vermont Yankee Nuclear Power Station
Location:	Vernon, Vermont
Dates:	July 28 - August 22, 1997
Inspectors:	Ram S. Bhatia, DRS George W. Morris, DRS Keith A. Young, DRS
Approved by:	William H. Ruland, Chief Electrical Engineering Branch Division of Reactor Safety

EXECUTIVE SUMMARY

Vermont Yankee Nuclear Power Station
NRC Inspection Report 50-271/97-80

This special team inspection was conducted: 1) to review corrective actions implemented in response to the fire protection and Appendix R program issues identified both during NRC inspections (NRC inspection reports 93-05 and 95-26) and by the licensee reports (LERs 95-14, Supplement 3, 96-09 and 96-18); and 2) to review corrective actions for resolving electrical issues identified during previous NRC inspections (NRC inspection reports 96-09, 97-03, and 97-04).

Fire Protection

- Overall, the licensee had made good progress in resolving fire protection and Appendix R program related issues; however, additional effort remains to be completed to achieve full compliance with 10 CFR 50, Appendix R requirements. The team determined that appropriate interim corrective actions were in place to compensate for identified Appendix R program deficiencies. (Section F8)

- VY staff's current understanding of fire protection and Appendix R requirements was found to be good. (Section F8)

- The NRC considers your initiatives to identify long-standing Appendix R and fire program design issues a strength. Corrective actions taken and planned to prevent recurrence of similar design deficiencies were comprehensive. Due to your comprehensive response, the NRC has decided to exercise discretion and not cite the violations of NRC requirements identified during your review of the Appendix R and fire protection programs. (Section F8.5)

- Four previously identified violations and three unresolved open items associated with the fire protection and Appendix R program were adequate and these items were closed. One additional Inspection Followup Item (IFI) was opened as result of this inspection. (Section F8)

Engineering

- Three out of the four previously identified straight-forward electrical open items were correctly resolved. (Section E8)

- VY continues to review and correct cable separation issues. The inspectors found some minor cable tray installation problems that did not affect equipment operability. (Section E8)

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Report Details

F8 Miscellaneous Fire Protection Issues Review (64150)

Background

In July 1995, the licensee identified that Vermont Yankee Nuclear Power Station (VY) was not in compliance with several Appendix R requirements. An NRC inspection (Inspection Report 50-271/95-26) was conducted in October 1995, that reviewed the identified deficiencies and their short term corrective actions. As a result of this inspection, an enforcement conference was held (January 11, 1996) and a Notice of Violation (NOV) issued (February 13, 1996). In response to the NOV, the licensee committed to review their safe shutdown capability analysis (SSCA) and verify the plant design requirements and other applicable documentation to ensure full compliance with 10 CFR 50, Appendix R requirements.

Two followup NRC inspections (Inspection Reports 96-08 and 96-10) were conducted in 1996 to review the licensee's response to the NOV. The licensee identified several significant fire protection and Appendix R issues during their comprehensive review of the fire protection program. To address these issues, the licensee made significant changes in their safe shutdown capability analysis (SSCA). In addition, several exemption requests were submitted to the NRC to address issues where literal compliance was not feasible or practical.

This special team inspection was conducted: 1) to review licensee's corrective actions taken during response to the fire protection and Appendix R program issues identified during previous NRC inspections (Inspection Reports 93-05 and 95-26) and by the licensee (LERs 95-14, 96-09 and 96-18); and 2) to review licensee's corrective actions for resolving electrical issues identified during previous NRC inspections (Inspection Reports 96-09, 97-02, 04, and 05).

F8.1 (Closed) Violation EA 95-268 1013, Multiple Appendix R Program Deficiencies

a. Inspection Scope (64150)

This violation was cited for several problems regarding procedures and plant equipment for safely shutting down the plant, in the event of a postulated fire. Details of the concerns are provided in the NRC Inspection Report 50-271/95-26 and LER 95-14. The team reviewed the licensees's corrective actions taken in response to these deficiencies.

b. Observations and Findings

In response to the NOV dated March 12, 1996, the team noted that the licensee had promptly instituted interim compensatory measures to mitigate the consequences of fire in various plant areas. These interim compensatory measures were thoroughly evaluated and found acceptable during two NRC follow up inspection in 1996 (96-08 and 96-11). The team observed that these measures

were still in place because the licensee had not completed updating the applicable procedure and providing the appropriate training to the staff to implement the safe shutdown strategy.

The team noted that VY had completed the following long term corrective actions:

1. Re-analyzed and updated the safe shutdown capability analysis (SSCA).

2. Updated the applicable procedures (OP-3126, OP-3020, and others) to account for compensatory measures and the new SSCA methodology.

3. Eliminated the reliance on fuse replacement "repairs" to achieve hot shutdown.

4. Provided Appendix R design basis training to plant personnel responsible for maintaining it.

5. Evaluated all the existing exemptions and submitted new requests.

6. Reviewed and enhanced the VY licensing process for submitting requests to the NRC.

The team verified a selected sample of VY's long term corrective actions, including the SSCA, updated shutdown procedures, new SSCA strategy, equipment availability, and staff training. The team found that the revised SSCA was comprehensive and contained a sufficient level of detail. The team reviewed the SCCA for its use of selected safe shutdown equipment, and found no concerns. The team found that VY's interim shutdown procedure, OP 3126, "Shutdown Using Alternate Shutdown Methods," and OP-3020, "Fire Emergency Response Procedure," clearly indicated the systems relied upon to safely shutdown the plant in the event of a fire. The team checked the procedure by directly observing that the installed plant equipment controls were consistent with the procedural steps. The team interviewed several plant personnel and Yankee Atomic Electric Company (YAEC) design engineers responsible for the fire protection program activities, and determined that they were knowledgeable of fire protection program requirements. The training module, used for fire protection and Appendix R training, was thorough with a strong emphasis with new SSCA strategy and program procedure requirements.

The team reviewed selected safe shutdown system components and their locations and determined that the licensee had eliminated the vulnerabilities to the ADS, RHR, lighting and support systems listed in ten modifications used to implement the revised SSCA.

c. Conclusions

The team concluded that the licensee had appropriately implemented the long term corrective actions stated in their response to the NOV. The interim compensatory

measures were still in place where needed. The licensee proposed actions and schedule to revise the applicable procedures and train the staff for the new SCCA strategy were appropriate. This item is closed.

F8.2 (Closed) Violation EA95-268 0204 Pertaining to the Appendix R Fuse Exemption Deficiencies in the Emergency Diesel Generator Support Systems (64150)

a. Inspection Scope

This violation documented a noncompliance regarding the replacement of EDG fuses for shutdown repairs which was inconsistent with an NRC Appendix R exemption. The team reviewed the actions taken to resolve this noncompliance.

b. Observations and Findings

The licensee response letter to the NOV, dated March 12, 1996, stated that the cause of this was due to an inadequate review by the technical staff.

The team noted the licensee had completed the following corrective actions to address this concern:

- VY completed a review of all the Appendix R program exemptions for accuracy and consistency with the license conditions to assure compliance with the Appendix R requirements. All identified deficiencies were appropriately addressed.

- VY performed a self-assessment of their Safety Evaluation Reports (SERs) of fire protection and environmental qualification, inservice testing (IST) and containment leak rate (Appendix J) testing programs to assure no similar concerns existed in those program areas.

- VY implemented design change EDCR 96-401, which installed several isolation switches and redundant backup fuses in the EDG support system control circuits. This design change eliminated the reliance on fuse replacement as part of the alternate shutdown procedure (OP3126). VY plans to withdraw the existing fuse exemption when full compliance with Appendix R requirements is achieved.

  The team verified that the licensee had correctly installed the EDG related switches and redundant fuses in various control panels. In addition, existing pre-staged fuses were also available for fuse replacement.

- The team reviewed the results of VY's Appendix R self-assessment of various exemptions in their Appendix R program, as documented in their internal memorandum OPVY 305-96, and noted that the licensee had appropriately resolved all identified exemption deficiencies. The assessment was comprehensive and the licensee had reviewed all relevant correspondence, SERs, and exemptions. The team also reviewed the self-

assessment audit report (VY-96-27 dated January 17, 1997), conducted by a VY specialist and an independent contractor, to assess the adequacy and effectiveness of the VY's inservice testing (IST) program. The self-assessment results indicated that the licensee's calculations and assumptions were correct and well documented.

- The team noted that VY was testing a new process (Trial use of new procedure VYP-132) to assure that all NRC-approved SERs issued are reviewed for consistency with the proposed change prior to their implementation. A draft administrative procedure was issued to various departments for their review and concurrence. The team verified that the VY staff effectively reviewed the accuracy and the adequacy of license amendments and exemptions.

c. Conclusions

The team concluded that the licensee had appropriately implemented corrective action to resolve the deficiency cited in the violation. This item is closed.

F8.3 (Closed) URI 95-26-01; Inadequate Post-Fire Shutdown Procedure

a. Inspection Scope (64150)

During a 1995 NRC Appendix R inspection (Inspection Report 50-271/95-26), a concern was identified that the post-shutdown procedures were inadequate, in that, in the event of the control room evacuation, fire in the cable vault, or other plant fire areas, the procedure only provided instructions for alternate shutdown with a concurrent loss of offsite power. Consequently, the operators were not required to verify the component status to protect various equipment against possible spurious operation if the loss of offsite power (LOOP) does not occur. At that time, the licensee stated that they would evaluate this general concern in their ongoing Appendix R program review and take appropriate corrective action.

b. Observations and Findings

Vermont Yankee revised operating procedures and modified selected control circuits to accommodate a fire-induced control room evacuation without a LOOP. The team verified that selected steps in the procedures were modified or added to address the above condition. For example, the inspectors found a procedure step that placed the reactor feed pumps in pull-to-lock, which would prevent overfilling the reactor vessel if off-site power was still available. The procedure changes were supported by a change to the SSCA and installing modification EDCR 96-411.

c. Conclusions

Based on the above review, the team concluded that the licensee had appropriately addressed the above plant equipment issues. This item is closed.

F8.4 (Closed) URI 95-026-02; Spurious Operation of RHR System

a. Inspection Scope (64150)

The NRC highlighted a previously-identified VY concern that during certain postulated fires, the RHR system may be susceptible to water hammer flow diversion, and pump damage due to coincident start of a pump with the failure to open the discharge or minimum flow line valves. These issues were documented in the licensee's plant event report (ER 95-0628), LER 95-14, and the NRC Inspection Report 50-271/95-26.

b. Observations and Findings

To address the RHR system water hammer concern, VY reviewed the results of their integrated ECCS test OP-4100, conducted during the 1996 refueling outage. In addition, licensee determined that two valves have to spontaneously open to cause a water hammer. Based on the above, the licensee concluded that the RHR system was not susceptible to water hammer.

The licensee evaluated the potential of RHR flow diversion and concluded that a possible diversion path may exist due to the spurious operation of the RHR pump motor-operated discharged valves. To address this concern in the interim, VY enhanced the existing shutdown procedure to preclude any diversion paths. A planned design change was being tracked as a part of the remaining SSCA work to provide a more permanent resolution of this issue. The team concluded that the potential diversion path was appropriately addressed by the implemented procedure changes.

The team noted that the licensee's SSCA system analysis evaluated the potential for a spurious RHR pump starting with the simultaneous failure of the RHR discharge or minimum flow line valves to open. The evaluation determined that this concern was only applicable to "A" RHR pump for alternate shutdown scenarios. The licensee revised the procedure to include a step to place this pump control switch in the "pull-to-lock" position, and subsequently trip the pump breaker locally in the switchgear room. The licensee had also implemented a design change (EDCR 96-411) that enhanced the existing control circuits of the RHR valves to alleviate this concern.

The team reviewed the design documentation and performed a walkdown of selected portions of the installation.

c. Conclusions

The team concluded that the licensee had appropriately addressed the issues associated with the RHR system equipment. The team concluded that the root cause of these issues was similar to the previous escalated enforcement issue (SSCA design assumption errors) and since they were identified by the licensee in their event report (LER 95-14 and ER 95-0628), and were promptly appropriately

corrected, they satisfied the Enforcement Policy, of Section VII.B.4. Therefore, the NRC is exercising discretion and not citing this violation. This item is closed.

F8.5 Miscellaneous Fire Protection Issues Identified in LERs (95-14, 96-13, and 96-18)

a. Inspection Scope (64150)

The team reviewed the licensees's corrective actions taken for the resolutions of the following deficiencies identified in their fire protection program as documented in Licensee Event Reports 95-14, Supplement 3, 96-009, 96-13, and 96-18.

b. Observations and Findings

The team reviewed the resolutions of the documented fire protection issues described in the LERs and in the previous NRC inspection reports (95-26, 93-05, 96-08, and 96-10). The team noted that these issues were identified by the licensee as a result of their ongoing fire protection improvement program (FPIP) and Appendix R upgrading efforts conducted in response to the previous NRC escalated enforcement violations (EA 95-268).

Fire Induced Hot Short Concerns

LER 95-14, Rev. 3, Identified March 13, 1996

The ammeter cables running from the 4160 V Alternate Shutdown (ASD) load in the switchgear room through the cable vault to the control room were not isolated by the original ASD design changes. A fire in the control room or cable vault, due to the existing wiring configuration of the ammeter cables, could have caused safe shutdown system loads such as RHR and SW to trip, and the operator may not have been able to recover operation of the systems at the alternate shutdown panel.

The licensee had promptly instituted a compensatory measure and corrected this problem by implementing a design change to isolate the ammeter circuits, as part of the fuse replacement modification (EDCR 96-401). The team verified the circuits by reviewing the design documents and found no concerns.

Routing Concern

LER 95-14, Rev. 3, Identified December 5, 1995

In the event of a fire in Reactor Building (fire zone RB-3), the RCIC system would become unavailable due to loss of normal and alternate 125 Vdc feed supply to RCIC turbine controller. RCIC was required for safe shutdown of the plant in fire zone RB-3 per the Safe Shutdown Capability Analysis (SSCA) and plant alternative shutdown procedure.

The licensee had promptly instituted a compensatory measure upon discovery of this problem. The team noted that the above change is no longer needed, because under the new approved safe shutdown methodology, the licensee no longer relies on the RCIC system.

LER 95-14, Rev. 3, Identified March 26, 1996

The RHR pump room coolers automatically start when either of the RHR pumps or CS pumps in their respective rooms start. The auto start control circuit consists of control wiring between Bus 3 & 4 in the West and East Switchgear Rooms, respectively. This wiring between busses was necessary when the RHR pumps were cross powered in 1974. The effect on Appendix R is that for a West Switchgear Room fire, when the "A" RHR pumps are powered from Bus 4 in the East Room, the "A" train room cooler may not be available. The same is true for the "B" RHR pump.

A fire in either of the switchgear rooms (east or west), due to an induced "hot short" in control logic circuitry of ECCS system valves and loss of HVAC in the ECCS corner room, would have rendered both RHR trains inoperable. The licensee had corrected this routing problem by implementing a design change (EDCR 96-411).

Assumption error in VY's SSCA (outside the existing Appendix R Design Basis)

LER 95-14, Rev. 3, Identified March 28, 1996

- A potential loss of drywell cooling was not considered in evaluating the safe shutdown systems that may be required to safely shutdown the plant in the event of fire in the plant. No provision existed to monitor drywell temperature and other conditions in the remote shutdown panels to safely shutdown and maintain the plant in cold shutdown condition. The loss of drywell cooling may result in: (1) exceeding the containment shell design temperature, (2) reactor vessel level instrumentation reference leg may flash, (3) the EQ qualification for important components such as the SRV solenoids may be challenged.

- The current SSCA identifies that the safe shutdown strategy for a fire in the Reactor Building (zone RB-4) involves depressurizing the reactor, flooding the vessel using the CS system, supplying a return path to the torus through an open SRV, and cooling the torus with the RHR system. In order to maintain this flow path for a long term shutdown nitrogen or air must be supplied to the SRV's via the containment air system. Neither the SSCA nor the operating procedures, identify the need to restore makeup to the containment air system in the event of a RB-4 fire.

In the event of a fire in RB-4, for a long-term shutdown, restoration of nitrogen or make up air to operate a SRV, required to safely shutdown and maintain the plant in cold shutdown was not considered. In addition, appropriate shutdown procedures were not established.

To alleviate this concern, the licensee completed a containment heat up analysis (VYC-1457) and now has established a new strategy to monitor and control the parameters in the drywell or to initiate the containment spray system, if required, if the drywell temperature is increased to unacceptable level. In addition, a new temperature indicator (EDCR 96-404) was added and the applicable shutdown procedures were updated to deal with this condition to ensure temperature stays within the acceptable level.

The licensee had also included the contingency plan to use nitrogen supply located in the yard to operate required SRVs. The necessary exemption to use the emergency lighting in the yard for performing the above actions to satisfy the Appendix R lighting requirements was obtained from the NRC.

Fire Barrier Assumption Concerns

LER 95-14, Rev. 3, Identified October 31, 1995

- Two uncoated cables (R131SI in torus area N.W. corner and R237SI located 280 elevation East) were found within a combustible free zone (CFZ). Per the NRC approved exemption (1985), VY was required to protect all wires in the CFZ with thermo-plastic to satisfy the exemption.

  The team noted that VY instituted compensatory measures and corrected this condition by applying the required coating (WOs 95-8876 and 8875 for R237SI). The root cause was human error in assuring vendor barrier installation was completed.

- Several conduits in a CFZ between RB-3 and RB-4 were not sealed. These were required to be sealed with fire stops to satisfy the NRC exemption.

  The team noted that VY established a prompt compensatory measure. The licensee had sealed these conduits appropriately to permanently resolve this problem.

LER 95-14, Rev. 3, Identified June 19, 1996

- The separation horizontal distance between the two safe shutdown redundant trains (bus 9B; RHR and CS systems in the northwest corner area of the reactor building; RB-3) was found to be 15 feet instead of the 18 feet in the approved exemption.

VY has been granted an exemption for the 15 feet separation and additional actions are being taken to satisfy the sprinkler system installation in this area.

- Two redundant 24 Vdc system feeds to redundant ECCS instrumentation permissive interlocks for LPCI and CS system were not protected by an equivalent of three-hour barrier in the 280 feet reactor building area or equivalent of 1-hour barrier, with detection and suppression, in the area of recirculation MG set. In the event of a fire, under this condition, above redundant safety system may not have been able to safely shutdown the plant as required per the SSCA.

  The team noted that VY had posted fire watches in the areas due to other issues and upon discovery of this issue, they promptly changed the ongoing hourly roving fire watch to a continuous fire watch to mitigate the consequence of this deficiency. In addition, VY corrected this problem by rerouting the redundant cables.

LER 96-009, Rev. 0, Identified April 4, 1996

A combustible polystyrene material was still present in the joints between the turbine, radwaste, reactor, and control building fire areas. Three-hour fire-rated seals were required to separate the redundant safety systems in all fire areas.

The team noted that VY promptly started roving fire watches in these areas. The team verified by walkdown that this deficient condition was corrected by removing the unwanted material and adding appropriate foam seal to meet the 3-hour barrier requirements by implementing station work orders (96-06314-00) and minor Modification No. 96-024.

LER 96-18, Rev. 0, Identified August 15, 1996

A potential degraded 1-hour rated fire wrap (1/4 x 3-inch gap) was not installed since 1986 modification on conduits of redundant safe shutdown cables at elevation 232 in the northwest corner of the reactor building. As a result of this condition, it could have affected the performance of redundant safe shutdown systems (CS, ADS, RCIC) in the event of fire in area RB-1.

The team noted that VY had compensatory measures already in place in this area. The gap was repaired (W.O NO 96-5068-91) as per the manufacturer design requirements and the licensee provided training to the staff, including supervisors, to prevent the recurrence.

LER 95-14, Rev. 3, Identified July 11, 1996

A radwaste hallway (fire area FA-13) was not separated by a barrier from the adjacent fire area (turbine building). As a result of this condition, normal and alternate 24 Vdc to RCIC system control circuitry, could have been rendered unavailable due to a fire in either area.

The team verified that VY has upgraded the existing door in the radwaste hallway to provide the appropriate fire barrier to satisfy the Appendix R requirements of III.G.2.c.

LER 95-14, Rev. 3, Identified January 19, 1996

An exemption did not exist for the use of Fire Zone R cable (Rockbestos) in the cable vault area instead of a fire barrier system to meet the Appendix R Section III.G.2.c requirements.

The team noted that VY had promptly implemented a fire watch and standing order 17 to alleviate any fire concern and on May 28, 1996, requested an exemption from the requirements of 10 CFR 50, Appendix R, Section III.G.2.c. By NRC letter dated June 9,1997, an exemption was granted by the NRC.

LER 96-13, Rev. 0, Identified May 23, 1996

The foam deluge systems were found inadequate to meet its intended design requirements (foam suppression system as per the manufacturers recommended design coverage to achieve NFPA 16 requirements of a 0.16 gpm/ft^2 discharge density of 30 psi). The existing system provided foam on the reactor building 280-foot elevation above the reactor recirculation motor-generator sets.

The team noted that the licensee had established fire watches promptly and had modified the system (EDCR 96-407) by replacing the 4 inch pipe with 8 inch pipe from TB supply line to the RB supply line with two isolation valves. In addition, VY also modified the entire array of suppression nozzles, replaced the 100 gallon foam tank with a 150 gallon tank.

LER 96-13, Rev. 0, Identified June 7, 1996

The hydraulic calculations for the pre-action fire suppression system (water sprinkler system for the reactor building 252/232-foot elevation above and below the cable trays area) failed to included the loss of pressure of 4-inch main feed line located in the turbine building area. As a result, VY's commitment to satisfy a 0.30 gpm/ft^2 discharge density value was not satisfied.

The team verified that VY had modified the applicable sprinkler system piping to provide necessary suppression water to satisfy the fire code.

The team noted that upon discovery these deficiencies, the licensee had implemented prompt compensatory measures in accordance with Technical Specification (TS) requirements. The team determined that the operability determinations and compensatory measures were appropriate. The licensee had completed the evaluations and actions and was in the final implementation phase for the last two exemptions issued on August 12, 1997.

c. Conclusions

The above problems were violations of NRC requirements of 10 CFR Part 50.48, Appendix R, Section III.G, which would normally be considered for escalated enforcement and subject to a civil penalty. However, the NRC will not issue a Notice of Violation and not propose a civil penalty in this case in accordance with the provision of Section VII.B.4 of the NRC's Enforcement Policy. This decision was made after consideration that the violations: (1) were identified by licensee staff as part of the corrective action for the previous issue with the Appendix R related issues with the RCIC, ADS and EDG systems; (2) had the same root cause as the previous issue; and (3) did not substantially change the safety significance or the character of the regulatory concern arising out of that finding. Additionally, corrective actions, both taken and planned, were comprehensive and reasonable.

F8.6 (Closed) URI 94-31-02: Pertaining to Appendix R Emergency Lighting Issue (64150)

a. Background and Inspection Scope

The team reviewed the licensee's corrective actions taken to resolve the emergency lighting deficiencies.

b. Observations and Findings

The team found that the licensee had completed upgrading and reevaluating lighting deficiencies identified in their shutdown procedures based on assumptions made in the Safe Shutdown Capability Analysis (SSCA). The team verified that appropriate analysis had been conducted by the licensee which resulted in additional emergency lighting units being installed in the RCIC east switchgear, "A" diesel generator, and in front of the 4kv switchgear bus 4 areas to augment existing lighting. Installations were also completed in the cable vault and near the SRM-IRM drive control panel as well as other appropriate areas of the plant. The installations and appropriate emergency lighting were verified by the team during a walkdown of procedure OP3126, "Shutdown Using Alternate Shutdown Methods," Revision 14, October 26, 1996. Procedures were also appropriately revised by the licensee to reflect concerns for deficient emergency lighting.

c. Conclusion

The team concluded that the licensee performed adequate analysis to correct emergency lighting deficiencies in various areas of the plant. Additional emergency lighting units were installed and shutdown procedures were adequately revised to address alternate shutdown capability emergency lighting deficiencies. This item is closed.

F8.7 (Closed) Violation (93-05-01) Pertaining to the Penetration Seals Deficiencies

a. Inspection Scope (64150)

The team reviewed the licensee's corrective actions taken to address penetration seals deficiencies. NRC noted that several fire barrier penetration seals required to protect the reactor building, control room, and diesel generator rooms were not intact (degraded). Specifically, numerous fire barrier penetration seals were found degraded due to: 1) inadequate depth of penetration fill material; 2) unqualified fill material; 3) installation of unqualified designs; and 4) through wall cracking of fire barriers. These issues were documented in NRC inspection report 50-271/93-05.

b. Observations and Findings

The licensee root cause analysis of the deficient penetration seals determined that the cause was due to inadequate documentation of assumptions, inadequate procedures, and human error. To address deficiencies, the licensee performed a walkdown of the fire barrier seal installations (walls, floors, ceilings) to verify the adequacy of configuration of the seals. In 1993, several design changes were made to seal configurations to bring the penetration seals into compliance with design requirements. The licensee also made changes in their administrative and surveillance procedures to clearly define the staff responsibilities and the acceptable seal designs. Since 1993, the licensee has performed routine surveillance inspections of barrier and seals, and the defective seals had been appropriately repaired. The team inspected penetration seals in various plant areas and noted that the penetration seals were acceptable.

The team reviewed VY's self-assessment performed during the 1995 refueling outage which concluded that fire penetration seals integrity was significantly improved. The corrective action for degraded seals was being adequately tracked. The VY staff interviewed were well aware of the penetration seal design requirements and inspection procedures.

Per discussion with the licensee, the team noted that the VY's special project team was again performing a special walkdown of all penetration seals. A limited sample review of penetration seal data gathered at this time indicated no concerns. Based on the walkdown verification and the completed corrective actions to-date, the inspectors concluded that the licensee actions completed were adequate to address the penetration seal issue. However, the licensee efforts to develop a database was still ongoing and was expected to be completed by the end of this year. This

item will remain open pending NRC review of the licensee's completed penetration seal database **(IFI NO. 97-80-01)**.

c. Conclusions

The inspectors concluded that the licensee had appropriately addressed the issues associated with the penetration barrier seals. This item is closed.

F8.8 (Closed) Violation No. 93-05-02 Regarding Surveillance Testing Not Being Performed Per the T.S. for Plant Penetration Seals

a. Inspection Scope (64150)

The team reviewed the licensees's corrective actions taken in response to Notice of Violation (NOV) issue associated with a lack of adequate surveillance procedure for testing fire protection penetration seals as required per their Technical Specification. The NRC was concerned that the procedure lacked the specific acceptance criteria and direction to accept/reject seals. As a result of inspections conducted from 1980 until 1992, the licensee failed to identify that numerous penetrations were not functional due to various deficiencies and degradation.

b. Observations and Findings

The team noted that the licensee, in their response to the NOV dated June 11, 1993, indicated that the problem occurred due to an inadequate surveillance procedure. Because the procedure OP 4019, "Surveillance of Plant Fire Barriers and Fire Rated Assemblies," lacked the specific acceptance criteria and direction for their inspection personnel.

The licensee has revised the surveillance procedure (OP-4019, Revision 12), several times since the issuance of the violation in 1993. The review indicated that the licensee had added detailed guidance and specific barrier acceptance criteria to perform a thorough inspection for both smoke/gas seals and penetration seals.

VY conducted surveillance testing during the 1993, 1995 and 1996 refueling cycles to ascertain the operability of penetration barriers/seals.

c. Conclusions

The team concluded that the licensee had appropriately revised the procedure to include sufficient detail and were appropriately conducting the required surveillance. This item is closed.

E8 Miscellaneous Electrical Items

E8.1 (Update) URI 97-03-02 Pertaining to Electrical Cable Separation Concerns (37551)

a. Scope of Inspection

Previous NRC Inspection Report 97-03 documented an initial evaluation of concerns that the licensee had found in the area of electrical cable separation. The licensee had issued several event reports (ER) recently on cable separation concerns as well as two licensee event reports (LERs). LER 96-028, issued on November 18, 1996, identified inadequate field labeling of safety class wiring and drawing updates related to a 1976 design change installation. LER 97-006 documented the licensee's finding of numerous examples of failure to maintain proper electrical separation. The licensee indicated that they were still addressing the cable separation problems, therefore, the team limited their review to the use of non-nuclear safety (NSS) breakers and a limited field walkdown of cable.

b. Observations and Findings

The team learned that the licensee planned to use NNS circuit breakers as a second level circuit protection to assure cable insulation protection. The team noted that some of the circuits were 125 Volt dc control circuits and questioned the dc interrupting rating of the molded case circuit breakers (MCCBs) on dc circuits. The licensee was able to confirm that the upstream dc panel had fault duties up to 9000 Amps. However, the licensee confirmed that some of the MCCBs were located in remote control panels and had a 5000 Amp dc interrupting rating but could not substantiate the interrupting fault duty at the questionable MCCBs. The licensee also indicated that these NNS MCCBs located in remote control panels were not included in any maintenance program.

During walkdowns of the cable raceway system, the team noted four examples where cable tray covers or conduit (wireway) covers were not left in their designed configuration. In the cable spreading room area, the team noted two examples of conduit being used as cable risers where the covers were missing and one example where a division SII covered cable tray, routed within a division SI open cable tray, did not have its cover placed properly, exposing the enclosed division SII cables. The team also noted in the switchgear area, a cover on a power cable tray lying directly on the cables and not supported by the cable tray rails as designed. In response, the licensee issued event report (ER) 97-1003 to add these issues to the planned system engineer material inspection walkdowns.

c. Conclusions

The team concluded that the licensee's corrective actions in the cable separation area were still being developed. The use of NNS MCCBs for circuit protection required additional development, especially for dc control circuits. This item will remain open pending NRC review of the licensee's corrective action. The team concluded that the misplaced cable tray covers did not, by themselves, introduce a safety concern because they did not directly expose cables of different separation groups to each other. However, the misplaced covers did indicate a weakness in workmanship.

E8.2 (Closed) IFI 50-271/97-04-06, Pertaining to 480 Volt Switchgear Bus 9 Loading Concern (92701)

a. Scope of Inspection

The team reviewed the follow-up effort by the licensee associated with 480 Volt switchgear Bus 9 overloading concern raised during a previous inspection (IR 97-04.) The earlier inspection questioned the modeling of the loading due to fire pump load. The scope of this inspection was to assess the licensee's corrective action.

b. Observations and Findings

The team reviewed the 480 Volt one-line diagram G-191301, Sheet 1, Rev 2, dated May 20, 1996 and sheet 2, Rev. 3, dated November 15, 1995, which showed the load that could be connected to Bus 9 consisted of large 460 Volt motor loads and feeds to motor control centers (MCCs.) The team noted that Bus 9 had a continuous rating of 1600 Amps but the transformer feeding it, T-9-1A, had a rating of 100/1150 kVA OA/FA for a 55°C temperature rise. The transformer also had a 1120/1288 kVA OA/FA rating for a 65°C rise. This higher rating indicated the transformer was capable of carrying up to 1549 Amps. The team estimated that if all loads that could be connected to Bus 9 were running, including the fire pump, the total connected load could exceed the load center transformer 65°C rise ratings by 52 kVA. The licensee presented calculation VYC-791, MCC Loading Calculation, Rev. 5, dated October 5, 1995, to demonstrate the worst-case MCC running load was less than the total MCC connected load estimated by the team. The team reviewed Section 5, Results and Conclusions, from that calculation, and noted the worst case MCC total load on Bus 9 was 85 kVA less than the team's estimate of total connected MCC load.

The licensee initiated event report (ER) 97-0999 in response to the earlier NRC observation documented in NRC Inspection Report 50-271/97-04 to address a 1992 modeling of the 480 Volt Bus 9 during starting of the fire pump. The licensee prepared calculation VYC-1688, Transient Voltage on 480 Volt Power System, dated August 22, 1997, to demonstrate the ability of the fire pump to start on a fully loaded bus. The inspectors reviewed this calculation and confirmed the analysis was conservative in that it assumed minimum voltage, non-accident conditions and normal power available. Accident conditions or loss of normal power would result in less load on Bus 9 when the fire pump was started. The results confirmed that the fire pump could be successfully started on Bus 9 at the minimum operable voltage of 435 Volts without overloading the bus or the load center transformer. The result of that calculation also indicated the normal steady state current through transformer T9 to Bus 9 would be 1071 amps. This provided further assurance that neither the bus nor the transformer would be overloaded beyond their current carrying capability.

c. Conclusion

The team concluded the normal and emergency running load would not overload 480 Volt Bus 9. The team also concluded the licensee's analysis of the transient loading of Bus 9 following the addition of the fire pump was responsive to the team's concerns. This item is closed.

E8.3 (Closed) URI 96-09-06, Pertaining to Main Battery Service Test Acceptance Criteria (92701)

a. Scope of Inspection

NRC inspection report (IR) 50-271/96-09 documented inadequate battery service test acceptance criteria associated with the main station batteries. A previous NRC inspection, 97-04, found that the licensee had revised the battery sizing and dc voltage drop calculations associated with the main station batteries but had not yet factored the results of those calculations into the station test procedures. The team reviewed the main station battery service test procedure to assess the licensee's final corrective action.

b. Observations and Findings

The team reviewed procedure OP 4215, Main Station Battery Performance/Service Test, Rev. 7, dated July 3, 1997. The team found that the acceptance criteria contained in section 4.6 of the procedure was consistent with the required voltage established by section 5.1 of calculation VYC 1349, 125 V dc Voltage Drop Calculation, Rev. 1, dated April 30, 1997.

c. Conclusions

The team concluded the licensee had correctly interpreted the requirements of the dc voltage drop calculation into the acceptance criteria for the main station battery service test. This item is closed.

E8.4 (Closed) URI 97-02-05, Advanced Offgas (AOG) System Crossed Neutrals Wiring (92701)

a. Scope of Inspection

NRC inspection report (IR) 50-271/97-02 said that the licensee's root cause evaluation of the AOG system crossed-tied neutrals, failed to assess the extent of condition to other safety-related or important-to-safety systems. The licensee responded to the inspection report with their letter dated July 8, 1997, indicating their corrective action for their extent of condition evaluation. The team reviewed the licensee's back-up information to assess their extent-of-condition review.

b. Observations and Findings

The team found the licensee's response letter failed to address the extent-of-condition to other systems. The team reviewed specification 33600-3401-00-1, Rev. 2, dated August 2, 1972, Specification for Catalytic Hydrogen-Oxygen Recombiner and Charcoal Offgas Retention System, the AOG system supplied for VY. The team confirmed that the specification required the vendor to provide a prewired control panel. The team reviewed the vendor's drawings A719-0206, Electrical Schematic Diagram, Rev. H, dated April 2, 1975 and A719-3205, Control Panel Wiring Diagram, Sheet 4, Rev. G, and Sheet 5, Rev. F, both dated April 3, 1975. The team confirmed that the vendor's schematic drawing was correct, but the vendor's wiring diagram was in error.

The licensee indicated they had performed a preliminary review of their drawing database and had not found any evidence that the AOG panel vendor had supplied drawings for any other system. They had documented this review in their internal memorandum from YAEC to VY (WO# 4100-XX) dated January 30, 1997. This memorandum had formed part of the basis of VY's closeout of event report 97-0092 on May 7, 1997. In their letter to the NRC (No. BVY 97-89, dated July 8, 1997), the licensee summarized their ER and stated that there were no current AOG wiring problems which could degrade safety-related, or important to safety systems. However, in response to the team's questions, the licensee was not initially prepared to demonstrate that the AOG panel vendor had not performed any other work at VY.

Later, the licensee indicated that they had performed additional review of the accounts payable records for the life of the plant and concluded the AOG panel vendor had not performed other work at VY. The licensee documented this review in their memorandum VYE 83/97, dated August 21, 1997.

c. Conclusions

Based on the observed drawing errors, the team concluded the problem with the AOG crossed neutrals originated with the supplier equipment (AOG control panel).

The team concluded the licensee was able to provide reasonable evidence that the AOG panel vendor had not performed other work for VY and they, therefore, could eliminate the possibility of similar wiring problems that systems due to this vendor's work. The team concluded this was an isolated event from 1975. This item is closed.

E9 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 inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. No discrepancies were noted.

E10 Exit Meeting Summary

The team leader met with the licensee representatives on August 22, 1997. 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.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Maret, Plant Manager
P. Corbett, Project Engineering Manager
J. DeVincentis, Asst. to Engineering Director
R. Sojka, Licensing Manager
J. Boothroyd, Fire Protection
D. Maidrand, Systems Engineer
J. Laughney, QA Supervisor
R. Cox, Electrical Design
K. Horelik, Electrical Design
P. Johnson, Electrical Design
R. January, Electrical, Inst. & Control Engineering Manager
L. Doane, Operations

NRC

W. Ruland, Branch Chief, Electrical Engineering Branch, DRS
B. Cook, Senior Resident Inspector, Vermont Yankee

INSPECTION PROCEDURES USED

64704	Fire Protection Program
64150	Triennial Postfire Safe Shutdown Capability Reverification
92903	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

CLOSED

Violation EA 95-268: Multiple Appendix R Program Deficiencies
Violation EA 95-268: Pertaining to the Appendix R fuse exemption deficiencies
URI 95-26-01: Inadequate post-fire shutdown procedure
URI 95-026-02; Spurious operation of RHR system
Miscellaneous: Fire Protection issues identified in LERs (95-14, 96-009, 96-13, and 96-187
URI 94-31-02: Pertaining to Appendix R Emergency Lighting Issue
Violation 93-05-01; Pertaining to the Penetration Seals deficiencies
Violation 93-05-02; Regarding surveillance testing not being performed per the T.S. for plant penetration seals

Miscellaneous Electrical Items:

IFI 97-04-06; Pertaining to 480 Volt Switchgear Bus 9 Loading Concern
URI 96-09-06; Pertaining to Main Battery Service Test Acceptance Criteria Issue

URI 97-02-05; Pertaining to the Advance Offgas System Crossed Neutral Wiring Issue

DISCUSSED AND UPDATED

URI 97-03-02: Pertaining to Electrical Cable Separation Concerns

OPEN

IFI 97-80-01 Penetration Barriers and Seal Design Verification