December 23, 1997

98-016

EA No: 97-532

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

SUBJECT: NRC INSPECTION REPORT 50-271/97-11 AND NOTICE OF VIOLATION

Dear Mr. Reid:

On December 6, 1997, the NRC completed an inspection at your Vermont Yankee reactor facility. The enclosed report presents the results of that inspection.

During the 5-week period covered by this inspection period, your conduct of activities at the Vermont Yankee facility was generally characterized by safety-conscious operations, sound engineering and maintenance practices, and careful radiological work controls.

Based upon the inspectors' observations this period, one violation of regulatory requirements was cited. This violation was the consequence of Vermont Yankee (VY) having not provided effective direct oversight of switching orders initiated by Vermont Electric Power Company and performed by VY auxiliary operators in the owner-controlled 345 KV switchyard. We understand that the automatic reactor scram on November 25, 1997 was precipitated by auxiliary operators manipulating 345 KV disconnects, while unisolated from their power source. Fortuitously, no personnel injuries occurred. However, the event could have been avoided had the switchyard activities been better controlled.

This violation is cited in the enclosed Notice of Violation, and the circumstances surrounding the violation are described in detail in the enclosed report. Please note that you are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Included in this report is the closeout of two Unresolved Items. Unresolved Item 96-08-02 involves the event documented in VY LER 96-015, where the VY staff used industry operating experience to identify the absence of thermal overpressure protection in certain piping systems penetrating primary containment. VY identified and corrected this "old design issue" by a combination of pipe modifications to preclude the conditions creating the containment vulnerability.

I-9

This design issue is an apparent violation of NRC requirements which could 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 provided in Section VII.B.3 of the NRC's Enforcement Policy. This decision was made after consideration that: (1) the violation was identified by your staff's good questioning attitudes during voluntary initiatives; (2) corrective actions were comprehensive and timely; (3) the condition was subtle in nature and not likely to be disclosed through routine surveillance or quality assurance activities; and (4) the violation is not reasonably linked to current performance. The exercise of discretion acknowledges your good effort to identify and correct subtle violations that would not be identified by routine efforts before the degraded safety systems are called upon.

The second unresolved item involved the potential violation of 10 CFR 70.24, Criticality Accident Requirements. We have determined that your facility was in violation of this regulation, in that VY did not have, in place, either a criticality monitoring system for storage and handling of new (non-irradiated) fuel or an NRC approved exemption to the regulation. Numerous other facilities have similar circumstances. The NRC has reconsidered this violation and concluded based on the information discussed in the report that, although a violation did exist, it is appropriate to exercise enforcement discretion for Violations Involving Special Circumstances in accordance with Section VII B.6 of the Enforcement Policy. The bases for exercising this discretion are: the lack of safety significance of the failure to meet 10 CFR 70.24; the failure of the NRC staff to recognize the need for an exemption during the licensing process; the prior NRC position concerning the lack of a need for an exemption; and finally, the NRC's intention to amend 10 CFR 70.24 through rulemaking to provide for administrative controls in lieu of criticality monitors. Therefore, VY's noncompliance with 10 CFR 70.24 was not cited. Pending the amendment to 10 CFR 70.24, further enforcement action will not be taken for failure to meet 10 CFR 70.24 provided you obtain an exemption to this regulation before the next receipt of fresh fuel or before the next planned movement of fresh fuel.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practices," a copy of this letter and the enclosed Notice will be placed in the NRC Public Document Room.

Sincerely,

Original Signed By:

Hubert J. Miller
Regional Administrator

Enclosures:
1. Notice of Violation
2. NRC Inspection Report 50-271/97-11

Docket No. 50-271

cc w/encl:
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G. Sen, Licensing Manager, Vermont Yankee Nuclear Power Corporation
J. Gilroy, Director, Vermont Public Interest Research Group, Inc.
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ENCLOSURE 1

NOTICE OF VIOLATION

Vermont Yankee Nuclear Power Corporation
Vermont Yankee

Docket No. 50-271
License No. DPR-28

During the NRC inspection conducted from October 31 - December 6, 1997, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

Technical Specifications 6.5, "Plant Operating Procedures," states that detailed written procedures, involving both nuclear and non-nuclear safety, shall be prepared, approved, and adhered to covering the areas of preventive and corrective maintenance operations which could have an effect on the safety of the reactor.

Contrary to the above, plant auxiliary operators were implementing an unapproved (by Vermont Yankee) Vermont Electric Power Company (VELCO) switching order on November 25, 1997 to open the 345 KV Scobie Line No. 379-3 disconnects when at 6:48 A.M. an automatic reactor scram resulted from the manipulation of the disconnects while the line was energized per the switching sequence.

This is a Severity Level IV violation (Supplement 1).

Pursuant to the provisions of 10 CFR 2.201, Vermont Yankee Nuclear Power Corporation is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region I, and a copy to the NRC Resident inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without reaction. However, if you find it necessary to include such information, you should clearly indicate the specific information request for withholding the information from the public.

Dated at King of Prussia, Pennsylvania
this 23rd day of December, 1997

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No. Licensee No.	50-271 DPR-28
Report No.	97-11
Licensee:	Vermont Yankee Nuclear Power Corporation
Facility:	Vermont Yankee Nuclear Power Station
Location:	Vernon, Vermont
Dates:	October 31 - December 6, 1997
Inspectors:	William A. Cook, Senior Resident Inspector Edward C. Knutson, Resident Inspector
Approved by:	Curtis J. Cowgill, III, Chief, Projects Branch 5 Division of Reactor Projects

EXECUTIVE SUMMARY

Vermont Yankee Nuclear Power Station
NRC Inspection Report 50-271/97-11

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a five week period of resident inspection.

Operations

Control room operator response to the November 25 reactor scram was good and in accordance with off-normal and emergency operating procedures. A significant contributing cause of the automatic reactor scram was a failure to provide an appropriate level of review and approval of the 345 KV switching orders which was contrary to Technical Specification 6.5 and a cited violation (VIO 97-11-01). Station management and PORC response to this transient and the resolution of plant technical and human performance concerns prior to unit start-up were appropriate. The temporary modification and associated actions to address the generator runback circuit problem were adequately implemented, but operator training on the TM was weak and administrative oversight not comprehensive.

Violation 96-05-01, plant operation impact of high pressure coolant injection (HPCI) system design change, was reviewed. VY's corrective actions appropriately addressed the causes of the violation, as evidenced by no similar occurrences since the event. Accordingly, this violation was closed.

During a regularly scheduled NSAR committee meeting, the inspector observed that the presentations were informative and that many of the NSAR committee members actively engaged the presenters in additional discussion. The inspector determined that the agenda had been revised to focus routinely on the known problem areas of corrective action effectiveness and human performance. Based upon this review and the resident inspector staff's routine inspection and assessment of the off site committee's plant oversight functions, inspection follow item 96-200-06 was closed.

Maintenance

VY's quality control measures for the brazing activities witnessed during inspection 50-271/95-10 were determined to have been in accordance with ANSI B31.1 requirements and established VY welding and QC inspection guidelines. Accordingly, unresolved item 95-10-01 was closed.

Unresolved item 95-14-01, concerning maintenance requirements for the alternate cooling system, was reviewed. Routine preventive maintenance requirements for alternate cooling system and RHRSW piping were found to have been properly implemented, and the unresolved item was closed.

Unresolved item 95-25-04, concerning an administrative error in the test gas mixture for Advanced Offgas (AOG) system hydrogen monitor Technical Specification (TS) surveillance testing was reviewed. The licensee's actions to address this error were appropriate and the previously observed weaknesses in VY's biennial procedural revision process have been adequately addressed. It was concluded that the error did not constitute a violation of NRC requirements. This unresolved item was closed.

The inspector identified an inconsistency in the licensee's documentation pertaining to the expected life of the AOG hydrogen recombiner catalyst. The inspector reviewed the individual action plan items and concluded that VY had implemented reasonable and timely measures to monitor hydrogen recombiner catalyst performance. The inconsistency did not constitute a violation of NRC requirements. The associated unresolved item, URI 96-03-07, was closed.

The inspector reviewed Licensee Event Report (LER) 97-015, concerning an instrument check that was not completed in accordance with Technical Specifications due to personnel error. Review of the event had been previously documented in inspection report no. 50-271/97-06 and dispositioned as a non-cited violation. The LER was determined to have been appropriately written and properly submitted in accordance with 10 CFR 50.73, and was closed.

Engineering

Unresolved item 95-19-01, regarding the adequacy of controls for the conduct of system manipulations for performance monitoring, was reviewed. This issue was the result of an inspector observation of system performance monitoring, during which a procedural limit was found to have been exceeded. Inspector follow-up and day-to-day monitoring of plant activities indicated that the type of intrusive system performance monitoring conducted during inspection period 95-19 was isolated and not a programmatic concern. The failure to have properly operated the RBCCW system per OP-2181 was a violation of minor safety significance and not cited. Accordingly, the unresolved item was closed.

Violation 96-05-04, concerning deficiencies in VY's addressal of a battery room wall seismic qualification issue, was reviewed. The inspector noted that a significant number of design issues have been identified by VY during 1997, and that these issues have been appropriately dealt with within the corrective action program. Accordingly, this violation was closed.

Violation EA 96-210, concerning a single failure vulnerability of the RHR minimum flow valves, was reviewed. Although VY was credited both with identifying the issue and taking prompt, comprehensive corrective action, enforcement action was taken due to the length of time that the condition had existed and the number of prior opportunities that existed to identify and correct the condition. The significant number of design issues that have been identified through VY's design basis documentation program indicates that the licensee has taken action to address this issue. Accordingly, this violation was closed.

The inspectors reviewed LER 96-015, concerning the omission of overpressure relief protection for piping sections that are isolated by the primary containment isolation system. Based upon the VY staff having identified this design issue and having implemented comprehensive and timely corrective actions to resolve this "old design issue," it was not cited, in accordance with the NRC Enforcement Policy, Section VII.B.3. LER 96-015 and associated unresolved item, URI 96-08-02, are closed.

Plant Support

Unresolved item 97-03-07, criticality accident requirements, was reviewed. This issue involved the failure to have in place either a criticality monitoring system for storage and handling of new (non-irradiated) fuel or an NRC approved exemption to this requirement contained in 10 CFR 70.24. Although it was concluded that a violation of 10 CFR 70.24 existed, the NRC determined that, for this issue, it is appropriate to exercise enforcement discretion for violations involving special circumstances in accordance with Section VII B.6 of the Enforcement Policy. Pending the amendment to 10 CFR 70.24, further enforcement action will not be taken for failure to meet 10 CFR 70.24 provided an exemption to this regulation is obtained before the next receipt of fresh fuel or before the next planned movement of fresh fuel.

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Report Details

Summary of Plant Status

At the beginning of the inspection period, Vermont Yankee (VY) was operating at 100 percent power. During the period of November 7-12, plant power was reduced to approximately 60 percent for single rod scram time testing and for localized core power suppression testing to identify the location of a leaking fuel rod that developed on October 26. This testing succeeded in identifying the single leaking fuel bundle. A control rod configuration was developed to suppress power generation in that location and allow the return to full power operation. The plant was returned to 100 percent power on November 16. On November 25, an electrical transient on the main generator due to a maintenance activity in the VY 345 KV switchyard resulted in a reactor scram. The scram was uncomplicated, and plant restart was commenced the following day. The main generator was closed on the grid on November 27 and full power operation was achieved on December 1. With the exception of a power reduction to conduct planned surveillance testing, the plant operated at 100 percent power for the remainder of the inspection period.

I. Operations

O1 Conduct of Operations[1]

O1.1 Reactor Scram Due To Switchyard Electrical Transient

a. Inspection Scope (93702, 71707)

At 6:48 A.M. on November 25, a reactor scram occurred due to an electrical fault that occurred during manipulation of a line disconnect in the 345 KV switchyard at VY. The inspectors reviewed the circumstances that led up to the event, observed operator response to the scram, and monitored VY's actions to identify the cause and prepare for plant restart.

b. Observations and Findings

Background

During power operations, VY is normally connected to the electrical grid by three outgoing power lines. These lines are normally supplied by the main generator through a ring bus in the VY 345 KV switchyard. Circuit breakers (interrupt devices that are designed to be operated under electrical load) in the ring bus allow the individual outgoing lines to be de-energized, if required. Disconnect switches (interrupt devices that are used for maintenance isolation and not designed to be operated under load) on the outgoing lines allow the lines to remain isolated from the ring bus, so that the circuit breakers originally used to isolate a line can be re-closed. This ability is important because the configuration of the ring bus affects

[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.

the amount of load that the main generator is able to transmit to the grid. The alignment of the distribution system is controlled by Vermont Electric Power Company (VELCO) through an offsite dispatcher. Operations department personnel perform switching orders, as directed by VELCO, on that portion of the distribution system that is located at VY.

The Electrical Transient

On the morning of November 25, VELCO was preparing to conduct maintenance on one of the outgoing lines from VY (the Scobie line, circuit 379), which required this line to be de-energized. The standard process for de-energizing an outgoing line was to open the associated circuit breakers in the ring bus (one on either side of the line), and then to open and lock the disconnect switch. This sequence precluded any operation of the disconnect switch when the conductors on either side of the switch were energized. On this occasion, however, VELCO was concerned for ring bus stability, due to the existing grid conditions. To expedite reestablishing the ring bus, the sequence that was used was to open the associated circuit breakers, open the disconnect, and then re-close the circuit breakers prior to locking the disconnect open. As a result, the VY side of the circuit 379 disconnect was energized at the time that operators were directed to lock the disconnect open.

Control over the position of a disconnect switch during maintenance can be maintained either by danger-tagging power to the motor operator in the off position, or by disconnecting the operating mechanism from the motor operator and installing a lock to prevent the mechanism from being rotated. In this instance, VELCO directed that the latter technique be used. Two VY auxiliary operators (AOs) were dispatched to the 345 KV switchyard to perform this operation. After the operating mechanism had been disengaged, the operators found that the holes for attaching the lock (one hole in the rotating mechanism and the other in a fixed portion of the unit) were not aligned closely enough to allow the lock to be installed. One of the operators attempted to rotate the operating mechanism slightly to align the holes; however, a significant amount of force was required to achieve breakaway torque. As a result, when movement did occur, the mechanism rotated further than had been desired. Rotation of the operating mechanism caused the three conducting arms of the disconnect switch (located about 50 feet above the operators) to rotate back towards the closed position. As a result, the conducting arms were sufficiently close to the main line contacts to draw an arc from the energized VY side through to the de-energized circuit 379. The operator noted that this condition existed for several seconds before he was able to rotate the mechanism back to the open position and install the lock.

Plant Response

The resulting electrical transient on the main generator initiated a turbine runback to reduce generator load. The protective feature that actuated was the stator cooling runback circuit, which initiated upon detecting a mismatch between stator water cooling system flow and main generator output current (which was high due to the arcing across the circuit 379 disconnect). The turbine runback is designed to reduce main generator output to approximately 28 percent over a period of two minutes, and should only occur while a mismatch condition exists. In this instance, however, the turbine runback continued to completion, even though the condition that required it existed for only a few seconds. The stator cooling runback circuit is not part of the reactor protection system; its function is equipment protection, and it is classified as a non-nuclear safety system.

The affect of a turbine runback is to cause the ten main steam bypass valves to automatically open, as required, to maintain 100 percent steam flow from the reactor. A secondary affect is that feedwater heating is reduced, because the steam supply for the feedwater heaters is extraction steam from the main turbine; as turbine load is reduced, less steam is available for feedwater heating. The result of lower feedwater temperature, in turn, is to add positive reactivity to the reactor (due to the negative temperature coefficient of reactivity) and therefore, to cause reactor power to increase. To prevent an automatic reactor protective action, operator action is required to compensate for this response by reducing reactor power. In this instance, operators responded appropriately in accordance with operating procedure OT-3119, "Loss of Stator Cooling" and reduced reactor power by reducing reactor recirculation flow. However, reduced recirculation flow also causes the high flux reactor scram setpoint to be reduced (due to flow biasing), and additional power reduction by control rod insertion is required to avoid an automatic scram. As the control room operators attempted to continue the manual reactor power reduction by control rod insertion, an automatic reactor scram occurred at approximately 85 percent reactor power.

Plant response to the scram was uncomplicated and safety systems responded as designed. The inspector observed operator response and determined that it was deliberate and in accordance with appropriate procedures. In working to stabilize plant conditions, the inspector noted that the operators were concerned about lowering main condenser vacuum. Steam to the main condenser air ejectors had been secured earlier in the recovery to reduce the reactor cooldown rate; however, vacuum decreased more rapidly than anticipated and was approaching the point at which the main steam isolation valves (MSIVs) would be required to be shut. Closure of the MSIVs was not desirable because it would eliminate the main condenser's availability as a heat sink, and require use of emergency core cooling systems for residual heat removal. Operators succeeded in arresting the decrease in vacuum, and eventually in restoring normal main condenser vacuum, without having to close the MSIVs. Securing auxiliary steam loads to control plant cooldown rate is allowed by procedure; however, the inspector considered that securing steam to the air ejectors to control a cooldown that was the result of feedwater addition may have been premature, given that a significant amount of steam was still being rejected to the main condenser. This observation was acknowledged by Operations management who stated that it would be examined as part of the post-trip critique.

Management Oversight

The inspectors attended a series of department manager and Plant Operations Review Committee (PORC) meetings conducted following the November 25 plant scram and prior to unit start-up on November 26. The inspectors observed that these meetings were well attended, purposeful, and generally effective. The department manager meetings held shortly after the scram established clear assignments for event/problem follow-up and clear expectations for the plant staff assigned to these tasks. The post-scram/pre-start-up PORC meeting (meeting No. 97-90) held on November 26 was observed by the inspector to have been especially useful in identifying the human performance shortcomings which directly contributed to this event. It was also useful in developing the appropriate recommendations to the plant manager for addressing these shortcomings prior to unit restart.

The inspector noted that the PORC members' review of the preliminary root cause evaluation for this event was extensive. The PORC identified that a significant contributor to the November 25 reactor scram and the April 24, 1997 reactor scram was the failure to provide effective oversight for an activity that had plant impact potential. Specifically, the licensee did not review the tagging order. The order revised the standard practice of maintaining the appropriate portion of the ring bus completely deenergized while disengaging the drive gear. This failure to have reviewed and approved the 345 KV switchyard switching orders was a violation of Technical Specification 6.5 and was cited **(VIO 97-11-01)**.

The inspector's observation of the PORC's examination of the stator cooling runback sequence and generator protection circuit response, again, noted excellent PORC member participation and safety assessment. Troubleshooting and testing of the circuit was inconclusive in determining why the runback signal sealed-in for one minute and 18 seconds. The voltage/current transient resulting from the disconnect arcing lasted only five cycles (5/60th of a second) as recorded by the generator supervisory circuit and plant process computer. Additionally, technical staff examination of the runback circuit identified that the current/flow comparator was a 24 VDC supplied component functioning in a 125 VDC protective circuit. VY engineering staff review of this anomaly concluded that this was not a desirable design, but not necessarily the cause of the circuit seal-in malfunction. Based upon the above technical information and inconclusive troubleshooting, the PORC examined numerous recommendations for resolving this equipment problem. PORC concluded that modifying the generator runback circuit (bypassing the current/flow comparator) via a temporary modification was the most prudent course of action. The inspectors viewed this interim corrective action as appropriate.

Temporary Modification

VY modified the runback circuit via a temporary modification (TM No. 97-26). TM 97-26 removed the comparator input signal and left the remaining stator cooling system high temperature and low system pressure inputs available. Design engineering staff review concluded that the current/flow comparator was installed in the runback circuit in 1979, as an anticipatory loss of stator cooling function which would detect an insufficient stator cooling flow condition before either the high temperature or low pressure conditions developed. Early detection and the resultant generator runback was viewed to lessen the potential for generator damage and potentially lessen the severity of the reactor plant transient.

Inspector review of the implementation of this TM identified that limited communication and training was provided to the operating shift crews on the impact of TM 97-26 on plant response and/or alarm response procedures. The inspectors determined that there was an absence of a clear understanding of the modified generator runback circuit by the operators interviewed. However, inspector discussions with a number of shift operators determined that, in spite of the lack of training on TM 97-26, the control room and local stator cooling alarm response procedures provided adequate procedural guidance to address a degraded or loss of stator cooling condition. The inspector identified that control room annunciator 7-B-5, "Stator Cooling Runback" was "orange dotted" in accordance with Operations Procedure (OP)-3140, "Alarm Response," to reflect the modified runback circuit. However, the local stator cooling panel annunciator SC-A-3, "turbine runback," was not similarly "orange dotted." This administrative oversight was promptly corrected by the on shift supervisor when identified by the inspector. The inspector noted that training was supplemented after his observations.

c. Conclusions

Overall, control room operator response to the November 25 reactor scram was good and in accordance with off-normal and emergency operating procedures. A significant contributing cause of the automatic reactor scram was a failure to provide an appropriate level of review and approval of the 345 KV switching orders which was contrary to Technical Specification 6.5 and a cited violation (**VIO 97-11-01**). Station management and PORC response to this transient and the resolution of plant technical and human performance concerns prior to unit start-up were appropriate. The temporary modification and associated actions to address the generator runback circuit problem were adequately implemented, but operator training on the TM was weak and administrative oversight not comprehensive.

O8 Miscellaneous Operations Issues

O8.1 (Closed) Violation 96-05-01: Plant operation impact of high pressure coolant injection (HPCI) system design change (92901)

On December 7, 1995, replacement of the HPCI turbine trip pushbutton per the Installation and Test (I&T) procedure for Engineering Design Change 95-408 caused an unanticipated automatic system realignment. The cause of the realignment was that the electrical isolation that was specified in the I&T procedure for replacement of the pushbutton (CRP9-39 fuses 23A-F1 and -F2 removed) also caused the HPCI suction transfer logic to initiate, thus causing the HPCI suction to realign from the condensate storage tank to the torus. Subsequent investigation revealed that the primary containment isolation system (PCIS) logic for HPCI had also been degraded as a result of the fuse removals. This event was discussed in detail in inspection report 50-271/95-25 and was cited as a violation of 10 CFR 50, Appendix B, Criterion XIV, "Inspection, Tests, and Operating Status," in inspection report 50-271/96-05.

The inspector reviewed licensee event report (LER) 96-005, "Failure to identify and take Technical Specification required actions during design change installation activities due to human error," dated March 28, 1996, which discussed this event. The cause of the event was determined to be human errors in preparation of the installation procedure and the design change. The governing procedures were considered to have contained sufficient guidance to have avoided the problem, and corrective actions focused on promulgating lessons learned from the event. In addition, members from appropriate departments were designated to participate in advanced preparation and review for future similar on-line maintenance and design installation activities.

In their reply to the notice of violation, dated July 19, 1996, VY indicated that, "additional review and assessment of the event determined that a contributing factor was that the expectations for Operations Department review of I&T procedures was not well understood (e.g. expectation that a thorough review of proposed tagouts be performed consistent with the level of review performed under the switching and tagging process)." The reply indicated that procedure AP-4001, "Installation, Test and Special Test Procedures," would be revised to incorporate these expectations. The inspector reviewed AP-6001, revision 20, dated September 26, 1997. This revision incorporated recommended changes that resulted from review of the I&T processes and procedures used by six other utilities. Enhancements included increased interdepartmental participation during walkdowns and procedure generation, and improved coordination with operations during equipment tagout and alignment evolutions.

The inspector concluded that VY's corrective actions appropriately addressed the causes of the violation, as evidenced by no similar occurrences since the event. Accordingly, violation **VIO 96-05-01** is closed.

O8.2 <u>(Closed) Inspection Follow Item 96-200-06: Follow-up on Nuclear Safety and Audit Review (NSAR) Committee plans to resolve known problem areas</u> (92901)

The inspector attended portions of the regularly scheduled NSAR committee meeting of November 13 and 14, 1997. The inspector observed presentations and committee member review of selected Quality Assurance (QA) audits, a Systems Engineering Program presentation, a status report on the service water system, a Human Performance Improvement Program report, and a presentation on the site Corrective Action Program which included a discussion of recent performance trends. The inspector observed that the presentations were informative and that many of the NSAR committee members engaged the presenters in further explanation of the prepared information and challenged the bases for the presenters' assessments. A few new committee follow-up items were assigned and previously opened NSAR committee items were assessed for closure.

The inspector determined from a follow-up discussion with the NSAR committee chairman that the agenda had been revised to focus routinely on the known problem areas of corrective action effectiveness and human performance. In addition to the formal agenda topic presentations, the inspector learned that various site-generated monthly performance trend reports were now routinely distributed to the committee members. The inspector also noted that NSAR committee-sponsored QA audits have recently placed more focus in the areas of procedural quality and adherence, at the request of the committee.

Based upon this review and the resident inspector staff's routine inspection and assessment of the off site committee's plant oversight functions, inspection follow item **IFI 96-200-06** is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Observations

a. Inspection Scope (62707)

The inspectors observed portions of plant maintenance activities to verify that the correct parts and tools were utilized, the applicable industry code and technical specification 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 operability was verified upon completion of post maintenance testing.

b. Observations, Findings, and Conclusions

The inspector observed portions of the following maintenance activities:

- Reactor power suppression testing to identify the location of a leaking fuel rod, observed November 7-9

  A small leak developed from a fuel rod on October 26. as indicated by a step increase on the steam jet air ejector radiation monitor. Grab sample results indicated that the activity discharge rate at the air ejector had increased from the pre-event level of 600 microcuries per second (μCi/sec) to 2600 μCi/sec. A reactor coolant sample showed that iodine concentration had also increased from 1.5E-4 μCi/ml (dose equivalent I-131) to 3.1E-4 μCi/ml. The suppression testing succeeded in identifying the fuel bundle that contains the leaking fuel rod.

The inspector observed that this activity was well controlled. Although the activity required several days of reduced power operation, the inspector observed no evidence of time pressure on the operators. VY's decision to complete core-wide testing after the leaking fuel rod was identified (thereby verifying that there were no other leaking rods) also indicated management's support for obtaining as much information as possible about the condition prior to resuming full power operation.

- Scram solenoid pilot valve (SSPV) replacement for hydraulic control unit 06-19, observed November 10

  The SSPV was replaced after it was determined to be buzzing; this condition has been demonstrated to be an indicator of incipient failure, as discussed in inspection report 50-271/97-04. The inspector observed no problems during the replacement and post-maintenance testing.

- "A" service water subsystem maintenance, observed November 10-12

  This limiting condition for operation (LCO) maintenance period included preventive maintenance on the "C" SW pump and motor. No problems were noted, and activities were completed well within the 15-day allowed outage time.

M1.2 Surveillance Observations

a. Inspection Scope (61726)

The inspector observed portions of surveillance tests to verify proper calibration of test instrumentation, use of approved procedures, performance of work by qualified personnel, conformance to LCOs, and correct post-test system restoration.

b. Observations, Findings, and Conclusions

The inspector observed portions of the following surveillance test:

- High Pressure Coolant Injection System Quarterly Surveillance, observed December 3, 1997

The inspector observed that the test was well controlled. No problems were noted.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) Unresolved Item 95-10-01: Quality Control inspection and verification of in-process brazing

a. Background and Inspection Scope (92902)

During an earlier inspection, the inspector witnessed brazing activities associated with the replacement of the reactor recirculation units (RRUs) 7 and 8 heat exchangers. The inspector observed no independent Quality Control (QC) inspector oversight of those activities, other than the QC verifications of consumable materials issued from the warehouse. As a result, the inspector questioned the adequacy of quality assurance measures for this activity, even though no specific brazing problems were observed by the inspector. The licensee acknowledged the inspector's observation and conducted a review of their brazing control processes to ensure compliance with established practices and industry standards. The inspector reviewed the licensee's response to this item and verified the minimum regulatory requirements were being satisfied.

b. Observations and Findings

The inspector reviewed VY's internal response to this issue documented by memorandum dated April 5, 1996. The memo confirmed VY's initial response to the inspector that the Quality Assurance (QA) Program had been adequately adhered to, for the brazing activities observed. The inspector independently verified, by a records review, that the brazer who performed the work was qualified to ASME Section IX standards, as required by ANSI B31.1, Section 128.

No in-process QC inspections were required by ANSI B31.1, nor were in-process inspections stipulated in the work package by the responsible weld engineer. Consequently, the inspector determined that appropriate QC measures were invoked for this brazing activity and that the qualified brazer was principally responsible per code for ensuring quality in this special process. The inspector noted that VY did clarify their contractor Quality Control Inspection Guidelines, (VY/FPS-70010) as a result of the inspector's observations.

c. Conclusions

The licensee's quality control measures for the brazing activities witnessed during inspection 95-10 were in accordance with ANSI B31.1 requirements and established VY welding and QC inspection guidelines. This unresolved item (URI 95-10-01) is closed.

M8.2 (Closed) Unresolved Item 95-14-01: Verify maintenance requirements for alternate cooling system (92902)

This unresolved item was initiated to provide a tracking item to verify the licensee's implementation of routine preventive maintenance (PM) requirements for the alternate cooling system. As previously documented in inspection report 50-271/95-14, section 2.2.2, the NRC staff concluded that VY's 10-year test interval to demonstrate satisfactory hydraulic performance per Generic Letter (GL) 89-13, Action II, was acceptable. However, the alternate cooling system and residual heat removal service water (RHRSW) piping PM requirements had not yet been formalized per GL 89-13, Action III.

With the assistance of the service water systems engineer, the inspector reviewed the Maintenance Planning and Control (MPAC) computer data base, (VY's formal maintenance department record system), to verify that alternate cooling system and RHRSW piping PM requirements were entered and being implemented. The inspector examined a sampling of the most recently completed Work Orders (WOs) for the associated PM tasks and verified satisfactory implementation of the selected PM activities. For example, the inspector confirmed that the safety related portion of the alternate cooling system (west cooing tower, CT-2) was receiving routine (repetitive) PM in accordance with VY's PM Basis No. M307, revision 3. WO No. 97-06891, completed May 30, 1997, documented the results of the last CT-2 structural inspection per Vermont Yankee Equipment Manual (VYEM) No. 146. Deficient conditions were appropriately resolved via separate WOs, as annotated in the MPAC records. The inspector also examined a sampling of the completed PM records for the RHRSW piping and verified that the piping inspections had been performed in accordance with OP-5265, "Service Water component inspection and acceptance criteria." This unresolved item (URI 95-14-01) is closed.

M8.3 <u>(Closed) Unresolved Item 95-25-04: Advanced Offgas (AOG) system hydrogen monitor Technical Specification (TS) surveillance requirement identifies an incorrect test gas mixture</u> (92902)

During an earlier inspection period, the licensee identified that AOG hydrogen monitor surveillance requirements specified in TS (TS Table 4.9.2, note 4) were incorrect. As documented in inspection report 50-271/95-25, prompt corrective action was taken to ensure proper testing of the hydrogen monitors and an amendment to TS Table 4.9.2 was initiated. By letter dated February 5, 1996, VY submitted proposed change No. 180, Administrative Change to Vermont Yankee Technical Specification to correct typographical errors and text inconsistencies. The NRC staff's review of this February 5, 1996 submittal was ongoing, as of the close of this inspection period. The VY staff has taken reasonable and timely action to resolve the TS administrative error.

Included in the inspector's original review of this issue was a conclusion that the biennial procedure review process had not been sufficiently thorough to have identified this TS error earlier (the hydrogen monitor TS was revised in the early 1980's). To address this observation, the licensee revised their administrative control procedure (AP)-0037, "Plant Procedure," to significantly enhance the biennial procedure review criteria. The revised procedure ensured a "comprehensive review" of the entire procedure, with a verification of all procedure steps against the FSAR, Technical Specification, Quality Assurance Program (YOQAP-1-A), and applicable program requirements (IST, Fire Protection, EQ, E-Plan, etc.). The inspectors have observed, over the past several months, improved biennial review of procedures. This has been demonstrated by the generation of Event Reports for significant deficiencies identified via the biennial review, and the sometime extensive summary-of-changes memorandums addressed to the Plant Operations Review Committee, which are attached to the current revision of a plant procedure.

The licensee's actions to address the administrative error in the TS were appropriate and the previously observed weaknesses in the VY procedures biennial revision process have been adequately addressed. The inspector concluded that no violations of regulatory requirements occurred. This unresolved item **(URI 95-25-04)** is closed.

M8.4 <u>(Closed) Unresolved Item 96-03-07: Resolution of inconsistent recombiner catalyst lifetime in the AOG system</u> (92902)

During an inspection of the advanced offgas (AOG) system, the inspector identified an inconsistency in the licensee's documentation pertaining to the expected life of the AOG hydrogen recombiner catalyst. In response, the VY staff initiated an action plan via their commitment tracking system (CTS) to address this issue.

By engineering staff memorandum VYI-59/96 Revision 1, dated August 28, 1996, the licensee formulated an action plan to verify the current performance characterization of the hydrogen recombiner catalysts and to routinely monitor their performance to better project their replacement, if necessary, prior to the end of the facility's lifetime (same as the estimated catalyst's lifetime). The inspector reviewed the individual action plan items (CTS items INS 960307-01 through 06) and verified that recombiner temperature monitoring has been implemented in accordance with AP-0150, "Conduct of Operations and Operator Rounds," Revision 31. Based upon the above, the inspector concluded that VY had implemented reasonable and timely measures to monitor hydrogen recombiner catalyst performance and that no violation occurred. This unresolved item **(URI 96-03-07)** is closed.

M8.5 <u>(Closed) LER 97-015: Instrument check not completed in accordance with Technical Specifications due to personnel error</u> (92700)

The event discussed in LER 97-015, dated September 24, 1997 was previously reviewed by the inspectors, as documented in inspection report 50-271/97-06, section M1.3. As documented in report 97-06, this violation was not cited. LER 97-015 was reviewed using the guidance of Inspection Procedure 92700 and determined to have been appropriately written and properly submitted in accordance with 10 CFR 50.73. **LER 97-015** is closed.

M8.6 <u>(Closed) Unresolved Item 96-03-04: Torus water temperature administrative limit reduced</u>

This unresolved item was examined by the NRC staff in inspection report 50-271/97-201, section E1.1.2.2. As documented in report 97-201, the inspectors examined plant operation with this concern and the timeliness of the licensee's corrective actions for this design issue. Consequently, two separate unresolved items were generated (URI 97-201-01 and URI 97-201-02). Accordingly, URI 96-03-04 is administratively closed, and the two new unresolved items will track the individual issue's resolution and NRC staff disposition of the same.

III. Engineering

E8 Miscellaneous Engineering Issues

E8.1 (Closed) NCV 97-11-02 and Unresolved Item 95-19-01: Adequacy of controls for the conduct of system manipulations for performance monitoring (92903)

This item was initiated based upon the inspector observing an instance where the reactor building closed cooling water (RBCCW) heat exchanger outlet valve was throttled, per the operating procedure (OP-2181) for the purpose of collecting data on heat exchanger thermal performance and flow-induced vibration. During the review of this activity, the inspector observed that both heat exchanger service water inlet-to-outlet differential pressure readings exceeded the operating procedure maximum limit of 10 psid. Neither of the individuals involved with the data collection were aware of this condition nor took action to address it prior to the inspector's intervention. The inspector was concerned that the RBCCW system may not have been appropriately operated per procedure and that the controls for system performance monitoring may not be adequate.

Follow-up inspection of this activity identified that VY management developed similar concerns. The auxiliary operator log sheets were revised to include the acceptable RBCCW service water flow differential pressure (d/p) band and all heat exchanger d/p limits and operating parameters were verified. The procedure biennial review process was also modified to include a verification of all system design and limiting parameters. In addition, an assessment of this event and the lessons learned was included in the Operations Department "Read and Sign" notebook (log #96-34) for mandatory Operations Department staff review. The "Read and Sign" re-enforced VY management expectations with respect to procedural adherence and expected supervisory involvement when operating systems for data collection.

Inspector follow-up and day-to-day monitoring of plant activities indicated that the type of intrusive system performance monitoring conducted during inspection period 95-19 was isolated and not a programmatic concern. There has been no observed recurrence of this type of on-line system performance monitoring. The failure to have properly operated the RBCCW system in accordance with OP-2181 constituted a violation of minor safety significance and was treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy. NCV 97-11-02 and unresolved item URI 95-19-01 are closed.

E8.2 <u>(Closed) Violation 96-05-04: Battery room block wall seismic qualification methodology acceptability and operability determination impact</u> (92903)

On March 5, 1996, the VY engineering staff identified that the block walls that separate and support the station batteries were not seismically qualified per the current design basis requirements. This condition was not entered into the corrective action process until March 12, 1996, which constituted an inappropriate delay in the plant staff's operability assessment of the station batteries. Additionally, VY's initial projection that the condition would not be corrected until the first quarter of 1997 represented acceptance of an excessive delay in implementing corrective action. This condition was discussed in detail in inspection report 50-271/96-03, and was cited as a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," in inspection report 50-271/96-05.

The inspector reviewed LER 96-008, "Error identified in block wall calculation due to personnel error which places plant outside of design basis." The cause of the event and corrective actions discussed in that document relate to the deficient material condition rather than the timeliness of corrective action. The inspector reviewed VY's reply to the notice of violation, which identified the cause as being personnel error on the part of engineering management, in that reporting the condition to plant management was delayed pending completion of an engineering analysis. Corrective actions consisted of discussions and training to emphasize the need for timeliness of operability determinations with respect to issue revelation. The reply also discussed appointment of a new engineering management team and reorganization of the VY engineering organization, and pointed out their progress in lowering the threshold of problem discovery.

The inspector noted that a significant number of design issues have been identified by VY engineering during the year 1997. In general, these issues have been expeditiously entered into the corrective action program and immediate operability determinations have been promptly performed. Where appropriate, continued operation with degraded or nonconforming conditions has been justified through the licensee's Basis for Maintaining Operation (BMO) process. The inspector assessed that the quality of operability determinations resulting from this process was good, and that the timeliness of proposed corrective actions was reasonable. Accordingly, violation **VIO 96-05-04** is closed.

E8.3 <u>(Closed) LER 96-001, Supplement 2: Technical Specification 4.6.E not met due to components not included in the Inservice Test Program scope</u> (92700)

The Inservice Testing Program issues documented in LER 96-001, Supplement 2, dated January 16, 1997, were previously examined as documented in inspection reports 50-271/96-03, section 3.3.1, and 50-271/96-11, section E8.2. Supplement 2 updated the corrective actions implemented by the VY staff. The inspector reviewed LER 96-001, Supplement 2 using the guidance of Inspection Procedure 92700 and found the licensee's updated response to this event to have been appropriate. **LER 96-001, Supplement 2** is closed.

E8.4 (Closed) Violation EA 96-210: RHR Minimum Flow Valve Single Failure Vulnerability (92903)

This issue was examined by the NRC staff in inspection report 50-271/96-07 and reported by the licensee in LER 96-010. As discussed in the notice of violation, dated August 23, 1996, VY was credited both with identifying the issue and taking prompt, comprehensive corrective action. In this case, however, enforcement action was taken due to the length of time that the condition had existed and the number of prior opportunities that existed to identify and correct the condition. The significant number of design issues that have been identified through VY's design basis documentation program indicates that the licensee has taken action to address this issue. Accordingly, violation EA 96-210 is closed.

E8.5 (Closed) NCV 97-11-03 and LER 96-15: "Original B31.1 ANSI Code Section that required overpressure relief for isolated piping was not considered during original design.

a. Inspection Scope (92700)

The inspectors reviewed LER 96-015, to assess the adequacy of the VY staff's actions with respect to identification, reporting, evaluation, and resolution of that issue. This review included the examination of the associated Event Report (ER), immediate operability determination, Basis for Maintaining Operation (BMO), and the related safety evaluation. The inspector verified proper implementation of immediate corrective actions, verified and discussed the adequacy of interim compensatory measures with responsible engineering and operations staff, and assessed the overall adequacy and timeliness of proposed actions to ensure the proper resolution and/or a comprehensive programmatic review of the design issue.

b. Observations and Findings

This issue involved the licensee's determination that the original piping construction code (ANSI B31.1, 1967) stipulated that portions of piping that could exceed design pressure limits, if isolated, should be protected from thermal over-pressurization by a pressure relieving device. Based upon industry operating experience review, the VY staff determined that a number of piping systems penetrating containment were susceptible. This issue was previously reviewed in inspection report 96-08 and an unresolved item (URI 96-08-02) was assigned for further NRC staff review and follow-up.

As stated in LER 96-015, based upon the event being an original design and construction issue, the root cause determination was not conclusive. However, it appeared that personnel involved with initial plant design and construction failed to apply Section 101.4.2 of ANSI B31.1-1967 to those plant systems penetrating containment which could be isolated in the event of a LOCA and resultant primary containment isolation system initiation. As previously documented in inspection report 50-271/96-08, the affected systems included portions of reactor building closed cooling water (RBCCW) piping, residual heat removal (RHR) and shutdown cooling piping, main steam line drains piping, liquid radwaste piping, and recirculation system sample piping.

By letter dated January 28, 1997, VY responded to Generic Letter (GL) 96-06, "Assurance of equipment operability and containment integrity during design basis accident conditions," dated September 30, 1996. As stated in VY's September 30 letter, to address the potential thermal overpressurization of isolated piping penetrating containment, VY implemented engineering design changes during the Fall 1996 refueling outage. Relief valves were installed in the RBCCW and radwaste system piping, and check valves were installed in the main steam line drains, RHR shutdown cooling, and recirculation sample piping.

The inspectors found that the above licensee identified "old design issue" (reference Enforcement Guidance Memorandum 96-005, dated October 21, 1996) was appropriately dispositioned in a timely fashion within the guidance of Generic Letter 91-18 and 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." The inspectors observed that the licensee's reporting of these issues was consistent with 10 CFR 50.72 and 50.73 requirements. The inspector observed that VY promptly investigated this issue and developed corrective action in advance of the NRC staff's issuance of GL 96-06.

c. Conclusions

The extended period without operability of the containment system is a violation of 10 CFR Appendix B, Criteria III, Design Control. The violation was identified by the licensee as a result of a voluntary initiative, corrective actions were prompt and comprehensive, the violation was not likely to be identified by routine licensee efforts such as normal surveillance or quality assurance activities and the violation is not reasonably linked to current performance. As a result, this apparent violation of NRC requirements will not be cited in accordance with Section VII.B.3 of the NRC Enforcement Policy. **NCV 50-271/97-11-03, LER 96-015** and **URI 96-08-02** are closed.

E8.6 (Closed) LER 50-271/96-028: Inadequate field labeling of safety class wiring and drawing updates result in the failure to maintain electrical separation requirements described in nuclear safety system design bases. (92700)

This LER describes a failure to meet cable separation criteria resulting from an improper labeling of wiring during a 1976 modification. While the original modification error only involved mislabeling of the wiring, subsequent modifications to the mislabeled wiring resulted in a loss of physical separation between safety divisions. The NRC identified electrical separation concerns in NRC inspection report 50-271/97-03 and opened an unresolved item (URI 97-03-02) at that time. Additional similar concerns were raised by the NRC AE team as described in inspection report 50-271/97-201. The NRC continues to follow the licensee's activities regarding electrical separation as part of the resolution of the previously identified unresolved item. Therefore, this LER is administratively closed and all further NRC response will be tracked via URI 97-03-02.

IV. Plant Support

P8 Miscellaneous EP Issues

P8.1 (Closed) NCV 97-11-04 and Unresolved Item 97-03-07: Criticality Accident Requirements (92904)

This issue involved the failure to have in place either a criticality monitoring system for storage and handling of new (non-irradiated) fuel or an NRC approved exemption to this requirement contained in 10 CFR 70.24.

10 CFR 70.24 requires that each licensee authorized to possess more than a small amount of special nuclear material (SNM) maintain in each area in which such material is handled, used, or stored a criticality monitoring system which will energize clearly audible alarm signals if accidental criticality occurs. The purpose of 10 CFR 70.24 is to ensure that, if a criticality were to occur during the handling of SNM, personnel would be alerted to that fact and would take appropriate action.

Most nuclear power plant licensees were granted exemptions from 10 CFR 70.24 during the construction of their plants as part of the Part 70 license issued to permit the receipt of the initial core. Generally, these exemptions were not explicitly renewed when the Part 50 operating license was issued, which contained the combined Part 50 and Part 70 authority. In August 1981, the Tennessee Valley Authority (TVA), in the course of reviewing the operating licenses for its Browns Ferry facilities, noted that the exemption to 10 CFR 70.24 that had been granted during the construction phase had not been explicitly granted in the operating license. By letters dated August 11, 1981, and August 31, 1987, TVA requested an exemption from 10 CFR 70.24. On May 11, 1988, NRC informed TVA that "the previously issued exemptions are still in effect even though the specific provisions of the Part 70 licenses were not incorporated into the Part 50 license." Notwithstanding the correspondence with TVA, the NRC has determined that, in cases where a licensee received the exemption as part of the Part 70 license issued during the construction phase, both the Part 70 and Part 50 licenses should be examined to determine the status of the exemption. The NRC view now is that unless a licensee's licensing basis specifies otherwise, an exemption expires with the expiration of the Part 70 license. The NRC intends to amend 10 CFR 70.24 to provide for administrative controls in lieu of criticality monitors.

The NRC has concluded that a violation of 10 CFR 70.24 existed. The NRC has also determined that numerous other licensees have similar circumstances that were caused by confusion regarding the continuation of an exemption to 10 CFR 70.24 originally issued prior to issuance of the Part 50 license. After considering all the factors that resulted in these violations, the NRC has concluded that while a violation did exist, it is appropriate to exercise enforcement discretion for Violations Involving Special Circumstances in accordance with Section VII B.6 of the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. Pending the amendment to 10 CFR 70.24, further enforcement action will not be taken for failure to meet 10 CFR 70.24, provided an exemption to this regulation is obtained before the next receipt of fresh fuel or before the next planned movement of fresh fuel. **NCV 97-11-04** and unresolved item **URI 97-03-07** are closed.

F8 Miscellaneous Fire Protection Issues

F8.1 <u>(Closed) Unresolved Item 95-21-02: Appendix R and Fire Protection Program implementation issues</u> (92904)

This unresolved item broadly addressed the large number of related Appendix R and Fire Protection (FP) Program issues which were identified by the VY staff in late 1995. As a result of several of these identified discrepancies, compensatory fire watches were established in accordance with TS-pending resolution of the technical or programmatic issue. The inspector observed that initially, a number of minor fire watch performance lapses occurred as a result of either poor understanding of the compensatory measures or miscommunication of the fire watch requirements. Significant VY management attention was focused on these performance issues and the broader Appendix R and FP Program technical issues, resulting in sweeping changes and noteworthy improvements in the program and overall VY staff performance.

During an earlier inspection this year (reference inspection report 50-271/97-80, dated October 16, 1997) the NRC concluded that VY's overall response to the cited violations and additional licensee identified and corrected problems were appropriate. As a consequence, enforcement discretion was exercised for the related Appendix R and FP Program violations identified via the licensee's corrective action processes. Accordingly, this unresolved item is administratively closed, as it identified no additional technical or performance issues that had not already been addressed by the VY staff. Unresolved item **URI 95-21-02** is closed.

V. Management Meetings

X1 Exit Meeting Summary

The resident inspectors met with licensee representatives periodically throughout the inspection and following the conclusion of the inspection on December __, 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.

X3 Review of Updated Final Safety Analysis Report (UFSAR)

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or 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 area inspected. The inspectors verified that the UFSAR wording was consistent with the observed practices and procedures and/or parameters.

INSPECTION PROCEDURES USED

61726	Surveillance Observations
62707	Maintenance Observations
71707	Plant Operations
92700	Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities
92901	Followup - Plant Operations
92902	Followup - Maintenance
92903	Followup - Engineering
92904	Followup - Plant Support
93702	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

OPEN

VIO 97-11-01	Failure to review and approve switchyard switching orders resulted in reactor scram

CLOSED

VIO 96-05-01	Plant operation impact of high pressure coolant injection (HPCI) system design change
IFI 96-200-06	Follow-up on Nuclear Safety and Audit Review (NSAR) Committee plans to resolve known problem areas
URI 95-10-01	Quality Control inspection and verification of in-process brazing
URI 95-14-01	Verify maintenance requirements for alternate cooling system
URI 95-25-04	Advanced Offgas (AOG) system hydrogen monitor TS surveillance requirement identifies an incorrect test gas mixture
URI 96-03-07	Resolution of inconsistent recombiner catalyst lifetime in the AOG system
LER 97-015	Instrument check not completed in accordance with Technical Specifications due to personnel error
URI 96-03-04	Torus water temperature administrative limit reduced
URI 95-19-01 & NCV 97-11-02	Adequacy of controls for the conduct of system manipulations for performance monitoring
VIO 96-05-04	Battery room block wall seismic qualification methodology acceptability and operability determination impact

LER 96-001, Supp2	Technical Specification 4.6.E not met due to components not included in the Inservice Test Program scope
VIO EA 96-210	RHR minimum flow valve single failure vulnerability
LER 96-015 & NCV 97-11-03	Original B31.1 ANSI Code Section that required overpressure relief for isolated piping sections was not considered during original design
URI 96-08-02	Design deficiency in containment piping penetrations
URI 97-03-07 NCV 97-11-04	Criticality accident requirements
URI 95-21-02	Appendix R and fire protection program implementation issues
LER96-028	Inadequate field labeling of safety class wiring and drawing updates result in the failure to maintain electrical seperation criteria described in the design bases.

PARTIAL LIST OF PERSONS CONTACTED

G. Maret, Plant Manager
F. Helin, Tech. Services Superintendent
E. Lindamood, Director of Engineering
K. Bronson, Operations Manager
M. Watson, Maintenance Superintendent
M. Desilets, Radiation Protection Manager
R. Gerdus, Chemistry Manager
G. Morgan, Security Manager

LIST OF ACRONYMS USED

BMO	Basis for Maintaining Operation
CFR	Code of Federal Regulation
CR	control room
CS	core spray
EDG	emergency diesel generator
ER	Event Report
GL	Generic Letter
HPCI	high pressure coolant injection
IFI	Inspector follow item
IN	Information Notice
KV	kilovolt
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LPCI	low pressure coolant injection
MCC	motor control center
NRC	Nuclear Regulatory Commission
NNS	Non-nuclear safety
PORC	Plant Operations Review Committee
QA	Quality Assurance
RHR	residual heat removal
RP	radiation protection
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	unresolved item
VAC	volts alternating current
VDC	volts direct current
VY	Vermont Yankee