July 30, 2003

Mr. Jay K. Thayer
Site Vice President - Vermont Yankee
Entergy Nuclear Vermont Yankee, LLC
P.O. Box 0500
185 Old Ferry Road
Brattleboro, Vermont 05302-0500

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000271/2003005

Dear Mr. Thayer:

On June 28, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Vermont Yankee Nuclear Power Station. The enclosed report documents the inspection findings which were discussed on July 17, 2003, with Mr. K. Bronson and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two NRC-identified findings of very low safety significance (Green) that were also determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective actions program, the NRC is treating these two findings as non-cited violations (NCVs), consistent with Section VI.A of the NRC's Enforcement Policy. If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Vermont Yankee Nuclear Power Station.

Since the terrorist attacks on September 11, 2001, NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year (CY) '02, and the remaining inspection activities for Vermont Yankee are scheduled for completion in CY '03. The NRC will continue to monitor overall safeguards and security controls at Vermont Yankee.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Clifford J. Anderson, Chief
Projects Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 05000271/2003005
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-271
Licensee No.	DPR-28
Report No.	05000271/2003005
Licensee:	Entergy Nuclear Vermont Yankee, LLC
Facility:	Vermont Yankee Nuclear Power Station
Location:	320 Governor Hunt Road Vernon, Vermont 05354-9766
Dates:	March 30, 2003 - June 28, 2003
Inspectors:	David L. Pelton, Senior Resident Inspector Beth Sienel, Resident Inspector Jennifer Bobiak, Reactor Engineer Joseph T. Furia, Senior Health Physicist Roy Fuhrmeister, Senior Reactor Inspector Robert Berryman, Reactor Inspector Timothy O'Hara, Reactor Inspector Gregory C. Smith, Senior Physical Security Inspector
Approved by:	Clifford J. Anderson, Chief Projects Branch 5 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000271/2003-005; 03/30/03-06/28/03; Vermont Yankee Nuclear Power Station; Fire Protection and Maintenance Effectiveness.

This report covered a 13-week period of baseline inspection conducted by resident inspectors. Additionally, announced inspections were performed by regional inspectors in the areas of radiation protection; permanent plant modifications; and evaluations of changes, tests, and experiments. Two Green non-cited violations (NCVs) were identified. The significance of findings is indicated by their color (Green, White, Yellow, Red) using inspection manual chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” for the failure to take effective corrective actions to address cable separation deficiencies in the cable vault.

  This finding is considered to be greater than minor because it affected the Mitigating Systems Cornerstone objective of equipment availability. Specifically, cable separation deficiencies continue to be identified by NRC inspectors in the safety-related cable vault despite corrective actions taken by the licensee to address previous NRC-identified cable vault cable separation issues. The finding was determined to be of very low safety significance because no actual loss of safety function was identified. (Section 1R05)

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” for the failure to take effective corrective actions to address continued lift setpoint testing failures of standby liquid control (SLC) system relief valves.

  This finding is considered to be greater than minor because the on-going history of SLC system relief valve testing failures affected the Mitigating Systems Cornerstone objective of equipment reliability. The finding was determined to be of very low safety significance since relief valve failures would not have resulted in a loss of SLC system safety function. (Section 1R12)

B. Licensee Identified Findings

None.

REPORT DETAILS

Summary of Plant Status

Vermont Yankee Nuclear Power Station entered the inspection period operating at or near full power and with the exception of minor power reductions for control rod pattern adjustments, continued at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

.1 Readiness for Seasonal Susceptibilities

a. Inspection Scope

The inspectors reviewed measures established by the licensee for restoring from cold weather operations. The inspectors reviewed Vermont Yankee Operating Procedure (OP) 2196, “Preparations for Cold Weather Operations,” Form VYOPF 2196.02, “Cold Weather Restoration Operations Checklist,” to ensure all required actions to restore from cold weather operations had been identified and completed. Additionally, the inspectors reviewed event reports (ERs) related to cold weather or warm weather operations to verify that there had been no issues identified that could affect the plant’s ability to operate safely under warm weather conditions.

b. Findings

No findings of significance were identified.

.2 Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

On May 2, 2003, the inspectors reviewed actions taken by the licensee due to the presence of severe thunderstorms in the vicinity of the plant. During this storm, a nearby lightning strike caused a drywell high range radiation monitor (RM) 16-19-1B and plant process radiation monitor (PRM) 17-359 to alarm. The inspectors observed operator actions taken as a result of the alarm conditions and reviewed annunciator response sheets (ARS) to ensure all required actions were taken. Also, several plant process computer points were temporarily lost including drywell temperature indication, discharge gate position, and safety relief valve baseline temperature indication. The inspectors reviewed OP 3127, “Natural Phenomena,” Appendix B, “Lightning Damage Indicator Walkdown Check List,” and observed operators perform the required panel walkdowns. The inspectors performed independent walkdowns of safety and risk significant control room panels and reviewed the plant process computer to determine the extent of the loss and to ensure no loss of safety function occurred as a result of the lightning strike.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed samples of safety evaluations to verify that changes and tests were reviewed and documented in accordance with 10CFR50.59, “Changes, Tests, and Experiments,” and when required, NRC approval was obtained prior to implementation. The sample included a review of safety evaluations performed to support Design Change Package (DCP) changes. The inspectors assessed the adequacy of the safety evaluations through interviews with the cognizant plant staff and review of supporting information, such as calculations, engineering analyses, design change documentation, the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS) and plant drawings. In addition, the inspectors reviewed the administrative procedures that control the screening, preparation, and issuance of the safety evaluations to ensure that the procedures adequately implemented the requirements of 10CFR50.59.

The inspectors reviewed a sample of changes that the licensee had evaluated using their 10CFR50.59 screening process. The inspectors performed this review to determine if the licensee’s conclusions with respect to 10CFR50.59 applicability were appropriate.

The inspectors reviewed issues that had been entered into the corrective action program to determine if the licensee had been effective in identifying problems associated with the 10CFR50.59 safety evaluation process. A sample of these issues was selected for further review during which the inspectors assessed the adequacy of the corrective actions which had been implemented for the selected issues.

The safety evaluations and screenings were selected based on the safety significance of the affected Structures, Systems and Components (SSCs). A listing of the safety evaluations, safety evaluation screens and other documents reviewed is provided in Attachment A.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed four partial system walkdowns of risk significant systems to verify system alignment and to identify any discrepancies that could impact system operability. Observed plant conditions were compared with the standby alignment of equipment specified in the licensee's system operating procedures. The inspectors also observed valve positions, the availability of power supplies, and the general condition of selected components to verify there were no obvious deficiencies. The inspectors verified the alignment of the following systems:

- The "B" Train of residual heat removal service water (RHRSW) on April 7, 2003;
- The "B" Train of the residual heat removal (RHR) system on April 28, 2003;
- The "D" Train of RHRSW on May 14, 2003; and
- The "B" Train of the standby gas treatment system on June 10, 2003.

b Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Routine Fire Area Inspection

a. Inspection Scope

The inspectors identified fire areas important to plant risk based on a review of the licensee's Safe Shutdown Capability Analysis, Revision 6, as well as the Individual Plant Examination of External Events (IPEEE). Additional plant areas were selected based on their increased significance due to on-going plant maintenance. The inspectors toured these plant areas important to safety in order to verify the suitability of the licensee's control of transient combustibles and ignition sources, and the material condition and operational status of fire protection systems, equipment, and barriers. In addition, the inspectors discussed attributes of several of the areas with the fire protection engineer. The following fire areas were inspected:

- The service water system pump room (Fire Area FZ-15);
- The cable vault (Fire Area FZ-2);
- The diesel fuel oil storage tank and transfer pump house (Fire Area FA-12);
- The startup transformers (Fire Area STFRM);
- The augmented off gas building (Fire Area AOG);
- The reactor building, 280-foot elevation, recirculation pump MG [motor generator] area (Fire Area SZ RB-MG);
- The reactor building, 280-foot elevation, North side (Fire Area FZ RB5);
- The reactor building, 252-foot elevation, North side (Fire Area FZ RB3);
- The reactor building, 252-foot elevation, South side (Fire Area FZ RB4).

b. Findings

Introduction: A Green, inspector identified NCV was identified for a failure to take effective corrective actions, as required by 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," to address cable separation issues in the cable vault.

Description: The cable vault contains both safety and non safety related control, instrument, and power cables routed in cable trays and conduit. Licensee Specification VYS-027, "Separation Criteria for Reactor Protection, Engineered Safety Feature and Auxiliary Support Systems-Related Electrical Equipment and Wiring," specifies criteria used to provide physical separation and electrical isolation of circuits and components so that the safety functions required during and following any design basis event can be accomplished. Since May 2000, the licensee has documented several separate instances of cable separation issues in the cable vault. Specifically, ERs 2000-0767 and 2000-0826 documented NRC and licensee identified cable separation issues (i.e., missing cable tray covers) and ERs 2002-2811 and 2002-2826 documented NRC and licensee identified reactor protection system (RPS) cable separation issues (i.e., multiple channels of RPS cabling commingled in the same tray). In each of the above examples, the licensee took immediate actions to evaluate the specific cable separation issues for system operability concerns and to ensure the deficiencies were corrected. However, corrective actions taken by the licensee, to date, have been insufficient to ensure the extent-of-condition of cable separation deficiencies was fully understood. As a result, inspectors continue to identify cable vault cable separation deficiencies including a recent safety-related control and instrumentation cable separation issue wherein safety-related cable risers had missing covers (ER 2003-1381).

Analysis: The inspectors determined that this finding affected the objectives of the Reactor Safety Strategic Performance Area and the Mitigating Systems Cornerstone as discussed in NRC IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening." Specifically, cable separation deficiencies in the safety-related cable vault continue to be identified by NRC inspectors despite corrective actions having been taken by the licensee to address previously identified cable vault cable separation deficiencies. The inspectors determined that this finding was of very low safety significance through a review of the Significance Determination Process (SDP) because an actual loss of safety function did not occur. This finding was associated with the Cross Cutting area of Problem Identification and Resolution, in that corrective actions taken to date have not effectively precluded the recurrence of the issue.

Enforcement: 10CFR50, Appendix B, Criterion XVI "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Vermont Yankee procedure AP 0009, "Event Reports," Revision 14, describes the licensee's requirements for the identification and correction of conditions adverse to quality including assigning corrective actions that preclude recurrence. Contrary to the above, on June 25, 2003, NRC inspectors identified multiple cable separation deficiencies in the cable vault despite previous corrective actions implemented by the licensee to address similar cable separation deficiencies. Because the finding was of very low safety significance and has been entered into the licensee's Corrective Actions Program (Event Report 2003-1399), this violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy **(NCV 50-271/03-05-01).**

.2 Annual Plant Fire Drill Inspection

a. Inspection Scope

On May 19, 2003, the inspectors observed the performance of a licensee fire drill involving a simulated fire within the main transformer and a failure of the associated deluge system. The inspectors evaluated the readiness of the licensee's fire brigade against the drill objective acceptance criteria established within the drill scenario including:

- Donning of protective clothing;
- Use of self-contained breathing apparatus (SCBA) equipment;
- Fire brigade control of the effected area;
- Use and availability of fire fighting equipment; and
- Communications between the fire brigade, the main control room, and security personnel.

The inspectors also observed debriefing activities between the drill evaluators and the fire brigade to ensure lessons learned were fed back to fire brigade members.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the licensee's established flood protection barriers and procedures for coping with external flooding events. The inspectors reviewed external flooding information contained in the licensee's IPEEE and compared it to required flooding actions delineated in OP 3127, "Natural Phenomena," Revision 16. The inspectors performed walkdowns of flood vulnerable areas and ensured equipment needed to mitigate an external flooding event (e.g., sump pumps, floor drain plugs, sand bags, etc.) was available and in working order. The inspectors also reviewed a sample of problems identified in the licensee's corrective action program to verify that the licensee has identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed a performance test conducted on the “A” RHR heat exchanger in accordance with OP 4124, “Residual Heat Removal and RHR Service Water System Surveillance,” Revision 55. The inspectors reviewed test data taken, verified the licensee’s execution and monitoring of biofouling controls, and cleanliness of heat exchanger tubes. The inspectors ensured inspection results were categorized against pre-established engineered acceptance criteria, ensured that test acceptance criteria considered differences between testing and accident conditions, and that the frequency of testing was sufficient to detect degradation prior to loss of heat removal capabilities below design basis values.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

1. Inspection Scope

On June 3, the inspectors observed a simulator training session to assess the performance of the licensed operators and the ability of the licensee’s Training Department staff to evaluate licensed operator performance.

The inspectors evaluated the crew’s performance in the areas of:

- Clarity and formality of communications;
- Ability to take timely actions;
- Prioritization, interpretation, and verification of alarms;
- Procedure use;
- Control board manipulations;
- Oversight and direction from supervisors; and
- Group dynamics.

Crew performance in these areas was compared to licensee management expectations and guidelines as presented in the following documents:

- Vermont Yankee Administrative Procedure (AP) 0151, “Responsibilities and Authorities of Operations Department Personnel,” Revision 9;
- AP 0153, “Operations Department Communication and Log Maintenance,” Revision 20; and
- Vermont Yankee Department procedure (DP) 0166, “Operations Department Standards,” Revision 3.

The inspectors also compared simulator configurations with actual control board configurations. For any weaknesses identified, the inspectors observed the licensee evaluators to verify that they also noted the issues and discussed them in the critique at the end of the session.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed licensee actions taken in response to the following equipment problems to assess the effectiveness of the licensee's maintenance activities.

- SLC system pump discharge relief valve failures;
- Cracked weld in discharge piping of the "A" emergency diesel generator air compressor; and
- Problems encountered with new [design] relays installed in backup scram circuitry.

The inspectors reviewed each system's maintenance rule scoping document, most recent system health report, maintenance rule functional failure determination, and corrective actions taken in response to the equipment problem in accordance with station procedures and the requirements of 10CFR50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The inspectors also confirmed that the licensee appropriately tracked the occurrences against the systems' performance criteria, both for functional failures and unavailability time, as required.

b. Findings

Introduction: A Green NCV was identified for a failure to take corrective actions, as required by 10CFR50, Appendix B, Criterion XVI, "Corrective Actions," to address repeated as-found testing failures for SLC system relief valve SR-11-39A.

Description: Technical Specification (TS) Surveillance Requirement (SR) 4.4.A.3 requires that the relief set point of SLC system pump discharge relief valves be periodically tested to ensure the relief valves will lift at a pressure of between 1400 and 1490 pounds per square inch (PSI). The licensee accomplishes required testing of SLC system relief valves by removing the valves from the system and replacing them with bench-tested valves each refueling outage. The removed relief valves are then "as-found" tested to determine if the setpoint had changed since installation during the previous refueling outage. The licensee has also performed as-found testing of SLC system relief valves suspected of having incorrect relief setpoints due to observed premature lifting of the relief valves at or near TS testing conditions as documented in ERs 98-1632 and 03-0962.

Since April 1995, the licensee has documented four separate as-found setpoint test failures for valve SR-11-39A, the “A” train SLC system pump discharge relief valve. Specifically, ER 95-0340 documented that valve SR-11-39A had exceeded it’s as-found setpoint test, lifting at 1530 PSI during bench testing; ER 98-1653 documented that valve SR-11-39A prematurely lifted at a pressure of 1395 PSI during bench testing; ER 02-2334 documented that valve SR-11-39A leaked at 1300 PSI and lifted prematurely at 1360 PSI during bench testing; and, most recently, ER 03-1057 documented that valve SR-11-39A prematurely lifted at 1360 PSI during bench testing. In each of the above examples, the licensee took immediate action to ensure the suspect SLC system relief valves were removed from the system and replaced with valves known to have passed bench testing and to ensure continued SLC system operability. However, the licensee had not identified an adverse trend with the testing of SLC relief valves, no cause of the continued as-found testing failures had been identified, and no action had been taken to ensure that future changes in the relief valve setpoint would not impact SLC system availability or operability. The inspectors determined that there was no similar failure history regarding the SLC system “B” train pump discharge relief valve, SR-11-39B.

Analysis: The inspectors determined that this finding affected the objectives of the Reactor Safety Strategic Performance Area and the Mitigating Systems Cornerstone as discussed in NRC IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening.” Specifically, the finding affected the continued reliability of the SLC system and was therefore more than minor. The inspectors evaluated this finding in accordance with NRC IMC 0609, “Significance Determination Process.” The inspectors determined that although relief valve SR-11-39A had repeatedly failed as-found testing, the test data indicated that the relief valve would not have inadvertently lifted under anticipated transient without a scram (ATWS) conditions. Therefore the finding was determined to be of very low safety significance through a review of the SDP Phase 1 Worksheet, because the safety function was unaffected. The finding was associated with the Cross Cutting area of Problem Identification and Resolution, in that previous corrective actions had been ineffective regarding continued SLC system relief valve setpoint testing failures.

Enforcement: 10 CFR 50, Appendix B, Criterion XVI states, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Vermont Yankee procedure AP 0009, “Event Reports,” Revision 14, describes the licensee’s requirements for the identification and correction of conditions adverse to quality including determining the cause of the event, assigning corrective actions that preclude recurrence, and evaluation of similar events to determine if an adverse trend exists. Contrary to the above, between April 1995 and May 2003 the licensee documented four separate as-found setpoint test failures for the “A” train SLC system pump discharge relief valve, SR-11-39A; however, no cause of the continued as-found testing failures had been identified, no adverse trend had been identified, and no action had been taken to prevent future changes in the relief valve setpoints. Because the finding was of very low safety significance and has been entered into the licensee’s corrective actions program (Event Report 2003-0186), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy **(NCV 50-271/03-05-02)**.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

a. Inspection Scope

The inspectors evaluated on-line risk management for four planned and one emergent maintenance activities. The inspectors reviewed maintenance risk evaluations, work schedules, recent corrective actions, and control room logs to verify that other concurrent or emergent maintenance activities did significantly increase plant risk. The inspectors also compared these items and activities to requirements listed in procedures AP 0125, "Equipment Release," Revision 11 and AP 0172, "Work Schedule Risk Management - Online," Revision 4. The inspectors determined the following work activities were effectively managed for on-line risk:

- The planned limiting condition for operation (LCO) maintenance period for the “B” RHRSW pump;
- The planned LCO maintenance period for the “A” train of the RHR system;
- The planned LCO maintenance period for the “D” RHRSW pump;
- The planned LCO maintenance period for the “A” train of the standby gas treatment system; and
- The replacement of the “A” standby liquid control system pump discharge relief valve concurrent with an RHR/RHRSW outage.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

.1 Main Steam Line High Radiation Alarm Encountered During Power Suppression Testing

a. Inspection Scope

The inspectors reviewed the circumstances surrounding a previous event wherein control room operators encountered a main steam line high radiation alarm during power suppression testing. The event occurred in February of 2002. The inspectors reviewed actions taken by the control room operators as documented in ER 2002-0366; Vermont Yankee Annunciator Response Sheet (ARS) 3-F-1, “Main Steam Line Rad Monitoring,” Revision 0; Vermont Yankee Administrative Procedure (AP) 6100, “Infrequently Performed Tests or Evolutions,” Revision 0; and Vermont Yankee Operational Transient Procedure (OT) 3112, “Main Steam Line High Radiation,” Revision 14. The inspectors also reviewed the corrective actions listed in ER 2002-0366 to ensure actions assigned were appropriate and had been completed.

b. Findings

No findings of significance were identified.

.2 Inadvertent Actuation of the Unit Auxiliary Transformer Deluge System.

a. Inspection Scope

The inspectors observed the licensee’s response to an inadvertent actuation of the unit auxiliary transformer (UAT). The inspectors observed actions taken by Operations and Security Department personnel as well as the response by the licensee’s fire brigade. The inspectors performed walkdowns of affected areas (i.e., the immediate vicinity of the UAT and the turbine building) once the licensee had secured from the event. The inspectors performed a walkdown of electrical system control panels in the main control room and discussed the event with the shift manager and plant manager. Additionally, the inspectors reviewed deluge system problems identified in the licensee’s corrective action program to verify that the licensee has identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed a sample of operability determinations prepared by the licensee. The inspectors evaluated the selected operability determinations against the requirements and guidance contained in NRC Generic Letter 91-18, “Resolution of Degraded and Nonconforming Conditions,” as well as procedure AP 0167, “Operability Determinations,” Revision 1. The inspectors verified the adequacy of following evaluations of degraded or non-conforming conditions:

- East and west switchgear room ventilation system out of service;
- Excessive tubercles in service water pipe to and from the “A” emergency diesel generator coolers;
- Incorrect classification of the high pressure coolant injection system inverter as non-safety and use of replacement parts that were not designated as non-safety;
- “B” core spray system pump keepfill pressure found out of specification.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspectors reviewed the cumulative effect of operator work-arounds on continued plant reliability and availability as well. The inspectors also reviewed work-arounds for their potential impact on the operators ability to respond to plant transients and accidents as well as for increased potential of mis-operation of affected systems. The inspectors performed walkdowns, both in-plant and in the main control room, in order to identify equipment likely to contribute to operator work-arounds and reviewed the licensee's program for the control and management of operator work-arounds and burdens contained in Vermont Yankee Department Procedure (DP) 0166, "Operations Department Standards," Revision 7. The inspectors also reviewed a sample of problems identified in the licensee's corrective action program to verify that the licensee has identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

.1 Biennial Permanent Plant Modifications Inspection

a. Inspection Scope

The inspectors reviewed selected permanent plant modification packages to verify that the design bases, licensing bases, and performance capability of risk significant SSCs had not been degraded through plant modifications. Plant changes were selected for review based on risk insights for the plant and included SSCs associated with the initiating events and mitigating systems cornerstones. The inspection included walkdowns of selected plant systems and components, interviews with plant staff, and the review of applicable documents including procedures, calculations, modification packages, engineering evaluations, drawings, corrective action documents, the UFSAR and the TS. The inspectors verified that selected attributes were consistent with the design and licensing bases. These attributes included component safety classification, licensee requirements supplied by supporting systems, instrument set-points, and control system interfaces. Design assumptions were reviewed to verify that they were technically appropriate and consistent with the UFSAR. For each modification the 10CFR50.59 screenings or evaluations were reviewed. The inspectors verified that procedures, calculations and the UFSAR were properly updated with revised design information and operating guidance. The inspectors also verified that the as-built configuration was accurately reflected in the design documentation and that post-modification testing was adequate to ensure the SSCs would function properly. The inspectors also reviewed issues that had been entered into the corrective action program to determine if the licensee had been effective in identifying problems associated with the plant modification process and activities. A sample of these issues was selected for further review during which the inspectors assessed the adequacy of the corrective actions which had been implemented for the selected issues. A listing of documents reviewed is provided in Attachment A.

b. Findings

No findings of significance were identified.

.2 Annual Permanent Plant Modifications Inspection

a. Inspection Scope

The inspectors reviewed minor modification package 2202-021, “Replacement of P-8-1B Suction Barrel,” and compared the package to the requirements of AP 0020, “Control of Temporary and Minor [Permanent] Modifications,” Revision 25. The inspectors performed in-field observations of the suction barrel replacement and of removed RHRSW system piping and components. The inspectors also reviewed the adequacy of the design of the modification; the licensee’s preparation, staging, and implementation of modification, and the testing performed to verify proper installation of the modification and to ensure the reestablishment of system operability.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed post-maintenance test (PMT) activities on risk significant systems to verify that the effect of the test on the plant had been evaluated adequately. Where the testing was specifically observed, the inspectors verified test equipment was appropriate and controlled and the test was properly performed in accordance with station procedures. The inspectors either directly observed or reviewed completed PMT documentation to verify the test data met the required acceptance criteria contained in the licensee’s TS, UFSAR, and in-service testing program; the test activity was adequate to verify system operability and functional capability following maintenance; systems were properly restored following testing; and that discrepancies were appropriately documented in the corrective action process. The inspectors reviewed the following PMT activities:

- Replacement of “B” RHRSW system pump suction barrel;
- Disassembly, inspection, and refurbishment of the “A” residual heat removal (RHR) system heat exchanger discharge service water valve, V10-89A;
- Replacement of “D” RHRSW system pump suction barrel;
- Removal of standby gas treatment (SBGT) system testing valves;
- Replacement of the “A” SLC system pump discharge relief valve; and
- Testing following planned maintenance on the AS-1 Battery.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed and observed surveillance testing to verify that the test acceptance criteria was consistent with TS and UFSAR requirements, the test was performed in accordance with the written procedure, the test data was complete and met procedural requirements, and the system was properly returned to service following testing. The inspectors reviewed Administrative Procedure (PP) 7013, "Inservice Testing Program Implementation," Revision 11, AP 4000, "Surveillance Testing Program," Revision 22; and observed pre-job briefs for the test activities. The inspectors verified that systems were properly restored following testing and that discrepancies were appropriately documented in the corrective action process. The inspectors verified that the following surveillance tests met all applicable requirements:

- OP 4120, "High Pressure Coolant Injection System Surveillance," Revision 38;
- OP 4127, "John Deere Diesel Generator Quarterly Surveillance," Revision 9;
- OP 4114, "Standby Liquid Control System Surveillance, Revision 33;
- OP 4400, "Calibration of the Average Power Range Monitoring System to Core Thermal Power, Revision 21;
- OP 4115, "Primary Containment Surveillance," Revision 43; and
- OP 4195, "Fuel Oil Transfer System Surveillance," Revision 27.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed temporary modification (TM) 2002-013, "Metallic Polymer (Belzona) Repair of RRU-17A Coil," to ensure that the modification did not adversely affect the availability, reliability, or functional capability of any risk-significant structures, systems, and components. The inspectors compared the information in TM 2002-13 to the licensee's TM requirements contained in AP 0020, "Control of Temporary and Minor Modifications," Revision 25. The inspectors also walked down accessible portions of this TM to verify that required tags and markings were applied and that the TM was properly maintained. The inspectors also reviewed a sample of TM-related problems identified in the licensee's corrective action program to verify that the licensee has identified and implemented appropriate corrective actions.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

On June 3, 2003, the inspectors observed an operator crew evaluate events using the station emergency action levels during a licensed operator requalification simulator session. The inspectors discussed the performance expectations and results with the lead instructor, operations training manager, and emergency preparedness staff. The inspectors focused on the ability of licensed operators to perform event classification and make proper notifications in accordance with the following station procedures and industry guidance:

- AP 0156, "Notification of Significant Events," Revision 24;
- AP 0153, Operations Department Communications and Log Maintenance," Revision 20; and
- AP 3125, "Emergency Plan Classification and Action Level Scheme (Implementing Procedure for the licensee's VY Emergency Plan)," Revision 19;
- DP 0093, "Emergency Planning Data Management," Revision 2; and
- OP 3540, "Control Room Actions During an Emergency," Revision 3; and
- NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspectors reviewed exposure-significant work areas (i.e., high radiation areas, locked high radiation areas, very high radiation areas, and airborne radioactivity areas) in the plant and associated controls and surveys of these areas to determine if the controls (e.g., surveys, postings, barricades) were acceptable. For these areas, the inspectors reviewed radiological job requirements and attended job briefings to determine if radiological conditions in the work area were adequately communicated to workers through briefings and postings. The inspectors also verified radiological controls, radiological job coverage, and contamination controls to ensure the accuracy of surveys and applicable posting and barricade requirements.

The inspectors determined if prescribed radiation work permits (RWPs), procedures, and engineering controls were in place; whether licensee surveys and postings were complete and accurate; and if air samplers were properly located. The inspectors conducted reviews of RWPs used to access exposure significant work areas to identify the acceptability of work control instructions or control barriers specified. The inspectors reviewed electronic pocket dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy. The inspectors reviewed the locked high radiation area key inventory and surveillance list (i.e., Form VYOPF 0532.03), the key control log (i.e., Form VYOPF 0532.02), and conducted a physical inventory of all locked high radiation area keys to verify that the licensee’s key control program was being implemented as written. Procedure OP 0532, "Locked High Radiation Area Door Key Control," Rev 20, documents the licensee’s program for implementing this program area. The controls implemented by ENVY were compared to those required under TS 6.5 and the requirements contained in 10CFR20, Subpart G, “Control of Exposure from External Sources in Restricted Areas.”

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspectors reviewed work being performed during calender year 2003, including the exposure goal established for this non-outage year of 52 person-rem. This goal was established administratively, and does not include any emergent work activities during the year. The inspectors reviewed as low as reasonably achievable (ALARA) job evaluations, exposure estimates, and exposure mitigation requirements and compared ALARA plans with the results achieved.

A review of actual exposure results versus exposure estimates for work performed was conducted including: comparison of estimated and actual dose rates and person-hours expended; determination of the accuracy of estimations to actual results; and determination of the level of exposure tracking detail, and exposure report timeliness and exposure report distribution to support control of collective exposures to determine conformance with the requirements contained in 10CFR20.1101(b), “Radiation Protection Programs.”

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspectors reviewed field instrumentation utilized by health physics technicians and plant workers to measure radioactivity including portable field survey instruments, “friskers,” portal monitors and small article monitors, which were utilized to ensure that occupational exposures were maintained in accordance with 10CFR20.1201, “Occupational Dose Limits for Adults.” The inspectors visually verified that field instrumentation, including hand-held survey instruments located throughout the radiological controlled area (RCA) had current calibration stickers, and had been verified for proper daily operation through source checking.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

The inspectors reviewed the licensee's program for the training of personnel involved in the radioactive waste (radwaste) and radioactive materials transportation program with regard to the requirements contained in NRC Bulletin 79-19 and 49CFR, Subpart H. The inspectors attended portions of the licensee's continuing training program given to the radiation protection technicians as part of three training cycles during the year. Topics included triennial refresher training for hazardous materials transportation and radwaste burial requirements.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP3 Response to Contingency Events

a. Inspection Scope

The inspectors reviewed the licensee's Physical Security Plan, Revision 0; Safeguards Plan, Revision 0, and associated implementing procedures. The inspectors reviewed relevant event reports and interviewed members of the licensee's security department management and staff.

b. Findings

No findings of significance were identified.

3PP4 Security Plan Changes

a. Inspection Scope

An in-office review was conducted of changes to the licensee's Physical Security Plan, identified as Revision 0, Training & Qualification Plan, identified as Revision 0, and the Contingency Plan, identified as Revision 0. These documents were completely re-written and submitted to the NRC on September 23, 2002, in accordance with the provisions of 10CFR50.54(p), "Safeguards Contingency Plan Procedures." The review was conducted to confirm that the changes were made in accordance with 10CFR50.54(p), and did not decrease the effectiveness of the above listed Plans. The

NRC recognizes that some requirements contained in these Plans may have been superceded by the February 2002 Interim Compensatory Measures Order.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed licensee event reports (ERs), portions of operator logs, and chemistry records for the period of April 2002 to May 2003 to assess the accuracy and completeness of performance indicator (PI) data submitted by the licensee. The definitions provided in NEI 99-02, "Regulatory Assessment of Performance Indicator Guideline," Revision 2, were used to evaluate this information. The inspectors verified that the licensee accurately reported the following PIs:

- Reactor Coolant System Leakage; and
- Reactor Coolant System Activity.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify these issues were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. A listing of documents reviewed is included in the Attachment to this report.

b. Findings

No findings of significance were identified.

.2 Cross-Reference to PI&R Findings Documented Elsewhere

Section 1R05 describes a finding wherein the licensee had established ineffective corrective actions regarding safety-related cable separation deficiencies in the cable vault. Additionally, Section 1R12 describes a finding wherein the licensee had established ineffective corrective actions regarding continued SLC system relief valve setpoint testing failures.

4OA3 Event Followup

(Closed) LER 50-271/2002-003-00, "Reactor Building-to-Torus Vacuum Breakers Exceed Testing Acceptance Criteria due to Common Cause."

On November 4 and November 6, 2002, the licensee performed quarterly in-service testing (IST) of the two reactor building-to-torus vacuum breaker valves V16-19-12A and 12B. The valves are 20 inch Walworth swing check valves. During testing, both valves exceeded the IST program lift force acceptance criteria and one valve, V16-19-12A, failed to lift under the TS required minimum lifting force of 0.5 pounds per square inch differential (PSID). The licensee was able to cause valve V16-19-12A to lift after applying a force of 0.52 PSID. Although the valve failed to meet TS testing requirements, lifting at 0.52 PSID did meet the licensee's design bases lift pressure requirement of 2.0 PSID. The seats for these valves had been replaced with elastomeric seating material during the October 2002 refueling outage. The licensee performed a root cause analysis and determined that the above IST surveillance testing was performed subsequent to pressurization of the up-stream side of these valves to 44 PSI during local leak rate testing (LLRT). The LLRT pressurization caused "firm" seating of the valves against the newly installed elastomeric seating surfaces, resulting in a higher breakaway force being needed to unseat the valves. The licensee subsequently LLRT tested these valves at approximately 27 PSI, equating to a closing force equal to the postulated peak torus accident pressure for Vermont Yankee. Under these test conditions, the licensee determined that their LLRT requirements, the testing requirements of 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," as well as all IST and TS requirements would be satisfied. Corrective actions taken by the licensee included evaluation and future revision of LLRT procedures, evaluation of the continued suitability of the elastomeric seating material, and performance of a root cause analysis. The licensee determined that no similar events had occurred at Vermont Yankee or within the industry as a whole. The LER and associated root cause analysis were reviewed by the inspectors taking into consideration the licensee's selection of seating materials; maintenance performed on the valves; specified post-maintenance testing; 10 CFR 50, Appendix J requirements; and industry operating history. No findings of significance were identified. The licensee documented the issue in ER 2002-2716. This LER is closed.

4OA5 Other Activities

The inspectors reviewed the results of an Institute of Nuclear Power Operations (INPO) Evaluation conducted at Vermont Yankee during the weeks of March 3 and 10, 2003. The inspectors also reviewed the results of the INPO evaluation of Vermont Yankee's Training Program conducted the week of March 31, 2003.

4OA6 Meetings, including Exit

On July 17, 2003, the resident inspectors presented the inspection results to Mr. K. Bronson, and other members of his staff who acknowledged the findings presented. The inspectors asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel:

J. Thayer, Site Vice President
M. Balduzzi, Site Vice President, Operations
K. Bronson, General Plant Manager
P. Corbett, Maintenance Manager
M. Desilets, Technical Services Manager
J. Geyster, Radiation Protection Superintendent
D. Giorowall, Programs Supervisor
S. Goodwin, Mechanical Design Department Manager
M. Gosekamp, Superintendent of Operations Training
D. Leach, Director of Engineering
F. Marcussen, Security Operations Manager
R. Morissette, Principal ALARA Engineer
M. Pletcher, Radiation Protection Supervisor - Instruments
K. Stupak, Technical Training
C. Wamser, Operations Manager
R. Wanczyk, Director of Nuclear Safety

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-271/03-05-01	NCV	Failure to Take Effective Corrective Actions Regarding Safety-Related Electrical Cable Separation (Section 1R05)
50-271/03-05-02	NCV	Failure to Take Effective Corrective Actions Regarding Continued Failures Standby Liquid Control System Relief Valves (Section 1R12)

Closed

50-271/03-05-01	NCV	Failure to Take Effective Corrective Actions Regarding Safety-Related Electrical Cable Separation (Section 1R05)
50-271/03-05-02	NCV	Failure to Take Effective Corrective Actions Regarding Continued Failures Standby Liquid Control System Relief Valves (Section 1R12)
50-271/2002-003-00	LER	Reactor Building-to-Torus Vacuum Breakers Exceed Testing Acceptance Criteria due to Common Cause (Section 4OA3.1)

LIST OF DOCUMENTS REVIEWED

Section 1R02: Evaluations of Changes, Tests, or Experiments

10CFR50.59 Safety Evaluations

1998-004, “Fuel Rack Installation”
2001-004, “Replacement of 24 VDC [volts direct current] ECCS [emergency core cooling system] Batteries with DC Power Supplies”
2001-015, “Add 150 Ampere Thermal Magnetic Breaker to 125 VDC Panel DC-2"
2001-019, “Reload 21/Cycle 22"
2001-024, “Change Mode Switch to Shutdown before coolant temperature reaches 212 degrees”
2001-029, “HPCI [high pressure coolant injection] and RCIC Turbine Overspeed Operability Evaluation”
2001-030, “Replacement of Springs in [valve] V13-6/7"
2001-032, “Set Point Change, RCIC [reactor core isolation cooling] Turbine Exhaust High Pressure Trip”
2001-033, “Gathering Data for FRV [feed water regulating valve]”
2001-036, “Core Spray System Revisions”
2002-003, “Main Turbine EPR [electronic pressure regulator] Replacement”
2002-006, “FW [Feed Water] Control System Replacements]”
2002-023, “New Test Loop for Service Water IST [in-service testing] and Surveillance Testing”
2002-040, “Connect Battery Charger BC-1-1A to allow corrective maintenance to Battery Charger BC-1-1A”

10 CFR 50.59 Safety Evaluation “Screenings”

2001-016, “Modification to Tritium Sampling for Continuous Sampling”
2001-024, “Replace Pre-Cast Tank Motor”
2001-026, “Main Steam Line SRV [safety relief valve] and SRV Exhaust Temperature Indication”
2001-042, “Replace TRU [turbine building recirculation unit] 5 Cooling Coil, Condensate Pump Air Coolers”
2002-003, “Addition of Appendix R Emergency Light”
2002-006, “Recirc MG Set - Loss of DC”
2002-012, “Main Steam Line Radiation Monitor, Reactor Trip, and MSIV [main steam isolation valve] Closure Elimination”
2002-013, “Scram Contactor Auxiliary Contact Replacement”
2002-016, “Diesel 1B Frequency Meter Replacement”

50.59 Applicability Determinations Reviewed:

Vermont Yankee Design Change (VYDC) 2001-002, “FW Control System Replacement - Phase 1 RBCCW Design Basis Document Interim Change”
VYDC 2001-003, “RCIC Turbine Exhaust Check Valve Replacement”
Interim Change RO-007 Regarding PCIS [Primary Containment Isolation System] and Design Basis Document Interim Changes for PCIS

Minor Modification (MM) 2002-012, “Main Steam Line Radiation Monitoring Reactor Trip and MSIV Closure Elimination”
MM 2001-036, “Install Pressure Relief Valves on Fire Protection System”
VYDC 2002-006, “MG Set - Loss of DC”

Procedures

OP 4300, “SRM Functional/Calibration Procedure,” Revision 16
OP 0105, “Reactor Operations,” Revision 7

Miscellaneous Documents

Moore Process Automation Solutions Software Release Memo SR 353-7, Rev. 1
Moore Process Automation Solutions Software Release Memo SR 353-6, Rev. 1
VYAPF 0145.01, “Cyclic Limitations For Plant Transients And Operating Events”
VY Basis for Maintaining Operation (BMO) 2007-07, “Potential Over-pressurization of HPCI/RCIC on Turbine Over-speed Trip”
GE SIL 623 dated 10/22/1999

Section 1R17: Permanent Plant Modifications

Drawings

DWG 5920-279, “Recirc Flow Control System,” Revision 10
DWG 5920-1034, Recirc Flow Control System,” Revision 7
DWG 5920-5134Sh.1, “Recirc Set W/D Speed Control Fluid Drive,” Revision 3
DWG 5920-12901Sh.1, “Actuator Mounting Assembly-RH,” L/R
DWG 5920-12901Sh.2, “Actuator Mounting Assembly-LH,” L/R

Design Calculations

VYDC-2000-002, “Upgrade of Recirc Pump Speed Control Loops”
VYDC 2000-027, “Main Turbine EPR Replacement”
VYDC-2001-002, “Feedwater Control System Replacement - Phase 1"

Section 4OA2.1: Routine Review of Problem Identification and Resolution

Event Reports

1995-0340	Safety Relief Valve (SR-11-39A) exceeded its setpoint while being tested
1996-0075	“C” & “F” APRM’s [Average Power Range Monitor] GAF [Gain Adjustment Factor] > 1.0
1998-1632	SLC System Relief Valve lift
1998-1653	Test failure of safety & relief valves
1999-1006	APRM “E” declared inoperable
2000-1599	TS 3.1/Procedure OP 4401 requirements disparity

2001-0249	Inconsistent master recirculation controller response
2001-1377	Recirc MG “B” speed controller erratic operation
2001-1448	Work not completed per intended sequence
2001-1824	Operation of the master recirc flow controller doesn’t allow flow to be reduced to 27.0-27.5 mlb/hr
2001-2317	APRM “E” GAF>1.0
2002-0119	“E” APRM GAF erratic
*2002-0366	Communication breakdown during an infrequently performed evolution (Power Suppression Testing)
2002-0705	Change in APRM reading due to possible changes in APRM GAF
2002-0929	Recirc flow controller swap from auto to manual
2002-1502	Erratic operation of recirc pump master flow controller
2002-1815	APRM “A & B” GAF’s high
2002-2067	Questionable system response based on feedwater level controller model
2002-2334	SR-11-39A failed bench leakage and pressure testing
2002-2716	Failed surveillance of DW to torus vacuum breaker
2002-2725	Feedwater level controller trouble
2002-2811	RPS [reactor protection system] division RB1 cables found laying in an RA1 cable tray
2002-2826	Potential cable separation violations - RPS trays
2002-2852	Cable vault tray risers mislabeled
2003-0328	“A” recirc controller auto shift to manual
2003-0370	Step decrease change in recirc flow master controller
2003-0626	Excessive tubercles in service water pipe to and from the “A” emergency diesel generator coolers
2003-0658	Cracked weld in discharge piping of the “A” emergency diesel generator air compressor
2003-0599	Containment cam filter housing cover plate incorrectly installed
*2003-0847	Late entry into LCO
*2003-0848	Torus level indicator & associated level recorder effected by torus cooling flow
*2003-0866	CRD flow controller found out of auto
2003-0874	Fuses found installed did not match fuse list
*2003-0912	Discrepancy on instrument reading during IST surveillance
*2003-0956	IST surveillance acceptance criteria changed without a supporting change in the reference data set
2003-0957	Missed local leak rate test for PMT
2003-0960	Incomplete corrective action from ER96-0421
2003-0961	Work entered a surveyed HRA in the overhead on a non-HRA RWP
*2003-0962	SLC relief valve lifted during pump surveillance
2003-0993	Potential increased level drop during shut down cooling initiation
2003-0995	Lightning strike
2003-1012	Adverse trend for RPS [reactor protection system] scram relays
2003-1039	The “B” core spray pump keepfill pressure found out of specification
2003-1057	Failed relief valve setpoint test
2003-1113	Problems with the new relays (backup scram relays) installed during last RFO [refueling outage]
2003-1156	Several operator aids missing, damaged, or with wrong revision

*2003-1319	Fire Hazards Analysis incorrectly lists an NRC commitment which is no longer valid
*2003-1381	Missing covers on risers in cable spreading room
2003-1399	Adverse trend for cable vault cable separation issues identified by external sources
2003-1408	Trip of fire system deluge valve for aux transformer
*2003-1411	Accidental removal of badge from site
*2003-1476	NRC questions licensee operator requalification training performance indicators

* Inspector-identified issue.

LIST OF ACRONYMS

ADAMS	Automated Document Access Management System
ALARA	As Low as is Reasonably Achievable
AP	Vermont Yankee Administrative Procedure
ARS	Annunciator Response Sheets
CFR	Code of Federal Regulation
CY	Calendar Year
DP	Vermont Yankee Department Procedure
EPR	Electronic Pressure Regulator
ER	Event Report
FW	Feed Water
HPCI	High Pressure Coolant Injection
ICM	Interim Compensatory Measures
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IPEEE	Individual Plant Evaluation of External Events
IR	Inspection Report
IST	In-Service Testing
LER	Licensee Event Report
LCO	Limited Condition for Operation
LLRT	Local Leak Rate Testing
MG	Motor Generator
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OP	Vermont Yankee Operating Procedure
PI	Performance Indicator
PMT	Post Maintenance Testing
PP	Administrative Procedure
PRM	Process Radiation Monitor
PSI	Pounds per Square Inch
PSID	Pounds per Square Inch; Differential
RCIC	Reactor Core Isolation Cooling
RM	Radiation Monitor
RPS	Reactor Protection System
RFO	Refueling Outage
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RWP	Radiation Work Permit
SBGT	Standby Gas Treatment
SDP	Significance Determination Process
SLC	Standby Liquid Control
SR	Surveillance Requirement
SSC	Structures, Systems and Components
SW	Service Water
TS	Technical Specification

UAT	Unit Auxiliary Transformer
UFSAR	Updated Final Safety Analysis Report
VDC	Volts Direct Current
VY	Vermont Yankee
VYDC	Vermont Yankee Design Change