October 13, 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 PLANT - NRC INSPECTION REPORT NO. 05000271/2003008

Dear Mr. Thayer:

On August 29, 2003, the U. S. Nuclear Regulatory Commission (NRC) completed a team inspection at the Vermont Yankee Power Station. The enclosed report documents the inspection findings which were discussed on August 29, 2003, with Mr. K. Bronson and other members of your staff.

The inspection was an examination of activities conducted under your license as they relate to the identification and resolution of problems, compliance with the Commission's rules and regulations, and with the conditions of your license. Within these areas, the inspection involved examination of selected procedures and representative records, observation of activities, and interviews with personnel.

On the basis of the samples selected for review, the team concluded that in general, problems were properly identified, evaluated and corrected. There were two green findings identified during this inspection related to the effectiveness of the corrective action program. Specifically, for a problem originally identified by the Vermont Yankee staff on the position of control fuses for the Vernon tie line, the team identified that a non-cited violation of 10 CFR 50, Appendix R, Section III had occurred. The team found that the plant staff had not identified or fully evaluated the non-compliance with Appendix R and had not performed an appropriate evaluation for the impact on the ability of the operators to achieve and maintain safe shutdown. The second finding involved the adequacy of corrective actions to address problems identified during the testing of relief valves. These findings were determined to be violations of NRC requirements. However, because of their very low safety significance and because they were entered into your corrective action program, the NRC is treating these two findings as non-cited violations, in accordance with Section VI.A.1 of the NRC Enforcement Policy. If you deny these noncited violations, you should provide a response with the basis of your denial within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Vermont Yankee Nuclear Station.

In addition, several examples of minor problems were identified by the team that your staff entered into the corrective action program. Some of these items involved corrective actions that were ineffectively tracked or had not been implemented. None of these minor deficiencies resulted in a challenge to system operability or reliability.

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,

Raymond K. Lorson, Chief
Performance Evaluation Branch
Division of Reactor Safety

Docket Nos.: 50-271
License Nos.: DPR-28

Enclosure: Inspection Report 05000271/2003008
w/Attachment: Supplemental Information

cc w/encl:
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-271
License No:	DPR-28
Report No:	05000271/2003008
Licensee:	Entergy Nuclear Vermont Yankee, LLC
Facility:	Vermont Yankee Nuclear Power Station
Location:	320 Governor Hunt Road Vernon, Vermont 05354-9766
Dates:	August 11-15 and August 25-29, 2003
Inspectors:	Frederick W. Jaxheimer, Reactor Inspector (Team Leader) Frank Arner, Senior Project Engineer Jennifer Bobiak, Reactor Engineer Chris Cahill, Senior Reactor Inspector Beth Sienel, Vermont Yankee Resident Inspector Harold Eichenholz, Senior Reactor Inspector
Approved by:	Raymond K. Lorson, Chief Performance Evaluation Branch Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000271/2003-008; Vermont Yankee Nuclear Power Station; 8/11/2003 - 8/29/2003; biennial baseline inspection of problem identification and resolution of problems. Violations were identified in the areas of Appendix R and corrective actions.

This inspection was conducted by five regional inspectors and one resident inspector. Two green findings of very low safety significance were identified during this inspection and were classified as noncited violations. The findings were evaluated using the significance determination process (SDP).

Identification and Resolution of Problems

The team determined that generally Vermont Yankee (VY) properly identified, evaluated and corrected problems. Nevertheless, the NRC team identified two findings which indicated deficiencies with the effectiveness of corrective actions. Excluding these two findings and the specific performance tied to previous inspection findings identified by the Resident inspectors during this two year assessment period, the team found that VY adequately prioritized and evaluated problems that were entered into the corrective action program. Other than the reported inspection issues, the inspection team found that in general, corrective actions were implemented in a timely manner. Audits and self-assessments were found to be acceptable. On the basis of interviews conducted during the inspection, workers at the site felt free to input safety findings into the corrective action program.

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix R, Section III, “Alternate and Dedicated Shutdown Capability,” paragraph L.3, which requires that “the alternate shutdown capability shall be independent of the specific fire area(s) and shall accommodate post fire conditions where offsite power is available and where offsite power is not available for 72 hours.” The primary, alternate shutdown power source control power fuses were found in the off position. In this condition, the alternate shutdown capability was not independent for a fire in the control room or cable spreading room.

  This finding was greater than minor because fuses were improperly installed which impacted the ability to implement an alternate shutdown independent of a fire in the control room or cable spreading room. The finding was determined to be of very low significance (Green) since its safety function (i.e., restoration of power) could be accomplished before core damage would occur through the use of the "A" EDG (Section 4OA2b).

- Green. The team identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” for the failure to establish effective corrective actions to address quality issues identified during in-service relief valve testing.

This finding is greater than minor since the failure to develop adequate corrective actions for in-service relief valve test failures could allow similar problems to remain undetected in other potentially affected relief valves and adversely impact mitigating system reliability. This finding was determined to be of very low significance (Green) since an actual loss of the safety system function had not occurred as a result of this problem (Section 4OA2.c).

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

a. Effectiveness of Problem Identification

(1) Inspection Scope

The inspection team reviewed the procedures describing the corrective action program at the Vermont Yankee Nuclear Power Station (VY). The team reviewed items selected from various licensee processes and activities to determine if personnel were properly identifying, characterizing and entering problems into the corrective action program for evaluation and resolution.

The team reviewed logs, control room deficiencies and operator work-arounds, system health reports, temporary modifications, operating experience reviews, and procedures. In addition, the team interviewed plant staff and management to determine their understanding of and involvement with the corrective action program. The specific documents reviewed and referenced during the inspection are listed in the attachment to this report.

The team reviewed a sample of nuclear safety assessment audits and assessments, as well as departmental and program self-assessments. The team evaluated the effectiveness of the audits and self-assessments by comparing the associated results against self-revealing and NRC-identified findings.

The team conducted several plant walkdowns of safety-related, risk significant areas to determine if observable system equipment and plant material adverse conditions were identified and entered into the corrective action program (CAP). Team members attended daily review and management meetings, where event reports (ERs) were reviewed for screening and assignment. The team attended these meetings to understand the threshold for identifying problems and to assess management involvement with the corrective action program. The team also assessed the interface between the corrective action program and the work control process.

(2) Observations and Findings

No findings of significance were identified.

Overall, the licensee's effectiveness at problem identification was acceptable. Audits and self-assessments were self-critical and generally consistent with the team's findings. Vermont Yankee staff members promptly initiated ERs as appropriate in response to inspection team identified deficiencies.

b. Prioritization and Evaluation of Issues

(1) Inspection Scope

The team reviewed the action items and ERs listed in the attachment to this report to assess whether the licensee adequately prioritized and evaluated problems. The team selected the ERs in areas to cover the seven cornerstones of safety identified in the NRC Revised Oversight Program. These reviews evaluated the causal assessment of each issue (i.e., root cause analysis or apparent cause evaluation); and for significant conditions adverse to quality, the extent of condition and determination of corrective actions to preclude recurrence. Additionally, the team attended the daily meetings to observe the ER review process and to understand the basis for assigned significance levels and root cause levels.

The team also selected a sample of ERs associated with previous NRC non-cited violations (NCVs) to determine whether the licensee evaluated and resolved problems associated with compliance to applicable regulatory requirements. The team reviewed the evaluation of industry operating experience information for applicability to VY. The team also reviewed the VY's assessment of equipment operability, reportability requirements, and extent of condition.

(2) Observations and Findings

The team concluded that, in general, VY evaluated and categorized problems contained within the ER process at the correct significance level. The staff was generally effective at classifying and performing operability evaluations and reportability determinations for discrepant conditions. However, the team noted some examples where problem evaluations did not contain sufficient depth in causal determination or did not contain sufficient basis to support the associated conclusions. For example, the team noted two discrepancies associated with the reportability review of ER 2002-2105 which involved a condition where alternate shutdown fuses were found to be in the "off" position. Specifically, the review did not identify an Appendix R compliance issue (discussed below) and also did not incorporate relevant information that would have challenged the effectiveness of assumed operator actions that were credited to mitigate this problem. These discrepancies were discussed with VY and subsequently entered into the corrective action program as ER2003-1761. Additionally, a revision was subsequently issued to the potentially reportable occurrence report PRO022105.

Implementation of Alternative Shutdown Capability

a. Inspection Scope

The team reviewed ER 20022105 titled "Both Alternate Shutdown Fuses Found in the Off Position (RUT and RUC Fuses)" to determine the impact on the safe shutdown capability at VY. The team performed walkdowns of the alternate shutdown procedure, discussed the procedural guidance for restoring electrical power with licensed operators and fire protection engineers, reviewed the inventories of the post-fire shutdown tools contained in the Appendix R toolboxes in the switchgear rooms, and reviewed the availability of personal protective equipment necessary for shutdown outside the control room. Additionally, the team compared the manual actions for shutdown outside the control room to the criteria listed in NRC Inspection Procedure 71111.05, dated March 6, 2003, to determine feasibility of the actions called for in the safe shutdown procedure and to evaluate whether VY's corrective actions for this problem were appropriate.

b. Findings

Introduction

The inspectors identified a non-cited violation (NCV) for the failure to maintain the alternate shutdown capability independent of a postulated fire in either the control room or cable spreading room as required by 10 CFR 50, Appendix R, Section III, paragraph L.3.

Description

Vermont Yankee received an exemption from 10 CFR 50, Appendix R, Section III, "Alternate and Dedicated Shutdown Capability," paragraph L.3, in August 1997. This exemption allowed the use of the Vernon Tie line as an acceptable alternative to an onsite emergency diesel generator for fire events where offsite power was not available. Vermont Yankee determined that an exemption request was required in order to meet their Safe Shutdown Capability Analysis (SSCA). Specifically, VY memorandum dated October 16, 1996, titled "Timelines operator actions in OP3126 rev. 14" identified that the newly issued SSCA documented that the time available for the restoration of alternating current (AC) power and initiation of reactor core isolation cooling system (RCIC) had been reduced to approximately 22.5 minutes. Vermont Yankee determined that AC power could be restored in approximately 10 minutes by using the Vernon Tie line and in approximately 30 minutes by using an onsite emergency diesel generator (EDG).

To permit the use of the Vernon Tie as part of the post-fire alternate shutdown system, VY modified the controls of the Vernon Tie circuit breaker to isolate control room cables and transfer control of the breaker to local control switches at the switchgear. To eliminate the potential for hot-shutdown repairs, redundant fuses (RUT and RUC) were installed and designed to be automatically switched into the circuit upon operation of the transfer switches. On September 30, 2002, while performing electrical integrity testing on the alternate shutdown fuse blocks under work order 02-002652-000, technicians found that the Vernon Tie line redundant fuses (RUT and RUC) were installed in the "off" position. In this position the fuses are physically removed from the control circuitry and would not be switched into the circuit upon operation of the transfer switches. Vermont Yankee determined that this condition was present for approximately 16 months.

Procedure OP 3126, revision 16, "Shutdown Using Alternate Shutdown Methods" is used to shutdown the plant following fires in the control room or cable spreading room requiring evacuation of the control room. The procedure identified the Vernon Tie as the primary source of protected AC power. The procedure directs the operators to use the "A" EDG in the event that the Vernon Tie is unavailable. Although the SSCA predicted that core uncovery would occur in approximately 22.5 minutes, the Appendix R Boil-Off Sensitivity Study predicted that the fuel peak centerline temperature (PCT) would be low enough to preclude core damage for up to 50 minutes during the control room evacuation fire scenario. The team reviewed the protection scheme for the "A" EDG and walked down the procedure to verify that the "A" EDG could provide AC power and RCIC could be initiated in the time required to prevent core damage.

Analysis

The finding adversely impacted the ability of the operators to achieve and maintain safe shutdown conditions in the event that shutdown from outside the control room was required. Because the finding is associated with the reactor safety mitigating system cornerstone and affects the protection against external factors attribute (i.e., fire), the finding is more than minor.

The finding was evaluated in accordance with MC 0609, Appendix A, Attachment 1, Significance Determination Process (SDP) Phase I screening for mitigating systems. The finding was determined not to be associated with design or qualification deficiency and did not represent an actual loss of safety function. The finding was determined to be potentially risk significant due to a fire initiating event. The team determined that in the event that the Vernon Tie was not available, its safety function, restoration of AC power, could be accomplished before core damage would occur through the use of the "A" EDG. As a result this finding was determined to have very low safety significance and screened as a Green finding in Phase I of the SDP.

Enforcement

10 CFR 50, Appendix R, Section III, "Alternate and Dedicated Shutdown Capability," paragraph L.3, requires that "the alternate shutdown capability shall be independent of the specific fire area(s) and shall accommodate post fire conditions where offsite power is available and where offsite power is not available for 72 hours." Contrary to the above, in September 2002, the alternate shutdown fuses for the primary alternate shutdown AC power source were found in the “off” position. In this condition, the alternate shutdown capability was not independent for a fire in the control room or cable spreading room. This is a violation of 10 CFR 50, Appendix R, Section III, Paragraph L.3.

Because the failure to maintain the safe shutdown capability independent of a fire in the cable spreading room and/or the control room was of very low safety significance and has been entered into the corrective action program (ER 20022105 and ER 20031761), this violation is being treated as a non-cited violation consistent with section VI.A of the NRC enforcement policy: **NCV 05000271/2003008-01**, Failure to Maintain the Safe Shutdown Capability Independent of a Fire in the Cable Spreading Room and/or Control Room.

c. Effectiveness of Corrective Actions

(1) Inspection Scope

The team reviewed the corrective actions associated with selected ERs to determine whether the actions addressed the identified causes of the problems. The team reviewed ERs for repetitive problems to determine whether previous corrective actions were effective. The team also reviewed the licensee's timeliness in implementing corrective actions and their effectiveness in precluding recurrence of significant conditions adverse to quality. The team also reviewed non-cited violations issued since the last inspection of the licensee's corrective action program to determine if issues placed in the program had been properly evaluated and corrected.

(2) Observations and Findings

Overall, the team concluded the licensee developed and implemented corrective actions that appeared reasonable to address the identified problems. Based on the sample reviewed, the team determined that, in general, corrective actions were completed or scheduled to be completed in a timely manner commensurate with the potential significance of the issue. The team identified one finding related to the corrective actions for safety relief valve test failures (discussed below) and also identified some minor examples where operability determinations, self-assessments and corrective actions were not thorough or fully implemented to resolve identified problems. Event Reports generated to document and review these issues are listed in the attachment to this report.

Response to Safety Relief Valve Test Failures

a. Inspection Scope

The team reviewed the in-service testing (IST) of relief valves to determine whether VY's actions for previous RCIC system and service water system relief valve test failures were appropriate. The team examined associated program documents, test data, system diagrams, work documents, self-assessment and event reports and interviewed personnel during this review.

b. Findings

Introduction

A Green NCV was identified for a failure to establish appropriate corrective actions for relief valve test failures as required by 10 CFR 50 Appendix B Criterion XVI.

Description.

The SR-13-25 relief valve on the suction piping of the RCIC system was removed from service on February 24, 2003 and bench tested on July 8, 2003, for partial complement testing, as part the IST program. The team noted that the testing was not completed within three months following removal of the valve from the system as specified by the ASME Code. The test results indicated that the valve had excessive seat leakage that precluded it from being pressurized to the required setpoint range of 148-152 psig. Vermont Yankee initiated a number of actions on July 8-9, 2003, to address the failure to meet the 3-month ASME code testing requirement and the setpoint test failure. Specifically, work order (WO) 03-58379 and WO 03-3714-000 were initiated to conduct the ASME Code required expansion testing on the SR-13-26 relief valve.

Event report 03-1491 documented that the SR 13-26 valve would need to be tested as soon as reasonably possible, and VY System Engineering Memorandum 2002-100 stated that expanded testing was to be completed within the three month period specified by the ASME Code for completion of the partial complement testing unless approved by the plant IST Coordinator. The team noted that, based on the reported lack of an available replacement valve, VY did not plan to complete the expanded testing on the SR 13-26 valve until December 2003. The VY event investigation attributed the SR 13-25 test failure to a loose nozzle that had been identified during the initial pre-test inspection of the valve. However, after tightening, an informational retest was performed and the SR 13-25 valve produced the same failure results. The team concluded that VY had not determined the cause for the relief valve test failure and noted that if SR 13-26 exhibited similar test results (i.e. premature lifting of the relief valve) then the RCIC lubrication oil cooling flow could be adversely affected thus challenging the operability and/or reliability of the RCIC system. Vermont Yankee subsequently issued ER 03-1855 to address their lack of proper follow-up on the SR 13-26 relief valve.

The team also identified the area of VY's testing of relief valves in the service water system as involving inadequate corrective actions to address conditions adverse to quality. Specifically, ER 03-1640, documented that the SR 70-2A relief valve failed to lift at the required setpoint of between 148-154 psig on July 29, 2003. In fact, the valve did not lift up to an applied value of 187 psig (125% of setpoint). Visual inspection of the inlet nozzle did not show any corrosion, silt or other buildup that may have affected the valve setpoint. The ER documented that the previous setpoint test of SR 70-2A in November 1999 also identified that the valve failed to lift. The SR 70-2B relief valve history showed that the valve experienced service water fouling and failure to lift during setpoint testing in December 1999 (ER 00-0003). The assessment of similar conditions in ER 03-1640 specified that fouling of the inlet piping/relief valve nozzles, combined with failure to lift, had occurred in six other relief valves, including the diesel generator jacket cooler relief valve SR 70-16B. This later valve experienced minor silting in November 1999 and had a setpoint that failed high (ER 99-1739) and failed again in April 2002 (WO No. 01-4329-00), but had no ER. The event analysis for ER 03-1640 documented that service water applications of relief valves showed a history of failing to lift due to sticking and required further investigation. The team determined that the described conditions involving relief valves in the service water system represented a condition adverse to quality, and the longstanding existence (at least since 1999) of numerous failures during IST valve testing had not been appropriately addressed by the VY to preclude recurrence.

Analysis

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 failure to develop adequate corrective actions for in-service relief valve test failures could have allowed similar problems to remain undetected in other potentially affected relief valves and adversely impacted mitigating system reliability. The Team evaluated this finding in accordance with NRC IMC 0609, “Significance Determination Process,” and determined that, while the failure to develop adequate corrective actions for previous RCIC and service water system relief valve IST failures could have allowed a degraded system to remain in-service undetected, there was no evidence that a loss of the safety function of either system had occurred. Therefore, this finding was determined to be of very low significance (Green) in accordance with the Phase I worksheet of the SDP.

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. Contrary to the above, since November 1999, VY failed to implement prompt and appropriate corrective actions for multiple relief valve test failures. Because this low risk violation has been entered into the corrective action program as ER’s 03-1855 and 03-1910, it is being treated as a non-cited violation consistent with Section VI.A of the NRC enforcement policy: **NCV 05000271/2003008-02,** Failure to Implement Adequate Corrective Actions for Relief Valve Test Failures.

d. Assessment of Safety Conscious Work Environment

(1) Inspection Scope

During this inspection, the team interviewed a cross section of plant staff to determine if conditions existed that would result in personnel being hesitant to raise safety concerns to their management and/or the NRC.

(2) Observations and Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. Kevin Bronson and other members of Vermont Yankee staff on August 29, 2003. During the inspection, no proprietary information was examined or retained by the team.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

T. Autry	Technical Support Superintendent
R. Booth	Component Engineer
R. Burns	System Engineer
M. Empey	EP Drill & Exercise Coordinator
P. Imm	Radiation Tech
S. Jonasch	System Engineer
L. Lukens	IST Engineer
M. McCluskie	Operations Standards Supervisor
B. Naeck	System Engineer
W. Penniman	Project Coordinator
M. Romeo	Nuclear Training Superintendent
R. Rusin	Programs and Components Supervisor
J. Stasolla	System Engineer
C. Tabone	Sr. Operations Instructor
L. Tkaczyk	Emergency Preparedness Manager
F. Underkoffler	Code Programs Engineer
R. Wanczyk	Director, Nuclear Safety
G. Wierzbowski	System Engineering

ITEMS OPENED AND CLOSED

Opened and Closed

05000272/2003008-01	NCV	Alternate Shutdown Capability was not Independent for a Fire in the Control room or Cable Spreading Room
05000272/2003008-02	NCV	Failure to Implement Adequate Corrective Actions for Relief Valve Test Failures

LIST OF DOCUMENTS REVIEWED

Policies & Procedures

AP 0009	Event Reports, Rev. 14, 3/11/03
AP 0047	Work Request, Rev. 7, January 31, 2002
AP 0077	Barrier Control Process, Rev.2, 9/27/02
AP 0140	Vermont Yankee Local Control Switching Rules, Rev.23, 3/5/02
AP 0167	Operability Determinations, Rev. 1, 7/30/02
AP 0168	Vermont Yankee Work Management, Rev. 2, January 31, 2002
AP 0502	Radiation Work Permits
DP 0166	Operations Department Standards, Rev. 7, 10/18/02
ENN-PL-125	Employee Concerns Policy, Rev. 1, 3/14/03
OP 0105	Reactor Operations, Rev.10, 10/18/02
OP 3126	Shutdown Using Alternate Shutdown Methods, Rev 16
OP 4123	Core Spray System Surveillance, Rev. 36, 7/2/03
OP 4124	Residual Heat Removal (RHR) and RHR Service Water System Surveillance, Rev. 56, 7/29/03
OP 4261	Safety and Relief Valve Testing, Rev. 5, June 16, 2000
OP 7013	Inservice Testing Program, Revision 12, June 24, 2003
PP 7013	Inservice Testing Program, Draft Revision 13
PP 7017	Corrective Action Program Procedure, Rev. 2, 7/12/02
PP 7038	Vermont Yankee Human Performance Program, Rev. 1, 4/8/03
PP 7204	Safety & Relief Valve Program, Rev. 2

Audits and Self-Assessments

SA 01-02-04	Operations Procedure Place Keeping (focused assessment), 6/20/02
SA 01-02-04A	Operations Procedure Usage (focused assessment), 6/20/02
SA 16-04-04	ERO Members Included in CAN Database (EP focused assessment), 7/8/02
SA 2001-001-01	Engineering Self-Assessment
OPVY-2003-009-01	Self-Assessment of Control Room Protocol & Formality (Operations focused assessment), 4/23/03
OPVY-2003-010-01	Self-Assessment of Reactivity Management (Operations focused assessment), 6/19/03
OPVY-2003-024-01	Assessment of Emergency Planning Program (Fleet Assessment), 5/03
OPVY-2003-030-01	Self-Assessment of Quality of RP Worker Practices
RPSA 2003-04	Self-Assessment on RP Human Performance Related Event Reports
SRVY 2002-023	Quality Assurance Emergency Planning Assessment, 9/3/02)
06-02-06	Self-Assessment, Human Performance/Procedure Adherence
2001-037	Work Process Implementation
2002-025	Maintenance Programs Assessment
OPVY-2003-065-01	Safety and Relief Valve Program On-Going Self-Assessment

Engineering Human Performance Database Monthly Report

Event Reports

2001-1828	2001-1938	2001-2025	2001-2119
2001-2183	2001-2306	2001-2307	2001-2335
2001-2336	2001-2385	2001-2431	2001-2564
2001-2565			

2002-0037	2002-0109	2002-0126	2002-0136
2002-0182	2002-0258	2002-0431	2002-0529
2002-0587	2002-0638	2002-0724	2002-0795
2002-0885	2002-0887	2002-0894	2002-0909
2002-1111	2002-1182	2002-1193	2002-1231
2002-1232	2002-1263	2002-1340	2002-1393
2002-1446	2002-1458	2002-1509	2002-1791
2002-2019	2002-2067	2002-2095	2002-2105
2002-2105	2002-2183	2002-2186	2002-2264
2002-2319	2002-2363	2002-2414	2002-2451
2002-2476	2002-2494	2002-2501	2002-2675
2002-2735	2002-2749	2002-2753	2002-2787
2002-2813	2002-2897	2002-2954	

2003-0026	2003-0039	2003-0040	2003-0080
2003-0119	2003-0145	2003-0169	2003-0181
2003-0222	2003-0243	2003-0284	2003-0288
2003-0331	2003-0361	2003-0458	2003-0459
2003-0481	2003-0618	2003-0626	2003-0667

2003-0680	2003-0700	2003-0774	2003-0777
2003-0782	2003-0785	2003-0788	2003-0814
2003-0835	2003-0843	2003-0860	2003-0889
2003-0893	2003-0896	2003-0918	2003-0935
2003-0948	2003-0957	2003-0961	2003-0965
2003-1039	2003-1030	2003-1041	2003-1052
2003-1059	2003-1169	2003-1279	2003-1304
2003-1616	2003-1306	2003-1335	2003-1336
2003-1338	2003-1339	2003-1418	2003-1436
2003-1489	2003-1491	2003-1494	2003-1497
2003-1530	2003-1591	2003-1640	2003-1732
2003-1738 ** (GFI Extension Cord)	2003-1761**	2003-1764 ** (Fuse Size Discrep)	2003-1855 ** (Lack of follow-up on RCIC -26)
2003-1859 ** (0028 Commitment not issued)	2003-1860 ** (MRFF missed)	2003-1861 ** (Vernon Tie ACE vs PRO)	2003-1870 ** (Approval Process)
2003-1871 ** (Missed Adverse Trend flag on ER)	2003-1872 ** (ERs not flagged as Human Performance)	2003-1878 ** (2003-1640 and 2003-1304 MRFF Determinations were Inadequate)	2003-1879** **(Failure to Initiate Corrective Action)
2003-1880** (Possible Adverse Trend in I&C Related Equipment Status Control Events)	2003-1884** (Part 5 MRFF Determination Does Not Match The Investigation Write-Up)	2003-1910** (Safety Relief Valve Programmatic Deficiencies)	

** - Event Reports issued as a result of NRC inspection activities

Safety Evaluation Reports & Exemptions

April 4, 1996, Letter, Request for Exemption From 10 CFR Part 50, Appendix R, Item III.L.3 Alternative and Dedicated Shutdown Capability

August 12, 1997, NRC Safety Evaluation Report, Exemption from Certain Requirements of Section III.G and III.L of Appendix R to 10 CFR Part 50

Non-Cited Violations (NCV) and Findings (FIN)

NCV 50-271/02-04-03
NCV 50-271/02-04-04
NCV 50-271/03-03-01

Training Change Requests (TCR)

TCR 03-0229
TCR 03-0230

Action Items/Regulatory Commitment

ER-2002-1791-01, Perform an Annual Self-Assessment of Lube Oil Analysis Program
ER-2003-05-01, Failure to take effective corrective actions
INF-2003-003-01, NRC Information Notice 2003-03
INF-2001-019-01, Improper Maintenance and Reassembly of Automatic Oil Bubblers
INF-2002-003-01 Highly Radioactive Particle Control Problems During Spent Fuel Pool Cleanout
INF-2002-004-01, Wire Degradation at Breaker Cubicle Door Hinges
INF-2002-005-01, Foreign material in standby liquid control (SLC) storage tanks, 1/21/02
INF-2002-012-01, Submerged Safety-Related Electrical Cables
INF-2002-022-01, NRC Information Notice 2002-022
INF-2002-025-01, Challenges to ability to provide prompt public notification and information, 9/10/02
OE-12059-01, Review of OE 12059 “Hot Particle Incident”
UND-2003-223-01, Install Plantronics headsets

Drawings

B191301, Sh325 and 325A, 4KV SWGR No. 3

Work Orders

01-004341-001	Disassemble/Inspect & Rebuild 1'' Crosby Relief Valve
03-003714-000	Perform Relief Valve Replacement and Expansion Testing of SR-13-26 in Accordance with OP 4261
03-002752-000	Perform Valve Replacement of SR-72-10B, Reactor Building Railroad Airlock Nitrogen Supply Relief Valve
03-002754-000	Perform Valve Replacement of SR-72-10A, Reactor Building Railroad Airlock Nitrogen Supply Relief Valve
03-002755-000	Perform Valve Replacement of SR-72-9B, Reactor Building Railroad Airlock Nitrogen Supply Relief Valve
03-002756-000	Perform Valve Replacement of SR-72-10B, Reactor Building Railroad Airlock Nitrogen Supply Relief Valve

Miscellaneous

AOR-23-305	Auxiliary Operator Requal Training Program Instructor Guide, Responses, Rev. 1, 05/03
BMO 2001-007	24 VDC Converters Dedication Testing Deficiencies
IN 2002-05	Foreign Material in SLC Storage Tanks, 1/17/02
IN 2002-25	Challenges to Licensees' Ability to Provide Prompt Public Notification & Information During an Emergency Preparedness Event, 8/26/02
LOR-23-305	Licensed Operator Requal Training Program Instructor Guide, Responses, Rev. 0, 02/03
RWP 00-00073	SFP and MS Pit Clean up Pre-Job Brief Package

Appendix R Boil-Off Sensitivity Study

ENVY Plan for Ever-Increasing Excellence in Organizational Performance

October 16, 1996, Memorandum, Subject: Timeliness operator actions in OP 3126 rev. 14

Vermont Yankee Switchgear Control Circuit Fuse Selection verification

03-0058379-00, Work Request, Relief Valve Expansion Testing Due to Setpoint Failure of SR-13-25 on July 8, 2003

Systems Engineering Memorandum SYSENG 2002-100, December 4, 2002 - Relief Valve Expanded Sample Testing Requirements

Vermont Yankee Technical Specifications, Section 4.6, Surveillance Requirements, Amendment 196

Vermont Yankee Sponsored Work Team Action Items List, dated August 26, 2003

AP 0028 Commitment IST-2003-002-31, Revise OP 4261to Incorporate Time Limits to Perform As-found Testing of Removed Relief Valves

Potentially Reportable Occurrence Report PRO-031491, ER-20031491; Relief Valve (V13-25) Not Tested In Accordance With IST Code

LIST OF ACRONYMS

AC	Alternating Current
CAP	Corrective Action Program
CFR	Code of Federal Regulations
EDG	Emergency Diesel Generator
ENVY	Entergy Nuclear Vermont Yankee
ER	Event Report
FIN	Finding
IMC	Inspection Manual Chapter
IPE	Individual Plant Examination
IST	Inservice Testing
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
P&ID	Piping and Instrumentation Drawing
PCT	peak centerline temperature
RCIC	Reactor Core Isolation cooling system
ROP	Reactor Oversight Process
TCR	Training Change Request
SDP	Significant Determination Process
SSCA	Safe Shutdown Capability Analysis
VY	Vermont Yankee
WO	Work Order