April 25, 2005

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

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000271/2005002

Dear Mr. Thayer:

On March 31, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Vermont Yankee Nuclear Power Station (VY). The enclosed report documents the inspection findings which were discussed on April 8, 2005, with you and 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 one self-revealing finding of very low safety significance (Green). If you contest any NCV in this report, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U.S. 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, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Vermont Yankee Nuclear Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the 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 05000271/2005002
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.	05000271/2005002
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Vermont Yankee Nuclear Power Station
Location:	320 Governor Hunt Road Vernon, Vermont 05354-9766
Dates:	January 1, 2005 - March 31, 2005
Inspectors:	David L. Pelton, VY Senior Resident Inspector Beth E. Sienel, VY Resident Inspector Frank J. Arner, Senior Reactor Inspector Steven Dennis, Senior Operations Engineer Lawrence L. Scholl, Senior Reactor Inspector Jeffery E. Josey, Reactor Inspector Jeffery A. Kulp, Reactor Inspector Thomas P. Sicola, Reactor Inspector
Approved by:	Clifford J. Anderson, Chief Projects Branch 5 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000271/2005002; 01/01/05 - 03/31/05; Vermont Yankee Nuclear Power Station; Equipment Alignment.

This report covered a 13-week period of inspection by resident inspectors and announced inspections by two regional senior reactor inspectors, a senior operations engineer, and three regional reactor inspectors. One Green finding was identified. The significance of most 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. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. A very low safety significance, self-revealing finding was identified because an instrumentation and control (I&C) technician did not follow instructions contained in a maintenance procedure. Rather than isolating the air supply to condensate demineralizer system valve SP-52-E1C, the technician inadvertently isolated the air supply to an adjacent valve which ultimately resulted in a small reactor vessel level and power perturbation.

This finding is greater than minor since it is associated with the Configuration Control-Operating Equipment Lineup attribute of the Initiating Events Cornerstone and because it affects the associated Cornerstone Objective to limit the likelihood of those events that upset plant stability during power operations. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Findings for At-Power Situations," the inspectors conducted an SDP Phase 1 screening. The inspectors determined that the finding is of very low safety significance since it does not contribute to the likelihood of a primary or secondary system loss of coolant accident, does not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions would not be available, and does not increase the likelihood of a fire or internal/external flood.

A contributing cause of this finding is related to the personnel subcategory in the cross-cutting area of human performance. The I&C technician did not apply the required self-checking techniques (i.e., did not read the valve identification tag to verify he was manipulating the correct valve) while attempting to close valve SP-52-E1C. (Section 1R04)

B. Licensee Identified Findings

None.

REPORT DETAILS

Summary of Plant Status

Vermont Yankee Nuclear Power Station began the inspection period at or near full power and, with the exception of power reductions for control rod pattern adjustments and turbine valve testing, 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 (71111.01)

1. Readiness for Seasonal Susceptibilities

a. Inspection Scope (two samples)

The inspectors reviewed measures established by Entergy for ensuring cold weather availability and operability of the condensate storage tank (CST) and associated emergency core cooling system supply (ECCS) valves and the emergency diesel generators (EDGs). The inspectors performed walkdowns of these systems and compared the current system alignments and operation to the requirements of Vermont Yankee Operating Procedure (OP) 2196, "Preparations for Cold Weather Operations"; OP 3127, "Natural Phenomena"; OP 2192, "Heating, Ventilating, and Air Conditioning System"; OP 0150, "Conduct of Operations and Operator Rounds"; and Technical Specifications (TS). Additionally, the inspectors reviewed condition reports (CRs) related to cold weather to ensure problems regarding the CST and EDGs were properly addressed for resolution.

b. Findings

No findings of significance were identified.

2. Readiness for Impending Adverse Weather Conditions

a. Inspection Scope (one sample)

On January 12, the inspectors reviewed actions taken by Entergy due to the severe cold weather (ambient outside temperature of less than -15 EF) in the vicinity of the plant. The inspectors reviewed procedure OP 3127, "Natural Phenomena," Appendix D, "Extreme Low Temperature Walkdown Check List" to ensure required plant walk-downs were being completed. The inspectors performed independent walkdowns of systems listed in procedure OP 3127 including high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), EDGs, and the instrument air (IA) system to determine the impact of severe cold weather on these systems.

b. Findings

No findings of significance were identified.

1R02 Evaluation of Changes, Tests, or Experiments (IP 71111.02)

a. Inspection Scope (twelve samples)

The inspectors reviewed a sample of six safety evaluations for the initiating events, barrier integrity and mitigating systems cornerstones to verify that changes and tests were reviewed and documented in accordance with 10 CFR 50.59. The inspectors assessed the adequacy of the safety evaluations through interviews with Entergy staff and review of supporting information, such as calculations, engineering analyses, design change documentation, the Updated Final Safety Analysis Report (UFSAR), the Technical Specifications, and plant drawings. In addition, the inspectors reviewed the Entergy administrative procedures that control the screening, preparation, and issuance of the safety evaluations to ensure that the procedures adequately implemented the requirements of 10 CFR 50.59, "Changes, Tests, and Experiments."

The inspectors reviewed a sample of six changes that Entergy had evaluated (using a screening process) and determined to be outside of the scope of 10 CFR 50.59; therefore, did not require a full safety evaluation. The inspectors performed this review to assess if Entergy's conclusions with respect to 10 CFR 50.59 applicability were appropriate. The sample of issues that were screened out with respect to safety evaluations included design changes and procedure changes.

The safety evaluations and screens 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 the attachment to this report.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope (five samples)

The inspectors performed five 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 to the standby alignment of equipment specified in Entergy'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. Additionally, the inspectors reviewed operator control room logs; the Piping and Instrumentation Drawing G191274, Condensate Demineralizers"; Licensed Operator Requalification Training

(LOR) Lesson Plan LOR-00-257, “Condensate Demineralizers”; the Vermont Yankee Operating License; CRs 2005-0502 and 2005-0504; Vermont Yankee Administrative Procedure (AP) 0095, “Plant Procedures”; AP 0021, “Work Orders”; and Work Orders (WOs) 04–0107, “Replace Actuator Positioner,” and 04-4182, “Rebuild Actuator.” Finally, the inspectors interviewed control room operators and interviewed I&C personnel. The inspectors verified the alignment of the following systems:

- The “B” train of the standby liquid control (SLC) system during planned maintenance on the “A” train of SLC on January 10, 2005;
- The “A” train of residual heat removal service water (RHRSW) during planned maintenance on the “B” train of RHRSW on January 19;
- The diesel driven fire pump during planned maintenance on the “A” service water pump on January 31;
- The condensate demineralizers following the misalignment of the “B” demineralizer during planned maintenance activities on February 17 (see finding below); and
- The “A” emergency diesel generator (EDG) during planned maintenance on the “B” EDG on March 23.

b. Findings

Introduction: A Green, self-revealing finding was identified because an I&C technician did not follow instructions contained in a maintenance procedure. Rather than isolating the air supply to condensate demineralizer system valve SP-52-E1C, the technician inadvertently isolated the air supply to an adjacent valve which ultimately resulted in a small reactor vessel level and power perturbation.

Description: On February 14, 2005, operators removed the “C” condensate demineralizer from service to support planned maintenance on its air-operated discharge isolation valve, SP-52-E1C. The applicable maintenance procedure (WO 04-0107) included instructions to tag out the demineralizer including isolating the air supply to valve SP-52-E1C. Rather than isolating the air supply to valve SP-52-E1C, an I&C technician inadvertently isolated the air supply to an adjacent valve, SP-52-E1B. Valve SP-52-E1B is the “B” condensate demineralizer discharge isolation valve. Isolating the air supply to valve SP-52-E1B caused the valve to close resulting in an increase in condensate demineralizer system differential pressure and the automatic opening of the condensate demineralizer bypass valve, SB-52-8. The opening of valve SB-52-8 resulted in an increase in condensate and feedwater system flow and an increase in reactor vessel level of approximately one inch. The feedwater level control system responded automatically to the increasing vessel level, returning level to its pre-transient value in approximately 10 minutes. The increase in vessel level caused an increase in vessel pressure, a reduction in void fraction, and a momentary increase in reactor power from 1592 megawatts to 1611 megawatts (or approximately 1.13 percent).

The inspectors determined that the momentary increase in reactor power did not result in the 8-hour average thermal power exceeding the Vermont Yankee Operating License limit of 1593 megawatts.

This issue was entered into Entergy's corrective actions program (CRs 2005-0502 and 2005-0504). In the associated Apparent Cause Evaluation, Entergy determined that the I&C technician did not apply the required self-checking techniques (i.e., did not read the valve identification tag to verify he was manipulating the correct valve) while attempting to close valve SP-52-E1C. Overconfidence in his knowledge of the condensate demineralizer system along with being overly focused on completing the work led to a lapse in judgement regarding the need to use self-checking.

Analysis: The performance deficiency associated with this finding was that an I&C technician did not follow instructions contained in a maintenance procedure. Procedure AP 0095, "Plant Procedures," states that plant personnel shall adhere to the requirements of applicable Vermont Yankee procedures, including maintenance procedures. The finding is greater than minor since it is associated with the Configuration Control-Operating Equipment Lineup attribute of the Initiating Events Cornerstone and because it affects the associated Cornerstone Objective to limit the likelihood of those events that upset plant stability during power operations. Specifically, an I&C technician did not follow instructions contained in WO 04-0107. Rather than isolating the air supply to condensate demineralizer discharge isolation valve SP-52-E1C, the technician inadvertently isolated the air supply to an adjacent valve which ultimately resulted in a small reactor vessel level and power perturbation. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Findings for At-Power Situations," the inspectors conducted an SDP Phase 1 screening. The inspectors determined that the finding is of very low safety significance (Green) since it does not contribute to the likelihood of a primary or secondary system loss of coolant accident, does not contribute to both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions would not be available, and does not increase the likelihood of a fire or internal/external flood.

A contributing cause of this finding is related to the personnel subcategory in the cross-cutting area of human performance. The I&C technician did not apply the required self-checking techniques (i.e., did not read the valve identification tag to verify he was manipulating the correct valve) while attempting to close valve SP-52-E1C.

Enforcement: No violation of NRC regulatory requirements was identified because the I&C technician's incorrect action was associated with non-safety related equipment. **FIN 05000271/2005002-01, Technician Did Not Follow Non-Safety Related Maintenance Procedure Which Resulted in a Reactor Water Level and Power Perturbation.**

1R05 Fire Protection (71111.05)

8. Annual Plant Fire Drill Inspection (71111.05A)

a. Inspection Scope (one sample)

On January 11, the inspectors observed the performance of a fire drill involving a simulated fire in an overhead cable tray on the 252 foot elevation of the reactor building (above the control rod drive valve aisle). The inspectors evaluated the readiness of the fire brigade against the drill objective acceptance criteria established within the drill scenario including:

- Donning of protective clothing;
- Use of self-contained breathing apparatus 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.

2. Routine Fire Area Inspections (71111.05Q)

a. Inspection Scope (nine samples)

The inspectors identified nine fire areas and zones important to plant risk based on a review of Entergy's Safe Shutdown Capability Analysis for Vermont Yankee and the Individual Plant Examination External Events (IPEEE). The inspectors toured these plant areas important to safety in order to verify the suitability of Entergy's control of transient combustibles and ignition sources, and to evaluate 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 nine fire areas (FAs) and fire zones (FZs) were inspected:

- Control room (FZ-1);
- Cable vault (FZ-2);
- Battery room (FZ-3);
- East switchgear room (FA-4);
- West switchgear room (FA-5);
- "A" EDG room (FA-8);
- "B" EDG room (FA-9);
- Main and auxiliary transformers (no specific fire designation); and

- Start-up transformers (no specific fire designation).

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope (one sample)

The inspectors reviewed Entergy's established flood protection barriers and procedures for coping with internal flooding in the HPCI room including Vermont Yankee Off-Normal Procedure (ON) 3148, Loss of Service Water; ON 3158, Reactor Building High Area Temperature/Water Level; and Emergency Operating Procedure (EOP) 4, Secondary Containment and Radioactive Release. The inspectors reviewed internal flooding and HPCI system design information contained in Entergy's IPEEE, the UFSAR, and in the Internal Flooding and HPCI Design Basis Documents (DBDs). Finally, the inspectors performed a walkdown of the HPCI room to ensure equipment and structures needed to mitigate an internal flooding event were as described in the IPEEE and the DBD and discussed observations with design and system engineering personnel.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope (one sample)

On February 7, the inspectors observed a simulator examination for one operating crew to assess the performance of the licensed operators and the ability of Entergy's Training and Operations Department staff to evaluate licensed operator performance. Crew performance was evaluated during a simulated turbine trip with a failure to scram, failure of fast transfer, and failure of an EDG to start.

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 Entergy management expectations and guidelines as presented in the following documents:

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

The inspectors verified that the crew completed the critical tasks identified in the simulator evaluation guide. The inspectors also compared simulator configurations with actual control board configurations. For any weaknesses identified, the inspectors observed Entergy evaluators to verify that they also noted the issues to be discussed with the crew.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope (three samples)

The inspectors performed two issue/problem-oriented inspection of actions taken by Entergy in response to RCIC system high suction pressure following a surveillance and torus room cement wall water intrusion. Additionally, the inspectors performed one system/function performance history-oriented inspection of the augmented offgas (AOG) system. The inspectors reviewed the UFSAR, system DBD, operating procedures, system’s maintenance rule scoping document, list of historical condition reports written for AOG system problems, applicable maintenance rule functional failure determinations, and corrective actions taken in response to the equipment problems in accordance with station procedures and the requirements of 10 CFR 50.65.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation (71111.13)

a. Inspection Scope (seven samples)

The inspectors evaluated on-line risk management for six planned maintenance activities and one emergent condition. 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 not significantly increase plant risk. The inspectors compared reviewed items and activities to requirements listed in procedures AP 0125, "Plant Equipment," and AP 0172, "Work Schedule Risk Management - Online." The inspectors reviewed the following on-line work activities:

- Planned limiting condition for operation (LCO) outage of the “A” train of the SLC system;
- Planned LCO outage of the “B” train of the RHRSW system to install a minor modification on the “B” and “D” pumps’ motor cooling lines;
- Planned LCO outage of the “A” train of RHRSW to install a minor modification on the “A” and “C” pumps’ motor cooling lines;
- Planned LCO outage of the “A” service water pump;
- Emergent online risk condition during activities to reseat the RCIC discharge check valve, V-22;
- Planned LCO outage of cooling tower 2-1; and
- Planned LCO outage of the “B” EDG.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions (71111.14)

a. Inspection Scope (one sample)

The inspectors assessed control room operators performance during a January 13 planned power reduction to approximately 60 percent power in support of rod pattern adjustments and main turbine control valve/stop valve testing. The inspectors evaluated the adequacy of personnel performance, procedure compliance, and use of the corrective action process against the requirements and expectations contained in the following station procedures:

AP 0151, “Responsibilities and Authorities of Operations Department Personnel”;
AP 0153, “Operations Department Communication and Log Maintenance;”
DP 0166, “Operations Department Standards;”
OP 0105, “Reactor Operations;” and
OP 2403, “Control Rod Sequence Exchange with the Reactor Online.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (six samples)

The inspectors reviewed five operability determinations prepared by Entergy. The inspectors evaluated operability determinations against the requirements and guidance contained in NRC Generic Letter 91-18, "Resolution of Degraded and Nonconforming Conditions," as well as procedure ENN-OP-104, "Operability Determinations." The inspectors verified the adequacy of the following evaluations of degraded or non-conforming conditions:

- Lack of periodic testing of EDG room tornado dampers;
- Potential non-conservative TS instrument setting for automatic swap of condensate storage tank suction;
- Failure of several service water system heat trace circuits;
- Cooling tower 2-1 structural damage;
- "A" EDG pressure control valve (73A) diaphragm leakage greater than 60 drops per minute; and
- HPCI system non-conservative design basis post-accident injection pressure.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope (one sample)

The inspectors reviewed the cumulative effect of operator workarounds on the reliability, availability, and potential mis-operation of systems and the potential to affect the ability of operators to respond to plant transients and events. The inspectors reviewed identified operator burdens, control room deficiencies, disabled or illuminated control room alarms, and component deviations and discussed them with responsible operations personnel to ensure they were appropriately categorized and tracked for resolution. In addition, in-plant and control room tours were performed to identify any workarounds not previously identified in accordance with procure DP 0166, "Operations Department Standards."

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

2. Annual Review (71111.17A)

a. Inspection Scope (two samples)

The inspectors reviewed two permanent plant changes to verify that the design bases, licensing bases, and performance capability of risk significant SSCs had not been degraded. These reviews focused on the impact the permanent plant change had on operator actions and changes to key safety functions. The selection of permanent plant changes for review was based on risk insights for the plant and included SSCs associated with the initiating events, barrier integrity 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, engineering requests, the UFSAR and TS.

The inspectors verified that selected attributes were consistent with the design and licensing bases. These attributes included where applicable, component safety classification, energy requirements supplied by supporting systems, seismic qualification, instrument set-points, uncertainty calculations, electrical loads analysis, and equipment environmental qualification. Design assumptions were reviewed to verify that they were technically appropriate and consistent with the UFSAR. For selected modifications, the 50.59 screens or evaluations were reviewed as described in section 1R02 of this report. 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 SSC would function properly. The inspectors evaluated how effective the changes were in resolving identified problems. A listing of documents reviewed is provided in the attachment to this report. The inspector reviewed the following permanent plant modifications:

- MM 2004-001, “Removal of remote position indication for SLC manual isolation valve”
- Revisions made to procedure ON 3155, “Loss of Auto Transformer”

The sample associated with revisions made to procedure ON 3155, “Loss of Auto Transformer,” was selected, in part, due to the fact that a Green NCV had been previously identified regarding the adequacy of procedure ON 3155. The NCV was originally documented in NRC Inspection Report 2004-008 as NCV 05000271/2004008-02, “Procedures for Assessing Off-Site Power Operability.” In addition to the inspection attributes discussed above, the inspectors also reviewed this sample in terms of corrective actions taken by Entergy to address this NCV. Specifically, the inspectors reviewed condition reports generated from the NCV, training program instructor guides, procedure changes, administrative reviews, operations department most probable cause reports, and 10 CFR 50.59 evaluations. The effectiveness of the corrective actions taken was evaluated by comparing the specific items noted in Inspection Report 2004-008 against completed and proposed actions. The inspectors evaluated the condition reports generated to ensure the concerns listed in the NCV were properly translated into the corrective action program. Entergy's actions to address the problems listed were then evaluated by examining changes to procedures dealing with the system in question, and reviewing the associated 10 CFR 50.59 screens and evaluations. Entergy also provided training documentation for senior reactor operator (SRO) re-licensing which addressed the changes made to the procedures. The inspectors also conducted walkdowns of the control room panels and interviewed control room operators and SROs to determine if the documented changes had been fully implemented and to test the operator's knowledge and/or understanding of these changes.

b. Findings

No findings of significance were identified.

3. Biennial Review (71111.17B)

a. Inspection Scope (six samples)

The inspectors reviewed six permanent plant changes to verify that the design bases, licensing bases, and performance capability of risk significant SSCs had not been degraded. The selection of permanent plant changes for review was based on risk insights for the plant and included SSCs associated with the initiating events, barrier integrity 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, engineering requests, the UFSAR and TSs.

The inspectors verified that selected attributes were consistent with the design and licensing bases. These attributes included where applicable, component safety classification, energy requirements supplied by supporting systems, seismic qualification, instrument set-points, uncertainty calculations, electrical loads analysis, and equipment environmental qualification. Design assumptions were reviewed to verify that they were technically appropriate and consistent with the UFSAR. For selected modifications, the 50.59 screens or evaluations were reviewed as described in section 1R02 of this report. 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 SSC would function properly. The inspectors evaluated how effective the changes were in resolving identified problems. A listing of documents reviewed is provided in the attachment to this report. The inspector reviewed the following permanent plant modifications:

- MM 2000-002, "Recirculation Pump Speed Control Modification"
- MM 2000-029, "Main Station Battery Chargers Modification"

- MM 2001-031, “Plug reactor building railroad airlock floor drains”
- MM 2002-022, “B RHRSW subsystem motor bearing line”
- MM 2003-005, “Upgrade to NSSS and TAP supports”
- MM 2003-032, “Furmanite seal on flange connection (TM conversion)”

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope (six samples)

The inspectors reviewed six post-maintenance test (PMT) activities on risk-significant systems. The inspectors either directly observed or reviewed completed PMT documentation to verify that the test data met the required acceptance criteria contained in the TS, UFSAR, and in-service testing program. Where testing was directly observed, the inspectors verified that installed test equipment was appropriate and controlled and that the test was performed in accordance with applicable station procedures. The inspectors also ensured that the test activities were adequate to ensure system operability and functional capability following maintenance; that systems were properly restored following testing; and that any discrepancies were appropriately documented in the corrective actions program. The inspectors reviewed the following PMT activities:

- Testing performed following planned maintenance on the “A” SLC pump;
- Testing performed following the installation of a minor modification on the “B” RHRSW pump;
- Testing of the “F” average power range monitor (APRM) following troubleshooting activities;
- Testing performed following planned maintenance on the “A” service water pump;
- Testing performed following the planned cable replacement from the Vernon tie transformer to 3V4 breaker; and
- Testing following planned maintenance on the “B” EDG air start solenoid valves.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (six samples)

The inspectors observed surveillance testing to verify that each test was performed in accordance with the written procedure, the acceptance criteria specified for each test was consistent with the requirements of the TS and UFSAR requirements, test data was complete and met procedural requirements, and the system was properly returned to service following the completion of testing. The inspectors observed selected pre-job briefings supporting testing. The inspectors also verified that discrepancies identified were entered into the corrective action program. The inspectors verified that testing in accordance with the following procedures met the above requirements:

- OP 4113, "Main and Auxiliary Steam System Surveillance," Section A, "MSIV Full Closure Timing and RPS Relay Actuation Function Test";
- OP 4120, "High Pressure Coolant Injection System Surveillance," Section A, "Pump Operability and Flow Rate Test";
- OP 4123, "Core Spray System Surveillance," Section C, "Core Spray Pump Operability Test";
- OP 4160, "Turbine Generator Surveillance," Section V.A.6, "Bypass Valve Test";
- OP 4181, "Service Water/Alternate Cooling System Surveillance," Section D, "Service Water Pump Capacity Test"; and
- OP 4424, "Control Rod Scram Time Testing," Section B, "Single Rod Scram Using ERFIS Data Collection."

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope (one sample)

The inspectors reviewed temporary modification (TM) 2004-036, "Installation of Jumper in SLC Pump P-45-1A MCC [motor control center] Cubicle to Maintain Pump Operability," to ensure that the modification did not adversely affect the availability, reliability, or functional capability of any risk-significant structures, systems, or components. The inspectors compared the information in the TM package to Entergy's TM requirements contained in AP 0020, "Control of Temporary and Minor Modifications." The inspectors observed the installation of the TM and subsequently walked down the TM to verify that required tags and markings were applied and that the TM was properly maintained.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope (one sample)

On February 7, 2005, the inspectors observed an operating crew evaluate a simulator-based event using the station emergency action levels (EALs) during licensed operator requalification training activities. The inspectors discussed the performance expectations and results with the lead instructor. 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 0153, "Operations Department Communications and Log Maintenance";
- AP 0156, "Notification of Significant Events";
- AP 3125, "Emergency Plan Classification and Action Level Scheme";
- DP 0093, "Emergency Planning Data Management";
- OP 3540, "Control Room Actions During an Emergency"; and
- Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

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 they were being entered into Entergy's corrective action system at an appropriate threshold and that adequate attention was being given to timely corrective actions. Additionally, in order to identify repetitive equipment failures and/or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into Entergy's corrective action program. This review was accomplished by reviewing selected hard copies of condition reports (a listing of CRs reviewed is included in the Attachment to this report) and/or by attending daily screening meetings.

b. Findings

No findings of significance were identified.

2. Annual Sample Review of Non-Conservative RCIC System Discharge Pressure

a. Inspection Scope (one sample)

CR 2004-2815 was initiated in August 2004 to address an identified non-conservative value for required post-accident discharge pressure in the design calculation for the RCIC system. The inspectors selected CR 2004-2815 for review because Entergy identified this non-conservatism while addressing questions raised during a 2004 NRC engineering team inspection and because the RCIC system is a risk-significant system. This CR was reviewed to verify that an appropriate extent of condition review was performed, all technical aspects of the issue were properly evaluated, and appropriate corrective actions were assigned and prioritized. The inspectors also reviewed the design basis documents (DBDs) for both the RCIC and HPCI systems and interviewed Design Engineering personnel.

b. Findings and Observations

No findings of significance were identified. However, the inspectors identified that Entergy did not take appropriate action in August 2004 to address a potential HPCI system operability issue identified during the extent of condition review for CR 2004-2815. When Entergy initially identified the non-conservative value for RCIC system discharge pressure, they immediately developed an operability determination (OD) which demonstrated the continued operability of the RCIC system. During their extent of condition review, Entergy determined that the HPCI system was also subject to the same discharge pressure non-conservatism as the RCIC system. However, Entergy did not take action to develop an OD to support the continued operability of the HPCI system, as required by Entergy Procedure ENN-OP-104, "Operability Determinations." Additionally, Entergy Management performed a closure review of CR 2004-2815 in November 2004. At that time, Entergy Management concluded that there were no further operability concerns associated with the identified RCIC and HPCI system non-conservatisms despite the fact that an OD was not developed supporting the continued operability of the HPCI system. Entergy subsequently entered this issue into their corrective actions program (CR 05-0547) and developed an OD which supported the continued operability of the HPCI system.

The inspectors reviewed CR 05-0547 and the associated apparent cause evaluation. Entergy's failure to develop an OD to address the continued operability of the HPCI system constitutes a violation of minor significance. The inspectors considered the issue to be of minor significance because the HPCI system remained operable and the failure to develop an OD to demonstrate the continued operability of the HPCI system was entered into Entergy's corrective actions program. As such, the finding is not subject to enforcement action in accordance with Section VI of the NRC's Enforcement Policy.

3. Annual Sample Review of RCIC Operation From the Alternate Shutdown Panel

a. Inspection Scope (one sample)

The inspectors selected this item for review because a previous inspection finding (Non-Cited Violation (NCV) 05000271/2004008-08) identified that the demonstrated time for operators to start the RCIC system from the alternate shutdown panel had increased. Also, operation at proposed power uprate power levels would reduce the time available before reactor vessel level would decrease to below the top of active fuel, and therefore reduce the time available to start the RCIC system to prevent core uncovery. Additionally, the increased RCIC startup times had not been communicated to Engineering Department personnel and as a result the Safe Shutdown Capability Analysis Report was not properly updated. The inspectors reviewed the associated CRs (CRs 2004-2614 and 2004-2552) to evaluate the adequacy of the apparent cause and extent-of-condition evaluations and to ensure appropriate corrective actions had been identified and prioritized. The inspectors interviewed the cognizant Operations Department staff and performed a timed walkdown of the operator actions contained in procedure OP 3126, "Shutdown Using Alternate Shutdown Methods," to verify that actions taken were effective in reducing the RCIC startup times.

b. Findings and Observations

No findings of significance were identified. The inspectors found that the apparent cause evaluation of the issue was appropriate and that actions taken to address the RCIC startup were comprehensive and effective. During the timed procedure OP 3126 walkthrough, the inspectors confirmed that the RCIC system could be started from alternate operating panels in approximately 14 minutes.

However the inspectors identified that Entergy had not included all of the corrective actions identified during its extent of condition review into its corrective action program. During the extent of condition review for CR 2004-2552, Entergy developed a comprehensive list of operator actions that were required to be completed within a time that was determined from a calculation. Entergy had established Corrective Action (CA) CA-14 to evaluate the list. The evaluation identified that 12 operator actions on the list require additional evaluation to support current plant operation. Additionally, the disposition for a number of other items in the comprehensive list of operator actions indicated the need for additional review or evaluation to support future changes in plant operation. For example, several operator actions were identified that could be affected by a proposed power uprate or alternate source term activities. Entergy closed out CA-14 but did not assign a CA for the evaluation of these items.

The licensee subsequently formalized the identified additional CAs for the items requiring additional evaluation, updated the disposition descriptions on the operator action list, and initiated CR 2005-0884 to document the failure to initially enter the additional CAs into the CAP.

4OA3 Event Followup

4. (Closed) Licensee Event Report (LER) 05000271/2004004-00: Standby Liquid Control Discharge Relief Valves Fail Set Point Testing.

Vermont Yankee has had a history of SLC relief valves lifting outside the TS-required set point range (1400 to 1490 psig) during surveillance testing. These testing failures have recurred over several years, affected both relief valves in the system, and have occurred when tested during outages as well as when tested mid-cycle. Inspectors had previously identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," because ENtergy had not determined the cause(s) of previous SLC relief valve failures and for not assigning appropriate CAs to prevent recurrence. Entergy performed an evaluation of the history of SLC relief valve testing failures and determined that although many issues could have contributed to the testing failures (e.g., test stand setup, procedure quality, test stand operator training, etc.), no specific cause could be determined. Entergy also concluded that the history of SLC relief valve testing failures was not reportable under 10 CFR 50.73, "Licensee Event Report System."

While reviewing recent, additional SLC relief valve testing failures, the inspectors challenged Entergy regarding the reportability of these multiple test failures under 10 CFR 50.73(a)(2)(i)(B), "Operation or Condition Prohibited by Technical Specifications." The inspectors believed the history of SLC relief valve testing failures was reportable as discussed in NUREG-1022, "Event Reporting Guidelines for 10 CFR 50.72 and 50.73," Section 3.2.2 which states that the existence of similar discrepancies in multiple valves is an indication that the failures may well have arisen over time which would then be reportable. The inspectors held several meetings with Engineering and Licensing Department Management to discuss the SLC relief valve testing failure history, the reportability aspects, and the continued operability of the SLC system.

Based on discussions with the inspectors and on further review of NUREG-1022, Entergy agreed that the history of SLC system relief valve testing failures did represent a condition prohibited by the TS that should have been reported to the NRC via a licensee event report (LER). Entergy entered the issue into their CAP (CR 2004-3174) and developed an OD in which they concluded that the SLC system remained operable based on the fact that none of the testing failures would have prevented the SLC system from performing its design functions. On December 9, 2004, Entergy issued LER 05000271/2004004-00, "Standby Liquid Control Discharge Relief Valves Fail Set Point Testing."

The inspectors performed a review of the LER and associated root cause analysis. CAs taken by Entergy included a historical review (considering both internal and external operating experience) of previous relief valve failures, work with an independent testing laboratory to determine a more appropriate testing methodology for these relief valves, procurement of improved testing equipment and enhancement of applicable testing procedures, training sessions with management to increase sensitivity to safety-related relief valve performance issues, and submittal of a TS amendment request to change the SLC TS surveillance requirements to evaluate the performance of the relief valves in accordance with the in-service testing program; which is consistent with the relief valve testing methodology used with other safety systems.

Although no findings of significance were identified, Entergy's failure to report the history of SLC relief valve testing failures via an LER (until prompted by the inspectors) constitutes a violation of minor significance. The inspectors considered the issue to be of minor significance because the SLC system remained capable of performing all required design functions, there was no other impact on the regulatory process, and the issue was entered into Entergy's corrective actions program. As such, the finding is not subject to enforcement action in accordance with Section VI of the NRC's Enforcement Policy. This LER is closed.

4OA4 Cross-Cutting Aspects of Findings

Section 1R04 discusses a Green finding with a contributing cause related to the personnel subcategory in the cross-cutting area of human performance. The I&C technician did not apply the required self-checking techniques (i.e., did not read the valve identification tag to verify he was manipulating the correct valve) while attempting to close valve SP-52-E1C. As a result he operated the wrong valve and caused a small reactor water level and power perturbation.

4OA5 Other Activities

1. (Closed) Unresolved Item (URI) 05000271/2004006-01: Adequacy of Testing for Emergency Diesel Generator Room Tornado Dampers.

The "A" and "B" EDG rooms each have one tornado damper (three louvers per damper) installed to provide a pressure relief path to prevent the EDG room walls from collapsing during a design basis tornado event. Preventive maintenance (PM) is performed on these dampers and includes a visual inspection, lubrication, and manual cycling open and closed every 18 months.

Although the PM applied to the dampers provided evidence that the dampers would open manually, the inspectors questioned whether manually cycling the dampers open and closed was sufficient to demonstrate that the dampers would open within the design basis differential pressure range (0.28 to 0.35 pounds per square inch differential (psid)) specified in Entergy engineering design change request (EDCR) 97-407.

Entergy initially entered the issue into the CAP (CR 2004-3293) and developed an OD which demonstrated that the EDG room tornado dampers would perform their design function if called upon. Entergy has since performed testing of the EDG room tornado dampers. Using a calibrated force gauge, a pulling force was applied to the damper spring mechanism to determine the force necessary to cause the dampers to open. Once the opening force value was determined, engineers translated this value to an equivalent differential pressure across the damper louvers. Both the "A" and "B" EDG room dampers opened within the design basis differential pressure range specified in EDCR 97-407.

The inspectors determined that this testing was appropriate to validate that the dampers would open within the design basis differential pressure range and that the test met the testing requirements of 10 CFR 50, Appendix B, Criterion XI, “Test Control.” Entergy plans to enhance their current 18-month PM performed on the EDG room tornado dampers to include the opening force test described above. This URI is closed.

4OA6 Meetings, Including Exit

Resident Exit

On April 8, the resident inspectors presented the inspection results to Mr. Jay Thayer and members of his staff. The inspectors asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

Annual Assessment Meeting

A public meeting was conducted with Mr. Jay Thayer, Vermont Yankee Site Vice President, and other members of Entergy’s staff at the Governor Hunt House in Vernon, Vermont on March 17, 2005. The meeting was held to discuss the Annual Assessment of the Vermont Yankee Nuclear Power Station. The assessment letter and slides used during the meeting can be found in ADAMS (Accession Numbers ML050610121 and ML050690139, respectively).

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

J. Callaghan, Design Engineering Manager
P. Corbett, Maintenance Manager
J. Dreyfuss, Project Engineering Manager
J. Devincentis, Licensing Manager
M. Gosekamp, Superintendent of Operations Training
M. Hamer, Licensing
M. Metell, Engineering
W. Maguire, General Plant Manager
J. Thayer, Site Vice President
C. Wamser, Operations Manager
R. Wanczyk, Director of Nuclear Safety

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000271/2005002-01	FIN	Technician Did Not Follow Non- Safety Related Maintenance Procedure Which Resulted in a Reactor Water Level and Power Perturbation (Section 1R04)

Closed

05000271/2004004-00	LER	Standby Liquid Control Discharge Relief Valves Fail Set Point Testing (Section 4OA3.1)
05000271/2004006-01	URI	Adequacy of Testing for Emergency Diesel Generator Room Tornado Dampers (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

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

10 CFR 50.59 Safety Evaluations

Safety Evaluation (SE) 2000-024, "Modification to support installation of 4 inch ss line to cross-tie RHRSW
SE 2000-004, "Incorporating new main steam line break analysis at hot standby condition"
SE 2004-001, "Removal of remote position indication for SLC manual isolation"
SE 2000-002 "Evaluation for Recirculation Pump Speed Controls Modification"
SE 2003-004, "Revise OT 3121, Inadvertent opening of a relief valve"

SE 2000-029, “Main station battery chargers modification”

10 CFR 50.59 Safety Evaluation Screens

Procedure OP 4114, “Standby Liquid Control Surveillance”
MM 2004-015, “Standby Liquid Control relief valve tolerances”
MM 2002-029, “RCIC steam supply line drain pot level switch replacement”
ENN-LI-101 50.59, “Screen Control Form for OP 2141, Rev. 17"
ENN-LI-101 50.59, “Screen Control Form for OP 2140, Rev. 17"
ENN-LI-101 50.59, “Screen Control Form for OP 3155, Rev. 09"

Section 1R17: Permanent Plant Modifications; Annual Review and Biennial Reviews

Design References and Calculations

Vermont Yankee Calculation (VYC) 2148, “Pipe stress analysis for the cross tie”
VYC-1910, “Main steam line break at power”
VYC-1828, “Reactor building masonry wall review for HELB [high energy line break] loadings”
VYC-2041, “Reactor building internal wall differential pressure and EQ [environmental qualification] data”
VYC-2292, “Evaluation of torus attached pipe supports for power uprate conditions”
5319-RCIC-HD63C, “VY nuclear power plant pipe wall stress calculations”
Basis for Maintaining Operability (BMO) 99-08, “Basis for maintaining operability”
BMO 02-06, “RHRSW pump operation with potentially degraded cooling”
P&ID G-191176, Sheet 1
Station drawing 5920-3594

Procedures

OP 0150, Rev 40 – Conduct of Operations and Operator Rounds
OP 2140, Rev 26 – 345 kV Electrical System
OP 2141, Rev 17 – 115 kV Switchyard
VYAPF 0097.01 for OP 2140, Rev. 26 – Preapproved LPC form
VYAPF 0097.01 for ON 3155, Rev. 09 – Preapproved LPC form
AP 0096, Cross-Discipline review Checklist for OP 2141, Rev. 17
AP 0096, Cross-Discipline review Checklist for OP 2140, Rev. 26
AP 0096, Cross-Discipline review Checklist for ON 3155, Rev. 09

Engineering Requests

ER 04-0502, Remove Valve Indication From Component
ER 04-0563, Raise Setpoint of SR-11-39A&B
ER 04-0604, Provide Jumpers Capability for SLC Injection Control Switch
ER 02-0760, Implementation of MM 2002-022

Other

Licensed Operator Requal Training Program Instructor Guide (LOR) 24-505
Vermont Yankee Administrative Procedure Form (VYAPF) 0097.01 for OP 2140, Rev. 26 – Preapproved Limited Procedure Change (LPC) form
VYAPF 0097.01 for ON 3155, Rev. 09 – Preapproved LPC form
ENN-LI-100 Process Applicability Procedure for OP 2141, Rev. 17
ENN-LI-100 Process Applicability Procedure for OP 2140, Rev. 26
ENN-LI-100 Process Applicability Procedure for ON 3155, Rev. 09
TE-2000-09 Transient Analysis Impact of VYDC 2000-02
Procedure EMI98P0200/1, Electromagnetic Interference Test Procedure for Class 1E Dedication and Qualification of Moore Products Co. Digital Controllers
Test SHAT00C0930/1, "Software/Hardware Acceptance Test for Moore 353 Recirc Motor Generator Set Speed Controller"

Section 4OA2: Review of Problem Identification and Resolution

Condition Reports

2000-1509	Documentation discrepancies discovered during NRC inspection
2000-1578	Inadequate 50.59(a)(1) screening associated with SCR98C-068
2000-1596	Tech. Spec. Low CST water level trip settings
2001-0692	AOG exhaust pre-filters found damaged (adverse trend)
2003-2059	Lack of documentation and action plans to address goals
2004-0040	AOG has exceeded it's performance criteria of $\leq$ 2 MRFFs per 3 year period
2004-0692	Possible non-conservative AOG system valve failure positions
2004-1829	AOG performance evaluation not completed in the time frame required by ENN-DC-121
2004-2803	NRC assessment ON-3155 potential weakness issues
2004-2804	Lack of clear procedural guidance on declaring the115kv immediate access power source inoperable
2004-3042	AOG HEPA pre-filter found detached from filter rack
2004-3519	AOG exhaust fan EF-100-1A found tripped
2004-3763	AOG pre-filters discovered "blown off." This is a repeat problem
2004-3802	AOG compressor and fans HE-109-1B tripped causing charcoal bed temperature to rise
2004-2923	Operations performance indicators- red
2004-3160	Long standing operational work-arounds and burdens not included in Op. Aggregate Index
*2004-3174	SLC pump discharge relief valve lift test failures
2004-3689	Heat trace freeze protection problems
2005-0002	"A" EDG air compressor air leak
2005-0022	RV chloride >1ppb
2005-0029	Circuit breaker in MCC-8B–7C appears to have exceeded its EQ installed life
2005-0038	Hinsdale NH fire station siren inoperable
2005-0079	Overstress condition on single failure proof crane trolleys
2005-0132	APRM-F power supply failure

2005-0196	Questions on diesel fuel analysis
2005-0198	Thermal overload relay installed without adequate engineering review
2005-0205	Questions on secondary containment capability test
2005-0243	Low CST enclosure temperature
*2005-0319	Timeliness of submittal of Technical Specification proposed change
2005-0332	4KV breaker undergoing maintenance testing failed to completely charge
2005-0350	Potential for CST tank vent blockage during cold weather
2005-0370	Equipment deficiencies result in tank overflow
2005-0372	Unable to clear HRA areas during RWCU resin transfer
2005-0444	EDG “A” SW pressure control valve leak
2005-0451	Torus lower level west walls starting to show deterioration
2005-0502	Wrong valve was isolated
2005-0504	Reactor power excursion during normal 100% power operation
2005-0512	Fitness for Duty random testing was not all inclusive
*2005-0547	Follow up CR to NRC assessment RCIC CR
2005-0555	Valve packing leak RHR V10-25A
2005-0586	Fuel oil storage tank level indicator tube empty
2005-0643	RCIC suction pressure high alarm came in repeatedly
2005-0675	RCIC MOV 13-21 leaks past seat
2005-0697	ORAM “Schedule” module is not reflecting actual plant work schedule
2005-0710	Degraded structural members found in cooling tower cell CT 2-1
2005-0763	Fan blade leading edge protection loosening on CT 2-1 fan blades
*2005-0884	Additional corrective actions not generated in PCRS
2005-0931	Core spray valve 26A did not fully close when performing CS surveillance
2005-0946	Core spray “A” full flow test valve, V14-26A, inoperable

*Inspector-identified issues

Section 4OA2.2: Annual Sample Review of RCIC Operation From the Alternate Shutdown Panel

Procedures

AP 0096	Procedure Development, Review, Issuance, and Cancellation
AP 0097	Limited Procedure Changes
OP 3126	Shutdown Using Alternate Shutdown Methods
LOR 24-405-2	Licensed Operator Requal Training Lesson Plan

LIST OF ACRONYMS

ADAMS	Automated Document Access Management System
AOG	Augmented Off-Gas
AP	Vermont Yankee Administrative Procedure
APRM	Average Power Range Monitor
BMO	Basis for Maintaining Operability
CFR	Code of Federal Regulations

CA	Corrective Action
CAP	Corrective Actions Program
CST	Condensate Storage Tank
DBD	Design Basis Document
DP	Vermont Yankee Department Procedure
EAL	Emergency Action Level
EOP	Emergency Operating Procedure
EDCR	Engineering Design Change Request
EDG	Emergency Diesel Generator
FA	Fire Area
FZ	Fire Zone
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
IA	Instrument Air
I&C	Instrumentation and Control
IPEEE	Individual Plant Examination External Events
LER	Licensee Event Report
LCO	Limiting Condition for Operation
LOR	Licensed Operator Requalification Training Lesson Plan
LPC	Limited Procedure Change
MCC	Motor Control Center
MM	Minor Modification
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OD	Operability Determination
ON	Vermont Yankee Off-Normal Procedure
OP	Vermont Yankee Operating Procedure
PM	Preventive Maintenance
PMT	Post Maintenance Testing
RCIC	Reactor Core Isolation Cooling
RHRSW	Residual Heat Removal Service Water
SDP	Significance Determination Process
SE	Safety Evaluation
SLC	Standby Liquid Control
SRO	Senior Reactor Operator
SSC	Systems, Structures, and Components
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VY	Vermont Yankee
VYAPF	Vermont Yankee Administrative Procedure Form
VYC	Vermont Yankee Calculation
VYDC	Vermont Yankee Design Calculation
WO	Work Order