November 8, 2004

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 05000271/2004005

Dear Mr. Thayer:

On September 30, 2004, 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 October 14, 2004 with 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 NRC-identified finding and one self-revealing finding of very low safety significance (Green). Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. If you contest this non-cited violation, you should provide a response within 30 days of the date of this inspection report with the basis for your denial 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 the 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 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). (Note: Public access to ADAMS has been temporarily suspended so that security reviews of publicly available documents may be performed and potentially sensitive information removed. Please check the NRC website for updates on the resumption of ADAMS access.)

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000271/2004005
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.	05000271/2004005
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Vermont Yankee Nuclear Power Station
Location:	320 Governor Hunt Road Vernon, Vermont 05354-9766
Dates:	July 1, 2004 - September 30, 2004
Inspectors:	David L. Pelton, VY Senior Resident Inspector Beth E. Sienel, VY Resident Inspector Jennifer A. Bobiak, Electrical Engineering Inspector James D. Noggle, Senior Health Physicist Amar C. Patel, Reactor Engineer William J. Raymond, Pilgrim Senior Resident Inspector David M. Silk, Senior Emergency Preparedness Inspector Keith A. Young, Electrical Engineering Inspector
Approved by:	Clifford J. Anderson, Chief Projects Branch 5 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000271/2004005; 07/01/04 - 09/30/04; Vermont Yankee Nuclear Power Station; Event Followup and As Low As Is Reasonably Achievable (ALARA) Planning and Controls.

This report covered a 13-week period of baseline inspection conducted by resident inspectors and announced inspections by a senior health physicist and senior emergency preparedness, electrical engineering and reactor inspectors. Two Green findings were 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 (Green) self-revealing finding was identified because Entergy did not effectively incorporate industry operating experience into the preventive maintenance strategy for the 22 Kilovolt (KV) electrical system as required by Entergy’s preventive maintenance program. Specifically, Entergy’s preventive maintenance strategy for the 22 KV electrical system did not effectively include information from industry operating experience related to inspections of isophase bus bars and flexible connections or the periodic testing of surge arresters or capacitors located in the generator potential transformer cabinets. As a result, degraded conditions on the “B” phase bus bar flexible connection and within the “A” phase surge arrester went unidentified resulting in a two-phase electrical fault-to-ground that ignited a fire on top of the main transformer and ultimately resulted in an automatic reactor scram.

The finding is greater than minor since it is associated with the Equipment Performance-Maintenance attribute of the Initiating Events Cornerstone and because it affects the associated Cornerstone objective. In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Findings for At-Power Situations,” the inspectors conducted a SDP Phase 1 screening and determined that an SDP Phase 3 analysis was required since the finding resulted in a reactor scram and a fire on the main transformer but did not result in exceeding Technical Specification limits for identified reactor coolant system leakage and did not result in a total loss of safety function of a mitigating system. The Region I senior reactor analyst conducted a Phase 3 analysis and determined that the finding is of very low safety significance (Green) due to the resultant small increase in both core damage and large early release frequencies. This issue has been entered into Entergy’s corrective action program. (Section 4OA3.1)

Cornerstone: Occupational Radiation Safety

Green. The inspector identified a finding of very low safety significance (Green) because Entergy exceeded the original as low as reasonably achievable (ALARA) exposure estimate for reactor reassembly during the Spring 2004 refueling outage by 72% due to ineffective coordination and control of radiological work activities which were within its ability to foresee and correct.

The finding is greater than minor since it is associated with the Program and Process (ALARA Planning) attribute of the Occupational Radiation Safety Cornerstone and because it affects the associated Cornerstone objective. The finding is of very low safety significance (Green) because although it involved ALARA planning and work controls, the 3-year rolling average collective dose was less than 240 person-rem. This issue has been entered into Entergy's corrective action program. (Section 2OS2)

B. Licensee Identified Findings

A violation of very low safety significance, which was identified by Entergy, has been reviewed by the inspectors. Corrective actions taken or planned by Entergy have been entered into Entergy's corrective action program. This violation and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Vermont Yankee Nuclear Power Station began the inspection period with the reactor shutdown following a June 18, 2004, forced outage associated with a main transformer fire and reactor scram. Reactor startup activities began on July 5, following the completion of investigation, repair and replacement activities associated with the forced outage. The reactor was returned to full power operation on July 9, and with the exception of minor power reductions for control rod pattern adjustments, continued at, or near, full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

1. Readiness for Impending Adverse Weather Conditions

a. Inspection Scope (1 sample)

On August 11, 2004, the inspectors reviewed actions taken by Entergy in response to severe thunderstorm activity including a lightning strike in the 115 kilovolt (KV) switchyard. The inspectors reviewed Vermont Yankee Operating Procedure (OP) 3127, “Natural Phenomena,” Appendix B, “Lightning Damage Indicator Walkdown Checksheet,” and observed operators perform the actions of Appendix B. The inspectors performed independent walkdowns of selected equipment from Appendix B as well as walkdowns of the main control room panels and the 115 KV switchyard.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope (3 samples)

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

- The “A” train of the residual heat removal (RHR) system during planned maintenance on the “B” train of the RHR system;

- The high pressure coolant injection (HPCI) system, main feedwater pumps, and an administrative review of the availability of the automatic depressurization system during planned maintenance on the reactor core isolation cooling (RCIC) system; and
- The “B” emergency diesel generator (EDG) during planned maintenance on the “A” EDG.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope (9 samples)

The inspectors identified fire areas important to plant risk based on a review of the Vermont Yankee Safe Shutdown Capability Analysis, the Fire Hazards Analysis, and the Individual Plant Examination External Events (IPEEE). The inspectors toured plant areas important to safety in order to verify the suitability of Entergy’s control of transient combustibles and ignition sources, and the material condition and operational status of fire protection systems, equipment, and barriers. The following fire areas were inspected:

- Reactor building, 303 foot elevation (FZ RB7);
- Reactor building, 280 foot elevation, North (FZ RB5);
- Reactor building, 280 foot elevation, South (FZ RB7);
- Radwaste corridor (FA 13);
- RCIC corner room, 232 foot elevation (FZ RB1S);
- RCIC corner room, 213 foot elevation (FA RCIC);
- Service water system pump room (FZ 15);
- Augmented offgas system building (no fire designation); and
- Diesel fuel oil storage tank and transfer pump house (FA 12).

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope (1 sample)

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

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

1. Quarterly Review of Licensed Operator Requalification Testing and Training Activities

a. Inspection Scope (1 sample)

The inspectors observed simulator sessions for one operating crew to assess the performance of the licensed operators and the ability of Entergy's Training Department staff to evaluate licensed operator performance. Operating crew performance was evaluated during 1) a simulated positive reactivity addition followed by a leak outside primary and secondary containment and 2) a simulated loss of a 480 V motor control center, recirculation loop leak/break, and emergency core cooling system malfunctions. 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:

- Vermont Yankee Administrative 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 also compared simulator configurations with actual control board configurations. For any weaknesses identified, the inspectors observed the Entergy evaluators to verify that they also noted the issues and discussed them with the crew.

b. Findings

No findings of significance were identified.

2. Operator's License - Medical Requirements

a. Inspection Scope

The inspectors evaluated actions taken by Entergy to evaluate the medical condition of a licensed operator who returned to work after experiencing a medical condition that might be reportable. The inspectors reviewed the operator's medical records, American National Standard (ANS) 3.4, "American National Standard Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants;" 10 VFR 55.21, "Medical Examinations;" 10 CFR 55.25, "Incapacitation Because of Disability or Illness;" and Entergy Nuclear Northeast Organization and Management Procedure ENN-OM-117, "Medical Program Procedure." The inspectors interviewed Operations Department personnel including the licensed operator with the medical condition.

b. Findings

During the review of the affected licensed operator's medical records, the inspectors identified the following:

- Operations Department Management had cleared the licensed operator to return to licensed duties without first consulting with Vermont Yankee's medical records officer (i.e., the medical doctor responsible for evaluating licensed operator medical conditions).

- The licensed operator had been prescribed medication by a personal physician following the medical event but had not discussed this with the medical records officer prior to returning to work.

- Once prompted by the inspectors, the licensed operator provided to the medical records officer information regarding the prescribed medication. The medical records officer's response indicated that the licensed operator's license should be restricted as "no solo" (i.e., would require another qualified individual to be present while performing licensed duties). An error by a medical files administrator resulted in the restriction not being forwarded to the individual or to Operations Department Management.

Following the inspectors' identification of the above issues, Entergy took immediate action to place an administrative "no solo" restriction on the licensed operator's license. The inspectors determined that the licensed operator had not been assigned to licensed duties since returning to work following the medical event. Entergy has also forwarded the changes in the licensed operator’s medical condition, including the prescribed medication, to the NRC for further evaluation. This issue has been entered into Entergy’s corrective action program (CR 2004-2158). Pending the outcome of the NRC’s review of the licensed operator’s medical condition against the requirements of 10 CFR 55 and ANS 3.4, this issue is considered to be an unresolved item (URI): **URI 05000271/2004005-01, Issues Identified with the Evaluation and Reporting of Changes in Licensed Operator Medical Conditions**.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope (3 samples)

The inspectors performed three issue/problem-oriented inspections of actions taken by Entergy in response to the following issues:

- Main steam isolation valve (MSIV) as-found local leakage rate testing (LLRT) failures;
- HPCI turbine exhaust inboard check valve (V23-3) as-found LLRT failures; and
- Reactor building railroad airlock doors (secondary containment) seal failures.

The inspectors reviewed applicable system maintenance rule scoping documents, system health reports, corrective actions taken in response to the equipment problems, maintenance rule functional failure determinations, and applicable a(1) action plans. In addition, the issues were discussed with the responsible engineers.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope (5 samples)

The inspectors evaluated on-line risk management for five planned maintenance activities. The inspectors reviewed maintenance risk evaluations, work schedules, recent corrective actions, and control room logs to verify that other concurrent or emergent maintenance activities did not significantly increase plant risk. The inspectors compared these items and activities to requirements listed in AP 0125, "Equipment Release" and AP 0172, "Work Schedule Risk Management - Online." The inspectors reviewed the following planned work activities:

- Planned maintenance on the “B” train of the RHR system;
- Planned maintenance on the RCIC system;
- Planned maintenance on alternate shutdown battery charger CAS-1;
- Planned maintenance on the “A” EDG; and

- Installation of a temporary modification to replace failing Weidmuller fuse holders.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Evolutions and Events (71111.14)

a. Inspection Scope (1 sample)

The inspectors assessed control room operator performance during an unplanned, partial primary containment isolation system Group V isolation of the reactor water cleanup (RWCU) system following the failure of the “B” RWCU pump casing vent on September 10, 2004. Response to this isolation included entry into emergency operating procedure (EOP) 4, “Secondary Containment Control Procedure,” due to elevated temperatures in the reactor building.

The adequacy of personnel performance, procedure compliance, and use of the corrective action process were evaluated against the requirements and expectations contained in Technical Specifications and the following station procedures, as applicable:

- AP 0151, “Responsibilities and Authorities of Operations Department Personnel;”
- AP 0153, “Operations Department Communications and Log Maintenance;”
- DP 0166, “Operations Department Standards;” and
- OP 2112, “Reactor Water Cleanup System.”

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (4 samples)

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

- “C” residual heat removal service water (RHRSW) system pump upper bearing oil site glass repeat leakage;

- "A" standby liquid control (SLC) system pump south plunger leakage greater than procedure acceptance criteria;
- Review of safety function evaluation for Rockwell valves susceptible to packing follower leakage; and
- HPCI system steam admission line drain valve failure.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope (1 sample)

The inspectors reviewed the cumulative effect of operator workarounds resulting from main control board annunciator unavailability 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, and disabled or illuminated control room alarms and discussed them with responsible operations personnel to ensure they were appropriately categorized and tracked for resolution. In addition, control panel walkdowns were performed to identify if any potential workarounds existed that had not been previously identified in accordance with procedure DP 0166, "Operations Department Standards."

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope (8 samples)

The inspectors reviewed completed documentation for eight post-maintenance test (PMT) activities to verify the test data met the required acceptance criteria contained in Entergy's Technical Specifications, Updated Final Safety Analysis Report (UFSAR), and in-service testing program, and that the PMT was adequate to verify system operability and functional capability following maintenance. The inspectors reviewed the PMTs performed after the following maintenance activities:

- Strengthening of the steam dryer;
- Modifications to the main steam line high flow trip instrumentation;
- Modifications to recirculating pump runback circuitry;
- Maintenance on the "A" standby gas treatment fan;
- Maintenance on the "B" train of the RHR system;
- Maintenance on the RCIC system;

- Replacement of the “A” reactor protection system motor generator set voltage regulator; and
- Planned maintenance on the “A” EDG.

The inspectors verified that systems were properly restored following testing and that discrepancies were appropriately documented in the corrective action process. The inspectors also discussed the PMT results with the responsible engineers, as needed.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope (1 sample)

The inspectors observed portions of the heatup and startup of the reactor plant following the completion of the forced outage resulting from the main transformer fire on June 18, 2004. The inspectors reviewed the requirements of OP 0105, “Reactor Operations,” and ensured that prerequisite conditions had been met prior to startup, ensured observed startup activities were performed in accordance with approved procedures, and ensured control room personnel were appropriately focused on plant operations.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (5 samples)

The inspectors observed surveillance testing to verify that the test acceptance criteria specified for each test was consistent with Technical Specification and UFSAR requirements, was performed in accordance with the written procedure, the test data was complete and met procedural requirements, and the system was properly returned to service following testing. The inspectors observed selected pre-job briefs for the test activities. The inspectors also verified that discrepancies were appropriately documented in the corrective action program. The inspectors verified that testing in accordance with the following procedures met the above requirements:

- OP 4124, “Residual Heat Removal and RHR Service Water Surveillance,” Section G, “RHR Service Water Pump and Valve Operability and Full Flow Test,” for the “A” and “C” RHRSW system pumps;
- OP 4124, Section H, “RHR Pump Operability Test,” for the “A” and “C” RHRSW system pumps;

- OP 4126, “Diesel Generator Surveillance,” Section C, “Diesel Generator Overspeed Trip Test” for the “A” EDG;
- OP 4126, Section G, “Diesel Generator Air Compressor Capacity Test and Discharge Check Valve Test,” for the “A” EDG; and
- OP 4127, “John Deere Diesel Generator Surveillance.”

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope (1 sample)

The inspectors reviewed temporary modification (TM) 2004-028, Bypass of Feedwater Weidmuller Fuse Holders, 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 in the control room 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

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope (8 samples)

The inspector conducted the following activities to verify that Entergy had properly implemented physical, engineering, and administrative controls for access to high radiation areas and other radiologically controlled areas, and that workers were adhering to these controls when working in these areas. Implementation of the access control program was reviewed against the criteria contained in 10 CFR 20, site technical specifications, and procedures. This inspection activity represents completion of eight samples relative to this inspection area.

- The inspector walked down portions of the plant to verify there were no posted airborne radioactivity areas or potential internal exposure accessible work areas greater than 50 millirem (mrem) committed effective dose equivalent (CEDE);
- Dosimetry records were reviewed to verify that there were no recorded internal exposures greater than 50 mrem CEDE during 2004;
- Controls for the under water storage of highly activated reactor components in the spent fuel pool were observed;
- Radiation Protection Audit No. QA-14-2004-VY-01, dated July 8, 2004, was reviewed;
- Condition reports (CRs) were reviewed to ensure the radiation protection audit was identifying any repetitive deficiencies in the radiation protection program. A listing of CRs reviewed is included in the Attachment to this report;
- Performance Indicator incident information relative to the Occupational Radiation Safety Cornerstone was reviewed for the previous four quarters;
- The inspectors reviewed procedure OP 0532, “Locked High Radiation Area Door Key Control,” and AP 0541, “Access to High and Very High Radiation Areas,” and discussed their implementation with Radiation Protection Department personnel; and
- Utilizing the latest high radiation area checklist, the inspector walked down the plant and verified the postings, barricades, and locked status of all the plant locked high radiation areas. In addition, the locked high radiation area keys were inventoried.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope (3 samples)

The inspectors conducted the following activities to verify that Entergy was properly maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). Implementation of the ALARA program was reviewed against the criteria contained in 10 CFR 20.1101(b) and Entergy’s procedures. This inspection activity represents completion of three samples relative to this inspection area.

- Procedure AP 0537, Rev. 16, “ALARA Implementation for Design Changes and Work Analysis” was reviewed with respect to processes used to estimate, re-estimate, and track work activity exposures;
- Post-job review reports were reviewed for the following five highest exposure work activities for the Spring 2004 refueling outage and were analyzed for any doses exceeding estimates: Staging, miscellaneous drywell, steam dryer modifications, reactor disassembly/reassembly, and miscellaneous valve work. Follow-up discussions were held with ALARA specialists and the ALARA Engineer; and

- Evaluation of Entergy's method for adjusting exposure estimates were performed through a review of procedure AP 0537, "ALARA Implementation for Design Changes and Work Analysis," review of refueling outage (RFO) 24 post-job reviews, and discussions with the Radiation Protection Department personnel.

b. Findings

Introduction. The inspector identified a finding of very low safety significance (Green) because Entergy exceeded the original ALARA exposure estimate for reactor reassembly during the Spring 2004 refueling outage by 72% due to ineffective coordination and control of radiological work activities which were within their ability to foresee and correct.

Description. While planning RFO 24 reactor disassembly and reassembly work activities, Entergy developed an ALARA estimate of 5.67 person-rem for the job. Once reactor reassembly work had been completed, the accountable exposure was 9.74 person-rem. Compared to the original 5.67 person-rem estimate, this actual exposure represented a 72% increase. Various refueling floor coordination and work control issues contributed to spending three times the hours and dose usually required for reactor reassembly. Examples of issues that contributed to additional exposure include:

- Reactor head nuts and washers had originally been cleaned and organized for reassembly on the refuel floor. The in-service inspection group subsequently inspected them and left them in a disorganized condition. The washers and nuts were resorted in the reactor cavity area (a high dose area) instead of transporting them back to the refueling floor for reorganizing in a low dose area. Additionally, the nuts and washers were installed on the wrong studs, resulting in more rework in the reactor cavity.

- Refueling cavity decontamination was conducted while several inches of water remained on the cavity floor. The remaining water was drained from the cavity and the cavity bellows was refilled for shielding purposes. Draining the water from the cavity created a high contamination work area for installing the reactor head. The reactor head work subsequently resulted in 14 contamination events, extending the hours and exposure for this work activity.

- Due to the multiple contamination events, Entergy decided to decontaminate the floor of the cavity. Because the cavity bellows had been previously filled and the drain closed, the individuals assigned to the cavity floor decontamination could not flush the bellows drain (as was usual practice) resulting in an incomplete decontamination and higher dose levels.

Analysis. Entergy exceeded the original ALARA program estimate for RFO 24 reactor reassembly work by 72% due to ineffective coordination and control of radiological work activities which were within their ability to foresee and correct. The finding is greater than minor since it is associated with the Program and Process (ALARA Planning) attribute of the Occupational Radiation Safety Cornerstone and because it affects the associated cornerstone objective. Specifically, it affected the cornerstone objective in that the ineffective coordination and control of radiological work practices associated with RFO 24 reactor reassembly work challenged Entergy's ability to ensure the adequate protection of worker health and safety from exposure to radiation. IMC 0308, "Technical Basis for Occupational Radiation Safety Significance Determination Process," states that situations where the unplanned, unintended collective dose for a work activity exceeds 50% of the planned collective dose and the actual total dose for the work activity is greater than or equal to 5 person-rem should be considered more than minor. In this instance, the unplanned and unintended collective dose for the reactor reassembly was 72% of the planned collective dose and the actual total dose was 9.74 person-rem. In accordance with IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," the inspectors determined that the finding was of very low safety significance (Green). The inspectors determined that the finding involved ALARA planning and work controls but that the 3-year rolling average collective dose was less than 240 person-rem (the 3-year rolling collective dose accrued at Vermont Yankee was 115 person-rem).

Enforcement. No violation of regulatory requirements occurred. 10 CFR 20.1101(b) states that, "The licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA)." The Statements of Consideration published with the above regulation states that "compliance with this requirement will be judged on whether the licensee has incorporated measures to track and, if necessary, to reduce exposures and not whether exposures and doses represent an absolute minimum or whether the licensee has used all possible methods to reduce exposure." As there was no overall breakdown in maintaining exposures within exposure estimates during the refueling outage, and the three-year average collective exposures at VY have been below the median industry average, no violation of 10 CFR 20.1101(b) occurred: **FIN 05000271/2003005-02, Entergy exceeded the original ALARA estimate for reactor reassembly by 72% due to ineffective coordination and control of radiological work activities.**

Cornerstone: Public Radiation Safety (PS)

2PS1 Gaseous and Liquid Effluents (71122.01)

a. Inspection Scope (10 samples)

The inspectors reviewed the following documents to evaluate the effectiveness of Entergy's radioactive gaseous and liquid effluent control programs at Vermont Yankee. The requirements for radioactive effluent controls are specified in the Technical Specifications and the Offsite Dose Calculation Manual (TS/ODCM):

- The 2002 and 2003 Radiological Annual Effluent Release Reports were reviewed including projected public dose assessments;

- The current ODCM (Revision 30) was reviewed including technical justifications for any changes made since the previous revision;
- UFSAR Sections 9.4 and 10.12 were reviewed which describe the gaseous radioactive waste system and station ventilation systems; and
- The latest quality assurance audit (RETS/REMP/ODCM Audit Report, QA-6-2004-VTY-1) was also reviewed.

The inspectors observed the following plant equipment and work activities to evaluate the effectiveness of Entergy's radioactive gaseous and liquid effluent control programs:

- The radioactive liquid/gaseous effluent radiation monitoring systems (RMS);
- Sampling and laboratory measurement techniques;
- Air cleaning systems; and
- Plant stack gaseous effluent sample collection, counting and analysis.

There were no radioactive liquid waste releases reported in the 2002 and 2003 radioactive annual effluent release reports and no radioactive liquid waste releases through August 2004. Two radioactive gaseous release permits (nos. 2003-01 and 2003-02) were selected and reviewed with respect to ODCM and procedural requirements.

There were no instances of unplanned effluent RMS unavailability that would require compensatory sampling and analysis between October 2002 and August 2004.

Changes to the ODCM (Revision 30) were reviewed along with the technical justification for each change. This included a calculation method to correlate offsite dose due to main steam line radiation measurements.

Effluent release dose calculations were reviewed for each month of 2003 and the first seven months of 2004 with respect to TS/ODCM calculation methodology and 10 CFR 50, Appendix I public dose requirements.

The inspectors reviewed the most recent air cleaning system filter surveillance test results required by Technical Specifications (visual inspection, pressure differential, in-leakage tests, laboratory charcoal efficiency test, and air flow capacity tests, as appropriate) for the following systems:

- The standby gas treatment system;
- The shutdown iodine system;
- The radwaste roof filtration system; and
- The augmented off-gas filtration system.

The inspectors reviewed the most recent calibration results for the gaseous and liquid effluent RMS radiation monitors and associated flow rate measurement devices, as required by the ODCM for the following monitors:

- The liquid radwaste effluent (RM-17-350);

- The service water effluent (RM-17-351);
- The cooling tower influent (RM-17-359);
- The reactor building closed cooling water (RM-17-352);
- The steam jet air ejector (RM-17-150A/B);
- The augmented off-gas (RM-3127/3128); and
- The plant stack I & II primary and secondary calibrations.

Effluent liquid and gas sample radiation measurement equipment calibrations were reviewed for currently in-use high purity germanium gamma spectrometers and liquid beta scintillation counter.

Implementation of the measurement laboratory quality control program was reviewed, including effluent intra-laboratory and inter-laboratory comparisons. In addition, the inspectors reviewed the 2004 quality assurance audit (RETS/REMP/ODCM Audit Report, QA-6-2004-VTY-1) of the radioactive liquid and gaseous effluent control program and the ODCM.

CRs reviewed are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope (5 samples)

The inspectors reviewed Entergy submittals for the performance indicators (PIs) listed below. The PI definitions and guidance contained in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” and AP 0094, “NRC Performance Indicator Reporting,” were used to verify the accuracy and completeness of the PI data reported.

Mitigating Systems Cornerstone

- Safety System Unavailability, High Pressure Injection System (HPCI); and
- Safety System Unavailability, Heat Removal System (RCIC).

The inspectors reviewed licensee event reports (LERs), portions of operator logs, maintenance rule out of service logs, and CRs to verify the accuracy and completeness of the PI data for the period from July 1, 2003 through June 30, 2004. The inspectors also interviewed personnel responsible for the PI data collection and evaluation.

Emergency Preparedness Cornerstone

- Drill/Exercise Performance (DEP);
- Emergency Response Organization Drill Participation (ERO); and
- Alert and Notification System Reliability.

The inspectors reviewed Entergy's 2003/2004 drill and exercise reports, training records and American Nuclear Society testing data to verify the accuracy of the reported data. Data generated since the August 2003 emergency preparedness PI verification was reviewed during this inspection. The inspectors also interviewed personnel associated with the PI data collection, evaluation, and distribution.

b. Findings

No findings of significance were identified.

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 entered into Entergy's corrective action system at an appropriate threshold and that adequate attention was 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 Main Steam Isolation Valve Leakage Testing Failures

a. Inspection Scope (1 sample)

The inspectors selected MSIV LLRT failures as a sample due to three MSIV LLRT failures during the most recent refueling outage and the history of failed tests at Vermont Yankee. The inspectors focused on LLRT failures in the past five years and reviewed the CRs listed in the Attachment to this report. The CRs were reviewed to ensure the full extent of documented issues were identified, appropriate evaluations were performed, and appropriate corrective actions were specified, with a particular focus on the recent adverse trend CR 2004-0918. The inspectors evaluated CRs against the requirements of AP 0009, "Condition Reports." The inspectors reviewed the root cause for the recent “B” inboard MSIV failed LLRT and the recent adverse trend common cause evaluation and discussed the results with the responsible engineer to understand the basis of Entergy’s conclusions and long term corrective actions.

b. Findings and Observations

No findings of significance were identified. The inspectors concluded that although there have been several failed MSIV LLRTs in the past five years, there was no common cause for the failures. Completed immediate and short term and planned long term corrective actions were appropriate.

4OA3 Event Followup (71153)

1. (Closed) LER 05000271/2004003-00, Automatic Reactor Scram due to a Main Generator Trip as a result of an Isophase Bus Duct Two-Phase Electrical Fault.

a. Inspection Scope

The inspectors reviewed LER 05000271/2004003-00 and Root Cause Analysis (RCA) Report CR-VTY-2004-2015, “Electrical Fault/Fire,” to verify that Entergy had identified the causes of the conditions resulting in a main transformer fire and reactor scram and had identified reasonable corrective actions. The inspectors interviewed station personnel involved with the development of the above LER and RCA report and reviewed AP 0214, “Preventive Maintenance Program Implementation.”

b. Findings

Introduction. A very low safety significance (Green) self-revealing finding was identified because Entergy did not effectively incorporate industry operating experience into the preventive maintenance strategy for the 22 Kilovolt (KV) electrical system as required by Entergy’s preventive maintenance program. Specifically, Entergy’s preventive maintenance strategy for the 22 KV electrical system did not effectively include information from industry operating experience related to inspections of isophase bus bars and flexible connections or the periodic testing of surge arresters or capacitors located in the generator potential transformer cabinets. As a result, degraded conditions on the “B” phase bus bar flexible connection and within the “A” phase surge arrester went unidentified resulting in a two-phase electrical fault-to-ground that ignited a fire on top of the main transformer and ultimately resulted in an automatic reactor scram.

Description. On June 18, 2004 at 0640, with the plant operating at full power, a two-phase fault-to-ground occurred on the 22 KV electrical system. The “B” phase faulted to ground in the low voltage bushing box located on top of the main transformer and the “A” phase faulted to ground in the surge arrester cubicle of the generator potential transformer (PT) cabinet through the “A” phase surge arrester.

The event began when a laminate layer from a flexible connector on the “B” phase isophase bus bar suffered a failure due to low cycle fatigue on the leading (upstream) edge of the laminate. Each flexible connector is made up of laminated assemblies grouped in pairs on each side of each hollow, square, tubular bus bar for a total of eight assemblies per connector. Each assembly contains 22 aluminum laminations measuring 17 inches long, 4 inches wide and 0.20 inches thick stacked one on top of another and welded at each end. When the isophase bus duct cooling system was in operation, the failed leading edge of the laminate was caught by the air stream. Subsequently, the trailing edge of the laminate failed and the laminate became detached from the flexible connector. The air stream within the “B” isophase bus duct caused the detached laminate to tumble towards the main transformer along the “B” phase isophase bus duct for approximately 40 feet until it reached a vertical section of the bus duct. Entergy determined that this movement of the laminate occurred during the most recent refueling outage (RFO 24) at a time when the isophase bus was not energized but the bus duct cooling system was operating. This is evidenced by a lack of arc strikes along this 40-foot horizontal run of bus ducting.

Following the re-energization of the “B” bus, the piece of laminate, now located in the bottom of a vertical section of the “B” isophase bus duct, was moved by the air stream along the remaining 35 feet to the “B” low voltage bushing and bushing box. This movement of the laminate was evidenced by numerous arc strikes that occurred en route, each arc strike representing a momentary ground of the “B” phase bus. Once in the bushing box, the laminate provided a pathway for arcing to ground. This arcing caused the phase-to-ground voltage on the “B” bus to decrease to zero and the “A” and “C” phase-to-ground voltages to increase from 12.4 KV to 21.5 KV. During this arcing event, the “A” surge arrester failed resulting in a two-phase ground between the “A” and “B” phases. The main generator protective relaying sensed the two-phase fault between the “A” and “B” phases and isolated the generator from the grid. A turbine-generator trip and automatic reactor scram followed. The voltage transient also resulted in a trip of both recirculation pumps and other minor alternating current anomalies. The plant response to the voltage transient, turbine-generator trip, and reactor scram was as expected.

Following the initial electrical faults-to-ground from the “A” and “B” phases, arcing and ionization in the “B” phase low voltage bushing box carried over to the “C” phase low voltage bushing box. The electrical faults also resulted in a slight mechanical displacement of a mechanical flanged joint in a 1 inch oil piping line between the main transformer oil expansion tank and the “C” phase low voltage bushing, resulting in an oil leak. The arcing and heat in the vicinity of the “C” phase low voltage bushing ignited the leaking oil resulting in a fire. The main transformer fire protection systems automatically activated and functioned as designed. Additionally, the VY fire brigade was dispatched to fight the fire. The VY fire brigade initiated fire hose spray from a nearby hydrant and successfully quenched the fire. Local fire departments arrived on-site and assisted the VY fire brigade as needed.

Entergy declared an Unusual Event (UE) at 0650 due to the fact that the on-site fire was not extinguished within 10 minutes. The UE was exited at 1245, once the fire was extinguished and site personnel were certain that no fire reflash was possible. Offsite electrical power sources and station emergency power sources remained available throughout the event.

Entergy determined that there were two root causes of this event. The first was that the preventive maintenance performed on the 22KV isophase bus bars and flexible connections were not adequate. Although the isophase bus ducting was subjected to preventive maintenance, such as cleaning, inspection, and testing each refueling outage, the scope of maintenance did not include an evaluation of the condition of the bus bars or the condition of the flexible connections. Industry operating experience (OE) indicated the need for inspections of the flexible connectors due to previous failures similar to that experienced at VY. Additional inspections to evaluate the condition of the bus bars and flexible connectors at VY would have allowed for the detection of the degraded flexible connector. The second root cause identified by Entergy was that no testing was performed on the surge arresters or capacitors located in the generator PT cabinets. Industry OE had revealed that surge arresters degrade over time due to a combination of age, service environment, and service conditions. Periodic testing would have detected degradation and allowed for replacement prior to failure.

During RFO 24, Entergy replaced the isophase bus duct cooling unit in support of a proposed 20 percent power uprate. The new cooling unit increased the bus duct air flow from about 10,500 standard cubic feet per minute (SCFM) to about 17,300 scfm [this equates to an increase in air speed from about 34 feet per second to about 63 feet per second]. In their RCA report, Entergy concluded that this increase in bus duct cooling air flow likely accelerated the failure timetable for the laminate but that ultimately the failure would have occurred at some point in the future at the original system flow rate. This conclusion was supported by Entergy's discovery that the weld on the leading edge of the failed flexible connector had been subjected to excessive grinding during manufacturing, visual evidence of low cycle fatigue cracking and oxidation along the leading edge of the failed laminate indicating the failure developed slowly over time, and no evidence of flow induced vibration resulting from the increase in turbulent air flow over the laminate as indicated by the fact that none of the other seven adjacent connectors exhibited signs of degradation similar to the failed laminate or otherwise.

Entergy entered this event into their corrective actions program as CR 2004-2015.

<u>Analysis</u>. The performance deficiency associated with this finding is that Entergy did not effectively incorporate operating experience (OE) into the PM strategy for the 22 KV electrical system as required by procedure AP 0214, "Preventive Maintenance Program Implementation." AP 0214 requires that PM strategies be developed using, in part, a reliability-based maintenance evaluation process which considers industry, vendor, and plant experience to support continuous improvement of the PM program. The finding is greater than minor since it is associated with the Equipment Performance-Maintenance 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 and challenge critical safety functions during power operations. Specifically, Entergy

performed neither inspections of the 22 KV electrical system isophase bus bars and flexible connections nor periodic testing of associated surge arresters or capacitors located in the generator PT cabinets despite having industry OE indicating the need to perform such PM-related activities. As a result, degraded conditions on the “B” phase aluminum bus bar flexible connector and within the “A” phase surge arrester in the generator PT cabinet went unidentified resulting in a two-phase electrical fault-to-ground which ignited a fire on top of the main transformer and ultimately resulted in an automatic reactor scram.

In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Findings for At-Power Situations,” the inspectors conducted a SDP Phase 1 screening and determined that an SDP Phase 3 analysis was required since the finding resulted in a reactor scram and a fire on the main transformer but did not result in exceeding Technical Specification limits for identified reactor coolant system leakage and did not result in a total loss of safety function of a mitigating system.

The Region I senior reactor analyst (SRA) conducted a Phase 3 analysis and determined that the finding was of very low safety significance (Green) for the increase in both core damage and large early release frequencies (i.e., ΔCDF and ΔLERF). In performing the analysis, the SRA used information from the VY IPEEE regarding the potential for and effects of a main transformer fire. The VY IPEEE documented that the frequency of core damage due to a fire on the main transformer was in the low E-6 per year range with an initiating event frequency in the range of mid E-3. Such a fire event is assumed to result in a plant trip due to generator load rejection. The IPEEE analysis also assumed that the fire would propagate to the turbine building, resulting in losses of the main feedwater, condensate, and main condenser systems leaving only safety-related systems to prevent core damage. To reflect these conditions in the Phase 3 analysis, the SRA applied the Transient Without Power Conversion System (TPCS) SDP worksheet and revised the initiating event likelihood from a baseline value of E-1 per year to a value of mid E-3 per year. After applying this change, the baseline TPCS core damage estimate was on the order of mid E-8 per year, which was consistent with the IPEEE estimate. Because the finding resulted in a fire on the main transformer, the SRA increased the fire initiating event frequency by one order of magnitude to E-2 per year. This resulted in an estimated increase in core damage frequency on the order of mid E-7 per year. The dominant core damage sequences were 1.) failure to remove decay heat from the containment with a failure to vent the containment, and 2.) a failure of high pressure injection with a failure of the operator to depressurize the plant. Further, in accordance with IMC 609 Appendix A, the SRA determined that the increase in ΔLERF was in the mid E-8 range, considering the ability of operators to depressurize the reactor and flood the containment following core damage.

Enforcement. No violation of NRC regulatory requirements was identified. Although Entergy did not implement its requirements in its preventive maintenance program related to the 22 KV electrical system, this aspect of the preventive maintenance program is not an NRC regulatory requirement. This LER is closed. **FIN 0500271/2004005-03, Did Not Effectively Incorporate Operating Experience into the Preventive Maintenance Strategy for the 22 Kilovolt Electrical System**

2. (Closed) LER 05000271/2004001-00, Main Steam Isolation Valve Leakage Exceeds a Technical Specification Leakage Rate Limit

On April 5, 2004, with the plant shutdown for refueling, Entergy identified that the “B” inboard MSIV exceeded its allowable Technical Specification 3.7.A.4 leakage rate limit of 31 standard cubic feet per hour (SCFH) at 44 psig during a local leak rate test (LLRT). Entergy determined the LLRT failure was due to binding of the valve stem and packing follower from scoring and galling which was worsened by cycling the valve for scheduled surveillance activities. The scoring and galling was the result of inadequate maintenance instructions, provided during previous maintenance activities, that did not specify required tolerances. Corrective actions prior to reactor startup included disassembly, repair, and testing of the “B” inboard MSIV and inspection of all other MSIVs for stem scoring, with the subsequent repair of two others with indications of scoring. Long term corrective actions include development of a controlled MSIV maintenance procedure and review and revision of MSIV drawings to specifically identify critical design dimensions. This finding is more than minor because it had a credible impact on safety, in that if the redundant (outboard) “B” MSIV failed to close on a containment isolation signal, containment integrity would not be ensured. The finding affects the Barrier Integrity Cornerstone and was considered to have very low safety significance (Green) using Appendix H of the SDP because the likelihood of an accident leading to core damage was not affected and the MSIV leakage was less than 10,000 scfh through the best-sealing valve in any steam line. (In fact, the “B” outboard MSIV remained operable with leakage of less than 1.0 scfh.) This licensee-identified finding was a violation of TS 6.4, Procedures. The enforcement aspects of this violation are discussed in Section 4OA7. This LER is closed.

4OA5 Other Activities

2. (Closed) Unresolved Item (URI) 05000271/2004003-02, Weaknesses Identified with the Preventive Maintenance Performed on the 22 KV Electrical System Resulted in Main Transformer Fire.

This URI is considered closed based on the results of the inspectors’ review of the June 18, 2004 main transformer fire as documented in Section 4OA3.1 of this report.

2. Strike Contingency Planning (92709)

a. Inspection Scope

Entergy developed the Vermont Yankee Management Alternative Plan (MAP) to provide a sufficient number of qualified personnel to continue Vermont Yankee operations in the event that International Brotherhood of Electrical Workers (IBEW) union personnel engaged in a job action upon the expiration of their contract on August 19, 2004. Using the guidance of NRC Inspection Procedure 92709, “Licensee Strike Contingency Plans,” the inspectors reviewed Entergy’s plans to address a potential job action at the site. The inspection included an evaluation of the MAP content and the actions needed to implement the plan; a review to determine whether the number of qualified personnel needed for the proper operation and safety of the facility would be available; a review to determine if reactor operation and facility security would be maintained as required; and, a review to determine if the plan complied with technical specification requirements and other NRC requirements. On August 19, IBEW union personnel approved a contract and no job action was taken.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

Resident Exit

On October 14, 2004, the resident inspectors presented the inspection results to Mr. Kevin Bronson and members of his staff. The inspectors asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following finding of very low safety significance (Green) was identified by Entergy and is a violation of NRC requirements, which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as an NCV:

- Technical Specification 6.4, “Procedures,” requires that written procedures be established, implemented, and maintained covering preventive and corrective maintenance operations, which could have an effect on the safety of the reactor. Contrary to this, Entergy identified a failure to provide an adequate procedure for corrective maintenance on the “B” inboard MSIV which resulted in a failed TS required LLRT of the valve on April 5, 2004. This was entered into Entergy’s corrective action program as CRs 2004-0841 and 2004-0955. This finding was of very low safety significance because the “B” outboard MSIV remained operable.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

J. Allen, Design Engineering
K. Bronson, General Plant Manager
P. Corbett, Maintenance Manager
J. Dreyfuss, Project Engineering Manager
J. Devincentis, Licensing Manager
W. Fadden, Design Engineering
J. Geyster, Radiation Protection Superintendent
D. Giorowall, Programs Supervisor
D. Girroir, Programs Supervisor
S. Goodwin, Mechanical Design Department Manager
M. Gosekamp, Superintendent of Operations Training
M. Hamer, Licensing
D. Johnson, Design Engineering
D. King, ISI Coordinator
M. Layton, ALARA Specialist
R. Morissette, Principal As Low As Reasonably Achievable (ALARA) Engineer
M. Pletcher, Radiation Protection Supervisor - Instruments
P. Rainey, Design Engineering
K. Stupak, Technical Training
J. Thayer, Site Vice President
C. Wamser, Operations Manager
R. Wanczyk, Director of Nuclear Safety

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000271/2004005-01	URI	Issues Identified with the Evaluation and Reporting of Changes in Licensed Operator Medical Conditions (Section 1R11.2)

Opened/Closed

05000271/2004005-02	FIN	Entergy exceeded the original ALARA estimate for reactor reassembly by 72% due to ineffective coordination and control of radiological work activities (Section 2OS2)
05000271/2004005-03	FIN	Did Not Effectively Incorporate Operating Experience into the Preventive Maintenance Strategy for the 22 Kilovolt Electrical System (Section 4OA3.1)

Closed

05000271/2004003-00	LER	Automatic Reactor Scram due to a Main Generator Trip as a result of an Isophase Bus Duct Two-Phase Electrical Fault (Section 4OA3.1)
05000271/2004001-00	LER	Main Steam Isolation Valve Leakage Exceeds a Technical Specification Leakage Rate Limit (Section 4OA3.2)
05000271/2004003-02	URI	Weaknesses Identified with the Preventive Maintenance Performed on the 22 KV Electrical System Resulted in Main Transformer Fire (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

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

Condition Reports

1996-0343	Potential impact on the diesel generators ability to perform their safety function in the event of a design basis tornado
1999-0481	Broken locking mechanism on tornado damper
2000-0265	EDG tornado relief damper plunger failed during PM
2001-0083	AP 4000 “Surveillance Testing Program” is not remaining current with station practices or changes
2001-2020	Configuration control inconsistencies for tornado relief dampers
2001-2101	Incorrect spring position found on tornado relief damper
2001-2349	Insufficient testing requirements specified by mod 97-409
2003-0615	Switchgear ventilation drawing from turbine building
2003-0976	WOSE 2002-044 does not comply with AP 0020
2003-1710	Tritium Concentrations above Background
2003-2093	Missing Master Key Assigned to RP
2003-2245	Steam leak from packing on 3/4" test line manual isolation valve
2003-2342	Adverse trend Rockwell packing follower failure
2003-2346	RP Technician incorrect response to AMS-4 alarm
2004-0203	HVAC air flow is occurring from an area of potentially higher radioactive contamination to an area normally not expected
2004-0255	Reactor building outer door does not operate freely
2004-0277	Outer reactor building railroad airlock door seal pulls out of seal support clips
2004-0565	Outer reactor building railroad door bladder rolled out of retaining clips
2004-0624	Operability determination for CR 2003-2342 didn’t address specific safety-related functions
2004-0661	Outer reactor building railroad airlock door clips found off the seal
2004-0745	Outer railroad airlock seal dislodged
2004-0751	Outer reactor building door seal found out of place
2004-0769	Fire Protection DGFP exceeded its unavailability performance criteria
2004-0838	MSIV B Outboard valve (86B) failed leak rate testing
2004-0841	MSIV B Inboard valve (80B) failed leak rate testing

2004-0909	Checkpoint Frisking
2004-0955	As-found condition of V2-80B (inboard MSIV) included a galled stem
2004-0968	Unable to decontaminate worker
2004-1013	Zone 15 Fire alarm received in Control Room
2004-1017	HPCI valve V23-3 failed Appendix J local leakage rate testing
2004-1211	Internal contamination of rad workers
2004-1353	RP technician failed to implement proper radiological controls
2004-1440	Inner railroad air lock door seal found out of clips
2004-1723	Lack of adherence to RWP and good rad worker practices may have caused intakes of Radioactive material
2004-1725	Error identified in technical specifications
2004-1728	Relief valves failed as-found setpoint testing
2004-1957	Several recent RP surveys of maintenance shop and RCA tool crib have found tools with loose surface contamination and items not marked as radioactive
2004-2095	"A" SLC pump south plunger leakrate greater than surveillance acceptance criteria
*2004-2158	Medical return to work process error occupational health
2004-2218	Failure to Notify the NRC of the need for a Conditional Operator License (Glasses)
2004-2229	No formalized process of notifications between departments for Operator Physicals
2004-2295	Numerous SLC Valves Have Boron Buildup During Operator Rounds
*2004-2345	Posting critical plant equipment signs process needs to be formalized
2004-2505	Purchase Order VY018176 to Peerless Pump specified 1760 RPM, per 5920-3259 drawing, for the Electric Driven Fire Pump P-40-1B
2004-2571	SLC Valves identified to have slight process leaks to atmosphere
2004-2646	Potential decrease in margin exists between the pump discharge pressure and the pump discharge relief valve setpoint on SLC Pump "A" (P-45-1A")
2004-2859	RWCU isolation on high area temperature
*2004-2927	Low "B" EDG jacket cooling expansion tank level

*Inspector-identified issues.

Section 4OA2.2: Annual Sample Review of MSIV Leakage Rate Testing Failures

1998-0476	Hot as-found MSIV LLRTs did not meet TS acceptance criteria
1999-1537	Stem scratch identified on MSIV during PM, valve passed as-found LLRT
2001-0888	MSIV 80D local leak rate test results exceed the acceptance criteria
2002-2211	MSIV 80B local leak rate test results exceed the acceptance criteria
2002-2212	MSIV 86B local leak rate test results exceed the acceptance criteria
2003-2139	Scored stem and minor packing leak on MSIV 86D
2003-2165	Visual inspection identified minor stem scoring on inboard MSIVs 80A and 80B
2004-0836	MSIV A Outboard valve (86A) failed leak rate testing
2004-0838	MSIV B Outboard valve (86B) failed leak rate testing
2004-0839	MSIV C Outboard valve (86C) failed leak rate testing
2004-0841	MSIV B Inboard valve (80B) failed leak rate testing
2004-0918	Adverse trend - MSIV Appendix J test failures

2004-0955	As-found condition of V2-80B (inboard MSIV) included a galled stem

LIST OF ACRONYMS

ΔCDF	Change in Core Damage Frequency
ΔLERF	Change in Large Early Release Frequency
AC	Alternating Current
ADAMS	Automated Document Access Management System
ALARA	As Low As Is Reasonably Achievable
ANS	American National Standard
AP	Vermont Yankee Administrative Procedure
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
CR	Condition Report
DP	Vermont Yankee Department Procedure
EDG	Emergency Diesel Generator
ENN-OM	Entergy Nuclear Northeast Organization and Management Procedure
EOP	Emergency Operating Procedure
FA	Fire Area
FZ	Fire Zone
HPCI	High Pressure Coolant Injection
IBEW	International Brotherhood of Electrical Workers
IMC	Inspection Manual Chapter
IPEEE	Individual Plant Examination External Events
IR	Inspection Report
KV	Kilovolt
LER	Licensee Event Report
LLRT	Local Leakage Rate Testing
MAP	Management Alternative Plan
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
ODCM	Offsite Dose Calculation Manual
OE	Operating Experience
OP	Vermont Yankee Operating Procedure
PI	Performance Indicator
PMT	Post Maintenance Testing
PT	Potential Transformer
QA	Quality Assurance
RCIC	Reactor Core Isolation Cooling
RFO	Refueling Outage
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RMS	Radiation Monitoring System

RWCU	Reactor Water Cleanup
RWP	Radiation Work Permit
SCFH	Standard Cubic Feet Per Hour
SCFM	Standard Cubic Feet Per Minute
SDP	Significance Determination Process
SLC	Standby Liquid Control
SRA	Senior Reactor Analyst
TM	Temporary Modification
TPCS	Transient Without Power Conversion System
TS	Technical Specifications
UE	Unusual Event
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VY	Vermont Yankee