UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

November 7, 2007

Mr. Theodore A. Sullivan
Site Vice President
Entergy Nuclear Operations, Inc.
Vermont Yankee Nuclear Power Station
320 Governor Hunt Road
Vernon, VT 05354

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000271/2007004

Dear Mr. Sullivan:

On September 30, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Vermont Yankee Nuclear Power Station. The enclosed report documents the inspection results which were discussed on October 9, 2007, with you and members of your staff.

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

This report documents two self-revealing findings of very low safety significance (Green). One of these findings was determined to involve a violation of NRC requirements. In addition, a licensee-identified violation of very low safety significance is listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest either of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 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 the

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Sincerely,
/RA by Ronald R. Bellamy For/

Raymond J. Powell, Chief
Projects Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 05000271/2007004
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.:	05000271/2007004
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Vermont Yankee Nuclear Power Station
Location:	320 Governor Hunt Road Vernon, Vermont 05354-9766
Dates:	July 1, 2007 through September 30, 2007
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Approved by:	Raymond J. Powell, Chief Projects Branch 5 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000271/2007004; 07/01/07 - 09/30/07; Vermont Yankee Nuclear Power Station; Maintenance Effectiveness.

This report covered a 13-week period of inspection by resident and region-based inspectors. Two Green self-revealing findings, one of which was a non-cited violation (NCV), 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 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. A self-revealing NCV of TS 6.4, “Procedures,” was identified for Entergy’s failure to effectively incorporate readily available industry operating experience (OE) into the cooling tower (CT) inspection program and processes. Specifically, Entergy had not recognized the importance of performing hands-on inspections of CT structural members which were located in heavily loaded and normally inaccessible areas for detecting degraded conditions. As a result of not performing adequate inspections of “B” and “C” columns within the fill area, a partial collapse of non-safety CT cell 2-4 occurred on August 21, 2007. This event resulted in Entergy rapidly reducing reactor power from 100 percent to approximately 35 percent power, although no significant degradation of the safety-related cell was identified.

The performance deficiency is that Entergy did not incorporate readily available OE into the CT inspection process and procedures to detect degraded structural components. This finding is more than minor because it is associated with the Equipment Performance attribute of the Initiating Events Cornerstone and affects the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors conducted a Phase 1 screening in accordance with IMC 0609, Appendix A. The finding was determined to be of very low safety significance because it did not contribute to both the likelihood of a reactor scram and the likelihood that mitigating equipment or functions would not be available. This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Operating Experience, because Entergy did not implement and institutionalize OE through changes to station processes and procedures for the CT, as appropriate. [P.2(b)] (Section 1R12.1)

Green. A self-revealing Finding of very low safety significance was identified for Entergy's failure to specify adequate preventive maintenance (PM) for main turbine stop valve number two (TSV-2). As a result, during troubleshooting activities on the TSV-2 bypass control mechanism, a reactor scram occurred when all four main turbine stop valves closed.

The performance deficiency is the failure of Entergy to have an adequate PM strategy for TSV-2. This finding is more than minor because it is associated with the Equipment Performance attribute of the Initiating Events Cornerstone and affects the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, during troubleshooting of the TSV-2 bypass control mechanism, an automatic reactor scram occurred. The inspectors conducted a Phase 1 screening of the finding in accordance with IMC 0609, Appendix A. The finding was determined to be of very low safety significance because it did not contribute to both the likelihood of a reactor scram and the likelihood that mitigating equipment or functions would not be available. The finding has a cross-cutting aspect in the area of Human Performance, Resources component, because Entergy did not maintain an effective PM program for TSV-2. Specifically, the PM for TSV-2 did not specify a periodic activity to inspect, rebuild, and lubricate the bell crank assembly portion of the bypass control mechanism. [H.2(c)] (Section 1R12.2)

B. Licensee-Identified Findings

One violation of very low safety significance (Green), which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee'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 (VY) Nuclear Power Station began the inspection period operating at 100 percent power. On August 4, 2007, power automatically decreased to approximately 80 percent when the "C" reactor feed pump unexpectedly tripped. Entergy determined that the feed pump tripped as a result of blockage in the system's feed flow sensing line. Following repairs, operators returned the "C" feed pump to service and increased reactor power to 100 percent on August 5.

On August 21, operators reduced power to approximately 35 percent in response to a partial collapse of the non-safety related, forced draft, west cooling tower (CT) cell 2-4. Following the event, operators secured circulating water to the west CT to investigate and repair. From August 21 through August 30, power was maintained between approximately 35 to 65 percent.

On August 30, an automatic reactor scram occurred from approximately 63 percent power after all four main turbine stop valves unexpectedly closed during stop valve maintenance activities. On September 1, the reactor was restarted and placed on the grid after the cause of the scram had been identified and corrected. Operators increased power to approximately 50 percent on September 2. From September 2 until September 15, power was maintained between approximately 50 to 70 percent.

On September 15, the west CT was placed back in service. Power was subsequently increased to 100 percent on September 16. On September 16, a downpower to approximately 70 percent was performed for a planned control rod pattern adjustment. On September 17, the plant was restored to approximately 100 percent power, where it remained for the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope (1 sample)

The inspectors reviewed Entergy's measures to ensure the readiness of the circulating water, service water (SW), and alternate cooling (ACS) systems for summer weather. The intent of the inspection was to ensure adequate cooling remained available for both the non-safety related main condenser and for the safety-related SW and ACS systems during peak summer temperatures. The inspectors performed walkdowns of accessible portions of the cooling water systems to determine if they were properly aligned and maintained, in accordance with station procedures. The inspectors reviewed heat sink temperature and main condenser back pressure data to determine if adequate cooling was available. In addition, the inspectors reviewed condition reports (CRs) to determine if Entergy had properly identified and addressed known deficiencies. A listing of documents reviewed is provided in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

a. Inspection Scope (7 evaluations and 15 applicability determinations)

The inspectors reviewed safety evaluations to determine if changes and tests were evaluated and documented in accordance with 10 CFR 50.59; and, if required, Entergy obtained NRC approval prior to implementation. The inspectors assessed the adequacy of the safety evaluations through interviews with Entergy personnel and review of supporting information, such as calculations, engineering analyses, design change documentation, the Updated Final Safety Analysis Report (UFSAR) and Technical Specifications (TS). In addition, the inspectors reviewed the administrative procedures that control the screening, preparation, and issuance of the safety evaluations to evaluate whether the procedures adequately implemented the requirements of 10 CFR 50.59, "Changes, Tests, and Experiments." The inspectors also reviewed changes that Entergy had evaluated, using their applicability determination process, and determined that safety evaluations were not required. The inspectors performed this review to assess Entergy's conclusions with respect to 10 CFR 50.59 applicability. The safety evaluations, applicability determinations, and screenings were selected based on the safety significance of the affected structures, systems, and components (SSCs). The inspectors also reviewed issues that had been entered into the corrective action program to determine whether Entergy had been effective in identifying and resolving problems associated with the 10 CFR 50.59 safety evaluation process. A listing of the safety evaluations, safety evaluation screenings, applicability determinations, 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)

.1 Partial Equipment Alignment (71111.04Q)

a. Inspection Scope (3 samples)

The inspectors performed partial system walkdowns of the following risk-significant systems to evaluate the 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 applicable Entergy operating procedures (OP) and piping and instrumentation drawings (P&IDs). The inspectors observed valve positions, the availability of power supplies, and the general condition of selected components. A listing of documents reviewed is provided in the Attachment of this report.

- Diesel-driven fire pump while the “C” SW pump was out of service for emergent repairs;
- “A” Residual Heat Removal (RHR) and “A” RHR Service Water (RHRSW) systems while the “B” RHR was out of service for planned maintenance; and
- SW system while the ACS was inoperable with the deep basin drained.

b. Findings

No findings of significance were identified.

.2 Complete Equipment Alignment (71111.04S)

a. Inspection Scope (1 sample)

The inspectors performed a complete equipment alignment inspection of the accessible portions of the SW system. The inspectors compared the actual system configuration to approved P&IDs, the UFSAR, the SW system design basis document (DBD), OPs, and vendor manual requirements. The inspectors reviewed required support systems and ancillary equipment to determine if they interfered with the operation of system valves and if pump oil levels were maintained in the nominal band, and whether deficiencies had been entered into the corrective action program. The inspectors evaluated whether major system components were properly ventilated, and whether hangers and supports were correctly installed and functional. In addition, the inspectors evaluated a sample of previously identified deficiencies to determine if they had been properly addressed, and whether open items impacted system operability. A listing of documents reviewed is provided in the Attachment of this report.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope (12 samples)

The inspectors selected fire areas (FAs) important to plant risk, based on a review of Entergy’s Safe Shutdown Capability Analysis, the Fire Hazards Analysis, and the Individual Plant Examination External Events (IPEEE). The inspectors toured these areas to determine 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. A listing of documents reviewed is provided in the Attachment of this report. The following FAs and fire zones (FZs) were inspected:

- High pressure coolant injection (HPCI) corner room (FZ RB1);
- Reactor core isolation cooling (RCIC) corner room, 232 foot elevation (FZ RB1S);
- “B” emergency core cooling system (ECCS) corner room (FZ RB2);
- Reactor building, 318 foot elevation (FZ RB7);

- Reactor building, 345 foot elevation (FZ RB7);
- Circulating water pump room (FZ 14);
- Service water pump room (FZ 15);
- West cooling tower (FA 16);
- RCIC corner room, 213 foot elevation (FA RCIC);
- Discharge structure (No fire designation);
- Alternate offgas (AOG) building (No fire designation); and
- East Cooling Tower (No fire designation).

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 internal flooding in the Reactor Building 318 foot elevation. The inspectors reviewed internal flooding design information associated with this location in Entergy's IPEEE, the UFSAR, and in the Internal Flooding DBD. The inspectors conducted a walkdown of the area to determine if equipment and structures needed to mitigate an internal flooding event were available, as described in Entergy design documents. A listing of documents reviewed is provided in the Attachment of this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

a. Inspection Scope (1 sample)

The inspectors observed a simulator-based licensed operator requalification annual exam on August 8, 2007. The inspectors evaluated crew performance in the areas of clarity and formality of communications; ability to take timely actions; prioritization, interpretation, and verification of alarms; procedure usage; control board manipulations; and command and control. Crew performance in these areas was compared to Entergy management expectations and guidelines as presented in Administrative Procedure (AP) 0151, "Responsibilities and Authorities of Operations Department Personnel," AP 0153, "Operations Department Communication and Log Maintenance," and Department Procedure (DP) 0166, "Operations Department Standards." The inspectors also compared the simulator configuration with the actual control board configuration. Finally, the inspectors observed the Entergy evaluators discuss identified weaknesses with the crew and/or individual crew members, as appropriate. A listing of documents reviewed is provided in the Attachment of this report.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope (3 samples)

The inspectors reviewed Entergy's evaluation of degraded conditions, involving safety-related SSCs for maintenance effectiveness during this inspection period. The inspectors reviewed Entergy's implementation of the Maintenance Rule, 10 CFR 50.65, to determine if the conditions associated with the referenced CRs were appropriately evaluated against applicable Maintenance Rule functional failure criteria, as found in Entergy scoping documents and procedures. The inspectors discussed these issues with the system engineers and Maintenance Rule coordinators to determine if they were appropriately tracked against each system's performance criteria and that the systems were appropriately classified in accordance with Maintenance Rule implementation guidance. Documents reviewed during the inspection are listed in the Attachment. The following issues and/or systems reviewed were:

- RCIC outboard steam isolation valve, V13-16, failed local leakage rate test;
- Partial collapse of CT cell 2-4; and
- Reactor scram during #2 main turbine stop valve troubleshooting.

b. Findings

.1 Inadequate Inspection Program Resulted in a Partial Collapse of Cooling Tower Cell 2-4

Introduction: A self-revealing non-cited violation (NCV) of TS 6.4, "Procedures," was identified for Entergy's failure to effectively incorporate readily available industry operating experience (OE) into the CT inspection program and processes. Specifically, Entergy had not recognized the importance of performing hands-on inspections of CT structural members which were located in heavily loaded and normally inaccessible areas for detecting degraded conditions. As a result of not performing adequate inspections of "B" and "C" columns within the fill area, a partial collapse of non-safety CT cell 2-4 occurred on August 21, 2007. This event resulted in Entergy rapidly reducing reactor power from 100 percent to approximately 35 percent power.

Description: On August 21, 2007, a portion of CT cell 2-4 collapsed while the plant was operating at full power. To maintain State thermal discharge requirements, Entergy reduced power from 100 percent to approximately 35 percent. The inspectors' assessment of Entergy's performance in response to the event is in Section 4OA3.3 of this report.

There are two cooling towers at Vermont Yankee; each tower contains 11 cells. The non-safety function of the cooling towers is to reduce the temperature of the water used to cool the plant before it is returned to the Connecticut River. Only one cell (cell 2-1) is safety-related. The safety-related function of cell 2-1 is to provide an alternate means to remove reactor heat following a plant shutdown in the unlikely event that the normal heat removal system becomes unavailable. Adjacent cell 2-2 is not safety-related, but is structurally robust and is designed to physically separate and protect cell 2-1.

Entergy completed a root cause analysis (RCA) to identify the causes associated with the partial cell collapse. Entergy determined that root causes included both a mechanistic and programmatic aspect:

- Mechanistic Aspect - Entergy concluded that a number of the wooden 4x4 “B” vertical columns located inside the fill area of cell 2-4 had failed prior to the cell collapsing. The columns were already heavily loaded from the weight of the water distribution deck and the circulating water header. The columns were further weakened due to stresses from a chemical iron-salt attack related to iron bolting used to connect the wooden columns, a biological fungal attack in areas that the wooden columns had been affected by the iron-salt, and from over-tightened bolts at spliced locations. The column failures caused the distribution deck to sag and the water header to separate; this resulted in additional water on the distribution deck, which increased the loading on the support columns. The added weight on columns that were already stressed caused additional columns to buckle, resulting in the collapse of a portion of the cell.

- Programmatic Aspect - The CT inspection program did not require inspections of the “B” and “C” columns in the normally inaccessible fill area. Specifically, Entergy had routinely performed remote boroscopic and/or visual inspections, but had not recognized the importance of utilizing hands-on inspections techniques to detect degraded structural conditions, such as iron-salt and fungal attack or over-tightened bolts at spliced locations.

Entergy’s RCA team reviewed internal and external OE reports related to CT structural events, which shared a common theme related to the importance of performing detailed inspections in the normally inaccessible fill areas. Additionally, the OE stated that traditional inspection methods of looking for signs of wood distress were not sufficient to identify degraded columns. Entergy identified that, although the OE had been received from offsite sources and Nuclear Electric Insurance Limited reports, no actions had been developed for Entergy’s CT inspection programs. The inspectors determined that multiple opportunities existed for Entergy to identify and incorporate the recommendations of the external and internal OE into the CT inspection process.

The performance deficiency is that Entergy did not incorporate readily available OE into the CT inspection process and procedures to detect degraded structural components.

Analysis: This finding is more than minor because it is associated with the Equipment Performance attribute of the Initiating Events Cornerstone and affects the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors conducted a Phase 1 screening of the finding in accordance with Inspection Manual Chapter (IMC) 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations.” The finding was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor scram and the likelihood that mitigating equipment or functions would not be available.

This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Operating Experience, because Entergy did not implement and institutionalize OE through changes to station processes and procedures for the CTs, as appropriate. [P.2(b)]

Enforcement: TS 6.4, "Procedures," requires, in part, that written procedures be established, implemented, and maintained for "preventive and corrective maintenance operations which could have an effect on the safety of the reactor." Although no significant degradation of the safety-related cell was identified, there are substantive differences between the construction and maintenance of the safety-related and non-safety related cells, aspects of the inspection inadequacies associated with cell 2-4 are applicable to safety-related cell 2-1. As such, contrary to the above, work orders (WOs) which implemented vendor manual VY-EM-0146, "Cooling Tower - Operation, Maintenance, Repair, and Inspections CT 2-1," did not incorporate readily available OE into the CT inspections to detect degraded structural components (i.e., the WOs did not specify periodic fill removal and/or hands on inspection of timbers). Corrective actions, taken or planned, included a physical inspection of both CTs, focusing on "B" and "C" columns, and the repair of identified structural components; the incorporation of OE into the procedures for future inspections; ensuring all identified cooling tower deficiencies are documented into the corrective action program; and completion of a corrective action effectiveness review within one year. Because this issue is of very low safety significance (Green) and has been entered into Entergy's corrective action program (CR-2007-03243), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **NCV 05000271/2007004-01, Inadequate Inspection Program Resulted in the Partial Collapse of a Non-Safety-Related Cooling Tower Cell**

.2 Reactor Scram During Main Turbine Stop Valve Troubleshooting

Introduction: A self-revealing Finding of very low safety significance (Green) was identified for Entergy's failure to specify adequate preventive maintenance (PM) for main turbine stop valve number two (TSV-2). As a result, during troubleshooting activities on the TSV-2 bypass control mechanism, a reactor scram occurred when all four main turbine stop valves closed.

Description: On August 30, 2007, Entergy and vendor personnel were troubleshooting the failure of TSV-2 to re-open during TSV surveillance testing. During the troubleshooting, the bell crank assembly portion of the bypass control mechanism was observed to be sticking and not allowing TSV-2 to re-open. When the linkage was depressed, the assembly that had been sticking was suddenly freed, causing the TSV to open faster than normal. This caused a hydraulic pressure transient condition in the emergency trip oil system, which caused all four TSVs to close, and resulted in a reactor scram. The plant responded to the TSV closure and reactor scram as designed. An assessment of operator performance in response to the event is documented in Section 4OA3.4 of this report.

The function of the TSVs is to close on a turbine trip condition. The valves also provide a signal to the reactor protection system if 3 or more of the TSVs are less than 90 percent open with the reactor greater than 25 percent power, as occurred during this event. The design of TSV-2 is different from the other three TSVs. TSV-2 has an internal bypass valve which allows for turbine warmup, and an associated bypass control mechanism.

Entergy determined the root cause of the event to be the lack of a PM activity to inspect, rebuild, and lubricate the bell crank assembly portion of the bypass control mechanism. Although the 18 month PM, along with other maintenance, was performed on the TSV-2 bypass control mechanism in the May 2007 refueling outage, this did not include disassembly and inspection of the bell crank mechanism. The failure of Entergy to have an adequate PM strategy for TSV-2 constituted a performance deficiency.

Analysis: This finding is more than minor because it is associated with the Equipment Performance attribute of the Initiating Events Cornerstone and affects the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, during troubleshooting of the degraded condition of the TSV-2 bypass control mechanism, an automatic reactor scram occurred. The inspectors conducted a Phase 1 screening of the finding in accordance with IMC 0609, Appendix A. The finding was determined to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor scram and the likelihood that mitigating equipment or functions would not be available.

The finding has a cross-cutting aspect in the area of Human Performance, Resources component, because Entergy did not maintain an effective PM program for TSV-2. Specifically, the PM for TSV-2 did not specify a periodic activity to inspect, rebuild, and lubricate the bell crank assembly portion of the bypass control mechanism. [H.2(c)]

Enforcement: No violation of regulatory requirements occurred, because the event involved non-safety related systems. Entergy entered this condition into their corrective action program (CR-2007-3349). Corrective actions identified in the root cause report include creation of a PM activity to rebuild the bypass control mechanism; inspection, rebuild, and lubrication of the TSV-2 bell crank and bypass control assemblies during the next refueling outage; and a review of turbine valve PMs to ensure linkages and bell cranks are properly lubricated and rebuilt at intervals. Since this issue is of very low safety significance and has been entered into Entergy's corrective action program, it is classified as a finding. **FIN 05000271/2007004-02, Reactor Scram During Troubleshooting Due to Inadequate Main TSV Preventive Maintenance**

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

a. Inspection Scope (6 samples)

The inspectors evaluated on-line risk management for four planned maintenance activities and two emergent repair activities. The inspectors reviewed maintenance risk evaluations, work schedules, corrective actions, and control room logs to determine if concurrent or emergent maintenance activities significantly increased the plant risk. The inspectors compared reviewed items and activities to requirements listed in AP 0125, "Plant Equipment," and AP 0172, "Work Schedule Risk Management - Online." A list of documents reviewed is provided in the Attachment of this report. The inspectors walked down areas of the plant containing equipment that was determined to have higher risk significance during the following work activities:

- Planned maintenance on the “B” RHR system and orange risk configuration;
- Planned maintenance on the “C” RHR pump discharge check valve and yellow risk configuration;
- Planned ACS deep basin draining and inspection and yellow risk configuration;
- Planned maintenance for the week of August 20 and green risk configuration;
- Emergent work associated with the “C” SW pump and yellow risk configuration; and
- Emergent work associated with the “A” SW pump and yellow risk configuration.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (5 samples)

The inspectors reviewed five operability evaluations associated with degraded or non-conforming conditions to ensure that operability and functionality was justified. The inspectors evaluated the operability evaluations against the guidance contained in NRC Regulatory Issue Summary 2005-20, Revision to Guidance Formerly Contained in NRC Generic Letter 91-18, “Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability,” as well as Entergy procedure ENN-OP-104, “Operability Determinations.” A listing of documents reviewed is provided in the Attachment of this report. The inspectors also discussed the conditions with operators and system and design engineers, as necessary. The following evaluations were reviewed.

- Leak in “C” SW discharge piping;
- Fire header pressure lowered unexpectedly;
- Condensate storage tank (CST) temperature high out of band;
- CST leakage; and
- Cooling tower cell 2-1 operability

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17B)

a. Inspection Scope (9 samples)

The inspectors reviewed nine permanent plant modification packages to determine if the design bases, licensing bases, and performance capability of risk significant SSCs had been degraded through plant modifications. The inspectors performed walkdowns of selected plant systems and components, interviewed plant staff, and reviewed applicable documents, including procedures, calculations, modification packages, engineering evaluations, drawings, corrective action program documents, the UFSAR, and TS.

The inspectors reviewed selected attributes (component safety classification, energy requirements supplied by supporting systems, seismic qualification, instrument setpoints, uncertainty calculations, electrical coordination, electrical loads analysis, and equipment environmental qualification) to determine if they were consistent with the design and licensing bases. Design assumptions were reviewed to evaluate if they were technically appropriate and consistent with the UFSAR. For each modification, the 10 CFR 50.59 screenings or evaluations were reviewed, as described in section 1R02 of this report. The inspectors reviewed procedures, calculations, and the UFSAR to determine if they were properly updated with revised design information and operating guidance. The inspectors also reviewed the as-built configuration to determine if it was accurately reflected in the design documentation and that post-modification testing was adequate to ensure the SSCs would function properly. The inspectors also reviewed issues that had been entered into the corrective action program to determine whether Entergy had been effective in identifying and resolving problems associated with the plant modification process and activities. A listing of modifications and documents reviewed is provided in the Attachment to this report.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope (5 samples)

The inspectors reviewed post-maintenance testing (PMT) activities on risk-significant systems. The inspectors either observed the PMT or reviewed completed PMT documentation to evaluate if the test data met the required acceptance criteria contained in the applicable WO, TS, UFSAR, and/or inservice 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 verified that the test activities were adequate to ensure system operability and functional capability following maintenance, systems were properly restored following testing, and any discrepancies were appropriately documented in the corrective action program. The inspectors reviewed the PMTs performed for the following maintenance activities:

- Testing of the “B” RHR pump suction from recirculation loop “A” valve (V10-15B), PMT in accordance with WO 51191102;
- Mechanical troubleshooting #2 main TSV, PMT in accordance with WO 121155;
- Inspection and repair of “C” RHR pump discharge check valve (V10-48C), PMT in accordance with WO 51072384;
- Repair of “C” SW discharge pressure indicator piping leak, PMT in accordance with WO 115614; and
- Replacement of control rod hydraulic control unit (HCU) 06-31 accumulator, PMT in accordance with WO 51079490.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope (1 sample)

The inspectors evaluated the below forced outage activities, after the August 30, 2007, scram, to determine if Entergy considered risk when developing outage schedules; adhered to administrative risk reduction methodologies for plant configuration control; and adhered to their operating license, TS requirements, and approved procedures:

- Monitoring of Shutdown Activities - The inspectors observed the shutdown of the reactor plant. The plant remained in hot shutdown throughout the forced outage;
- Control of Outage Activities - The inspectors reviewed the daily shutdown risk assessment and work schedule to evaluate if Entergy appropriately addressed risk; and
- Startup Activities - The inspectors observed portions of the startup of the reactor plant, including criticality and power increase, following the completion of the forced outage.

The inspectors also evaluated whether Entergy identified problems related to the forced outage and entered them into their corrective action program. Specifically, the inspectors evaluated if Entergy identified the cause of the scram and took appropriate corrective actions prior to restarting the reactor. A listing of documents reviewed is provided in the Attachment of this report.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (5 samples)

The inspectors observed surveillance testing to determine if the test acceptance criteria specified for each test was consistent with TS and UFSAR requirements, the test was performed in accordance with the written procedure, the test data was complete and met procedural requirements, and the system was properly returned to service following testing. The inspectors observed selected pre-job briefs for the test activities. The inspectors also determined if discrepancies were appropriately documented in the corrective action program. A listing of documents reviewed is provided in the

Attachment of this report. The inspectors verified that the following surveillance testing activities met the above requirements:

- “B” emergency diesel generator (EDG) monthly slow start operability test in accordance with OP 4126, Section B;
- RCIC comprehensive testing in accordance with OP 4121, Section C;
- “A” EDG six month fast start operability test in accordance with OP 4126, Section F;
- “A” core spray (CS) pump quarterly operability test in accordance with OP 4123, Section C; and
- “A” CS quarterly motor operated valve/injection check valve closure test in accordance with OP 4123, Section B.

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) EC-2428, “Temporary Modification to Eliminate Ground On DC-1." The inspectors reviewed the TM package to determine if it adversely affected the availability, reliability, or functional capability of any risk-significant SSCs. The inspectors reviewed the installation and control of the TM against the TS, UFSAR, and Entergy Corporate Procedure ENN-DC-136, “Temporary Alterations,” to evaluate if the license and design basis were properly maintained. A listing of documents reviewed is provided in the Attachment of this report.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope (13 samples)

During September 10-13, 2007, the inspector conducted the following activities to determine if Entergy was properly implementing 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, TS, and Entergy procedures. A listing of documents reviewed is provided in the Attachment of this report.

The inspector walked down exposure significant work areas of the plant (reactor building, turbine building, radwaste building, and auxiliary off-gas building) and reviewed Entergy controls and surveys to determine if Entergy surveys, postings, and barricades were acceptable and in accordance with regulatory requirements.

The inspector walked down exposure significant work areas of the plant and, with the assistance of a radiation protection technician, conducted independent surveys to determine whether prescribed radiation work permit and procedural controls were in place and whether Entergy surveys and postings were complete and accurate.

The inspector reviewed Entergy's physical and programmatic controls for highly activated materials stored underwater in the spent fuel pool and evaluated through observation and a review of the applicable access control procedure.

The inspector reviewed Entergy radiation protection program self-assessments and audits during 2007 to determine if identified problems were entered into the corrective action program for resolution.

Ten CRs associated with the radiation protection access control and as low as reasonably achievable (ALARA) areas between May 2007 and August 2007, were reviewed and discussed with Entergy staff to determine if the follow-up activities were being conducted in an effective and timely manner commensurate with their safety significance. Based on the CRs reviewed, repetitive deficiencies were screened to determine if the Entergy's self-assessment activities were identifying and addressing these deficiencies.

There were no Occupational Exposure Performance Indicator incidents during the current assessment period.

Changes to the high radiation area and very high radiation area procedures since the last inspection in this area were reviewed and management of these changes was discussed with the Radiation Protection Manager.

Controls associated with potential changing plant conditions, to anticipate timely posting and controls of radiation hazards, were discussed with a radiation protection supervisor.

All accessible locked high radiation area entrances in the plant were examined to determine if they were locked, including the key inventory. Also, the inspector reviewed the controls for the locked and very high radiation area keys.

Several CRs were reviewed to evaluate if the incidents were caused by radiation worker errors and to determine if there were any trends or patterns and if the Entergy's corrective actions were adequately addressing these trends.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope (2 samples)

During September 10-13, 2007, the inspector conducted the following activities to determine if Entergy was properly maintaining individual and collective radiation exposures ALARA. Implementation of the ALARA program was reviewed against the criteria contained in 10 CFR 20.1101(b) and Entergy procedures. A listing of documents reviewed is provided in the Attachment of this report.

The process for adjusting work activity exposure estimates was evaluated for emergent work and unexpected radiological conditions. The methodology for the exposure estimate adjustments was evaluated with respect to radiation protection and ALARA principles.

There was one declared pregnant worker during the current assessment period. The applicable personnel exposure results and monitoring controls employed by Entergy were reviewed with respect to the requirements of 10 CFR 20.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety (PS)

2PS3 Radiological Environmental Monitoring Program (REMP) (71122.03)

a. Inspection Scope (10 samples)

The inspector reviewed the current Annual Radiological Environmental Operating Report, and Entergy assessment results, to determine if the REMP was implemented as required by TS and the offsite dose calculation manual (ODCM). The review included changes to the ODCM with respect to environmental monitoring commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, interlaboratory comparison program, and analysis of data. The inspector reviewed the ODCM to identify environmental monitoring stations. In addition, the inspector reviewed the following: Entergy self-assessments and audits, event reports, interlaboratory comparison program results, the UFSAR for information regarding the environmental monitoring program and meteorological monitoring instrumentation, and the scope of the audit program to determine if it met the requirements of 10 CFR 20.1101.

The inspector walked down seven air particulate and iodine sampling stations; one south river water sampling station; three dairy farms; and 25 thermoluminescent dosimeter (TLD) monitoring locations to determine if they were located as described in the ODCM, and to evaluate if the equipment material condition was acceptable.

The inspector observed the collection and preparation of a variety of environmental samples to determine if the environmental sampling was representative of the release pathways as specified in the ODCM, and if sampling techniques were in accordance with procedures.

Based on direct observation and review of records, the inspector evaluated whether the primary and backup meteorological tower instruments were operable, calibrated, and maintained in accordance with the guidance contained in the UFSAR, NRC Safety Guide 23, and Entergy procedures. The inspector reviewed the meteorological data readout and recording instruments in the control room and at the tower to determine if they were operable.

The inspector reviewed each event documented in the Annual REMP which involved a missed sample, inoperable sampler, lost TLD, or anomalous measurement for the cause and corrective actions. The inspector conducted a review of Entergy's assessment of any positive sample results.

The inspector reviewed any significant changes made by Entergy to the ODCM as the result of changes to the land census or sampler station modifications since the last inspection. The inspector also reviewed technical justifications for any changed sampling locations to determine if Entergy performed the required reviews to ensure that the changes did not affect its ability to monitor the impacts of radioactive effluent releases on the environment.

The inspector reviewed the calibration and maintenance records for air samplers. The inspector reviewed the following: the results of Entergy's interlaboratory comparison program to evaluate the adequacy of environmental sample analyses performed by Entergy, the quality control evaluation of the interlaboratory comparison program and the corrective actions for any deficiencies, the determination of any bias to the data and the overall effect on the REMP, and quality assurance audit results of the program to determine whether Entergy met the TS/ODCM requirements. The inspector evaluated the detector sensitivities for counting samples to determine if they in accordance with the TS/ODCM and reviewed the results of the quality control program, including the interlaboratory comparison program, to evaluate the adequacy of the program.

The inspector observed the radioactive material survey and release locations and inspected the methods used for control, survey, and release. The inspector also observed the performance of personnel surveying and releasing material for unrestricted use to determine if the work was performed in accordance with plant procedures.

The inspector reviewed the radiation monitoring instrumentation used for the release of material from the radiological controlled area to determine if it was appropriate for the radiation types present and was calibrated with appropriate radiation sources. The inspector reviewed Entergy's equipment to ensure the radiation detection sensitivities were consistent with the NRC guidance contained in Circular 81-07 and Information Notice 85-92 for surface contamination and HPPOS-221 for volumetrically contaminated material.

The inspector reviewed Entergy's audits and self-assessments related to the radiological environmental monitoring program since the last inspection to determine if identified problems were entered into the corrective action program, as appropriate. Selected

CRs were reviewed since the last inspection to determine if identified problems accurately characterized the causes and corrective actions were assigned to each commensurate with their safety significance. Any repetitive deficiencies were assessed to determine if Entergy's self-assessment activities were identifying and addressing these deficiencies.

A listing of documents reviewed is provided in the Supplemental Information attachment of this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope (3 samples)

Mitigating Systems Cornerstone

The inspectors reviewed Entergy submittals for the Safety System Functional Failures performance indicator (PI). The PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, and AP 0094, "NRC Performance Indicator Reporting," were used to verify the completeness of the PI data reported. The inspectors reviewed the licensee event reports (LERs), operator logs, and maintenance rule out of service logs to verify the accuracy and completeness of the PI data for the period from October 1, 2006, through June 30, 2007.

Occupational Radiation Safety Cornerstone

The inspector reviewed implementation of the Entergy's Occupational Exposure Control Effectiveness PI Program. Specifically, the inspector reviewed CRs and radiologically controlled area dosimeter exit logs for the past four calendar quarters. These records were reviewed for occurrences involving locked high radiation areas, very high radiation areas, and unplanned exposures against the criteria specified in NEI 99-02 to verify that all occurrences that met the NEI criteria were identified and reported as performance indicators.

Public Radiation Safety Cornerstone

The inspector reviewed a listing of relevant effluent release reports for the past four calendar quarters, for issues related to the RETS/ODCM Radiological Effluent PI against the criteria specified in NEI 99-02 . The inspector reviewed the monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases, quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases, and dose assessment procedures to verify the accuracy and completeness of the PI data.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, “Identification and Resolution of Problems,” the inspectors performed a screening of each item entered into Entergy’s corrective action program. This review was accomplished by reviewing printouts of each CR, attending daily screening meetings and/or accessing Entergy’s database. The purpose of this review was to identify conditions such as repetitive equipment failures or human performance issues that might warrant additional follow-up.

b. Findings

No findings of significance were identified.

.2 Annual Sample: Dropped Fuel Pin (1 Sample)

a. Inspection Scope:

The inspectors reviewed Entergy’s actions in response to the dropped fuel pin during gamma scanning on May 26, 2007. The Gamma Scan was being done on irradiated fuel rods stored in the spent fuel pool. The inspectors reviewed the CR, including the RCA, the extent of condition, the corrective actions, and the applicable operating procedures. The inspectors also interviewed the responsible reactor engineer.

b. Findings

No findings of significance were identified. The Entergy investigation, and the associated RCA report, appeared thorough. The description of the event included sufficient detail to support Entergy’s conclusions. The gamma scan equipment performed as designed to capture the dropped fuel pin, subsequent investigation of the fuel pin showed no signs of damage. All personnel associated with the event were qualified fuel handlers. The root cause of the dropped fuel pin was attributed to an inadequate process to compensate for a clamp design weakness; the clamp allowed two possible means to grab the fuel rod. The fuel handlers were not aware of this vulnerability.

4OA3 Event Followup (71153)

.1 Operator Response following a “C” Feedpump Trip

a. Inspection Scope (1 sample)

On August 4, 2007, with the plant at 100 percent power, operators responded to an unexpected trip of the “C” reactor feed pump. During the transient, an automatic reactor recirculation runback occurred reducing power to approximately 80 percent, as designed. The operators took prompt action to insert control rods, per a predetermined rapid shutdown sequence, to reduce reactor power to within the maximum extended load line limit analysis power to flow boundary, in accordance with station procedures. The inspectors evaluated the adequacy of operator actions in response to the reactor feed pump trip. A listing of documents reviewed is provided in the Attachment of this report.

b. Findings

No findings of significance were identified.

.2 Loss of Control Board Annunciators

a. Inspection Scope (1 sample)

On July 31, 2007, operators identified that a portion of control room alarm panel 9-3 would not light when tested and the associated alarm horn for the panel did not sound. The affected alarms were related to radiation process and area conditions. The resident inspectors responded to the control room to determine if operations had properly considered each alarm function lost in reference to the individual alarm response procedures. The inspectors reviewed the determination that no TS action statement, Emergency Action Level, or NRC reportability criteria had been met. Operations established compensatory actions involving hourly local monitoring of the affected radiation parameters. The inspectors reviewed the compensatory actions to evaluate if design assumptions remained valid. On August 2, 2007, after a power supply was replaced, the annunciators were returned to service and compensatory measures discontinued. A listing of documents reviewed is provided in the Attachment of this report.

b. Findings

No findings of significance were identified.

.3 Partial Collapse of a Non-Safety-Related Cell in the West Cooling Tower

a. Inspection Scope (1 sample)

On August 21, 2007, at 3:30 p.m., with the plant at 100 percent power, a non-safety related portion (cell 2-4) of the west CT collapsed. The resident inspectors responded to the control room and assessed the operator’s response to the event, and locally observed the CT structural damage. Immediately following the event, the plant continued to operate at 100 percent power and all associated safety-related SSCs remained operable. At approximately 4:00 p.m., the operators isolated circulating water flow to the CT and secured the cooling fans, the operators then reduced power to approximately 35 percent to comply with State thermal discharge limits. The inspectors observed these activities to evaluate Entergy's response to determine if it was consistent with station procedures and training. A listing of documents reviewed is provided in the Attachment of this report.

b. Findings

No findings of significance related to event response were identified. A performance deficiency associated with the cause of the unplanned power reduction is documented in Section 1R12.1.

.4 Operator Response to an Automatic Reactor Scram

a. Inspection Scope (1 sample)

On August 30, at 3:13 p.m., with the Unit at 63 percent power, all four main turbine stop valves unexpectedly closed during troubleshooting activities on TSV-2. As a result, an automatic reactor scram occurred. The resident inspectors responded to the control room to determine if operator response was in accordance with station procedures and training and that mitigating systems responded as expected. The inspectors evaluated whether Entergy appropriately considered emergency action level entry and notification requirements. In addition, the inspectors reviewed operator logs, plant computer data, CRs, and the Post Trip Review report. A listing of additional documents reviewed is provided in the Attachment of this report.

b. Findings

No findings of significance related to event response were identified. A performance deficiency associated with the cause of the scram is documented in Section 1R12.2.

.5 (Closed) LER 05000271/2007002-00, High Pressure Coolant Injection System Valve Failed to Open (1 sample)

On June 8, 2007, with the reactor at 81 percent power, Entergy identified that the HPCI pump injection valve (V23-19) did not open on a manual signal from the control room during a surveillance test. Entergy entered the condition into their corrective action program and a root cause evaluation was performed. Entergy determined that one of the motor operated valve (MOV) contacts (72/C) was in the intermediate position, causing electrical and mechanical interlocks that prevented the open contactor (82/O) from energizing. Entergy identified that the 72/C contacts were pitted and worn, causing the contact surfaces to overheat and weld together. Entergy determined that the PM performed on the valve control circuitry was inadequate, in that it did not contain sufficient guidance on how to determine contact wear and when the contacts should be replaced. The inadequate PM activity constituted a performance deficiency.

This finding is more than minor because it is associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and affects the cornerstone objective of assuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the Phase 1 screen required a Phase 2 analysis, because the inoperability of the HPCI system represented a loss of system safety function. The inspector conducted a Phase 2 SDP analysis, using the following assumptions, and the Risk-Informed Inspection Notebook for Vermont Yankee Nuclear Power Station, Revision 2: the exposure time was approximately six days and no operator recovery credit provided. Based upon the SDP Notebook Pre-Solved Excel Spreadsheet, this finding was preliminarily assessed as greater than Green.

Subsequently, a Region I Senior Reactor Analyst (SRA) performed a Phase 3 evaluation using the Vermont Yankee Standardized Plant Analysis Risk (SPAR) model using the same assumptions stated above. The SPAR model internal events risk assessment yielded a low E-7 increase in core damage frequency (CDF). Based upon the internal risk contribution being greater than 1E-7, the SRA reviewed the finding for external event risk contributors, consistent with IMC 0609, Appendix A, Attachment 3. No fire or flooding risk contributors were identified and the seismic risk contribution (principally seismically induced loss of offsite power events) was determined to be of minimal consequence (less than 1E-11). Accordingly, the total (internal and external) increase in CDF for this finding was low E-7, or very low safety significance (Green). The dominant postulated core damage scenarios involved a loss of the Division 2 Direct Current Bus followed by the failure to make-up (high pressure or low pressure injection, following depressurization) to the reactor vessel. Neither the RCIC system nor the low pressure injection systems, in conjunction with the automatic depressurization system, were adversely impacted by this finding and were available for accident mitigation while the HPCI system was degraded.

The issue was entered into Entergy’s corrective action program (CR 2007-2372). Corrective actions included: performing an extent of condition review to identify affected contacts in other systems, evaluating system operability, developing a prioritized replacement schedule based on risk significance, developing criteria for replacement during PM activities, and requiring periodic replacement of related heavily loaded MOV contactors. The inspectors reviewed the LER, root cause evaluation, extent of condition review, and corrective actions. A listing of documents reviewed is provided in the Attachment of this report.

This licensee-identified finding involved a violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings.” The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

4OA5 Other Activities

Strike Contingency Plan (92709)

a. Inspection Scope

Entergy developed a Management Alternative Plan (MAP) to ensure a sufficient number of qualified personnel were available to continue Entergy operations in the event that the International Brotherhood of Electrical Workers (IBEW) union personnel engaged in a job action upon the expiration of their contract on August 19, 2007. Using the guidance contained in 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 of the facility would be available; a review to determine if reactor operations and station security would be maintained, as required; and, a review to determine if the plan complied with TS requirements and other NRC requirements. On August 17, 2007, the contract was extended and negotiations continued until August 27, 2007, when IBEW union personnel approved a contract. No job action was taken.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On August 2, 2007, the radiation protection inspector presented the inspection results to Mr. Norman Rademacher, Director of Engineering, and other members of the VY staff. The inspectors confirmed that no proprietary information was provided or examined during the inspection.

On August 10, 2007, the permanent plant modifications inspectors presented the inspection results to Mr. William Maguire, General Manager of Plant Operations, and other members of the VY staff. The inspectors confirmed that no proprietary information was provided or examined during the inspection.

On September 14, 2007, the radiation protection inspector presented the inspection results to Mr. David Mannai, Licensing Manager, and other members of the VY staff. The inspectors confirmed that no proprietary information was provided or examined during the inspection.

On October 9, 2007, the resident inspectors presented the inspection results to Mr. Theodore Sullivan, Site Vice President, and other members of the VY staff. The inspectors confirmed that no proprietary information was provided or examined during the inspection.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by Entergy and is a violation of NRC requirements which meet the criteria in Section VI.A.1 of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV. A listing of documents reviewed is provided in the Attachment of this report.

10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” states that “Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.”

Contrary to the above, Entergy did not have a maintenance procedure describing appropriate acceptance criteria for determining when to replace the HPCI MOV contacts. As a result, HPCI was inoperable for approximately six days after MOV V23-19 failed to open during a planned surveillance. This finding was entered into Entergy’s corrective action program (CR 2007-2372). This finding is of very low safety significance (Green) as determined by a NRC Phase 3 significance determination (Section 4OA3.5).

ATTACHMENT: SUPPLEMENTAL INFORMATION

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

J. Dreyfuss, Director of Nuclear Safety
W. Maguire, General Manager of Plant Operations
D. Mannai, Licensing Manager
N. Rademacher, Director of Engineering
T. Sullivan, Site Vice President
G. Von Der Esch, Acting Operations Manager
S. Wender, Radiation Protection Manager

LIST OF ITEMS OPENED AND CLOSED

Opened and Closed

05000271/2007004-01	NCV	Inadequate Inspection Program Resulted in a Partial Collapse of a Non-Safety-Related Cooling Tower Cell (Section 1R12.1)
05000271/2007004-02	FIN	Reactor Scram During Troubleshooting Due to Inadequate Main TSV Preventive Maintenance (Section 1R12.2)

Closed

05000271/2007002-00	LER	High Pressure Coolant Injection (HPCI) System Valve Failed to Open (Sections 4OA3.5 and 4OA7)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Condition Reports
CR-2007-0292, Service Water Pump Motor Air Cooling Flow to be Found to be Less than Specified on Drawings
CR-2007-0814, “D” Service Water Pump High Motor Winding Temperatures
CR-2007-2915, Foaming of Oil in “D” Service Water Pump Sight glass
CR-2007-3070, Increase in Service Water Supply Temperature to 82.5 Degrees after Recirculation Gate was Shut

Procedures
OP 2181, Service Water/Alternate Cooling Operating Procedure, Revision 108
ON 3148, Loss of Service Water, Revision 12

Miscellaneous Documents
Circulating Water System Design Basis Document
Service Water System Design Basis Document
SYSENG Memo 2007-027, Increase in SW Supply Temperature Above Procedural Limits

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

10 CFR 50.59 Safety Evaluations
2005-01, 24V DC Power Distribution Improvements, Revision 1
2005-02, RCIC Torus Suction Valve Closure Upgrade, Revision 0
2006-01, ST1-05-VY1-0003-000 RHR Service Water Pump Hi Flow Test, Revision 0
2006-02, HPCI Manual Initiation of Auto Isolation, Revision 0
2006-03, Remove Ones Position (i.e., 01, 11, 21, 31, and 41) from CRD PIP Probe 26-07 to RPIS, Revision 0
2007-01, STP 2007-01, Hydraulic Performance Test of the ACS System, Revision 0
2007-02, OP-1403, OP-1405 and TS Basis 3.7.c, Revision 0

10 CFR 50.59 Safety Evaluation Screens
ER 04-0976, Replacement of V10-26B, Revision 0
ER 04-1336, MCC-89A and MCC-89B Refurbishment, 1/19/05
ER 05-0226, Redesign of Internal Shaft Bearings for Service Water Pump P-7-1D, 12/13/05
ER 05-0297, Fuel Oil Storage Tank Replacement Level Indicator, 9/11/05
ER 05-0316 Torus to Drywell Replacement Differential Pressure Transmitter, Revision 0
ER 05-0534, Add Flow Restrictor to RHRSW Motor Bearing Cooler Piping, Revision 0
ER 05-0687, Swap Degraded Penetration Port for Recirc MG Control Cables, 10/19/05
ER 05-0776, RFP B Auto Trip on Trip of Condensate Pump, Revision 0
ER 06-1099, Reactor Recirculation Runback Termination Point Change, Revision 0
Multiple Temporary Shielding Requests, Revision 0

10 CFR 50.59 Safety Evaluation Applicability Determinations
CR-VTY-2006-1384-CA-006, Scaffolding Not Receiving 10 CFR 50.59 Reviews, Revision 0
EC 1753, Replace DC Contactors in Local 23-19 with Stock Code 35DA123538 and Make Design of Remote Starter for V23-19 Similar to the Starter for V23-20, Revision 0
EC 748, Design Change to Replace LT-2-3-70 and Upgrade, Revision 0
OP 2143 Change 117, Revision 117, 480V and Lower Voltage AC System, Revision 0
OP 2145 Change 47, Revision 47 Normal 125 VDC Operation, Revision 0

Procedures
EN-AD-101, Procedure Process, Revision 7
EN-LI-100, Process Applicability Determination, Revision 4
EN-LI-101, 10 CFR 50.59 Review Program, Revision 3
EN-LI-113, Licensing Basis Document Change Process, Revision 1
OP-1403, Fuel Bundle Non-Destructive Testing and Reconstitution, Revision 25
OP-1405, Water Submersible Gamma Spectrometer, Revision 2

Calculations
VYC-2299, Radiological AST Fuel Handling Accident, Revision 0
VYC-2374, Suppression Pool Temperature for Appendix R without Containment Overpressure, Revision 0
VYC-2397, Maximum RHRSW Flow to RHR Heat Exchangers for Station Blackout and Appendix R, Revision 0
VYC-2398, Torus Temperature Calculation for a Station Blackout Event at Extended Power Uprate, Revision 0
VYC-2421, Suppression Pool Temperature for SRV Discharge Transients at Extended Power Uprate with Enhanced Cooling, Revision 0

Completed Tests
ER-STI-05-VY1-0003-000, RHR Service Water High Flow Test, March 24, 2006
STP 2007-001, Hydraulic Performance Test of the ACS System, Revision 0, May 22, 2007

Condition Reports
2006-1384 2007-0235

Drawings
G-191159 Sheet 1, Flow Diagram Service Water System, Revision74
G-191159 Sheet 2, Flow Diagram Service Water System, Revision89

Section 1R04: Equipment Alignment

Procedures
OP 2124, Residual Heat Removal System
OP 2181, Service Water/Alternate Cooling Operating Procedure
OP 2186, Fire Suppression Systems

Condition Reports
2007-0241, SW-FCV-28B Did Not Full Open
2007-0292, Service Water Pump Motor Air Cooling Flow to be Found to be Less than Specified on Drawings
2007-0435, Twenty-four Hour Shutdown LCO Entered for Service Water Pump Surveillance
2007-0567, Assumption in EDG DBD Reference Not in Accordance with Actual System Setpoint
2007-0814, “D” Service Water Pump High Motor Winding Temperatures
2007-1468, The P-7-IC, “C” Service Water Motor Windings are Showing Signs of Running Warmer than the “D” Service Water Motor
2007-1490, LO-OPX-2007-00077 CA-0015 Required an OE Review of the NRC Information Notice 2007-05: Vertical Draft Pump Shaft and Coupling Failures
2007-1929, SW “A” Pump Train is at Maintenance Rule Reliability Criteria
2007-2256, Vernon Tie Alternate Shutdown Capability Test Not Completed
2007-2915, Foaming of Oil in “D” Service Water Pump Sight glass
2007-3314, Engineering Change Did Not Consider Service Water Impingement When Opening SW-11

Drawings
G-191159, Flow Diagram Service Water System
G-191172, Flow Diagram Residual Heat Removal System

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SSC Performance History for Service Water, 6/30/2004-6/30/2007
EC 2646, Increase temperature limit for deep basin alignment of the SW system to 80 degrees in OP 2181

Section 1R05: Fire Protection

CR-2007-3295, Degraded Fire Barrier between Cooling Tower Cells CT 2-1 and CT 2-2
CR-2007-3427, Cigarette Butts Identified by NRC Resident Inspectors within the Fire Control Area Surrounding the West Cooling Tower (NRC identified)

Section 1R06: Flooding Protection Measures

CR-2007-2023, Water Back-Up from Floor Drain Scuppers on 318' Reactor Building, South-East Side
Vermont Yankee Nuclear Power Station Topical DBD for Internal Flooding, Revision 8

Section 1R11: Licensed Operator Requalification Program

Simulator Lesson Plan/Scenario implemented on August 8, 2007

Section 1R12: Maintenance Effectiveness

Condition Reports
CR-2007-2045, As-found local leakage rate testing (LLRT) not Performed
CR-2007-2070, Cooling Tower Structural Member Require Repair
CR-2007-2267, V13-16 Failed as-left Appendix J Local Leakage Rate Testing
CR-2007-2276, RCIC V13-16 Failed Appendix J Local Leakage Rate Testing
CR-2007-2393, RCIC - System is at Maintenance Rule Reliability Criteria
CR-2007-3223, Fan on Cooling Tower Cell 2-4 is Rubbing
CR-2007-3235, Evidence of Structural Damage/Degradation to Cooling Tower Cell 2-4
CR-2007-3243, Partial Failure of Cooling Tower Cell 2-4
CR-2007-3275, BLD - Cooling Towers Subsystem above Maintenance Rule Condition Monitoring Criteria
CR-2007-3282, Surface Degradation Noted on Column of CT-2-2
CR-2007-3314, Engineering Change did not Consider Service Water Impingement when Opening SW-11
CR-2007-3315, Degraded Conditions Resulting from Inspections in CT-2-1 and CT-2-2
CR-2007-3365, Pipe Support Discrepancies Noted in CT 1-1 & 1-2
CR-2007-3366, Degraded Conditions Identified in Cells CT 2-1 & 2-2 during Deep Basin Inspection
CR-2007-3439, “C” Column Splice Joints Installed in Wrong Location
CR-2007-3455, CT-2-2 Structural Inspection Findings
CR-2007-3550, CW Leak at North End of New Distribution Header Installed by Temp Mod EC
CR-2007-3555, Breaker 55 Trip Causing Loss of Cooling Tower Fans

CR-2007-3586, Work Order 121155, was Closed without all of the Associated Troubleshooting Documents Attached
CR-2007-3595, PI for EVENT was Red for the Month of August
CR-2007-3595, PI for Events (Human Performance) Red for Month of August because of Near Miss at the Cooling Tower

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WO 00119978-08, Structure Is Sagging and Needs Evaluation and Repair, CT-2-4
WO 00119978-09, Structure Is Sagging and Needs Evaluation and Repair, CT-2-4
WO 00120847-01, Perform Structural Repairs To Safety Class Cell 2-1, CT-2-1
WO 00120851-01, Perform Structural Repairs To Safety Class Cell 2-2, CT-2-2
WO 03-001243-003, Generic Pre-Plan For NPS Cooling Towers Repair Fall 2004, CT-2
WO 03-001243-004, Replace the Plywood and Overlay Material
WO 03-001243-008, Replace End Wall Panels on the North End Wall
WO 03-1243-09, End Wall Replacement/ Inspection
WO 05-000692-000, Perform Normal Fall Structural Repairs/Upgrades
WO 06-6156-00, Cooling Tower Structural Repairs/Upgrades
WO 51073974-01, Repair/Replace Degraded Lower West C Columns in Bents
WO 51078050-01, Drain Deep Basin, Remove Silt and Inspect Normally Submerged
WO 51078785-01, Structural Repairs As Directed By Maintenance Support, CT-2-1
WO 51078786-01, Structural Repairs As Directed By Maintenance Support, CT-2-2
WO 51079433-01, Above Waterline Cooling Tower Structural/Mechanical Inspection
WO 51079445-01, Complete Fall Mechanical Cooling Tower PM’s
WO 51080170-01, Above Waterline Cooling Tower CT-2-1 Structural/Mechanical
WO 51080285-01, Cooling Tower Structural Repairs/Upgrades
WO 51080923-01,Toeboards On Southeast End of Bravo Cooling Tower Need to be Replaced
WO 51081542-01, Above Waterline Cooling Tower Structural Inspection, CT-2

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Alternate Cooling DBD
Alternate Cooling Maintenance Rule Scoping Basis Document
Building & Structures (BLD) Maintenance Rule Scoping Basis Document
BLD 1st Quarter 2007 System Health Report
BLD SSC Performance History 8/31/2004 Through 8/31/2007
Cooling Tower Inspection Guideline
Circulating Water Maintenance Rule Scoping Basis Document
OP 4030, Types B and C Primary Containment Leakage Rate Testing
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RCIC 1st Quarter 2007 System Health Report
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RFO 24 RCIC V13-16 LLRT test results
RFO 25 RCIC V13-15 LLRT test results
RFO 26 RCIC V13-15 LLRT test results
RFO 26 RCIC V13-16 LLRT test results

Section 1R13: Maintenance Risk Assessment and Emergent Work Evaluation

Procedures
AP 0172, Work Schedule Risk Management, Revision 7

Miscellaneous
“B” RHR LCO Maintenance Plan 7/23-26/07
Risk Assessment of 7/23-26/07 “B” RHR LCO Maintenance
VYAPF 0172.02, Risk Management Worksheet for 7/10/07
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VYAPF 0172.02, Risk Management Worksheet for 8/25 - 8/27/07
VYAPF 0172.02, Risk Management Worksheet for 9/26/2007
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Section 1R15: Operability Evaluations

Condition Reports
2004-2600, Operability Statement for CR-VTY-2002-2942 was not Fully Conservative
2004-2799, OPL-3A for EPU Listed a Lower CST Temperature than Actually, Historically, Recorded on VY OPF 0150.01
2004-2793, Incomplete Corrective Action for CR-VTY-2004-2600
2007-2621, Electric Fire Pump Auto Start
2007-2728, Leak from “C” SW Pump Discharge Pipe
2007-2756, Fire Water Pressurizing Orifice Suspected of Plugging
2007-2757, Fire Header Pressure Lowered Unexpectedly during Monthly Surveillance
2007-2828, Auto Start of the Electric Fire Pump
2007-3076, Unexpected Electric Fire Pump Start with No Fire
2007-3182, Condensate Storage Tank Temperature High
2007-3264, CST leakage rate needs engineering evaluation
2007-3308, NRC Noted Cooling Tower Distribution Nozzles Partially Blocked
2007-3314, Engineering Change did not Consider Service Water Impingement when Opening SW-11 (NRC identified)
2007-3446, Three Questions from NRC Inspectors of Cooling Towers Required Disposition (NRC identified)
2007-3243, Partial Failure of Cooling Tower Cell 2-4
2007-3223, Fan on Cooling Tower Cell 2-4 is Rubbing
2007-3235, Evidence of Structural Damage/Degradation to Cooling Tower Cell 2-4
2007-3282, Surface Degradation Noted on Column of CT-2-2 (NRC identified)
2007-3315, Degraded Conditions Resulting from Inspections in CT-2-1 and CT-2-2
2007-3365, Pipe Support Discrepancies Noted in CT 1-1 & 1-2
2007-3366, Degraded Conditions Identified in Cells CT 2-1 & 2-2 during Deep Basin Inspection
2007-3425, Engineering Recommendation Considered as an Option
2007-3427, Cigarette Butts Identified during NRC Tour of West Cooling Tower
2007-3429, CT-2-4 Structural material Collection Methodology Hampered RCA Team Mechanistic Review
2007-3439, “C” Column Splice Joints Installed in Wrong Location
2007-3455, CT-2-2 Structural Inspection Findings
2007-3550, CW Leak at North End of New Distribution Header Installed by Temp Mod EC 2697 in Cooling Tower Cell 2-4

2007-3555, Breaker 55 Trip Causing Loss of cooling Tower Fans
2007-3595, PI for EVENT was Red for the Month of August

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Cooling Tower Inspection Guideline
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DP 4196, CST Leak Monitoring Program
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VY Online Schedule, Work Weeks 8/31/07-9/9/07
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WO 51078785-01, Structural Repairs As Directed By Maintenance Support, CT-2-1
WO 51078786-01, Structural Repairs As Directed By Maintenance Support, CT-2-2
WO 00119978-08, Structure Is Sagging and Needs Evaluation and Repair, CT-2-4
WO 00119978-09, Structure Is Sagging and Needs Evaluation and Repair, CT-2-4
WO 00120847-01, Perform Structural Repairs To Safety Class Cell 2-1, CT-2-1
WO 00120851-01, Perform Structural Repairs To Safety Class Cell 2-2, CT-2-2
WO 03-001243-003, Generic Pre-Plan For NPS Cooling Towers Repair Fall 2004, CT-2
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WO 03-001243-008, Replace End Wall Panels on the North End Wall
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WO 05-000692-000, Perform Normal Fall Structural Repairs/Upgrades
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WO 51080923-01,Toeboards On Southeast End of Bravo Cooling Tower Need to be Replaced
WO 51080170-01, Above Waterline Cooling Tower CT-2-1 Structural/Mechanical
WO 51079433-01, Above Waterline Cooling Tower Structural/Mechanical Inspection
WO 51080285-01, Cooling Tower Structural Repairs/Upgrades
WO 51079445-01, Complete Fall Mechanical Cooling Tower PM's
WO 51078050-01, Drain Deep Basin, Remove Silt and Inspect Normally Submerged
WO 51073974-01, Repair/Replace Degraded Lower West C Columns in Bents

Procedures
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Section 1R17: Permanent Plant Modifications

Modifications
EC 1552, MCC-8B-10DR Breaker Replacement, Revision 0
EC 1753, Local 23-19 Contactor Replacement Modification, Revision 0
ER 04-1272, RCIC Torus Suction Valve Closure Upgrade, Revision 0

ER 04-1273, HPCI Manual Initiation of Auto Isolation, Revision 0
ER 04-1337, 24V DC Power Distribution Improvements, Revision1
ER 05-0776, RFP B Auto Trip on Trip of Condensate Pump, Revision 0
ER 05-0921, HPCI Turbine Exhaust V23-3 Check Valve Modification, Revision 0
ER 05-0992, Emergency Diesel Generator KLF Loss of Field Relay, Revision 0
ER 06-1099, Reactor Recirculation Runback Termination Point Change, Revision 0

Procedures
EN-DC-112, Engineering Change Request and Project Initiation Programs, Revision 0
EN-DC-114, Project Management, Revision 4
EN-DC-115, Engineering Change Development, Revision 2
EN-DC-117, Post Modification Testing and Special Instructions, Revision 0
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ERT 04-1272-01-00, Post Installation Test of ER 04-1272, RCIC Torus Suction Valve Closure Upgrade, February 18, 2006
ERT 04-1273-01-00, Post Installation Test of ER-04-1273, HPCI Manual Initiation of Auto Isolation, January 18, 2007
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5920-1924, Elementary Wiring Diagram, RCIC System, Revision 26
5920-1925, Elementary Wiring Diagram, RCIC System, Revision 25
5920-1929, Elementary Wiring Diagram, RCIC System, Revision 9
5920-2748, Connection Diagram HPCI Relay Cabinet CRP 9-39
5920-38, Functional Control Diagram HPCI System, Revision 15
B-191031 Sheet 1146, Control Wiring Diagram Suppression Pool Upstream Isolation Valve V23-58, Revision 14
B-191031 Sheet 1182, RCIC Logic System and Turbine Trip, Revision 19
B-191031 Sheet 1195, RCIC Pump Suction from Suppression Chamber Valve V13-39, Revision 15
B-191031 Sheet 1196, RCIC Pump Suction from Suppression Chamber Valve V13-41, Revision 17

B-191031 Sheet 1451, Control Wiring Diagram HPCI Logic System, Revision 28
B-191301 Sheet 1440, Control Wiring Diagram, HPCI Pump Discharge Valve V23-19, Revision 17
B-191301 Sheet 1441, Control Wiring Diagram, Suppression Pool Upstream Isolation Valve V23-58, Revision 14
B-191301 Sheet 1451, Control Wiring Diagram, HPCI Logic System, Revision 28
B-191301 Sheet 551, Control Wiring Diagram, Reactor Feed Pump P-1-1B Stop Circuit, Revision 20
B-191301 Sheet 872, Analog Trip System 24V DC Bus A Power Supply ES-24DC-1, Revision11
B-191301 Sheet 873, Analog Trip System 24V DC Bus A Power Supply ES-24DC-2, Revision14
G-191157 Sheet 1, Flow Diagram, Condensate Feedwater and Air Evacuation Systems, Revision 62
G-191157 Sheet 3, Flow Diagram, Condensate Feedwater and Air Evacuation Systems, Revision 7
G-191169 Sheet 1, Flow Diagram, High Pressure Coolant Injection System, Revision 49
G-191169 Sheet 1, Flow Diagram High Pressure Coolant Injection System, Revision49
G-191169 Sheet 2, Flow Diagram High Pressure Coolant Injection System, Revision42
G-191174 Sheet 1, Flow Diagram, Reactor Core Isolation Cooling System, Revision 43
G-191341 Sheet 1, 480V Auxiliary One Line Diagram, Switchgear Bus 11, MCC-11B, MCC-10B, AC-DP-D1A, Revision 46
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TE 2002-054, Generic Insulation Removal Evaluation for RCIC, HPCI, and Torus Rooms
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Section 1R20: Refueling and Other Outage Activities

CR-2007-3350, Loss of Control Power to EPR
CR-2007-3352, Condensate Pump Amps Fluctuating
CR-2007-3375, APRM F Hi and Rod Block Received at 11 % during Startup with Mode Switch in RUN
CR-2007-3376, ”F” APRM Hi Alarm at 10 % Power
OP 0105, Reactor Operations, Revision 84

Section 1R22: Surveillance Testing

Condition Reports
2007-2865, “B” EDG Day Tank Level High OOS (NRC identified)
2007-2870, “B” EDG Does Not Meet Surveillance Defined Acceptance Criteria for Fuel Oil Filter Differential Pressure

Procedures
OP 4121, Reactor Core Isolation Cooling System Surveillance, Revision 77
OP 4123, Core Spray System Surveillance, Revision 43
OP 4126, Diesel Generators Surveillance, Revision 82

Miscellaneous
NRC Part 9900: Technical Guidance, Maintenance - Preconditioning of Structures, Systems and Components before Determining Operability
VY Inservice Testing (IST) Program Section SEP-IST-001, Revision 0

Section 1R23: Temporary Plant Modifications

Control Room Log, dated 8/15/2007
CR-2007-3204, 4-C-6 “Pump Motor A Oil LVL HI/LO” Alarm Cycling
Drawing B-191301, Sheet 701, Control Wiring Diagram, Revision 13
EC-2428, Temporary Modification to Eliminate Ground On DC-1
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Section OS: Occupational Radiation Safety

LO-VTYLO-2007-00068, Snapshot Self-Assessment of INPO Identified Radiation Protection Technician Deficiencies
Snap Shot Assessment of Quarterly Trend Data for Radiation Protection: 12/28/06, 12/06/07, 4/10/07, 7/10/07
Quality Assurance Surveillance Observations

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EN-RP-101, Access Control for Radiologically Controlled Areas, Revision 2
EN-RP-105, Radiation Work Permits, Revision 2
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Section 2PS3: Radiological Environmental Monitoring Program

Annual Radiological Effluent Release Reports - 2005 and 2006
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VYOPF 5335.02, Primary Meteorological System Wind Speed Monitoring Procedure
VYOPF 5335.03, Primary Meteorological System Wind Direction Monitoring Procedure
VYOPF 5335.05, Primary Meteorological System Rain Monitoring Procedure

Section 4OA2: Problem Identification and Resolution

Condition Reports
CR-2007-2065, Fuel Pin in Gamma Scan Device Released into Gamma Scan Capture Tube
CR-2007-3089, SBGT 3-B Failed to Auto Open
CR-2007-3104, Ground on DC-1
CR-2007-3128, Water was Observed Leaking through Threaded Area of Seat Ring in the "B" RHR Pump Discharge Check Valve V10-48B
CR-2007-3171, Inadvertent Trip of Plant Equipment
CR-2007-3173, An Increase in the Number of Control Rod Double Notching has been Observed that has Impacted the Reactivity Management PI and Required Further Investigation
CR-2007-3218, Ames Hill Transmitter OOS
CR-2007-3230, HPCI System Swapped Suction Paths from CST to Torus due to a Spurious CST Low Level Signal
CR-2007-3386, Excessive Vibration on "A" Recirc MG Lube Oil System
CR-2007-3552, PCIS Group 3 Actuated
CR-2007-3592, Unexpected Annunciator 50-K-3 (HVAC AH Units AH-1001A 7 1B Off)

Miscellaneous
OP 1403, Fuel Bundle Non-Destructive Testing and Reconstitution, Revision 17
OP 1405, Water Submersible Gamma Spectrometer, Revision 0

Section 4OA3: Event Followup

Condition Reports
CR-2007-2372, HPCI-19 Fails to Open
CR-2007-3023, Control Room 9-3 Annunciator Panel has a Section that does not Alarm
CR-2007-3071, Reactor Feed Pump Tripped on Spurious Low Flow
CR-2007-3088, Drawing not Revised for Installed Plant Equipment Changes
CR-2007-3224, Control Room Alarm Failure
CR-2007-3270, Misleading Wording in Root Cause Analysis (NRC identified)
CR-2007-3349, During Troubleshooting of #2 TSV Failure to Open and Plant Experience a Reactor Scram

Procedures
Alarm Response Procedures, CR 9-3
AP 0156, Notification of Significant Events, Revision 62
EOP-1, RPV Control
OP 2180, Circulating Water/Cooling Tower Operation, Revision 92
OP 4160, Turbine Generator Surveillance, Revision 49
OT 3100, Reactor Scram, Revision 8
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OT 3175, Recirculation Pump Runback due to Condensate or Feed Pump Trip, Revision 3

Miscellaneous Documents
Drawing No. 251R191, Revision E, Turbine Control Diagram
Event Notification Sheet, Reactor Scram, NRC Event Number 43610, dated 8/30/2007
LOT 00239, Main Steam System, Revision 20
LOT 00249, Mechanical Hydraulic Control System, Revision 21
Post Trip Report 2007-01

Section 4OA7: Licensee-Identified Violations

CR-2007-2372, HPCI-19 Failed to Open
CR-2007-3270, Misleading Wording in Root Cause Analysis, CR-2007-2372, HPCI MOV V23-19 Failed to Open, did not Clearly State the Duration of when all of the Safety Systems were Operable (NRC identified)

LIST OF ACRONYMS

ALARA	as low as reasonably achievable
ACS	alternate cooling system
ADAMS	Agencywide Documents Access and Management System
AOG	alternate offgas
AP	Administrative Procedure
CDF	core damage frequency
CFR	Code of Federal Regulations
CR	condition report
CS	core spray
CST	condensate storage tank
CT	cooling tower
DBD	design basis document
DP	department procedure
ECCS	emergency core cooling system
EDG	emergency diesel generator
FA	fire area
FIN	finding
FZ	fire zone
HCU	hydraulic control unit
HPCI	high pressure coolant injection
IBEW	International Brotherhood of Electrical Workers
IMC	Inspection Manual Chapter
IPEEE	individual plant examination external events
IST	inservice testing
LER	licensee event report
MAP	management alternative plan
MOV	motor operated valve
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission

ODCM	Offsite Dose Calculation Manual
OE	operating experience
OP	operating procedure
QA	quality assurance
PARS	Publicly Available Records System
PCIS	primary containment isolation system
PI	performance indicator
PI&D	piping and instrumentation drawing
PM	preventive maintenance
PMT	post maintenance testing
PS	Public Radiation Safety
QA	quality assurance
RCA	root cause analysis
RCIC	reactor core isolation cooling
REMP	radiological environmental monitoring program
RHR	residual heat removal
RHRSW	residual heat removal service water
SDP	significance determination process
SSCs	structures, systems, and components
SPAR	standardized plant analysis risk
SRA	Senior Reactor Analyst
SW	service water
TLD	thermoluminescent dosimeter
TM	temporary modification
TS	technical specification
TSV	turbine stop valve
UFSAR	Updated Final Safety Analysis Report
VY	Vermont Yankee
WO	work order