UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PA 19406

January 29, 2008

Mr. Peter T. Dietrich
Site Vice President
Entergy Nuclear Northeast
James A. FitzPatrick Nuclear Power Plant
Post Office Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000333/2007005

Dear Mr. Dietrich:

On December 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your James A. FitzPatrick Nuclear Power Plant. The enclosed inspection report documents the inspection results, which were discussed on January 9, 2008, with you and other members of your staff.

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

This report documents three self-revealing findings of very low safety significance (Green). Two of these findings were determined to be violations of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of their very low safety significance, and because they were entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a written response within 30 days of the date of this inspection report with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D. C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Senior Resident Inspector at the James A. FitzPatrick Nuclear Power Plant.

In accordance with 10 CFR Part 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the

NRC's document system (ADAMS). ADAMS is accessible from the NRC Web Site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Eugene W. Cobey, Chief
Projects Branch 2
Division of Reactor Projects

Docket No.: 50-333
License No.: DPR-59

Enclosure: Inspection Report 05000333/2007005
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-333
License No.:	DPR-59
Report No.:	05000333/2007005
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	268 Lake Road Scriba, New York 13093
Dates:	October 1, 2007 through December 31, 2007
Inspectors:	G. Hunegs, Senior Resident Inspector L. Casey, Resident Inspector S. Rutenkroger, PhD, Resident Inspector P. Frechette, Physical Security Inspector J. Noggle, Senior Health Physicist
Approved by:	Eugene W. Cobey, Chief Projects Branch 2 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000333/2007-005; 10/01/2007 - 12/31/2007; James A. FitzPatrick Nuclear Power Plant; Maintenance Risk Assessments and Emergent Work Control, and Event Followup.

The report covered a three-month period of inspection by resident inspectors and announced inspections by region based inspectors. Three Green findings, two of which were non-cited violations (NCVs), 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 Technical Specification 5.4, “Procedures,” was identified when operators did not implement certain steps specified in Operations Shift Standing Order 2007-020, “Lake Condition Monitoring,” Revision 4, which increased the likelihood of a scram. Entergy entered the condition into their corrective action program, revised the lake condition monitoring procedure, and discussed procedure adherence expectations with operators.

  The inspectors determined that this finding is more than minor because it is associated with the Human Performance attribute (human error) of the Initiating Events cornerstone; and it impacted the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety function during shutdown as well as power operations. The inspectors evaluated this finding using Phase 1 of IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At Power Situation,” and determined it to be of very low safety significance (Green) because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment functions would not be available.

  This finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that expectations regarding procedural compliance were met. (H.4(b)) (Section 4OA3)

- Green. A self-revealing finding was identified involving inadequate corrective actions when Entergy failed to correct the adverse condition of the feedwater low-flow control valve, 34FCV-137. Entergy also failed to implement corrective actions in a timely manner to remotely monitor feedwater flow rate through the feedwater low-flow control valve in order to support level control. This condition resulted in a low level scram and primary containment isolation system group two isolation on September 12, 2007, and October 28, 2007. This problem was entered into Entergy’s corrective action program.

Following the October 28, 2007, manual scram and subsequent low level scram, Entergy replaced the stem and packing box for the low-flow control valve and implemented an interim method to remotely monitor feedwater flow rate. In addition, Entergy has scheduled a design change to provide low-range feedwater flow rate instrumentation in the control room.

The inspectors determined that this finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events cornerstone, and it impacted 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 evaluated this finding using Phase 1 of Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined it to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available.

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not take appropriate corrective actions, in a timely manner, to address the feedwater low-flow control valve degradation and to provide a method to monitor the feedwater control system response following the low level scram and primary containment isolation system group two isolation on September 12, 2007. Consequently, another low level scram and primary containment isolation system group two isolation occurred on October 28, 2007. (P.1(d)) (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. A self-revealing NCV of 10 CFR Part 50.65 (a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was identified when Entergy failed to perform a risk assessment prior to commencing performance of Instrument Surveillance Procedure ISP-175A1, "Reactor Containment Cooling Instrument Functional Test/ Calibration." This was due to instrument and control technicians performing the procedure which was not in accordance with the plant work schedule. This problem was entered into Entergy's corrective action program. Corrective actions included communicating the error to personnel, conducting human performance training, and improving administrative control of procedures.

The inspectors determined that the finding impacted the Mitigating Systems cornerstone because it impacted the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding is more than minor because the licensee's risk assessment failed to consider risk significant structures, systems, and components (i.e., high pressure coolant injection and reactor core isolation cooling) that were unavailable during the maintenance period.

Using IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management SDP," Flowchart 1, "Assessment of Risk Deficit," the inspectors determined the incremental core damage probability deficit from Entergy's core damage frequency as a result of the actual duration of ISP-175A1 (1.07 hours). The inspectors calculated the incremental core damage probability deficit and determined it to be significantly lower than 1E-6. Because the calculated risk deficit was not greater than 1E-6 incremental core damage probability deficit, the inspectors determined that this finding was of very low safety significance (Green).

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the instrument and control technicians involved did not effectively implement the expected human error prevention techniques (e.g., self-checking, prejob briefs, and proper documentation of activities), to ensure the correct procedure was used in accordance with the work schedule. (H.4(a)) (Section 1R13)

B. Licensee-Identified Violations

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 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

The James A. FitzPatrick Nuclear Power Plant began the inspection period operating at full power. On October 3, 2007, Entergy reduced power to 65 percent to remove the 'B1' condenser waterbox from service as a result of a condenser tube leak. Following repairs, the plant was returned to full power on October 4, 2007. On October 13, 2007, Entergy shut down the plant due to lowering plant cooling water intake level. The lowering intake level was caused by algae intrusion resulting in excessive fouling of the intake traveling water screens. Following repairs to the traveling water screen system and execution of a monitoring plan to assure availability of cooling water systems, the plant was started up on October 15, 2007, and returned to full power on October 18, 2007. On October 28, 2007, operators initiated a manual reactor scram due to lowering plant cooling water intake level which was caused by algae intrusion in the intake. Following repairs to the traveling water screen system and execution of a monitoring plan to assure availability of cooling water systems, the plant was started up and returned to service on November 1, 2007. Reactor power was limited to 55 percent due to the unavailability of one of two feedwater pumps. On November 5, 2007, the main generator was removed from service and the plant was shutdown to repair the 'A' feedwater pump discharge isolation valve. Following repairs, the plant was started up and returned to full power on November 12, 2007. On November 16, 2007, reactor power was lowered to 65 percent as required by operations shift standing orders, due to environmental conditions that had the potential to result in excessive fouling of the traveling water screen system. When environmental conditions returned to normal, reactor power was raised to 100 percent later that day. The plant continued to operate at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample)

a. Inspection Scope

The inspectors completed one adverse weather protection sample. The inspectors reviewed and verified completion of the operations department cold weather preparation checklist contained in procedure Administrative Procedure AP-12.04, "Seasonal Weather Preparations," Revision 16. The inspectors reviewed the operating status of the reactor and turbine building heating and cooling systems, emergency diesel generators and fire protection water. Accessible areas of the reactor, turbine, and screen house buildings were inspected to assess the effectiveness of the ventilation systems. The inspections included discussions with operations and engineering personnel to ensure that they were aware of temperature restrictions and required actions. The documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q - 3 samples)

a. Inspection Scope

The inspectors performed three partial system walkdowns to verify the operability of redundant or diverse trains and components during periods of system train unavailability or following periods of maintenance. The inspectors referenced the system procedures, the Updated Final Safety Analysis Report (UFSAR), and system drawings in order to verify that the alignment of the available train was proper to support its required safety functions. The inspectors also reviewed applicable condition reports (CRs) and work orders to ensure that Entergy had identified and properly addressed equipment discrepancies that could potentially impair the capability of the available equipment train, as required by 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action.” The documents reviewed are listed in the Attachment. The inspectors performed a partial walkdown of the following systems which represented three inspection samples:

- Reactor core isolation cooling system when the high pressure coolant injection system was out of service for testing;
- Residual heat removal service water system during degraded conditions caused by intake debris ingestion; and
- High pressure coolant injection system when the reactor core isolation cooling system was out of service for testing.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 10 samples)

a. Inspection Scope

The inspectors conducted tours of several fire areas to assess the material condition and operational status of fire protection features. The inspectors verified, consistent with applicable administrative procedures, that: combustibles and ignition sources were adequately controlled; passive fire barriers, manual fire-fighting equipment, and suppression and detection equipment were appropriately maintained; and compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with Entergy’s fire protection program. The inspectors evaluated the fire protection program against the requirements of Licensee Condition 2.C.3. The documents reviewed are listed in the Attachment.

This inspection represented ten inspection samples for fire protection tours and was conducted in the following plant areas:

- Fire Area/Zone IX/RB-1A, elevation 344 foot;
- Fire Area Zone IE/TB-1, North elevation 252 foot;
- Fire Area/Zone IE/TB-1, South elevation 252 foot;
- Fire Area/Zone X/RB-1, elevation 272 foot;
- Fire Area/Zone XII/SP-1, elevation 255 Foot;
- Fire Area/Zone XIII/SP-2, elevation 255 foot;
- Fire Area/Zone ID/CT-4, elevation 286 foot;
- Fire Area/Zone XI/CT-3, elevation 286 foot;
- Fire Area/Zone II/CT-2, elevation 258 foot; and
- Fire Area/Zone II/CT-1, elevation 258 foot.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q - 1 sample)

a. Inspection Scope

On November 26, 2007, the inspectors observed licensed operator simulator training to assess operator performance during several scenarios to verify that operator performance was adequate and evaluators were identifying and documenting crew performance problems. The inspectors evaluated the performance of risk significant operator actions, including the use of emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, the implementation of appropriate actions in response to alarms, the performance of timely control board operation and manipulation, and the oversight and direction provided by the shift manager. The inspectors also reviewed simulator fidelity to evaluate the degree of similarity to the actual control room. Licensed operator training was evaluated against the requirements of 10 CFR Part 55, “Operators’ Licenses.” The documents reviewed are listed in the Attachment. This observation of operator simulator training represented one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 2 samples)

a. Inspection Scope

The inspectors reviewed performance-based problems involving selected in-scope structures, systems, or components (SSCs) to assess the effectiveness of the maintenance program. The reviews focused on:

- Proper Maintenance Rule scoping in accordance with 10 CFR Part 50.65;
- Characterization of reliability issues;
- Changing system and component unavailability;
- 10 CFR Part 50.65 (a)(1) and (a)(2) classifications;
- Identifying and addressing common cause failures;
- Trending of system flow and temperature values;
- Appropriateness of performance criteria for SSCs classified (a)(2); and
- Adequacy of goals and corrective actions for SSCs classified (a)(1).

The inspectors reviewed system health reports, maintenance backlogs, and Maintenance Rule basis documents. The inspectors evaluated the maintenance program against the requirements of 10 CFR Part 50.65. The documents reviewed are listed in the Attachment. The following Maintenance Rule samples were reviewed and represented two inspection samples:

- Low pressure coolant injection (LPCI) system batteries; and
- Standby liquid control system.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 5 samples)

a. Inspection Scope

The inspectors reviewed maintenance activities to verify that the appropriate risk assessments were performed prior to removing equipment for work. The inspectors verified that risk assessments were performed as required by 10 CFR Part 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The documents reviewed are listed in the Attachment. The review of the following activities represented five inspection samples.

- Week of October 1, 2007, which included ‘B’ scram discharge instrument volume vent and drain valve failure and an unplanned power reduction to repair a condenser tube leak;

- Week of October 15, 2007, which included plant startup with the 'B' LPCI system battery inoperable and implementation of Technical Specification Limiting Condition for Operation (LCO) 3.0.4.b;
- Week of November 5, 2007, which included shutdown risk assessment for a forced outage (FO184) to replace the 'A' reactor feedpump discharge valve 34MOV-100A;
- Week of November 12, 2007, which included 'A' LPCI battery replacement, cooling water intake algae intrusion risk due to high winds, and performance of Instrument Surveillance Procedure ISP-175A1, "Reactor Containment Cooling Instrument Functional Test," which affected the operability of both high pressure coolant injection and reactor core isolation cooling systems; and
- Week of November 26, 2007, which included 'B' standby liquid control pump replacement and increased trip risk due to winds greater than 50 miles per hour.

b. Findings

Introduction: A Green, self-revealing non-cited violation (NCV) of 10 CFR Part 50.65 (a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was identified when Entergy failed to perform a risk assessment prior to commencing performance of Instrument Surveillance Procedure ISP-175A1, "Reactor Containment Cooling Instrument Functional Test/ Calibration." Due to personnel error, the incorrect procedure was performed.

Description: On November 15, 2007, Entergy performed a risk assessment to perform ISP-175A2, "Reactor and Containment Cooling and ATWS Instrument Functional Test/ Calibration," in conjunction with planned maintenance on the 'A' LPCI Battery and '2A' turbine building closed loop cooling (TBCLC) pump. Entergy's Administrative Procedure AP-10.10, "On-line Risk Assessment," assigns a risk category color in risk significant order from Green, Yellow, Orange to Red based on core damage frequency calculated with the specific plant equipment out-of-service. The risk as assessed for performing ISP-175A2 in conjunction with planned maintenance on the 'A' LPCI Battery and '2A' TBCLC pump was determined to be Green per AP-10.10. However, due to personnel error, ISP-175A1, "Reactor Containment Cooling Instrument Functional Test/ Calibration," was performed instead of ISP-175A2. As a result, an appropriate risk assessment was not performed prior to performing ISP-175A1. Performance of ISP-175A1 made the high pressure coolant injection system inoperable during a portion of the test and the reactor core isolation cooling system inoperable for a portion of the test. Although high pressure coolant injection and reactor core isolation cooling were not inoperable at the same time, the 'A' LPCI system was out of service for planned maintenance during performance of ISP-175A1. Performance of ISP-175A1 in conjunction with planned maintenance on the 'A' LPCI Battery and '2A' TBCLC pump actually put the plant in a Yellow risk category per AP-10.10, "On-line Risk Assessment." The risk of performing ISP-175A1 in conjunction with planned maintenance on the 'A' LPCI battery and '2A' TBCLC pump was not assessed until ISP-175A1 was complete and instrument and control technicians realized that they had completed the incorrect procedure. The inspectors noted that there were other opportunities to identify the

incorrect procedure during the pre-job brief, and during interdepartmental communications and coordination with operations and the work control center.

<u>Analysis</u>: The inspectors determined that the performance deficiency was that an appropriate risk assessment was not performed prior to the start of Instrument Surveillance Procedure ISP-175A1 “Reactor Containment Cooling Instrument Functional Test/Calibration.” This was due to instrument and control technicians performing the procedure which was not in accordance with the plant work schedule. As a result, equipment was taken out-of-service which affected core damage frequency. Factoring in the actual equipment that was taken out-of-service would have resulted in a change in plant risk from Green to Yellow, thus requiring the implementation of additional risk management actions than were implemented on November 15, 2007. Traditional enforcement does not apply because there were no actual safety consequences or potential for impacting the NRC’s regulatory function, and the finding was not the result of any willful violation of NRC requirements or Entergy’s procedures.

The inspectors determined that the finding impacted the Mitigating Systems cornerstone because it impacted the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding was more than minor because the licensee’s risk assessment failed to consider risk significant structures, systems, and components (i.e., high pressure coolant injection and reactor core isolation cooling) that were unavailable during the maintenance period.

Using IMC 0609, Appendix K, “Maintenance Risk Assessment and Risk Management SDP,” Flowchart 1, “Assessment of Risk Deficit,” the inspectors determined the incremental core damage probability deficit from Entergy’s core damage frequency as a result of the actual duration of ISP-175A1 (1.07 hours). The inspectors calculated the incremental core damage probability deficit and determined it to be significantly lower than 1E-6. Because the calculated risk deficit was not greater than 1E-6 incremental core damage probability deficit, the inspectors determined that this finding was of very low safety significance (Green).

The problem was entered into Entergy’s corrective action program as CR-JAF-2007-04019. Corrective actions included communicating the error to personnel, conducting human performance training, and improving administrative control of procedures.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the instrument and control technicians involved did not effectively implement the expected human error prevention techniques (e.g., self-checking, prejob briefs, and proper documentation of activities), to ensure the correct procedure was used in accordance with the work schedule. (H.4(a))

<u>Enforcement</u>: 10 CFR Part 50.65 (a)(4), requires, in part, that before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to the above, on November 15, 2007, Entergy did not assess the increase in risk prior to performing ISP-175A1, “Reactor Containment Cooling Instrument Functional

Test/ Calibration." Because the finding was of very low safety significance and was entered into Entergy's corrective action program as CR-JAF-2007-04019, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000333/2007005-01, Failure to Perform a Risk Assessment When Required by 10 CFR Part 50.65(a)(4)**.

1R15 Operability Evaluations (71111.15 - 5 samples)

a. Inspection Scope

The inspectors reviewed operability determinations to assess the acceptability of the evaluations; when needed, the use and control of compensatory measures; and compliance with Technical Specifications (TS). The inspectors' review included a verification that the operability determinations were made as specified by ENN-OP-104, "Operability Determinations." The technical adequacy of the determinations was reviewed and compared to the TS, UFSAR, and associated design basis documents. The documents reviewed are listed in the Attachment. The following evaluations were reviewed and represented five inspection samples:

- CR 2007-03424, concerning a solenoid operated valve, 03SOV-31B, which affected the operability of the scram discharge instrument volume vent and drain valves;
- CRs 2007-03665 and 2007-03693, concerning emergency service water pinhole leaks;
- CR 2007-03747, concerning the operability of the emergency service water system and residual heat removal service water system following the cooling water intake algae intrusion and manual reactor scram on October 28, 2007;
- CR 2007-03580, concerning operability of the raw water systems including emergency service water and residual heat removal service water following the cooling water intake algae intrusion and plant shutdown on October 13, 2007; and
- CR 2007-03507, concerning the operability of the 'B' standby liquid control pump.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A - 1 sample)

a. Inspection Scope

Three recent cooling water intake algae intrusion events resulted in the traveling water screen system becoming non-functional. Entergy installed several modifications with a design objective to improve the reliability of the traveling water screen system during occurrences of high algae intrusion into the screenwell and onto the traveling water screen buckets. The inspectors reviewed the 10 CFR Part 50.59 screens, impact screening summary, and calculations for the modifications. In addition, portions of installation post-maintenance tests were reviewed and observed. The documents reviewed are listed in the Attachment. The following modifications were reviewed and represented one inspection program sample:

- Engineering Change 3551, “Provide An Alternate Stronger Shear Pin For the Traveling Water Screens;”
- Engineering Change 3702, “Traveling Water Screen Downstream Guides;”
- Engineering Change 3745, “Change Traveling Water Screen Motors from 2 Horsepower / 1800 revolutions per minute to 5 Horsepower / 3600 revolutions per minute;” and
- Engineering Change 3741, “Remove Falk Fluid Coupling.”

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 6 samples)

a. Inspection Scope

The inspectors reviewed six post-maintenance test procedures and associated testing activities for selected risk-significant mitigating systems to assess whether the effect of maintenance on plant systems was adequately addressed by control room and engineering personnel. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness, and were consistent with design basis documentation; test instrumentation had current calibrations and adequate range and accuracy for the application; and tests were performed, as written, with applicable prerequisites satisfied. Upon completion, the inspectors verified that equipment was returned to the proper alignment necessary to perform its safety function. Post-maintenance testing was evaluated against the requirements of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control.” The documents reviewed are listed in the Attachment. The following post-maintenance test activities were reviewed and represented six inspection samples:

- Work order 00125550 for replacement of cell 137 in the ‘B’ LPCI battery due to cell voltage falling below TS requirements and rendering the ‘B’ LPCI battery inoperable during performance of Maintenance Surveillance Test MST-071.11, “LPCI Battery Quarterly Surveillance Test,” on October 9, 2007;
- Work order 00130767 for replacement of the ‘D’ emergency diesel generator 4160 V output breaker during the week of November 26, 2007;
- Work order 51104406 for replacement of the entire ‘A’ LPCI battery from November 13, 2007 through November 16, 2007;
- Work order 00125751 for ‘B’ standby liquid control pump replacement from November 29, 2007 to November 30, 2007;
- Work order 00124350 for ‘A’ emergency service water pump 46P-2A replacement from December 11, 2007 to December 13, 2007; and
- Work order 51193346 for reactor building exhaust fan discharge damper 66-AOD-104A maintenance on December 17, 2007.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20 - 3 sample)

a. Inspection Scope

The inspectors observed and reviewed the following activities during the unplanned outages conducted from October 13, 2007, through October 15, 2007, and October 28, 2007, through November 1, 2007; and the scheduled maintenance outage from November 5, 2007, through November 11, 2007, to confirm that the Entergy had appropriately considered risk, industry experience, and previous site-specific problems in their outage plan. The documents reviewed are listed in the Attachment. During the outages, the inspectors observed portions of the shutdown and cooldown and monitored licensee controls over the outage activities listed below. This review represented three inspection samples.

- The inspectors reviewed outage schedules and procedures and verified that TS required safety system availability was maintained, shutdown risk was considered, and that contingency plans existed to restore key safety functions such as electric power and water inventory control;
- The inspectors observed portions of the plant shutdown and cooldown and verified that the TS cooldown rate limits were not exceeded;
- The inspectors periodically verified the proper alignment and operation of the shutdown cooling and reactor coolant makeup systems; and
- The inspectors observed portions of the reactor startup following the outage, and verified that safety-related equipment required for mode changes was operable, containment integrity was maintained, and reactor coolant boundary leakage was within TS limits.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 6 samples)

a. Inspection Scope

The inspectors witnessed performance of surveillance tests and/or reviewed test data of selected risk-significant SSCs to assess whether the SSCs satisfied TS, UFSAR, Technical Requirements Manual, and Entergy procedure requirements. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness, and were consistent with design basis documents; test instrumentation had current calibrations and adequate range and accuracy for the application; and tests were performed, as written, with applicable prerequisites satisfied. Upon surveillance test completion, the inspectors verified that equipment was returned to the status specified to perform its safety function. The inspectors evaluated the tests against the requirements in TS. The documents reviewed are listed in the Attachment. The following surveillance tests were reviewed and represented six inspection samples.

- Surveillance Test ST-4N, “High Pressure Coolant Injection Quick-start, Inservice and Transient Monitoring Test,” Revision 54;
- Surveillance Test ST-8Q, “Testing of the Emergency Service Water System,” Revision 36;
- Surveillance Test ST-20T, “Post Scram Control Rod Testing,” Revision 8;
- Surveillance Test ST-68, “IST Cold Shutdown Valve Testing,” Revision 17;
- Surveillance Test ST-24J, Reactor Core Isolation Cooling Flow Rate and Inservice Test,” Revision 37; and
- Surveillance Test ST-40D, “Daily Surveillance and Channel Check for RCS Leak Detection,” Revision 104.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 1 sample)

a. Inspection Scope

The inspectors reviewed Temporary Modification 3227, which was implemented in order to provide a method to control the scram discharge volume vent and drain valves while maintenance was performed on the scram discharge volume vent and drain solenoid valve, 03SOV-31B. The inspectors assessed the adequacy of the 10 CFR Part 50.59 evaluation for the temporary modification. The inspectors also verified that the installation was consistent with the modification documentation; that the drawings and procedures were updated as applicable; and that the post-installation testing was adequate. The documents reviewed are listed in the Attachment. This review represented one inspection sample.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors observed simulator activities associated with licensed operator requalification training on November 26, 2007. The inspectors verified that emergency classification declarations and notification activities were properly completed. The inspectors evaluated the drill against the requirements of 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities." This observation represented one inspection sample.

b. Findings

No findings of significance were identified.

2. **RADIATION SAFETY**

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 21 samples)

a. Inspection Scope

During December 18 through 21, 2007, the inspectors conducted the following activities to verify that 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 Part 20, TS, and Entergy's procedures. Documents reviewed are listed in the Attachment.

1) The inspectors determined that there were no occupational exposure cornerstone performance indicator (PI) incidents during the current assessment period.

2) The inspectors walked down exposure significant work areas of the plant in the reactor building, turbine building, and radwaste building and reviewed licensee controls and surveys to determine if licensee surveys, postings, and barricades were acceptable and in accordance with regulatory requirements.

3) The inspectors walked down exposure significant work areas of the plant in the reactor building, turbine building, and radwaste building and conducted independent surveys to determine whether prescribed radiation work permit and

procedural controls were in place and whether licensee surveys and postings were complete and accurate.

4) Radiation work permits (RWPs) were reviewed that provide access to exposure significant areas of the plant including high radiation areas. Specified electronic personal dosimeter alarm set points were reviewed with respect to current radiological condition applicability and workers were queried to verify their understanding of plant procedures governing alarm response and knowledge of radiological conditions in their work area.

5) The inspectors determined that there were no radiation work permits for airborne radioactivity areas with the potential for individual worker internal exposures of >50 mrem committed effective dose equivalent.

6) The inspectors determined that there were no internal dose assessments greater than 50 mrem during 2007.

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

8) A review of licensee radiation protection program self-assessments and audits during 2007 was conducted to determine if identified problems were entered into the corrective action program for resolution.

9) Three CRs associated with the radiation protection access control and as low as is reasonably achievable (ALARA) areas between February 2007 and December 2007, were reviewed and discussed with licensee staff to determine if the follow-up activities were being conducted in an effective and timely manner commensurate with their safety significance.

10) Based on the CRs reviewed, repetitive deficiencies were screened to determine if Entergy's self-assessment activities were identifying and addressing these deficiencies.

11) The inspectors determined that there were no Occupational Exposure PI incidents reported during the current assessment period.

12) During December 18 through 21, 2007, the following radiologically significant job was selected; the radiological job requirements were reviewed; and job performance was reviewed with respect to the radiological work requirements.

- Control rod blade underwater dose rate measurements

13) During observation of the radiologically significant job listed in (12) above, the adequacy of surveys, job coverage and contamination controls were reviewed.

14) The inspectors determined that there were no significant dose gradients requiring relocation of dosimetry for the radiologically significant job listed in (12) above.

15) Changes to the high radiation area and very high radiation area procedures and management of these changes were discussed with the Radiation Protection Manager.

16) Controls associated with transverse incore probe activation in the core and coordination with plant operations prior to allowing personnel entry into the transverse incore probe room was discussed with the duty watch radiation protection technician.

17) All accessible locked high radiation area entrances were verified to be locked through challenging the locks or doors.

18) During observation of the job listed in (12) above, radiation worker performance was evaluated with respect to the specific radiation protection work requirements and their knowledge of the radiological conditions in their work areas.

19) Several radiological CRs (see Section 4OA2) were reviewed to evaluate if the incidents were caused by radiation worker errors and determine if there were any trends or patterns and if Entergy's corrective actions were adequately addressing these trends.

20) During observation of the jobs listed in (12) above, radiation protection technician work performance was evaluated with respect to their knowledge of the radiological conditions, the specific radiation protection work requirements and radiation protection procedures.

21) Several radiological CRs (see Section 4OA2) were reviewed to evaluate if the incidents were caused by radiation protection technician errors and determine if there were any trends or patterns and if Entergy's corrective actions were adequately addressing these trends.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 2 samples)

a. Inspection Scope

During December 18 through 21, 2007, the inspectors conducted the following activities to verify that Entergy was properly maintaining individual and collective radiation exposures as low as is reasonably achievable. Implementation of the ALARA program was reviewed against the criteria contained in 10 CFR Part 20.1101(b) and Entergy's procedures. Documents reviewed are listed in the Attachment.

1) There were two declared pregnant workers during the current reactor oversight program assessment period. The personnel exposure records and procedural controls for the declared pregnant worker were reviewed with respect to 10 CFR Part 20 requirements.

2) Based on the CRs reviewed (see Section 4OA2), repetitive deficiencies were screened to determine if Entergy's self-assessment activities were identifying and addressing these deficiencies.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03 – 9 samples)

a. Inspection Scope

During November 13 through 15, 2007, the inspectors conducted the following activities to evaluate the operability and accuracy of radiation monitoring instrumentation, and the adequacy of the respiratory protection program for issuing self-contained breathing apparatus to emergency response personnel. Implementation of these programs was reviewed against the criteria contained in 10 CFR Part 20, applicable industry standards, and Entergy's procedures. Documents reviewed are listed in the Attachment.

1) The UFSAR describing the liquid radwaste system, solid radwaste system, and gaseous radwaste system was reviewed to identify applicable radiation monitors associated with transient high radiation areas in the plant for review.

2) The radiation protection (RP) instrument issue area provided for the selection of portable RP instruments that were available for use for job coverage of radiologically significant areas.

3) Current calibration records and applicable calibration procedures were reviewed for the following plant radiation monitors and portable RP instruments. In addition, the applicable calibrators utilized were reviewed for appropriate instrument calibration geometries and National Institute for Science and Technology standard traceability.

Plant Radiation Monitors
- Main steam line radiation monitors;
- Transverse in-core probe room area radiation monitor;
- Refuel floor area radiation monitors;
- Containment radiation monitors; and
- Steam jet air ejector gas monitors.

Portable RP Instruments
- 60 electronic dosimeters;
- Three radiation survey instruments;
- One extendable probe survey instruments;
- One neutron radiation survey instrument;
- Two continuous air monitors;
- Three air samplers;
- One personal lapel air sampler; and
- Three beta and alpha air sample counters.

Calibrators
- Two Shepherd 89 survey instrument calibrators; and
- One Shepherd 142-10 panoramic calibrator.

4) Radiological incidents involving internal exposures identified by CRs were reviewed for 2007. In addition, dosimetry electronic records were queried for any internal exposures >50 mrem committed effective dose equivalent. None were identified for further review.

5) Three CRs were reviewed with respect to radiation protection program deficiencies to determine if the deficiencies were appropriately characterized and corrected commensurate with their safety significance.

6) Based on the CRs reviewed, no repetitive deficiencies were identified for further followup.

7) With respect to the RP portable instruments listed in 3) above, the instrument's calibration expiration and response check stickers were reviewed. The applicable response check beta-source and instrument sign-out procedures were also reviewed.

8) Emergency plan-specified self-contained breathing apparatus equipment and qualified users were sampled based on Fitzpatrick Emergency Plan documents, (SAP-2, & RP-RESP-02.03). This included inspection of selected self-contained breathing apparatus located in the main control room, security building, and operations support center. Self-contained breathing apparatus qualification records for all on-shift reactor operators were verified for currency.

9) Selected self-contained breathing apparatus units in the main control room, security building and operations support center were examined for periodic air

cylinder hydrostatic testing and maintenance records. Review of approved replacement parts documentation and certification of the repair personnel was performed.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151 - 14 samples)

a. Inspection Scope

The inspectors reviewed PI data for the cornerstones listed below and used Nuclear Energy Institute 99-02, “Regulatory Assessment PI Guidance,” Revision 5, to verify individual PI accuracy and completeness.

Cornerstone: Initiating Events

- Unplanned Scrams;
- Unplanned Power Changes; and
- Unplanned Scrams with Complications.

The inspectors reviewed Entergy’s event reports, operator logs, and PI data sheets to determine whether Entergy adequately identified the number of scrams and unplanned power changes greater than 20 percent that occurred between July 2006 and June 2007. This number was compared to the number reported for the PI during the applicable quarter. The inspectors also verified the accuracy of the number of critical hours reported.

Cornerstone: Mitigating Systems

- Safety System Functional Failures;
- Mitigating Systems Performance Index (MSPI), Emergency AC Power System;
- MSPI, High Pressure Injection System;
- MSPI, Heat Removal System;
- MSPI, Residual Heat Removal System; and
- MSPI, Cooling Water Systems.

The inspectors reviewed data and plant records from January 2007 to June 2007. The records reviewed included PI data summary reports, licensee event reports, operator narrative logs, and maintenance rule records. The inspectors verified the accuracy of the number of critical hours reported, and interviewed the system engineers and operators responsible for data collection and evaluation.

Cornerstone: Occupational Radiation Safety

- Occupation Exposure Control Effectiveness

The inspectors reviewed implementation of Entergy's Occupational Exposure Control Effectiveness PI Program. Specifically, the inspectors reviewed CRs, and radiological controlled area dosimeter exit logs for the past 4 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 Nuclear Energy Institute 99-02, Regulatory Assessment PI Guideline, Revision 5, to verify that all occurrences that met the Nuclear Energy Institute criteria were identified and reported as PIs.

Cornerstone: Public Radiation Safety

- RETS/ODCM Radiological Effluent Occurrences

The inspectors reviewed a listing of relevant effluent release reports for the past 4 calendar quarters, for issues related to the public radiation safety PI, which measures radiological effluent release occurrences per site that exceed 1.5 mrem/qtr whole body or 5.0 mrem/qtr organ dose for liquid effluents; 5 mrads/qtr gamma air dose, 10 mrad/qtr beta air dose, and 7.5 mrads/qtr for organ dose for gaseous effluents. The review was against applicable criteria specified in Nuclear Energy Institute 99-02, Regulatory Assessment PI Guideline, Revision 5. The purpose of the review was to verify that occurrences that met the Nuclear Energy Institute criteria were recognized and identified as PI occurrences.

The inspectors reviewed the following documents to ensure Entergy met all requirements of the PI:

- 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.

Cornerstone: Physical Protection

- Fitness-for-Duty;
- Personnel Screening; and
- Protected Area Security Equipment.

The review included Entergy's tracking and trending reports, personnel interviews and security event reports for the PI data collected since the last security baseline inspection. The inspectors noted from Entergy's submittal that there were no reported failures to properly implement the requirements of 10 CFR Part 73 and 10 CFR Part 26 during the reporting period.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of all items entered into Entergy's corrective action program. The review was accomplished by accessing Entergy's computerized database for CRs and attending CR screening meetings.

In accordance with the baseline inspection procedures, the inspectors selected items across the initiating events, mitigating systems, and barrier integrity cornerstones for additional follow-up and review. Additionally, NRC specialist inspectors reviewed six CRs associated with the radiation protection program that were initiated between February 2007 and November 2007. The inspectors assessed Entergy's threshold for problem identification, the adequacy of the cause analyses, extent of condition review, operability determinations, and the timeliness of the specified corrective actions. The CRs reviewed are listed in the Attachment.

.2 Semi-Annual Review to Identify Trends (71152 - 1 sample)

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of Entergy's Corrective Action Program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment and corrective maintenance issues but also considered the results of daily inspector corrective action program item screening discussed in Section 4OA2.1. The review also included issues documented in system health reports, corrective maintenance work requests, component status reports, site monthly meeting reports and maintenance rule assessments. The inspectors' review nominally considered the six-month period of July 2007 through December 2007, although some examples expanded beyond those dates when the scope of the trend warranted. The inspectors compared and contrasted their results with the results contained in Entergy's latest integrated quarterly assessment report. Corrective actions associated with a sample of the issues identified in the trend report were reviewed for adequacy. The inspectors also evaluated the trend report specified in ENN-LI-102, "Corrective Action Process," and 10 CFR Part 50, Appendix B. The documents reviewed are listed in the Attachment.

b. Assessment and Observations

No findings of significance were identified.

Equipment, human performance and program issues were identified at an appropriate threshold and were entered into the corrective action program.

.3 Annual Sample: Reactor Scram Due to Low Intake Water Level (71152 - 1 sample)

a. Inspection Scope

The inspectors selected the following corrective action issue for detailed review. The report and supporting information were reviewed to ensure that a comprehensive evaluation was performed and appropriate corrective actions were specified. The inspectors evaluated the reports against the requirements of procedure ENN-LI-102, "Corrective Action Process," and 10 CFR Part 50, Appendix B.

- CR-2007-03202, "Reactor Scram Due to Low Intake Water Level."

b. Findings and Observations

No findings of significance were identified. The inspectors determined that the causal analysis, extent of condition review, and the timeliness of the specified recommendations and corrective actions were appropriate.

.4 Annual Sample: 'A' Feedwater Pump Discharge Isolation Valve, 34MOV-100A, Failure (71152 – 1 sample)

a. Inspection Scope

The inspectors selected the following corrective action issue for detailed review. The report and supporting information were reviewed to ensure that a comprehensive evaluation was performed and appropriate corrective actions were specified. The inspectors evaluated the reports against the requirements of procedure ENN-LI-102, "Corrective Action Process," and 10 CFR Part 50, Appendix B.

- CR-2007-03851, "34MOV-100A Stem/Disc Separation."

b. Findings and Observations

No findings of significance were identified. The inspectors determined that the causal analysis, extent of condition review, and the timeliness of the specified recommendations and corrective actions were appropriate.

4OA3 Event Followup (71153 - 1 sample)

.1 Operator Actions as a Result of Intake Water Blockage

a. Inspection Scope

Operators initiated a manual reactor scram on September 12, 2007, a shutdown on October 13, 2007 and a manual reactor scram on October 28, 2007 due to traveling water screen blockage by algae and a corresponding lowering intake water level. Following each of these events, the main condenser remained available for heat removal. On November 16, 2007, operators reduced power to 65 percent due to high winds as required by an operations shift standing order. Weather conditions warranted an additional power reduction per procedure to 65 percent on November 27, 2007 but this was not executed. For the shutdown and power reduction, the inspectors responded to the control room to monitor activities.

For the scram on September 12 and October 28, 2008, the inspectors responded to the control room and verified that the plant was stable and confirmed that all safety-related mitigating systems had operated properly. The inspectors discussed the scrams with operations, engineering, and licensee management personnel to gain an understanding of the events and to assess follow-up actions. The inspectors reviewed operator actions taken in accordance with licensee procedures and reviewed unit and system indications to verify that actions and system responses were as expected. The inspectors evaluated safety equipment and operator performance before and after the event by examining existing plant parameters, strip charts, plant computer historical data displays, operator logs, the alarm typewriter sequence of events printout, and the critical parameter trend charts in the post-transient evaluation. Particular attention was paid to the residual heat removal, emergency service water, service water and fire protection systems to assure that these systems remained operable as a result of the debris. The inspectors also interviewed responsible on shift operations personnel, examined the implementation of the applicable alarm response procedures and abnormal operating procedures and reviewed the written notification made in accordance with 10 CFR Part 50.72 associated with the scrams. In addition, the inspectors reviewed the plant conditions and compared them with the classification of emergency conditions to verify that licensee expectations were met in the emergency preparedness area.

b. Findings

Introduction: A Green, self-revealing NCV of TS 5.4, "Procedures," was identified when operators did not implement certain steps specified in Operations Shift Standing Order 2007-020, "Lake Condition Monitoring," Revision 4.

Description: On November 27, 2007, operators entered Abnormal Operating Procedure 13, "High Winds, Hurricanes, and Tornadoes," for sustained winds greater than 50 miles per hour. Due to the weather conditions, Operations Shift Standing Order 2007-020 was also required to be implemented. Loss of plant intake level due to large amounts of fine lake weed could cause rapid fouling of the traveling water screens. Lake level and traveling water screen parameter monitoring may not provide sufficient time margin to prevent excessive lowering of the intake level. The purpose of the operations shift standing order was to direct operators to implement prescribed actions to reduce power and remove the 'C' circulating water pump to prevent excessive fouling of the traveling water screen system and minimize the possibility of a plant shutdown.

On November 27, starting at 8:00 pm, a large influx of algae was experienced at the intake. Over the next seven hours, the operators changed out the traveling water screen debris fish basket two to three times an hour. Normal traveling screen wash was assisted by the use of the fire hoses during the event. Also, the service water strainer differential pressure alarm was illuminated. Operations Shift Standing Order 2007-020, "Lake Conditioning Monitoring," Revision 4, actions state that with a combination of wind direction and speed <u>and</u> evidence of debris intrusion (two or more), including service water strainer differential pressure alarm and frequent fish basket change out is required (less than or equal to hourly), then reduce power per OP-65 and remove 'C' circulating water pump from service. This was not done because operators misinterpreted the guidance. The inspectors determined that the performance deficiency was that Entergy failed to implement a procedure when the prerequisites for procedure implementation were met.

<u>Analysis</u>: The inspectors determined that the performance deficiency was that Entergy failed to implement a procedure when the prerequisites for procedure implementation were met. The inspectors determined that this finding impacted the Initiating Events cornerstone due to the increased likelihood of a scram due to algae intrusion. This was reasonably within Entergy's ability to foresee and prevent because there is extensive documentation highlighting procedure adherence expectations for operators. Traditional enforcement does not apply because the issue did not have an actual safety consequence or a potential for impacting the NRC's regulatory function, and it was not the result of any willful violation of NRC requirements.

The inspectors determined that this finding was more than minor because it was associated with the Human Performance attribute (human error) of the Initiating Events Cornerstone; and it impacted the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety function during shutdown as well as power operations.

The inspectors evaluated this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At Power Situation," and determined it to be of very low safety significance (Green) because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment functions would not be available.

This finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that expectations regarding procedural compliance were met. (H.4(b))

Enforcement: Technical Specification 5.4.1.a requires that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 1972. Applicable procedures recommended in RG 1.33, Appendix A, include procedures for abnormal, off-normal, or alarm conditions; and procedures for combating emergencies and other significant events. Contrary to the above, Operations Shift Standing Order 2007-020, "Lake Condition Monitoring," Revision 4 was not properly implemented on November 27, 2007. Specifically, operators did not reduce power and remove the 'C' circulating water pump as specified in the procedure actions. Because this issue was determined to be of very low risk significance (Green) and was entered into Entergy's corrective action program as CR 2007-04144, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy:
(NCV 05000333/2007005-02, Failure to Implement Procedure Associated with Lake Condition Monitoring).

.2 (Closed) Licensee Event Report (LER) 05000333/2007002-00 and Supplement 01, Manual Reactor Scram due to Blocked Circulating Water Intake Screens

a. Inspection Scope

The inspectors reviewed the LER and Supplement 01 and related documents to determine the appropriateness of corrective actions, whether any violations of regulatory requirements occurred, and whether the event revealed any generic concerns. Documents reviewed are listed in the Attachment.

b. Findings

Introduction: A Green self-revealing finding was identified involving inadequate corrective actions when Entergy failed to correct the adverse condition of sluggish operation of the feedwater low-flow control valve, 34FCV-137. Entergy also failed to implement corrective actions in a timely manner to remotely monitor feedwater flow rate through the feedwater low-flow control valve in order to support level control. This condition resulted in a low level scram and primary containment isolation system group two isolation on September 12, 2007, and October 28, 2007.

Description: On September 12, 2007, operators inserted a manual reactor scram due to low intake level caused by traveling water screen blockage. All plant equipment responded as expected and the plant was stable in Mode 3, Hot Shutdown. While transitioning to Mode 4, Cold Shutdown, the reactor protection system automatically actuated when the reactor pressure vessel level decreased to less than 177 inches above the top of active fuel. This resulted in a low level scram and a primary containment isolation system group two isolation. The impact of the group two isolation was that it complicated operator recovery actions.

Reactor level was subsequently restored, utilizing the reactor feed pump which remained available, and the cool down was completed satisfactorily. The cause of the manual scram was determined to be due to environmental debris overloading the traveling water screens, and the cause of the low level scram was attributed to sluggish operation of the feedwater low-flow control valve.

The feedwater low-flow control valve, 34FCV-137, is used at low power levels to supply water from the reactor feedpump to the reactor pressure vessel. It is an air-operated valve which receives a demand signal based on reactor pressure vessel level indication. The feedwater low-flow control valve was examined to determine the cause of its slow response. The investigation found that the feedwater low-flow control valve had a significant air leak on the actuator and that the actuator stem required replacement. To correct the air leak the valve positioner was rebuilt. Work order 122735-05 was written to replace the stem and packing box and perform diagnostic testing of the valve. The valve was returned to service prior to the degraded stem being replaced.

An apparent cause evaluation (CR-JAF-2007-03212) completed on October 11, 2007, on the September 12, 2007, low water level scram stated that, during low feedwater flow conditions, operators have limited instrumentation available to monitor feedwater control system response. The only indications available to operators are reactor pressure vessel level and the low-flow control valve position demand signal. The demand signal shows only the controller output to the flow control valve, not actual valve position. Consequently, operators are forced to wait for a response in reactor pressure vessel water level in order to determine if the feedwater low-flow control valve, 34FCV-137, is responding properly. The apparent cause evaluation proposed the corrective action of an interim method to remotely monitor feedwater flow rate through 34FCV-137, until a design change to provide low-range feedwater flow rate instrumentation in the control room is implemented in February, 2008.

On October 28, 2007, a similar event involving low intake level due to traveling water screen blockage from environmental debris resulted in a manual scram. While transitioning from Mode 3 to Mode 4, the reactor protection system automatically actuated when the reactor pressure vessel level decreased to less than 177 inches above the top of active fuel. This resulted in a low level scram and a primary containment isolation system group two isolation. The cause of the low level scram was again attributed to the sluggish operation of the feedwater low-flow control valve. Further, no interim method to remotely monitor feedwater flow rate through the low-flow control valve was in place to monitor feedwater system response. The inspectors determined that the performance deficiency was that Entergy did not correct the feedwater low-flow control valve degradation.

Following the October 28, 2007, manual scram and low level scram, Entergy replaced the stem and packing box for the low-flow control valve, 34FCV-137, and implemented an interim method to remotely monitor feedwater flow rate. In addition, Entergy has scheduled a design change to provide low-range feedwater flow rate instrumentation in the control room.

<u>Analysis</u>: The inspectors determined that the performance deficiency was that Entergy did not correct the feedwater low-flow control valve degradation as documented in CR-JAF-2007-03211, following the low level scram and primary containment isolation system group two isolation on September 12, 2007. Additionally, Entergy did not

implement in a timely manner the corrective action of establishing an interim method to remotely monitor feedwater flow rate through the low-flow control valve as documented in CR-JAF-2007-03212. This resulted in a low level scram and primary containment isolation system group two isolation on October 28, 2007. Entergy procedure EN-LI-102, “Corrective Action Process,” Revision 10, requires, in part, that corrective actions address the cause or resolve the deficiency. This was reasonably within Entergy’s ability to foresee and prevent. Traditional enforcement does not apply because the issue did not have an actual safety consequence or a potential for impacting the NRC’s regulatory function, and it was not the result of any willful violation of NRC requirements.

The inspectors determined that this finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events cornerstone, and it impacted 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 evaluated this finding using Phase 1 of Inspection Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” and determined it to be of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available.

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not take appropriate corrective actions, in a timely manner, to address the feedwater low-flow control valve degradation and to provide a method to monitor the feedwater control system response following the low level scram and primary containment isolation system group two isolation on September 12, 2007. Consequently, another low level scram and primary containment isolation system group two isolation occurred on October 28, 2007 (P.1(d)).

Enforcement: No violation of regulatory requirements occurred because corrective action issues related to the feedwater system are outside of the scope of 10 CFR Part 50 Appendix B. **(Finding (FIN) 05000333/2007005-03, Feedwater Low Flow Control Valve Degradation Led to Primary Containment Isolation System Group Two Isolation)** This LER is closed.

.3 (Closed) Licensee Event Report (LER) 05000333/2007001-00, Safety Relief Valve Setpoints Outside of Allowable Tolerances

On June 6, 2007, Entergy identified that it had operated during the previous operating cycle (Cycle 17) with less than nine operable safety relief valves (SRVs) as required by TS 3.4.3, “Safety/Relief Valves.” TS require nine operable SRVs when in Modes 1, 2 or 3. Seven SRVs that Entergy had removed during the previous refueling outage (RFO-17) had as-found lift setpoints outside the high tolerance limit allowed by TS 3.4.3.1. Additionally, one SRV could not be tested because it was damaged during removal. The root cause analysis determined that the most probably cause of the out of tolerance SRV setpoints was corrosion bonding between the SRV pilot disc and seat which is an industry generic problem. Corrective actions documented in CR -2007-02108 include:

- Evaluate use of new material for discs and seats based on industry experience;
- Reevaluate use of the ion deposition process;
- Evaluate improved processes for control of SRV insulation and environmental controls; and
- Send three valves for forensic analysis to confirm cause and to update the root cause analysis based on the results.

The failure to satisfy TS 3.4.3 during Cycle 17 was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone, and it impacted the cornerstone objective of ensuring the availability, reliability, and capability of the SRV system to respond to initiating events to prevent undesirable consequences. The condition at FitzPatrick was mitigated by two considerations: (1) while the SRVs did not lift within the TS prescribed high limit, they did actuate at higher pressures; and (2) a diverse SRV electronic pressure switch actuation system was available which would have actuated the valves. Because the plant continued to operate within the bounds of the design basis safety analyses, there was no loss of safety function. The inspectors evaluated this finding using Phase I of Inspection Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situation,” and determined that the condition was of very low safety significance (Green) because it did not represent a loss of safety function and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This licensee identified finding involved a violation of TS 3.4.3, “Safety Relief Valves.” The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

4OA6 Meetings, including Exit

Exit Meeting Summary

On January 9, 2008, the inspectors presented the inspection results to Mr. Peter T. Dietrich and other members of his staff. The inspectors asked Entergy whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-identified Violations

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

TS 3.4.3 requires that at least nine SRVs shall be operable in operating modes 1, 2, and 3. Contrary to this, on June 6, 2007, Entergy identified that it had operated in these modes during Cycle 17 with less than nine operable SRVs. Entergy documented this condition in CR-2007-02108.

This finding is of very low safety significance because it did not result in the loss of the overpressure relief safety function of the valves.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

P. Dietrich, Site Vice President
C. Adner, Manager Operations
S. Bono, Director Engineering
J. Costedio, Manager, Regulatory Compliance
P. Cullinan, Manager, Emergency Preparedness
M. Durr, Manager, System Engineering
B. Finn, Director Nuclear Safety Assurance
D. Johnson, Manager, Training
J. LaPlante, Manager, Security
K. Mulligan, General Manager, Plant Operations
J. Pechacek, Manager, Programs and Components Engineering
J. Solowski, Radiation Protection

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000333/2007005-01	NCV	Failure to Perform a Risk Assessment When Required by 10 CFR Part 50.65(a)(4) (Section 1R13)
05000333/2007005-02	NCV	Failure to Implement Procedure Associated with Lake Condition Monitoring (Section 4OA3.1)
05000333/2007005-03	FIN	Feedwater Low Flow Control Valve Degradation Led to Primary Containment Isolation System Group Two Isolation (Section 4OA3.2)

Closed

05000333/2007002-00 and 01	LER	Manual Reactor Scram due to Blocked Circulating Water Intake Screens (Section 4OA3.2)
05000333/2007001-00	LER	Safety Relief Valve Setpoints Outside of Allowable Tolerances (Section 4OA3.3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

OP-51A, "Reactor Building Ventilation and Cooling System," Revision 47
OP-52, "Turbine Building Ventilation," Revision 16
DBD-066, "Design Basis Document for the Reactor Building Heating, Ventilation and Air Conditioning (HVAC) Systems"
DBD-067, "Design Basis document for the Turbine Building HVAC systems"

Section 1R04: Equipment Alignment

OP-19, "Reactor Core Isolation Cooling," Revision 46
OP-13C, "Residual Heat Removal Service Water," Revision 9
OP-15, "High Pressure Coolant Injection," Revision 54

Section 1R05: Fire Protection

ENN-DC-161, "Transient Combustible Program,"
PFP-PWR27, Fire Area/Zone IX/RB-1A
PFP-PWR42, Fire Area Zone IE/TB-1
PFP-PWR43, Fire Area/Zone IE/TB-1
PFP-PWR21, Fire Area/Zone X/RB-1
PFP-PWR33, Fire Area/Zone XII/SP-1
PFP-PWR33, Fire Area/Zone XIII/SP-2
PFP-PWR06, Fire Area/Zone ID/CT-4
PFP-PWR07, Fire Area/Zone XI/CT-3
PFP-PWR01, Fire Area/Zone II/CT-2
PFP-PWR02, Fire Area/Zone II/CT-1

Section 1R11: Licensed Operator Requalification Program

Evaluation 60490-0, "Main Steam Line Break Outside Containment"
AOP-32, "Unexplained/Unanticipated Reactivity Change," Revision 10

Section 1R12: Maintenance Effectiveness

System Health Report for the DC Electrical System, 3rd Quarter 2007
JAF-RPT-ELEC-02302, "Maintenance Rule Basis Document for System 71-DC/ DC Electrical Distribution System," Revision 4
IEEE Std 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications"
CR-2007-03027, "MST-71.11 below acceptance criteria of 2.13 VDC"
CR-2007-03079, "LPCI Inverters need to be evaluated for (a)(1)"

CR-2007-03635, “During performance of MST-071.11, found four cells with electrolyte level below low level mark”
CR-2007-02485, “B LPCI Battery cell 102 below 2.12 VDC”
CR-2007-03519, “MST-071.11 cells did not meet acceptance 1 criteria”
CR-2006-04739, “MST-071.11 below level 1 acceptance criteria”
CR-2006-02472, “Performed MST-71.11, Quarterly Surveillance Test of 71BAT-3A”
CR-2005-02594, “During performance of MST-071.11 found cell # 116 float voltage to be 2.09 VDC”
CR-2004-05334, “Six cells found below Category A or B limits of TSs”
System Health Report for Standby Liquid Control System, 3rd Quarter 2007
JAF-RPT-SLC-02282, “Maintenance Rule Basis Document for System 011 Standby Liquid Control System,” Revision 3
Drawing FM-21A, “Flow diagram Standby Liquid Control,” Revision 36
CR-2005-01107, “B SLC pump plunger leakage”
CR-2006-01488, “SLC pump combined discharge local pressure indication fluctuation”
CR-2006-01981, “SLC tank has material suspended in it”
CR-2007-00187, “Erroneous SLC tank level indication”
CR-2007-01468, “SLC tank level indication is slowly trending upward”

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

ST-2AM, “RHR Loop B Quarterly Operability Test,” Revision 24
AP-10.09, “Outage Risk Assessment,” Revision 22
CR-2007-04027, “A Technical Specification LCO was not entered”
CR-2007-04019, “Performance of the wrong ISP”
JAF On-line Schedule C17-0506-P3, “0746 Risk and Impact Profile”
JAF On-line Schedule C17-0506-P3, “0748 Weekly Work”
JAF On-line Schedule C17-0506-P3, “0748 Risk and Impact Profile”
ISP-175A2, “Reactor and Containment Cooling and ATWS Instrument Functional Test/ Calibration,” Revision 18
ISP-175A1, “Reactor and Containment Cooling Instrument Functional Test/ Calibration,” Revision 12
AP-10.10, “On-line Risk Assessment, “ Revision 5
EN-WM-101, “On-line Work Management Process,” Revision 0
EN-WM-109, “Scheduling,” Revision 1
EN-DC-151, “PSA Maintenance and Update,” Revision 1
FO184 Risk Assessment
AP-05.13, “Maintenance During LCOs,” Revision 9
JAF-RPT-FWS-03079, “Maintenance Rule Basis Document for System 034 Feedwater System, “ Revision 0
System Health Report for the Feedwater System, 3rd Quarter 2007

CR-JAF-2007-03519, “When performing quarterly ST per MST-071.11 the following cells did not meet acceptance 1 criteria”

Section 1R15: Operability Evaluations

Vendor Drawing Number JVA-206-384, " 3-way Solenoid Valve"
FM-46B, "Flow Diagram, Emergency Service Water System," Revision 50
CR-2003-03708, "Wall thinning found below administrative limits"
CR-2003-01333,"A leak was identified on an ESW line in the East Crescent Unit Coolers"
ENN-DC-185, "Through-wall Leaks in ASME Section IX Class 3 Moderate Energy Piping Systems," Revision 0
JAF-CALC-SWS-02407, "Minimum Wall Thickness and Structural Integrity for JAF Service Water Piping," Revision 0
FM-21A, "Flow Diagram Standby Liquid Control System 11," Revision 36
ST-6HB, "Standby Liquid Control B Side Quarterly Operability Test," Revision 2
System Health Report for the Standby Liquid Control System, 2nd Quarter 2007
JAF-RPT-SLC-02282, "Maintenance Rule Basis Document for System 011 Standby Liquid Control System," Revision 3

Section 1R17: Permanent Plant Modifications

DBD-046, Service Water Systems, Revision 9
CR-JAF-2007-03202, "Initiated a manual reactor scram due to lowering intake water level"
CR-JAF-2007-03580, "Noted decreasing trend in intake water level"
Drawing 2.95-10, Traveling Water Screen Drive Assembly
WO 001126861 for contingency shear pin replacement
Drawing 2.95-4, Traveling Water Screen General Arrangement
Calculation 91-024, MCC loading, Revision 3
OP-4, Circulating Water System, Revision 37
FM-36A, Flow Diagram Circulating Water System, Revision 36
FM-7A, Machine Location Screenwell, Revision 31

Section 1R19: Post Maintenance Testing

MST-071.11, "LPCI Battery Quarterly Surveillance Test," Revision 19
MST-071.10, "LPCI Battery Weekly Surveillance Test," Revision 32
OP-43C, "LPCI Independent Power Supply System," Revision 17
System Health Report for DC Electrical Distribution System, 2nd Quarter 2007
St-9BB, "EDG B and D Full Load Test and ESW Pump Operability Test," Revision 8
CR-2007-03871, "EDG D Load Breaker failed to close on first attempt during ST-9BB"
Work Order 51523284, "ST- Quarterly Inspection on A LPCI Battery"
Work Order 51560470, "ST- Weekly Inspection and Surveillance of Batteries 71BAT-3"
CR-2007-04012, "While Performing A LPCI Battery Changeout an Arc Occurred"
ST-6HB, "Standby Liquid Control B Side Quarterly Operability Test," Revision 2
CR-2007-03507, "B SLC Pump failed ST-6HB Level one acceptance criteria for required flow rate"

ASME OM Code-2001, "Code of Operation and Maintenance of Nuclear Power Plants"
ST-8Q

Section 1R20: Refueling and Other Outage Activities

CR-2007-03747, “A manual reactor scram was inserted due to low intake water level”
CR-2007-03751, “While placing B RHR system in SDC mode, prior to starting the RHR pump, it was observed 10MOV17 & 18 had closed”
Operations shift standing order 2007-020, “Lake Condition Monitoring,” Revision 0
CR-2007-03808, “Broken Clevis on Condensate Support #H33-22”
Transient No. 07-003 Post Transient Evaluation
AP-03.01, “Post Transient Evaluation,” Revision 11
RAP-7.3.16, “Plant Power Changes,” Revision 41
OP-65, “Start-up and Shutdown Procedure,” Revision 106
CR-2007-03580, “Operability of Raw water systems based on algae intrusion on October 13, 2007”
Raw Water Systems Startup Monitoring Plan, October 15, 2007
CR-2007-03519, “When performing quarterly ST per MST-071.11 the following cells did not meet acceptance 1 criteria”
CR-2007-03592, “20AOV-95 leaks by seat at .3gpm”
CR-2007-03580, “10/13/2007 noted lowering trend on intake water level”
CR-2007-03605, “Pinhole leaks in 10 RHR 202 identified by RP”
CR-2007-03585, “When shutting down B condensate pump operators observed pump reverse rotating”
CR-2007-03586, “During unplanned reactor shutdown, a plant power/flow map exclusion zone entry occurred”
CR-2007-03594, “Strainer basket removed from 67YS-281 on 10/15/2007 was approximately 40% fouled with green algae”
TOP-373, ‘Fill and Vent Feedwater System During and Following 34MOV-100A Maintenance,” Revision 0
OP-3, “Condensate System,” Revision 49
CR-2007-03851, “During startup of the A RFP, the discharge valve, 34MOV-100A, would not open”
CR-2007-03912, “On 11/07/07 at 0300, operators observed algae/weed on the traveling screens and traveling screen effluent wash”

Section 1R22: Surveillance Testing

CR-2007-03570, “ESW target flow rates for unit coolers 66UC-22C and 67UC-16A were not met”
CR-2007-03776, “CRDU 26-11 required entry into AOP-24 to move from the 00 position to full out”
CR-2007-03811, “During performance of ST-20T, control rod 46-11 was missing position indication for positions 01, 11, 21, 31, 41”

Section 1R23: Temporary Plant Modifications

TOP-372, “Administrative Control of SDIV Vent and Drain Valves,” Revision 0
FM-39D, “Instrument Air Reactor Building System 39,” Revision 10
FM-39C, “Flow Diagram Instrument Air Reactor Building and Drywell System 39,” Revision 26

Section2OS1: Access Control to Radiologically Significant Areas

JAF Focused Self-assessment Report LO-JAFLO-2007-0109, Control of Radioactive Contamination and Radioactive Material
JAF Focused Self-Assessment Report LO-JAFLO-2007-0021, External Radiation dose Control
JAF Snapshot Self-Assessment Report LO-JAFLO-2007-0078, Temporary Shielding Program
Corporate Assessment of Radiation Protection Fundamentals, March 19-23, 2007
JAF Snapshot Self-Assessment Report LO-JAFLO-2007-0005CA00011, Temporary Shielding Program
JRP-APL-05-009, Radiation Field Control Plan

Section 2OS2: ALARA Planning and Controls

JAF Focused Self-Assessment Report LO-JAFLO-2007-0101, Radiation Dose Reduction
Fitzpatrick Five Year ALARA Plan, 2007-2011
JRP-APL-05-009, Radiation Field Control Plan

Procedures

EN-RP-108, Revision 5, Radiation Protection Posting
EN-RP-105, Revision 2, Radiation Work Permits
EN-RP-141, Revision 2, Job Coverage
EN-RP-101, Revision 2, Access Control for Radiological Controlled Areas
RP-OPS-03.03, Revision 9, Radiological Postings and Labels

Section 20S3: Radiation Monitoring Instrumentation and Protective Equipment

Procedures
SAP-2, Revision 43, Emergency Equipment Inventory
RP-RESP-02.03, Revision 7, Self-Contained Breathing Apparatus, Scott Pressure Pack 4.5
RP-RESP-03.03, Revision 4, Breathing Air Testing and Use
RE-INS-7CG-5, Revision 8, Calibration of the Merlin-Gerin Electronic Dosimeters Using WCDM 2000
RP-INST-04.08, Revision 2, MGPI Telepole WR Extendable GM Survey Meter
RP-INST-05.02, Electrometer, Victoreen Model 500
RP-INST-05.03, Revision 2, Calibrator, J. L. Shepherd Model 89
RP-INST-05.04, Revision 3, Irradiator, Shepherd Panoramic Model 142-10
RP-INST-02.01, Revision 2, Teletector Survey Meter, Model 6112B
RP-INST-02.04, Revision 5, Count Rate Meter, Ludlum Model 177
RP-INST-02.05, Revision 2, Geiger Meuller Survey Meter

RP-INST-02.06, Revision 2, Dose Rate Meter, Bicron Micro-Rem
RP-INST-02.08, Revision 2, Ion Chamber Dose Rate Meter
RP-INST-02.09, Revision 3, Mini-Scalar MS-2, MS-3
RP-INST-02.10, Revision 1, Scintillation Alpha Counter, Eberline Model SAC-4
RP-INST-02.12, Revision 2, Electronic Dosimeter, Merlin Gerin Products Instruments

RP-INST-03.01, Revision 3, Area Radiation Monitors
RP-INST-03.03, Revision 7, Containment Radiation Monitor System
RP-INST-04.01, Revision 4, Area Radiation Monitor, Dosimeter Corporation
RP-INST-04.07, Revision 2, Area Radiation Monitor, AMP-100/200

Section 4OA2: Identification and Resolution of Problems

Condition Reports

2007-03572
2007-03564
2007-04001
2007-04007
2007-04012
2007-03987
2007-03745
2007-03746
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2007-03770
2007-02814
2007-00827
2007-01111
2007-01035

James A. FitzPatrick Quarterly Trend Report

Root Cause Analysis Report, “Intake Cooling Water Blockage” including CRs 2007-03202, 2007-03580, and 2007-03747
LER 90-023, “Manual Scram, Blocked Intake Screens”
Post Transient Evaluation Numbers 07-001, 07-002 and 07-003
Root Cause Analysis Report, “34MOV-100A Stem/Disc Separation”
WO JF-020804400, 2004 34MOV100A rebuild
2007-02995, 34MOV-100A hardened grease and stem misalignment
General Electric Service Information Letter (SIL) No. 368, Revision 1, Gate Valve Lockup
FM-34A Flow Diagram Feedwater system 34, Revision 59
Drawing 6.37-113, Motor Operated Gate Valve, 34MOV-100A, Revision 1
EC 3945, Drill a 1/8 inch hole in downstream disk of 34MOV-100A
OP-2A, Feedwater System, Revision 59
WO 00120488, Repair 34MOV-100A

<u>Section 4OA3: Event Follow-up</u>

2007-03211, “34FCV-137 Has Sluggish Response”
LER-07-002 (CR-JAF-2007-03202), “Manual Reactor Scram due to Blocked Circulating Water Intake Screens”
2007-03212, “With the reactor in mode 3 and cool-down in progress, a low reactor pressure vessel water level scram and primary containment isolation system Group 2 initiation occurred due to an actual low water level condition”
2007-03757, “Feedwater pump discharge min flow control valve experienced sluggish response following the reactor scram”
OP-2A, “Feedwater System,” Revision 59

LIST OF ACRONYMS

ADAMS	agencywide documents access and management system
ALARA	as low as is reasonably achievable
CR	condition report
IMC	inspection manual chapter
LCO	limiting condition for operation
LPCI	low pressure coolant injection
mrem	millirem
MSPI	mitigating systems performance index
NCV	non-cited violation
NRC	Nuclear Regulatory Commission
Pars	Publicly Available Records
PI	performance indicator
RP	radiation protection
RWP	radiation work permit
SDP	significance determination process
SSC	structures, systems, or components
SRVS	safety relief valves
TS	technical specification
TBCLC	turbine building closed loop cooling
UFSAR	updated final safety analysis report