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

August 12, 2009

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 05000333/2009003

Dear Mr. Dietrich:

On June 30, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your James A. FitzPatrick Nuclear Power Plant (FitzPatrick). The enclosed inspection report documents the inspection results which were discussed on July 9, 2009, 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.

Based on the results of this inspection, five findings of very low safety significance (Green) were identified. Four of these findings were determined to be violations of NRC requirements. Additionally, two licensee-identified violations which were determined to be of very low safety significance are listed in this report. However, because of the very low safety significance, and because the violations were entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV, you should provide a response within 30 days of the date of the inspection report, with the basis for your denial, to the U. S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with a copy to the Regional Administrator, Region I; Office of Enforcement; U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspector at FitzPatrick. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Resident Inspectors at FitzPatrick. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR Part 2.390 of the NRC's "Rules of Practice," a copy of this letter and 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 Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Mel Gray, Chief
Projects Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000333/2009003
w/Attachment: Supplemental Information

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Sincerely,
/RA/
Mel Gray, Chief
Projects Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000333/2009003
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.:	05000333/2009003
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Scriba, New York
Dates:	April 1 through June 30, 2009
Inspectors:	G. Hunegs, Senior Resident Inspector S. Rutenkroger, PhD, Resident Inspector J. Commiskey, Health Physicist J. Heinly, Reactor Engineer J. Noggle, Senior Health Physicist M. Patel, Reactor Inspector A. Rosebrook, Senior Project Engineer
Approved by:	Mel Gray, Chief Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000333/2009003; 04/01/2009 - 06/30/2009; James A. FitzPatrick Nuclear Power Plant (FitzPatrick); Equipment Alignment; Operability Evaluations; Surveillance Testing; and ALARA Planning and Controls.

The report covered a three-month period of inspection by resident inspectors and announced inspections by region-based inspectors. Five Green findings, of which four were 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). The cross-cutting aspect for each finding was determined using IMC 0305, "Operating Reactor Assessment Program." 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.

Cornerstone: Mitigating Systems

- Green: The inspectors identified an NCV of very low safety significance of 10 CFR 50, Appendix B, Criterion III, "Design Control," because Entergy personnel did not maintain a high energy line break (HELB) barrier. Specifically, HELB door 76 FDR-DG-272-11, located between the 'A' division emergency diesel generator (EDG) switchgear room and the turbine building was in use as a HELB barrier but was not qualified due to a missing support. The issue was entered into Entergy's corrective program as condition report (CR)-JAF-2009-01895. Corrective actions included installing a lower bottom right side support to enable the door to be qualified for HELB.

  This finding is greater than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, Entergy's engineering calculation previously documented that the door could not be qualified with a missing lower support. The inspectors evaluated the significance of this finding using IMC 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The finding was determined to be of very low safety significance (Green) because the finding was a qualification deficiency confirmed not to result in loss of operability.

  The inspectors determined that this finding has a cross-cutting aspect in the area of human performance within the work practices component because Entergy personnel did not ensure that the secondary HELB barrier was qualified as a result of ineffective error prevention techniques. (H.4(a)) (Section 1R04)

- Green: A self-revealing NCV of very low safety significance of 10 CFR 50.55a, "Codes and Standards," was identified because Entergy personnel did not comply with the in-service testing (IST) program requirements contained within the applicable American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants. Specifically, Entergy personnel changed the reference value for the stroke time of the 23HOV-1, high pressure coolant injection (HPCI) turbine stop valve, without meeting the required ASME code criteria. Entergy's corrective actions included replacing the relay valve piston, lapping the relay valve seat, implementing procedure changes requiring additional evaluation within a decreased range of stroke times to open, and performing an extent of condition review of the IST program.

This finding is greater than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, Entergy personnel did not identify a prior adverse performance trend which resulted in an unplanned extension of the maintenance period for the HPCI system, extending the unavailable period from January 23, 2009 through January 31, 2009. The inspectors determined that the finding was of very low safety significance (Green) using the SDP Phase 3, in accordance with IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations."

The inspectors determined this finding had a cross-cutting aspect in the area of human performance within the resources component because Entergy personnel did not ensure that the procedures and other resources available for inspecting 23HOV-1 and evaluating its performance under the IST program were adequate to assure nuclear safety. (H.2(c)) (Section 1R15)

- Green: A self-revealing NCV of very low safety significance of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified because Entergy personnel did not identify and correct a condition adverse to quality related to the HPCI system which caused the system to be inoperable between January 30 and April 28, 2009. Specifically, the balance chamber pressure for the HPCI turbine stop valve, 23 HOV-1, was not set at a value to ensure proper operation of the HPCI turbine system and resulted in a HPCI high steam flow isolation during the performance of the surveillance test. Entergy personnel entered the condition into their corrective action program as CR-JAF-2009-01398. Corrective actions included the performance of a root cause analysis, adjustment of the balance chamber pressure to be higher in the acceptance band consistent with operating experience and increasing the frequency of HPCI surveillance testing.

  This finding is greater than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, Entergy personnel did not take adequate corrective action to establish the balance chamber pressure for 23 HOV-1, following an erratic fast opening of the valve on January 30, 2009. The inspectors determined that the finding was of very low safety significance (Green) using the SDP Phase 3, in accordance with IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations."

  The inspectors determined that this finding had a cross-cutting aspect in the area of human performance within the decision-making component because after reviewing the available data and industry operating experience, in January 2009, Entergy personnel incorrectly determined that balance chamber pressure margin was not a contributing cause of the erratic operation of the valve. (H.1(b)) (Section 1R22)

- Green: A self-revealing NCV of very low safety significance of 10 CFR 50, Criterion XVI, "Corrective Action," was identified because Entergy personnel did not identify and correct a condition adverse to quality related to the emergency diesel generator (EDG) system. Specifically, Entergy personnel did not properly identify and implement adequate actions required by their system monitoring program in response to a degraded generator rotor on the 'C' EDG revealed by an adverse performance trend with respect to the insulation resistance and polarization index. Entergy staff initiated CR-JAF-2009-01847 to determine the root causes and recommend further corrective actions. Entergy's corrective actions included rewinding of the affected pole of the 'C' EDG rotor.

  This finding is greater than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, Entergy personnel did not identify an adverse performance trend which resulted in an unplanned extension of the maintenance period for the 'C' EDG, extending the unavailable period from May 28 through June 11, 2009. The inspectors evaluated the significance of this finding using IMC 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The inspectors determined the finding was of very low safety significance (Green) because the finding was not a qualification or design deficiency, did not represent a loss of a safety function, and did not screen as potentially risk significant due to external initiating events.

  The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy personnel did not implement a corrective action program with a low threshold for identifying issues in that the adverse trend in the 'C' EDG rotor insulation was not identified. (P.1(a)) (Section 1R22)

Cornerstone: Occupational Radiation Safety

Green. A self-revealing finding of very low safety significance was identified because Entergy personnel did not adequately plan and prevent unnecessary exposure consistent with Radiation Work Permit No. 08-0524 controls. Specifically, Entergy staff work planning deficiencies relative to a main steam line strain gauge modification resulted in additional unplanned collective exposure (11.32 person-rem compared to a work activity original estimate of 6.1 person-rem). The job site conditions for installation of the strain gauges were not adequately evaluated by Entergy staff for interferences and the support work involving scaffolding and insulation removal were not adequately planned and coordinated to prevent additional unnecessary exposure. This finding was entered into the corrective action program as CR-JAF-2008-3181.

This finding is greater than minor because it is associated with the program and process attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine nuclear reactor operation. The inspectors evaluated the significance of this finding using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process." The inspectors determined this finding was of very low safety significance (Green) because it involved an actual collective exposure greater than 5 person-rem that was greater than 50% above the estimated or intended exposure.

This finding has a cross-cutting aspect in the area of human performance because Entergy's planned work activities did not adequately incorporate work site interferences or outage work coordination in the work control planning process. (H.3(a)) (Section 2OS2)

Other Findings

- Violations of very low safety significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. The violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The James A. FitzPatrick Nuclear Power Plant (FitzPatrick) began the inspection period operating at 100 percent reactor power. On April 13, 2009, operators reduced reactor power to 55 percent to repair condenser tube leaks. Following repairs, reactor power was restored to 100 percent on April 15, 2009. On May 3, 2009, operators reduced reactor power to 67 percent due to a loss of level control in a feedwater heater and following restoration of level control, restored reactor power to 100 percent the same day. On June 17, 2009, operators reduced reactor power to 60 percent to perform a planned control rod sequence exchange and restored reactor power to 100 percent the same 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 – 2 samples)

.1 Evaluate Summer Readiness of Offsite and Alternate AC Power Systems

a. Inspection Scope

The inspectors reviewed operating procedures to verify continued availability of offsite and alternate AC power systems. The inspectors also reviewed Entergy's agreements and protocols established with the transmission system operator to verify that the appropriate information is exchanged when issues arise that could impact the offsite power system. The documents reviewed are listed in the Attachment. This inspection represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Seasonal Weather Conditions

a. Inspection Scope

The inspectors reviewed and verified completion of the warm weather preparation checklist contained in procedure AP-12.04, "Seasonal Weather Preparations," Revision 17. The inspectors reviewed the operating status of the control room and battery room ventilation systems, reviewed the procedural limits and actions associated with elevated lake and air temperatures, and walked down accessible areas of the battery room and control room ventilation areas to assess the effectiveness of the ventilation systems. Discussions with operations and engineering personnel were conducted by the inspectors to ensure plant personnel were aware of temperature restrictions and required actions. The documents reviewed are listed in the Attachment. The inspection satisfied one inspection sample for seasonal weather conditions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdown (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 system procedures, the Updated Final Safety Analysis Report (UFSAR), and system drawings in order to verify 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 (WO) to ensure that Entergy personnel identified and properly addressed equipment discrepancies that could impair the capability of the available equipment train, as required by 10 CFR 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:

- RCIC system when the HPCI system was out of service due to emergent work;
- 'A' EDG subsystem when the 'B' EDG subsystem was out of service for maintenance; and
- 115 kilovolt (kV) offsite power sources and 'A' EDG subsystem switchgear when the 'C' EDG was out of service for emergent maintenance.

These activities constituted three partial system walkdown inspection samples.

b. Findings

Introduction: The inspectors identified an NCV of very low safety significance of 10 CFR 50, Appendix B, Criterion III, "Design Control," because Entergy personnel did not maintain an adequate high energy line break (HELB) barrier. Specifically, the inspectors identified that HELB door 76 FDR-DG-272-11 was used as a HELB barrier but was not qualified due to a missing bottom right side support.

Description: In the event of a HELB, credited structural barriers at the station are designed to withstand a differential pressure resulting from the HELB. These barriers function to separate harsh environmental areas from mild environmental areas, such that all safe shutdown components are properly qualified for the environmental conditions to which they might be subjected.

On May 29, 2009, Entergy personnel established the secondary HELB barrier, door 76FDR-DG-272-11, in order to breach the primary HELB barrier per AP-16.14, "Hazard Barrier Controls." Entergy staff had revised this procedure to allow removal of the primary HELB barrier, door 76FDR-E-272-3, to transport the 'C' EDG rotor offsite for repair. Secondary HELB barriers were qualified to allow breach activities, such as maintenance, while the plant remained in operation to provide the necessary protection from the effects of a potential HELB. After the rotor was removed, the inspectors reviewed the design basis for the removal path. The inspectors verified that calculation JAF-CALC-MISC-03340 documented that the secondary HELB barrier door 76FDR-DG-272-11 was an acceptable HELB door. However, the inspectors observed that the calculation identified that door 76FDR-DG-272-11 could not be qualified in its present condition due to the missing bottom right side support. The missing support was not noted by Entergy personnel involved with implementing the program to allow use of the secondary HELB barrier. On May 30, the inspectors observed that the bottom right side support was missing. The issue was entered into Entergy's corrective program as CR-JAF-2009-01895. Entergy personnel installed a lower bottom right side support to enable the door to be HELB qualified.

Analysis: The inspectors identified a performance deficiency in that Entergy personnel incorrectly designated an unqualified door to be a HELB barrier. This finding is greater than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the door could not be qualified for a HELB barrier with a missing lower support and there was visual evidence that the support was missing. The inspectors evaluated the significance of this finding using IMC 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and determined it to be of very low safety significance (Green) because the finding was a qualification deficiency confirmed not to result in loss of operability per "Part 9900, Technical Guidance, Operability Determination Process for Operability and Functional Assessment."

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance within the work practices component because Entergy personnel did not ensure that the secondary HELB barrier was qualified as a result of ineffective error prevention techniques. (H.4(a))

Enforcement: 10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Contrary to the above, Entergy personnel did not ensure appropriate quality standards were specified and controlled to ensure that a secondary HELB barrier met design requirements. Specifically, a bottom right side support from the HELB barrier, 76FDR-DG-272-11, was missing which resulted in the HELB door not meeting qualification requirements when it was in use on May 29 and 30, 2009. Because the violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000333/2009003-01, High Energy Line Break Door Missing Lower Support)**

.2 Complete System Walkdown (71111.04S – 1 sample)

a. Inspection Scope

The inspectors performed a complete system alignment inspection of the control and relay room ventilation systems to identify discrepancies between the existing equipment lineup and the required lineup. During the inspection, system drawings and operating procedures were used to verify proper equipment alignment and operational status. The inspectors reviewed the open maintenance WOs associated with the systems for deficiencies that could affect the ability of the systems to perform their function. Documentation associated with open design issues such as temporary modifications, operator workarounds and items tracked by plant engineering were also reviewed by the inspectors to assess their collective impact on system operation. In addition, the inspectors reviewed the CR database to verify equipment problems were being identified and appropriately resolved. The documents reviewed are listed in the Attachment.

These activities constituted one complete system walkdown inspection sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Review (71111.05Q – 5 samples)

a. Inspection Scope

The inspectors conducted inspections of 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 for conformance with the requirements of Licensee Condition 2.C.3. The documents reviewed are listed in the Attachment.

- Fire Area/Zone V/EG-1, EG-2, EG-5, elevation 272 foot;
- Fire Area/Zone VI/EG-3, EG-4, EG-6, elevation 272 foot;
- Fire Area/Zone III/BR-2, IV/BR-3, BR-4, XVI/BR-5, elevation 272 and 282 foot;
- Fire Area/Zone IX/RB-1A, elevation 369 foot; and
- Fire Area/Zone XII/SP-1, XIII/SP-2, IB/FP-1, FP-3, elevation 255 foot.

These activities constituted five quarterly fire protection inspection samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 – 1 sample)

a. Inspection Scope

The inspectors conducted tours of the EDG rooms and the adjacent switchgear rooms to assess internal flooding protection measures in those areas. The inspectors reviewed selected risk significant plant design features intended to protect the associated safety-related equipment from internal flooding events. The inspectors reviewed flood analysis and design documents, including the Individual Plant Examination, UFSAR, and engineering evaluations. The documents reviewed are listed in the Attachment.

These activities constituted one internal flood protection measures inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Quarterly Review (71111.11Q – 1 sample)

a. Inspection Scope

On April 7, 2009, the inspectors observed licensed operator simulator training to assess performance during scenarios to verify that crew 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. Licensed operator training was evaluated for conformance with the requirements of 10 CFR 55, "Operators' Licenses." The documents reviewed are listed in the Attachment.

This activity constituted one operator simulator training inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12 – 1 sample)

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 the following aspects when applicable:

- Proper maintenance rule scoping in accordance with 10 CFR 50.65;
- Characterization of reliability issues;
- Changing system and component unavailability;
- 10 CFR 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 the control and relay room ventilation systems including applicable system health reports, maintenance backlogs, and maintenance rule basis documents. The inspectors evaluated the maintenance program for conformance with the requirements of 10 CFR 50.65. The documents reviewed are listed in the Attachment.

These activities constituted one quarterly maintenance effectiveness inspection sample.

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 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 week of April 13, 2009, which included increased risk due to a control rod sequence exchange and condenser tube leak repairs, traveling water screens out of service for intake cleaning, and surveillances involving the ‘B’ residual heat removal (RHR) system;
- The week of April 20, 2009, which included emergent work on the HPCI system and emergent work resulting in placing the reactor protection system bus ‘A’ on the alternate power supply;
- The week of May 11, 2009, which included increased risk due to ‘B’ EDG maintenance and independent spent fuel storage cask heavy lifts;
- The week of May 25, 2009, which included increased risk due to ‘C’ EDG maintenance, independent spent fuel storage cask heavy lifts, and ‘A’ traveling water screen replacement; and
- The week of June 1, 2009, which included increased risk due to emergent ‘C’ EDG maintenance, independent spent fuel storage cask heavy lifts, and ‘A traveling water screen replacement.

These activities constituted five maintenance risk assessments and emergent work control samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 – 5 samples)

a. Inspection Scope

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

- CR-JAF-2009-01662, 345kV potential transformer corona noise and discoloration of connections;
- CR-JAF-2009-01692, Maintenance records, related to environmental qualifications for safety relief valve pilot solenoid operated valve connections during 2008 refueling outage (RO18), were not able to be located;
- CR-JAF-2009-01847, Common cause failure review for EDG abnormal electrical characteristics required by TS 3.8.1, required action B.3.1;
- CR-JAF-2009-02011, 'C' EDG exhibited abnormal drift during 110% load;
- CR-JAF-2009-01895, Lower support plate on the south emergency switchgear door was missing and the door was credited as providing the HELB barrier; and
- CR-JAF-2009-00350, HPCI valve 23HOV-1 failed to open (operability evaluation sample was previously credited in NRC inspection report 2009-002).

These activities constituted five operability evaluation samples.

b. Findings

Introduction: A self-revealing NCV of very low safety significance of 10 CFR 50.55a, "Codes and Standards," was identified because Entergy personnel did not comply with the IST program requirements contained within the applicable ASME Code for Operation and Maintenance of Nuclear Power Plants. Specifically, Entergy personnel changed the reference value for open stroke time of the 23HOV-1, HPCI turbine stop valve, without meeting the required code criteria.

Description: On January 19, 2009, operators entered TS 3.5.1, 'ECCS Operating" to conduct various planned HPCI maintenance activities. On January 23, Entergy staff performed ST-4N, "HPCI Quick-Start, Inservice, and Transient Monitoring Test," to complete post maintenance testing requirements. Entergy staff measured the stroke time of 23HOV-1 at 37.9 seconds which was outside the procedure's acceptance criteria range of 16.6 to 27.6 seconds. Operators completed initial corrective actions, such as venting the hydraulic oil system and installing additional instrumentation. On January 26, operators conducted additional tests in which 23HOV-1 failed to stroke in two successive tests. Entergy personnel documented the condition in CR-JAF-2009-0350.

Maintenance workers lapped the 23HOV-1 hydraulic control oil relay valve seat, improving the seat contact from 30% to 100%, replaced the auxiliary oil pump as a precautionary measure, and replaced the 23HOV-1 relay valve piston. The HPCI system was restored to an operable and available status on January 31. Entergy personnel determined that the root cause of the malfunction of 23HOV-1 to open was oil leakage through the relay valve which prevented adequate pressure from being available to open 23HOV-1 in the required time.

By design, hydraulic control oil inlet flow to the relay valve is limited by a 3/16 inch orifice, giving an effective inlet flow area of 0.0276 square inches. Due to the measured gap between the bore of the relay valve and the replaced relay valve piston, Entergy personnel determined the leakage flow area through this gap to be 0.0353 square inches. With the new relay valve piston installed, Entergy personnel determined the leakage flow area was reduced to 0.0235 square inches, reducing the oil leakage through the relay valve and allowing sufficient oil pressure to move the relay valve piston and thereby open 23HOV-1 within the required time.

Entergy's root cause evaluation determined that prior to the January 2009 maintenance outage, on August 10, 2005, Entergy personnel performed an IST evaluation, and increased the stroke time acceptance criteria for 23HOV-1 from the range 14.6 to 24.3 seconds to the range 16.6 to 27.6 seconds. The actual 23HOV-1 stroke time increased from approximately 18 seconds, beginning in 2000, to 25.12 seconds on January 25, 2007, which exceeded the previous acceptance criteria. In addition, the inspectors noted the stroke time measurements became increasingly erratic starting in 2007. For example, the stroke time to open increased from 18.18 seconds on May 4, 2007, to 24.28 seconds on August 24, 2007. For the last surveillance test prior to January 2009, on October 10, 2008, the stroke time to open had increased to 27 seconds from 23.15 seconds on August 15, 2008.

Entergy staff's root cause evaluation concluded that the change to the performance criteria was technically unsupported because it was performed as a re-baseline of the reference value from 19.45 seconds to 22.12 seconds with no reference to recent physical component work activity which would justify an increasing trend. A change to an IST reference value is allowed per ASME OM Code-2003 Addenda to ASME OM Code-2001, "Code for Operation and Maintenance of Nuclear Power Plants," provided that a documented verification is performed such that the new values represent acceptable operation. However, Entergy personnel did not document such verification.

Entergy personnel concluded the program change to the valve stroke to open time masked a degrading overall trend for the valve to stroke open. Although the actions procedurally required by the IST program were masked by the higher acceptance criteria established in 2005, the inspectors also determined that Entergy personnel did not recognize an adverse trend in the performance of the 23HOV-1 valve in 2007 and 2008 when the stroke time of the valve increased to a peak opening time of approximately 27 seconds on October 10, 2008. In particular, the inspectors determined it was reasonable for Entergy engineers to identify during IST surveillance test reviews that the documented stroke times compared to the respective previous stroke times indicated a degrading trend.

Entergy's corrective actions included replacing the relay valve piston, lapping the relay valve seat, implementing procedure changes requiring additional evaluation within a decreased range of stroke times to open, and performing an extent of condition review of the IST program.

Analysis: There was a self-revealing performance deficiency in that Entergy personnel changed the reference value for stroke time of the 23HOV-1, HPCI turbine stop valve, without meeting the required code criteria and did not identify a degraded trend with the valve's opening stroke time. This finding is greater than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the 23HOV-1 degraded valve performance resulted in unplanned work and extension of the maintenance period for the HPCI system, extending the unavailable period from January 23 through January 31, 2009.

The finding was determined to be of very low safety significance in accordance with IMC 0609, Appendix A, using SDP Phases 1, 2 and 3. In accordance with 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding was determined to require a detailed Phase 2 evaluation due to an actual loss of the safety function because the HPCI system is a single train system for the high pressure safety injection function. The inspectors conducted a Phase 2 evaluation using the FitzPatrick Pre-solved Risk-Informed Inspection Notebook, and concluded that a Phase 3 evaluation was needed to assess the significance. A Region I SRA conducted a Phase 3 analysis and concluded that the finding was of very low safety significance (Green).

The SRA used the FitzPatrick Standardized Plant Analysis Risk (SPAR) model assuming that HPCI was in an unplanned, non-recoverable maintenance condition for 8 days, which indicated an increase in the delta core damage frequency (ΔCDF) for internal initiating events in the range of 1 core damage accident in 5,000,000 years of reactor operation, in the low E-7 range per year. The dominant core damage sequences included the failure of both HPCI and RCIC systems and the failure of operators to depressurize the reactor following a loss of the ability to reject decay heat to the condenser.

The SRA assessed the impact of the finding on: 1) external events such as fire, seismic and flooding, determining, based on review of the FitzPatrick Individual Plant Examination for External Events, that the total ΔCDF (internal plus external) would not be above the 1E-6 threshold; and 2) the delta large early release frequency (ΔLERF), determining that given the operators ability, following core damage, to depressurize and inject water to the reactor from low pressure sources and to flood the containment, that the ΔLERF was in the low E-8 range.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance within the resources component because Entergy personnel did not ensure that the procedures and other resources available for inspecting 23HOV-1 and evaluating its performance under the IST program were adequate to assure nuclear safety. (H.2(c))

Enforcement: 10 CFR 50.55a, "Codes and standards," states, in part, that pumps and valves which are classified as ASME code Class 1, Class 2, and Class 3 must meet the inservice test requirements set forth in the ASME OM Code. Furthermore, inservice tests to verify operational readiness of pumps and valves, whose function is required for safety must comply with the requirements of the ASME OM Code. Contrary to this, from August 10, 2005, through January 23, 2009, Entergy personnel inappropriately implemented the ASME OM Code when they established and used a reference value for 23HOV-1 without appropriate technical justification and verification that the valve was operating acceptably. Entergy personnel took corrective actions to replace the relay valve piston and lap the relay valve seat. Because this violation was of very low safety significance and it was entered into Entergy's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (**NCV 05000333/2009003-02: Failure to Recognize an Adverse HPCI Performance Trend.)**

1R18 Plant Modifications (71111.18 – 1 sample)

a. Inspection Scope

The inspectors reviewed permanent plant modification EC-13018 which was implemented to eliminate the valves, 10SOV-101 A, B, C, and D, and re-route the cooling water supply piping to the residual heat removal service water (RHRSW) pump motor. The inspectors 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.

This activity constituted one permanent plant modification inspection sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed 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, 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 for conformance with the requirements of 10 CFR 50, Appendix B, Criterion XI, "Test Control." The documents reviewed are listed in the Attachment.

- WO 00147322, 10MOV-149A, RHRSW loop 'A' to residual heat removal cross-tie downstream isolation valve breaker maintenance;
- WO 00180283, HPCI system stop valve balance chamber adjustment;
- WO 51692500, 'B' EDG turbo-charger replacement;
- WO 00193991, Uninterruptible power supply motor generator set repair;
- WO 00148120, 'C' EDG rotor replacement; and
- WP 00148127, 'C' EDG rotor rewinding.

This inspection constituted six post-maintenance test samples.

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 (STs) and/or reviewed test data of selected risk-significant SSCs to assess whether the SSCs satisfied TSs, UFSAR, Technical Requirements Manual, and Entergy procedure requirements. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness, and were consistent with DBDs; test instrumentation had current calibrations, adequate range, and accuracy for the application; and tests were performed, as written, with applicable prerequisites satisfied. Upon ST completion, the inspectors verified that equipment was returned to the status specified to perform its safety function. The documents reviewed are listed in the Attachment. The following STs were reviewed:

- ST-23C, "Jet Pump Operability Test for Two Loop Operation," Revision 25;
- ST-4N, "HPCI Quick-Start, Inservice, and Transient Monitoring Test (IST)," Revision 56;
- ST-2AM, "RHR Loop 'B' Quarterly operability Test (IST)," Revision 27;
- ST-9QA, "EDG 'A' and 'C' Full Load Test (8 Hour Run)," Revision 6;
- ST-4N, "HPCI Quick-start, Inservice, and Transient Monitoring Test (IST)," Revision 56; and
- ST-09BA, "A and C Full Load Test and ESW Pump Operability Test," Revision 10 (with one-time temporary change effective only on June 22, 2009).

These activities represented six surveillance testing inspection samples.

b. Findings

.1 Balance Chamber Pressure for the HPCI Turbine Stop Valve Was Not Set at a Value to Ensure HPCI Operation

Introduction: A self-revealing NCV of very low safety significance (Green) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified because Entergy personnel did not identify and correct a condition adverse to quality related to the HPCI system which resulted in the HPCI system inoperability between January 30 and April 28, 2009. Specifically, the balance chamber pressure for the HPCI turbine stop valve, 23HOV-1, was not set at a value to ensure proper operation of the HPCI turbine system.

Description: On April 22, 2009, Entergy personnel performed a quarterly surveillance test on the HPCI system by conducting surveillance test ST-4N, "HPCI Quick-Start, Inservice, and Transient Monitoring Test (IST)." This was the first surveillance test after extensive HPCI maintenance was completed on January 30, 2009. During the initial HPCI startup sequence, a HPCI high steam flow isolation occurred with corresponding control room annunciators. HPCI steam line isolation valves closed as expected due to the isolation signal. Operators declared the HPCI system inoperable, placing the plant in a 14-day shutdown action statement in accordance with TS 3.5.1, "Emergency Core Cooling

Systems (ECCS)." Operators placed additional instruments on the HPCI system for monitoring and successfully started the HPCI system from relatively hot conditions on April 23 without the occurrence of a steam flow isolation signal. Entergy staff's analysis concluded the HPCI steam line isolation was caused by erratic fast opening of 23HOV-1 which caused a high steam flow condition and consequently the isolation. Entergy staff determined the direct cause of the erratic fast opening of 23HOV-1 was that the balance chamber pressure was adjusted too low for cold conditions. Entergy personnel implemented immediate corrective actions which included adjustment of 23HOV-1 balance chamber pressure and calibration checks of HPCI high steam flow transmitters subsequently followed by performance of HPCI hot and cold quick starts with satisfactory results. Entergy operators then performed a successful surveillance test on April 23 with the system hot and declared the HPCI system available. Entergy operators declared the HPCI system operable on April 28 after a successful performance of the surveillance test with the system at ambient cold conditions.

Prior to this occurrence, the last erratic fast opening of 23HOV-1 occurred on January 30, 2009. Entergy personnel determined that the root cause of the HPCI high steam flow isolation was the result of the "erratic fast opening of 23HOV-1 caused by balance chamber pressure set marginally low and an indeterminate effect resulting from maintenance performed on the valve in January 2009." Although the balance chamber pressure was within the range 100 to 180 pounds per square inch gauge (psig), as specified by IMP-23.12, "HPCI Stop valve Steam Balance Chamber Adjustment," Entergy personnel raised the pressure to 192 psig to assure proper valve operation.

The inspectors concluded that following the erratic opening of 23HOV-1 on January 30, Entergy personnel incorrectly determined that the cause was due to moisture carryover from the steam line due to operating the system under cold conditions. A cold quick start is more challenging to the balance chamber pressure margin than a hot start. Although the system had previously operated satisfactorily with the current balance chamber pressure, Entergy personnel did not fully consider the effect from the maintenance that was conducted on 23HOV-1 which resulted in the valve opening about 3 seconds sooner than previously. Additionally, Entergy personnel did not validate their cause and verify that the erratic valve condition had been corrected through the performance of a cold quick start which would have likely revealed a balance chamber pressure issue.

The inspectors determined that following erratic operation of the 23HOV-1 on January 30, Entergy personnel did not sufficiently evaluate available and applicable industry information related to the setting of 23HOV-1 balance chamber pressure. The inspectors determined the following operating experience, available to Entergy staff, were not adequately addressed:

- The "EPRI Terry Turbine Maintenance Guide, HPCI Application," dated November 2002 states that, with a nominal reactor pressure of 1000 psig, the balance chamber pressure range should be 150 to 200 psig (15% to 20% of steam line pressure). "When the balance chamber pressure is adjusted too low, the stop valve will experience erratic fast opening behavior." On January 30, 2009, the HPCI 23HOV-1 balance chamber pressure measured 144 psig. The EPRI manual also notes that the stop valve supplier has recommended a balance chamber pressure acceptance criteria of 10% to 15% of steam line pressure (100 to 150 psig). However, operating experience has shown the 10% value is too low for the cold quick startup transient.

The EPRI guide noted that there is a difference in balance chamber pressure between thermally hot and cold conditions, and it is critical that an adequate balance chamber pressure be demonstrated during the cold startup transient. If the HPCI turbine is thermally hot, the balance chamber pressure should be at the upper end of its tolerance.

- Additionally, operating experience in the form of a GE safety information letter (SIL) had shown that the 23HOV-1 balance chamber pressure needed to be raised to eliminate the potential for erratic fast opening. GE SIL No. 352, "HPCI Turbine Stop Valve Steam Balance Chamber Pressure Adjust," dated February 18, 1981, notes that "if the stop valve opening transient is erratic or unstable, balance chamber pressure adjustment will be required." The GE SIL No. 352 continues with "Problems with erratic opening of the HPCI turbine stop valve have been reported at several sites, identified primarily with the system "cold quick start transient."

Entergy personnel entered the condition into their corrective action program as CR-JAF-2009-01398. Corrective actions included the performance of a root cause analysis, adjustment of the balance chamber pressure to be high in the acceptance band and increasing the frequency of HPCI surveillance testing.

Analysis: There was a self-revealing performance deficiency identified in that Entergy personnel did not promptly identify and correct an adverse condition related to erratic opening of the HPCI turbine stop valve. This finding is greater than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, Entergy personnel did not take adequate corrective action to establish the balance chamber pressure for 23HOV-1, in accordance with applicable industry guidance, following an erratic fast opening of the valve on January 30, 2009. This resulted in a condition where HPCI was inoperable from January 30 to April 28, 2009, because system performance indicated it would have isolated on a high steam flow signal if called upon and would have required operator actions to restore its ability to supply water to the reactor coolant system.

The finding was determined to be of very low safety significance in accordance with IMC 0609 Appendix A, using SDP Phases 1, 2 and 3. In accordance with 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding was determined to require a detailed Phase 2 evaluation due to an actual loss of the safety function for greater than the allowable TS outage time. The inspectors conducted a Phase 2 evaluation using the FitzPatrick Pre-solved Risk-Informed Inspection Notebook, and concluded that a Phase 3 evaluation was needed to assess the significance. A Region I SRA conducted a Phase 3 analysis and concluded that the finding was of very low safety significance (Green).

The SRA used the FitzPatrick SPAR model, assuming that the HPCI system would isolate on high steam flow over an 87 day period, but be recoverable, under certain situations, by operator actions. Specifically, the SRA assumed that following a high steam flow isolation the operators could restore HPCI to operation given sufficient time following a failure of the RCIC system as long as the initiating event did not include a loss of RCS inventory. This assumption was supported by successful operation of the HPCI system from hot standby conditions on April 23, 2009. The non-recovery probability was conservatively assumed at a screening value of 0.1 (higher than the SPAR-human action calculation would assume) for situations where RCIC had failed and was not recoverable and 0.54, as calculated by the SPAR-human action calculation, for situations where RCIC was recoverable, but not recovered by the operators (i.e., a dependent operator action). This analysis indicated an increase in ΔCDF for internal initiating events in the range of 1 core damage accident in 2,000,000 years of reactor operation, in the mid E-7 range per year. The dominant core damage sequences included the operator failure to recover HPCI and RCIC and the failure of operators to depressurize the reactor following a loss of the ability to reject decay heat to the condenser. In accordance with IMC 0609 Appendix A, for a finding with an internal events ΔCDF above 1E-7, the SRA assessed the impact of the finding on: 1) external events such as fire, seismic and flooding, determining, based on review of the FitzPatrick Individual Plant Examination for External Events, that the total ΔCDF (internal plus external) would not be above the 1 E-6 threshold.; and 2) the ΔLERF, determining that given the operators' ability, following core damage, to depressurize and inject water to the reactor from low pressure sources and to flood the containment that the ΔLERF was in the high E-8 range.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance within the decision-making component because after reviewing the available data and industry operating experience in January 2009, Entergy personnel did not verify whether balance chamber pressure margin was a contributing cause of the erratic operation of the valve. (H.1(b))

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, from January 30 through April 28, 2009, Entergy personnel did not implement adequate measures related to a condition adverse to quality, associated with erratic HPCI turbine stop valve (23HOV-1) operation following an extended maintenance window, to assure the condition adverse to quality was identified and promptly corrected. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (**NCV 05000333/2009003-03: Balance Chamber Pressure for the HPCI Turbine Stop Valve Not Set at a Value to Ensure HPCI Operation)**

.2 Failure to Recognize an Adverse EDG Rotor Insulation Performance Trend

Introduction: A self-revealing NCV of very low safety significance of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified because Entergy personnel did not identify and correct a condition adverse to quality related to the emergency diesel generator (EDG) system. Specifically, Entergy personnel did not identify and implement adequate actions in response to a degraded generator rotor on the 'C' EDG revealed by an adverse performance trend with respect to the insulation resistance and polarization index.

Description: On May 26, 2009, Entergy personnel entered TS 3.8.1, "AC Sources – Operating," to conduct various planned EDG maintenance on the 'C' EDG. On May 27, Entergy personnel performed MP-093.04, "EDG Electrical Preventive Maintenance," to perform the electrical portion of the preventive maintenance activities. Entergy technicians measured the minimum 'C' EDG rotor (or field winding) insulation resistance to be below the acceptance criteria as specified in the procedure (0.039 Megohms versus 5.2 Megohms). In addition, Entergy technicians measured the minimum ‘C’ EDG field winding polarization index to be 1.0, the lowest value possible and below the acceptance criteria of 2.0.

Entergy personnel removed the rotor from the EDG generator and shipped the rotor to a vendor for repair. After receiving the repaired rotor, Entergy personnel restored the ‘C’ EDG, completed all post-maintenance testing, and exited TS 3.8.1 on June 11. Although the allowed outage time associated with this condition is 14 days which would have normally expired on June 9, 2009, Entergy staff submitted and the NRC approved TS Amendment 294 which provided a 3-day extension to the normal 14-day allowed outage time for TS 3.8.1 action B.4 for this specific issue only.

The inspectors determined that Entergy procedure EN-DC-159, “System Monitoring Program,” defines a degrading trend as an adverse change in measured or observed data that does not conform to expected/normal values after accounting for mode of operation, seasonal or environmental changes, or maintenance activity. EN-DC-159 also states that the required actions be taken when alert or action levels are exceeded as specified in the System Monitoring Plan. The System Monitoring Plan for System 093: Emergency Diesel Generators, specifies actions of increased frequency of monitoring and possibly rewind the generator when at or below a minimum polarization index of 1.25. The inspectors determined Entergy personnel did not previously take action when the minimum polarization index was found at 1.00 on September 18, 2007.

The inspectors determined that IEEE Standard 43-2000, “IEEE Recommended Practice for Testing Insulation Resistance of Rotating Machinery,” was used by Entergy personnel as a basis for the acceptance criteria in MP-093.04. The acceptance criteria were a minimum insulation resistance of 5.2 Megohms or polarization index less than 2.0. IEEE 43-2000 also notes that a sharp decline in the insulation resistance or polarization index from the previous reading may indicate surface contamination, moisture, or severe insulation damage, such as cracks. IEEE 43-2000 further indicates that a limitation of the insulation resistance test is that insulation resistance of a winding is not directly related to its dielectric strength and unless the defect is concentrated, it is not possible to specify the value of insulation resistance at which the insulation system of a winding will fail.

The inspectors concluded that with the significant drop in the minimum insulation resistance to 499 Megohm on June 28, 2005, followed by the significant drop in the polarization index to 1.00 on September 18, 2007, there was reasonable evidence that a condition adverse to quality existed and was not entered by Entergy personnel into the corrective action program. The Entergy EDG system monitoring program called for corrective actions involving increased monitoring or possibly rewinding the rotor and those actions were not completed.

Following May 27, 2009, Entergy’s corrective actions included rewinding the affected pole of the ‘C’ EDG rotor and initiating CR-JAF-2009-01847 in order to determine the root causes and recommend further corrective actions.

<u>Analysis:</u> There was a self-revealing performance deficiency in that Entergy personnel did not promptly identify and correct a condition adverse to quality associated with the ‘C’ EDG rotor. This finding is greater than minor because it is associated with the equipment

performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, Entergy personnel did not identify an adverse performance trend which resulted in an unplanned extension of the maintenance period for the ‘C’ EDG, extending the unavailable period from May 28 through June 11, 2009.

The inspectors evaluated the significance of this finding using IMC 0609.04, “Phase 1 – Initial Screening and Characterization of Findings,” and determined it to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not represent a loss of safety function, and did not screen as potentially risk significant due to external initiating events. The inspectors concluded the ‘C’ EDG continued to meet its safety function because the field winding degradation was not sufficient to render the ‘C’ EDG inoperable based on vendor analysis and successful monthly surveillance tests results.

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy personnel did not implement a corrective action program with a low threshold for identifying issues in that the adverse trend in the ‘C’ EDG rotor insulation was not identified. (P.1(a))

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requires, in part, that measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected. Contrary to the above, between September 18, 2007 and May 27, 2009, Entergy personnel did not implement measures to promptly identify and correct a condition adverse to quality associated with the ‘C’ EDG. Entergy personnel took corrective actions to rewind the affected pole of the ‘C’ EDG rotor. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (**NCV 05000333/2009003-04: Failure to Identify an Adverse EDG Rotor Insulation Performance Trend)**

Cornerstones: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 – 1 sample)

a. Inspection Scope

The inspectors observed simulator training activities associated with licensed operator requalification training on April 7, 2009. The inspectors reviewed emergency classification declarations and notifications to ensure they were properly completed. The inspectors evaluated the drill for conformance with the requirements of 10 CFR 50, Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities.” The inspectors observed Entergy staff’s critique and compared their self-identified issues with observations from the inspectors’ review to ensure that performance issues were properly identified. This evaluation represented one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupation Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 – 14 samples)

a. Inspection Scope

During June 8 through June 12, 2009, the inspectors conducted the following activities to verify that Entergy staff 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 by the inspectors for conformance with the criteria contained in 10 CFR Part 20, TS, and station procedures.

1. There were no occupational exposure cornerstone performance indicator incidents during the current assessment period.

2. The inspectors walked down accessible exposure significant work areas of the plant 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 accessible exposure significant work areas of the plant 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. During 2009, there were no internal dose assessments >10 mrem committed effective dose equivalent and therefore, no assessment of internal exposure calculations was performed.

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

6. The inspectors reviewed radiation protection (RP) program self-assessments and audits during 2009 to determine if identified problems were entered into the corrective action program for resolution.

7. The inspectors reviewed ten condition reports associated with the RP access control and ALARA areas, between January 2008 and June 2009, to determine if the follow-up activities by Entergy staff were being conducted in an effective and timely manner commensurate with their safety significance.

8. Based on the condition reports reviewed, the inspectors screened repetitive deficiencies to determine if Entergy staff's self-assessment activities were identifying

and addressing these deficiencies.

9. There were no Occupational Exposure performance indicator incidents reported during the current assessment period to evaluate utilizing the SDP.

10. Changes to the high radiation area and very high radiation area procedures since the last inspection in this area were reviewed by the inspectors and discussed with the RP manager.

11. Controls associated with potential changing plant conditions to anticipate timely posting and controls of radiation hazards was discussed by the inspectors with a RP supervisor.

12. The inspectors verified that accessible locked high radiation area entrances in the plant were locked through challenging the locks or doors. The inspectors also reviewed locked and very high radiation area key inventories and controls.

13. The inspectors reviewed radiological condition reports to evaluate if the incidents were caused by radiation worker errors and determine if there were any trends or patterns and if the licensee's corrective actions were adequately addressing these trends.

14. The inspectors reviewed radiological condition reports to evaluate if the incidents were caused by RP technician errors and determine if there were any trends or patterns and if the station's corrective actions were adequately addressing these trends.

This inspection constituted 14 access control to radiologically significant areas samples.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 11 samples)

a. Inspection Scope

During June 8 through June 12, 2009, the inspectors conducted the following activities to verify that Entergy personnel were properly maintaining individual and collective radiation exposures ALARA. Implementation of the ALARA program was reviewed for conformance with the criteria contained in 10 CFR 20.1101(c) and Entergy's procedures.

1. The inspectors reviewed collective personnel exposure historical results and the three-year rolling average exposure for 2005-2007 was determined to be 119 person-rem.

2. Site specific source term trends in collective exposures and source-term were reviewed by the inspectors, indicating an increasing trend reflecting second quartile boiling water reactor radiation levels which corresponds to the current collective exposure second quartile ranking.

3. The inspectors reviewed site specific procedures associated with maintaining occupational exposures ALARA including processes for estimating and tracking

collective exposures.

4. The inspectors reviewed work activities from the recent fall 2008 refueling outage and the highest actual exposure significant work activities greater than 5 person-rem were selected as listed below:

• In-Service Inspection	26.841 person-rem
• Reactor disassembly/reassembly	15.620
• N-2C pipe weld overlay	12.979
• Main steam line strain gauge modification	11.321
• Reactor defuel/refuel/inspection	9.225
• Safety relief valve replacement	8.516
• Control rod drive replacement	8.310
• RP drywell support	5.961
• Leak rate testing	5.210
• Drywell fan maintenance	5.034

5. The highest exposure significant work activities listed in (4) above were selected for detailed performance review to include the associated ALARA work activity evaluations, exposure estimates and exposure mitigation requirements. The inspectors performed this review with respect to sound RP principles to achieve ALARA.

6. For the refueling outage work activities listed in (4) above, the inspectors compared the exposure results achieved with the intended dose estimates and the reasons for dose overruns were evaluated to determine any significant performance deficiencies.

7. The inspectors reviewed the assumptions and basis for the 2009 annual collective exposure estimate. The estimate included both dose rate and man-hour estimate calculations which were reviewed in accordance with applicable procedures.

8. The station's method for adjusting exposure estimates, to incorporate work overruns, and to incorporate changes in work scope or emergent work were reviewed by inspectors to ensure accurate exposure estimates provide an effective measurement standard for job performance exposure evaluations.

9. The inspectors reviewed source-term data to assess an increasing trend (approximately 33%) from May 2000 to October 2008. Interviews were conducted with the ALARA supervisor and the RP manager relative to reactor water chemistry and source-term controls being evaluated to reduce the source term and occupational exposures.

10. The ALARA program self-assessments and RP program audit were reviewed by inspectors to determine if the station's overall audit program scope and frequency met the requirements of 10 CFR 20.1101.c.

11. With respect to the condition reports reviewed, the inspectors reviewed repetitive deficiencies that were identified with respect to the station's self-assessment and audit program identification and resolution.

This inspection represented 11 ALARA planning and controls samples.

b. Findings

Introduction: A self-revealing finding of very low safety significance was identified because Entergy personnel did not adequately plan and coordinate work activities to prevent unnecessary exposure consistent with the original dose estimate as described in Radiation Work Permit No. 08-0524. Specifically, work planning and coordination issues relative to a main steam line strain gauge modification resulted in an unplanned collective exposure of 11.32 person-rem compared to a an original work estimated dose of 6.1 person-rem.

Description: Entergy Radiation Work Permit No. 08-0524 was the applicable plan for dose execution related to the main steam line strain gauge instrument modification activity. The modification project was planned by Entergy personnel two months prior to the refueling outage, outside of the normal outage planning and scheduling process. The inspectors determined the actual versus planned job site conditions for installation of the strain gauges were not adequately evaluated by Entergy personnel for interferences and the support work involving scaffolding and insulation removal were not adequately planned and coordinated to prevent additional unnecessary exposure. Specifically, the inspectors determined there was a lack of in-field walkdowns prior to the modification design that resulted in strain gauge locations that were not accessible based on actual plant conditions. The inspectors noted these as-found interferences required removal and reinstallation of several strain gauges. Also, the inspectors noted additional work interferences occurred with station personnel scaffold erection conflicting with vessel nozzle door access, safety relief valve replacement path access, and fuel movement restricting access in the drywell. The inspectors determined this resulted in removal and re-erection of scaffolding by Entergy personnel that could have been avoided. In addition, inadequate insulation work package instructions used by Entergy personnel resulted in additional drywell entries to support strain gauge installation.

The inspectors determined these examples of additional in-field high radiation work resulted in additional collective exposure that could have been avoided by Entergy personnel had sufficient work activity planning and outage coordination occurred. The inspectors determined the actual work activity exposure of 11.321 person-rem was 55% greater than the inspectors' revised exposure estimate of 7.284 person-rem (original Entergy staff exposure estimate was 6.1 person-rem). The inspectors revised exposure estimate took into account necessary work that was not included in the original estimate and a higher effective dose rate for this work activity of 8.1%.

Entergy personnel entered the issue into the corrective action program as CR-JAF-2008-3181.

Analysis: A self-revealing performance deficiency was identified because Entergy personnel did not adequately plan and prevent unnecessary exposure during planned work activities. This finding is greater than minor because it is associated with the program and process attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine nuclear reactor operation. This finding is more than minor because it involved actual collective exposure greater than 5 person-rem that was greater than 50% above the estimated or intended exposure. Additionally, this finding is similar to the greater than minor examples example provided in IMC 0612, Appendix E (Example 6i related to ALARA planning). This finding was evaluated in accordance with IMC 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process”. The inspectors determined that the finding was of very low safety significance (Green) because it involved an ALARA planning issue and the 3-year rolling average collective dose history was less than 240 person-rem (119 person-rem average annual exposure for 2005-2007).

This finding has a cross-cutting aspect in the area of human performance because Entergy personnel’s planned work activities did not adequately incorporate the work site interferences or outage work coordination in the work control planning process. (H.3(a))

Enforcement: Enforcement action does not apply because the performance deficiency did not involve a violation of a regulatory requirement. Because this finding does not involve a violation of regulatory requirements and has very low safety significance, it is identified as **FIN 05000333/2009003-05: Inadequate Work Planning for Strain Gauge Resulted in Unplanned Exposure.**

Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (71122.01 – 3 samples)

a. Inspection Scope

During the period April 6 through April 10, 2009, the inspectors conducted the following activities to verify that Entergy personnel were properly maintaining the gaseous and liquid processing systems to ensure that radiological releases were properly mitigated, monitored, and evaluated with respect to public exposure. Implementation of these controls was reviewed for conformance with the criteria contained in 10 CFR Parts 20 and 50, TS, the Off-site Dose Calculation Manual (ODCM), and Entergy’s procedures.

The inspectors reviewed the 2007 (and data for the 2008) Annual Radiological Effluent Release Reports to verify that the effluent programs were implemented as required by the ODCM. As part of this review, changes made to the ODCM, including technical justifications, were evaluated to determine if the changes affected Entergy staff’s ability to maintain effluent doses ALARA. Applicable sections of the UFSAR were reviewed that describe the gaseous radioactive waste system and station ventilation systems. The inspectors reviewed correlations between the effluent release reports and the environmental monitoring results.

The inspectors walked down the major components of the gaseous and liquid effluent monitoring systems, with a cognizant engineer, to verify that the system configuration complied with the UFSAR description, to evaluate equipment material condition and availability; and to observe sampling collection, laboratory sample preparation, and analysis techniques.

The inspectors reviewed the relevant effluent monitoring procedures and observed station personnel collect particulate / iodine samples and noble gas grab samples from a sampling of effluent radiation monitors.

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

- Liquid radwaste effluent (17RM-350);
- SW effluent (17RM-351);
- Reactor building (RB) closed loop cooling monitor (17RM-352);
- Steam jet air ejector (17RM-150A/B);
- RB exhaust (17RM-452A/B);
- Refueling floor exhaust (17RM-456A/B);
- TB exhaust (17RM-431 and 432);'
- Radwaste building exhaust (17RM-458A/B);
- Control room ventilation (17RM-459); and
- Plant stack (17RM-50A/B).

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

- Standby gas treatment system;
- Control room exhaust ventilation air supply;
- Technical support center ventilation air supply system; and
- Off-gas filtration system.

The inspectors reviewed select pre- and post-discharge permits for adequacy, including release batch number 08-76 (B Waster Storage Tank). The inspectors observed Entergy personnel evaluate sample data, calculate discharge concentrations, and determine the radiation monitor alarm set points. There were no abnormal discharges during this inspection period.

The inspectors reviewed monthly dose projections for liquid and gaseous effluents performed since the last inspection in this area to verify that the effluent was processed and released in accordance with ODCM requirements. The inspectors confirmed that compensatory sampling was performed when installed monitors were out of service. The inspectors confirmed that no ODCM performance indicator criteria were exceeded for this time period.

The inspectors reviewed the calibration records for the currently in-use high purity germanium gamma spectrometers and liquid beta scintillation counters to determine if the required lower limits of detection were achievable and that the instruments were properly maintained. Selected counting equipment quality control charts were reviewed that documented continued operability of this equipment. Review included verification of National Institute of Standards and Technology traceability of sources.

The inspectors reviewed implementation of the measurement laboratory quality control program including effluent intra- and inter-laboratory comparisons.

The inspectors reviewed the validation and verification results for the radiological effluent dose calculation software to ensure that the software currently in use provides accurate dose projections.

The inspectors reviewed 19 condition reports relative to FitzPatrick's Effluents Program between June 2007 and April 2009 to evaluate the station's threshold for identifying, evaluating, and resolving problems in implementing the ODCM. The condition reports were also reviewed to determine if identified problems accurately characterized the causes and corrective actions were assigned to each, commensurate with their safety significance. The inspectors assessed repetitive deficiencies to ensure the staff's self-assessment activities were identifying and addressing these deficiencies.

This inspection represented three radioactive gaseous and liquid effluent treatment and monitoring systems samples.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (REMP) And Radioactive Material Control Program (71122.03 – 10 samples)

a. Inspection Scope

1. The inspectors reviewed the 2007 and 2008 Annual Radiological Environmental Operating Reports and Entergy's assessment results to verify that the REMP was implemented as required by TS and the ODCM. The inspectors' review included changes to the ODCM with respect to environmental monitoring commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, inter-laboratory comparison program, and analysis of data. The inspectors also reviewed the ODCM to identify environmental monitoring stations. In addition, the inspectors reviewed the following: Entergy staff's self-assessments and audits, event reports, inter-laboratory comparison program results, the UFSAR for information regarding the environmental monitoring program and meteorological monitoring instrumentation, and the scope of the audit program to verify that it met the requirements of 10 CFR 20.1101.

2. The inspectors walked down a sampling of air particulate and iodine sampling stations (12); drainage outfalls; water treatment stations; and thermo luminescent dosimeter (TLD) monitoring locations (25) to determine if they were located as described in the ODCM and the equipment material condition was acceptable.

3. The inspectors observed the collection and preparation of a variety of environmental samples including milk and verified that environmental sampling was representative of the release pathways as specified in the ODCM and that sampling techniques were in accordance with procedures.

4. Based on direct observation and review of records, the inspectors reviewed whether meteorological instruments were operable, calibrated, and maintained in accordance

with guidance contained in the UFSAR, NRC Safety Guide 23, and Entergy's procedures. The inspectors verified that the meteorological data readout and recording instruments in the control room and at the tower were operable.

5. The inspectors reviewed events documented in the Annual Radiological Environmental Monitoring Report which involved a missed sample, inoperable sampler, lost TLD, or anomalous measurement for the causes and corrective actions. The inspectors conducted a review of the staff's assessment of positive sample results.

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

7. The inspectors reviewed the calibration and maintenance records for environmental station equipment. The inspectors reviewed the following: the results of the station's inter-laboratory comparison program to verify the adequacy of environmental sample analyses; quality control evaluation of the inter-laboratory comparison program and the corrective actions for deficiencies; Entergy staff's determination of 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 inspectors reviewed whether the appropriate detection sensitivities with respect to TS/ODCM were utilized for counting samples and reviewed the results of the quality control program including the inter-laboratory comparison program to verify the adequacy of the program.

8. The inspectors observed the radioactive material survey and release locations and inspected the methods used for control, survey, and release to include observing the performance of personnel surveying and releasing material for unrestricted use and verifying the work was performed in accordance with plant procedures.

9. The inspectors verified that the radiation monitoring instrumentation used for the release of material from the radiological controlled area was appropriate for the radiation types present and was calibrated with appropriate radiation sources. The inspectors 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. Calibration records for select instruments were reviewed: (10) Ludlum-177, (2) SAC-4, (9) Miniscaler, (3) SAM, (7) PM-7, and (7) IPM.

10. The inspectors reviewed Entergy staff's audits and self-assessments related to the REMP since the last inspection to determine if identified problems were entered into the corrective action program, as appropriate. Selected corrective action reports 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 also assessed by the inspectors to ensure that self-assessment activities were identifying and addressing these deficiencies.

This inspection represented ten REMP and radioactive material control program samples.

b. Findings

No findings of significance were identified.

4. **OTHER ACTIVITIES** (OA)

4OA2 Identification and Resolution of Problems (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,” to 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. The inspectors assessed Entergy personnel’s threshold for problem identification, the adequacy of the cause analyses, and extent of condition review, operability determinations, and the timeliness of the specified corrective actions. The CRs reviewed are listed in the Attachment.

b. Assessment and Observations

No findings of significance were identified. The inspectors determined that Entergy staff identified equipment, human performance and program issues at an appropriate threshold and entered them into the corrective action program.

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

a. Inspection Scope

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 January 2009 through June 2009, although some examples expanded beyond those dates when the scope of the trend warranted. The inspectors compared and contrasted their results with the results documented in the last NRC integrated quarterly assessment report for FitzPatrick. 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 50, Appendix B. The documents reviewed are listed in the Attachment.

b. Assessment and Observations

No findings of significance were identified. The inspectors determined that Entergy personnel identified equipment, human performance, and program issues at an appropriate threshold and entered them into the corrective action program.

Entergy's Quality Assurance organization identified some examples of engineering programs not being implemented in accordance with code, procedural, or industry guidance. The programs included the inservice testing program, check valve program, preventive maintenance program for large motors, and air and motor operated valve trending two-year reviews. Entergy staff initiated CR-JAF-2009-01109, classified at the highest 'A' level, in order to correct the deficiencies.

Consistent with these results, the inspectors documented two self-revealing findings in this inspection report, regarding the HPCI turbine stop valve degradation and the 'C' EDG rotor winding degradation. These issues, in part, involved instances where station engineering programs did not appropriately identify adverse performance trends in accordance with station procedures.

.3 Annual Sample: Review of Repeat Loss of Shutdown Cooling Events during the FitzPatrick Refueling Outage (71152 – 1 sample)

a. Inspection Scope

The inspectors reviewed Entergy personnel's evaluation and corrective actions associated with two loss of shutdown cooling (SDC) events during the fall refueling outage. On September 16, 2008, while applying a tag out on the 'B' channel of the reactor protection system (RPS), an invalid primary containment isolation system (PCIS) initiation signal was generated and resulted in a loss of SDC for a period of approximately 53 minutes. This event was documented in NRC inspection report 05000333/2008004 and was determined to be a finding of very low safety significance related to managing risk during outage conditions.

On October 7, 2008, FitzPatrick experienced a loss of the 10600 vital bus, during a test of the trip and lock out relay associated with the 71-10402. This resulted in a PCIS initiation and the loss of power to the 'B' and 'D' RHR pumps which resulted in a loss of shutdown cooling for approximately 33 minutes. This event was documented in NRC inspection report 05000333/2008005 and was determined to be a finding of very low safety significance related to managing risk during outage conditions.

The inspectors assessed the adequacy of the information Entergy personnel used to identify and evaluate each event, the adequacy of the extent-of-condition reviews, and the appropriateness of the prioritization and timeliness of corrective actions associated with the loss of shutdown cooling events. The inspectors' review focused on determining whether Entergy personnel were completing corrective actions appropriate to address the deficiencies that resulted in the plant loss of shutdown cooling events. The inspectors reviewed Entergy staff's common cause analysis of human performance errors and events during the refueling outage, and reviewed an apparent cause evaluation related to work being performed on protected equipment during the refueling outage. The inspectors reviewed relevant operating procedures, abnormal operating procedures, and relevant work orders related to these events. Additionally, the inspectors interviewed cognizant plant personnel regarding each event. Specific documents reviewed are listed in the attachment to this report.

b. Findings and Observations

No findings of significance were identified.

The inspectors reviewed the two root cause analyses (RCAs) performed in response to the loss of SDC events and concluded that the RCAs appeared to have effectively identified several key process/programmatic and human performance issues which contributed to these events. Corrective actions were developed to address these issues. The inspectors determined a majority of the corrective actions were being implemented at FitzPatrick at the time of the inspection and those actions should be effective because the actions appeared to address the causes. For example, station personnel have implemented revisions to the work planning process, the shutdown risk assessment process, the protected equipment program, and the work authorization process. In addition, work planning tools were being implemented by Entergy personnel to identify potential work conflicts and unplanned system responses.

.4 Annual Sample: Apparent Cause Evaluation of Failure of Level 1 Acceptance Criteria of Two Remote Shutdown Safety/Relief Valve Circuits. (71152 - 1 sample)

a. Inspection Scope

The inspectors reviewed Entergy staff's evaluations and corrective actions associated with failure to meet level 1 acceptance criteria for two remote shutdown safety relief valves (SRVs) as documented in CR-JAF-2008-02865. The condition occurred during performance testing of the remote shutdown circuits for 02RV-71H, main steam line 'D' automatic depressurization system SRV, and 02RV-71J, main steam line 'D' manual SRV. During performance of MST-029.05, "SRV Remote Actuation Maintenance Testing," Revision 3, which demonstrates the operability of the remote shutdown actuation circuits for the SRVs, the measured resistances for 02RV-71H and 02RV-71J were 100 Megohms and 65 Megohms, respectively. The level 1 acceptance criteria of 140-500 Ohms were exceeded for 02RV-71H and 02RV-71J. Entergy operators declared the remote shutdown circuits for 02RV-71H and 02RV-71J inoperable and entered the appropriate TS LCO condition. The inspectors reviewed the apparent cause analysis and corrective actions to ensure that appropriate evaluations were performed and corrective actions were specified and prioritized. Documents reviewed during the inspection are listed in the Attachment.

b. Findings & Observations

No findings of significance were identified.

Entergy personnel determined that a high resistive film buildup in the NAMCO connector pins used in the SRV actuator circuits was the apparent cause and that the testing methodology to conduct the resistance measurement tests was inadequate. Entergy personnel identified the resistance test methodology used a standard digital multi-meter which used a 9 Vdc battery as the power source for resistance measurements. The normal SRV circuit voltage is 125 Vdc. Entergy personnel concluded that despite the test failure the normal SRV circuit voltage of 125 Vdc would burn through the resistive film build-up and actuate the SRV when required and therefore proposed an alternate test methodology to use a 100 Vdc megger to test the circuits as a corrective action in the event of a failed test using a standard 9 Vdc digital multi-meter.

The inspectors reviewed Entergy staff's apparent cause evaluation and determined that the proposed corrective action for testing the SRV circuits with 100 Vdc megger was adequate. However, the inspectors determined that the apparent cause analysis did not evaluate or document its review of an abnormal condition during the 2006 refueling outage regarding NAMCO connector pins. Specifically, the inspectors noted that NAMCO connectors associated with SRV actuation circuits were tested as a part of 2006 post-outage work activity related to pilot solenoid replacements and lubricating oil was observed on NAMCO connector pins. This abnormal condition was documented in Entergy's corrective action program as CR-JAF-2006-04678. The inspectors concluded that it would have been appropriate for Entergy's apparent cause analysis team to consider this abnormal condition as a possible contributor to the high resistance on the NAMCO connector pins with appropriate actions to address the issue. Although the lubricating oil that was observed by Entergy personnel in 2006 was not considered or documented by the apparent cause team as a possible contributor to the high resistance condition in 2008, the inspectors determined that corrective actions to address the high resistance on the NAMCO connector pins were appropriately implemented by Entergy staff.

4OA3 <u>Follow-up of Events and Notices of Enforcement Discretion</u> (71153 – 5 samples)

.1 <u>(Closed) LER 05000333/2006002-01</u>, High Pressure Coolant Injection (HPCI) System Declared Inoperable Due to Turbine Speed Oscillations, and Changing from MODE 2 to MODE 1 with HPCI System Inoperable

On November 4, 2006, with the plant operating in Mode 1, Entergy personnel identified that the HPCI system was inoperable due to turbine speed oscillations. The condition was discovered during post-maintenance testing following the 2006 refueling outage, and was caused by connecting two turbine hydraulic actuator oil lines to the incorrect oil ports. The enforcement aspects of this violation of maintenance procedures were documented as a licensee-identified violation in section 4OA7 of NRC Inspection Report 05000333/2006005. Entergy personnel entered the event into its corrective action program as CR-JAF-2006-04754.

The inspectors identified that the original submitted version of licensee event report (LER), LER 05000333/2006002 did not address an additional basis for reporting the condition related to 10 CFR 50.73(a)(2)(i)(B), "Operation or Condition Prohibited by Technical Specifications." Entergy personnel previously initiated CR-JAF-2006-04738 and documented that prior to low pressure testing of the HPCI system, surveillance procedure ST-4J, "HPCI Turbine Slow Roll," Revision 2, was aborted prior to completion because the test speed potentiometer was fully turned clockwise and the required minimum speed of the HPCI turbine could not be obtained. Entergy personnel attributed the malfunction of

ST-4J to faulty test equipment without validation and continued with the HPCI startup process. The inspectors determined this decision contributed to Entergy personnel making a change from Mode 2 to Mode 1 while HPCI was inoperable, which was prohibited by TS and also reportable.

The inspectors concluded the revised reportability aspects of the originally submitted LER 05000333/2006002 did not impact the regulatory process since no regulatory decisions would have differed. The inspectors determined this issue constituted a violation of minor significance that is not subject to enforcement action in accordance with the NRC's Enforcement Policy. Entergy personnel initiated CR-JAF-2009-01076 to address the issue. This LER is closed.

.2 (Closed) LER 05000333/2009002-00, Subsystem Inoperable in Excess of Technical Specifications Allowed Out-of-Service-Time

On January 28, 2009, Entergy personnel identified that the Technical Requirements Manual section B.3.7.A, "125 Vdc Battery Room Ventilation System," was developed assuming that each air handling unit (AHU) was capable of 100% redundant capacity when in fact each AHU has capacity sufficient for its respective battery and charger rooms only, without redundancy. Entergy personnel determined that if one AHU is not functional there may not be adequate cooling to maintain the operability of the associated battery charger room without realigning the ventilation system. Without adequate cooling to the battery charger room, the associated 125 Vdc subsystem should be declared inoperable according to TS 3.8.4, "DC Sources – Operating," which requires that the subsystem be restored to an operable state within 8 hours, or if not restored, the plant be in Mode 3 within 12 hours and Mode 4 in 36 hours.

In August of 1988, while operating under Custom Technical Specifications, Technical Specification Interpretation 06 was developed to provide guidance to the operations department on the operability of the battery room ventilation system. This guidance was developed using the Stone and Webster conceptual design notes dated November 23, 1970, which did not reflect the as-built configuration of the plant. The design notes described the ventilation system as having two 100% capacity redundant AHUs. However, due to interferences associated with the larger AHUs, the facility was constructed using AHUs with sufficient capacity for a single battery room only, such that with an AHU out of service the respective train of battery room ventilation should have been considered inoperable. This error was not identified during a March of 1999 revision to the Technical Specification Interpretation 06, nor during the July of 2001 conversion from Custom Technical Specifications to Improved Technical Specifications.

Entergy personnel reviewed the period starting January 2006 through February 2009, and identified two periods when a battery room ventilation system AHU was tagged out for greater than the allowed out-of-service time. The first occurrence was when 72AHU-30A was tagged out for approximately 34 hours during April 2006, and the second occurrence was when 72AHU-30B was tagged out for 77 hours during September 2008. Each of these occurrences constituted a past operation or condition which was prohibited by the plant's TS, thus requiring the LER according to 10 CFR 50.73(a)(2)(i)(B).

This condition at FitzPatrick was mitigated because during the periods of non-compliance the room temperatures were monitored and there was no adverse change in temperature.

In addition, the plant had in place specific procedures for supplying temporary cooling to the battery and battery charger rooms with operations department personnel trained to execute those procedures. This licensee-identified finding involved a violation of TS 3.8.4, "DC Sources – Operating." The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

.3 (Closed) LER 05000333/2009004-00, Loss of Control Room Envelope Boundary

On January 31, 2009, Entergy personnel identified door 70DOR-A-300-5, a control room envelope (CRE) boundary door between the control room chiller room and the control room HVAC room, to be unlatched and initiated CR-JAF-2009-00387. On March 19, 23, and 24, 2009, the inspectors identified the door to be unlatched and slightly ajar. The inspectors also identified that the door handle latch mechanism appeared degraded and that changes in differential pressure across the door due to the opening and closing of adjacent doors caused the latch to spontaneously unlatch.

The CRE supports the ability of the control room ventilation system to maintain control room habitability following an accident. Entergy personnel performed an engineering evaluation that concluded that the CRE cannot be maintained with the door unlatched. TS 3.7.3, "Control Room Emergency Ventilation Air Supply (CREVAS) System," condition B requires the plant to immediately initiate actions to implement mitigating actions with the CRE inoperable, and these actions were not initiated until March 24, 2009. Entergy personnel determined the event was reportable under 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(D).

The enforcement aspects of this violation were documented in section 1R22 of NRC Inspection Report 05000333/2009002. Entergy personnel entered the event into its corrective action program as CR-JAF-2009-01021 and CR-JAF-2009-01070. The inspectors reviewed this LER and no new findings were identified. This LER is closed.

.4 (Closed) LER 05000333/2009005-00, Safety Relief Valve Setpoints Outside of Allowable Tolerances

On April 20, 2009, Entergy personnel identified that it had operated during the previous operating cycle (Cycle 18) with less than nine operable safety relief valves (SRVs) as required by TS 3.4.3, "Safety/Relief Valves." TS 3.4.3 requires nine operable SRVs when in Modes 1, 2 or 3. Entergy personnel had removed SRVs during the previous refueling outage (RFO-18) and identified five SRVs had as-found lift setpoints outside the high tolerance limit allowed by TS 3.4.3.1. Entergy staff's root cause analysis determined that the most probable cause of the out of tolerance SRV setpoints for four of the malfunctions was corrosion binding between the SRV pilot disc and seat which is an industry generic problem. The root cause analysis determined that the most probable cause of the out of tolerance SRV setpoint for the fifth malfunction was significant pilot valve seat leakage which would have required additional steam pressure to overcome the leakage in order to lift this SRV. Corrective actions documented in CR-JAF-2007-02108 and CR-JAF-2007-02937 included:

- Installed enhanced insulation on pilot assemblies;
- Redirected ventilation to limit cooling effect; and
- Replaced pilot assemblies with recently refurbished, tested, and certified assemblies.

The condition at FitzPatrick was mitigated by two considerations: (1) while the SRVs did not lift within the TS prescribed high limit, they actuated at higher pressures; and (2) a diverse SRV electronic pressure switch actuation system was available which would have actuated the valves. 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.

.5 (Closed) LER 05000333/2009006-00, Inoperable High Pressure Coolant Injection System

On April 22, 2009, Entergy personnel performed a quarterly surveillance test on the HPCI system. During the initial HPCI startup sequence, a HPCI high steam flow isolation occurred and Entergy personnel declared HPCI inoperable. Entergy staff's analysis concluded that the 23HOV-1 balance chamber pressure was adjusted too low which caused an erratic fast opening of 23HOV-1, resulting in a high steam flow condition that caused the HPCI steam line isolation.

Entergy personnel reported the condition within 8 hours according to 10 CFR 50.72(b)(3)(v)(D) since the invalid HPCI steam line isolation temporarily rendered the HPCI system inoperable. Entergy personnel also determined the condition was reportable under 10 CFR 50.73(a)(2)(v)(D).

The inspectors reviewed this LER and a finding is documented in section 1R22 of this report. This LER is closed.

4OA5 Other Activities

.1 Independent Spent Fuel Storage Installation (60855)

An independent spent fuel storage installation (ISFSI) inspection was conducted from April 6 through April 10, 2009, utilizing inspection procedure 60855 to review the ongoing maintenance and surveillance activities for onsite dry storage of spent fuel. The ISFSI licensing basis documents and implementing procedures were reviewed as the inspection standards for the inspection. The inspection consisted of the following: observation of the condition of the nine Holtec Hi-Storm casks currently storing spent fuel inside the restricted area at Fitzpatrick; independent radiation survey of the nine spent fuel storage casks; observation of obtaining the daily air vent outlet temperature readings; verification of placement of perimeter area dosimeters; and review of surveillance records, including the annual SNM inventory inspection, monthly air vent inspections, and daily air vent outlet temperature readings.

b. Findings

No findings of significance were identified.

.2 TI 2515/173, Review of the Implementation of the Industry Ground Water Protection Voluntary Initiative

a. Inspection Scope

On May 4 through 8, 2009, an NRC assessment was performed of Entergy's implementation of the Nuclear Energy Institute – Ground Water Protection Initiative (dated August 2007, ML072610036).

Entergy personnel have identified systems, structures, and components that contain licensed radioactive material to determine potential leak or spill mechanisms. Entergy personnel have completed an initial site characterization of geology and hydrology to determine the predominant ground water gradients and potential pathways for ground water migration from on-site locations to off-site locations. An on-site ground water monitoring program has been implemented by the station to monitor for potential licensed radioactive leakage into groundwater. The ground water monitoring results are being reported in the annual effluent and/or environmental monitoring report.

Entergy personnel have identified the appropriate local and state officials and have conducted initial briefings on Entergy's ground water protection initiative.

b. Findings and Observations

No findings of significance were identified.

.3 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that these activities were consistent with Entergy security procedures and applicable regulatory requirements. Although these observations did not constitute additional inspection samples, they were considered an integral part of the normal, resident inspectors' plant status reviews during implementation of the baseline inspection program.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. P. Dietrich and other members of Entergy's management at the conclusion of the inspection on July 9, 2009. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified by Entergy personnel.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by Entergy personnel and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

- TS 3.8.4 requires that with one 125 Vdc electrical power subsystem inoperable for reasons other than an inoperable battery charger, the subsystem be restored to an operable state within 8 hours, or if not restored, the plant be in Mode 3 within 12 hours and Mode 4 in 36 hours. Contrary to this, on January 28, 2009, Entergy personnel identified it had remained in Mode 1 with an inoperable 125 Vdc electrical power subsystem for greater than the allowed out-of-service time on two occasions, April 5, 2006 and September 17, 2008. Entergy personnel documented this condition in CR-JAF-2009-00358. The inspectors evaluated this finding using IMC 0609.04, “Phase 1 – Initial Screening and Characterization of Findings,” and determined that the condition was of very low safety significance (Green) because it did not result in the loss of the 125 Vdc electrical power subsystems’ ability to provide emergency power given actual room temperatures in April 2006 and September 2008 and the plant’s ability to supply temporary cooling.

- TS 3.4.3 requires that at least nine SRVs shall be operable in operating modes 1, 2, and 3. Contrary to this, on April 20, 2009, Entergy personnel identified it had operated in these modes during Cycle 18 with less than nine operable SRVs per TS 3.4.3. Entergy personnel documented this condition in CR-JAF-2009-01429. The inspectors evaluated this finding using IMC 0609.04, “Phase 1 – Initial Screening and Characterization of Findings,” and determined that the condition was of very low safety significance (Green) because it did not result in the loss of the overpressure relief safety function of at least nine of the eleven SRVs.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

P. Dietrich, Site Vice President
C. Adner, Manager Operations
J. Barnes, Manager, Training and Development
P. Cullinan, Manager, Emergency Preparedness
B. Finn, Director Nuclear Safety Assurance
D. Johnson, Manager, System Engineering
J. LaPlante, Manager, Security
K. Mulligan, General Manager, Plant Operations
J. Pechacek, Licensing Manager
J. Solowski, Radiation Protection
M. Woodby, Director Engineering

LIST OF ITEMS OPEN, CLOSED, AND DISCUSSED

Opened and Closed

05000333/2009003-01	NCV	High Energy Line Break Door Missing Lower Support (Section 1R04)
05000333/2009003-02	NCV	Failure to Recognize an Adverse HPCI Performance Trend (Section 1R15)
05000333/2009003-03	NCV	Balance Chamber Pressure for the HPCI Turbine Stop Valve Was Not Set at a Value to Ensure HPCI Operation (Section 1R22)
05000333/2009003-04	NCV	Failure Regarding an Adverse EDG Rotor Insulation Performance Trend (Section 1R22)
05000333/2009003-05	FIN	Inadequate Work Planning for Strain Gauge Resulted in Unplanned Exposure (Section 2OS2)

Closed

05000333/2006002-01	LER	High Pressure Coolant Injection (HPCI) System Declared Inoperable Due to Turbine Speed Oscillations, and Changing from

		MODE 2 to MODE 1 with HPCI System Inoperable (Section 4OA3)
05000333/2009002-00	LER	Subsystem Inoperable in Excess of Technical Specifications Allowed Out-of-Service-Time (Section 4OA3)
05000333/2009004-00	LER	Loss of Control Room Envelope Boundary (Section 4OA3)
05000333/2009005-00	LER	Safety Relief Valve Setpoints Outside of Allowable Tolerances (Section 4OA3)
05000333/2009006-00	LER	Inoperable High Pressure Coolant Injection System (Section 4OA3)

<u>Discussed</u>

None

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WO: 00151622, 00147788
AOP-72, "115kV Grid Loss, Instability, or Degradation," Revision 2
AP-12.04, "Seasonal Weather Preparations," Revision 17
OP-51A, "RB Ventilation and Cooling System," Revision 47
OP-55B, "Control Room Ventilation and Cooling," Revision 30
OP-59A, "Battery Room Ventilation," Revision 6

Section 1RO4: Equipment Alignment
AOP-39, "Loss of Coolant," Revision 17
AOP-40, "Main Steam Line Break," Revision 10
AOP-44, "Dropped Fuel Assembly," Revision 7
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DBD-070, "Design Basis Document for the Control Room Relay Room Ventilation and Cooling Systems," Revision 13
FB-35E, "Flow Diagram Control Room Area Service & Chilled Water System 70," Revision 34
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JAF-CALC-RAD-00042, "Control Room Radiological Habitability Under Power Uprate Conditions and CREVASS Reconfiguration," Revision 3
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EN-CY-102, "Laboratory Analysis Quality Controls", Revision 3
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EN-RW-104, "Scaling Factors", Revision 4
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IMP-01-107.7, "Stack Exhaust Flow Indication Calibration", Revision 4
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IMP-69.2, "Radwaste Building Vent Exhaust Flow Indication Calibration", Revision 2
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S-ENVSP-3.2, “Garden/Irrigation Census & Food Product Sample Collection,” Revision 2
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LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ADS	automatic depressurization system
AHU	air handling unit
ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
CDF	core damage frequency
CFR	Code of Federal Regulations
CR	condition report
CRE	control room envelope
CREVAS	control room emergency ventilation air supply
DBD	design basis document
ECCS	emergency core cooling system
EDG	emergency diesel generator
Entergy	Entergy Nuclear Northeast
HELB	high energy line break
HPCI	high pressure coolant injection
IMC	inspection manual chapter
ISFSI	independent spent fuel storage installation
IST	in-service test
kV	kilovolt
LER	licensee event report
LERF	large early release frequency
LOCA	loss of coolant accident
NCV	non-cited violation
NMSS	Nuclear Material Safety and Safeguards
NRC	Nuclear Regulatory Commission
OA	other activities
ODCM	off-site dose calculation manual
PARS	Publicly Available Record
PCIS	primary containment isolation system
psig	pounds per square inch gauge
RB	reactor building
RCA	root cause analysis
RCIC	reactor core isolation cooling
REMP	radiological environmental monitoring program
RHR	residual heat removal
RP	radiation protection
SDC	shutdown cooling
SDP	significance determination process
SPAR	standardized plant analysis risk
SRA	senior reactor analysis
SRV	safety relief valve
SSC	structures, systems, or components
ST	surveillance test
SW	service water
TLD	thermoluminescent dosimeter

TS	technical specification
UFSAR	updated final safety analysis report
WO	work order