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

January 29, 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 05000333/2008005

Dear Mr. Dietrich:

On December 31, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your James A. FitzPatrick Nuclear Power Plant. The enclosed inspection report documents the inspection results, which were discussed on January 12, 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, two findings of very low safety significance (Green) were identified. These findings were determined to be violations of NRC requirements. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. However, because of the very low safety significance, and because the violations were entered into your corrective action program, the NRC is treating these violations as a non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a written response within 30 days of the date of this inspection report with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Senior Resident Inspector at the James A. FitzPatrick Nuclear Power Plant.

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

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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 05000333/2008005
w/Attachment: Supplemental Information

cc w/encl:
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Senior Manager, Nuclear Safety and Licensing, Entergy Nuclear Operations
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J. Spath, Program Director, New York State Energy Research and Development Authority

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-333
License No.:	DPR-59
Report No.:	05000333/2008005
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	268 Lake Road Scriba, New York 13093
Dates:	October 1, 2008 through December 31, 2008
Inspectors:	G. Hunegs, Senior Resident Inspector S. Rutenkroger, PhD, Resident Inspector R. Fuhrmeister, Senior Project Engineer 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 05000333/2008-005; 10/01/2008 - 12/31/2008; James A. FitzPatrick Nuclear Power Plant; Equipment Alignment, and Refueling and Other Outage Activities.

The report covered a three-month period of inspection by resident inspectors and announced inspections by region based inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process (SDP).” Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. An NRC identified NCV of 10 CFR 50 Appendix B, Criterion III, “Design Control,” was identified when Entergy did not assure that appropriate quality standards were specified and included in design documents and that deviations from such standards were controlled. Specifically, Entergy did not ensure the oil tubing within the high pressure coolant injection (HPCI) system remained properly supported and routed with an appropriate slope in accordance with design. The issue was entered into Entergy’s corrective action program as CR-JAF-2008-04040. Corrective actions included establishing work order 172913 to restore the original configuration properly supporting the HPCI tubing lines.

This finding is more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, reliability was affected because the unsupported span of tubing was more susceptible to personnel damage and vibration during HPCI operation, both during surveillance testing and also if called upon to perform its safety function. In addition, the tubing was more susceptible to damage and adverse routing changes during maintenance activities. Therefore, over time, the high pressure fittings associated with the lines would be more likely to suffer failures, retain air bubbles within the lines, and/or leak during pump operation affecting the long-term reliability of the system. This was reasonably within Entergy’s ability to foresee and prevent because the governing procedures require tube routings, including support locations, be provided during installation of Class I tubing, and a support bracket was available to attach the tubing. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” and determined it to be of very low safety significance (Green) because the finding represented a design or qualification deficiency confirmed not to result in loss of operability.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the design documents, procedures, and work packages used during the maintenance activities in September and October 2008, were not sufficiently complete to ensure design standards were implemented. (H.2(c)) (Section 1R04)

Green. A self-revealing NCV of 10 CFR Part 50.65 (a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was identified when Entergy did not manage the increase in risk during the conduct of relay testing associated with emergency buses. The conduct of the relay testing resulted in an unanticipated loss of shutdown cooling (SDC) function. Entergy implemented corrective actions that included communicating the error to personnel to reinforce management expectations for control of protected equipment and providing an additional level of work authorization review.

This finding is more than minor because it is associated with the Mitigating Systems cornerstone and is related to Entergy's performance in assessing and managing risk. A risk assessment review was not conducted prior to performance of a trip and lockout relay functional test associated with emergency buses. Specifically, this finding reflects inadequate risk management that contributed to a short duration loss of shutdown decay heat removal capability resulting from the inadvertent interruption of flow through the operating train of shutdown cooling with the plant in a cold shutdown condition. This was reasonably within Entergy's ability to foresee and prevent because there were opportunities to recognize and manage the potential risk of losing shutdown cooling and to schedule the maintenance activity at a more appropriate maintenance window or take actions to prevent the loss of shutdown cooling.

In accordance with IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and Appendix G, "Shutdown Operations Significance Determination Process," the inspectors determined that this finding was of very low safety significance (Green). The basis for this determination is that in accordance with IMC 0609, Appendix G, Table 1, "Losses of Control," and Checklist 8, "BWR Cold Shutdown or Refueling Operation Time to Boil > 2 Hours: RCS Level <23 feet Above Top of Flange," this finding did not require quantification and did not constitute a significant loss of thermal margin, based upon the slow reactor coolant system heat-up rate and minimal time of interruption in shutdown cooling system operation. The problem was entered into Entergy's corrective action program as CR-JAF-2008-03805.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not plan and coordinate work activities properly to manage the operational impact of work activities. Specifically, Entergy did not recognize that the emergency bus 10600 would be de-energized as a result of the trip and lockout relay functional test. (H.3(b)) (Section 1R20)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

The James A. FitzPatrick Nuclear Power Plant (FitzPatrick) began the inspection period shutdown to conduct a refueling outage. On October 8, 2008, the reactor was started up and on October 9, 2008, the generator was returned to service. On October 12, reactor power was increased to 100 percent and the plant continued to operate at or near 100 percent reactor 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)

a. Inspection Scope

The inspectors reviewed and verified completion of the cold weather preparation checklist contained in procedure AP-12.04, “Seasonal Weather Preparations.” The inspectors reviewed the operating status of the reactor and turbine building cooling systems, reviewed the procedural limits and actions associated with cold weather, and walked down accessible areas of the reactor and turbine buildings to assess the effectiveness of the heating and ventilation systems. Walkdowns were also conducted in the emergency diesel generator (EDG), emergency service water, and screenhouse rooms. Discussions with operations and engineering personnel were conducted to ensure that they 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.

On December 15, 2008, the site experienced high wind conditions and on December 23, the site experienced environmental conditions conducive to frazil ice formation. The inspectors reviewed operating procedure (OP) -4, “Circulating Water System,” Revision 62 and Routine Test-04.05, “Ice Potential Determination,” Revision 1, and discussed scheduled work activities and plant operating parameters and conditions with operators. In addition, the inspectors toured the switchyard and walked down portions of the intake structure. This inspection represented one inspection sample for the onset of adverse weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

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

- ‘B’ reactor protection system when ‘A’ reactor protection system was on a backup power supply due to system failure;
- HPCI while the ‘A’ reactor protection system motor-generator set was out of service and work was being performed on the ‘B’ reactor feed pump electrical stop; and
- Offsite power source 115 kV line number three when offsite power source 115 kV line number four was inoperable due to system maintenance.

b. Findings

Introduction: A Green NRC identified NCV of 10 CFR 50 Appendix B, Criterion III, “Design Control,” was identified when Entergy did not assure that appropriate quality standards were specified and included in design documents and that deviations from such standards were controlled. Specifically, Entergy did not ensure the oil tubing within the HPCI system remained properly supported and routed with an appropriate slope in accordance with design.

Description: On October 20, 2008, the inspectors performed a walkdown of the HPCI system while the ‘A’ reactor protection system motor-generator set was out of service and work was being performed on the ‘B’ reactor feed pump electrical stop. During this walkdown, the inspectors identified that four 3/8” stainless steel turbine governor hydraulic actuator oil tubing lines connecting the governor (23GOV-1) and the remote servo (23HYC-1) were not supported. The total tubing length was approximately seven and a half feet with the longest seismic support span (in the horizontal) of approximately six feet. In addition, an unconnected support bracket was observed adjacent to the tubing which appeared to have been utilized in the past.

The inspectors determined that in October 2000, Entergy installed modification number JE-00-035, “HPCI Governor/Servo Tubing Changes,” revision 2, including engineering change notices 001 and 002, to implement G.E. Field Disposition Instruction number 83-88595, item 6, issued on February 25, 1976. The modification specified that “All tubing installations shall be per IS-S-01 and routing shall remain consistent with the existing routing." IS-S-01, "Tubing and Support Installation," revision 6 specified that for work on this class of tubing, "QA Category I," that Design Engineering shall develop a detailed tube routing drawing per CES-8B showing "Supports and their locations" among other items and that the work shall locate the tubing supports per the detailed tubing routing drawing.

The configuration of the unsupported tubing did not comply with the general design criteria provided in CES-8B, "JAF Tubing Design Standard," Revision 0 which specified a maximum seismic tubing span of four feet nine inches and deadweight tubing span of four feet zero inches for this application. The vendor guidance contained within G.E. Field Disposition Instruction number 83-88595, item 6, stated "Tubing must be rigidly supported at three foot intervals." The guidance also stated "Continuously slope from the servo to the actuator, use as a minimum 3/8 inch stainless steel tubing, 0.035 wall, and rigidly support at three foot intervals." Modification number JE-00-035 also specified for the lines to be continuously sloped.

The inspectors noted that subsequent to the installation in October 2000, engineering personnel evaluated the condition in January 2001, performed as work order JF-000804300, "Engineering to Assess the Existing Tubing as Routed. However, the personnel did not incorporate the results of the evaluation into the design records when it recommended taking no further action to remedy the unsupported condition. In addition, the evaluation did not adequately address the long term reliability of the system because it did not address the specific recommendation for rigid supports in the vendor guidance intended to improve the reliability and availability of the system and it incorrectly concluded that the design configuration would be maintained without the supports. Specifically, since that time the tubing has been bent and malformed resulting in a loss of continuous sloping.

The tubing was last removed and installed during refueling outage 18, having been removed on September 18, 2008 and reinstalled on October 2, 2008, in accordance with work order 51193075 and MP-023.01, "HPCI Turbine Major Inspection," Revision 14. However, these documents did not reference tubing supports nor provide guidance on maintaining tube routing configurations in accordance with design.

The inspectors noted that, although the unsupported span of tubing exceeded the design standard, there was no evidence of current leaks, strains, or compromised integrity. In addition, seismic considerations would be expected to be of decreased importance because the HPCI skid is located at a point least susceptible to such concerns. However, vibrations induced by system operation would be greater in the unsupported configuration, and the lack of supports allows for greater susceptibility to inadvertent and unrecognized personnel damage. As such, the reliability of the system would be adversely affected, over time, as the personnel damage and vibrations during system operation impact the unsupported lines.

The inspectors determined that the failure to ensure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled was a performance deficiency.

Analysis: This finding is more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, reliability was affected because the unsupported span of tubing was more susceptible to personnel damage and vibration during HPCI operation, both during surveillance testing and also if called upon to perform its safety function. In addition, the tubing was more susceptible to damage and adverse routing changes during maintenance activities. Therefore, over time, the high pressure fittings associated with the lines would be more likely to suffer failures, retain air bubbles within the lines, and/or leak during pump operation and affect the long-term reliability of the system. This was reasonably within Entergy's ability to foresee and prevent because the governing procedures called out within modification JE-00-035 require that tube routings, including support locations, be provided during installation of Class I tubing, and a support bracket was visibly available to attach the tubing. The inspectors evaluated the significance of this finding using IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and determined it to be of very low safety significance (Green) because the finding represented a design or qualification deficiency confirmed not to result in loss of operability.

The issue was entered into Entergy's corrective action program as CR-JAF-2008-04040. Corrective actions included establishing work order 172913 to restore the original configuration to properly support the HPCI tubing lines.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the design documents, procedures, and work packages used during the maintenance activities in September and October 2008, were not sufficiently complete to ensure design standards were implemented. (H.2(c))

Enforcement: 10 CFR Part 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 did not ensure appropriate quality standards were specified and controlled to ensure the HPCI turbine governor hydraulic actuator oil lines were installed and supported properly on October 2, 2008. Because the finding was of very low safety significance and Entergy entered the finding into their corrective action program as CR-JAF-2008-04040, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: (**NCV 05000333/2008005-01, Quality Standards Not Specified in Design Documents that Resulted in Unsupported HPCI Oil Tubing.)**

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

a. Inspection Scope

The inspectors performed a complete system alignment inspection of the automatic depressurization system to identify any discrepancies between the existing equipment lineup and the required lineup. During the inspection, system drawings and OPs were used to verify proper equipment alignment and operational status. The inspectors reviewed the open maintenance work orders associated with the system for any deficiencies that could affect the ability of the system to perform its function. Documentation associated with unresolved 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 condition report database to verify that equipment problems were being identified and appropriately resolved. The documents reviewed are listed in the Attachment. The inspection represented one inspection sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q – 5 samples, 71111.05A – 1 sample)

.1 Quarterly Inspection

a. Inspection Scope (5 samples)

The inspectors conducted tours 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 against the requirements of Licensee Condition 2.C.3. The documents reviewed are listed in the Attachment.

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

- Fire Area/Zone VII/CS-1, elevation 272 foot;
- Fire Area/Zone VII/RR-1, elevation 286 foot;
- Fire Area/Zone Yard, elevation 272 foot;
- Fire Area/Zone IB/SH-1, elevation 235, 255 and 260 foot; and
- Fire Area/Zone IB/SH-1, elevation 272 foot.

b. Findings

No findings of significance were identified.

.2 Annual Inspection (1 sample)

a. Inspection Scope

The inspectors observed a fire drill on November 4, 2008, including the post-drill critique, and reviewed the disposition of issues and deficiencies that were identified. The drill was observed to evaluate the capability of the fire brigade to fight fires. Specific attributes evaluated were: (1) control room response; (2) effectiveness of fire brigade leader communications, command and control, and utilization of pre-planned strategies; (3) proper wearing of turnout gear and self-contained breathing apparatus; (4) proper use and layout of fire hoses; (5) sufficient fire fighting equipment brought to the scene; (6) employment of appropriate fire fighting techniques; (7) search for victims and propagation of the fire into other plant areas; (8) smoke removal operations; and (9) proper storage of fire fighting equipment. The inspectors evaluated the fire brigade capability to meet 10 CFR Part 50, Appendix R requirements. This inspection represented one sample.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07 – 2 samples)

a. Inspection Scope

The inspectors reviewed Entergy's programs for maintenance, testing, and monitoring of risk significant heat exchangers to verify whether potential deficiencies could mask degraded performance, and to assess the capability of the heat exchangers to perform their design functions. The inspectors assessed whether the FitzPatrick program conformed to Entergy's commitments to NRC Generic Letter 89 -13, "Service Water (SW) System Problems Affecting Safety-Related Equipment." In addition, the inspectors evaluated whether any potential common cause heat sink performance problems could affect multiple heat exchangers in mitigating systems or result in an initiating event.

Based on risk significance and prior inspection history, the following heat exchangers were selected:

- 'A' residual heat removal heat exchanger (10E-2A); and
- 'B' residual heat removal heat exchanger (10E-2B).

The heat exchangers are cooled by the safety-related residual heat removal (RHR) service water systems. The systems were designed to supply cooling water from the ultimate heat sink (Lake Ontario) to various heat loads to ensure a continuous flow of cooling water to systems and components necessary for plant safety both during normal operation and under abnormal conditions. The inspectors reviewed system health reports, performance tests, design specifications and calculations, inspection test results, and chemical control methods to ensure that the selected components conformed to Entergy's commitments to Generic Letter 89 -13, "SW System Problems Affecting Safety-Related Equipment." The inspectors compared the surveillance test and inspection results to the established acceptance criteria to verify that the results were acceptable and that the heat exchangers operated in accordance with design. The documents reviewed are listed in the Attachment. These observations represented two inspection samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On November 3, 2008, the inspectors observed licensed operator simulator training to assess operator performance during several scenarios to verify that operator performance was adequate and evaluators were identifying and documenting crew performance problems. The inspectors evaluated the performance of risk significant operator actions, including the use of emergency OPs. The inspectors assessed the clarity and effectiveness of communications, the implementation of appropriate actions in response to alarms, the performance of timely control board operation and manipulation, and the oversight and direction provided by the shift manager. The inspectors also reviewed simulator fidelity to evaluate the degree of similarity to the actual control room. Licensed operator training was evaluated against the requirements of 10 CFR Part 55, "Operators' Licenses." The documents reviewed are listed in the Attachment. This observation of operator simulator training represented one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 2 samples)

a. Inspection Scope

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

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

The inspectors reviewed system health reports, maintenance backlogs, and Maintenance Rule basis documents. The inspectors evaluated the maintenance program against the requirements of 10 CFR Part 50.65. The documents reviewed are listed in the

Attachment. The following maintenance effectiveness samples were reviewed and represented two inspection samples:

- Automatic depressurization system; and
- Reactor water cleanup system.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 4 samples)

a. Inspection Scope

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

- The week of October 13, 2008, which included instrument surveillance tests, a reactor core isolation cooling surveillance test, troubleshooting and repair of the ‘A’ reactor protection system motor-generator set, and work on the ‘B’ reactor feed pump electrical stop;
- The week of October 27, 2008, which included master trip unit surveillance tests and calibrations, a ‘B’ RHR surveillance test, a ‘B’ emergency diesel generator system surveillance test, planned maintenance on offsite power 115 kV Fitz-NMP line number 4 to replace surge arrestors and calibrate relays, and trip risk due to adverse weather conditions including high winds;
- The week of November 24, 2008, which included main turbine overspeed trip device test, reactor protection system testing and manual scram testing and emergent work on the ‘B” reactor feedwater pump control system; and
- The week of December 15, 2008, which included emergent work on ‘A’ reactor protection system and placing the system on the alternate power supply and periods of high winds.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 4 samples)

a. Inspection Scope

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

- CR 2008-03907, 'E' safety relief valve leakage;
- CR LO-LAR-2008-00020 and CR 2008-03623, Gas accumulation in emergency core cooling, decay heat removal, and containment spray systems;
- CR 2008-03508, Drywell liner pitting, concrete damage and coating degradation; and
- CR-2008-04040, HPCI governor oil lines without required tubing supports.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18 – 1 sample)

a. Inspection Scope

The inspectors reviewed the following temporary plant modification to verify the design bases, licensing bases, and performance capability of the system was not degraded by the modification. The inspectors reviewed the modification against the requirements of 10 CFR 50.59. The following temporary modification was reviewed and represented one inspection sample.

- The inspectors reviewed temporary modification EC-00612, which was implemented to disable the non-safety related reactor feedwater pump high vibration trip and restore the main turbine high vibration trip. Both the reactor feedwater pump and main turbine high vibration trips share portions of the same circuit. The modification bypassed the reactor feedwater pump vibration trips which have degraded vibration probes through lifting leads which allowed the main turbine high vibration trip to remain in service. The inspectors also verified that the installation was consistent with the modification documentation; that the drawings and procedure were updated as applicable; and that the post-installation testing was adequate.

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 against the requirements of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control." The documents reviewed are listed in the Attachment. The following post-maintenance test activities were reviewed and represented six inspection samples:

- Work order 00114044, maximum extended load line limit analysis testing for maximum extended operating domain;
- Work order 51102328, involving core spray 'A' preventive maintenance and motor refurbishment;
- Work order 51099652, involving high pressure core injection preventive maintenance and rotor replacement;
- Work order 51104636, involving torus exhaust inner isolation valve 27AOV-117 maintenance;
- Work order 51104630, involving torus exhaust outer isolation valve 27AOV-118 maintenance; and
- Work order 51208699, involving invasive cleaning and inspection of east crescent area unit cooler 66UC-22B.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20 – 1 sample)

a. Inspection Scope

The inspectors observed and reviewed selected refueling outage activities to verify that operability requirements were met and that risk, industry experience, and previous site specific problems were considered. The outage was in progress at the end of the previous inspection period, therefore this sample is a continuation of the inspection of refueling outage activities from the previous inspection period.

- The inspectors periodically verified proper alignment and operation of the shutdown cooling and alternate decay heat removal systems. The verification also included reactor cavity and fuel pool makeup paths and water sources, and administrative control of drain down paths.

- The inspectors reviewed procedures RAP-7.1.04B, "Refueling Procedure," and RAP-7.1.04C, "Neutron Instrument Monitoring During In-Core Fuel Handling," and the results of refueling platform interlock functional tests to ensure that the TS requirements for fuel movement were met. The inspectors also verified through review of procedure ST-39D, "Secondary Containment Leak Test," that containment requirements for refueling activities were met.

- The inspectors observed portions of the reactor startup on October 8 and 9, 2008, and verified through plant walkdowns, control room observations, and surveillance test reviews that the safety-related equipment required for mode change was operable, that containment integrity was set, and that reactor coolant boundary leakage was within TS limits. In addition, the inspectors conducted an inspection and walkdown of containment prior to reactor startup.

b. Findings

Introduction: A Green, self-revealing NCV of 10 CFR Part 50.65 (a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was identified when Entergy did not manage the increase in risk during the conduct of relay testing associated with emergency buses. The conduct of the relay testing resulted in an unanticipated loss of shutdown cooling (SDC).

Description: On October 7, 2008, a periodic trip test of a lockout relay on the 4.16 kV normal AC distribution bus, 10400, was being performed under work order 51192897-01. The test caused the 'B' 4.16 kV emergency AC distribution bus, 10600, to de-energize. This resulted in an automatic start of the 'B' and 'D' emergency diesel generators. De-energizing the 10600 bus resulted in the loss of the 'B' reactor protection system power supply and caused a primary containment isolation system group two isolation, including the closure of shutdown cooling suction valves, 10 MOV 17 and 10 MOV 18. 'D' RHR pump and 'B' RHR SW pump, which were operating in the shutdown cooling mode, were tripped by the 10600 bus load shedding circuit. All systems operated as designed in response to the loss of the 10600 bus.

Emergency bus 10600 is one of two redundant 4.16 kV emergency buses that supply power to safety related loads. At the time of the event, emergency bus 10600 was powered by bus 10400 and FitzPatrick was in cold shutdown (Mode 4). Operators entered appropriate procedures to restore power to the emergency bus and to restore shutdown cooling. Operators restored shutdown cooling within 33 minutes. During this time, reactor coolant temperature increased 6 degrees F.

Entergy determined that the cause of the event was that the trip and lockout relay test was re-scheduled outside of the original bus outage work window without performing a risk assessment review. The trip and lockout relay test work package was originally prepared for implementation during the bus outage and did not identify precautionary measures for conducting the test outside the bus outage work window. Per plant procedure, AP-10.09, "Maintenance Risk Assessment," Revision 24, a review for impact on key safety functions is required when an outage activity is rescheduled from within an approved system outage work window and this review was not performed. Additionally, the work package stated that the testing would result in several circuit breakers being affected that would impact emergency bus 10600 and, at the time, both the 10400 and 10600 bus were listed as protected equipment. The inspectors noted that these considerations were not addressed by plant personnel during the work review and approval process.

Analysis: This finding is more than minor because it is associated with the Mitigating Systems cornerstone and is related to maintenance risk assessment and management. A risk assessment review was not conducted prior to performance of a trip and lockout relay functional test associated with emergency buses. Specifically, this finding reflects inadequate risk management that contributed to a short duration loss of shutdown decay heat removal capability resulting from the inadvertent interruption of flow through the operating train of shutdown cooling with the plant in cold shutdown. This was reasonably within Entergy's ability to foresee and prevent because there were opportunities to recognize and manage the potential risk of losing shutdown cooling and to schedule the maintenance activity at a more appropriate maintenance window or take actions to prevent the loss of shutdown cooling.

In accordance with IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and Appendix G, "Shutdown Operations Significance Determination Process," the inspectors determined that this finding was of very low safety significance (Green). The basis for this determination is that in accordance with IMC 0609, Appendix G, Table 1, "Losses of Control," and Checklist 8, "BWR Cold Shutdown or Refueling Operation Time to Boil > 2 Hours: RCS Level <23 feet Above Top of Flange," this finding did not require quantification and did not constitute a significant loss of thermal margin, based upon the slow reactor coolant system heat-up rate and minimal time of interruption in shutdown cooling system operation. The problem was entered into Entergy's corrective action program as CR-JAF-2008-03805. Entergy implemented corrective actions that included communicating the error to personnel to reinforce management expectations for control of protected equipment and providing an additional level of work authorization review.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not plan and coordinate work activities properly to manage operational impact of work activities. Specifically, Entergy did not recognize that the emergency bus 10600 would be de-energized as a result of the trip and lockout relay functional test. (H.3(b))

Enforcement: 10 CFR Part 50.65 (a)(4), requires, in part, that before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), Entergy shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to the above, on October 7, 2008, Entergy did not manage the increase in risk prior to conducting trip and lockout relay functional testing associated with the emergency buses. Conduct of the testing resulted in loss of emergency bus 10600 and consequently a loss of shutdown cooling. Because this finding was of very low safety significance and was entered into Entergy's corrective action system as CR-JAF-2008-03805, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: (**NCV 05000333/2008005-02, Conduct of Relay Test Without Plant Impact Review Resulted in Loss of Emergency Bus and Shutdown Cooling.)**

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 design basis documents; 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 following STs were reviewed and represented six inspection samples:

- ST-39H, “RPV System Leakage Test and CRD Class 2 Piping Inservice Test,” Revision 27;
- ST-3F, “Core Spray Full Flow Test (IST),” Revision 4;
- ST-9BB, “B and D EDG Full Load Test and ESW Pump Operability Test,” Revision 9;
- ST-9AB, “EDG System B Fuel/Lube Oil Monthly Test,” Revision 1;
- ST-24J, “RCIC Flow Rate and Inservice Test (IST),” Revision 38; and
- ST-39F, “Primary Containment Integrated Leakage Rate (Type A) Test,” Revision 14.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 – 1 sample)

a. Inspection Scope

The inspectors observed simulator activities associated with licensed operator requalification training on November 3, 2008. The inspectors verified that emergency classification declarations and notification activities were properly completed. The inspectors evaluated the drill against the requirements of 10 CFR Part 50, Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities.” The inspectors observed Entergy’s critique and compared Entergy’s 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.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator (PI) Verification (71151 – 6 samples)

a. Inspection Scope

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

Cornerstone: Mitigating Systems

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

The inspectors reviewed data and plant records from July 2007 to July 2008. The records reviewed included PI data summary reports, licensee event reports, operator narrative logs, and maintenance rule records. The inspectors verified the accuracy of the number of critical hours reported, and interviewed the system engineers and operators responsible for data collection and evaluation.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," 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'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 appropriately 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 the daily inspector corrective action program item screening discussed in Section 4OA2.1. The review also included issues documented in system health reports, corrective maintenance work requests, component status reports, site monthly meeting reports and maintenance rule assessments. The inspectors' review nominally considered the six-month period of July 2008 through December 2008, although some examples expanded beyond those dates when the scope of the trend warranted. The inspectors compared and contrasted their results with the results contained in Entergy's latest integrated quarterly assessment report. Corrective actions associated with a sample of the issues identified in the trend report were reviewed for adequacy. The inspectors also evaluated the trend report specified in ENN-LI-102, "Corrective Action Process," and 10 CFR part 50 Appendix B. The documents reviewed are listed in the Attachment.

b. Assessment and Observations

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

.2 Annual Sample: Ultra Low Sulfur Fuel Oil and Fuel Oil Storage Tank Vortexing (71152 – 1 sample)

a. Inspection Scope

The inspectors reviewed Entergy's corrective actions following identification of two issues relating to the fuel oil storage tanks which were documented in CR-2007-2392. This issue was identified, in part, in a finding from a prior inspection report dated August 31, 2007 (ML072430509). Entergy identified on June 29, 2007 that the Technical Specification permitted American Petroleum Institute gravity range (27 degrees to 39 degrees) and 32,000 gallon minimum volume do not ensure a seven day supply of fuel oil within the fuel oil storage tanks to support emergency diesel generator operation for the analyzed load. NRC inspectors identified on July 10, 2007 that the evaluation for fuel oil quantity contained within the fuel oil storage tanks did not provide any allowance for submergence to prevent air entrainment from vortexing.

Specifically, the inspectors reviewed Entergy's calculations and compensatory measures to ensure the procedurally controlled fuel oil volumes maintained an adequate seven day supply, Entergy's analysis with respect to reportability concerns, and Entergy's actions towards obtaining a revision of the Technical Specifications in order to correct the non-conservative aspects.

b. Assessments and Observations

The revised surveillance tests and fuel oil volume limits are controlled based upon API gravity sampling and are more limiting by accounting for vortexing in the revised calculations. However, the inspectors identified an instance where the application of an equation accounting for vortexes was not well supported in the calculations. Specifically, the equation used for vortexing in Entergy's analysis was referenced in the text authored by J. Knauss, "Swirling Flow Problems at Pump Intakes," as "Chang, 1979." However, this text did not recommend using the Chang equation, nor did the text provide information with respect to ranges of applicability and appropriateness of the equation under various conditions. In addition, Entergy did not have other references which could provide insight into the suitability of the equation or its range of applicability/accuracy nor did Entergy consult the original work referenced as "Chang, 1979" in order to perform an assessment of its appropriateness and suitability.

In response to the inspector's questions, Entergy provided supporting analysis which indicated that the Chang equation yielded a more conservative result for submergence than would be obtained using an equation Entergy derived from analyzing NUREG/CR-2772, "Hydraulic Performance of Pump Suction Inlets for Emergency Core Cooling Systems (ECCS) in Boiling Water Reactors." The inspectors questioned the appropriateness of Entergy's supporting calculation because it included results from a derived equation that differed from empirical data and used data from NUREG/CR-2772 obtained with a strainer at the fuel oil storage tank suction pipe inlet. Fitzpatrick did not have a strainer at the fuel oil storage tank suction pipe inlet. However, the inspectors determined that sufficient information was contained within NUREG/CR-2772, including an analysis without a suction strainer, in order to bound the calculated submergence. As a result, the inspectors determined that the Chang approach provided a comparable value for submergence relative to the NUREG/CR-2772 bounding value and the fuel oil volume limits provided for sufficient oil to meet the design basis. Therefore these issues are minor.

.3 Annual Sample: Aging and Material Degradation Problems Are Adversely Affecting Reliability of Plant Equipment (71152 – 1 sample)

a. Inspection Scope

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

- CR-2008-00726, Aging and Material Degradation Problems Are Adversely Affecting

Reliability of Plant Equipment

b. Assessment and Observations

No findings of significance were identified related to aging and material degradation problems. The inspectors determined that Entergy appropriately identified equipment, human performance and program issues at an appropriate threshold and entered them into the corrective action program.

.4 Annual Sample: ‘A’ Reactor Protection System Deenergized Due to Fault

a. Inspection Scope

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

- CR-2008-03946, ‘A’ Reactor Protection System Deenergized Due to Fault

b. Assessment and Observations

No findings of significance were identified related to the ‘A’ reactor protection system. The inspectors determined that Entergy appropriately identified equipment, human performance and program issues at an appropriate threshold and entered them into the corrective action program.

4OA3 Event Follow-up (71153- 4 samples)

.1 (Closed) LER 05000333/2008001-00, Loss of Shutdown Cooling Resulting From Invalid PCIS [Primary Containment Isolation System] Actuation Signal

On September 16, 2008, shutdown cooling flow was isolated while hanging a tagout on the ‘B’ reactor protection system. When removing fuses as directed by the tagout, the isolation logic for the shutdown cooling suction valves was actuated. At the time of the isolation, the reactor was in the refueling mode and reactor cavity flood-up was in progress. The enforcement aspects of this violation were documented in section 1R20 of NRC Inspection Report 05000333/2008004. Entergy entered the event into its corrective action program as CR-2008-02997. The inspectors reviewed this LER and no new findings were identified. This LER is closed.

.2 (Closed) LER 05000333/2008002-00, Reactor Pressure Vessel Recirculation Inlet Nozzle Axial Flaw Indication, Discovered During Refueling Outage, Consistent With Inter-Granular Stress Corrosion Cracking

On September 23, 2008, with the plant shutdown and in the refueling mode, an ultrasonic examination was performed in accordance with the in-service inspection program on reactor pressure vessel reactor recirculation inlet nozzle, N2C that showed an inner diameter axial flaw indication approximately 0.8 inches long with a 0.5 inch wall depth. The flaw indication was located in the dissimilar metal weld area of the reactor pressure vessel N2-C nozzle to safe-end weld and was consistent with inter-granular stress corrosion cracking. Entergy performed a full structural weld overlay using an alternative repair procedure in accordance with an approved NRC relief request. The current refueling outage dissimilar metal weld inspection scope for the remaining reactor pressure vessel nozzle to safe-end examinations were completed with no other flaw indications identified. The LER was reviewed by the inspectors and no findings of significance were identified and no violation of NRC requirements occurred. Entergy documented the flaw indication in CR 2008-03311. This LER is closed.

.3 (Closed) LER 05000333/2008003-00, Loss of Emergency Bus and Auto-Start of 'B' EDG(s) Due To Rescheduled Relay Functional Test Without Risk Assessment Review

On October 7, 2008, during testing of a lockout relay on the 4.16 kV normal AC distribution bus, 10400, the 'B' 4.16 kV emergency AC distribution bus, 10600 was de-energized. This resulted in an automatic start of the 'B' and 'D' emergency diesel generators which re-energized the 10600 bus. De-energizing the 10600 bus resulted in the loss of the 'B' reactor protection system power supply and caused a primary containment isolation system group two isolation, including closing the shutdown cooling suction valves, 10 MOV 17 and 10 MOV 18. 'D' RHR pump and 'B' RHR SW pump which were operating in the shutdown cooling mode, were tripped by the 10600 bus load shedding circuit. All systems operated as designed in response to the loss of the 10600 bus. The enforcement aspects of this violation were documented in Section 1R20 of this inspection report. Entergy entered the event into its corrective action program as CR-2008-03805. The inspectors reviewed this LER and no new findings were identified. This LER is closed.

.4 (Closed) LER 05000333/2008004-00, Loss of Power Instrumentation Inoperable and Technical Specification Required Action Time Exceeded Due to Relay Set Point Drift

On September 23, 2008, with the plant shutdown and in the refueling mode, the 4.16 kV emergency bus degraded voltage time delay relay failed to meet the as-found TS surveillance acceptance criteria of greater than or equal to 41.0 seconds and less than or equal to 46.6 seconds. The relay was found to actuate at 47.29 seconds. Entergy determined the cause to be that the relay, while widely used in this type of application, was only marginally acceptable in this application due to instrument drift. Therefore, the cause is an original design deficiency.

Additional corrective actions completed or planned included replacement of the relay, increasing the frequency of the instrument surveillance procedure and replacement of the relay with electronic timer relays. This finding is more than minor because it is associated with equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and determined it to be of very low safety significance (Green) because the finding represented a design or qualification deficiency confirmed not to result in loss of operability. Though the as-found value for the degraded time delay relay exceeded the TS allowed value, the relay remained capable of performing its safety function as the as-found value was within the plant design bases accident analyses credited calculated allowable value for the non-loss of coolant accident degraded voltage time delay relay of 58.5 seconds.

No new findings were identified in the inspector's review. This licensee-identified finding involved a violation of TS 3.3.8.1. Entergy entered the event into its corrective action program as CR-2008-03796. The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

4OA5 Other Activities

.1 Implementation of Temporary Instruction (TI) 2515/176, "Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing"

a. Inspection Scope

The objective of TI 2515/176, "Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing," is to gather information to assess the adequacy of nuclear power plant emergency diesel generator (EDG) endurance and margin testing as prescribed in plant-specific technical specifications (TS). The inspectors reviewed emergency diesel generator ratings, design basis event load calculations, surveillance testing requirements, and emergency diesel generator vendor's specifications and gathered information in accordance with TI 2515/176.

The inspector assessment and information gathered while completing this TI was discussed with Entergy personnel. This information was forwarded on to the Office of Nuclear Reactor Regulation for further review and evaluation.

b. Findings

No findings of significance were identified.

.2 10 CFR 50.46 Annual Report, "Errors in Emergency Core Cooling System Evaluation Models"

On December 22, 2008, in accordance with the requirements of 10 CFR 50.46 (a)(3)(ii), Entergy submitted a report for an October 2008 change in reactor fuel type that resulted in a change to the calculated peak clad temperature. The report also identified that an error related to the top peaked power shape for small break loss of coolant accident analysis was discovered in the evaluation in July 2006. Entergy identified that a report should have been submitted per 10 CFR 50.46 (a)(3)(ii) in 2007, but was not.

The inspectors reviewed the report and the reporting requirements of 10 CFR 50.46. The error discovered in July 2006, did not result in a change in peak clad temperature and thus would be considered minor (an error resulting in a change in peak clad temperature of 50 degrees or greater is considered significant and requires a 30 day report). However, all errors are required to be reported via an annual report, thus the failure to make a timely report is a violation of regulatory requirements. Failure to make a required report to the

NRC has the potential to impact the regulatory process and is evaluated under the traditional enforcement process. Using the NRC Enforcement Policy (NUREG 1600) Supplement I "Reactor Operations," this issue is most similar to Severity Level IV example D4: "A failure to meet regulatory requirements that have more than minor safety or environmental significance". However, as stated above, the underlying technical issue, an error related to the top peaked power shape for a small break loss of coolant accident analysis, was determined to be minor because it did not result in a change in the peak clad temperature. Additionally, this error would not have influenced regulatory decision making. As a result, the inspectors determined that this issue was a minor violation of a regulatory requirement. Entergy identified this issue, reported the failure to make this timely report to the NRC, and entered the issue into their corrective action program as CR-JAF-2008-4624. This issue is closed.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

Exit Meeting Summary

On January 12, 2009, the inspectors presented the inspection results to Mr. Peter T. Dietrich and other members of his staff. The inspectors asked Entergy whether any of the material examined during the inspection should be considered proprietary. Entergy did not identify any material as proprietary information.

4OA7 Licensee-Identified Violations

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

- TS 3.3.8.1 requires that the 4.16 kV emergency bus undervoltage (degraded voltage) time delay (non-loss of coolant accident) channel be placed in trip in one hour if the allowable value is out of specification. Contrary to this on September 23, 2008, the as found time delay failed to meet the TS allowed value limit and the TS required action time was exceeded without placing the channel in trip. The inspectors evaluated the significance of this finding using IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and determined it to be of very low safety significance (Green) because the finding represented a design or qualification deficiency confirmed not to result in loss of operability. This was entered into Entergy's corrective action program as CR-2008-03796.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

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

LIST OF ITEMS OPEN, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000333/2008005-01	NCV	Quality Standards Not Specified in Design Documents that Resulted in Unsupported HPCI Oil Tubing
05000333/2008005-02	NCV	Conduct of Relay Test Without Plant Impact Review Resulted in Loss of Emergency Bus and Shutdown Cooling

Closed

05000333/2008001-00	LER	Loss of Shutdown Cooling Resulting From Invalid PCIS [Primary Containment Isolation System] Actuation Signal
05000333/2008002-00	LER	Reactor Pressure Vessel Recirculation Inlet Nozzle Axial Flaw Indication, Discovered

		During Refueling Outage, Consistent With Inter-Granular Stress Corrosion Cracking
05000333/2008003-00	LER	Loss of Emergency Bus and Auto-Start of ‘B’ EDG(s) Due To Rescheduled Relay Functional Test Without Risk Assessment Review
05000333/2008004-00	LER	Loss of Power Instrumentation Inoperable and Technical Specification Required Action Time Exceeded Due to Relay Set Point Drift

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1RO1: Adverse Weather Protection
OP-51A, “Reactor Building Ventilation and Cooling System,” Revision 47
OP-52, “Turbine Building Ventilation,” Revision 16
DBD-066, “Design Basis Document for the Reactor Building Heating, Ventilation and Air Condition (HVAC) Systems”
DBD-067, “Design Basis Document for the Turbine Building HVAC Systems”

Section 1RO4: Equipment Alignment
OP-18, “Reactor Protection System,” Revision 27
OP-44, “115kV System,” Revision 16
OP-68, “Automatic Depressurization System,” Revision 18
Work Order 000804300, Engineering Assessment of HPCI Tubing
GE Field Disposition Instruction 83/88595, “HPCI and RCIC Turbines”

Section 1RO5: Fire Protection
Fire Area/Zone VII/CS-1, elevation 272 foot – PFP-PWR-11
Fire Area/Zone VII/RR-1, elevation 286 foot – PFP-PWR-12
Fire Area/Zone Yard, elevation 272 foot – PFP-PWR-49
Fire Area/Zone IB/SH-1, elevation 235, 255 and 260 foot – PFP-PWR-34
Fire Area/Zone IB/SH-1, elevation 272 foot – PFP-PWR-35

Section 1R07: Heat Sink Performance
Procedures
AP-09.02, “Zebra Mussel Control Program,” Revision 7
AP-19.12, “SW Inspection Program,” Revision 5
AP-19.14, “Eddy Current Testing of Heat Exchanger Tubes,” Revision 9
MDSO-14, “Heat Exchanger Tube Plugging,” Revision 8
OP-4, “Circulating Water System,” Revision 61

OP-7A, “Chlorine Injection System,” Revision 22
OP-42A, “SW Chemical Cleaning System,” Revision 5

<u>Surveillances</u>
ST-2YA, “RHR Heat Exchanger A Performance Test,” Revision 0
ST-2YB, “RHR Heat Exchanger B Performance Test,” Revision 0

<u>Engineering Evaluations and Calculations</u>
JAF-CALC-RHR-02953, “RHR Heat Exchanger K-Value with Reduced Tube Side Fouling Factor,” Revision 0

<u>System Health Report</u>
RHR and residual heat removal SW, 3rd quarter 2008

<u>Miscellaneous</u>
NED-M-090-107, “James A. FitzPatrick Nuclear Power Plant Temporary Chlorine Water Treatment System for SW Systems Modification F1-90-038," Memorandum dated September 14, 1990
JAF-RPT-MULTI-01267, “Generic Letter 89-13 Program Plan,” Revision 4
JAF-RPT-MULTI-02294, “Maintenance Rule Basis Document for SW Systems Including System 010-000 RHR SW System 046-ESW Emergency SW System 046-000 Normal SW,” Revision 7
WO JAF-05-15659
WO 51522220

<u>Drawings</u>
4.95-5, “10E-2A RHR Heat Exchanger ‘A’ Tube Plugging Map,” Revision 1
4.95-6, “10E-2B RHR Heat Exchanger ‘B’ Tube Plugging Map,” Revision 1

<u>Section 1R11: Licensed Operator Requalification Program</u>
Evaluation 2008A, Loss of RWR Pump; Unisolable RCIC Steam Leak in Reactor Building; Failure to Scram; Emergency Depressurization with degraded SRV response – Use of Alternate Depressurization Systems
AOP-8, “Loss or Reduction of Reactor Coolant Flow,” Revision 29
AOP-1, “Reactor Scram,” Revision 43

<u>Section 1R12: Maintenance Effectiveness</u>
<u>Procedures</u>

EN-DC-203, "Maintenance Rule Program," Revision 0
EN-DC-204, "Maintenance Scope and Basis," Revision 0
EN-DC-205, “Maintenance Rule Monitoring," Revision 0
EN-DC-324, “Preventive Maintenance Process,” Revision 3
EN-LI-102, "Corrective Action Process," Revision 10
ENN-DC-171, “Maintenance Rule Monitoring,” Revision 2

<u>Miscellaneous</u>
ENN-MS-S-004-JAF, “System Categorization – JAF,” Revision 2
ENN-MS-S-009-JAF, “JAF Safety System Function Sheets,” Revision 1

Maintenance rule quarterly report 1st quarter 2008
Maintenance rule quarterly report 2nd quarter 2008
JAF-RPT-MISC-02272, “Maintenance Rule Basis Document for Plant Level Performance,” Revision 7
Maintenance Rule Quarterly Report, 3rd quarter 2008
JAF-RPT-07-00030, “Maintenance Rule Basis Document/ System 02/ Automatic Depressurization System,” Revision 2
Automatic Depressurization System Health Report, 4th Quarter 2008
Automatic Depressurization system Health Improvement Plan Action
JAF-RPT-RWCU-02283, “Maintenance Rule Basis Document/ System 12/ Reactor Water Cleanup System,” Revision 4
Reactor Water Cleanup System Health Report, 1st half 2008
Reactor Water Cleanup System Health Improvement Plan Actions
Reactor Water Cleanup System Monitoring Agenda
CR-JAF-2007-01019
CR-JAF-2007-01128
CR-JAF-2007-01925
CR-JAF-2007-01929
CR-JAF-2007-01980
CR-JAF-2007-02434
CR-JAF-2007-02745
CR-JAF-2007-02777
CR-JAF-2007-02921
CR-JAF-2008-00700
CR-JAF-2008-00754
CR-JAF-2008-02148
CR-JAF-2008-02712
CR-JAF-2008-02810

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

AP-12.12, “Protective Equipment Program,” Revision 3
AP-10.10, On-Line Risk Assessment; Revision 6

Section 1R15: Operability Evaluations

JAF-RPT-03-00056, “Operational Leakage Action Levels for Target Rock Two-Stage Safety/Relief Valves,” Revision 0
VT-1 Report Numbers 08VT154 and 08VT149, Primary Containment Moisture Barrier Area
CES-8B, “JAF Tubing Design,” Revision 0

Section 1R19: Post Maintenance Testing

Engineering Change EC 5000018317, “Maximum Extended Operating Domain”
TST-103, “Testing of ESW Loop B (IST),” Revision 5
OP-21, “Emergency Service Water (ESW),” Revision 35
FM-46B, “Flow Diagram Emergency Service Water System 46 and 15,” Revision 50
FB-10H, “Flow Diagram Reactor Building Service Water Cooling System 66,” Revision 43

Section 1R20: Refueling and Other Outage Activities

AOP-30, “Loss of Shutdown Cooling,” Revision 19
AOP-19, “Loss of 10600 Bus,” Revision 12
AOP-60, “Loss of RPS Bus B Power,” Revision 5
CR 2008-03805

Section 1R22 Surveillance Testing
JAF-RPT-PC-02342, Primary Containment Leakage Rate Testing Program Plan
AP-19.05, Pump and Valve Inservice Testing, Revision 8
ST-39B, Type B and C LLRT of Containment Penetrations (IST), Revision 32

Section 4OA2: Identification and Resolution of Problems

JAF-CALC-07-00019, “Volume in EDG Underground Fuel Oil Storage Tanks as a Function of Level,” Revision 0
JAF-CALC-07-00020, “Revised Emergency Diesel Generator (EDG) Fuel Oil Storage Quantities for 7 Day and 6 Day Supplies,” Revision 0
EC No.: 11904
J. Knauss, “Swirling Flow Problems at Pump Intakes,” A.A. Balkema, 1987
EN-ME-G-001, “Evaluation of Pump Protection from Low Submergence,” Revision 0

Condition Reports

2007-01888	2008-03986	2008-04112
2007-02392	2008-04023	2008-04115
2007-02490	2008-04037	2008-04122
2007-03183	2008-04040	2008-04124
2008-03892	2008-04044	2008-04126
2008-03906	2008-04045	2008-04136
2008-03907	2008-04070	2008-04138
2008-03915	2008-04071	2008-04140
2008-03946	2008-04082	2008-04148
2008-03951	2008-04089	2008-04150
2008-03969	2008-04105	2008-04151
2008-03971	2008-04111	2008-04153

Section 4OA5: Implementation of Temporary Instruction (TI) 2515/176 – Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing

Completed Surveillance Procedures

ST-9BA, Rev. 9, A and C Full Load Test and ESW Pump Operability Test, completed July 21, June 23, and May 26, 2008
ST-9BB, Rev. 9, B and D Full Load Test and ESW Pump Operability Test, completed August 9, July 7, and June 9, 2008

Procedures

ST-9BA, Rev. 9, A and C Full Load Test and ESW Pump Operability Test
ST-9BB, Rev. 9, B and D Full Load Test and ESW Pump Operability Test
ST-9QA, Rev. 6, EDG A and C Full Load Test (8 Hour Run)

Calculations

14629-E-77-01, Rev. 1, Emergency Diesel Generator Load Review
E77-01, Rev. 1, EC#4599, Emergency Diesel Generator load Review at 61.2 Hz
JAf-CALC-EDG-03358, JAF Single EDG Loading

Other Documents

Updated Final Safety Analysis Report, Section 8.6, Rev. 1, Emergency AC Power System
Updated Final Safety Analysis Report, Section 14.6.1.3, Rev. 5, Loss of Coolant Accident
LBDCR 07-013, Revise UFSAR Table 8.6-1 to Reflect Tech Spec Maximum Frequency of 61.2 Hz

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
CFR	Code of Federal Regulations
CR	condition report
ECCS	emergency core cooling system
EDG	emergency diesel generator
Entergy	Entergy Nuclear Northeast
FitzPatrick	James A. FitzPatrick Nuclear Power Plant
HPCI	high pressure coolant injection
IMC	inspection manual chapter
IST	inservice test
NCV	non-cited violation
NRC	Nuclear Regulatory Commission
OA	other activities
OP	operating procedure
PARS	Publicly Available Records
PI	performance indicator
RHR	residual heat removal
SDC	shutdown cooling
SDP	significance determination process
SSC	structures, systems, or components
ST	surveillance test
SW	service water
TI	Temporary Instruction
TS	technical specification
UFSAR	updated final safety analysis report