UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BOULEVARD, SUITE 100
KING OF PRUSSIA, PENNSYLVANIA 19406-2713

April 24, 2013

Mr. Michael J. Colomb
Site Vice President
James A. FitzPatrick Nuclear Power Plant
Entergy Nuclear Northeast
P. O. Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000333/2013002

Dear Mr. Colomb:

On March 31, 2013, 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 April 12, 2013, 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.

The report documents three Severity Level IV violations. However, because of the very low safety significance, and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs), consistent with Section 2.3.2 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of the inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspector at FitzPatrick.

In accordance with Title 10 of the *Code of Federal Regulations* 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 component of the NRC's Agencywide Documents Access Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Arthur L. Burritt, Chief
Reactor Projects Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report No. 05000333/2013002
w/Attachment: Supplementary Information

cc w/encl: Distribution via ListServ

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Sincerely,
/RA/
Arthur L. Burritt, Chief
Reactor Projects Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report No. 05000244/2013002
w/Attachment: Supplementary Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-333
License No.:	DPR-59
Report No.:	05000244/2013002
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Scriba, New York
Dates:	January 1 through March 31, 2013
Inspectors:	E. Knutson, Senior Resident Inspector B. Sienel, Resident Inspector J. Laughlin, Emergency Preparedness Inspector S. McCarver, Project Engineer E. Miller, Resident Inspector
Approved by:	Arthur L. Burritt, Chief Reactor Projects Branch 2 Division of Reactor Projects

Enclosure

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SUMMARY

IR 05000333/2013002; 01/01/2013 - 03/31/2013; James A. FitzPatrick Nuclear Power Plant (FitzPatrick); Maintenance Effectiveness, Problem Identification and Resolution, and Other Activities.

This report covered a 3-month period of inspection by resident inspectors, an announced inspection at the Entergy Nuclear Northeast office performed by regional inspectors, and an in-office review performed by headquarters personnel. Inspectors identified three Severity Level IV non-cited violations (NCVs). The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP), dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Components Within the Cross-Cutting Areas," dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated June 7, 2012. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4.

Cornerstone: Mitigating Systems

- Severity Level IV. The inspectors identified a Severity Level (SL) IV non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.73, "Licensee Event Report [LER] System," because failure of an isolation valve in the high-pressure coolant injection (HPCI) system torus suction line to fully open on demand caused the automatic suction swap function to be inoperable, but this condition was not reported to the NRC as a condition that could have prevented fulfillment of a safety function per 10 CFR 50.73(a)(v) within 60 days of when it should reasonably have been discovered. Specifically, while this condition existed, an automatic suction swap from the condensate storage tanks (CSTs) to the torus would not have gone to completion, but rather would have stopped with both suction paths open. Depending on whether or not HPCI was running at the time, this would either result in air entrainment in the HPCI pump suction, causing a loss of HPCI, or an increase in suppression pool level due to drainage from the CSTs. However, this condition was not reported to the NRC as a condition that could have prevented fulfillment of a safety function per 10 CFR 50.73(a)(v) within 60 days of when it should reasonably have been discovered. This issue was entered into the corrective action program (CAP) as condition report (CR)-JAF-2013-01768.

  The inspectors determined that the failure to submit an LER within 60 days in accordance with 10 CFR 50.73 was a performance deficiency that was reasonably within Entergy's ability to foresee and correct. Because the issue impacted the regulatory process, in that a safety system functional failure was not reported to the NRC within the required timeframe, thereby delaying the NRC's opportunity to review the matter, the inspectors evaluated this performance deficiency in accordance with the traditional enforcement process. Using example 6.9.d.9 from the NRC Enforcement Policy, the inspectors determined that the violation was a SL IV (more than minor concern that resulted in no or relatively inappreciable potential safety or security consequence) violation, because Entergy personnel failed to

make a report required by 10 CFR 50.73 when information that the report was required had been reasonably within their ability to have identified. In accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," traditional enforcement issues are not assigned cross-cutting aspects. (Section 1R12)

- Severity Level IV. The inspectors identified a Severity Level (SL) IV non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.73, "Licensee Event Report [LER] System," because a violation of technical specification (TS) 3.0.4 for a reactor mode change being made from Mode 4 to Mode 2 without satisfying the TS required conditions for alignment of the containment air dilution and standby gas treatment (SGT) systems in Mode 2 was not reported to the NRC within 60 days of when it should reasonably have been discovered. Specifically, in Modes 1, 2, and 3, TS surveillance requirement (SR) 3.6.1.3.1 allows the 20-inch and 24-inch primary containment vent and purge valves to be open for inerting, deinerting, pressure control, or other reasons provided that valve 27MOV-120 in the full flow line to the SGT system is closed. This is to ensure that there would be no damage to the SGT filters if a loss-of-coolant accident (LOCA) were to occur with the vent and purge valves open. However, on November 24, 2012, operators transitioned the reactor from Mode 4 to Mode 2 while the 20-inch and 24-inch containment vent and purge valves, and valve 27MOV-120, were open. This condition was not reported to the NRC within 60 days of when it should reasonably have been discovered. As immediate corrective action, FitzPatrick staff entered the issue into the corrective action program as condition report (CR)-JAF-2013-01097.

  The inspectors determined that the failure to submit an LER within 60 days in accordance with 10 CFR 50.73 was a performance deficiency that was reasonably within Entergy's ability to foresee and correct. Because the issue impacted the regulatory process, in that a violation of site TSs was not reported to the NRC within the required timeframe, thereby delaying the NRC's opportunity to review the matter, the inspectors evaluated this performance deficiency in accordance with the traditional enforcement process. Using example 6.9.d.9 from the NRC Enforcement Policy, the inspectors determined that the violation was a SL IV (more than minor concern that resulted in no or relatively inappreciable potential safety or security consequence) violation, because Entergy personnel failed to make a report required by 10 CFR 50.73 when information that the report was required had been reasonably within their ability to have identified. In accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," traditional enforcement issues are not assigned cross-cutting aspects. (Section 4OA2)

- Severity Level IV. The inspectors identified a Severity Level (SL) IV non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, "Changes, Tests, and Experiments," because Entergy personnel implemented a change to the technical specification (TS) definition of core quadrant without prior review and approval by the NRC staff in accordance with 10 CFR 50.59(c)(1)(i). Specifically, Entergy staff changed the definition of core quadrant in Revision 5 of Reactor Analyst procedure RAP-7.1.04C, "Neutron Instrumentation Monitoring During In-Core Fuel Handling," which allowed operators to interpret what constitute core quadrant boundaries, such that core alterations could be performed anywhere in the core provided any three source range [neutron] monitors (SRMs) were operable. As immediate corrective action to the task interface agreement (TIA) final response, FitzPatrick staff withdrew RAP-7.1.04C pending revision of the core quadrant

definition. The inspectors verified that TS 3.3.1.2.2 had been satisfied during all core alterations that were performed during the 2010 and 2012 refueling outages, using the standard definition of a core quadrant. Entergy staff entered this issue into the corrective action program (CAP) as condition report (CR)-HQN-2013-00034.

The inspectors determined that Entergy staff's implementation of a redefinition of core quadrant prior to its review and approval by the NRC staff as specified in 10 CFR 50.59(c)(1)(i) was a performance deficiency that was reasonably within Entergy staff's ability to foresee and correct. Because this was a violation of 10 CFR 50.59, it was considered to be a violation that potentially impedes or impacts the regulatory process. Therefore, this violation was characterized using the traditional enforcement process. The violation was determined to be more than minor in accordance with the NRC Enforcement Manual, Section 7.3.E.6, because there was a reasonable likelihood that the change to the definition of what constituted a "core quadrant boundary" would require Commission review and approval prior to implementation. Additionally, the inspectors noted that, in accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Screening," the underlying performance deficiency would screen as more than minor because, if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, potentially inadequate SRM coverage during refueling operations could affect the TS bases function to provide early indication of unexpected subcritical multiplication that could be indicative of an approach to criticality. NRC Enforcement Manual Section 7.3 provides guidance to assess 10 CFR 50.59 violations through the significance determination process (SDP). In this case, the inspectors determined the violation could be evaluated using the SDP in accordance with IMC 0609 Appendix G, "Shutdown Operations Significance Determination Process," Checklist 7, "BWR Refueling Operation with RCS Level Greater Than 23 Feet." The finding affected the Reactivity Guidelines attribute that assumes existing core alteration TS are being met. Since this attribute does not require quantitative assessment, the finding was screened as Green in accordance with Section 3.3, "Mitigation Capability." In accordance with the NRC Enforcement Policy, Section 6.1.d.2, this violation was categorized as SL IV because the issue was evaluated by the SDP as having very low safety significance (Green). The finding did not have a cross-cutting aspect because the performance deficiency did not occur within the past three years and therefore was not reflective of present performance. (Section 4OA5)

REPORT DETAILS

Summary of Plant Status

James A. FitzPatrick Nuclear Power Plant (FitzPatrick) began the inspection period at 100 percent power. On January 30, 2013, operators reduced power to 63 percent to conduct a control rod sequence exchange, perform control blade interference testing, and perform main turbine valve testing. Operators restored power to 100 percent later that day. On February 1, 2013, operators reduced power to 50 percent to address main condenser tube leakage. Following identification and repair, operators restored power to 100 percent the following day. On February 3, 2013, operators reduced power to 85 percent to conduct a control rod pattern adjustment and restored power to 100 percent later that day. On February 23, 2013, operators reduced power to 50 percent to address main condenser tube leakage. Following identification and repair, operators restored power to 100 percent the following day. On February 25, 2013, operators reduced power to 80 percent to conduct a control rod pattern adjustment and restored power to 100 percent later that day. On March 2, 2013, operators commenced a shutdown for repair of tube leaks in the 6A feedwater heater. Following completion of repairs, operators commenced a reactor startup on March 7, 2013. Operators returned the unit to 100 percent power on March 9, 2013. On March 18, 2013, operators reduced power to 50 percent to address main condenser tube leakage. Following identification and repair, operators restored power to 100 percent on March 21, 2013. On March 22, 2013, operators reduced power to 50 percent to address main condenser tube leakage. Following identification and repair, operators restored power to 100 percent on March 23, 2013. On March 24, 2013, operators reduced power to 85 percent to conduct a control rod pattern adjustment and restored power to 100 percent later that day. The plant remained at or near 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample)

Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

The inspectors reviewed FitzPatrick's preparations for the onset of high winds on January 31, 2013. The inspectors reviewed the implementation of adverse weather preparation procedures before the onset of and during this adverse weather condition. The inspectors walked down the plant exterior to identify loose or inadequately protected equipment and materials. The plant did not experience any significant operational issues as a result of the high winds. Documents reviewed for each section of this inspection report are listed in the Attachment.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns (71111.04 - 4 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- ‘B’ control room emergency ventilation air supply (CREVAS) system while the ‘A’ CREVAS system was inoperable for maintenance on January 29, 2013
- ‘B’ residual heat removal (RHR) system while ‘A’ RHR system was inoperable for maintenance on February 5, 2013
- ‘A’ core spray system while ‘B’ RHR system was inoperable for maintenance on February 20, 2013
- ‘B’ core spray system while ‘A’ core spray system was inoperable for maintenance on March 26, 2013

The inspectors selected these systems based on their risk-significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors reviewed applicable operating procedures, system diagrams, the Updated Final Safety Analysis Report (UFSAR), technical specifications (TSs), condition reports (CRs), and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have impacted system performance of their intended safety functions. The inspectors performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and were operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no deficiencies. The inspectors also reviewed whether Entergy staff had properly identified equipment issues and entered them into the corrective action program (CAP) for resolution with the appropriate significance characterization.

b. Findings

No findings were identified.

.2 Full System Walkdown (71111.04S - 1 sample)

a. Inspection Scope

On March 26, 2013, the inspectors performed a complete system walkdown of accessible portions of the high-pressure coolant injection (HPCI) system to verify the existing equipment lineup was correct. The inspectors reviewed operating procedures, surveillance tests, drawings, equipment line-up check-off lists, and the UFSAR to verify the system was aligned to perform its required safety functions. The inspectors also

reviewed electrical power availability, component lubrication and equipment cooling, hanger and support functionality, and operability of support systems. The inspectors performed field walkdowns of accessible portions of the system to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no deficiencies. Additionally, the inspectors reviewed a sample of related CRs and work orders (WOs) to ensure FitzPatrick personnel appropriately evaluated and resolved any deficiencies.

b. Findings

No findings were identified.

1R05 Fire Protection

Resident Inspector Quarterly Walkdowns (71111.05Q - 5 samples)

a. Inspection Scope

The inspectors conducted tours of the areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that FitzPatrick personnel controlled combustible materials and ignition sources in accordance with administrative procedures. The inspectors verified that fire protection and suppression equipment was available for use as specified in the area pre-fire plan, and passive fire barriers were maintained in good material condition. The inspectors also verified that station personnel implemented compensatory measures for out of service, degraded, or inoperable fire protection equipment, as applicable, in accordance with procedures.

- Relay room, fire area/zone VII/RR-1 on January 11, 2013
- East cable tunnel, fire area/zone II/CT-2 on February 7, 2013
- Reactor building 272 foot elevation, fire area/zone IX/RB-1A and X/RB-1B on February 12, 2013
- East electric bay, fire area/zone IE/TB-1 and OR-2 on March 5, 2013
- Condenser bay, fire area/zone IE/TB-1 and OR-2 on March 6, 2013

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06 - 1 sample)

Internal Flooding Review

a. Inspection Scope

The inspectors reviewed the UFSAR, FitzPatrick's individual plant examination, and plant procedures to assess susceptibilities involving internal flooding. The inspectors

also reviewed the CAP to determine if FitzPatrick staff identified and corrected flooding problems and whether operator actions for coping with flooding were adequate. The inspectors focused on the turbine building 252-foot elevation to evaluate potentially safety-significant vulnerabilities to flooding from the normal service water (SW) system.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11Q - 2 samples)

.1 Quarterly Review of Licensed Operator Requalification Testing and Training

a. Inspection Scope

The inspectors observed a licensed operator simulator exam on March 11, 2013, which included the failure of various plant equipment and required entry into the emergency plan and related procedures. The inspectors evaluated operator performance during the simulated event and verified completion of risk significant operator actions, including the use of abnormal and emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, implementation of actions in response to alarms and degrading plant conditions, and the oversight and direction provided by the control room supervisor. The inspectors also verified the accuracy and timeliness of the emergency classifications made by the shift manager. Additionally, the inspectors assessed the ability of the crew and training staff to identify and document crew performance problems.

b. Findings

No findings were identified.

.2 Quarterly Review of Licensed Operator Performance in the Main Control Room

a. Inspection Scope

On January 30, 2013, the inspectors observed control room operators during a power reduction to 63 percent for the performance of a control rod sequence exchange, control blade interference testing, and main turbine valve testing. The inspectors observed crew performance to verify that procedure use, crew communications, and coordination of activities between work groups met established expectations and standards.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q - 3 samples)

a. Inspection Scope

The inspectors reviewed the samples listed below to assess the effectiveness of maintenance activities on structure, system, or component (SSC) performance and reliability. The inspectors reviewed system health reports, CAP documents, and maintenance rule basis documents to ensure that FitzPatrick staff was identifying and properly evaluating performance problems within the scope of the maintenance rule. For each sample selected, the inspectors verified that the SSC was properly scoped into the maintenance rule in accordance with 10 CFR Part 50.65 and verified that the (a)(2) performance criteria established by FitzPatrick staff was reasonable. As applicable, for SSCs classified as (a)(1), the inspectors assessed the adequacy of goals and corrective actions to return these SSCs to (a)(2). Additionally, the inspectors ensured that FitzPatrick staff was identifying and addressing common cause failures that occurred within and across maintenance rule system boundaries.

- HPCI
- Containment air dilution
- Reactor building ventilation

b. Findings

Introduction. The inspectors identified a SL IV NCV of 10 CFR 50.73, "Licensee Event Report [LER] System," because failure of an isolation valve in the HPCI system torus suction line to fully open on demand caused the automatic suction swap function to be inoperable, but this condition was not reported to the NRC as a condition that could have prevented fulfillment of a safety function per 10 CFR 50.73(a)(v) within 60 days of when it should reasonably have been discovered.

Description. On June 7, 2012, during a HPCI system planned maintenance period, testing of valve 23MOV-57, HPCI pump suction from suppression pool downstream isolation valve, identified that the valve opened only to 38 percent rather than to 100 percent on demand. The cause was determined to be a high resistance contact in the open torque switch. The contact was burnished, and the valve subsequently operated satisfactorily. The issue was entered into the CAP as CR-JAF-2012-03298. The condition was determined not to be reportable to the NRC because engineering evaluated that the valve would have provided adequate flow to maintain HPCI operability at 38 percent open. Also, operation of the valve had been satisfactory over the previous 3 years, including quarterly operation during routine system surveillance testing.

On July 3, 2012, FitzPatrick System Engineering initiated CR-JAF-2012-03825 due to concern that the impact of the June 7 issue with 23MOV-57 on past HPCI system operability had not been adequately addressed by CR-JAF-2012-03298. Specifically, the corrective action did not verify adequate net positive suction head (NPSH) would have been available, did not evaluate the potential for reduced margin to the low suction pressure pump trip, and did not address the effect that the condition would have had on

positioning interlocks with other HPCI system valves. This CR was closed on October 5, 2012.

FitzPatrick staff performed calculations to determine the NPSH available to the HPCI pump with 23MOV-57 opened to 38 percent along with the effect it would have on margin to the low suction pressure pump trip. The results indicated that the resultant flow restriction would only have a small effect on available NPSH and would not challenge HPCI system operability.

FitzPatrick staff evaluated the impact of the 23MOV-57 issue on interlocks with other HPCI system valves, specifically the condensate storage tank (CST) suction isolation valve, 23MOV-17, and the full flow test return to the CST valves, 23MOV-21 and 23MOV-24. The HPCI pump suction is normally aligned to the CSTs and the suppression pool suction isolation valves, 23MOV-57 and 23MOV-58, are closed. If CST level reaches the low level set point, the HPCI pump suction automatically realigns to the suppression pool. The full flow test return valves are normally closed, but would be open if the HPCI system were being operated in the pressure control mode. If a CST-to-suppression pool swap over were to occur, these valves would receive an automatic close signal when either 23MOV-57 or 23MOV-58 was fully open. Therefore, since 23MOV-58 was operating normally, the issue with 23MOV-57 would not affect operation of the interlocks with 23MOV-21 and 23MOV-24.

However, to prevent a loss of suction to the HPCI pump during the swap over, the suppression pool suction isolation valves must both be fully open before a close signal is generated for 23MOV-17. Therefore, a CST-to-suppression pool swap over while the 23MOV-57 issue existed would result in the HPCI pump operating with both the CST and suppression pool suction paths open indefinitely. With no operator action, level in the CSTs would continue to lower until air was entrained in the HPCI pump suction, causing a loss of HPCI. If HPCI were not running, this condition would cause an increase in suppression pool level due to drainage from the CSTs.

FitzPatrick staff evaluated whether this condition constituted a loss of safety function for the HPCI system. They noted that control room operators would be alerted to a CST-to-suppression pool automatic swap over by ‘A’ and ‘B’ CST low level alarms, annunciators 09-3-3-07 and -08. The corresponding annunciator response procedures, ARP-09-3-3-07 and -08, direct operators to ensure the suction shift occurs: 23MOV-57 and 23MOV-58 open; and when they are full open, 23MOV-17 closes. In the case at hand, FitzPatrick staff considered that, since 23MOV-57 would not fully open, operators would attempt to open it from the control room; and when this was unsuccessful (due to the failure mechanism being a high resistance contact in the open torque switch), they would have a field operator manually open 23MOV-57 using the hand wheel. Also, control room operators could close 23MOV-17 from the control room to prevent draining the CSTs. To confirm that these actions would be taken, FitzPatrick staff interviewed a Shift Manager and conducted a tabletop exercise with a shift crew, with comparable satisfactory results. FitzPatrick staff concluded that failure of the 23MOV-57 valve interlocks would not have prevented HPCI from performing its design function, due to proceduralized mitigating operator actions.

To evaluate FitzPatrick staff's position, the inspectors utilized NUREG-1022, "Event Report Guidelines 10 CFR 50.72 and 50.73," Revision 3. The inspectors noted that Section 3.2.7, "Event or Condition that Could Have Prevented Fulfillment of a Safety Function," states that an LER is required when: 1) there is a determination that the SSC is inoperable in a required mode or other specified condition in the TS Applicability, 2) the inoperability is due to one or more personnel errors including (among others) equipment failures, and 3) no redundant equipment in the same system was operable. Concerning determination of whether an SSC is operable, it goes on to reference NRC Regulatory Issue Summary 2005-20, Revision 1. In that document, Section C.5, "Use of Temporary Manual Action in Place of Automatic Action in Support of Operability," states that the licensee should have written procedures in place and personnel should be trained on the procedures before any manual action is substituted for the loss of an automatic action.

The inspectors noted that NUMARC [Nuclear Management and Resources Council] 93-01, "Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," although not directly applicable to 10 CFR 50.73 reporting requirements, does provide an industry standard for the level of operator action that is necessary to substitute for a safety system automatic action. Concerning SSC availability during testing, it states: "SSCs out of service for testing are considered unavailable, unless . . . the function can be promptly restored either by an operator in the control room or a dedicated operator stationed locally for that purpose. Restoration actions must be contained in a written procedure, must be uncomplicated (a single action or a few simple actions), and must not require diagnosis or repair . . . The intent of this paragraph is to allow licensees to take credit for restoration actions that are virtually certain to be successful (i.e., probability nearly equal to 1) during accident conditions." The inspectors concluded that the CST low level alarm response procedures did not provide the level of detail that would be required for the stated condition, such that operator action could be credited for maintaining HPCI operability, and therefore that the condition required an LER. This issue was entered into the CAP as CR-JAF-2013-01768.

Analysis. The inspectors determined that the failure to submit an LER within 60 days in accordance with 10 CFR 50.73 was a performance deficiency that was reasonably within Entergy's ability to foresee and correct. Because the issue impacted the regulatory process, in that a safety system functional failure was not reported to the NRC within the required timeframe, thereby delaying the NRC's opportunity to review the matter, the inspectors evaluated this performance deficiency in accordance with the traditional enforcement process. Using example 6.9.d.9 from the NRC Enforcement Policy, the inspectors determined that the violation was a SL IV (more than minor concern that resulted in no or relatively inappreciable potential safety or security consequence) violation, because Entergy personnel failed to make a report required by 10 CFR 50.73 when information that the report was required had been reasonably within their ability to have identified. In accordance with IMC 0612, "Power Reactor Inspection Reports," traditional enforcement issues are not assigned cross-cutting aspects.

The inspectors also reviewed the safety significance of the HPCI system inoperability as a result of the 23MOV-57 suppression pool suction valve not opening completely. Using

IMC 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, “The Significance Determination Process [SDP] for Findings At-Power,” the inspectors determined that per Exhibit 2 of Appendix A, a detailed risk evaluation was warranted. The Senior Reactor Analyst (SRA) performed a detailed risk evaluation using SAPHIRE 8 and the FitzPatrick SPAR model. Based upon the observed valve degradation, the SRA conservatively assumed the isolation valve failed to open upon receipt of the swap-over signal from the CST to the suppression pool, and assumed an exposure period of 45 days (in accordance with IMC 0308, T/2 is used when the exact time of failure cannot be determined between the surveillance interval, 90 days). Based upon use of an E-12 truncation value, the estimated increase in core damage frequency for this condition was calculated at mid E-8, or very low safety significance (Green). The dominant core damage sequences involved an inadvertent relief valve opening and subsequent failure of high-pressure injection and failure of operators to depressurize to permit low-pressure injection.

Enforcement. 10 CFR 50.73(a)(2)(B) requires, in part, that licensees shall submit a Licensee Event Report within 60 days after the discovery of any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident.

Contrary to the above, between July 3 and October 5, 2012, Entergy personnel determined that, on June 7, 2012, FitzPatrick’s HPCI system had been in a condition that could have prevented the fulfillment of its safety function to mitigate the consequences of an accident, but this condition was not reported to the NRC in an Licensee Event Report within 60 days after discovery. Specifically, the failure of valve 23MOV-57 to fully open during an automatic swap of the HPCI suction from the CSTs to the torus would have resulted in both suction sources being aligned simultaneously, thereby creating the possibility for air entrainment and loss of the HPCI system (in the case that the HPCI system was operating) or for overfilling of the torus due to drainage from the CSTs (in the case that the HPCI system was in standby). In their evaluation of this condition, Entergy personnel inappropriately credited operator action for maintaining the HPCI system operable if this condition were to have developed. Because this SL IV violation was of very low safety significance, was not repetitive or willful, and was placed in the licensee’s CAP as CR-JAF-2013-01768, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000333/2013002-01, Failure to Submit an LER for a Condition That Could Have Prevented Fulfillment of the HPCI System Safety Function)**

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

a. Inspection Scope

The inspectors reviewed station evaluation and management of plant risk for the maintenance and emergent work activities listed below to verify that FitzPatrick staff performed the appropriate risk assessments prior to removing equipment for work. The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that FitzPatrick personnel performed risk assessments as required by 10 CFR 50.65(a)(4)

and that the assessments were accurate and complete. When FitzPatrick staff performed emergent work, the inspectors verified that operations personnel promptly assessed and managed plant risk. The inspectors also reviewed the TS requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

- One day of planned maintenance on the 'A' control rod drive system pump, one day of planned maintenance on administration building ventilation fresh air supply isolation damper 72AOD-171, which caused the technical support center ventilation system to be inoperable, 'A' RHR system quarterly surveillance test for increased frequency monitoring of pump vibrations, and testing of 'A' main station battery charger 71BC-1A which required transferring the 'A' main station battery to the temporary charger during the week of January 7, 2013
- 'B' and 'D' emergency diesel generators (EDGs) operation test from the remote shutdown panel per ST-43D, "Remote Shutdown Panel 25ASP-3 Component Operation and Isolation Verification," a two day maintenance period for 'B' standby gas treatment (SGT) system, 'B' standby liquid control system quarterly surveillance test, 'B' core spray system quarterly surveillance test, and emergent maintenance to troubleshoot the cause of a loss of Division 2 vital bus 10600 during the conduct of ST-43D during the week of January 14, 2013
- 'A' core spray system quarterly surveillance test, 'A' RHR SW quarterly surveillance test, a one day maintenance outage for the 'A' CREVAS system, a power reduction to 63 percent for a control rod sequence exchange and to perform control blade interference testing and main turbine valve testing, reactor core isolation cooling (RCIC) system quarterly surveillance test, and emergent maintenance to address main condenser tube leakage that required power to be reduced to 50 percent during the week of January 28, 2013
- A three day maintenance period for the 'A' RHR system and emergent maintenance to repair a steam leak associated with the feedwater low flow control valve, a condition which, concurrent with the 'A' RHR maintenance, placed the plant in an elevated (Orange) risk state during the week of February 4, 2013
- A three day maintenance period for the 'B' RHR system and emergent maintenance to address main condenser tube leakage that required power to be reduced to 50 percent during the week of February 18, 2013

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15 - 5 samples)

a. Inspection Scope

The inspectors reviewed operability determinations for the following degraded or non-conforming conditions:

- CR-JAF-2013-00061, concerning a level switch for the RCIC condensate storage tank to torus swap-over that initially did not function during surveillance testing on January 4, 2013
- CR-JAF-2013-00223 concerning the effect of automatic high temperature shutdowns of the ‘B’ and ‘D’ EDGs on the operability of these EDGs on January 15, 2013
- CR-JAF-2013-01110 concerning failure of the stationary auxiliary switch in the ‘D’ EDG cubicle and its effect on ‘D’ EDG operability, as well as extent-of-condition considerations for other safety class 4160-volt alternating current circuit breakers on March 2, 2013
- CR-JAF-2013-01261 concerning the impact on secondary containment operability of 20RDW-101D, reactor building equipment drainage to condenser hotwell isolation valve, being found out of position open during the plant startup on March 7, 2013
- CR-JAF-2013-01413, concerning the possible effects on RCIC system operability of seat leakage past the normally closed RCIC turbine steam inlet isolation valve, 13MOV-131, on March 15, 2013

The inspectors selected these issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the operability determinations to assess whether TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TSs and UFSAR to FitzPatrick personnel’s evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled by FitzPatrick personnel. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18 - 1 sample)

Permanent Modifications

a. Inspection Scope

The inspectors evaluated a modification to the 125-volt direct current electrical system implemented during refueling outage 20 by engineering change (EC) 26750, “Replace ‘B’ 125-Volt Station Battery, 71SB-2.” This change was implemented because the existing ‘B’ station battery was approaching the end of its life expectancy. The engineering change also increased the battery capacity from 2,416 amp-hours to 2,552 amp-hours to provide longer capability during a station blackout event. It did so using higher capacity cells which have one additional positive and negative plate, but are otherwise functionally equivalent to the old battery cells. The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the modification. In addition, the inspectors reviewed modification

documents associated with the design change, and performed a walkdown of the 'B' station battery.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19 - 7 samples)

a. Inspection Scope

The inspectors reviewed the post-maintenance tests (PMTs) for the maintenance activities listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the test procedure to verify that the procedure adequately tested the safety functions that may have been affected by the maintenance activity, that the acceptance criteria in the procedure was consistent with the information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed test data to verify that the test results adequately demonstrated restoration of the affected safety functions.

- WO 339051 to replace a failed loss of voltage relay for the EDG 'B' and 'D' tie breaker on emergency bus 10600 on January 30, 2013
- WOs 5235588, 52410034, 52410035 for 'B' CREVAS system maintenance on February 15, 2013
- WO 338722 to address incorrect dual indication for drywell nitrogen isolation valve 27SOV-145 on February 21, 2013
- WO 343533 to replace the stationary auxiliary switch on the 'D' EDG output breaker on March 6, 2013
- WO 00345375 to replace the scram solenoid pilot valves, 03SOV-117 and 03SOV-118, for hydraulic control unit 42-35 on March 21, 2013
- WO 52211924 to replace a core spray loop 'A' safety valve on March 26, 2013
- WO 52206097 to replace type CR-2810 relay 01-125-69A-1SGTA01 in the 'B' SGT system initiation logic on March 29, 2013

b. Findings

No findings were identified.

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

March 2, 2013, Planned Outage

a. Inspection Scope

On March 2, 2013, operators commenced a plant shutdown to identify and plug leaking tubes in the 6A feedwater heater. The inspectors reviewed FitzPatrick staff's implementation of forced outage plans and schedules to verify that risk, industry

experience, previous site-specific problems, and defense-in-depth were considered. The inspectors observed portions of the cooldown, heatup, and startup processes, and monitored controls associated with the following outage activities:

- Configuration management, including maintenance of defense-in-depth for the key safety functions and compliance with the applicable TS when taking equipment out of service
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and instrument error accounting
- Status and configuration of electrical systems and switchyard activities to ensure that TSs were met
- Monitoring of decay heat removal operations
- Identification and resolution of problems related to outage activities

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 - 6 samples)

a. Inspection Scope

The inspectors observed performance of surveillance tests and/or reviewed test data of selected risk-significant SSCs to assess whether test results satisfied technical specifications, the UFSAR, and station procedure requirements. The inspectors verified that test acceptance criteria were clear, tests demonstrated operational readiness and were consistent with design documentation, test instrumentation had current calibrations and the range and accuracy for the application, tests were performed as written, and applicable test prerequisites were satisfied. Upon test completion, the inspectors considered whether the test results supported that equipment was capable of performing the required safety functions. The inspectors reviewed the following surveillance tests:

- ISP-106A, "MSIV [main steam isolation valve] Closure High Steam Line Flow Response Time Test (ATTS [analog transmitter trip system])**," on January 11, 2013
- ST-9BA, "EDG A and C Full Load Test and ESW [emergency service water] Pump Operability Test," on January 28, 2013
- ST-2XA, "RHR Service Water Loop A Quarterly Operability Test (IST)," on January 29, 2013
- RAP-7.3.39, "Channel - Control Blade Interference Monitoring," on January 30, 2013
- ST-1B, "MSIV Fast Closure Test (IST)," on March 3, 2013
- ISP-16, "Drywell Floor Drain Sump Flow Loop Functional Test/Calibration*," on March 20, 2013

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04 - 1 sample)

a. Inspection Scope

NRC staff from the Office of Nuclear Security and Incident Response performed an in-office review of the latest revisions of various emergency plan implementing procedures and the emergency plan located under ADAMS accession numbers ML130390419 and ML130230023.

FitzPatrick determined that in accordance with 10 CFR 50.54(q), the changes made in the revisions resulted in no reduction in the effectiveness of the plan and that the revised plan continued to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. The NRC review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, this revision is subject to future inspection.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06 - 1 sample)

Training Observations

a. Inspection Scope

The inspectors observed a simulator exam for FitzPatrick licensed operators on March 11, 2013, which required emergency plan implementation by an operations crew. FitzPatrick staff planned for this evolution to be evaluated and included in performance indicator (PI) data regarding drill and exercise performance. The inspectors observed event classification and notification activities performed by the crew. The inspectors also attended the post-evolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that FitzPatrick evaluators noted the same issues and entered them into the CAP.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures (1 sample)

a. Inspection Scope

The inspectors sampled FitzPatrick's submittals for the Safety System Functional Failures PI for the period of April 1 through December 31, 2012. To determine the accuracy of the PI data reported during those periods, inspectors used definitions and guidance contained in Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 10 CFR 50.73." The inspectors reviewed FitzPatrick's operator narrative logs, operability assessments, condition reports, licensee event reports, and NRC integrated inspection reports to validate the accuracy of the submittals.

b. Findings

No findings were identified.

.2 Unplanned Power Changes (1 sample)

a. Inspection Scope

The inspectors reviewed FitzPatrick's submittals for the following Initiating Events Cornerstone PIs for the period of July 1 through December 31, 2012.

- Unplanned Power Changes

To determine the accuracy of the PI data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02. The inspectors reviewed FitzPatrick's operator narrative logs, CRs, and NRC integrated inspection reports to validate the accuracy of the submittals.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152 - 1 sample)

.1 Routine Review of Problem Identification and Resolution Activities

a. Inspection Scope

As required by Inspection Procedure 71152, "Problem Identification and Resolution," the inspectors routinely reviewed issues during baseline inspection activities and plant

status reviews to verify that FitzPatrick staff entered issues into the CAP at an appropriate threshold, gave adequate attention to timely corrective actions, and identified and addressed adverse trends. In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the CAP and periodically attended CR screening meetings.

b. Findings

Introduction. The inspectors identified a SL IV NCV of 10 CFR 50.73, "Licensee Event Report System," because a violation of TS 3.0.4 for a reactor mode change being made from Mode 4 to Mode 2 without satisfying the TS required conditions for alignment of the containment air dilution and SGT systems in Mode 2 was not reported to the NRC within 60 days of when it should reasonably have been discovered.

Description. FitzPatrick was shut down during the period of November 11-24, 2012, due to a fire in one of the two main transformers and subsequent transformer replacement. For the majority of this time, the plant was in Mode 4, cold shutdown. While in Mode 4, the primary containment was deinerted to support limited maintenance but was never opened for general access.

In Modes 1, 2, and 3, TS surveillance requirement (SR) 3.6.1.3.1 allows the 20-inch and 24-inch primary containment vent and purge valves to be open for inerting, deinerting, pressure control, or other reasons provided the full flow line (with valve 27MOV-120) to the SGT system is closed and one or more SGT system reactor building suction valves are open. This is to ensure that there would be no damage to the SGT filters if a loss-of-coolant accident (LOCA) were to occur with the vent and purge valves open, since excessive differential pressure is not expected with 27MOV-120 shut and one or more of the SGT system reactor building suction valves open.

The "Cold Startup Checkoff" checklist, Attachment 1 to OP-65, "Startup and Shutdown Procedure," had been started several days in advance of the anticipated startup. The Shift Manager (SM) signed that 27MOV-120 was verified closed on November 21, 2012. On November 23, 2012, operators commenced a drywell and torus vent and purge to lower drywell pressure in accordance with OP-37, "Containment Air Dilution [CAD] System," Section G.19, "Vent and Purge Operation with Drywell and Torus Depressurized and Primary Containment not Required." At the time, the plant was in Mode 4 and primary containment was not required. As allowed by Section G.19, 27MOV-120 was opened to provide a higher flow rate. The drywell and torus purge/inert supply and exhaust valves were all opened as well, as allowed by the procedure.

Through personnel error, the containment vent and purge was subsequently noted on the Control Room Supervisor shift turnover checklist as having been secured, when no such action had actually been taken. Other status tracking processes also failed to identify the actual condition of the CAD system, such that operators were no longer

tracking the system as being in operation. At 11:30 a.m. on November 24, 2012, the SM completed OP-65 Attachment 2, the “Final Prestart Checkoff,” which, in part, acknowledges completion of the Cold Startup Checkoff. Five minutes later, operators placed the reactor mode switch in “Startup/Hot Standby,” transitioning the reactor to Mode 2.

The oncoming watch section subsequently identified the noncompliant condition of the CAD system during preparations to assume the watch. The containment vent and purge was secured and 27MOV-120 was shut at 6:39 p.m. on November 24.

FitzPatrick personnel considered that this event was not reportable under 10 CFR 50.73(a)(2)(i)(B), an operation or condition prohibited by TSs, because the TS 3.6.1.3 completion time had not been exceeded. Specifically, the TS bases for SR 3.6.1.3.1 state that if a purge valve is open in violation of this SR, the valve is considered inoperable; since all of the 20-inch and 24-inch containment vent and purge valves were open, TS 3.6.1.3 Condition B, one or more penetration flow paths with two or more primary containment isolation valves (PCIVs) inoperable, was entered at 11:35 a.m. on November 24, 2012. After the 1-hour completion time for this condition had expired, TS 3.6.1.3 Condition F was entered. The required action for this condition is to be in Mode 3 within 12 hours. Since the noncompliant condition was eliminated at 6:39 p.m. (6 hours and 4 minutes into the available 12 hours), the completion time was not exceeded.

The inspectors determined that this event was not reportable under 10 CFR 50.73 as a violation of TS 3.6.1.3 consistent with the guidance in NUREG-1022, “Event Report Guidelines 10 CFR 50.72 and 50.73,” and the NRC Enforcement Manual. However, the inspectors further determined that this event was reportable under 10 CFR 50.73 as a violation of TS 3.0.4, which states, “when an LCO [limiting condition for operation] is not met, entry into a Mode or other specified condition in the Applicability shall only be made: a. When the associated Actions to be entered permit continued operation in the Mode or other specified condition in the Applicability for an unlimited period of time; b. After performance of a risk assessment addressing inoperable systems and components, considerations of the results, determination of the acceptability of entering the Mode or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or c. When an allowance is stated in the individual value, parameter, or other Specification.” Specifically: a. TS 3.6.1.3 does not permit continued operation in Mode 2 with the 20-inch and 24-inch primary containment vent and purge valves inoperable for an unlimited period of time; b. a risk assessment had not been performed; and c. no allowance is stated in TS 3.6.1.3. As immediate corrective action, FitzPatrick staff entered the issue into the CAP as CR-JAF-2013-01097.

<u>Analysis</u>. The inspectors determined that the failure to submit an LER within 60 days in accordance with 10 CFR 50.73 was a performance deficiency that was reasonably within Entergy’s ability to foresee and correct. Because the issue impacted the regulatory process, in that a violation of site TSs was not reported to the NRC within the required timeframe, thereby delaying the NRC’s opportunity to review the matter, the inspectors evaluated this performance deficiency in accordance with the traditional enforcement process. Using example 6.9.d.9 from the NRC Enforcement Policy, the inspectors

determined that the violation was a SL IV (more than minor concern that resulted in no or relatively inappreciable potential safety or security consequence) violation, because Entergy personnel failed to make a report required by 10 CFR 50.73 when information that the report was required had been reasonably within their ability to have identified. In accordance with IMC 0612, traditional enforcement issues are not assigned cross-cutting aspects.

The inspectors also reviewed the safety significance of the CAD system inoperability for approximately seven hours on November 24, 2012. In accordance with IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," the inspectors determined that the condition constituted a Type B finding (related to a degraded condition that has potentially important implications for the integrity of the containment, without affecting the likelihood of core damage). Per Table 6.2, "Phase 2 Risk Significance - Type B Findings at Full Power," for failure of Mark 1 containment systems/components critical to suppression pool integrity/scrubbing (vacuum breakers or other bypass mechanisms) with an exposure time of less than three days, the condition was indicated to have been of very low safety significance.

<u>Enforcement</u>. 10 CFR 50.73(a)(2)(B) requires, in part, that licensees shall submit an LER within 60 days after the discovery of any operation or condition which was prohibited by the plant's TS (with certain exceptions). FitzPatrick renewed operating license DPR-59, Condition 2.C.2, states that Entergy personnel shall operate the facility in accordance with the TS. FitzPatrick TS SR 3.6.1.3.1 allows, in Modes 1, 2, and 3, the 20-inch and 24-inch primary containment vent and purge valves to be open for pressure control, provided the full-flow line to the SGT system is closed and one or more SGT system reactor building suction valves are open. FitzPatrick TS SR 3.6.1.3.1 bases indicate that the full-flow line to the SGT system is closed using valve 27MOV-120; these bases further indicate that if a purge valve is open in violation of this SR, the valve is considered inoperable. FitzPatrick TS 3.0.4 states, in part, "when an LCO is not met, entry into a Mode or other specified condition in the Applicability shall only be made: a. When the associated Actions to be entered permit continued operation in the Mode or other specified condition in the Applicability for an unlimited period of time; b. After performance of a risk assessment addressing inoperable systems and components, considerations of the results, determination of the acceptability of entering the Mode or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications; or c. When an allowance is stated in the individual value, parameter, or other Specification."

Contrary to the above, at 11:35 a.m. on November 24, 2012, with 27MOV-120 open and the 20-inch and 24-inch primary containment vent and purge valves open for pressure control, operators repositioned the reactor mode switch from "Shutdown" to "Startup/Hot Standby," thereby transitioning the reactor from Mode 4 to Mode 2, contrary to the requirements of TS 3.0.4, in that: a. TS 3.6.1.3 does not allow continued operation in Mode 2 with the 20-inch and 24-inch primary containment vent and purge valves inoperable for an unlimited period of time; b. Entergy staff did not first perform a risk assessment addressing the inoperable systems and components, considerations of the results, determination of the acceptability of entering Mode 2, and establishment of risk

management actions, if appropriate; and c. no allowance is stated in TS 3.6.1.3. Because this SL IV violation was of very low safety significance, was not repetitive or willful, and was placed in the licensee's CAP as CR-JAF-2013-01097, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000333/2013002-02, Failure to Submit an LER for a Condition Prohibited by TS 3.0.4)**

.2 Annual Sample: 'C' Condensate Booster Pump Failure

a. Inspection Scope

The inspectors performed an in-depth review of FitzPatrick staff's root cause analysis and corrective actions associated with CR-JAF-2012-00329 concerning failure of the 'C' condensate booster pump. Specifically, while operating at 100 percent power on January 17, 2012, operators observed the 'C' condensate booster pump minimum flow valve open unexpectedly and the running current for all three condensate booster pumps increase to above the maximum allowed values. Operators reduced power to within the capacity of two condensate booster pumps (65 percent) and secured the 'C' pump. Pump disassembly revealed that the diaphragm that separates the suction and discharge of the single stage pump was loose in the pump casing and that all 24 fasteners that held it to the pump casing were either broken or had backed out. The dislodged diaphragm caused damage to the impeller, wear rings, and pump casing.

The inspectors assessed FitzPatrick staff's problem identification threshold, cause analyses, extent-of-condition reviews, compensatory actions, and the prioritization and timeliness of FitzPatrick's corrective actions to determine whether FitzPatrick staff was appropriately identifying, characterizing, and correcting problems associated with this issue and whether the planned or completed corrective actions were appropriate. The inspectors compared the actions taken to the requirements of FitzPatrick's CAP and 10 CFR 50, Appendix B.

b. Findings and Observations

No findings were identified.

FitzPatrick staff's investigation found that the heads for 17 of the 24 diaphragm fasteners (cap screws) were missing. The remaining seven had backed out of the pump casing; two were found at the bottom of the casing, and the others, along with all 24 bolt head locking strips (designed to keep the cap screws from coming out), were missing. Laboratory analysis established that the cap screws were made of 410-stainless steel (SS) and identified high cycle fatigue cracks in the threaded area. FitzPatrick staff determined that the cap screw failures were the result of excessive torque that had been used on the cap screws during installation of the pump diaphragm, which produced high stress over a short length of the bolt. This, along with the susceptible material and suitable environmental conditions, supported intergranular stress corrosion cracking (IGSCC) which led to cap screw failure. These failures led to diaphragm movement which caused a high-cycle fatigue stress on the remaining fasteners.

FitzPatrick staff identified a missed opportunity to have identified susceptibility to this failure mode through the use of previous industry operating experience (OE). Multiple equipment failures had been reported that were attributed to IGSCC-induced failure of 410 SS components. This OE had been individually dispositioned for specific components involved, and no effort had been made to identify other applications of 410 SS in the plant. FitzPatrick staff considered that improvements in the OE program since the 1980-1990 timeframe increased the probability that such overlap would be identified in the present. Specifically, current OE applicability criteria include consideration of whether similar equipment or components are used at the station, although not necessarily in the same application.

FitzPatrick staff also identified that a change had been made to the vendor manual for the condensate booster pumps in 1988 that changed the material for the diaphragm fasteners from 410 SS to 17-4PH SS. This change was apparently not communicated to FitzPatrick because the station's vendor manual control program includes only safety related components, and the condensate booster pumps are not safety related.

Preventive maintenance for the condensate booster pumps is based on predictive measures such as vibration monitoring and oil analysis. The vendor manual states that pump overhaul should be performed approximately every 12 years of continuous operation. Although this recommendation was not incorporated, FitzPatrick staff concluded that the preventive maintenance strategy was consistent with Entergy and industry preventive maintenance template requirements.

As corrective action, the 'C' condensate booster pump was rebuilt with a new rotating assembly, using 17-4 PH SS cap screws for diaphragm installation. During refueling outage 20, the 'A' and 'B' condensate booster pumps were opened and the diaphragm cap screws were replaced with 17-4 PH SS cap screws. A loose parts analysis was performed which concluded that, if the lost parts were present in the reactor vessel, they did not present a concern for flow blockage to the fuel bundles, interference with the scram function, or corrosion / adverse chemical reaction with other reactor materials.

The inspectors determined that FitzPatrick staff's cause determination for failure of the 'C' condensate booster pump was reasonable. The inspectors noted that there had been multiple avenues by which the condition could have been identified prior to failure, but had not, because the condensate booster pumps are not safety related equipment. As such, the failure to identify the susceptibility to failure created by use of the 410 SS fasteners did not constitute a violation of regulatory requirements. However, the inspectors considered that, in light of the inability of predictive monitoring activities to have identified incipient failure in this case, consideration of a pump internals inspection at the frequency of the vendor-recommended pump overhaul may be appropriate. The inspectors determined that FitzPatrick staff's subsequent corrective action to replace the subject fasteners in the remaining condensate booster pumps and to evaluate the use of 410 SS fasteners in the condensate and feedwater systems was appropriate.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 - 3 samples)

.1 (Closed) LER 05000333/2012-004-00 and -01: Control Rods Inoperable While Entering Plant Outage

The FitzPatrick reactor is potentially susceptible to fuel channel - control blade interference. Under certain conditions (i.e., reduced reactor pressure), this could result in affected control rods not fully inserting into the core during a scram. The extent of the condition is primarily a function of the amount of neutron exposure that any given fuel channel has undergone. Based on the reactor core design for a given operating cycle (i.e., the placement of new and previously used fuel bundles in the core), the fuel vendor calculates a parameter for each control rod, called the cell friction metric (CFM), to identify which control rods have a higher probability to develop fuel channel - control blade interference. The values of CFMs increase over the two year core operating cycle. At the time of shutdown for refueling outage 20, the fuel vendor's guidance was that friction testing of high CFM control rods should be performed within 14 days of operation at reduced reactor pressure; if this was not done, these control rods should be fully inserted into the core prior to operation at reduced reactor pressure or be declared inoperable.

FitzPatrick staff did not perform control rod friction testing within 14 days of the shutdown for refueling outage 20. During this shutdown, due to reactor cool down primarily by ambient losses, reactor pressure dropped below the CFM threshold value for control rod operability before all high CFM control rods had been fully inserted into the core. As a result, FitzPatrick staff declared these 52 control rods inoperable. TS 3.1.3, "Control Rod Operability," Condition E, require that with nine or more control rods inoperable, the reactor be in Mode 3 within 12 hours. All control rods were fully inserted within approximately 2.5 hours of the CFM reduced pressure threshold having been reached. FitzPatrick staff experienced no issues with control rod movement during the shutdown.

The inspectors reviewed this LER and its revision (made to correct several technical details) and identified no violations of regulatory requirements. FitzPatrick staff's action to declare the 52 subject control rods inoperable was consistent with vendor guidance and the applicable TS required actions were performed within the allowed completion time. This LER is closed.

.2 (Closed) LER 05000333/2012-005-00: Transformer Installation Error Causes Loss of Off-Site Power

During refueling outage 20, reserve station service transformers (RSSTs) 71-T2 and 71-T3 were replaced. The RSSTs transform offsite 115 kilovolt power to 4160 volt to supply station loads when the plant is shut down. On October 5, 2012, the new RSSTs were placed in service. Later that day, a maintenance activity applied the first significant load to the RSSTs and caused a trip of 71-T3 which resulted in a loss of offsite power. The cause of the transformer trip was an unnecessary actuation of a phase differential

protection relay due to current transformer shorting bars not having been removed as specified in the RSST installation procedure. The four EDGs automatically started and reenergized the emergency buses, although one of the EDG output breakers did not close.

The enforcement aspects of this issue were addressed in NRC Integrated Inspection Report 05000333/2012005. The inspectors did not identify any new issues during the review of the LER. This LER is closed.

.3 (Closed) LER 05000333/2012-006-00: Loss of Safety Function for RHR Shutdown Cooling Isolation Function Due to Reference Leg Leakage During Maintenance

On October 16, 2012, with the reactor in Mode 4 during refueling outage 20, operators noted indication of rising reactor vessel (RV) water level on two channels of RV level indication. Actual RV level, as indicated by other instrumentation, remained constant. Operators determined that the affected RV level instruments used the 3A reference leg, and that the rising level indications had started when the backfill system for the 3A reference leg was being removed from service for maintenance. Operators further determined that the RV level instrument that provided the low level isolation function for the in-service ‘A’ loop of RHR shutdown cooling also used the 3A reference leg and was, therefore, unreliable. The ‘A’ RHR shutdown cooling low level isolation function was declared inoperable and, as required by TS 3.3.6.1.J, operators immediately initiated action to restore the function by transferring shutdown cooling to ‘B’ RHR.

FitzPatrick staff determined that the cause of the level indication issue was leakage past isolation valves that had been closed to support backfill system maintenance. This allowed the 3A reference leg to drain through the downstream bleed valve which had been opened to maintain the backfill system vented. The inspectors reviewed this LER and identified no violations of regulatory requirements. This LER is closed.

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000333/2011003-02: Source Range Monitor Operability Requirements during Core Alterations

a. Inspection Scope

In the second quarter of 2011, the inspectors reviewed refueling operations during the 2010 refueling outage as documented in NRC Integrated Inspection Report 05000333/2011003. During core alterations (movement of fuel or control rods within the reactor vessel), TS SR 3.3.1.2.2 requires an operable source range [neutron] monitor (SRM) to be located in the core quadrant where core alterations are being performed. However, the inspectors noted that FitzPatrick staff considered that refueling operations could proceed in any location with any single SRM inoperable based on a definition of “core quadrant” that had been developed and adopted by the station in 2004. The inspectors reviewed the issue of SRM operability requirements during refueling operations to determine if FitzPatrick’s core quadrant definition was consistent with the requirements of TSs. The inspectors concluded that a determination would be required

by the NRC Office of Nuclear Reactor Regulation (NRR) as to whether there was sufficient data and analyses to support the rotated core quadrants approach to TS requirements for SRM operability during core alterations.

As a result, the inspectors initiated task interface agreement (TIA) 2012-01, "James A. FitzPatrick Nuclear Power Plant Definition of Core Quadrant." The NRR staff response, a memorandum dated November 8, 2012, titled, "Final Response to Task Interface Agreement 2012-01, James A. FitzPatrick Nuclear Power Plant Definition of Core Quadrant," concluded: (a) Entergy's definition of core quadrant represented a change to the plant's TSs that required prior review and approval by the NRC staff as specified in 10 CFR 50.59 before it could be implemented and was, therefore, not acceptable; and (b) implementing Entergy's definition of core quadrant did not satisfy SRM TS operability requirements during core alterations. This URI is closed.

b. Findings

Introduction. The inspectors identified a SL IV NCV of 10 CFR 50.59, "Changes, Tests, and Experiments," because Entergy personnel implemented a change to the TS definition of core quadrant without prior review and approval by the NRC staff in accordance with 10 CFR 50.59(c)(1)(i). Specifically, Entergy staff changed the definition of core quadrant in Revision 5 of Reactor Analyst procedure RAP-7.1.04C, "Neutron Instrumentation Monitoring During In-Core Fuel Handling," which allowed operators to interpret what constitute core quadrant boundaries, such that core alterations could be performed anywhere in the core provided any three SRMs were operable.

Description. On September 15, 2010, FitzPatrick personnel commenced reactor refueling operations. At the time, one of the four installed SRMs was inoperable (SRM 'A'). During core alterations (movement of fuel or control rods within the reactor vessel), TS SR 3.3.1.2.2 requires that an operable SRM be located in the core quadrant where core alterations are being performed. The inspectors questioned how fuel movements were being controlled such that no movements would be performed in the core quadrant that contained SRM 'A'. The control room operators responded that, in accordance with procedure OSP-66.001, "Management of Refueling Activities," Revision 1, refueling operations could proceed in any core location with any single SRM out of service. Entergy personnel indicated this conclusion was based on a definition of "core quadrant" that had been developed and adopted by the station in 2004.

The FitzPatrick reactor core consists of 560 fuel assemblies, arranged symmetrically in an octagonal configuration. Due to this symmetry, the core can be divided into four equal quadrants, using two perpendicular axes (000-180° and 090-270°) that cross at the geometric center of the core. The reactor core also contains four installed SRMs with one in each of the quadrants as described above. The inspectors considered that this was the basis of the TS 3.3.1.2.2 requirements.

However, Entergy personnel had developed a definition of "core quadrant" with placement of the axes that was based on the SRM locations. This orientation results in quadrant axes that are rotated approximately 18 degrees clockwise from the arrangement that was described above. This results in quadrant boundaries that bisect

individual fuel assemblies; Entergy personnel considered that such fuel assemblies could be considered to reside in either of the adjacent quadrants. Entergy personnel used this concept to establish two quadrant boundaries, one rotated clockwise by 16° and the other rotated clockwise by 20°, such that the SRM would partially reside in both quadrants using either boundary. This created a set of fuel assemblies along a quadrant boundary that could be considered to be a part of either of the adjacent quadrants. Entergy personnel determined that, by selecting the appropriate boundary in the case that a single SRM is inoperable, this quadrant arrangement supports the requirement of TS 3.3.1.2.2, while allowing movement of fuel anywhere in the core.

FitzPatrick originally adopted the rotated quadrant definition of core quadrant in procedure RAP-7.1.04C, "Neutron Instrumentation Monitoring during In-Core Fuel Handling," Revision 5, effective September 28, 2004. During review of this revision, FitzPatrick personnel had concluded that, in accordance with procedure ENN-LI-100, "Process Applicability Determination," a 10 CFR 50.59 evaluation was not required. The inspectors concluded that the station's analysis did not provide an evaluation that adequately supported Entergy's interpretation of what constituted a "core quadrant boundary" such that core alterations could be performed anywhere in the core provided any three SRMs were operable.

After discussions with NRR personnel, the inspectors initiated TIA 2012-01 to determine whether there was sufficient data and analyses to support the rotated core quadrants approach to TS requirements for SRM operability during core alterations. In response to the TIA, NRR staff concluded Entergy's definition of core quadrant represented a change to the plant's TSs that required prior review and approval by the NRC staff as specified in 10 CFR 50.59 before it could be implemented and was therefore not acceptable, and implementing Entergy's definition of core quadrant did not satisfy SRM TS operability requirements during core alterations.

As immediate corrective action to the TIA final response, FitzPatrick staff withdrew RAP-7.1.04C pending revision of the core quadrant definition. The inspectors verified that TS 3.3.1.2.2 had been satisfied during all core alterations that were performed during the 2010 and 2012 refueling outages, using the standard definition of a core quadrant. Entergy entered this issue into the CAP as CR-HQN-2013-00034.

Analysis. The inspectors determined that Entergy staff's implementation of a redefinition of core quadrant prior to its review and approval by the NRC staff as specified in 10 CFR 50.59(c)(1)(i) was a performance deficiency that was reasonably within Entergy staff's ability to foresee and correct. Because this was a violation of 10 CFR 50.59, it was considered to be a violation that potentially impedes or impacts the regulatory process. Therefore, this violation was characterized using the traditional enforcement process. The violation was determined to be more than minor in accordance with the NRC Enforcement Manual, Section 7.3.E.6, because there was a reasonable likelihood that the change to the definition of what constituted a "core quadrant boundary" would require Commission review and approval prior to implementation. Additionally, the inspectors noted that, in accordance with IMC 0612, Appendix B, "Issue Screening," the underlying performance deficiency would screen as more than minor because, if left uncorrected, it would have the potential to lead to a more significant safety concern.

Specifically, potentially inadequate SRM coverage during refueling operations could affect the TS bases function to provide early indication of unexpected subcritical multiplication that could be indicative of an approach to criticality.

Although the SDP is not designed to assess traditional enforcement violations that potentially impact the regulatory process, NRC Enforcement Manual Section 7.3 provides guidance to assess 10 CFR 50.59 violations through the SDP. In this case, the inspectors determined the violation could be evaluated using the SDP in accordance with IMC 0609 Appendix G, “Shutdown Operations Significance Determination Process,” Checklist 7, “BWR Refueling Operation with RCS Level Greater Than 23 Feet.” The finding affected the Reactivity Guidelines attribute that assumes existing core alteration TS are being met. Since this attribute does not require quantitative assessment, the finding was screened as Green, in accordance with Section 3.3, “Mitigation Capability.” In accordance with the NRC Enforcement Policy, Section 6.1.d.2, this violation was categorized as SL IV because the issue was evaluated by the SDP as having very low safety significance (Green).

The finding did not have a cross-cutting aspect because the performance deficiency did not occur within the past three years and therefore was not reflective of present performance.

Enforcement. 10 CFR 50.59.59, “Changes, Tests, and Experiments,” Section (c)(1) states, in part, “A licensee may . . . make changes in the procedures as described in the final safety analysis report (as updated) . . . without obtaining a license amendment pursuant to paragraph 50.90 only if: (i) A change to the technical specifications incorporated in the license is not required . . .”

Contrary to the above, on September 28, 2004, Entergy personnel implemented a change to the TS definition of core quadrant without prior review and approval by the NRC staff, through implementation of Revision 5 to procedure RAP-7.1.04C, “Neutron Instrumentation Monitoring during In-Core Fuel Handling.” Specifically, RAP-7.1.04C established operational guidance that re-defined the TS 3.3.1.2.2 bases for what constituted a core quadrant boundary at FitzPatrick without sufficient data and analyses to support that this approach was consistent with the current licensing basis as established by the UFSAR and TS requirements. In accordance with the NRC Enforcement Policy Section 6.1, this violation was classified as a SL IV violation because the underlying technical issue was evaluated as having very low safety significance by the SDP. Because this violation was of very low safety significance, was not repetitive or willful, and was entered into Entergy’s CAP as CR-HQN-2013-00034, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000333/2013002-03, Failure to Obtain NRC Staff Review and Approval Prior to Changing the TS Definition of a Core Quadrant)**

.2 Follow-Up on Alternative Dispute Resolution Confirmatory Order (92702)

Background

NRC Confirmatory Order (CO) EA-10-090 / EA-10-248 / EA-11-106 was issued to Entergy on January 26, 2012, to confirm commitments made to the NRC during a mediation session held on November 9, 2011. The mediation session was conducted at the request of Entergy, in response to the NRC's offer of Alternative Dispute Resolution (ADR) regarding apparent violations identified by the NRC. As part of the settled agreement for the CO, Entergy agreed to take additional actions to ensure that the effectiveness of corrective actions previously taken for the issues identified are extended to the Entergy fleet and to the industry.

The objective of this inspection was to verify the actions required of Entergy, as documented in the CO, have been implemented. The inspectors used guidance contained in Inspection Procedure 92702 to conduct the reviews.

Other aspects of the CO were previously inspected and the results documented in NRC Integrated Inspection Reports 05000333/2012003 (ML1220A278) and 05000333/2012005 (ML1308A174).

.A (1) Inspection Scope

CO Section V, Paragraph 4.E: Within 360 days of the date of the CO, Entergy will deliver a presentation to Regional Utility Groups (RUGs) or Plant Managers at Regions I, II, III, and IV which will discuss the events that led to the CO, the lessons learned, and actions taken. If any of the RUGs or Plant Managers meetings will not support completion of this action, Entergy will contact the Regional Administrator, Region I, to provide notice and to resolve the scheduling issue.

(2) Findings and Observations

No findings were identified. As discussed in NRC Integrated Inspection Report 05000333/2012003, Section 4OA5.2, Entergy initiated CR-JAF-2012-00966 to address actions to be taken in response to the CO. As documented in corrective actions (CAs) 48 through 51 of this CR, Entergy developed a presentation and delivered it to each of the Regional Utility Groups in Regions I, II, III, and IV. The last presentation was completed in August 2012, which met the requirement to complete the presentations within 360 days of the date of the CO.

The inspectors conducted a review of the presentation slides and speaker's notes at Entergy offices in White Plains, NY, and determined the presentation provided a summary of the events leading to the CO, lessons learned, and actions taken. This closes item 4.E.

.B (1) Inspection Scope

CO Section V, Paragraph 4.F: Within 360 days of the date of the CO, Entergy will develop an assessment plan and conduct an assessment consistent with that plan of the Radiation Protection (RP) departments at the nine Entergy commercial nuclear power plants. That assessment will review the rigor with which members of the RP departments perform and document routine department activities. If those assessments identify performance or documentation issues, Entergy will enter those issues into its Corrective Actions Programs. Prior to the conduct of the first assessment, Entergy will make the assessment plan available to the NRC for review.

(2) Findings and Observations

No findings were identified. As discussed in NRC Integrated Inspection Report 05000333/2012003, Section 4OA5.2, Entergy initiated CR-JAF-2012-00966 to address actions to be taken in response to the CO. As documented in CAs 52 through 63 of this CR, Entergy developed an assessment plan, notified the NRC by letter dated June 21, 2012, that the plan was available for NRC review, and conducted assessments at each of their nine commercial nuclear power plants. All the assessments were completed within 360 days of the date of the CO, as required.

The inspectors conducted a review of the assessment plan at Entergy offices in White Plains, NY, and determined plan development was consistent with guidance in Entergy procedure, EN-LI-104, "Self Assessment and Benchmark Process," Revision 8, and that the scope of the assessment plan was sufficient to accomplish the stated purpose of reviewing the rigor with which members of the RP departments perform and document routine department activities. The scope included reviews of RP documentation for radiation surveys, dosimetry, source checks, respiratory protection inspection records, and radioactive waste shipments, condition reports documenting RP procedure non-compliances or unsatisfactory documentation events, Quality Assurance audits or assessments of RP activities and processes completed within the last 12 months, RP self assessment reports, and documentation of respiratory fit testing. Also included were interviews with personnel from various levels, functions, and shifts the RP department at each site, and field observations of RP activities.

The assessment plan indicated the assessments would be conducted by a four person team consisting of: "the Team Leader, an RP department member for the site being assessed, and two RP department members from other Entergy sites, on a rotational basis." Also, the "Team Leader is intended to be the same individual at all sites to support consistent implementation of the assessment methods." The inspectors observed the team leader was not the same individual for all assessments. However, the inspectors noted at least one team member for eight of the assessments had previously participated in these assessments. This, coupled with the assessment plan, provided reasonable consistency in the assessment methods used at each of the sites.

The inspectors also selected for review the results of self assessments performed at three of the nine Entergy commercial nuclear power plants, specifically, Palisades, Indian Point, and River Bend. The inspectors determined the individual site

assessments were performed in accordance with the assessment plan described above and issues identified during the assessments were entered into each site's corrective action program as appropriate. Additionally, the inspectors reviewed a sampling of condition reports resulting from the assessments at the remaining six Entergy commercial nuclear power plants and determined the types of issues identified were similar to those of the three assessments that received in depth review. The issues were administrative in nature and included items such as missing or illegible signatures, survey data not entered onto survey maps, inconsistent terminology, and supervisory reviews not completed within Entergy timeliness expectations. This closes item 4.F.

.3 National Nuclear Accrediting Board Report Review

a. Inspection Scope

The inspectors reviewed the final report for the National Nuclear Accrediting Board team evaluation of the FitzPatrick maintenance, chemistry, radiological protection, and engineering technical training programs conducted in May 2012. The inspectors evaluated this report to ensure that NRC perspectives of FitzPatrick performance were consistent with any issues identified during the assessment. The inspectors also reviewed this report to determine whether the team identified any significant safety issues that required further NRC follow-up.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

Exit Meeting

On April 12, 2013, the inspectors presented the inspection results to Mr. Michael Colomb, Site Vice President, and other members of the FitzPatrick staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Colomb, Site Vice President
C. Adner, Manager, Licensing
C. Brown, Manager, Quality Assurance, Entergy
B. Finn, Director, Nuclear Safety Assurance
T. Hunt, Manager, Corrective Action and Assessment
K. Irving, Manager, System Engineering
D. Poulin, Manager, Operations
T. Redfearn, Manager, Security
M. Reno, Manager, Maintenance
B. Sullivan, General Manager, Plant Operations
D. Wallace, Director, Engineering
E. Wolfe, Manager, Radiation Protection

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened/Closed

05000333/2013002-01	NCV	Failure to Submit an LER for a for a Condition That Could Have Prevented Fulfillment of the HPCI System Safety Function (Section 1R12)
05000333/2013002-02	NCV	Failure to Submit an LER for a Condition Prohibited by TS 3.0.4 (Section 4OA2)
05000333/2013002-03	NCV	Failure to Obtain NRC Staff Review and Approval Prior to Changing the TS Definition of a Core Quadrant (Section 4OA5)

Closed

05000333/2012-004-00 / -01	LER	Control Rods Inoperable While Entering Plant Outage (Section 4OA3)
05000333/2012-005-00	LER	Transformer Installation Error Causes Loss of Off-Site Power (Section 4OA3)

05000333/2012-006-00	LER	Loss of Safety Function for RHR Shutdown Cooling Isolation Function Due to Reference Leg Leakage during Maintenance (Section 4OA3)
05000333/2011003-02	URI	Source Range Monitor Operability Requirements during Core Alterations (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
AOP-13, "High Winds, Hurricanes, and Tornadoes," Revision 19
OP-4, "Circulating Water System," Revision 72

Section 1R04: Equipment Alignment

Procedures
OP-13, "Residual Heat Removal System," Revision 95
OP-13A, "Low-Pressure Coolant Injection," Revision 16
OP-14, "Core Spray System," Revision 34
OP-15, "High-Pressure Coolant Injection," Revision 59
OP-55B, "Control Room Ventilation and Cooling," Revision 35
ST-4B, "HPCI Monthly Operability Test," Revision 59
ST-4N, "HPCI Quick Start, Inservice, and Transient Monitoring Test (IST)," Revision 63

Documents
System Health Report for HPCI System for Third and Fourth Quarters 2012

Drawings
FM-23A, "Flow Diagram Core Spray System 14," Revision 49
FM-25A, "Flow Diagram High-Pressure Coolant Injection System 23," Revision 74

Condition Reports

CR-JAF-2011-00553	CR-JAF-2012-02194	CR-JAF-2012-07511
CR-JAF-2011-01351	CR-JAF-2012-02984	CR-JAF-2012-07838
CR-JAF-2011-02226	CR-JAF-2012-03357	CR-JAF-2012-07890
CR-JAF-2011-02769	CR-JAF-2012-03407	CR-JAF-2012-07990
CR-JAF-2011-04023	CR-JAF-2012-05573	CR-JAF-2013-00779
CR-JAF-2012-01465	CR-JAF-2012-07308	

Section 1R05: Fire Protection

Procedures
PFP-PWR01, "East Cable Tunnel / Elevation 258 feet, Fire Area/Zone II/CT-2," Revision 3

PFP-PWR12, “Relay Room / Elevation 286 feet, Fire Area/Zone VII/RR-1,” Revision 4
PFP-PWR20, “Reactor Building - East / Elevation 272 feet, Fire Area/Zone IX/RB-1A,” Revision 4
PFP-PWR21, “Reactor Building - West / Elevation 272 feet, Fire Area/Zone X/RB-1B,” Revision 5
PFP-PWR46, “Turbine Building - South / Elevation 272 feet, Fire Area/Zone IE/TB-1, OR-2,” Revision 4

Section 1R06: Flood Protection Measures

Documents
JAF-RPT-MULTI-2107, “James A. FitzPatrick Nuclear Power Plant Individual Plant Examination,” Revision 1

Section 1R11: Licensed Operator Requalification Program

Procedure
OP-65, “Startup and Shutdown Procedure,” Revision 114

Section 1R12: Maintenance Effectiveness

Documents
DBD-023, “Design Basis Document for the High-Pressure Coolant Injection System,” Revision 12
ENN-MS-S-009-JAF, “James A. FitzPatrick Safety System Function Sheets,” Revision 1
JAF-RPT-CAD-02312, “Maintenance Rule Basis Document System 27 Primary Containment Atmosphere Control and Dilution,” Revision 11
JAF-RPT-HPCI-02289, “Maintenance Rule Basis Document System 23 High-Pressure Coolant Injection System,” Revision 7
JAF-RPT-RBC-02295, “Maintenance Rule Basis Document System 066 Reactor Building Ventilation System,” Revision 3
JENG-APL-12-005, “James A. FitzPatrick Reactor Building Ventilation System (a)(1) Action Plan,” Revision 0
JENG-APL-12-006, “James A. FitzPatrick High-Pressure Coolant Injection Maintenance Rule (a)(1) Action Plan,” Revision 0
JENG-11-0041, “James A. FitzPatrick Containment Air Dilution Maintenance Rule (a)(1) Action Plan,” Revision 3
JENG-12-0035, “Reactor Building Ventilation System Maintenance Rule (a)(1) Evaluation”
Maintenance Rule Quarterly Report for Fourth Quarter 2012
System Health Report for CAD System for First through Fourth Quarters 2012
System Health Report for HPCI System for First through Fourth Quarters 2012
System Health Report for Reactor Building Ventilation for First through Fourth Quarters 2012

Procedures
EN-DC-203, “Maintenance Rule Program,” Revision 1
EN-DC-204, “Maintenance Rule Scope and Basis,” Revision 2
EN-DC-205, “Maintenance Rule Monitoring,” Revision 4
EN-DC-206, “Maintenance Rule (a)(1) Process,” Revision 2
OP-15, “High-Pressure Coolant Injection,” Revision 59

Condition Reports

CR-JAF-2009-02509	CR-JAF-2010-02019	CR-JAF-2010-03903
CR-JAF-2010-01712	CR-JAF-2010-03239	CR-JAF-2010-03920

CR-JAF-2010-08030	CR-JAF-2012-00881	CR-JAF-2012-04132
CR-JAF-2010-08031	CR-JAF-2012-00920	CR-JAF-2012-04479
CR-JAF-2010-08327	CR-JAF-2012-01913	CR-JAF-2012-04591
CR-JAF-2011-01049	CR-JAF-2012-02217	CR-JAF-2012-04699
CR-JAF-2011-01351	CR-JAF-2012-02897	CR-JAF-2012-04853
CR-JAF-2011-01595	CR-JAF-2012-02984	CR-JAF-2012-04994
CR-JAF-2011-01973	CR-JAF-2012-03015	CR-JAF-2012-05020
CR-JAF-2011-02000	CR-JAF-2012-03275	CR-JAF-2012-05064
CR-JAF-2011-02405	CR-JAF-2012-03298	CR-JAF-2012-05068
CR-JAF-2011-02512	CR-JAF-2012-03357	CR-JAF-2012-05740
CR-JAF-2011-02584	CR-JAF-2012-03359	CR-JAF-2012-05916
CR-JAF-2011-02923	CR-JAF-2012-03363	CR-JAF-2012-07153
CR-JAF-2011-03964	CR-JAF-2012-03380	CR-JAF-2012-07511
CR-JAF-2011-05067	CR-JAF-2012-03565	CR-JAF-2012-07515
CR-JAF-2011-06586	CR-JAF-2012-03779	CR-JAF-2012-08125
CR-JAF-2012-00428	CR-JAF-2012-03825	CR-JAF-2013-01674

Work Orders

WO 00321531	WO 52360656
WO 00321533	WO 52452866

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures

AP-10.10, "On-Line Risk Assessment," Revision 8
EN-OP-119, "Protected Equipment Postings," Revision 5
EN-WM-104, "On-Line Risk Assessment," Revision 7
ST-43D, "Remote Shutdown Panel 25ASP-3 Component Operation and Isolation Verification," Revision 18

Section 1R15: Operability Evaluations

Procedures

AP-12.08, "LCO Tracking and Safety Function Determination Program," Revision 15
EN-OP-104, "Operability Determination Process," Revision 6

Condition Reports

CR-JAF-2013-00061	CR-JAF-2013-01110	CR-JAF-2013-01413
CR-JAF-2013-00223	CR-JAF-2013-01261	

Section 1R18: Plant Modifications

Documents

DBD-071 Tab III, "Design Basis Document for the Electrical Distribution System 125 V and 24 V DC Power Systems," Revision 3
EC 26750, "Replace 'B' 125-Volt Station Battery, 71SB-2"

Section 1R19: Post-Maintenance Testing

Procedures

ISP-90, “4KV Emergency Power (Buses 10500 and 10600) Undervoltage Relay (Loss of Voltage) Calibration,” Revision 14
MP-54.02, “4.16KV Bus and Metal-Clad Switchgear Maintenance,” Revision 14
OP-14, “Core Spray System,” Revision 34
RAP-7.4.01, “Control Rod Scram Time Evaluation,” Revision 26
ST-1C, “Primary Containment Isolation Valve Exercise Test (IST),” Revision 54
ST-18BB, “CREVAS A Operability Test,” Revision 2
ST-34B, “Reactor Building Exhaust Rad Monitors Instrument/Logic System Functional and Simulated Automatic Actuation Test,” Revision 41
ST-39B-X57C, “Type C Leak Test of Instrument Nitrogen Supply Line Valves (IST),” Revision 7
ST-41D, “Remote Valve Position Indication Verification (IST),” Revision 18

Work Orders

WO 00345375	WO 00343533	WO 52410035
WO 00338722	WO 05235588	WO 52206097
WO 00339051	WO 52410034	WO 52211924

Section 1R20: Refueling and Other Outage Activities

Procedures
OP-13D, “RHR - Shutdown Cooling,” Revision 24
OP-65, “Startup and Shutdown Procedure,” Revision 115
ST-26J, “Heatup and Cooldown Temperature Checks,” Revision 23

Section 1R22: Surveillance Testing

Procedures
ISP-16, “Drywell Floor Drain Sump Flow Loop Functional Test /Calibration*,” Revision 38
ISP-106A, “MSIV Closure High Steam Line Flow Response Time Test,” Revision 1
RAP-7.3.39, “Channel-Control Blade Interference Monitoring,” Revision 3
ST-1B, “MSIV Fast Closure Test (IST),” Revision 25
ST-2XA, “RHR SW Loop ‘A’ Quarterly Operability Test,” Revision 13
ST-9BA, “EDG ‘A’ and ‘C’ Full Load Test and ESW Pump Operability Test,” Revision 14

1EP4: Emergency Action Level and Emergency Plan Changes

Documents
Emergency Plan, Section 5, Revision 46
Evacuation Time Estimate Study Update

4OA1: Performance Indicator Verification

Document
NEI-99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6

4OA2: Problem Identification and Resolution

Procedures
EN-LI-102, “Corrective Action Process,” Revision 20
EN-LI-121, “Entergy Trending Process,” Revision 12

Condition Reports

CR-JAF-2012-08462	CR-JAF-2013-00449	CR-JAF-2013-01085
CR-JAF-2013-00055	CR-JAF-2013-00493	CR-JAF-2013-01166
CR-JAF-2013-00061	CR-JAF-2013-00713	CR-JAF-2013-01169
CR-JAF-2013-00196	CR-JAF-2013-00736	CR-JAF-2013-01261
CR-JAF-2013-00214	CR-JAF-2013-00773	CR-JAF-2013-01268
CR-JAF-2013-00316	CR-JAF-2013-00864	
CR-JAF-2013-00388	CR-JAF-2013-01007	

Section 4OA5: Other Activities

Documents

LO-HQNLO-2012-0076, Entergy Focused Assessment Planning Worksheet for Fleet Radiation Protection Department Performance

Indian Point Energy Center Mid-Cycle Assessment, Conducted December 3 to 12, 2012, Radiological Protection Area Portion Only

Indian Point Energy Center Radiation Protection Assessment for FitzPatrick Confirmatory Order HQNLO-2012-0076 and CR-JAF-0966, CA 57, Conducted December 3 to 6, 2012

Palisades Radiation Protection Assessment for FitzPatrick Confirmatory Order HQNLO-2012-0076, Conducted September 24 to 28, 2012

River Bend Radiation Protection Assessment for FitzPatrick Confirmatory Order HQNLO-2012-0076, Conducted December 10 to 14, 2012

Procedures

EN-LI-104, "Entergy Self-Assessment and Benchmark Process," Revisions 8 and 9

Condition Reports

CR-ANO-2012-02190	CR-JAF-2012-04503	CR-RBS-2012-07652
CR-ANO-2012-02239	CR-JAF-2012-04504	CR-VTY-2012-04915
CR-GGN-2012-11879	CR-JAF-2012-04505	CR-VTY-2012-04918
CR-IP2-2013-00318	CR-JAF-2012-04507	CR-VTY-2012-04923
CR-IP3-2012-03867	CR-PLP-2012-07849	CR-VTY-2012-04924
CR-IP3-2012-03900	CR-PLP-2012-07850	CR-VTY-2012-04925
CR-JAF-2012-00966	CR-PLP-2012-07851	CR-WF3-2012-03316
CR-JAF-2012-04398	CR-PNP-2012-04140	CR-WF3-2012-03476
CR-JAF-2012-04399	CR-RBS-2012-07650	CR-WF3-2012-03480
CR-JAF-2012-04501	CR-RBS-2012-07651	CR-WF3-2012-03482

Miscellaneous

Entergy Presentation, NRC Confirmatory Order, Radiation Protection Technician Misconduct at FitzPatrick, Entergy Lessons Learned

LIST OF ACRONYMS

10 CFR	Title 10 of the *Code of Federal Regulations*
ADAMS	Agencywide Documents Access and Management System
ADR	Alternate Dispute Resolution
CA	corrective action
CAD	containment air dilution
CAP	corrective action program
CFM	cell friction metric
CO	confirmatory order
CR	condition report
CREVAS	control room emergency ventilation air supply
CST	condensate storage tank
EC	engineering change
EDG	emergency diesel generator
Entergy	Entergy Nuclear Northeast
ESW	emergency service water
FitzPatrick	James A. FitzPatrick Nuclear Power Plant
gpm	gallons per minute
HPCI	high-pressure coolant injection
IGSCC	intergranular stress corrosion cracking
IMC	Inspection Manual Chapter
IST	inservice test
LCO	limiting condition for operation
LER	licensee event report
LOCA	loss of coolant accident
MSIV	main steam isolation valve
NCV	non-cited violation
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NUMARC	Nuclear Management and Resources Council
OE	operating experience
PCIV	primary containment isolation valve
PI	performance indicator
PMT	post-maintenance test
RCIC	reactor core isolation cooling
RHR	residual heat removal
RP	radiation protection
RSST	reserve station service transformer
RUG	Regional Utility Group
RV	reactor vessel
SDP	significance determination process
SGT	standby gas treatment
SL	severity level
SM	Shift Manager
SR	surveillance requirement

SRA	senior reactor analyst
SRM	source range monitor
SS	stainless steel
SSC	structure, system, and component
SW	service water
TIA	task interface agreement
TS	technical specification
UFSAR	Updated Final Safety Analysis Report
URI	unresolved item
WO	work order