UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BLVD., SUITE 100
KING OF PRUSSIA, PA 19406-2713

April 28, 2015

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000333/2015001

Dear Mr. Sullivan:

On March 31, 2015, 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 23, 2015, with you and other members of your staff.

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

This report documents two violations of NRC requirements, both of which were of very low safety significance (Green). 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, consistent with Section 2.3.2.a of the NRC Enforcement Policy. If you contest the non-cited violations in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at FitzPatrick. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Resident Inspector at FitzPatrick.

In accordance with Title 10 of the *Code of Federal Regulations* (CFR) 2.390 of the NRCs "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's Public Document Room or from the Publicly Available Records component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000333/2015001
w/Attachment: Supplementary Information

cc w/encl: Distribution via ListServ

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-333
License No.	DPR-59
Report No.	05000333/2015001
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Scriba, NY
Dates:	January 1, 2015 through March 31, 2015
Inspectors:	E. Knutson, Senior Resident Inspector B. Sienel, Resident Inspector E. Burket, Emergency Preparedness Specialist J. Patel, Reactor Inspector J. Schoppy, Senior Reactor Inspector
Approved by:	Arthur L. Burritt, Chief Reactor Projects Branch 2 Division of Reactor Projects

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SUMMARY

IR 05000333/2015001; 01/01/2015 - 03/31/2015; James A. FitzPatrick Nuclear Power Plant (FitzPatrick); Operability Determinations and Problem Identification and Resolution.

This report covered a three-month period of inspection by resident inspectors, announced inspections performed by regional inspectors, and an in-office review performed by regional inspectors. The inspectors identified two findings of very low safety significance (Green), both of which were 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, "Aspects Within the Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated July 9, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5.

Cornerstone: Initiating Events

- Green. A self-revealing, Green NCV of Technical Specification (TS) 5.4, "Procedures," was identified for failure to institute appropriate processes and procedures for periodic maintenance activities of the reactor water recirculation motor generators (RWR MGs). During startup from refueling outage 21, degraded material conditions led to tripping of an RWR MG, with the resultant loss of the associated RWR pump and down power transient, on three occasions. Specifically, one trip was due to carbon dust buildup within the 'A' RWR MG exciter, and two trips were due to a high resistance connection between the 'B' RWR MG generator field winding and a slip ring. Additionally, a fourth trip occurred during performance of an inadequately prepared RWR MG test procedure. As corrective action, the high resistance connection associated with the 'B' RWR MG was eliminated, voltage regulator tuning for the 'B' RWR MG was successfully completed, and temporary instrumentation was connected to both RWR MGs to monitor various key parameters pending the implementation of long term corrective actions. The RWR MG trips were entered into the corrective action program (CAP) through individual condition reports (CRs) that were subsequently consolidated under CR-JAF-2014-06258 for root cause evaluation (RCE).

  The finding was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the transient initiation of single RWR loop operations challenges the reactor feedwater and vessel level control systems such that a more significant plant transient could result, and challenges plant operators in establishing allowable single RWR loop operating conditions. In accordance with IMC 0609.04, "Initial Characterization of Findings," and Exhibit 1 of IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors determined that the finding was of very low safety significance (Green) because the performance deficiency was a transient initiator that did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. The finding had a cross-cutting aspect in the area of Human Performance, Resources, because FitzPatrick staff did not ensure that procedures for RWR preventive maintenance (PM) and voltage regulator tuning were adequate to support nuclear safety [H.1]. (Section 4OA2)

Cornerstone: Barrier Integrity

- Green. A self-revealing, Green NCV of Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified because the existence of a partially seated fuel support piece at reactor cell location 38-39 was not identified when FitzPatrick staff performed the procedure for reactor core verification at the conclusion of refueling operations during the 2014 refueling outage (RO21). Specifically, the fact that the four fuel assemblies associated with cell 38-39 were elevated by an estimated 1.5 inches above the top of the rest of the fuel assemblies in the reactor core was not identified during visual verification of fuel assembly seating performed after the conclusion of core alterations in accordance with procedure EN-RE-210, “BWR [boiling water reactor] Reactor Core and MPC [multi-purpose canister] Cask Fuel Verification.” As immediate corrective action, FitzPatrick staff engaged the fuel vendor, who provided an interim thermal limit penalty to be applied to the four affected fuel assemblies pending completion of a formal analysis. The issue was entered into FitzPatrick’s CAP as CR-JAF-2015-00789.

  The finding was more than minor because it was associated with the Configuration Control attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the fuel support piece not being completely fitted into the top of the control rod guide tube resulted in increased bypass flow around the cell 38-39 fuel assemblies, which reduced the margin to thermal limits for these assemblies during normal, transient, and accident conditions. Since the performance deficiency associated with the finding occurred during shutdown operations and also had potential safety significance during normal at-power operations, the inspectors screened the finding for significance using both IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” and IMC 0609 Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process.” The inspectors determined that the finding was of very low safety significance (Green) because the displaced fuel bundles did not have any negative impact on safety during shutdown conditions, and through application of a thermal limit penalty, did not negatively impact the safe operation of the reactor at power. This finding had a cross-cutting aspect in the area of Human Performance, Procedure Adherence, because FitzPatrick staff did not follow the procedure requirement for reactor core verification to verify that the tops of the fuel channels and bail handles were all at approximately the same height [H.8]. (Section 1R15)

REPORT DETAILS

Summary of Plant Status

FitzPatrick began the inspection period at 100 percent power. On January 14, 2015, operators reduced power to 65 percent for a control rod sequence exchange, single control rod scram time testing, and turbine valve testing. Operators restored power to 100 percent the following day. On January 22, 2015, operators reduced power to 75 percent for a control rod pattern adjustment. Operators restored power to 100 percent the following day and 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

1R04 Equipment Alignment

.1 Partial System Walkdown (71111.04 - 4 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- ‘A’ residual heat removal (RHR) system during planned maintenance on the ‘B’ low pressure coolant injection (LPCI) inverter on January 14, 2015
- ‘A’ 125 volt direct current (VDC) station battery during ‘B’ 125 VDC station battery testing on February 17, 2015
- ‘A’ and ‘C’ emergency diesel generators (EDGs) during emergent offsite maintenance on 115 kilovolt (kV) Line 3 on March 17, 2015
- ‘B’ RHR system during planned maintenance on the ‘A’ RHR system on March 25, 2015

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), TSs, 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 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 February 24 and March 18, 2015, the inspectors performed a complete system walkdown of accessible portions of the standby liquid control system to verify the existing equipment lineup was correct. The inspectors reviewed operating procedures, 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 Entergy 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 Entergy 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 26, 2015
- East cable tunnel, fire area/zone II/CT-2, on January 27, 2015
- Recirculation MG set room, fire area/zone IA/MG-1, on February 13, 2015
- Reactor building 272 foot elevation, fire area/zone IX/RB-1A, on February 19, 2015
- Reactor building 300 foot elevation, fire areas/zones VIII/RB-1C, IX/RB-1A, X/RB-1B, on March 18, 2015

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

Internal Flooding Review (1 sample)

a. Inspection Scope

The inspectors reviewed the UFSAR, the site flooding analysis, and plant procedures to assess susceptibilities involving internal flooding. The inspectors also reviewed the CAP to determine if Entergy staff identified and corrected flooding problems and whether operator actions for coping with flooding were adequate. The inspectors focused on the EDG building, which contains the ‘A’ and ‘B’ train EDGs and associated switchgear, to verify the adequacy of floor and water penetration seals, common drain lines, and flood barriers.

b. Findings

No findings were identified.

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

.1 Quarterly Review of Licensed Operator Requalification Testing and Training

a. Inspection Scope

The inspectors observed licensed operator simulator training on March 2, 2015, which included a feedwater level controller failure, a reactor water cleanup system leak, and anticipated transient without scram and emergency depressurization. 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. 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 14, 2015, the inspectors observed control room operators during a control rod sequence exchange which required a reactor downpower to approximately 60 percent. The inspectors observed crew briefs, reactivity manipulations using control rods and the RWR system, single control rod scram time testing, and 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 Entergy 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 50.65 and verified that the (a)(2) performance criteria established by Entergy staff was reasonable. 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 Entergy staff was identifying and addressing common cause failures that occurred within and across maintenance rule system boundaries.

- RWR flow control
- Instrument air
- Containment atmosphere dilution

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed maintenance activities to verify that the appropriate risk assessments were performed prior to removing equipment for work. The inspectors reviewed whether risk assessments were performed as required by 10 CFR 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors reviewed whether plant risk was promptly reassessed and managed. The inspectors also walked down selected areas of the plant which became more risk significant because of the maintenance activities to ensure they were appropriately controlled to maintain the expected risk condition. The reviews focused on the following activities:

- Planned ‘B’ LPCI inverter maintenance and control rod sequence exchange during the week of January 12, 2015
- Planned ‘A’ reactor protection system (RPS) maintenance during the week of February 2, 2015
- Planned ‘B’ 125 VDC battery charger maintenance during the week of February 16, 2015

- Emergent offsite maintenance on 115 kV Line 3 followed by planned maintenance on 115 kV Line 4 during the week of March 16, 2015
- Planned 'A' RHR system maintenance during the week of March 23, 2015

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-2015-00617 concerning the effect of not being able to perform a procedurally required monthly verification of the 'B' leading edge flow meter (LEFM) correction factor (CF) on the validity of the current CF, on February 6, 2015. The LEFM CF is used to reduce the margin of error associated with the feedwater flow input to the reactor thermal power calculation, which, in turn, is used to limit reactor power to less than or equal to 100 percent; therefore, use of an incorrect CF could result in the reactor being operated at greater power than its licensed limit
- CR-JAF-2015-00789 concerning the possibility that the reactor fuel support piece associated with control rod 38-39 is not fully seated, on February 18, 2015. Based on industry operating experience, this could be why that control rod drive mechanism (CRDM) has been experiencing elevated temperature since startup from the 2014 refueling outage. If the fuel support piece is not fully seated, the resultant bypass flow could also affect fuel thermal limits, and bundle misalignment could affect the scram time of control rod 38-39
- CR-JAF-2015-01004 concerning unsatisfactory post-maintenance calibration testing of one of the torque wrenches that was used to tension the drywell head bolts at the conclusion of the 2014 refueling outage, which resulted in the drywell possibly not being able to hold design pressure, on March 2, 2015
- CR-JAF-2015-01238 concerning the potential inoperability of 'A' train RPS high drywell pressure transmitters because the calibration procedure as-left acceptance criteria were not revised as required by a previous engineering change (EC), on March 12, 2015
- CR-JAF-2015-01294 concerning the operability of three Target Rock 3-stage safety relief valves currently in use at FitzPatrick, in light of internal degradation that was recently identified in the same model valve at another commercial nuclear power plant, on March 17, 2015

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 Entergy staff'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 Entergy staff. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations.

b. Findings

Introduction. A self-revealing, Green NCV of 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified because the existence of a partially seated fuel support piece at reactor cell location 38-39 was not identified when FitzPatrick staff performed the procedure for reactor core verification at the conclusion of refueling operations during RO21. Specifically, four fuel assemblies associated with cell 38-39 were elevated by an estimated 1.5 inches above the top of the rest of the fuel assemblies in the reactor core, and this was not identified during visual verification of fuel assembly seating performed after the conclusion of core alterations in accordance with procedure EN-RE-210, “BWR Reactor Core and MPC Cask Fuel Verification.”

Description. Operators performed a reactor startup at the conclusion of RO21 on October 7, 2014. Three days later, operators received a high temperature alarm at 250°F for CRDM 38-39. This control rod was fully withdrawn at the time, and the design of the control rod drive (CRD) system allows greater cooling water flow through the drive mechanism when the rod is not fully withdrawn. However, repeated attempts to remedy the condition by cycling the rod in one step from fully withdrawn and then back out were unsuccessful. As a result, reactor engineering provided instructions to leave the rod inserted one step, which resulted in the drive temperature stabilizing at approximately 270°F.

As background, CRDMs are attached outside the bottom of the reactor pressure vessel (RPV) bottom head, with the mechanism penetrating the bottom head. On top of the CRDM is the control rod guide tube, which serves to provide lateral support for the control rod, to separate the CRDM cooling water flow from the water in the lower plenum region (which is at higher pressure than inside the guide tube due to this being the discharge point for the RWR system), and to provide vertical support the fuel assemblies. The control rod guide tube terminates at the core plate, which divides the lower plenum region from the in-core region. The fuel support piece is slip fit to the top of the control rod guide tube and provides the slip fit bottom connections for the four fuel assemblies that surround the control rod. Four flow holes in the control rod guide tube (below the core plate) align with orifices in the fuel support piece to direct water from the lower plenum region though the four fuel assemblies.

FitzPatrick staff subsequently identified operating experience from another class 4 boiling water reactor, where a high CRDM temperature was found to be due to an issue with the assembly of parts within the reactor vessel. In this case, the fuel support piece was not completely fitted into the top of the control rod guide tube. This increased the amount of leakage flow between the lower plenum region and the control rod guide tube (some leakage is expected, since the fuel support piece connection to the control rod guide tube is a slip fit). The resultant increase in backpressure against the normal CRDM cooling water flow coming up the control rod guide tube caused the CRDM temperature to increase. Additional considerations for this condition were that the increased bypass flow also resulted in reduced flow through the fuel assemblies, which could present a challenge to fuel thermal limits (that is, limits, based on factors such as reactor power and reactor coolant flow, to maintain the integrity of the fuel cladding during normal, transient, and accident conditions). Additionally, fuel assembly misalignment due to the incomplete seating of the fuel support piece could result in increased friction on the associated control rod and therefore, a longer scram time.

On or about February 17, 2015, FitzPatrick staff reviewed photographs of cell 38-39 taken during reactor core verification, performed on September 15, 2014, at the conclusion of RO21 refueling operations and determined that these four fuel assemblies were higher than the surrounding fuel assemblies. Evaluation of the available information by the fuel vendor concluded that the cell 38-39 fuel assemblies were approximately 1.5 inches higher than the rest of the core, apparently due to incomplete seating of the fuel support piece. The fuel vendor determined that this amount of offset would result in sufficient bypass flow around the affected fuel assemblies to adversely affect their thermal limits. The fuel vendor provided an interim thermal limit penalty to be applied to the four affected fuel assemblies pending completion of a formal analysis. The issue was entered into the CAP as CR-JAF-2015-00789.

The inspectors reviewed the reactor core verification photographs of cell 38-39 and verified that these fuel assemblies were visibly elevated above the tops of the surrounding fuel assemblies. Entergy procedure EN-RE-210, "BWR Reactor Core and MPC Cask Fuel Verification," had been used by FitzPatrick staff to perform the reactor core verification at the conclusion of RO21 refueling operations. Attachment 9.3 to this procedure, "Fuel Assembly Seating Verification Using Camera(s)," states, "Verify that the tops of the fuel channels and bail handles are all at approximately the same height . . ." The inspectors concluded that the reactor core verification photographs of cell 38-39 adequately showed that its fuel assemblies were elevated such that it should have been identified by FitzPatrick staff during the performance of EN-RE-210.

Analysis. The inspectors determined that FitzPatrick staff's failure to identify the elevated fuel assemblies associated with cell 38-39 during reactor core verification at the conclusion of RO21 refueling operations was a performance deficiency that was within FitzPatrick staff's ability to foresee and correct, and should have been prevented. The finding was more than minor because it was associated with the Configuration Control attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the fuel support piece not being completely fitted into the top of the control rod guide tube resulted in increased bypass flow around the cell 38-39 fuel assemblies, which reduced the margin to thermal limits for these assemblies during normal, transient, and accident conditions.

The inspectors determined that since the performance deficiency associated with the finding occurred during shutdown operations and also had potential safety significance during normal at-power operations, both the at-power and shutdown significance determination processes would be applied. Accordingly, the inspectors screened the finding for significance using both IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," and IMC 0609 Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process." Additionally, the inspectors consulted with the Regional Senior Reactor Analyst (SRA) for a determination of potential risk significance for this issue. The inspectors determined that the finding was of very low safety significance (Green) because the displaced fuel bundles did not have any negative impact on safety during shutdown conditions, and through application of a thermal limit penalty, did not negatively impact the safe operation of the reactor at power.

This finding had a cross-cutting aspect in the area of Human Performance, Procedure Adherence, because FitzPatrick staff did not follow the procedure requirement for reactor core verification to verify that the tops of the fuel channels and bail handles were all at approximately the same height [H.8].

Enforcement. 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures or drawings. . ." Entergy procedure EN-RE-210, "BWR Reactor Core and MPC Cask Fuel Verification," Attachment 9.3, "Fuel Assembly Seating Verification Using Camera(s)," states, in part, "Verify that the tops of the fuel channels and bail handles are all at approximately the same height . . ."

Contrary to the above, on September 15, 2014, FitzPatrick staff performed EN-RE-210, Attachment 9.3, but did not identify that the fuel assemblies associated with cell 38-39 were elevated approximately 1.5 inches above the top of the rest of the reactor core. As a result, the apparent incomplete connection of the cell 38-39 fuel support piece to its control rod guide tube was not identified until subsequent plant operations at power on or about February 17, 2015. This condition results in increased bypass flow and reduced flow through the four associated fuel assemblies which challenges the fuel thermal limits. Additionally, the potential for fuel assembly misalignment could result in increased friction on the associated control rod and therefore, a longer scram time. Because this violation was of very low safety significance (Green) and FitzPatrick staff entered this issue into their CAP as CR-JAF-2015-00789, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000333/2015001-01, Incomplete Fuel Support Piece Seating Not Identified During Post-Refueling Core Verification)**

1R18 Plant Modifications (71111.18 - 1 sample)

Temporary Modifications

a. Inspection Scope

The inspectors evaluated temporary modification EC 53770, "Change Setpoint from 250°F to 300°F for CRD Temperature Alarm (Up to 10)." This change was instituted due to the persistent high temperature condition of CRDM 38-39 (as discussed in Section 1R15 of this report), although it allows the increased alarm setpoint to be applied to a total of 10 CRDMs. With CRDM 38-39 operating at approximately 270°F, this modification allowed the high temperature alarm to be cleared, thereby preventing other potential high temperature alarms from being masked. The inspectors verified that, according to long standing vendor guidance, continued operation of a CRDM in the 250°F to 300°F range has no detrimental effect on scram time performance. The inspectors reviewed the 10 CFR 50.59 documentation and reviewed the EC to verify that the temporary modification did not degrade the design bases, licensing bases, and performance capabilities of the CRD system.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed the post-maintenance tests 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 00396371 to replace the scram solenoid pilot valve for hydraulic control unit (HCU) 46-23 on January 14, 2015
- WO 52434946 and WO 52434947 to inspect and clean the ‘A’ RPS MG electrical protection assemblies on February 4, 2015
- WO 51105496 to install grease relief piping on ‘A’ emergency service water (ESW) test valve 46MOV-102A on February 26, 2015
- WO 52407979 for maintenance on battery room ‘A’ air handling unit AHU-30A recirculation isolation damper 72MOD-101A on March 3, 2015
- WO 00407787 to replace manual rod control system relay 03A-K32 for control rod 38-23 on March 5, 2015
- WO 00403893 to troubleshoot the cause of a spurious ‘C’ high reactor level trip that occurred while performing surveillance test ST-27BA, “Inverter 71INV-1A Backup Power Test,” on March 12, 2015

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 - 7 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 TSs, 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:

- ST-9BA, "EDG A and C Full Load Test and ESW Pump Operability Test," on January 26, 2015
- ST-5BA, "APRM [average power range monitor] System A Channel Functional Test," on February 5, 2015
- ST-3PB, "Core Spray Loop B Quarterly Operability Test (IST)," on February 10, 2015
- ST-4F, "HPCI [high pressure coolant injection] Automatic Isolation Logic System Functional and Simulated Automatic Actuation Test," on March 12, 2015
- ISP-49, "Reactor Water Clean-Up Area High Temperature Instrument Functional Test/Calibration**," on March 24, 2015
- ST-02AL, "RHR Loop A Quarterly Operability Test (IST)," on March 26, 2015
- ISP-16, "Drywell Floor Drain Sump Flow Loop Functional Test/Calibration*," on March 26, 2015

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

Entergy implemented various changes to the Fitzpatrick Emergency Action Levels (EALs), Emergency Plan, and Implementing Procedures. Entergy had determined that, in accordance with 10 CFR 50.54(q)(3), any change made to the EALs, Emergency Plan, and its lower-tier implementing procedures, had not resulted in any reduction in effectiveness of the Plan, and that the revised Plan continued to meet the standards in 50.47(b) and the requirements of 10 CFR 50 Appendix E.

The inspectors performed an in-office review of all EAL and Emergency Plan changes submitted by Entergy as required by 10 CFR 50.54(q)(5), including the changes to lower-tier emergency plan implementing procedures, to evaluate for any potential reductions in effectiveness of the Emergency Plan. This review by the inspectors was not documented in an NRC Safety Evaluation Report and does not constitute formal NRC approval of the changes. Therefore, these changes remain subject to future NRC inspection in their entirety. The requirements in 10 CFR 50.54(q) were used as reference criteria.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors observed a simulator training evolution for licensed operators on March 2, 2015, which required Emergency Plan implementation by an operations crew. Entergy 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 Entergy evaluators noted the same issues and entered them into the CAP.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

.1 Unplanned Power Changes (1 sample)

a. Inspection Scope

The inspectors reviewed FitzPatrick staff's submittals for the following Initiating Events Cornerstone PIs for the period of January 1 through December 31, 2014.

- Unplanned Power Changes

To determine the accuracy of the PI data reported during that period, the inspectors used definitions and guidance contained in Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7. 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.

.2 Safety System Functional Failures (1 sample)

a. Inspection Scope

The inspectors sampled FitzPatrick staff's submittals for the Safety System Functional Failures PI for the period of January 1 through December 31, 2014. To determine the accuracy of the PI data reported during that period, inspectors used definitions and guidance contained in NEI Document 99-02 and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 10 CFR 50.73." The inspectors reviewed FitzPatrick's licensee event reports (LERs) and NRC integrated inspection reports to validate the accuracy of the submittals.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152 - 3 samples)

.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 Entergy 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

No findings were identified.

.2 Annual Sample: Reactor Water Level Controller Malfunction

a. Inspection Scope

The inspectors performed an in-depth review of Entergy staff’s apparent cause evaluation and corrective actions associated with an unexpected reactor water level increase during steady state conditions at power. Specifically, the reactor water level controller level set point increased without operator action causing the reactor water level to unexpectedly increase approximately three inches, which required operator manual action to control water level on May 12, 2014 (CR-JAF-2014-02428).

The inspectors assessed Entergy staff’s problem identification threshold, cause analysis, extent-of-condition reviews, operator actions, and the prioritization and timeliness of corrective actions to evaluate whether Entergy staff was appropriately identifying, characterizing, and correcting problems associated with this issue and whether the planned and/or completed corrective actions were appropriate. The inspectors compared the actions taken to the requirements of Entergy’s operating and alarm response procedures, Entergy’s CAP, 10 CFR 50 Appendix B, and the Maintenance Rule. The inspectors interviewed operations and engineering personnel to gain an understanding of potential operational challenges, planned and completed corrective actions, and feedwater control system performance. In addition, the inspectors performed a walkdown of the feedwater level controller, including associated control room instrumentation and alarm panels, to independently assess material condition, controller operation, operator interaction, and configuration control.

b. Findings and Observations

No findings were identified.

The feedwater control system maintains RPV water level within a desired level band during normal plant operation. While in normal three-element control, the feedwater control system receives inputs from RPV water level, main steam flow rate, and feedwater flow rate transmitters and programming amplifiers. During normal operation, the master level controller (06LC-83) automatically controls a motor gear unit at each reactor feed pump (RFP) turbine to control RPV water level. The motor gear unit adjusts the position of the RFP turbine control valves, which regulates the speed of the RFP turbines. The master level controller maintains RPV level at a desired level by comparing the level setpoint (desired level) to actual level and developing an error signal that is amplified and applied as a controller output signal to correct any deviation. Normally, operators manually adjust the level setpoint by turning the "Setpoint Adjust" knob, a mechanical wire wound potentiometer, to input the desired RPV water level.

On May 12, 2014, the 06LC-83 level controller's setpoint changed from 208 to 211 inches, without operator action, which resulted in an increase in the speed of both RFPs and caused RPV water level to increase by three inches. Approximately six seconds after the RFP speed began to increase, control room operators took manual control of the master level controller and restored RPV water level to the desired band.

On May 14, 2014, Entergy staff replaced the 06LC-83 controller (WO 00382538) and sent the previously installed controller to the manufacturer for a detailed failure analysis. The controller was tested for approximately one month during which time no specific component could be proven defective since the setpoint shift did not recur during the observation period. The manufacturer concluded that the problem was an intermittent issue and that the unit would have to be tested for an indeterminate amount of time before the problem recurred. However, based on additional circuit analysis the manufacturer subsequently determined that a faulty setpoint resistor network had caused a change in resistance resulting in the setpoint shift. The faulty resistor network consists of three potentiometers. The manufacturer replaced all three potentiometers in the resistor network and performed a satisfactory functional test of the controller.

The inspectors concluded that Entergy staff had taken timely and appropriate actions in accordance with Entergy's procedures and CAP, the Maintenance Rule, 10 CFR 50 Appendix B, and NRC Regulatory Issue Summary 2007-21, "Adherence to Licensed Power Limits." The inspectors determined that Entergy staff's associated apparent cause evaluation was sufficiently thorough and based on the best available information, sound judgment, and relevant operating experience. Entergy staff's assigned corrective actions were aligned with the identified causal factors, adequately tracked, appropriately documented, and completed as scheduled. Based on a review of operating logs, alarm response procedures, and reactor power graphs, the inspectors determined that operators took prompt and appropriate actions in response to the unexpected RPV water level increase. Based on the documents reviewed, control room walkdowns, and discussions with engineering and operations personnel, the inspectors noted that the controller setpoint shift issue did not recur and that Entergy personnel identified problems and entered them into the CAP at a low threshold.

.3 Annual Sample: Review of 'A' and 'C' EDG Tie Breaker Failures

a. Inspection Scope

The inspectors performed an in-depth review of Entergy staff's equipment-apparent cause evaluation and corrective actions associated with CR-JAF-2014-01567, and RCE and corrective actions associated with CR-JAF-2014-03534. CR-JAF-2014-01567 documented an occurrence on March 31, 2014, where the 'A' and 'C' EDG tie breaker

(71-10504) failed to close automatically upon simultaneous start of the ‘A’ and ‘C’ EDGs during the performance of the monthly surveillance test (ST-9BA). CR-JAF-2014-03534 documented failure of the same breaker during the performance of ST-9BA surveillance test on July 14, 2014. Both of these events were determined to be independent and common cause failure was not identified. These events resulted in failed surveillance tests. As required by Entergy’s TS, Limiting Condition for Operation 3.8.1, Condition B, was entered. Entergy staff’s immediate corrective actions were to replace the failed breakers, complete the surveillance test to meet the TS requirements, and initiate a WO to troubleshoot the issue and understand the cause of the failures.

FitzPatrick’s emergency alternate current power system consists of four EDGs which supply power to two safety-related switchgears during a loss of offsite power. Each of the two independent and redundant emergency power systems (i.e. divisions) consists of an EDG pair connected to emergency switchgear in a parallel configuration by utilizing the generator tie-breaker. Each EDG starts and accelerates to 200 revolutions per minute (RPM), and if both EDGs achieve 200 RPM within three seconds of each other, then the common generator tie breaker gets an automatic close signal. If the three second permissive is not met, the automatic tie breaker closure is blocked and the EDGs will not force-parallel. A simultaneous manual start of both EDGs in a division during the monthly surveillance test verifies this auto-closure function of the tie breaker.

The inspectors assessed Entergy staff’s problem identification threshold, causal analysis, extent of condition reviews, and the prioritization and timeliness of Entergy staff’s corrective actions to determine whether Entergy 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 Entergy's CAP and 10 CFR 50 Appendix B, Criterion XVI, “Corrective Action.” In addition, the inspectors reviewed documentation associated with this issue, interviewed engineering and maintenance personnel, and performed a visual inspection of the breaker.

b. Findings and Observations

No findings were identified.

The inspectors found that Entergy staff took appropriate actions to identify the cause of the March 31, 2014 issue. Entergy determined the apparent cause of the breaker failure event to be a loose 52IS (interlock switch) switch causing open circuit condition in the breaker closing coil circuitry. Specifically, the set screws that hold this switch in place were found to be loose. The switch contacts are actuated by an interlock switch arm when the breaker is in the racked-in or in the test position. The interlock switch arm pushes on the plunger, which actuates the 52IS switch contacts. Therefore, the loose switch did not fully engage with the plunger and caused intermittent continuity in the breaker closing coil circuitry. Entergy determined the contributing cause to be a lack of guidance in the 4.16 kV Magne-Blast Breaker PM Procedure (MP-054.01) and in the Overhaul Procedure (MP-054.03) to verify the 52IS switch set screws are tight. Prior to this, no tightness check or continuity checks were performed during the 12-year switch replacement PM or the 16-year breaker overhaul PM. Entergy staff promptly replaced the failed breaker and updated the procedures to verify the tightness of the 52IS switch set screws. Entergy staff also performed an extent of condition review on the breakers that were scheduled to be replaced in RO21. The review did not identify any set screw looseness.

Entergy staff performed a RCE for the July 14, 2014, breaker failure event. Entergy staff determined the possible root cause to be TBRX2 relay high contact resistance from increased residue on the contacts. The TBRX2 is a permissive relay which allows the in-series K8 relay to actuate when the EDG achieves 200 RPM. Without the K8 relay energized, the permissive to allow the breaker to close is not satisfied. While troubleshooting could not confirm the failure was caused by high resistance, Entergy staff took prompt corrective action to replace five logic relays (A-TBR, C-TBR, TBRX1, TBRX2, and K8) associated with the tie breaker closing circuit for the 'A' train. In addition, Entergy staff created a corrective action to replace these similar relays on the 'B' train. Entergy staff also created a "circle-back corrective action" to perform an analysis of the 'B' train relays and update the RCE if appropriate. Entergy staff's troubleshooting on the failed breaker included verification that the 52IS was not loose. The inspectors reviewed work orders and troubleshooting results to verify that there was no common cause between the two events.

The inspectors determined that Entergy staff's overall response to the breaker failure issue was commensurate with its safety significance, and the actions taken and planned were reasonable to restore the nonconforming conditions.

.4 Annual Sample: Multiple RWR Pump Trips during Startup from Refueling Outage 21

a. Inspection Scope

The inspectors performed an in-depth review of Entergy's cause analysis and corrective actions associated with CR-JAF-2014-06258 concerning four RWR pump trips that occurred in close succession during the startup from RO21. Specifically, on October 9, 2014, with reactor power at 74 percent, the 'A' RWR pump tripped due to tripping of its associated MG; following troubleshooting, perceived repair, and recovery of the 'A' pump, the 'B' RWR pump tripped from 100 percent reactor power on October 11, 2014, also due to a trip of its associated MG; after its return to service, the 'B' RWR pump/MG again tripped from 100 percent reactor power on October 13, and yet again on October 15, 2014. Extensive troubleshooting prior to the final restart of 'B' RWR pump was successful in correcting the immediate causes of the repeated pump trips.

The inspectors assessed Entergy's problem identification threshold, cause analyses, extent-of-condition reviews, compensatory actions, and the prioritization and timeliness of Entergy's corrective actions to determine whether Entergy 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 Entergy's CAP and the plant's TSs.

b. Findings and Observations

Introduction. A self-revealing, Green NCV of TS 5.4, "Procedures," was identified for failure to institute appropriate processes and procedures for periodic maintenance activities for the RWR MGs. During startup from RO21, this led to tripping of an RWR MG, with the resultant loss of the associated RWR pump and down power transient, due to degraded material conditions on three occasions, and a fourth trip due to an inadequately prepared RWR MG test procedure.

Description. FitzPatrick started up from RO21 on October 7, 2014. During power escalation on October 9, 2014, with reactor power at 74 percent, the ‘A’ RWR pump tripped. After the initial power decrease, operators took action to further reduce reactor power to 50 percent to support single RWR loop operations. The cause of the pump trip was determined to be that the ‘A’ RWR MG had tripped due to actuation of the 02A-K36A overload relay due to an overcurrent condition. Troubleshooting activities included replacement of the tachometer generator, the 02A-K36A relay, and a diode that indicated intermittent failure. A piece of metallic foreign material was also identified in the exciter field. FitzPatrick staff considered that this foreign material had the potential to have shorted a field coil to ground, with the resultant high current having caused the 02A-K36A relay to trip. Burn marks on the foreign material, presumably due to electrical contact, supported this cause evaluation. The ‘A’ RWR pump was returned to service on October 10, 2014, and power ascension was resumed.

On October 11, 2014, FitzPatrick achieved 100 percent power. However, shortly thereafter, the ‘B’ RWR pump tripped and operators reduced power to approximately 60 percent for single RWR loop operations. The cause of the pump trip was determined to be that the ‘B’ RWR MG had tripped due to actuation of the 02A-K10B loss of generator field voltage relay. Troubleshooting identified the apparent cause to have been that a connection to one of the exciter coils had worn through its insulation and was touching the generator casing. The loss of an exciter coil would cause the exciter to be unable to maintain adequate field strength for full power operation. The insulation was repaired and the ‘B’ RWR pump was returned to service on October 12, 2014.

On October 13, 2014, FitzPatrick again achieved 100 percent power and, as before, the ‘B’ RWR pump tripped shortly thereafter. Operators reduced power to approximately 60 percent for single RWR loop operations. The cause of the pump trip was again determined to be that the ‘B’ RWR MG had tripped due to actuation of the 02A-K10B loss of generator field voltage relay. Troubleshooting activities included replacement of the tachometer generator, voltage regulator stability capacitors, and the voltage regulator. Having addressed what were considered to be the most probable trip causes, FitzPatrick staff determined that the ‘B’ RWR pump could be restarted with increased monitoring during power escalation.

The ‘B’ RWR pump was returned to service on October 15, 2014, and power ascension resumed. During the performance of IMP-02-184.8, “Recirculating MG Set Voltage Regulator Tuning**,” at 67 percent reactor power, technicians were in the process of decreasing voltage to the exciter field by 3 percent when the ‘B’ RWR MG tripped. The cause of the trip was again determined to be due to actuation of the 02A-K10B loss of generator field voltage relay. Troubleshooting determined that the loss of generator field had occurred due to an error in the test procedure. Specifically, the procedure allowed the 3 percent “bump” test to be performed under initial conditions that would place the voltage regulator outside its design capability. As a result, when the technicians had inserted the voltage decrease, the silicon controlled rectifiers in the voltage regulators stopped firing and the generator field collapsed. Bench testing of the voltage regulator subsequently confirmed that the silicon controlled rectifiers turned off under the same conditions as had existed during the test.

Further review of the first two trips of the ‘B’ RWR MG (October 11 and 13, 2014) determined that the cause had been that a loose termination point between the generator field winding and one of the slip rings created a high resistance connection. Under the right conditions, this connection could momentarily open, resulting in a loss of field. To correct this problem, the termination point (a tapered pin connection to the slip ring) was tightened and staked. The ‘B’ RWR pump was returned to service and voltage regulator tuning was successfully completed. There have been no subsequent trips of either RWR MG to date. Temporary instrumentation has been connected to both RWR MGs to monitor various key parameters pending the implementation of long term corrective actions.

CRs were written for each of the RWR pump trips. These were later consolidated under CR-JAF-2014-06258, through which a RCE of all the trips was performed. The FitzPatrick RCE team concluded that the causes of the four RWR pump trips were all related to maintenance inadequacies associated with the RWR MGs. Specifically:
1) The trip of the ‘A’ RWR MG was determined not to have been due to a short circuit induced by metallic foreign material intrusion, but rather to have been caused by a grounding condition caused by carbon dust buildup within the machine in general, and the exciter in particular; no PM existed for periodic cleaning, as specified by the vendor manual; 2) the first and second trips of the ‘B’ RWR MG were determined to have been caused by a high resistance connection between the generator field winding and a slip ring; this was not previously identified because a PM to measure the slip ring/coil resistance did not exist; and 3) the third trip of the ‘B’ RWR MG was determined to have been caused by an inadequate test procedure for performing RWR MG voltage regulator tuning, in that testing was performed with the voltage regulator operating in an area beyond its design capability. The inspectors reviewed the RCE and determined that the conclusions were appropriate.

Analysis. The inspectors determined that FitzPatrick staff’s failure to institute an effective PM strategy and voltage regulator tuning procedure for the RWR MGs was a performance deficiency that was within their ability to foresee and correct, and should have been prevented. The finding was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the transient initiation of single RWR loop operations challenges the reactor feedwater and vessel level control systems such that a more significant plant transient could result, and challenges plant operators in establishing allowable single RWR loop operating conditions. In accordance with IMC 0609.04, “Initial Characterization of Findings,” and Exhibit 1 of IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” the inspectors determined that the finding was of very low safety significance (Green) because the performance deficiency was a transient initiator that did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition.

The finding had a cross-cutting aspect in the area of Human Performance, Resources, because FitzPatrick staff did not ensure that procedures for RWR PM and voltage regulator tuning were adequate to support nuclear safety [H.1].

Enforcement. TS 5.4, "Procedures," states, in part, "Written procedures shall be established, implemented, and maintained covering . . . the applicable procedures recommended in Regulatory Guide 1.33, Appendix A, November 1972." Regulatory Guide 1.33, Appendix A, November 1972, Section I, "Procedures for Performing Maintenance," states, in part, "Maintenance which can affect the performance of safety related equipment should be properly preplanned and performed in accordance with written procedures . . . Preventive maintenance schedules should be developed to specify . . . inspections of equipment . . ." Regulatory Guide 1.33, Appendix A, November 1972, Section D, "Procedures for Startup, Operation, and Shutdown of Safety Related BWR Systems," includes the nuclear steam supply system recirculating system as such a system.

Contrary to the above, prior to startup from RO21, Entergy had not developed and implemented adequate PM schedules for inspections of the RWR MGs, nor had they developed an adequate written procedure for performing RWR MG voltage regulator tuning. As a result, RWR MG trips, and the resultant plant down power transient due to loss of the associated RWR pump, occurred on October 9, 11, 13, and 15, 2014. Because this violation was of very low safety significance (Green) and FitzPatrick staff entered this issue into their CAP as CR-JAF-2014-06258, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000333/2015001-02, Inadequate Preventive Maintenance Strategy and Test Procedure for RWR MG Resulted in Multiple Plant Transients)**

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 - 1 sample)

(Closed) LER 05000333/2014-002-00: Secondary Containment Vacuum Below Technical Specification Limit

On October 28, 2014, secondary containment pressure relative to the external environment dropped below the TS-required minimum value of greater than or equal to 0.25 inches of vacuum water gauge on two occasions while altering the reactor building ventilation (RBV) system lineup. The first instance occurred at 1655 while isolating the RBV system and placing the standby gas treatment system in service. In the isolation sequence, the RBV exhaust valves take five seconds to close, whereas the intake valves take 15 seconds to close. The period of time with air entering the secondary containment but not being removed resulted in secondary containment vacuum falling to less than 0.25 inches of vacuum water gauge for approximately one minute. The second instance occurred at 1708 when RBV was being restored to service. When the 'A' RBV system had been placed in service and the standby gas treatment system was secured, secondary containment vacuum fell below 0.25 inches of vacuum water gauge because a damper in the RBV system had not fully opened. In response, operators placed the 'B' RBV system in service, which restored secondary containment vacuum to above the TS limit within approximately one minute.

The inspectors reviewed this event when it occurred, as documented, along with the enforcement aspects of this issue, in NRC Integrated Inspection Report 05000333/2014005, Section 1R15. Although that review covered only the second occurrence (time 1708) of low vacuum, the fact that there was another would not have altered the dispositioning of this event. The inspectors did not identify any other new issues during review of the LER. This LER is closed.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 23, 2015, the inspectors presented the inspection results to Mr. Brian Sullivan, 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

B. Sullivan, Site Vice President
C. Adner, Manager, Regulatory Assurance
B. Benoit, Manager, Systems and Components Engineering
W. Drews, Manager, Design and Program Engineering
R. Heath, Manager, Radiation Protection
J. Jones, Manager, Emergency Planning
S. McAllister, Director, Regulatory and Performance Improvement
T. Peter, Manager, Operations
D. Poulin, Director, Engineering
T. Redfearn, Manager, Security
M. Reno, Manager, Training
S. Vercelli, General Manager, Plant Operations

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Open/Closed

05000333/2015001-01	NCV	Incomplete Fuel Support Piece Seating Not Identified During Post-Refueling Core Verification (Section 1R15)
05000333/2015001-02	NCV	Inadequate Preventive Maintenance Strategy and Test Procedure for RWR MG Resulted in Multiple Plant Transients (Section 4OA2)

Closed

05000333/2014-002-00	LER	Secondary Containment Vacuum Below Technical Specification Limit (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Documents
DBD-010, "Design Basis Document for the Residual Heat Removal System," Revision 13
DBD-071 Tab III, "Design Basis Document for the Electrical Distribution System 125V and 24V DC Power Systems," Revision 3

Procedures
OP-13, "Residual Heat Removal System," Revision 97
OP-17, "Standby Liquid Control System," Revision 50
OP-21, "Emergency Service Water," Revision 38
OP-22, "Diesel Generator Emergency Power," Revision 59
OP-43A, "125 VDC Power System," Revision 29
OP-60, "Diesel Generator Room Ventilation," Revision 8

Drawings
FM-20A, "Flow Diagram Residual Heat Removal System 10," Revision 72
FM-21A, "Flow Diagram Standby Liquid Control System 11," Revision 37

Condition Reports
CR-JAF-2013-00158
CR-JAF-2013-04033
CR-JAF-2013-04087
CR-JAF-2013-05687
CR-JAF-2014-02026
CR-JAF-2015-00649
CR-JAF-2015-00934
CR-JAF-2015-01083

Work Orders
WO 00359177

Section 1R05: Fire Protection

Documents
JAF-RPT-04-00478, "JAF Fire Hazards Analysis," Revision 1

Procedures
FPP-3.56, Portable Fire Extinguisher Inspection Procedure," completed on November 13, 2014
FPP-3.56, Portable Fire Extinguisher Inspection Procedure," Revision 1
OP-33, "Fire Protection," Revision 54
OP-56, "Relay Room Ventilation and Cooling," Revision 21
PFP-PWR01, "East Cable Tunnel / Elev. 258' Fire Area/Zone II/CT-2," Revision 3
PFP-PWR12, "Relay Room / Elevation 286' Fire Area/Zone VII/RR-1," Revision 5
PFP-PWR20, "Reactor Building - East / Elev. 272' Fire Area/Zone IX/RB-1A," Revision 4
PFP-PWR21, "Reactor Building - West / Elev. 272' Fire Area/Zone X/RB-1B," Revision 5
PFP-PWR23, "Motor Generator Set Room / Elev. 300' Fire Area/Zone IA/MG-1," Revision 5
PFP-PWR24, "Reactor Building - East / Elev. 300' Fire Area/Zone IX/RB-1A, VIII/RB-1C," Revision 5
PFP-PWR25, "Reactor Building - West / Elev. 300' Fire Area/Zone X/RB-1B, VIII/RB-1C," Revision 3

Condition Reports
CR-JAF-2014-06728
CR-JAF-2015-00402
CR-JAF-2015-00409
CR-JAF-2015-00419

Section 1R06: Flood Protection Measures

Documents
JAF-NE-09-00001, “JAF Probabilistic Safety Assessment, Appendix C1 - Internal Flooding Analysis,” Revision 0

Procedures
ST-40K, “Periodic Tests and Inspections,” completed March 1, 2015

Section 1R12: Maintenance Effectiveness

Documents
JAF-RPT-CAS-02304, “Maintenance Rule Basis Document System 39 Instrument Air System,” Revisions 5 & 6
JENG-APL-13-002, “Maintenance Rule (a)(1) Action Plan System 39,” Revisions 0, 1 & 2
System Health Report, Instrument Air System/Service Air System/Breathing Air System, first quarter 2012, 1st through 4th quarter 2014
System Health Reports for Recirculation Flow Control System for fourth quarter 2014 through first quarter 2014
JAF-RPT-RFC-02315, “Maintenance Rule Basis Document / System 02-184 / Reactor Water Recirculation Flow Control System,” Revision 8
JAF-RPT-MISC-02272, “Maintenance Rule Basis Document for Plant Level Performance,” Revision 8
First quarter through fourth quarter system health reports for CAD system
JAF-RPT-CAD-02312, “Maintenance Rule Basis Document System 27 Primary Containment Atmosphere Control and Dilution,” Revision 14
JENG-11-0041, “Containment Air Dilution (CAD) Maintenance Rule (a)(1) Action Plan,” Revision 5

Procedures
EN-DC-203, “Maintenance Rule Program,” Revision 2
EN-DC-204, “Maintenance Rule Scope and Basis,” Revision 3
EN-DC-205, “Maintenance Rule Monitoring,” Revision 5
EN-DC-206, “Maintenance Rule (a)(1) Process,” Revision 3
OP-39, “Breathing, Instrument , and Service Air System,” Revision 35

Condition Reports

CR-JAF-2012-02388	CR-JAF-2013-00773	CR-JAF-2013-02373
CR-JAF-2012-04853	CR-JAF-2013-00795	CR-JAF-2013-02744
CR-JAF-2012-05344	CR-JAF-2013-00839	CR-JAF-2013-02813
CR-JAF-2012-07074	CR-JAF-2013-01702	CR-JAF-2013-02970
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Section 1R13: Maintenance Risk Assessments and Emergent Work Control

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Section 1R15: Operability Determinations and Functionality Assessments

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Work Orders
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LIST OF ACRONYMS

10 CFR	Title 10 of the *Code of Federal Regulations*
APRM	average power range monitor
BWR	boiling water reactor
CAP	corrective action program
CF	correction factor
CR	condition report
CRD	control rod drive
CRDM	control rod drive mechanism
EAL	emergency action level
EC	engineering change
EDG	emergency diesel generator
Entergy	Entergy Nuclear Northeast
ESW	emergency service water
FitzPatrick	James A. FitzPatrick Nuclear Power Plant
HCU	hydraulic control unit
IMC	Inspection Manual Chapter
IST	inservice test
kV	kilovolt
LEFM	leading edge flow meter
LER	licensee event report
LPCI	low pressure coolant injection
MG	motor-generator
MPC	multi-purpose canister
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PI	performance indicator
PM	preventive maintenance
RBV	reactor building ventilation
RCE	root cause evaluation
RFP	reactor feed pump
RHR	residual heat removal
RO21	refueling outage 21
RPM	revolutions per minute
RPS	reactor protection system
RPV	reactor pressure vessel
RWR	reactor water recirculation
SSC	structure, system, and component
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VDC	volts direct current
WO	work order