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

May 13, 2016

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 – INTEGRATED INSPECTION REPORT 05000333/2016001

Dear Mr. Sullivan:

On March 31, 2016, 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 21, 2016, 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 four violations of NRC requirements, all of which were very low safety significance (Green or Severity Level IV). 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 any non-cited violation 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* (10 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 5
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000333/2016001
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.	05000333/2016001
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Scriba, NY
Dates:	January 1, 2016, through March 31, 2016
Inspectors:	E. Knutson, Senior Resident Inspector B. Sienel, Resident Inspector J. Schoppy, Senior Reactor Inspector J. Schussler, Project Engineer
Approved by:	Arthur L. Burritt, Chief Reactor Projects Branch 5 Division of Reactor Projects

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SUMMARY

Inspection Report 05000333/2016001; 01/01/2016 - 03/31/2016; James A. FitzPatrick Nuclear Power Plant (FitzPatrick); Follow-Up of Events and Notices of Enforcement Discretion.

This report covered a three-month period of inspection by resident inspectors and announced inspections performed by regional inspectors. The inspectors identified one Severity Level (SL) IV non-cited violation (NCV) and three NCVs of very low safety significance (Green). 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," dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated December 4, 2014. All violations of U.S. Nuclear Regulatory Commission (NRC) requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated February 4, 2015. 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: Mitigating Systems

- <u>Green</u>. The inspectors identified a Green NCV of Title 10 of the *Code of Federal Regulations* (10 CFR) 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for failure to maintain a condition specified in an emergency operating procedure. Specifically, while operating the high pressure coolant injection (HPCI) system in the pressure control mode, operators failed to override automatic transfer of the HPCI pump suction from the condensate storage tank (CST) to the suppression pool prior to the transfer actually occurring. As a result, operators had to revert to using the safety/relief valves (S/RVs) for pressure control, which introduced additional, unnecessary plant challenges. As immediate corrective action, operators secured HPCI, overrode the automatic HPCI pump suction transfer, realigned the pump suction to the CST, and restarted HPCI in the pressure control mode. The issue was entered into the corrective action program (CAP) as condition report (CR)-JAF-2016-00765.

  The finding was more than minor because it was associated with the human performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the operators' failure to timely override automatic transfer of the HPCI suction to the suppression pool resulted in an additional, avoidable post-scram pressure and level transient being placed on the reactor pressure vessel (RPV) and unnecessarily reduced the thermal capacity of the suppression pool. In accordance with IMC 0609.04, "Initial Characterization of Findings," and Exhibit 2 of IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors determined that this finding was of very low safety significance (Green) because the performance deficiency was not a design or qualification deficiency, did not involve an actual loss of a safety function of a single train for greater than its technical specification (TS) allowed outage time, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. The finding had a cross-cutting aspect in the area of Human Performance, Procedure Adherence, because operators did not follow guidance of EOP-2 for the HPCI pump suction to be aligned to the CST by bypassing the HPCI pump suction swap to the suppression pool in a timely manner, such that the swap actually occurred [H.8]. (Section 4OA3)

- Green. The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for failure to take actions specified in the procedure for initiation of shutdown cooling. Specifically, prior to placing the ‘A’ loop of the residual heat removal (RHR) system into shutdown cooling, an operator was not stationed to close the condensate transfer system cross-connect valve to the ‘A’ RHR loop (10RHR-274), nor was the valve immediately closed after initiation of shutdown cooling, as specified by the operating procedure. This resulted in a significant loss of operational control, in that RPV level increased to the point of putting water down the main steam lines. As immediate corrective action, operators closed 10RHR-274, thus stopping the RPV inventory increase. The issue was entered into the CAP as CR-JAF-2016-00273.

  The finding was more than minor because it was associated with the human performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the resultant loss of RPV level control represented a significant loss of operational control that could have affected the operability of the HPCI and reactor core isolation cooling (RCIC) systems, as well as the S/RVs, had their use again been required in the near term. In accordance with IMC 0609.04, “Initial Characterization of Findings,” and Exhibit 2 of IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” the inspectors determined that this finding was of very low safety significance (Green) because the performance deficiency was not a design or qualification deficiency, did not involve an actual loss of a safety function of a single train for greater than its TS allowed outage time, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. The finding had a cross-cutting aspect in the area of Human Performance, Challenge the Unknown, because operators did not stop when faced with uncertain conditions. Specifically, without otherwise having maintained status control on the condensate transfer system cross-connect valve to the ‘A’ RHR loop, operators did not stop to positively establish the condition of the valve when it appeared in a conditional step in the procedure (that is, “if 10RHR-274 is open, then station an operator at 10RHR-274”) [H.11]. (Section 4OA3)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a self-revealing NCV of TS 5.4, “Procedures,” for FitzPatrick staff’s failure to perform adequate post-maintenance testing (PMT) following maintenance on a limit switch in the reactor building ventilation system in August 2014, that, along with another unrelated component failure in the reactor building ventilation system, resulted in secondary containment pressure, relative to the outside pressure, exceeding the TS limit of 0.25 inches of vacuum water gauge. As immediate corrective action, operators started both trains of the standby gas treatment system (SBGTS), which restored secondary containment pressure to within the TS limit. Operators subsequently secured the ‘A’ refuel floor exhaust train and placed the ‘B’ train in service. The issue was entered into the CAP as CR-JAF-2015-04166.

  The finding was more than minor because it was associated with the configuration control attribute of the Barrier Integrity cornerstone and 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, as a result of this event, secondary containment was not preserved, in that secondary containment pressure exceeded the limit of TS surveillance requirement (SR) 3.6.4.1.1. In accordance with IMC 0609.04, “Initial Characterization of Findings,” and

Exhibit 3 of IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors determined that this finding was of very low safety significance (Green) because the performance deficiency was not a pressurized thermal shock issue, did not represent an actual open pathway in the physical integrity of the reactor containment, did not involve an actual reduction in function of hydrogen igniters in the reactor containment, and only represented a degradation of the radiological barrier function provided by the reactor building and SBGTS. The finding had a cross-cutting aspect in the area of Human Performance, Resources, because FitzPatrick staff did not ensure that procedures for PMT of the reactor building refuel floor exhaust damper limit switch following maintenance performed in August 2014, were adequate to support the nuclear safety function of the secondary containment [H.1]. (Section 4OA3)

- <u>Severity Level IV</u>. The inspectors identified a SL IV NCV of 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," because unplanned inoperability of the secondary containment system was not reported to the NRC within eight hours of the occurrence, as required by 10 CFR 50.72(b)(3)(v), "Event or Condition That Could Have Prevented Fulfillment of a Safety Function." Specifically, following reasonable resolution of questions regarding the reliability of secondary containment differential pressure (d/p) instrumentation indications, FitzPatrick staff did not promptly report that, during a transfer from normal reactor building ventilation in service to the reactor building being isolated with the SBGTS in service, reactor building d/p briefly dropped below the TS required minimum value of 0.25 inches of vacuum water gauge and therefore caused the secondary containment system to be inoperable. As immediate corrective action, the event was reported to the NRC in accordance with 10 CFR 50.72(b)(3)(v). The issue was entered into the CAP as CR-JAF-2015-05244 and CR-JAF-2015-05265.

  The inspectors determined that the failure to inform the NRC of the secondary containment system inoperability within eight hours in accordance with 10 CFR 50.72(b)(3)(v) was a performance deficiency that was reasonably within Entergy's ability to foresee and correct. The inspectors evaluated this performance deficiency in accordance with the traditional enforcement process 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. 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 50.72 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. (Section 4OA3)

REPORT DETAILS

Summary of Plant Status

FitzPatrick began the inspection period at 100 percent power. On January 16, 2016, operators reduced power to 65 percent for a control rod sequence exchange and turbine valve testing, and restored power to 100 percent. On January 22, 2016, operators reduced power to 60 percent to perform maintenance on control rod drive hydraulic control units (HCUs). During power ascension on January 23, 2016, operators inserted a manual scram due to lowering lake water intake level caused by frazil ice. The scram response was complicated by failure of the automatic fast transfer of house loads to the reserve station service transformers, which resulted in a loss of essentially all non-vital plant loads. Following recovery, troubleshooting, and repairs, operators performed a reactor startup on January 29, 2016, and synchronized the main generator to the grid on January 31, 2016. During power ascension, anomalous temperature indications for the reactor water recirculation (RWR) motor generators (MGs) led operators to reduce power from 99 percent to 70 percent, based on guidance developed in response to multiple RWR MG trips that occurred in October 2014. Following troubleshooting and repair, operators completed power ascension to 100 percent on February 3, 2016. FitzPatrick continued to operate 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

On March 28 and 29, 2016, the inspectors reviewed FitzPatrick’s preparations for high winds (average wind speed greater than 30 miles per hour) due to an arriving weather front. The inspectors walked down exterior portions of the plant to identify loose or inadequately protected equipment and materials. The inspectors verified that the circulating water and service water systems were operated in accordance with procedural requirements for high wind conditions. The plant did not experience any significant operational issues as a result of the high wind conditions. 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 Walkdown (71111.04 - 4 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- ‘A’ containment atmosphere dilution (CAD) system during maintenance on the ‘B’ CAD system on February 18, 2016
- ‘B’ RHR system during maintenance on the ‘A’ RHR system on February 23, 2016
- ‘B’ and ‘D’ emergency diesel generators (EDGs) during maintenance on the ‘A’ RHR system on February 23, 2016
- ‘A’ and ‘C’ EDGs during maintenance on 115 kilovolt (kV) offsite Line 4 on March 28, 2016

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 March 7–10, 2016, the inspectors performed a complete system walkdown of accessible portions of the ‘B’ core spray 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 February 9, 2016
- East cable tunnel, fire area/zone II/CT-2, on February 9, 2016
- Reactor building, 272' elevation, fire area/zone IX/RB-1A, on February 24, 2016
- Reactor building, 300' elevation, fire areas/zones VIII/RB-1C, IX/RB-1A, X/RB-1B, on March 3, 2016
- RWR MG set room, fire area/zone IA/MG-1, on March 15, 2016

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 February 22, 2016, which included loss of a main condensate pump, a half scram due to failure of the 'A' reactor protection system (RPS) MG set, unintended opening of two main steam bypass valves that led to a high pressure reactor scram, failure of the control rods to insert following receipt of the scram signal, and failure of both standby liquid control system pumps. 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 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 22, 2016, operators performed a power reduction to approximately 60 percent to facilitate maintenance on 32 HCUs. The inspectors observed the power decrease, including reactivity manipulations using control rods and the RWR system. The inspectors also observed the beginning of shift crew brief following turnover between the night and day shift operators. 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 - 2 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.

- Reactor building closed loop cooling
- Reactor building ventilation

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 maintenance on the HPCI system the week of February 8, 2016
- Planned maintenance on six control rod drive system HCUs and emergent maintenance on a pressure control valve in the ‘B’ CAD system the week of February 15, 2016
- Planned maintenance on the ‘A’ RHR system, the ‘A’ spent fuel pool cooling pump, and six HCUs the week of February 22, 2016
- Calibration of the local power range monitor system using the traversing in-core probe system, ‘A’ standby liquid control system quarterly surveillance test, torus-to-drywell vacuum breaker surveillance test, and maintenance on ‘B’ CAD system the week of March 14, 2016
- Planned maintenance outage on 115 kV Line 4 with emergent high winds, followed by a planned maintenance outage on 345 kV Line 1, and RCIC system quarterly surveillance test the week of March 28, 2016

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15 - 6 samples)

a. Inspection Scope

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

- CR-JAF-2016-00244 concerning the failure of house loads to fast transfer to the reserve station service transformers following the scram of January 23, 2016; specifically, the PMT that had been done to verify, prior to startup, the fast transfer feature would function properly given that actual testing of the feature was impractical on January 30, 2016
- CR-JAF-2016-00493 concerning the ability to monitor gross activity with both offgas radiation monitors inoperable on February 3, 2016
- CR-JAF-2016-00590 concerning the effect on HPCI system operability of a disagreement between an electrical schematic of the control circuitry for the HPCI pump discharge to the reactor outboard isolation valve, 23MOV-20, which shows a mechanical interlock between the opening and closing relays and the actual configuration which has no such interlock on February 12, 2016

- CR-JAF-2016-00697 concerning the operability of the ‘B’ CAD system following the failure of the ‘B’ ambient vaporizer nitrogen inlet valve to meet its inservice testing opening stroke time during surveillance testing on February 19, 2016
- CR-JAF-2016-01005 concerning an electrical burn mark found on ‘A’ low pressure coolant injection power supply armored output cable and its possible implications to operability of the inverter or the overall power supply on March 15, 2016
- CR-JAF-2016-01057 concerning decreasing trends in ‘E’ S/RV first and second stage temperatures and their possible impacts to operability of the valve’s safety and relief functions on March 22, 2016

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

No findings were identified.

1R18 Plant Modifications (71111.18 - 1 sample)

Temporary Modification

a. Inspection Scope

The inspectors reviewed a procedurally controlled temporary modification to install and then remove clamps on the drywell instrument nitrogen normal pressure control valve (27PCV-120) isolation valves 27AOV-129A and B. The clamps were installed to facilitate preventive maintenance (PM) on 27PCV-120. The temporary modification was controlled by OP-37, “Containment Atmosphere Dilution System,” Attachment 6, “Procedural Temp Mod Control Form for Clamp Installation/Removal for 27AOV-129A/B.” The inspectors conducted a field walkdown of the modification to verify proper installation. The inspectors reviewed the temporary modification to verify it did not degrade the design basis, licensing basis, or performance capabilities of the CAD system.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed the 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 52386816 to perform PM on reactor building ventilation below refuel floor exhaust fan 66FN-12B breaker on January 15, 2016
- WO 52467267 and WO 52552759 to perform PM on HPCI valves 23HOV-1 (turbine stop valve) and 23HPI-61 (booster pump suction check valve from torus) on February 12, 2016
- WO 51193117 to perform PM on instrument nitrogen normal pressure control valve 27PCV-120 on February 18, 2016
- WO 00426173 to replace the coil on crescent area unit cooler 66UC-22D on February 28, 2016
- WO 00432095 to replace CAD ambient vaporizer 'B' inlet valve solenoid valve 27SOV-126B on March 15, 2016
- WO 00441200 to replace the S/RV electric lift system Division 1 125 volts direct current (VDC) in/24 VDC out instrument power supply 02P/S-1 on March 28, 2016

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors monitored the station's work schedule and outage risk management for the forced outage that occurred on January 24 through January 31, 2016. The inspectors reviewed FitzPatrick staff's development and implementation of outage plans and schedules to verify that risk, industry experience, previous site-specific problems, and defense-in-depth were considered. During the outage, the inspectors observed portions of the shutdown and cooldown processes and monitored controls associated with the following activities:

- Configuration management, including maintenance of defense-in-depth, to maintain the key safety functions and compliance with the applicable TSs when taking equipment out of service
- Implementation of clearance activities and confirmation that equipment was appropriately configured to safely support the associated work or testing

- Configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication
- Status and configuration of electrical systems and switchyard activities to ensure that TSs were met
- Monitoring of decay heat removal operations
- Activities that impacted the ability of the operators to operate the spent fuel pool cooling system
- Reactor water inventory controls, including flow paths, configurations, and alternative means for inventory additions
- Activities that could affect reactivity
- Maintenance of secondary containment as required by TSs
- Tracking of startup prerequisites, walkdown of the drywell to verify that debris had not been left which could block the emergency core cooling system suction strainers, and startup and power ascension
- 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 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:

- ISP-16, "Drywell Floor Drain Sump Flow Loop Functional Test/Calibration*," on February 2, 2016
- ST-9AB, "EDG System B Fuel Oil Monthly Test," on March 7, 2016
- ST-9BB, "EDG B and D Full Load Test and Emergency Service Water Pump Operability Test," on March 7, 2016
- ST-6HA, "Standby Liquid Control A Side Quarterly Operability Test (IST)," on March 15, 2016
- ISP-100C-RPS, "RPS Instrument Functional Test/Calibration Analog Transmitter Trip System**," on March 24, 2016
- ISP-8A, "Above Core Plate to Core Spray Line at Reactor Pressure Vessel Differential Pressure Instrument Functional Test/Calibration," on March 29, 2016

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 - 2 samples)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine FitzPatrick emergency drill on March 17, 2016, to identify any weaknesses and deficiencies in the classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the simulator and technical support center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the technical support center drill critique to compare inspector observations with those identified by FitzPatrick staff in order to verify whether the FitzPatrick staff was properly identifying weaknesses and entering them into the CAP.

b. Findings

No findings were identified.

.2 Training Observations

a. Inspection Scope

The inspectors observed a simulator training evolution for licensed operators on February 22, 2016, 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 Verification (71151 - 2 samples)

.1 Unplanned Power Changes

a. Inspection Scope

The inspectors reviewed FitzPatrick staff's submittals for the following Initiating Events cornerstone PI for the period of January 1, 2015, through December 31, 2015.

- 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

a. Inspection Scope

The inspectors sampled FitzPatrick staff’s submittals for the safety system functional failures PI for the period of January 1, 2015, through December 31, 2015. 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 – 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 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: Torus-to-Drywell Differential Pressure Concerns

a. Inspection Scope

The inspectors performed an in-depth review of Entergy's evaluations and corrective actions associated with maintaining torus-to-drywell d/p. Specifically, following the refueling outage (RF21) in October 2014, operators initiated CR-JAF-2014-06207 for a concern regarding challenges to maintaining drywell to torus d/p compared to pre-outage. Operators noted that they had to perform drywell nitrogen additions and torus venting activities more frequently than normal (approximately four times more often). This increased frequency resulted in a burden on operators, as well as requiring increased run-time on SBGTS equipment, and increased cycling of large primary containment isolation valves to support the make-up and venting activities.

The inspectors assessed Entergy's problem identification threshold, cause analyses, extent-of-condition reviews, operator actions, and the prioritization and timeliness of corrective actions to evaluate whether Entergy was appropriately identifying, characterizing, and correcting problems associated with these issues 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; FitzPatrick's TSs; 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 torus-to-drywell vacuum breaker performance. In addition, the inspectors performed several walkdowns of accessible portions of the torus-to-drywell vacuum breakers, including associated control room and relay room instrumentation and alarm panels, to independently assess the material condition, operating environment, operator awareness and response, and configuration control.

b. Findings and Observations

No findings were identified.

Entergy troubleshooting performed via a failure modes analysis determined that the most likely cause of the leakage from the drywell to the torus was through one or more of the five torus-to-drywell vacuum breakers (27VB-1 through 27VB-5). In an attempt to limit this potential leakage path by re-seating the vacuum breaker disc in its seat, operators performed the torus-to-drywell vacuum breaker quarterly surveillance test (ST-15J) with limited success (the frequency of make-up and vent instances decreased but was still more frequent than prior to RF21). In addition, Entergy implemented a one-time change to procedure ST-15J to place additional torque in the closing direction on the vacuum breaker valves; however, this too proved unsuccessful in correcting the condition.

In November 2014, Entergy used their operational decision-making issue (ODMI) process to develop and implement an ODMI plan to monitor and manage the adverse condition (torus-to-drywell d/p) while awaiting an opportunity to perform vacuum breaker inspections and repairs, as needed. Entergy's associated ODMI plan included: (1) documenting each drywell nitrogen make-up and torus vent activity in the narrative logs, (2) monitoring SBGTS run hours to ensure margin to charcoal replacement requirements, (3) calculating the drywell to torus leak rate daily and documenting in the narrative logs, (4) trending the make-up frequency and nitrogen usage, and (5) monitoring nitrogen tank inventory. The inspectors noted that Entergy established appropriate and conservative ODMI trigger points and actions commensurate with nuclear safety. Entergy developed and planned detailed WOs (00397546, 00397751, 00397752, 00397753, and 00397754) for the inspection and repair, as needed, for all five torus-to-drywell vacuum breakers and scheduled the work for the next outage of sufficient duration.

During a meeting to discuss the path forward to repair the vacuum breaker valves, engineering identified that no PM tasks existed to periodically replace the soft seats or elastomers on the valves. On December 9, 2014, engineering initiated corrective action CR-JAF-2014-07095 to address this potential gap. In response, engineering developed a periodic PM to open, inspect, repair, and/or replace the vacuum breaker valve internals and submitted it to the PM coordinator for implementation (AR 214419). Entergy initiated a CAP action item to track implementation of the PM (CR-JAF-2014-07095 CA 3).

Based on a review of the vacuum breaker vendor manual and related internal and external operating experience, the inspectors determined that Entergy's actions prior to October 2014 were reasonable and the absence of a periodic PM to perform internal inspection and repair of the torus-to-drywell vacuum breakers did not represent an Entergy performance deficiency. The inspectors concluded that, following identification of the concern, Entergy had taken timely and appropriate actions in accordance with Entergy's procedures and CAP, TSs, the NRC Maintenance Rule, and 10 CFR 50, Appendix B. The inspectors determined that Entergy's associated evaluations were sufficiently thorough and based on the best available information, sound judgment, and relevant operating experience. Entergy's assigned corrective actions were aligned with the identified causal factors, adequately tracked, appropriately documented, and completed as scheduled. Based on a review of operations' narrative logs and alarm response procedures, the inspectors determined that operators took prompt and appropriate actions in response to the torus-to-drywell d/p concerns. Based on the documents reviewed, control room and plant walkdowns, and discussions with engineering and operations personnel, the inspectors noted that the drywell make-up and torus venting returned to pre-RF21 frequency and that Entergy personnel identified problems and entered them into the CAP at a low threshold. The inspectors did not identify any issues or concerns that had not been appropriately entered into the CAP for evaluation and resolution. In response to several questions and minor equipment deficiencies identified by the inspectors during plant walkdowns, Entergy personnel promptly initiated CRs and/or took immediate action to address the issues.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 - 6 samples)

.1 Complicated Reactor Scram Due to Circulating Water System Icing (1 sample)

a. Inspection Scope

On January 22, 2016, operators reduced power to 60 percent to perform maintenance on control rod drive HCUs. During subsequent power ascension on the night of January 23, 2016, with reactor power at 89 percent, operators received an alarm for low screenwell intake level at 242 feet; normal lake level was about 244 feet. Based on lake and outside environmental conditions, this was considered likely to be due to frazil ice.

Operators entered AOP-56, "High Traveling Screen or Trash Rack Differential Level," and began reducing power. When power was less than 75 percent, operators secured one of the three circulating water pumps (lower water velocity tends to slow the formation of frazil ice). Intake level increased slightly, but then resumed its lowering trend. Operators continued to reduce power, but when intake level reached the AOP-56 override point of less than 240 feet at 10:40 p.m., operators inserted a manual scram.

Following the reactor scram and turbine trip, the expected automatic "fast" transfer of station electrical loads (a seamless transfer of power such that no operating equipment is lost) from the main generator through the normal station service transformer, to offsite power through the two reserve station service transformers, did not occur. Within three seconds, the backup automatic "residual" transfer did occur, but with the resultant loss of all previously operating non-vital equipment. Operators shut the main steam isolation valves due to the loss of all circulating water system pumps. In this mode of operation, the suppression pool, cooled by the RHR system, provides the heat sink for the reactor plant; and HPCI, RCIC, and the S/RVs provide RPV pressure and level control. Operators used these systems to perform a slow plant cooldown while they worked to restore normally operating plant systems to service. RHR shutdown cooling was placed in service at 10:59 p.m. on January 24, 2016.

The inspectors responded to the plant to observe plant parameters, observe and review personnel performance, and evaluate the performance of mitigating systems. The inspectors communicated the plant events to appropriate regional personnel, and compared the event details with criteria contained in IMC 0309, "Reactive Inspection Decision Basis for Reactors," for consideration of potential reactive inspection activities. The inspectors verified that FitzPatrick staff made appropriate emergency classification assessments and properly reported the event in accordance with 10 CFR 50.72 and 50.73. The inspectors reviewed FitzPatrick's follow-up action related to the event to assure that FitzPatrick staff implemented appropriate corrective actions commensurate with their safety significance.

b. <u>Findings</u>

A. <u>Unintended HPCI Pump Suction Transfer during Pressure Control Mode Operation</u>

<u>Introduction</u>. The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for failure to maintain a condition specified in an emergency operating procedure. Specifically, while operating HPCI in the pressure control mode, operators failed to override automatic transfer of the HPCI pump suction from the CST to the suppression pool prior to the transfer actually occurring. As a result, operators had to revert to using the S/RVs for pressure control, which introduced additional, unnecessary plant challenges.

<u>Description</u>. After the January 23, 2016, scram due to frazil ice and subsequent failure of electrical loads to fast transfer to the reserve station service transformers, operators used HPCI for RPV pressure control in accordance with OP-15, "High Pressure Coolant Injection," Section D.2, "Manual Startup for RPV Pressure Control." This mode of pressure control is preferable to using the S/RVs because it eliminates the possibility of the undesirable pressure/temperature transient that would result were an S/RV to stick open, as well as providing smoother pressure control. Additionally, HPCI exhausts lower enthalpy steam to the suppression pool (due to its having been used to operate the

turbine) than steam from the S/RVs, thereby conserving thermal margin in the suppression pool.

EOP-2, "RPV Control," directs that, in this mode, the HPCI suction be aligned to the CST, if available. At the time of the event, suppression pool level was gradually increasing due to HPCI and RCIC being in service. On the morning of January 24, 2016, as level approached the point at which the HPCI suction would automatically transfer to the suppression pool, the control room supervisor directed operators to bypass the transfer in accordance with EP-2, "Isolation/ Interlock Overrides," Section 5.13, "HPCI Pump Suction Valves 23MOV-57, 58, and 17: Suction Swap Prevention." However, this action was not completed before the automatic swap occurred. OP-15 states that, if the HPCI pump suction is lined up to the torus, HPCI cannot be used in RPV pressure control mode. This required operators to secure HPCI to realign the suction to the CST. As a result, operators had to revert to using the S/RVs for pressure control until HPCI was again available.

The inspectors discussed this issue with FitzPatrick personnel, who entered the issue into the CAP as CR-JAF-2016-00765.

Analysis. The inspectors determined that the failure to override the HPCI suction swap to the suppression pool in a timely manner, such that HPCI was required to be secured and pressure control maintained using S/RVs, was a performance deficiency that was within FitzPatrick's ability to foresee and correct, and should have been prevented. This finding was more than minor because it was associated with the human performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the operators' failure to timely override automatic transfer of the HPCI suction to the suppression pool resulted in an additional, avoidable post-scram pressure and level transient being placed on the RPV and unnecessarily reduced the thermal capacity of the suppression pool.

In accordance with IMC 0609.04, "Initial Characterization of Findings," and Exhibit 2 of IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors determined that this finding was of very low safety significance (Green) because the performance deficiency was not a design or qualification deficiency, did not involve an actual loss of a safety function of a single train for greater than its TS allowed outage time, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event.

This finding had a cross-cutting aspect in the area of Human Performance, Procedure Adherence, because operators did not follow guidance of EOP-2 for the HPCI pump suction to be aligned to the CST by bypassing the HPCI pump suction swap to the suppression pool in a timely manner, such that the swap actually occurred [H.8].

Enforcement. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, "Activities affecting quality shall be prescribed by . . . procedures . . . of a type appropriate to the circumstances and shall be accomplished in accordance with these . . . procedures . . ." EOP-2, "RPV Control," specifies that, if HPCI is being used for pressure control, then align the suction to the CST, if available. EP-2, "Isolation/Interlock Overrides," Section 5.13, "HPCI Pump Suction Valves 23 MOV-57, 58, and 17: Suction Swap Prevention," provides directions on how to bypass the HPCI pump suction automatic transfer from the CST to the suppression pool due to high level in the suppression pool.

Contrary to the above, on January 24, 2016, with HPCI being operated in the pressure control mode with the pump suction aligned to the CST, operators did not take timely action in accordance with EP-2, Section 5.13, to bypass the HPCI pump suction automatic transfer from the CST to the suppression pool, such that the transfer actually occurred. As a result, an additional, avoidable pressure and level transient was placed on the RPV and the thermal capacity of the suppression pool was unnecessarily reduced. Because this violation was of very low safety significance (Green) and Entergy entered this issue into their CAP as CR-JAF-2016-00765, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy.
(NCV 05000333/2016001-01, Unintended HPCI Pump Suction Transfer during Pressure Control Mode Operation)

B. Uncontrolled RPV Level Increase after Initiation of RHR Shutdown Cooling

Introduction. The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for failure to take actions specified in the procedure for initiation of shutdown cooling. Specifically, prior to placing the ‘A’ loop of RHR into shutdown cooling, an operator was not stationed to close the condensate transfer system cross-connect valve to the ‘A’ RHR loop, nor was the valve immediately closed after initiation of shutdown cooling, as specified by the operating procedure. This resulted in a significant loss of operational control, in that RPV level increased to the point of putting water down the main steam lines.

Description. On January 24, 2016, operators were preparing to place ‘A’ RHR in shutdown cooling in accordance with OP-13D, “RHR - Shutdown Cooling,” Revision 27, Subsection D.1, “RHR Loop A Shutdown Cooling Startup/Shifting Shutdown Cooling Loops.” Step D.1.8 directs operators to flush the RHR loop per Subsection G.2, “RHR Loop A Flush/Vent.” The flush is accomplished using the condensate transfer system through valve 10RHR-274, “RHR loop A containment spray keep-full condensate transfer connection valve.” At the completion of Subsection G.2, 10RHR-274 remains open to maintain the loop full.

Prior to initiating shutdown cooling, step D.1.31 directs that, if 10RHR-274 is open, then station an operator at the valve. After shutdown cooling has been initiated, step D.1.37 states that, if 10RHR-274 is open, then immediately close it. Operators apparently did not recognize that 10RHR-274 was opened, therefore an operator had not been stationed at the valve. When shutdown cooling was initiated, the action to close 10RHR-274 was not taken and RPV level began to rise. By the time that operators identified the problem and shut 10RHR-274, RPV level had reached the main steam lines and water had entered the lines. Depending on the degree of main steam line flooding, this condition could affect operability of the HPCI and RCIC systems, as well as the S/RVs.

This issue was entered into FitzPatrick’s CAP as CR-JAF-2016-00273. The inspectors noted that, the CR stated that the issue was caused by an inadequate procedure; specifically 10RHR-274 was opened in Subsection G.2 but was not reclosed, and that there was a similar procedural deficiency in Subsection G.3 for flushing the ‘B’ RHR loop. However, the inspectors reviewed OP-13D and determined the procedure for initiating shutdown cooling could successfully be performed as written, provided that the operators maintained accurate status control of 10RHR-274. The inspectors concluded that the cause of this event was that operators did not recognize that 10RHR-274 had remained opened at the conclusion of OP-13D, Subsection G.2, and therefore, did not station a watch to immediately close it after shutdown cooling was initiated. OP-13D was subsequently revised to include a note with step D.1.31 that 10RHR-274 will be in the open position if Subsection G.2 was performed per step D.1.8.

Analysis. The inspectors concluded that failure to station an operator at 10RHR-274 to immediately shut the valve after shutdown cooling was initiated, as specified by OP-13D, was a performance deficiency that was within FitzPatrick's ability to foresee and correct, and should have been prevented. This finding was more than minor because it was associated with the human performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the resultant loss of RPV level control represented a significant loss of operational control that could have affected the operability of the HPCI and RCIC systems, as well as the S/RVs, had their use again been required in the near term.

In accordance with IMC 0609.04, "Initial Characterization of Findings," and Exhibit 2 of IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors determined that this finding was of very low safety significance (Green) because the performance deficiency was not a design or qualification deficiency, did not involve an actual loss of a safety function of a single train for greater than its TS allowed outage time, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event.

This finding had a cross-cutting aspect in the area of Human Performance, Challenge the Unknown, because operators did not stop when faced with uncertain conditions. Specifically, without otherwise having maintained status control on 10RHR-274, operators did not stop to positively establish the condition of 10RHR-274 when it appeared in a conditional step in the procedure (that is, "if 10RHR-274 is open, then station an operator at 10RHR-274") [H.11].

Enforcement. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, "Activities affecting quality shall be prescribed by . . . procedures . . . of a type appropriate to the circumstances and shall be accomplished in accordance with these . . . procedures . . ." OP-13D, "RHR - Shutdown Cooling," Revision 27, Subsection D.1, "RHR Loop A Shutdown Cooling Startup/Shifting Shutdown Cooling Loops," step D.1.31, requires that, if 10RHR-274 is open, then station an operator at 10RHR-274, and, after shutdown cooling has been initiated, step D.1.37, requires that, if 10RHR-274 is open, then immediately close 10RHR-274.

Contrary to the above, on January 24, 2016, while placing the 'A' loop of RHR into shutdown cooling in accordance with OP-13D, Subsection D.1, with 10RHR-274 open, operators did not station an operator at 10RHR-274, and after shutdown cooling was initiated with 10RHR-274 open, did not immediately close 10RHR-274. As a result, before the cause was identified and corrected, RPV level increased to the level of the main steam lines, and water was introduced into the main steam lines, a condition that could have affected the operability of the HPCI and RCIC systems, as well as the S/RVs, had their use again been required in the near term. Because this violation was of very low safety significance (Green) and Entergy entered this issue into their CAP as CR-

JAF-2016-00273, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000333/2016001-02, Uncontrolled RPV Level Increase after Initiation of RHR Shutdown Cooling)**

.2 (Closed) LER 05000333/2015-004-00: Concurrent Opening of Reactor Building Airlock Doors (1 sample)

On September 17, 2015, personnel inside and outside the secondary containment simultaneously opened both doors to the north reactor building 272' elevation secondary containment airlock. TS SR 3.6.4.1.3 requires that one secondary containment access door in each access opening is closed at all times. Both parties were aware of this requirement, and both withdrew and closed their respective doors within a period of approximately five seconds. However, during the period that both doors were simultaneously open, TS SR 3.6.4.1.3 was not satisfied and therefore, secondary containment was inoperable. At the time of discovery, operators briefly entered TS limiting condition for operation 3.6.4.1, which requires that secondary containment be restored to operable status within four hours. However, because secondary containment is a single train system, this occurrence was reportable under 10 CFR 50.72, even though the condition existed for less than the TS allowed outage time.

The inspectors noted that the subject secondary containment access is equipped with three sets of green and red position indication lights, one set outside the airlock on either side, and the third set inside the airlock. All the green lights are energized if both air lock doors are closed. However, if either door is opened, then the green lights extinguish and all the red lights are energized. This scheme functions adequately to alert an individual preparing to enter an airlock that another individual is already in the process of entering from the other side. However, in the event that both doors are operated simultaneously, the indications for both individuals are as expected for a single door being opened, and neither recognizes a problem exists until they are positioned to see the other open door (at which point, it is too late). Therefore, the inspectors determined that this event was not due to a human performance deficiency. Given that secondary containment d/p remained within specification, along with the short duration of the event, the inspectors determined that secondary containment had remained capable of performing its design function throughout the event. Additionally, the issue was timely reported in accordance with the requirements of 10 CFR 50.72, so it did not constitute a traditional enforcement issue. Because the failure to comply with TS SR 3.6.4.1.3 was corrected within the allowed outage time, no violation of regulatory requirements occurred. This LER is closed.

.3 (Closed) LER 05000333/2015-005-00: Damper Failure Leads to Secondary Containment Vacuum below Technical Specification Limit (1 sample)

a. Inspection Scope

On September 18, 2015, secondary containment pressure became positive for approximately three minutes due to a malfunction of the 'A' reactor building ventilation system. Specifically, the discharge damper for the 'A' refuel floor exhaust fan failed partially closed, without initiating the automatic transfer to the 'B' refuel floor exhaust train in service (that is, the exhaust fan running with its discharge damper open), as should have occurred in this condition. As a result, the partially obstructed exhaust flow from the reactor building caused pressure to increase above the TS limit of 0.25 inches of vacuum water gauge.

Operators were alerted to the condition when control room annunciator 09-75-1-29, "Exhaust from Refuel Floor Air Flow Low," alarmed. Operators initially started both trains of the SBGTS, which restored secondary containment pressure to within the TS limit. Operators subsequently secured the 'A' refuel floor exhaust train and placed the 'B' train in service. The cause of this event was a failed diaphragm in one of the two air operators for the 'A' train discharge damper, combined with an inadequate setup of a damper position switch which prevented the automatic transfer to the 'B' refuel floor exhaust train in service. This LER is closed.

b. Findings

Introduction. The inspectors identified a self-revealing violation of TS 5.4, "Procedures," for FitzPatrick staff's failure to perform adequate PMT following maintenance on a limit switch in the reactor building ventilation system in August 2014, that, along with another unrelated component failure in the reactor building ventilation system, resulted in secondary containment pressure, relative to the outside pressure, exceeding the TS limit of 0.25 inches of vacuum water gauge.

Description. There are two trains of reactor building ventilation exhaust from the refuel floor; during operation, one train is running and the other is in auto standby. Each train consists of a fan and an air operated exhaust damper. The exhaust dampers consist of an upper and lower damper section, each of which has its own air operator. These operators use air to open and spring pressure to close. The two damper operators are connected by a connecting rod to ensure they operate together. Each damper has two limit switches, one to indicate not full open (NFO) and the other to indicate not full closed. These switches provide damper position indication and input to the system control logic. When an exhaust fan receives a start signal, its exhaust damper also receives an open signal, which causes air to be applied to the two operators. The fan is interlocked to the damper such that the fan will not start until the damper is fully open (that is, when the NFO limit switch is open). If the operating exhaust fan trips, air is vented from the operators for the associated exhaust damper, and spring pressure causes the damper to close. This causes the associated NFO limit switch to close, which sends a start signal to the standby train.

In the case at hand, the 'A' train was in operation and the 'B' train was in standby. On September 18, 2015, one of the air operators for the 'A' exhaust damper lost air pressure due to a diaphragm failure. Its spring pressure acted to try to close the damper, while the other, intact air operator, continued to try to maintain it open. As a result, the damper partially closed, which restricted ventilation exhaust flow. When the partial damper closure occurred, the 'A' NFO limit switch should have closed, causing exhaust fan 66FN-13A to trip and initiating the start sequence for the 'B' exhaust train. However, this did not occur, and the issue was entered into the CAP as CR-JAF-2015-04166.

FitzPatrick's root cause evaluation for this event identified that approximately one year earlier, on August 17, 2014, operators had found that the reactor building refuel floor exhaust ventilation apparently had automatically swapped in-service trains. Although the cause of the swap was not known, the fact that there had been no noticeable impact from this transfer supports that the exhaust ventilation control logic had functioned properly. Troubleshooting to determine the cause of this swap was performed under WO 00391716, utilizing procedure EN-MA-125, “Troubleshooting Control of Maintenance Activities.” The closure notes for this WO indicated that ‘A’ NFO limit switch had been replaced. However, the WO did not document what process or procedure had been used to replace and set up the switch. Additionally, no PMT was documented in the WO.

On August 28, 2014, a loud noise was identified to be coming from the refuel floor exhaust fans. Investigation revealed that the exhaust damper for the in-service train was not fully open but that the ‘A’ supply fan was still running. This supports that the ‘A’ NFO limit switch had not been set up properly 11 days earlier, since partial closure of the exhaust damper should have operated the limit switch and initiated a transfer to the standby exhaust ventilation train.

The inspectors concluded that the cause of the September 18, 2015, secondary containment positive pressure event was that maintenance performed in August 2014 on the ‘A’ train NFO limit switch was not identified to be inadequate during PMT. As a result, the imbalance between supply and exhaust ventilation flow, caused by the partially closed ‘A’ exhaust damper, caused secondary containment pressure, relative to the outside pressure, to increase above the TS SR 3.6.4.1.1 limit of greater than or equal to 0.25 inch of vacuum water gauge. This issue was entered into FitzPatrick’s CAP as CR-JAF-2015-04166.

Analysis. The reactor building ventilation system is a non-safety class system and, therefore, is not subject to the requirements of 10 CFR 50, Appendix B. Nevertheless, the inspectors determined that Entergy’s failure to perform adequate PMT on reactor building refuel floor exhaust damper ‘A’ NFO limit switch in August 2014, which, in combination with a failure of ‘A’ reactor building refuel floor exhaust damper actuator on September 18, 2015, resulted in secondary containment exceeding its TS limit, was a performance deficiency that was within FitzPatrick’s ability to foresee and correct, and should have been prevented. This finding was more than minor because it was associated with the configuration control attribute of the Barrier Integrity cornerstone and 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, as a result of this event, secondary containment was not preserved, in that secondary containment pressure exceeded the limit of TS SR 3.6.4.1.1.

In accordance with IMC 0609.04, “Initial Characterization of Findings,” and Exhibit 3 of IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” the inspectors determined that this finding was of very low safety significance (Green) because the performance deficiency was not a pressurized thermal shock issue, did not represent an actual open pathway in the physical integrity of the reactor containment, did not involve an actual reduction in function of hydrogen igniters in the reactor containment, and only represented a degradation of the radiological barrier function provided by the reactor building and SBGTS.

The finding had a cross-cutting aspect in the area of Human Performance, Resources, because FitzPatrick staff did not ensure that procedures for PMT of the reactor building refuel floor exhaust damper limit switch 66PNS-106A1 following maintenance performed in August 2014, were adequate to support the nuclear safety function of the secondary containment [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 (RG) 1.33, Appendix A, November 1972. RG 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, documented instructions, or drawings appropriate to the circumstances." RG 1.33, Appendix A, November 1972, Section D, "Procedures for Startup, Operation, and Shutdown of Safety Related Boiling Water Reactor Systems," includes the reactor building heating and ventilation system as such a system.

Contrary to the above, in and around August 2014, maintenance that affected the performance of safety-related equipment was not appropriate to the circumstances. Specifically, PMT for maintenance performed in August 2014 on reactor building ventilation system refuel floor exhaust damper 'A' NFO limit switch did not identify that the limit switch would not perform its functions to shut down 'A' refuel floor exhaust fan and initiate startup of the 'B' refuel floor exhaust train, in the event that the 'A' refuel floor exhaust train was in service and its exhaust damper was not fully open. As a result, on September 18, 2015, secondary containment pressure, relative to the outside pressure, exceeded the TS SR 3.6.4.1.1 limit of 0.25 inches of vacuum water gauge when the 'A' exhaust damper partially closed due to failure of one of its two air operators. This caused secondary containment to be inoperable for a period of approximately three minutes. Because this issue was of very low safety significance (Green) and Entergy entered this issue into their CAP as CR-JAF-2015-04166, this finding is being treated as an NCV, consistent with the NRC Enforcement Policy. (**NCV 05000333/2016001-03, Inadequate Post-Maintenance Testing of the Reactor Building Ventilation System Resulted in Short-Term Inoperability of Secondary Containment)**

.4 (Closed) LER 05000333/2015-006-00 and 05000333/2015-006-01: Transitory Secondary Containment Differential Pressure Excursions (2 samples)

a. Inspection Scope

On September 22, 2015, at 5:03 p.m., during a surveillance test that involved an automatic isolation of secondary containment and initiation of the SBGTS, operators noted that secondary containment pressure, relative to the outside pressure (d/p) became positive for a brief period (approximately 10 seconds) during the transition. TS SR 3.6.4.1.1 requires secondary containment vacuum be maintained greater than or equal to 0.25 inch of vacuum water gauge. Subsequent investigation indicated that the observed short duration pressure transient had actually occurred, and that similar events had occurred on 12 occasions over the previous three years. This LER and its revision are closed.

b. Findings

Introduction. The inspectors identified an SL IV NCV of 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," because unplanned inoperability of the secondary containment system was not reported to the NRC within eight hours of the occurrence, as required by 10 CFR 50.72(b)(3)(v), "Event or Condition that Could Have Prevented Fulfillment of a Safety Function." Specifically, FitzPatrick staff did not promptly report that reactor building d/p briefly dropped below the TS required minimum value of 0.25 inches of vacuum water gauge which caused the secondary containment system to be inoperable.

Description. Following the September 22, 2015, reactor building pressure transient, FitzPatrick staff questioned whether the observed condition was due to an issue with their instrumentation, or if secondary containment pressure was actually responding in this rapid manner. The issue was entered into the CAP as CR-JAF-2015-04198. On October 18, 2015, engineering staff responded that this event had been an expected response to isolating the reactor building, as previously documented in CR-JAF-2014-07227, and that corrective action was to be a license amendment to address the transitory secondary containment pressure response that is observed during isolation of the reactor building. On October 29, 2015, operations staff monitored all available indications associated with reactor building d/p and determined that all indications responded similarly. On November 3, 2015, FitzPatrick staff concluded that reactor building d/p instrumentation was, in fact, indicating accurately, and notified the NRC of the September 22, 2015, event in accordance with 10 CFR 50.72(b)(3)(v)(C). Failure to meet the 10 CFR 50.72 reporting time requirement was entered into the CAP as CR-JAF-2015-04893.

The subject LER acknowledged that additional time would be required to identify the cause of the event and to determine the number of past similar occurrences. Revision 1 to the subject LER reported that the cause of the event was the difference in design closure times for the reactor building ventilation system supply and exhaust isolation valves. The exhaust valves close within five seconds and the supply valves close within 15 seconds; since the supply fans operate in both the normal and recirculation modes of operation, the pressure in secondary containment increases during the transition, while the exhaust valves are closed and the supply valves are open. If a train of the SBGT system is already in operation, as is the case during a planned isolation of the reactor building, the pressure increase is mitigated. However, in the case of an automatic reactor building isolation, the SBGT system is not already in operation and secondary containment d/p may exceed the TS limit during the transition. FitzPatrick staff identified 12 occurrences during the past three years when secondary containment d/p exceeded the TS limit; all of these were during performance of surveillance tests that simulate automatic reactor building isolations.

FitzPatrick staff concluded that these events had no significant safety impact. The short period that secondary containment d/p exceeded the TS limit would limit exfiltration to a very low level. In the case of a design basis loss of coolant accident, reactor building isolation would occur early in the event, prior to any postulated fuel damage. And, in the case of a design basis refueling accident, Engineering Evaluation JAF-SE-96-071, "Impact of Increased Isolation Time of Reactor Building Ventilation System on FSAR

Analyzed Events," demonstrated that the amount of exfiltration would result in control room and offsite doses that were below regulatory limits. The inspectors determined that these conclusions were reasonable.

The inspectors determined that sufficient information was available to FitzPatrick staff to recognize the need to report the event per 10 CFR 50.72(b)(3)(v), "Event or Condition that Could Have Prevented Fulfillment of a Safety Function," before the actual report date of November 3, 2015. Although the inspectors determined that reasonable question existed immediately after the event as to the validity of the indications, by October 18, 2015, engineering staff had presented sufficient evidence that the condition was real and had been previously addressed. Nonetheless, when the validity of the indications were again demonstrated 11 days later, it still took FitzPatrick staff an additional five days to report the condition in accordance with 10 CFR 50.72.

Analysis. Due to the low safety significance of these events, the inspectors concluded that the failures to comply with TS SR 3.6.4.1.1 did not constitute violations of regulatory requirements because, in all cases, secondary containment was restored within the TS allowed outage time. However, the inspectors determined that the failure to inform the NRC of the secondary containment system inoperability within eight hours in accordance with 10 CFR 50.72(b)(3)(v) was a performance deficiency reasonably within Entergy's ability to foresee and correct. The inspectors evaluated this performance deficiency in accordance with the traditional enforcement process 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. 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 did not make a report required by 10 CFR 50.72 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.

Enforcement. 10 CFR 50.72(b)(3)(v)(C) requires, in part, that licensees shall notify the NRC within eight hours of the occurrence of any event or condition that at the time of discovery could have prevented the fulfillment of a safety function of structures or systems that are needed to control the release of radioactive material. Contrary to this, between September 22, 2015, at 5:03 p.m., and November 3, 2015, at 4:19 p.m., Entergy did not notify the NRC within eight hours of the occurrence of a condition that at the time of discovery could have prevented the fulfillment of a safety function of structures or systems needed to control the release of radioactive material. Specifically, positive pressure in secondary containment that occurred during transition between normal reactor building in service and the reactor building being isolated was not promptly identified by Entergy personnel as a condition that was reportable to the NRC within eight hours in accordance with 10 CFR 50.72(b)(3)(v)(C) and consequently was not reported until 4:19 p.m. on November 3, 2015, a period of approximately 42 days. Because this SL IV violation was of very low safety significance (Green), was not repetitive or willful, and was placed in the CAP as CR-JAF-2015-05244 and CR-JAF-2015-05265, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000333/2016001-04, Untimely 10 CFR 50.72 Notification of Inoperable Secondary Containment)**

.5 (Closed) LER 05000333/2014-002-01: Secondary Containment Vacuum below Technical Specification Limit (1 sample)

On October 28, 2014, secondary containment d/p 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 system lineup. The first instance occurred while isolating the reactor building and placing the SBGTS in service; as discussed in Section 4OA3.4 above, secondary containment d/p may exceed the TS limit during this transition. The second instance occurred shortly thereafter, when reactor building ventilation was being restored to service. The cause of this occurrence was attributed to failure of the 'A' exhaust fan discharge damper upper air operator due to a failed air piston diaphragm.

Revision 1 to this LER was written as a result of information gained through investigation of a similar failure of the reactor building ventilation system, documented in LER 2015-005-00 and reviewed in Section 4OA3.3 of this report. This information indicated that the cause of the 2014 issue (that is, the subject of LER 2014-002) was the same as the 2015 issue (LER 2015-005), that being failure of one of the air operators for the 'A' exhaust damper, combined with inadequate setup of the 'A' exhaust damper NFO limit switch.

Enforcement aspects with respect to the late reporting of the LER 2014-002 event were addressed in NRC Integrated Inspection Report 05000333/2014005, Section 1R15, while enforcement aspects with respect to the inadequate setup of the 'A' exhaust damper NFO limit switch are addressed in Section 4OA3.3 of this report. In reviewing this LER supplement, the inspector did not identify any new technical or regulatory issues, therefore, LER 2014-002-01 is closed.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 21, 2016, 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, Regulatory Assurance
J. Richardson, Manager, Systems and Components Engineering
W. Drews, Manager, Regulatory Assurance
R. Heath, Manager, Radiation Protection
J. Jones, Manager, Emergency Planning
T. Peter, Director, Regulatory and Performance Improvement
D. Poulin, Director, Engineering
T. Redfearn, Manager, Security
M. Reno, Manager, Training
T. Restuccio, Manager, Operations
S. Vercelli, General Manager, Plant Operations

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Open/Closed

05000333/2016001-01	NCV	Unintended HPCI Pump Suction Transfer during Pressure Control Mode Operation (Section 4OA3)
05000333/2016001-02	NCV	Uncontrolled RPV Level Increase after Initiation of RHR Shutdown Cooling (Section 4OA3)
05000333/2016001-03	NCV	Inadequate Post-Maintenance Testing of the Reactor Building Ventilation System Resulted in Short-Term Inoperability of Secondary Containment (Section 4OA3)
05000333/2016001-04	NCV	Untimely 10 CFR 50.72 Notification of Inoperable Secondary Containment (Section 4OA3)

Closed

05000333/2014-002-01	LER	Secondary Containment Vacuum Below Technical Specification Limit (Section 4OA3)

05000333/2015-004-00	LER	Concurrent Opening of Reactor Building Airlock Doors (Section 4OA3)
05000333/2015-005-00	LER	Damper Failure Leads to Secondary Containment Vacuum Below Technical Specification Limit (Section 4OA3)
05000333/2015-006-00 and 05000333/2015-006-01	LER	Transitory Secondary Containment Differential Pressure Excursions (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
AOP-13, "Severe Weather," Revision 25
AOP-56, "Intake Water Level Trouble," Revision 11
OP-4, "Circulating Water System," Revision 75

Section 1R04: Equipment Alignment

Documents
DBD-014, "Design Basis Document for the Core Spray System," Revision 10

Procedures
ODSO-4, "Shift Turnover and Log Keeping," Revision 12
OP-13, "Residual Heat Removal System," Revision 97
OP-14, "Core Spray System," Revision 36
OP-21, "Emergency Service Water," Revision 38
OP-22, "Diesel Generator Emergency Power," Revision 60
OP-37, "Containment Atmosphere Dilution System," Revision 82
OP-60, "Diesel Generator Room Ventilation," Revision 8

Condition Reports

CR-JAF-2014-00651	CR-JAF-2014-05480	CR-JAF-2015-03478
CR-JAF-2014-00844	CR-JAF-2014-07086	CR-JAF-2015-03482
CR-JAF-2014-02037	CR-JAF-2015-00865	CR-JAF-2016-00380
CR-JAF-2014-03337	CR-JAF-2015-02149	CR-JAF-2016-00661
CR-JAF-2014-04367	CR-JAF-2015-03309	CR-JAF-2016-00679

Work Orders
WO 00380981
WO 00402224

Section 1R05: Fire Protection

Documents
JAF-RPT-04-00478, "JAF Fire Hazards Analysis," Revision 2

Procedures
PFP-PWR01, “East Cable Tunnel / Elevation 258 Foot Fire Area/Zone II/CT-2,” Revision 3
PFP-PWR12, “Relay Room Elevation 286 Foot Fire Area VII/Fire Zone RR-1,” Revision 5
PFP-PWR20, “Reactor Building - East / Elevation 272 Foot Fire Area/Zone IX/RB-1A,” Revision 5
PFP-PWR21, “Reactor Building - West / Elevation 272 Foot Fire Area/Zone X/RB-1B,” Revision 5
PFP-PWR24, “Reactor Building - East / Elevation 300 Foot Fire Area/Zone IX/RB-1A, VIII/RB-1C,” Revision 5
PFP-PWR25, “Reactor Building - West / Elevation 300 Foot Fire Area/Zone X/RB-1B, VIII/RB-1C,” Revision 3
PFP-PWR23, “Motor Generator Set Room / Elevation 300 Foot Fire Area/Zone IA/MG-1,” Revision 5
FPP-3.56, “Portable Fire Extinguisher Inspection Procedure,” completed November 16, 2015
ST-16JT1, “Control Room and Relay Room Emergency Lighting Test,” completed January 23, 2016

Work Orders
WO 00436210

Section 1R11: Licensed Operator Requalification Program and Licensed Operator Performance

Procedures
AOP-6, “Malfunction of EHC Pressure Regulator,” Revision 8
AOP-41, “Feedwater Malfunction,” Revision 11
AOP-59, “Loss of RPS Bus A Power,” Revision 8
EN-RE-215, “Reactivity Maneuver Plan,” Attachment 9.4, dated January 22, 2016
EOP-2, “RPV Control,” Revision 9
EOP-3, “Failure to Scram,” Revision 10
EOP-4, “Primary Containment Control,” Revision 8
EOP-5/6, “Secondary Containment Control / Radioactivity Release Control,” Revision 8
EP-3, “Backup Control Rod Insertion,” Revision 11
EP-4, “Boron Injection Using CRD System,” Revision 3
OP-65, “Startup and Shutdown Procedure,” Revision 120

Section 1R12: Maintenance Effectiveness

Procedures
EN-DC-205, “Maintenance Rule Monitoring,” Revision 5
OP-40, “Reactor Building Closed Loop Cooling,” Revision 51
OP-51A, “Reactor Building Ventilation and Cooling System,” Revision 50

Documents
EN-DC-203, “Maintenance Rule Program,” Revision 3
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
JAF-RPT-RBCLC-02809, “Maintenance Rule Basis Document for System 015 RBCLC,” Revision 6

JAF-RPT-RBC-02295, “Maintenance Rule Basis Document System 066 Reactor Building Ventilation System,” Revision 4
JENG-16-0002, “Evaluation of the Reactor Building Above Refuel Train “A” for (a)(1) Status,” dated March 2, 2016
System Health Reports for Reactor Building Ventilation System, Second and Third Quarters 2015, and First through Fourth Quarters 2014
System Health Reports for the Reactor Building Closed Loop Cooling System, Fourth Quarter 2013, Second Quarter 2014, and Second Quarter and Fourth Quarter 2015

Condition Reports

CR-JAF-2014-02521	CR-JAF-2014-06888	CR-JAF-2015-05433
CR-JAF-2014-03231	CR-JAF-2015-00009	CR-JAF-2016-00081
CR-JAF-2014-03986	CR-JAF-2015-00245	CR-JAF-2016-00089
CR-JAF-2014-04168	CR-JAF-2015-01118	CR-JAF-2016-00115
CR-JAF-2014-04535	CR-JAF-2015-02342	CR-JAF-2016-00123
CR-JAF-2014-04984	CR-JAF-2015-03415	CR-JAF-2016-00161
CR-JAF-2014-05019	CR-JAF-2015-04166	CR-JAF-2016-01155
CR-JAF-2014-05195	CR-JAF-2015-04227	
CR-JAF-2014-05421	CR-JAF-2015-04376	

Maintenance Rule Functional Determinations for CRs

CR-JAF-2014-03986	CR-JAF-2014-06498	CR-JAF-2015-04166
CR-JAF-2014-04168	CR-JAF-2015-00285	CR-JAF-2015-05244
CR-JAF-2014-04535	CR-JAF-2015-03260	CR-JAF-2016-00089
CR-JAF-2014-05673	CR-JAF-2015-04201	

Work Orders

WO 00392756	WO 00425478	WO 00434802
WO 00397029	WO 00427026	WO 51192659

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
AP-10.10, “On-Line Risk Assessment,” Revision 9
EN-OP-119, “Protected Equipment Postings,” Revision 7
EN-WM-104, “On Line Risk Assessment,” Revision 12

Section 1R15: Operability Determinations and Functionality Assessments

Documents
EC 62923, Concerning the Operability Impact of the Lack of a Mechanical Interlock Between the 41-1O and 42-1C Contactors in the 23MOV-20 Breaker Control Circuit
EC 63725 Concerning Operability of 02RV-71E, ADS Main Steam Line C Safety/Relief Valve
EN-LI-118-08, Attachment 9.1, Failure Mode Analysis for Breaker 10042, “71PCB-10042 Did Not Operate or Indicate Position As Expected”
EN-OP-111 Attachment 9.2, “Operational Decision-Making Issue Regarding 71PCB-10042 (Fitz/Scriba 10 345 kV Line Circuit Breaker)”
Engineering White Paper, “Justification to Support Confidence in Fast Transfer Circuitry Post Replacement of the 10042 Auxiliary Contact Switches”
SEP-IST-007, “Inservice Testing for Pumps and Valves Fourth Ten-Year Interval,” Revision 7

Procedures
ST-25BB, “CAD System B Quarterly Operability Test (IST),” completed November 27, 2015
ST-40D, “Daily Surveillance and Channel Check,” Revision 110

Condition Reports

CR-JAF-2015-04427	CR-JAF-2016-00590	CR-JAF-2016-01057
CR-JAF-2016-00244	CR-JAF-2016-00697	
CR-JAF-2016-00493	CR-JAF-2016-01005	

Section 1R18: Plant Modifications

Procedures
OP-37, “Containment Atmosphere Dilution System,” Revision 82

Condition Reports
CR-JAF-2016-00661
CR-JAF-2016-00679

Section 1R19: Post-Maintenance Testing

Procedures
IMP-71.27, “Analog Transmitter Trip System Power Supply Failure Annunciator Test*,” Revision 3, completed March 28, 2016
MP-027.04, “Valtek Mark I, II, and Mark IV Valve Maintenance,” Revision 7, completed February 18, 2016
MP-027.05, “Valtek Spring Cylinder and Manual Actuator Maintenance,” Revision 4, completed February 18, 2016
MP-055.01, “600 V Air Circuit Breakers,” Revision 44, completed January 15, 2016
MP-066.01, “Unit Cooler Maintenance*,” Revision 9
ST-4M, “HPCI Torus Suction Operability Test,” Revision 19, completed February 12, 2016
ST-8Q, “Testing of the Emergency Service Water System (IST),” Revision 46, completed February 28, 2016
ST-25BB, “CAD System B Quarterly Operability Test (IST),” Revision 4, completed March 15, 2016
ST-41D, “Remote Valve Position Indication Verification Online (IST),” Revision 20, completed March 15, 2016

Work Orders

WO 00426173	WO 51193117	WO 52552759
WO 00432095	WO 52386816	
WO 00441200	WO 52467267	

Section 1R22: Surveillance Testing

Procedures
ISP-8A, “Above Core Plate to Core Spray Line at RPV Differential Pressure Instrument Functional Test/Calibration,” Revision 3
ISP-16, “Drywell Floor Drain Sump Flow Loop Functional Test/Calibration*,” Revision 44
ISP-100C-RPS, “RPS Instrument Functional Test/Calibration**,” Revision 41
ST-6HA, “Standby Liquid Control A Side Quarterly Operability Test (IST),” Revision 7

ST-9AB, "EDG System B Fuel Oil Monthly Test," Revision 4
ST-9BB, "EDG B and D Full Load Test and ESW Pump Operability Test," Revision 15

Condition Reports
CR-JAF-2016-00926
CR-JAF-2016-00929

Section 4OA1: Performance Indicator Verification

EN-LI-114, "Performance Indicator Process," Revision 7
JAF-SE-96-071, "Impact of Increased Isolation Time of Reactor Building Ventilation System on FSAR Analyzed Events," Revision 2

Section 4OA2: Problem Identification and Resolution

Documents
5321-X-202B/G, "Vent Pipe Penetration Thermal Growth Piping Analysis," Revision 1
733, Pipe Stress Reanalysis Program - 11825-MSK-168A1, Revision 0
JAF-CALC-CAD-04450, "Shaft Breakaway Torque, Corresponding to 0.5 psid for Vacuum Breakers 27VB-1 thru 5," Revision 0
DBD-016A, "Primary Containment Penetrations and Isolation Devices Design Basis Document," Revision 5
DBD-027, "Air Treatment Systems Design Basis Document," Revision 11
CR-JAF-2012-00057 CA 1, "Torus Downcomer Vacuum Breaker Equipment Failure Evaluation," dated January 18, 2012
CR-JAF-2014-06207 CA 5, "Long Term CA Classification Evaluation," dated January 8, 2015
EC 53544, "One Time Change to Procedure ST-15J, 'Torus to Drywell Vacuum Breakers Quarterly Test (IST)'," dated October 24, 2014
EC 58839, "CR-JAF-2015-03183 Operability Input," dated July 6, 2015
EN-DC-205 Attachment 9.1, "CR-JAF-2014-06207 Functional Failure Determination," dated January 2, 2015
EN-LI-118 Attachment 9.1, "CR-JAF-2014-06207 Failure Modes Analysis," dated October 10, 2014
WT-WTJAF-2014-0026 CA 93, "Engineering Evaluation of Addition of Alarm Set Point on Low Drywell to Torus Differential Pressure," dated December 11, 2014
NRC Information No. 97-16,"Preconditioning of Plant Structures, Systems, and Components before ASME Code Inservice Testing or Technical Specification Surveillance Testing," dated April 4, 1997
NUREG 1482, "Guidelines for Inservice Testing at Nuclear Power Plants," Revision 2
AR 214419, "Torus Downcomer Vacuum Breaker PM Change Request," performed March 24, 2015
CR-JAF-2014-06207 CA 4, "Drywell to Torus Differential Pressure ODMI Implementation Action Plan," dated November 3, 2014
CR-JAF-2014-06207 CA 5, "Due Date Extension Request," dated January 5, 2015
EN - Valve - Check - Various PM Basis Template, dated February 24, 2010
Operations Narrative Log for the Period January 13, 2016, through January 19, 2016, and February 25, 2016, through March 2, 2016
CAD ODMI Trend Data, for the period October 11, 2014, through February 25, 2016
Primary Containment Atmosphere Control and Dilution System Health Report, Second Quarter 2015 and Further Quarter 2015

ST-15J Results for 27VB-1, for the Period November 25, 2007, through December 20, 2015
A585-0317, “Instruction Manual for Vacuum Breaker Valves W/Disc Modification,” dated December 9, 1988

Procedures
ARP 09-3-3-39, “Torus to DW Vac Bkr Vlv Open,” Revision 2
ARP 09-5-1-34, “DW Press Alarm Hi or Lo,” Revision 5
ARP 27-CAD-2, “Liquid N2 Tk 7A Lvl Lo,” Revision 2
EN-LI-102, “Corrective Action Program,” Revision 25
EN-LI-118, “Cause Evaluation Process,” Revision 22
MP-007.04, “Containment Vacuum Breakers,” Revision 5
OP-37, “Containment Atmosphere Dilution System,” Revision 82
ST-15J, “Torus to Drywell Vacuum Breaker Quarterly Test (IST),” performed September 30, 2015, and December 20, 2015
ST-39E, “Pressure Suppression Chamber - Drywell Vacuum Breaker Leak Test (IST),” performed May 19, 2014, and February 2, 2016
ST-40D, “Daily Surveillance and Channel Check,” performed December 20, 2015, through January 3, 2016, and January 31, 2016, through February 13, 2016

Drawings
11825-6.44-16m “30” Vacuum Breaker Valve,” Revision F
FM-18B, “Drywell Inerting C.A.D. Purge and Containment Differential Pressurization System 27 Flow Diagram,” Revision 40
FM-20B, “Residual Heat Removal System 10 Flow Diagram,” Revision 72
JAF Dwg 6.44-51, “Vacuum Breaking Valve (Atwood and Morrill Drawing 21755-H),” Revision C

Condition Reports
CR-JAF-2010-02187
CR-JAF-2010-07059
CR-JAF-2010-08022
CR-JAF-2012-00057
CR-JAF-2012-01482
CR-JAF-2012-04272
CR-JAF-2014-02061
CR-JAF-2014-06207
CR-JAF-2014-07087
CR-JAF-2014-07095
CR-JAF-2015-00213
CR-JAF-2015-03183
CR-JAF-2016-00043
CR-JAF-2016-00086
CR-JAF-2016-00110
CR-JAF-2016-00117
CR-JAF-2016-00176
CR-JAF-2016-00183
CR-JAF-2016-00241
CR-JAF-2016-00243
CR-JAF-2016-00245
CR-JAF-2016-00246
CR-JAF-2016-00257
CR-JAF-2016-00260
CR-JAF-2016-00263
CR-JAF-2016-00265
CR-JAF-2016-00280
CR-JAF-2016-00313
CR-JAF-2016-00322
CR-JAF-2016-00324
CR-JAF-2016-00357
CR-JAF-2016-00369
CR-JAF-2016-00411
CR-JAF-2016-00554
CR-JAF-2016-00612
CR-JAF-2016-00619
CR-JAF-2016-00707
CR-JAF-2016-00732
CR-JAF-2016-00898
CR-JAF-2016-00914
CR-JAF-2016-00918
CR-JAF-2016-00934
CR-JAF-2016-00935
CR-JAF-2016-00938
CR-JAF-2016-00939
CR-JAF-2016-00942
CR-JAF-2016-00951

Work Orders
WO 00396190
WO 00397752
WO 00397546
WO 00397751
WO 00397754
WO 00398941

LIST OF ACRONYMS

10 CFR	Title 10 of the *Code of Federal Regulations*
CAD	containment atmosphere dilution
CAP	corrective action program
CR	condition report
CST	condensate storage tank
EDG	emergency diesel generator
HCU	hydraulic control unit
HPCI	high pressure coolant injection
IMC	Inspection Manual Chapter
LER	licensee event report
MG	motor generator
NCV	non-cited violation
NEI	Nuclear Energy Institute
NFO	not full open
NRC	Nuclear Regulatory Commission, U.S.
ODMI	operational decision-making issue
PI	performance indicator
PM	preventive maintenance
PMT	post-maintenance test
RCIC	reactor core isolation cooling
RG	regulatory guide
RHR	residual heat removal
RPS	reactor protection system
RPV	reactor pressure vessel
RWR	reactor water recirculation
SBGTS	standby gas treatment system
S/RV	safety/relief valve
SL	severity level
SR	surveillance requirement
SSC	structure, system, and component
TS	technical specification
UFSAR	Updated Final Safety Analysis Report
WO	work order