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

November 6, 2015

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - INTEGRATED INSPECTION REPORT 05000333/2015003

Dear Mr. Sullivan:

On September 30, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your James A. FitzPatrick Nuclear Power Plant (FitzPatrick). The enclosed inspection report documents the inspection results which were discussed on October 15, 2015, with you and other members of your staff.

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

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

In accordance with Title 10 of the *Code of Federal Regulations* (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 2
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000333/2015003
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.	05000333/2015003
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Scriba, NY
Dates:	July 1, 2015, through September 30, 2015
Inspectors:	E. Knutson, Senior Resident Inspector B. Sienel, Resident Inspector P. Kaufman, Senior Reactor Inspector R. Rolph, Health Physicist
Approved by:	Arthur L. Burritt, Chief Reactor Projects Branch 2 Division of Reactor Projects

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SUMMARY

Inspection Report 05000333/2015003; 07/01/2015 - 09/30/2015; James A. FitzPatrick Nuclear Power Plant (FitzPatrick); Operability Determinations and Follow-Up of Events.

This report covered a three-month period of inspection by resident inspectors and announced inspections performed by regional inspectors. The inspectors identified two findings of very low safety significance (Green), both of which were non-cited violations (NCVs). The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process (SDP)," dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated 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: Initiating Events

- Green. A self-revealing NCV of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified because FitzPatrick staff failed to correct a condition adverse to quality. Specifically, Entergy failed to take effective corrective actions for condition report (CR)-JAF-2010-00287 to replace the control rod drive (CRD) hydraulic control unit (HCU) directional control valve (DCV) bolting material which had signs of corrosion after the same material was identified through operational experience as the cause of a control rod drift. As a result, on July 19, 2015, FitzPatrick control rod 10-07 drifted from the fully withdrawn to the fully inserted position in the reactor core leading to an immediate power reduction from 100 to 99 percent followed by a manual rapid power reduction to 56 percent. Entergy's subsequent corrective actions included an extent of condition review and completed or planned replacement of all susceptible directional control valve bolting.

  The performance deficiency was determined to be more than minor because it was associated with the equipment performance attribute of the Initiating Events cornerstone, and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors determined that this finding was of very low safety significance (Green) using Exhibit 1 of IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, because the finding did not cause both a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition (e.g. loss of condenser, loss of feed water). The inspectors determined that there was no cross-cutting aspect associated with this finding because the cause of the performance deficiency occurred more than three years ago, and was not representative of current plant performance. (Section 1R15)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a self-revealing violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” because FitzPatrick staff failed to provide instructions appropriate to the reactor building roof replacement project. Specifically, inadequate instructions were provided to ensure that roofing material removal would be performed in slow, deliberate manner, such that its effect on secondary containment could be assessed and operability maintained. As a result, this activity caused secondary containment to be inoperable for a period in excess of its four hour technical specification (TS) allowed outage time. As immediate corrective action, roofing material removal was stopped and the new roofing materials were installed to reseal the affected area of the reactor building roof. Secondary containment vacuum was restored to greater than the TS-required minimum after a period of 92 minutes and secondary containment was declared operable after a period of five hours and 26 minutes. The issue was entered into the corrective action program (CAP) as CR-JAF-2015-03260.

  The finding was more than minor because it is associated with the procedure quality attribute of the Barrier Integrity cornerstone, and affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system (RCS), and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the work order (WO) did not provide adequate instruction to ensure that roofing material removal would be performed in slow, deliberate manner, coordinated between operations and maintenance personnel, and allowing adequate time after actions that could impact secondary containment such that their effect on secondary containment could be assessed and operability maintained. 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 standby gas treatment system. The finding had a cross-cutting aspect in the area of Human Performance, Avoid Complacency, because FitzPatrick staff did not adequately plan for the possibility of latent issues and inherent risk associated with the reactor building roof replacement project, such that the commencement of work resulted in a loss of secondary containment [H.12].
  (Section 4OA3)

REPORT DETAILS

Summary of Plant Status

FitzPatrick began the inspection period at 100 percent power. On July 18, 2015, operators reduced power to 65 percent to perform a control rod sequence exchange and turbine valve testing, and restored power to 100 percent. On July 19, 2015, operators reduced power to 56 percent in response to an unplanned insertion of control rod 10-07 from fully withdrawn to fully inserted, that occurred due to a drive water leak from its associated HCU. After the HCU was isolated and disarmed, operators restored reactor power to 100 percent the following day. The HCU was repaired and rod 10-07 was returned to fully withdrawn on July 23, 2015. Due to the possibility of common cause failure, short duration power reductions were performed on four subsequent occasions (August 6, August 27, September 18, and September 29, 2015), to support maintenance on small numbers of HCUs at a time (two to eight) while their associated control rods were fully inserted. FitzPatrick operated at or near 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

.1 Partial System Walkdown (71111.04 - 3 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- ‘A’ core spray system during planned maintenance on the ‘A’ low pressure coolant injection (LPCI) independent power supply on July 14, 2015
- ‘B’ containment atmosphere dilution (CAD) system during planned maintenance on the ‘A’ CAD system on September 8, 2015
- ‘A’ and ‘C’ emergency diesel generators (EDGs) during planned maintenance on reserve station service transformer 71T-3 and 115 kilovolt (kV) line 4 on September 16, 2015

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

b. Findings

No findings were identified.

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

a. Inspection Scope

On September 2, 2015, the inspectors performed a complete system walkdown of accessible portions of the high pressure coolant injection 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 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.

- Reactor building 369 foot elevation, fire area/zone IX/RB-1A, on July 10, 2015
- Safety related pump rooms, fire area/zone XII/SP-1, XIII/SP-2, on August 12, 2015
- Reactor building east crescent area, fire area/zone XVII/RB-1E, on August 31, 2015
- Reactor building 344 foot elevation, fire area/zone IX/RB-1A, on September 14, 2015
- Main control room and control room ventilation equipment rooms, fire area/zone VII/CR-1, on September 28, 2015

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

.1 Internal Flooding Review (1 sample)

a. Inspection Scope

The inspectors reviewed the UFSAR, the site flooding analysis, and plant procedures to assess susceptibilities involving internal flooding. The inspectors also reviewed the CAP to determine if Entergy staff identified and corrected flooding problems and whether operator actions for coping with flooding were adequate. The inspectors focused on the north and south cable tunnels to verify the adequacy of floor and water penetration seals, common drain lines, and flood barriers.

b. Findings

No findings were identified.

.2 Annual Review of Cables Located in Underground Bunkers/Manholes (1 samples)

a. Inspection Scope

The inspectors examined manhole MH-6A and MH-8A in the 345 kV switchyard during FitzPatrick staff's annual inspection of manhole sump pumps. These manholes contain non-safety class electrical cables that could affect the reliability of the 345 kV system. The inspectors verified that cable insulation was not visibly degraded. The inspectors observed that many of the cable trays and supports were corroded. These manholes can be subject to flooding because there are no sump high level alarms to alert operators to a sump pump failure, and there is a history of such failures. The degraded conditions in these manholes did not constitute a violation of regulatory requirements because the manholes do not contain safety-related equipment.

b. Findings

No findings were identified.

1R07 Heat Sink Performance

.1 Annual Review (71111.07A - 1 sample)

a. Inspection Scope

The inspectors reviewed the 'B' residual heat removal (RHR) system heat exchanger performance to determine its readiness and availability to perform its safety functions. This heat exchanger is cooled by the RHR service water (SW) system. The inspectors reviewed the design basis for the component and verified Entergy's commitments to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." The inspectors reviewed the results of the performance testing of this heat exchanger performed on August 26, 2014, and verified that Entergy staff initiated appropriate corrective actions for identified deficiencies.

b. Findings

No findings were identified.

.2 Triennial Review (71111.07T - 4 samples)

a. Inspection Scope

Based on risk ranking of safety-related heat exchangers, a review of past heat sink inspections, and recent operational experience, the inspectors selected the ultimate heat sink, which included emergency service water (ESW) system piping integrity and intake structure functionality and operation.

The inspectors also selected for review the inspection, cleaning, and performance testing methods and frequency used to ensure the heat removal capabilities for east crescent area unit cooler 66UC-22F, west electric bay unit cooler 67UC-16A, and west cable tunnel cooler 67E-11, and compared them to the FitzPatrick license commitments made in response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

Emergency Service Water System

ESW functions as the ultimate heat sink to provide cooling water flow from Lake Ontario to the safety-related heat exchangers during normal operation and loss of offsite power. The inspectors reviewed the system design to evaluate the adequacy of system monitoring and performance testing. The inspectors reviewed procedures, calculations, and design drawings to verify they were consistent with the design and licensing basis. The inspectors performed walkdowns of the control room panels to verify that the instrumentation that operators rely on for decision making was available and functional. The inspectors reviewed operation of the ESW system, which encompassed procedures for intake structure operation, abnormal operations, adverse weather conditions, and leak isolation.

To assess the structural integrity of the ESW piping and ensure that any piping or intake structure degradation was appropriately identified and dispositioned, the inspectors performed walkdowns of accessible areas of the intake area (including ESW pumps, strainers, and traveling screens) reviewed station procedures and interviewed engineering personnel. The inspectors reviewed a sample of non-destructive examination (NDE) records, photographs, structural engineering evaluations of through-wall pipe leaks, including completed or planned corrective actions to assess the structural integrity condition of the ESW and SW piping. The inspectors reviewed pipe inspection records and performed a walkdown of accessible areas containing the ESW/SW piping to ensure that any leakage or degradation was appropriately identified and dispositioned.

The inspectors reviewed operational and maintenance history, system health reports, and in-service testing results for adverse trends and to verify that the ESW/SW systems functioned as designed. In addition, the inspectors reviewed the monitoring and testing of interface valves between safety-related ESW and non-safety-related piping systems to ensure that adequate system flow is available post-accident consistent with design basis assumptions. Surveillance test results were reviewed to verify that the systems and components functioned as designed to verify that the minimum calculated flow rates were properly maintained to essential safety-related components and met the acceptance criteria in the UFSAR and in accordance with American Society of Mechanical Engineers Code requirements.

Heat Exchangers Cooled by Emergency Service Water / Service Water

The inspectors reviewed the procedures for maintaining the safety function of the selected heat exchangers and verified that the heat exchangers were effectively monitored by means of inspection, cleaning, and performance testing and verified that these activities were consistent with the Electric Power Research Institute NP-7552, "Heat Exchanger Performance Monitoring Guidelines," and accepted industry practices.

The inspectors reviewed heat exchanger performance test results, inspection data records, photographs of the as-found condition, specification sheets, and preventative maintenance activities to verify that the heat exchangers were maintained consistent with design assumptions in the heat transfer calculations associated with normal, accident, and transient conditions, the description of these components in the UFSAR and in accordance with TS requirements. The inspectors compared performance test results and inspection data sheets to the established acceptance criteria to verify that the results were acceptable and that operation was consistent with the design.

The inspectors reviewed heat exchanger tube eddy current test results to verify that the number of plugged tubes was properly controlled and bounded by the engineering analyses. The inspectors reviewed the inspection records to verify that degradation trends were consistent with industry standards, and provided reasonable assurance of continued operability. The inspectors walked down a sample of the heat exchangers to assess the material condition of the components. The inspectors also reviewed the methods implemented for controlling biotic fouling such as hypochlorite injection and monitoring for zebra mussel growth to verify that these controls were effectively implemented.

The inspectors reviewed a sample of CRs related to the selected coolers and ESW system to ensure that Entergy appropriately identified, characterized, and corrected problems related to these essential systems structures and components performance.

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 August 11, 2015, which included various feedwater system malfunctions, an unplanned high pressure coolant injection system start, and a manual reactor scram. The inspectors evaluated operator performance during the simulated event and verified completion of risk significant operator actions, including the use of abnormal and emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, implementation of actions in response to alarms and degrading plant conditions, and the oversight and direction provided by the control room supervisor. Additionally, the inspectors assessed the ability of the crew and training staff to identify and document crew performance problems.

b. Findings

No findings were identified.

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

a. Inspection Scope

On July 18, 2015, the inspectors observed control room operators during a control rod sequence exchange which required a reactor power reduction to approximately 65 percent. The inspectors observed crew briefs, reactivity manipulations using control rods and the reactor water recirculation system, and turbine valve testing. The inspectors observed crew performance to verify that procedure use, crew communications, and coordination of activities between work groups met established expectations and standards.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q - 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.

- RHR
- Crescent area unit coolers

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 6 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 ‘A’ LPCI independent power supply during the week of July 13, 2015
- Planned power reduction to 65 percent for control rod sequence exchange during the week of July 13, 2015
- Emergent maintenance on ‘A’ reactor protection system and planned power reduction to 85 percent during the week of August 24, 2015
- Emergent troubleshooting on the ‘B’ RHR heat exchanger SW outlet isolation valve, 10MOV-89B, during the week of August 31, 2015
- Planned maintenance on offsite 115 kV line 4 during the week of September 14, 2015
- Planned maintenance on offsite 115 kV line 3 during the week of September 28, 2015

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15 - 4 samples)

a. Inspection Scope

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

- CR-JAF-2015-03000 concerning the operability of the 'B' ESW system with non-safety-related flanges installed on safety-related east cable tunnel unit cooler 67E-14, on July 1, 2015
- CR-JAF-2015-03252 concerning the implication for operability of the remaining HCUs, given the failure of two fasteners for a flange on HCU 10-07 which allowed the associated fully withdrawn control rod to fully insert into the core and caused operators to reduce reactor power by over 40 percent in accordance with the applicable abnormal operating procedure, on July 20, 2015
- CR-JAF-2015-03260 concerning the restoration of operability of the secondary containment following its loss due to planned removal of roofing material, given that the surveillance that was to be performed to verify secondary containment operability was not performed, on July 21, 2015
- CR-JAF-2015-03793 concerning the impact of the inability of control room operators to fully close the 'B' RHR heat exchanger SW outlet isolation valve, 10MOV-89B, on operability of the 'B' RHR system in the case of a loss of offsite power coincident with a loss of coolant accident, on August 27, 2015

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

b. Findings

Introduction. A self-revealing NCV of very low safety significance (Green) of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified for Entergy's failure to correct a condition adverse to quality. Specifically, Entergy failed to take effective corrective actions for CR-JAF-2010-00287 to replace the CRD HCU DCV bolting material which had signs of corrosion after the same material was identified through industry operational experience as the cause of a control rod drift. As a result, FitzPatrick control rod 10-07 drifted from the fully withdrawn to the fully inserted position in the core on July 19, 2015, leading to an immediate power reduction from 100 to 99 percent followed by a manual rapid power reduction to 56 percent.

Description. FitzPatrick has 137 control rods in the reactor to control power. As needed, automatic or manual action is taken to scram, rapidly insert, or normally insert and withdraw control rods in the reactor core. Each control rod has a HCU which is made up of a scram accumulator (a water volume pressurized with nitrogen) and various valves and piping to control movement of the control rod. Included in these valves are four directional control valves which are solenoid operated and are opened or closed as needed to effect rod motion. Each of the four DCVs are mounted to the HCU piping manifold using four allen head cap screws. These fasteners secure the DCVs to the manifold to provide a pressure boundary between the reactor building/secondary containment and CRD drive water (during rod movement) or the RCS (during steady state operation).

In 2009, another U.S. nuclear power plant had two of four cap screws on one of their HCU DCVs shear off, allowing leakage into the reactor building and causing the associated control rod to drift into the core. That plant determined that the cause of the cap screw failures was stress corrosion cracking (SCC) on two of the four cap screws that attached the DCV to the HCU. The failed cap screws were made of American Society for Testing and Materials (ASTM) A574, a zinc-plated, high strength material known to be susceptible to SCC. These cap screws were also part of a design which left the threads of the screws exposed to the reactor building environment. At that time, 60 to 75 percent of the HCUs at FitzPatrick used this type of cap screw. The remainder either had the original style DCVs, which did not have the threads on the cap screws exposed, or had cap screws made of the then-current vendor recommended material, ASTM A193 Gr. B7, a cadmium plated, lower strength material that is not susceptible to SCC.

In response to the Industry Operational Experience, FitzPatrick Engineering performed a walkdown of all HCUs and identified four HCUs where the cap screws showed significant indications of corrosion. Corrosion was identified on cap screws for both the 121 and 122 DCVs, for a total of 8 DCV cap screws affected per HCU. CR-JAF-2010-00287 was initiated and the Engineering input to the operability evaluation concluded that the condition was a minor hardware deficiency. WO 00225414 was written to replace all of the cap screws on the 121 and 122 DCVs for HCU 50-31. The removed cap screws were to be examined by the Maintenance Inspection Group, who would refer indications to the Engineering Inspection Group to perform NDE if needed. Based on the results of the inspections, CRs and additional corrective actions would be initiated/developed as necessary. WO 00225414 stated that “no defect of any kind was found during visual exam of the 8 carbon steel bolts.” Based on this, no NDE was performed on the removed bolts, WOs for the three other HCUs were never created, and no further corrective action was taken or planned.

On July 19, 2015, while FitzPatrick was at 100 percent power, a control rod drift alarm was received in the control room during the performance of ST-20C, “Control Rod Operability for Fully Withdrawn Control Rods and HCU Cooling Water Supply Check Valve Reverse Flow Check (IST [inservice test]).” Control rod 10-07 drifted from the fully withdrawn to the fully inserted position. This caused reactor power to decrease from 100 to 99.27 percent. Operators entered AOP-27, “Control Rod Malfunction,” and performed a rapid power reduction to approximately 56 percent. The operator sent to the reactor building observed water spraying from the HCU 10-07 DCV, 03SOV-121, exhaust line location. Operators isolated the HCU and noted that two of the four cap screws on the DCV were broken, which created a RCS leakage path from the exhaust header to the reactor building.

Entergy issued CR-JAF-2015-03252 to document and evaluate the control rod drift. Entergy determined that the mechanistic root cause of the control rod drift was hydrogen induced SCC which weakened the DCV cap screw threads and resulted in a complete fracture of two of the four DCV cap screws. This allowed RCS water to spray into the reactor building and caused control rod 10-07 to drift into the core with no operator action. There are three conditions necessary for hydrogen induced SCC to occur: 1) susceptible material; 2) tensile stress, a given in this application both from original installation and normal DCV operation; and 3) hydrogen, not uncommon in this application due to the presence of water and most easily identified by the presence of corrosion on the surface of the cap screws. FitzPatrick personnel determined that a contributing cause was a lack of engineering technical rigor by not implementing corrective actions in 2010 that would have eliminated the known SCC failure mode for the HCU DCVs. The inspectors determined that Entergy's root cause determination was reasonable.

Analysis. The inspectors determined Entergy's failure to take appropriate corrective actions for a condition adverse to quality in 2010 was a performance deficiency which was reasonably within Entergy's ability to foresee and correct and should have been prevented. In 2010, FitzPatrick staff neither performed NDE on the visually inspected bolts, which could have identified the presence of SCC, nor put in place corrective actions to periodically verify no SCC was present, nor performed a planned replacement of the susceptible bolting material. The performance deficiency was determined to be more than minor because it was associated with the equipment performance attribute of the Initiating Events cornerstone, and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors determined that this finding was of very low safety significance (Green) using Exhibit 1 of IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," because the finding did not cause both a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition (e.g. loss of condenser, loss of feed water). The inspectors determined that there was no cross-cutting aspect associated with this finding because the cause of the performance deficiency occurred more than three years ago, and was not representative of present plant performance.

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected." Contrary to the above, from January 21, 2010, until September 19, 2015, Entergy failed to promptly correct a condition adverse to quality, that being HCU DCV bolts that were susceptible to stress corrosion cracking, after the issue was identified by them through operating experience in 2010. Consequently, on July 19, 2015, the failure of two of these bolts caused pressure boundary leakage and control rod 10-07 to drift from the fully withdrawn to the fully inserted position in the reactor core and led to an immediate power reduction from 100 to 99 percent followed by a manual rapid power reduction to 56 percent. FitzPatrick personnel replaced the failed bolts, entered the issue into the CAP, completed an extent of condition review, and commenced a complete replacement of the susceptible bolting material on all HCU DCVs. Because this violation was of very low safety significance (Green), and Entergy has entered this performance deficiency into the CAP as CR-JAF-2015-03252, the NRC is treating this as an NCV in accordance with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000333/2015003-01, Inadequate Corrective Actions Result in Control Rod Drift and Reactor Power Reduction)**

1R18 Plant Modifications (71111.18 - 1 sample)

Temporary Modifications

a. Inspection Scope

The inspectors evaluated Temporary Modification engineering change (EC) 55301, "Level Switch 38LS-22A is Dysfunctional, Instead Use 38LS-22B to "Dump" Off Gas Drip Pot Until Issue With 38LS-22A is Remedied." This condition was causing off gas drip pot low level alarms in the control room, which would require an operator to cycle a manually operated valve to avoid degrading condenser vacuum due to air in-leakage. Repair was not practical due to the high radiation levels in the area of the main condenser during plant operation. However, the need to perform the manual draining three times per shift was projected to result in a cumulative exposure of 2.5 Rem to the operators before repairs would be able to be performed during the next refueling outage. The purpose of this modification was to install a remotely operated solenoid drain valve to allow the drip pot level to be controlled from outside the high radiation area. The inspectors reviewed the EC to verify that the temporary modification did not degrade the design bases, licensing bases, and performance capabilities of the off gas system, and vacuum priming and air removal system.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19 - 5 samples)

a. Inspection Scope

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

- WO 52495646 to perform preventive maintenance on the 'A' LPCI inverter on July 15, 2015
- WO 00420116 to replace two cap screws for HCU 10-07 directional control valves 121 and 122; PMT was RAP 7.4.1, "Control Rod Scram Time Evaluation (IST)**," for control rod 10-07 on July 23, 2015
- WO 52387521 to replace the 'A' EDG A2 air start receiver relief valve on August 12, 2015
- WO 00422314, replacement of HCU 14-27 directional control valve cap screws; PMT was to perform an external leak check of the DCVs and to withdraw and insert rod 14-27 one step in accordance with OP-25, "Control Rod Drive Hydraulic System," Section G.22, "Electrically Rearming a CRD," on August 27, 2015

- WO 52360258-06, replacement of the 27PCV-116A controller as part of overhaul of the actuator for 27FCV-103A, CAD Train A Nitrogen Make-up Flow Control Valve on September 15, 2015

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 - 4 samples)

a. Inspection Scope

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

- ST-1L, “Main Turbine Control Valve Instrument Channel and Valve Operability Check,” on July 18, 2015
- ISP-100B-PCIS, “PCIS [primary containment isolation system] Instrument Functional Test/Calibration (ATTS [analog transmitter trip system]),” on July 31, 2015
- ST-6KB, “B SLC [standby liquid control] System Class 2 Piping Leakage Test (ISI [in-service inspection]) and Operability Test (IST),” on August 4, 2015
- ST-2AL, “RHR Loop A Quarterly Operability Test (IST),” on September 8, 2015

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

The inspectors reviewed Entergy’s performance in assessing and controlling radiological hazards in the workplace. The inspectors used the requirements contained in 10 CFR 20, TSs, applicable Regulatory Guides (RGs), and the procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors reviewed the performance indicators for the occupational exposure cornerstone, radiation protection program audits, and reports of operational occurrences in occupational radiation safety since the last inspection.

Radiological Hazard Assessment

The inspectors reviewed recent plant radiation surveys and any changes to plant operations since the last inspection to identify any new radiological hazards for onsite workers or members of the public.

Instructions to Workers

The inspectors reviewed several occurrences where a worker's electronic personal dosimeter alarmed. The inspectors reviewed Entergy's evaluation of the incidents, documentation in the CAP, and whether compensatory dose evaluations were conducted when appropriate.

Radiological Hazards Control and Work Coverage

The inspectors evaluated in-plant radiological conditions and performed independent radiation measurements during facility walkdowns and observation of radiological work activities. The inspectors assessed whether posted surveys, radiation work permits, worker radiological briefings, the use of continuous air monitoring, and dosimetry monitoring were consistent with the present conditions. The inspectors examined the control of highly activated or contaminated materials stored within the spent fuel pool and the posting and physical controls for selected high radiation areas, locked high radiation areas, and very high radiation areas to verify conformance with the occupational performance indicator.

Problem Identification and Resolution

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were identified at an appropriate threshold and properly addressed in the CAP.

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

a. Inspection Scope

The inspectors reviewed the control of in-plant airborne radioactivity and the use of respiratory protection devices in these areas. The inspectors used the requirements in 10 CFR 20, RG 8.15, RG 8.25, NUREG/CR-0041, TS, and procedures required by TS as criteria for determining compliance.

Inspection Planning

The inspectors reviewed respiratory protection program procedures and current performance indicators for unintended internal exposure incidents.

Use of Respiratory Protection Devices

There were no work activities where respiratory protection devices were used to limit the intake of radioactive materials during this inspection. The inspectors reviewed the adequacy of Entergy's use of respiratory protection devices in the plant to include applicable As Low As is Reasonably Achievable evaluations, respiratory protection device certification, respiratory equipment storage, air quality testing records, and individual qualification records.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment (71124.04)

a. Inspection Scope

The inspectors reviewed the monitoring, assessment, and reporting of occupational dose. The inspectors used the requirements in 10 CFR 20, RGs, TSs, and procedures required by TSs as criteria for determining compliance.

Internal Dosimetry - Special Bioassay (In Vitro)

There were no internal dose assessments obtained using urinalysis or fecal sample results for the inspector to review.

Internal Dose Assessment - Airborne Monitoring

The inspector reviewed Entergy's program for dose assessment based on airborne monitoring and calculations of derived air concentration internal dose. Entergy did not perform any internal dose assessments using airborne/derived air concentration monitoring during the period reviewed.

Shallow Dose Equivalent

Entergy did not document any dose assessments for shallow dose equivalent during this inspection period.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 - 2 samples)

Unplanned Scrams and Unplanned Scrams with Complications

a. Inspection Scope

The inspectors reviewed Entergy's submittals for the following Initiating Events cornerstone performance indicators for the period July 1, 2014, through June 30, 2015:

- Unplanned Scrams
- Unplanned Scrams with Complications

To determine the accuracy of the performance indicator data reported during this period, inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 10 CFR 50.73." The inspectors reviewed 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)

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.

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

(Closed) LER 05000333/2015-003-00: Roof Maintenance Results in Secondary Containment Vacuum Below Technical Specification Limit

a. Inspection Scope

On July 20, 2015, while operating at 100 percent power, work commenced to replace the roofing materials (insulation, felt paper, and a mix of asphalt and gravel) above the metal deck of the reactor building roof. Shortly after work started, operators noted that reactor building differential pressure (d/p, measured relative to outside pressure) was degrading. TS 3.6.4.1, "Secondary Containment," requires, in Mode 1, 2, or 3, that secondary containment be maintained at a vacuum of greater than or equal to 0.25 inch of vacuum water gauge, relative to the outside. Operators placed both trains of the standby gas treatment system in service and isolated normal reactor building ventilation, however, these actions were not successful in preventing d/p from decreasing to less than 0.25 inches of vacuum. This condition placed the plant in a four hour TS action statement to restore secondary containment or be in Mode 3 within 12 hours and Mode 4 in 36 hours. Work on the roof was stopped and efforts were commenced to reseal the roof. These actions were successful and d/p was restored to greater than 0.25 inches of vacuum after a period of 92 minutes. Secondary containment operability was not restored within four hours due to continuing roof restoration work and testing to verify operability, however, it was restored prior to a plant shutdown being required. This represents a self-revealing Green NCV, which is discussed below. This LER is closed.

b. Findings

Introduction. The inspectors identified a self-revealing violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because FitzPatrick staff failed to provide instructions appropriate to the reactor building roof replacement project. Specifically, inadequate instructions were provided to ensure that roofing material removal would be performed in a slow, deliberate manner, such that its effect on secondary containment could be assessed and operability maintained. As a result, this activity caused secondary containment to be inoperable for a period in excess of its TS allowed outage time.

Description. The reactor building roof replacement project originated due to the need to eliminate leaks, as well as to extend the functionality of the roof to the life of the plant. The EC package that was developed to support the roof replacement, EC 56686, "Reactor Building Roof Replacement - 2015," concluded that the metal roof decking formed the secondary containment pressure boundary and that the portion of the roof that was to be replaced was not safety-related. Nonetheless, it was recognized that removal of the top materials could uncover degraded portions of the metal roof decking, as well as fastener holes, and thereby challenge secondary containment. As a result, the EC placed a limit of 740 square feet of decking that could be exposed at any one time.

When removal of the existing roofing material commenced on July 20, 2015, workers cleared the first 740 square foot area of roof. At that time, it was not recognized that the decking plates were not overlapped and welded together, but rather had gaps between them which allowed for significantly more air in-leakage than had been anticipated. As a result, workers sealed the anticipated fastener holes and proceeded to clear the next 740 square foot area. At the same time, plant operators were observing secondary containment vacuum degrade. Operators placed both trains of the standby gas treatment system in service and isolated normal reactor building ventilation in an effort to maintain secondary containment. These actions arrested the degradation, but only resulted in holding secondary containment vacuum at 0.15 inches of water gauge. Work on the reactor building roof was stopped and secondary containment was declared inoperable.

As a corrective action, the new roofing materials were installed to reseal the affected area of the reactor building roof. This was successful in restoring secondary containment vacuum to greater than 0.25 inches water gauge after a period of 92 minutes. After completion of work and satisfactory PMT, operators declared secondary containment operable after a period of five hours and 26 minutes. The issue was entered into the CAP as CR-JAF-2015-03260.

Analysis. The inspectors determined that FitzPatrick staff's failure to provide appropriate instructions for the reactor building roof replacement project, such that this work resulted in a loss of secondary containment for longer than the TS allowed action time, was a performance deficiency that was within Entergy's ability to foresee and correct, and should have been prevented. The finding was more than minor because it is associated with the procedure quality attribute of the Barrier Integrity cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, RCS, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the WO did not provide adequate instruction to ensure that roofing material removal would be performed in slow, deliberate manner, coordinated between operations and maintenance personnel, and allowing adequate time after actions that could impact secondary containment such that their effect on secondary containment could be assessed and operability maintained. As a result, secondary containment was rendered inoperable and remained so for longer than the TS-specified allowed outage time.

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 standby gas treatment system.

The finding had a cross-cutting aspect in the area of Human Performance, Avoid Complacency, because FitzPatrick staff did not adequately plan for the possibility of latent issues and inherent risk associated with the reactor building roof replacement project, such that the commencement of work resulted in a loss of secondary containment [H.12].

Enforcement. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings…" Contrary to the above, on July 20, 2015, instructions provided to FitzPatrick maintenance personnel for removal of reactor building roofing material were not of a type appropriate to the circumstances, in that they were inadequate to ensure that roofing material removal would be performed in a slow, deliberate manner, in coordination with Operations department personnel, such that its effect on secondary containment could be assessed and operability maintained. As a result, secondary containment vacuum could not be maintained at or above the TS Surveillance Requirement 3.6.4.1.1 limit of 0.25 inches of vacuum water gauge, and secondary containment remained inoperable for a period in excess of the TS 3.6.4.1.A allowed outage time of four hours. Because this violation was of very low safety significance (Green) and FitzPatrick staff entered this issue into the CAP as CR-JAF-2015-03260, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000333/2015-02, Inadequate Instructions for Reactor Building Roof Replacement Result in Inadvertent Loss of Secondary Containment)**

4OA5 Other Activities

(Closed) Temporary Instruction 2515/190: Inspection of the Proposed Interim Actions Associated with Near-Term Task Force Recommendation 2.1 Flooding Hazard Evaluations

a. Inspection Scope

The inspectors performed activities to verify Entergy's conclusion that no interim actions were required. The activities performed were based on questions provided by the NRC staff that reviewed Entergy's near-term task force recommendation 2.1 flood hazard re-evaluation submittal, as well as the inspector's assessment of the hazard posed to safety-related equipment by the predicted flood levels. The results of the inspection were provided to the associated NRC staff in separate correspondence.

The specific activities performed included:

1) The inspectors reviewed the James A. FitzPatrick Flooding Hazard Re-Evaluation Report, Document No. 51-9227066-001, and Engineering Report JAF-RPT-14-00035, "Fukushima Project Walkdown of Plant Features (i.e. doors, hatches, etc.) That Are Potentially Subject to BDBEE [beyond design basis external events] Flood Water Infiltration," to identify the flooding mechanisms, pathways of concern, and consequences of the postulated worst case flooding events.
2) The inspectors evaluated the impact of the postulated flooding on the safety-related equipment that would be potentially at risk.
3) The inspectors inspected the exterior doors that would present the path for water infiltration to assess the material condition of the doors and the door seals.
4) The inspectors discussed the NRC staff reviewer's questions with Entergy engineering personnel to directly obtain their responses.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 15, 2015, the inspectors presented the inspection results to Mr. Brian Sullivan, Site Vice President, and other members of the FitzPatrick staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

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

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Open/Closed

05000333/2015003-01	NCV	Inadequate Corrective Actions Result in Control Rod Drift and Reactor Power Reduction (Section 1R15)
05000333/2015003-02	NCV	Inadequate Instructions for Reactor Building Roof Replacement Result in Inadvertent Loss of Secondary Containment (Section 4OA3)

Closed

05000333/2015-003-00	LER	Secondary Containment Vacuum Below Technical Specification Limit (Section 4OA3)
05000333/2515/190	TI	Inspection of Licensee's Proposed Interim Actions as a Result of the Near-Term Task Force Recommendation 2.1 Flooding Evaluation (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Documents
DBD-023, "Design Basis Document for the High Pressure Coolant Injection System," Revision 12

Procedures
ODSO-4, "Shift Turnover and Log Keeping," Revision 117
OP-14, "Core Spray System," Revision 35
OP-15, "High Pressure Coolant Injection," Revision 61
OP-21, "Emergency Service Water (ESW)," Revision 38
OP-22, "Diesel Generator Emergency Power," Revision 60
OP-37, "Containment Atmosphere Dilution System," Revision 81
OP-60, "Diesel Generator Room Ventilation," Revision 8

Drawings
FM-18A, "Flow Diagram Drywell Inerting C.A.D. and Purge System 27," Revision 57
FM-25A, "Flow Diagram High Pressure Coolant Injection System 23," Revision 74

Condition Reports
CR-JAF-2013-00779
CR-JAF-2013-01195
CR-JAF-2013-01212
CR-JAF-2013-01218
CR-JAF-2013-01955
CR-JAF-2013-02118
CR-JAF-2013-02697
CR-JAF-2013-02721
CR-JAF-2013-05940
CR-JAF-2013-06377
CR-JAF-2014-00872
CR-JAF-2014-01766
CR-JAF-2014-01821
CR-JAF-2014-01832
CR-JAF-2014-02040
CR-JAF-2014-06284
CR-JAF-2014-06824
CR-JAF-2014-06857
CR-JAF-2015-03546

Section 1R05: Fire Protection

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

Procedures
FPP-3.56, Portable Fire Extinguisher Inspection Procedure," completed 8/14/15
PFP-PWR13, "Main Control Room & Control Room HVAC Equipment Rooms/ Elev. 300' Fire Area/Zone VII/CR-1," Revision 6
PFP-PWR14, "Crescent Area East, Elev. 227' and 242' Fire Area/Zone XVII/RB-1E," Revision 3
PFP-PWR27, "Reactor Building, Elev. 344' Fire Area/Zone IX/RB-1A," Revision 4
PFP-PWR28, "Reactor Building/ Elev. 369' Fire Area/Zone IX/RB-1A," Revision 7
PFP-PWR33, "Pump Rooms (Screenwell)/Elev. 255' Fire Area/Zone XII/SP-1, XIII/SP-2, IB/FP-1, FP-3," Revision 2

Condition Reports
CR-JAF-2015-03180

Section 1R06: Flood Protection Measures

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

Procedures
AOP-43, “Plant Shutdown from Outside the Control Room,” Revision 38
ESP-50.003, “PSA Related Floor Drain Flow Test,” completed March 22, 2015

Condition Reports
CR-JAF-2015-01190
CR-JAF-2015-04034

Section 1R07: Heat Sink Performance

Documents
DBD-066, “Reactor Building HVAC Systems Unit Coolers,” Revision 11
DBD-067, “Design Basis Document for the Turbine Building Ventilation and Cooling Systems,” Revision 10
EPRI NP-7552, “Heat Exchanger Performance Monitoring Guidelines,” December 1991
JPN-90-015, “Response to NRC Generic Letter 89-13, ‘Service Water System Problems Affecting Safety-Related Equipment’” dated February 13, 1990; updated April 18, 1991 and March 16, 1993
LO-JAFLO-2014-00034, “Triennial Heat Sink NRC Prep Snapshot Assessment”
QA-8-2015-JAF-1, “Quality Assurance Audit Report, Engineering Programs”
SW and ESW System Heath Reports, third quarter 2014 and second quarter 2015

Procedures
AOP-56, “Intake Water Level Trouble,” Revision 11
AP-09.02, “Zebra Mussel Control Program,” Revision 8
AP-19.12, “Service Water Inspection Program,” Revision 7
EN-DC-184, “NRC Generic Letter 89-13 Service Water Program,” Revision 3
EN-DC-315, “Flow Accelerated Corrosion Program,” Revision 12
OP-42, “Service Water System,” Revision 48
OP-42A, “Service Water Chemical Cleaning System,” Revision 7
SEP-HX-JAF-001, “JAF Eddy Current Testing Of Heat Exchangers,” Revision 5
SEP-SW-JAF-001, “NRC Generic Letter 89-13 Service Water Program,” Revision 2
ST-8Q, “Testing of the Emergency Service Water System (IST),” Revision 45
ST-2YB, “RHR Heat Exchanger B Performance Test,” completed October 8, 2012
ST-2YB, “RHR Heat Exchanger B Performance Test,” completed August 26, 2014
ST-8Q, “Testing of the Emergency Service Water System (IST),” completed May 15, 2015

Drawings
FM-46A, Flow Diagram: Service Water System 46, Revision 91
FM-46B, Flow Diagram: Emergency Service Water System 46 and 15, Revision 57

Calculations
EC 6732, “Evaluate Decrease in Surface Area as a Result of Plugging Tubes on the Crescent, Cable Tunnel, and Electric Bay Area Coolers,” Revision 0
JAF-CALC-09-00007, “Cable Tunnel Cooler Heat Load,” Revision 0

Condition Reports

CR-JAF-2012-07011
CR-JAF-2014-00887
CR-JAF-2014-04388
CR-JAF-2014-04415
CR-JAF-2015-01119
CR-JAF-2015-01194
CR-JAF-2015-01345
CR-JAF-2015-01583
CR-JAF-2015-01835
CR-JAF-2015-03603
CR-JAF-2015-03604

Work Orders

00363260-01, “Thermal Performance Test 67UC-16A,” completed March 16, 2014
52466866-01, “PM-Cooler and Y-Strainer Cleaning for West Cable Tunnel Vent Supply Cooling Coil 67E-11,” completed May 12, 2015

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

Procedures

AOP-1, “Reactor Scram,” Revision 45
AOP-39, “Loss of Coolant,” Revision 18
AOP-41, “Feedwater Malfunction (Rising Feedwater Flow - High RPV Water Level),” Revision 11
EP-5, “Termination and Prevention of RPV Injection,” Revision 7
EP-8, “Alternate Injection Systems,” Revision 3
EOP-2, “RPV Control,” Revision 9
EOP-4, “Primary Containment Control,” Revision 8
OP-26, “Control Rod Drive Manual Control System,” Revision 25
OP-65, “Startup and Shutdown Procedure,” Revision 120
RAP-7.3.16, “Plant Power Changes,” Revision 51

Section 1R12: Maintenance Effectiveness

Documents

DBD-010, “Design Basis Document for the Residual Heat Removal System,” Revision 13
DBD-046, “Design Basis Document for the Normal Service Water, Emergency Service Water, RHR Service Water,” Revision 19
DBD-066, “Design Basis Document for the Reactor Building HVAC Systems,” Revision 7
JAF-RPT-MULTI-02294, “Maintenance Rule Basis Document for Service Water Systems Including System 10 (RHRSW), 46 (Normal SW), and 46-ESW (Emergency SW),” Revision 11
JAF-RPT-RBC-02295, “Maintenance Rule Basis Document System 066 Reactor Building Ventilation System,” Revision 4
JAF-RPT-RHR-02281, “Maintenance Rule Basis Document System 10 Residual Heat Removal System,” Revision 12

ST-19IA, “West Crescent Area Unit Cooler Air Flow Verification Test,” performed April 24, 2015 and August 20, 2015
ST-19IB, “East Crescent Area Unit Cooler Air Flow Verification Test,” performed February 14, 2015 and August 29, 2015
SW and ESW System Heath Reports for the third quarter 2014
System Health Reports, System 10 RHR/RHRSW for third quarter 2014 through second quarter 2015

Procedures

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

Condition Reports

CR-JAF-2013-00385	CR-JAF-2013-05842	CR-JAF-2014-04598
CR-JAF-2013-01104	CR-JAF-2013-05872	CR-JAF-2014-05383
CR-JAF-2013-01218	CR-JAF-2013-06086	CR-JAF-2014-06437
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Maintenance Rule Functional Failure Determinations for the Following CRs
CR-JAF-2014-00887
CR-JAF-2014-03121
CR-JAF-2014-03179

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 11

Section 1R15: Operability Determinations and Functionality Assessments

Documents
DBD-046, “Design Basis Document for the Normal Service Water, Emergency Service Water and RHR Service Water,” Revision 19
DBD-067, “Design Basis Document for the Turbine Building Ventilation and Cooling Systems,” Revision 7
ENN-MS-S-009-JAF, “JAF Safety System Function Sheets,” Revision 2
OSSO-15-005, Operations shift standing order form for HCU DCVs, Revision 0
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Procedures
AOP-27, “Control Rod Malfunction,” Revisions 11 and 12
ARP 09-5-2-3, “Annunciator Response Procedure - Rod Drift,” Revision 12
EN-OP-104, “Operability Determination Process,” Revision 9

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CR-JAF-2010-00287	CR-JAF-2015-03252	CR-JAF-2015-03793
CR-JAF-2015-03000	CR-JAF-2015-03260	

Section 1R18: Plant Modifications

Documents
EC 55301, “Level Switch 38LS-22A is Dysfunctional, Instead Use 38LS-22B to “Dump” Off Gas Drip Pot Until Issue With 38LS-22A is Remedied,” Revision 2
OSSO 15-002, “Off-gas Drip Pot Level Control,” Revision 3

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CR-JAF-2015-01000
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Section 1R19: Post-Maintenance Testing

Procedures
MST-071.11, “LPCI Battery Quarterly Surveillance Test,” Revision 22, performed July 23, 2015
MST-071.30, “LPCI Charger-Inverter Performance and LPCI Battery Service Surveillance Test,” Revision 19, performed July 15, 2015
OP-25, “Control Rod Drive Hydraulic System,” Revision 85
RAP 7.4.1, “Control Rod Scram Time Evaluation (IST)**,” Revision 22, for control rod 10-07, performed July 23, 2015
ST-16GA, “LPCI MOV Independent Power Supply Monthly Test,” Revision 2, performed July 17, 2015
ST-25BA, “CAD System A Quarterly Operability Test (IST),” Revision 4, performed September 15, 2015

Condition Reports
CR-JAF-2015-03595

Work Orders

WO 52495646	WO 52387521	WO 52360258
WO 00420116	WO 00422314	

Section 1R22: Surveillance Testing

Documents
ST-6HA, “Standby Liquid Control A Side Quarterly Operability Test (IST),” Revision 6, performed April 24, 2015
U055-0005, Vendor Manual for Union Type 1-7/8 x 3 TD-60 Pump

Procedures
ISP-100B-PCIS, “PCIS Instrument Functional Test/Calibration (ATTS),” Revision 13
OP-17, “Standby Liquid Control System,” Revision 51
ST-1L, “Main Turbine Control Valve Instrument Channel and Valve Operability Check,” Revision 35
ST-6KB, “B SLC System Class 2 Piping Leakage Test (ISI) and Operability Test (IST), Revision 2
ST-2AL, “RHR Loop A Quarterly Operability Test (IST),” Revision 35

Condition Reports
CR-JAF-2015-03445
CR-JAF-2015-03500

Work Orders
WO 00421941
WO 52475400
WO 52533127

Section 2RS1: Access Control to Radiologically Significant Areas

Documents
LO-JAFLO-2014-00048, Contamination Event Reduction, July 1, 2015
LO-JAFLO-2014-00030, Radiation Safety Inspection Snapshot, February 10, 2015
LO-JAFLO-2013-00093, Radiation Protection Documentation (CA-00001), January 14, 2015
LO-JAFLO-2013-00091, Radiological Postings (Outage) (CA-00001), January 30, 2015

Procedures
EN-RP-101, "Access Control for Radiologically Controlled Areas", Revision 11
EN-RP-105, "Radiological Work Permits", Revision 14
EN-RP-106, "Radiological Survey Documentation", Revision 6
EN-RP-106-01, "Radiological Survey Guidelines", Revision 2
EN-RP-108, "Radiation Protection Posting", Revision 15
EN-RP-110-04, "Radiation Protection Risk Assessment Process", Revision 5
EN-RP-121, "Radioactive Material Control", Revision 11
EN-RP-123, "Radiological Controls for Highly Radioactive Objects", Revision 1
RP-OPS-02.05, "Response to Notifications and Alarms", Revision 13
RP-OPS-03.05, "Refuel Floor and Drywell Radiological Controls", Revision 16

Surveys
JAF-1508-0035, reactor building elev. 272', on August 6, 2015, at 0156
JAF-1509-0018, reactor building elev. 272', on September 2, 2015, at 1225
JAF-1509-0072, reactor building elev. 272', on September 9, 2015, at 1439
JAF-1505-0162, reactor building elev. 300', on May 19, 2015, at 0815
JAF-1508-0043, reactor building elev. 300', on August 6, 2015, at 1451
JAF-1505-0048, reactor building elev. 326', on May 6, 2015, at 1224
JAF-1508-0014, reactor building elev. 326', on August 4, 2015, at 0841
JAF-1505-0152, reactor building elev. 344', on May 18, 2015, at 1356
JAF-1508-0038, reactor building elev. 344', on August 6, 2015, at 1045
JAF-1508-0157, reactor building elev. 369', on August 20, 2015, at 1827
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CR-JAF-2014-02620
CR-JAF-2014-02621
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CR-JAF-2015-00127
CR-JAF-2015-00161
CR-JAF-2015-00170
CR-JAF-2015-00208
CR-JAF-2015-00520
CR-JAF-2015-00662
CR-JAF-2015-01637

Section 2RS3: In-plant Airborne Radioactivity Control and Mitigation

Procedures
EN-RP-309, "Operation and Calibration of the Eberline AMS-3 and AMS-3A Continuous Air Monitor," Revision 1
EN-RP-122, "Alpha Monitoring", Revision 8
EN-RP-131, "Air Sampling", Revision 13
EN-RP-309, "Operation and Calibration of the Eberline AMS-3 and AMS-3A Continuous Air Monitor", Revision 1
EN-RP-310, "Operation and Initial Setup of the Eberline AMS-4 Continuous Air Monitor, Revision 4

EN-RP-402, “DOP Challenge Testing of HEPA Vacuums and Portable Ventilation Units”, Revision 4
EN-RP-404, “Operation and Maintenance of HEPA Vacuum Cleaners and HEPA Ventilation Units”, Revision 6

Section 2RS4: Occupational Dose Assessment

Procedures
EN-RP-110-04, “Radiation Protection Risk Assessment Process”, Revision 5
EN-RP-122, “Alpha Monitoring”, Revision 8
EN-RP-203, “Dose Assessment”, Revision 7
EN-RP-204, “Special Monitoring Requirements”, Revision 7
EN-RP-208, “Whole Body Counting / In-Vitro Bioassay”, Revision 6

Section 4OA2: Problem Identification and Resolution

Procedures
EN-LI-102, “Corrective Action Program,” Revision 24
EN-LI-118, “Cause Evaluation Process,” Revision 21

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CR-JAF-2015-02691	CR-JAF-2015-02911	CR-JAF-2015-03284
CR-JAF-2015-02692	CR-JAF-2015-02980	CR-JAF-2015-03598
CR-JAF-2015-02698	CR-JAF-2015-03000	CR-JAF-2015-03636
CR-JAF-2015-02748	CR-JAF-2015-03038	CR-JAF-2015-03713
CR-JAF-2015-02769	CR-JAF-2015-03043	CR-JAF-2015-03924
CR-JAF-2015-02888	CR-JAF-2015-03183	

LIST OF ACRONYMS

10 CFR	Title 10 of the *Code of Federal Regulations*
ASTM	American Society for Testing and Materials
CAD	containment atmosphere dilution
CAP	corrective action program
CR	condition report
CRD	control rod drive
DCV	directional control valve
EC	engineering change
EDG	emergency diesel generator
ESW	emergency service water
HCU	hydraulic control unit
IMC	Inspection Manual Chapter
IST	inservice test
kV	kilovolt
LER	licensee event report
LPCI	low pressure coolant injection
NCV	non-cited violation
NDE	non-destructive examination
NRC	Nuclear Regulatory Commission
PMT	post-maintenance test
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
SCC	stress corrosion cracking
SSC	structure, system, and component
SW	service water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
WO	work order