FitzPatrick
1Q/2016 Plant Inspection Findings

Initiating Events

Significance: G Sep 30, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
Inadequate Corrective Actions Result in Control Rod Drift and Reactor Power Reduction
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 As a result, on July 19, 2015, FitzPatrick control rod 10-07 drifted from the fully withdrawn to the fully inserted position in the reactor core leading to an immediate power reduction from 100 to 99 percent followed by a manual rapid power reduction to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.

Inspection Report# : 2015003 *(pdf)*

Significance: G Jun 26, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Failure to Adequately Assess the Impact of SRV Leakage on Operability
The inspectors identified a Green non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (10 CFR) 50 Appendix B Criterion III, Design Control, associated with Fitzpatrick’s failure to adequately assess and control the acceptance criteria specified in engineering analysis in EC-JAF-56258, “Operability Input for CR-JAF-2015-01271 SRV G Tailpipe Temperature Increase” which referenced JAF-RPT-03-0056 “Operational Leakage Action Levels for Target Rock Two-Stage Safety/Relief Valves.” Fitzpatrick concluded that a 2-stage Target Rock Safety Relief Valve (SRV) was operable with pilot valve leakage provided the leak rate was less than 1000 lbm/hr. This conclusion was not adequately supported by the available industry and plant data on setpoint drift and the references provided. As a result, Fitzpatrick did not declare 2-stage Target Rock Pilot valves inoperable when the leak rate exceeded 600 lbm/hr in 2007 and 2009. Fitzpatrick entered this issue into the corrective action system (CR-JAF-2015-02850) and is

reassessing the appropriate operability criteria.

This performance deficiency is more than minor because it adversely affects the equipment performance attribute of the initiating events cornerstone in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," to limit the likelihood of events that upset plant stability and challenge critical safety functions during power operations by ensuring RCS barrier integrity. This finding screens to Green using IMC 0609, "Significance Determination Process," Attachment 4, "Initial Characterization of Findings," and IMC 0609, Appendix A, Exhibit 1, "Initiating Events Screening Questions," Section A, "LOCA Initiators," as the finding could not result in leakage exceeding that of a small break LOCA nor could it have resulted in an interfacing system LOCA. The inspectors determined that this performance deficiency had a cross-cutting aspect in human performance, conservative bias, where individuals use decision making-practices that emphasize prudent choices over those that are simply allowable. [H.14] Section 1R17.
Inspection Report# : 2015007 *(pdf)*

Mitigating Systems

Significance: G Mar 31, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
[DRAFT] Unintended HPCI Pump Suction Transfer during Pressure Control Mode Operation
[DRAFT] 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).
Inspection Report# : 2016001 *(pdf)*

Significance: G Mar 31, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
[DRAFT] Uncontrolled RPV Level Increase after Initiation of RHR Shutdown Cooling
[DRAFT] 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 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).
Inspection Report# : 2016001 *(pdf)*

Significance: G Dec 31, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Unintended Elevated Plant Risk During EDG Maintenance
The inspectors identified a Green NCV of Title10 of the Code of Federal Regulations (10 CFR) 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” for failure to adequately manage the increase in risk during planned maintenance on the ‘A’ emergency diesel generator (EDG). Specifically, Entergy staff action to make the ‘C’ EDG unavailable while the ‘A’ EDG was already unavailable resulted in an unplanned increase in overall plant risk and deviation from the approved EDG outage risk management plan from a risk category of Green to the next higher risk category of Yellow. As immediate corrective action, the issue was entered into the corrective action program (CAP) as condition report (CR)-JAF-2015-05242.

The finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the ‘C’ EDG was not available when it should have been, in accordance with the approved risk management plan, which resulted in an unplanned escalation of risk from Green to Yellow. Additionally, this finding was similar to example 7.e in IMC 0612, Appendix E, “Examples of Minor Issues.” 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 safety function, did not represent 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, Work Management, because FitzPatrick did not execute the 'A' EDG maintenance outage work activities as planned, and after deviating from that plan, did not identify and manage the risk of barring the 'C' EDG while the 'A' EDG was unavailable [H.5].
Inspection Report# : 2015004 *(pdf)*

Barrier Integrity

Significance: G Mar 31, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
[DRAFT] Inadequate Post-Maintenance Testing of the Reactor Building Ventilation System Resulted in Short-Term Inoperability of Secondary Containment
[DRAFT] 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 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).
Inspection Report# : 2016001 *(pdf)*

Significance: N/A Mar 31, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
[DRAFT] Untimely 10 CFR 50.72 Notification of Inoperable Secondary Containment
[DRAFT] 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.
Inspection Report# : 2016001 *(pdf)*

Significance: N/A Dec 31, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Untimely 10 CFR 50.72 Notification of Inoperable Secondary Containment
The inspectors identified a Severity Level (SL) IV NCV of 10 CFR 50.72, “Immediate Notification Requirements for Operating Nuclear Power Reactors,” because inoperability of the secondary containment system was not reported to the NRC within eight hours of when the need to do so should reasonably have been recognized, as required by 10 CFR 50.72(b)(3)(v), “Event or Condition that Could Have Prevented Fulfillment of a Safety Function.” Specifically, positive pressure in the secondary containment due to a previously unidentified equipment malfunction that occurred during transition between the reactor building being isolated and normal reactor building ventilation being in service was not promptly recognized as a condition that caused the single train secondary containment system to be inoperable and therefore to be reportable under 10 CFR 50.72. This 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 an 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.

Inspection Report# : 2015004 *(pdf)*

Significance: G Sep 30, 2015
Identified By: NRC

Item Type: NCV Non-Cited Violation
Inadequate Instructions for Reactor Building Roof Relacement Result in Inadvertent Loss of Secondary Containment
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].

Inspection Report# : 2015003 *(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Public Radiation Safety

Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

Miscellaneous

Significance: SL-III Dec 31, 2011
Identified By: NRC
Item Type: VIO Violation
EA-10-090/EA-10-248/EA-11-106 RP Technician Willful Violations
During NRC investigations initiated on July 1, 2009, February 5, 2010, and April 8, 2010, violations of NRC requirements were identified. The following requirements were violated: 10 CFR 20.1703, 'Use of individual respiratory protection equipment'; 10 CFR 20.1501, Subpart F, 'Surveys and Monitoring'; 10 CFR 50.9, 'Completeness and accuracy of information'. Contrary to the listed requirements, the licensee employees willfully violated multiple procedures and incorrectly documented completion of surveys and respirator fit tests.

These violations are categorized collectively as a Severity Level III violation. The NRC offered and Entergy accepted to conduct Alternative Dispute Resolution (ADR) for the above listed violations. The NRC has issued Confirmatory Order (CO) EA-10-090, EA-10-248, EA-11-106 in response to the agreed upon ADR actions. As addressed in the CO, no civil penalty was assessed based on previous actions completed and actions agreed to be completed by the licensee.

Inspection Report# : 2011009 *(pdf)*

Last modified : July 11, 2016