FitzPatrick
3Q/2015 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 As a result, on July 19, 2015, FitzPatrick control rod 10-07 drifted from the fully withdrawn to the fully inserted position in the reactor core leading to an immediate power reduction from 100 to 99 percent followed by a manual rapid power reduction to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)*

Significance: G Mar 31, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
Inadequate Preventive Maintenance Strategy and Test Procedure for RWR MG Resulted in Multiple Plant Transients
A self-revealing, Green non-cited violation (NCV) of Technical Specification (TS) 5.4, "Procedures," was identified for failure to institute appropriate processes and procedures for periodic maintenance activities of the reactor water recirculation motor generators (RWR MGs). During startup from refueling outage 21, degraded material conditions led to tripping of an RWR MG, with the resultant loss of the associated RWR pump and down power transient, on three occasions. Specifically, one trip was due to carbon dust buildup within the 'A' RWR MG exciter, and two trips were due to a high resistance connection between the 'B' RWR MG generator field winding and a slip ring. Additionally, a fourth trip occurred during performance of an inadequately prepared RWR MG test procedure. As corrective action, the high resistance connection associated with the 'B' RWR MG was eliminated, voltage regulator tuning for the 'B' RWR MG was successfully completed, and temporary instrumentation was connected to both RWR MGs to monitor various key parameters pending the implementation of long term corrective actions. The RWR MG trips were entered into the corrective action program (CAP) through individual condition reports (CRs) that were subsequently consolidated under CR-JAF-2014-06258 for root cause evaluation (RCE).

The finding was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the transient initiation of single RWR loop operations challenges the reactor feedwater and vessel level control systems such that a more significant plant transient could result, and challenges plant operators in establishing allowable single RWR loop operating conditions. In accordance with Inspection Manual Chapter (IMC) 0609.04, "Initial Characterization of Findings," and Exhibit 1 of IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors determined that the finding was of very low safety significance (Green) because the performance deficiency was a transient initiator that did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. The finding had a cross-cutting aspect in the area of Human Performance, Resources, because FitzPatrick staff did not ensure that procedures for RWR preventive maintenance and voltage regulator tuning were adequate to support nuclear safety (H.1).
Inspection Report# : 2015001 *(pdf)*

Mitigating Systems

Significance: N/A Dec 31, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Untimely 10 CFR 50.72 Notification of a Secondary Containment System Functional Failure
The inspectors identified a Severity Level (SL) IV non-cited violation (NCV) of Title10 of the Code of Federal Regulations (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, while restoring the normal reactor building ventilation system to service following maintenance, reactor building-to-ambient differential pressure dropped below the Technical Specification (TS) required minimum value of 0.25 inches of vacuum water gauge and therefore caused the secondary containment system to be inoperable. However, FitzPatrick staff did not promptly recognize this as a condition reportable under 10 CFR 50.72. As corrective action, FitzPatrick staff reported the condition to the NRC in accordance with 10 CFR 50.72 (b)(3)(v) and entered it into the corrective action program (CAP) as condition report (CR)-JAF-2014-06498.

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 Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," traditional enforcement issues are not assigned cross-cutting aspects.

Inspection Report# : 2014005 *(pdf)*

Significance: G Dec 31, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
TS Actions for Inoperable ECCS Not Performed Within the TS Allowed Completion Time
The inspectors identified a Green non-cited violation (NCV) for two violations of Technical Specification (TS) 3.5.1, "ECCS [emergency core cooling systems] - Operating," associated with the non-functionality of east crescent area ventilation and cooling (CAVC) subsystem unit cooler 66UC-22H. Specifically, during the periods May 5 through May 21, 2010, and March 15 through March 25, 2011, the Technical Requirements Manual (TRM) requirements for east crescent unit cooler operability were not satisfied for longer than the allowed outage time (AOT), which caused the ECCS in the east crescent to become inoperable and remain so for longer than the TS AOT without completion of the required plant mode changes. As immediate corrective action, Entergy personnel reconditioned the fan motor contactor for the affected unit cooler to obtain satisfactory low voltage pickup response. The issue was entered into Entergy's corrective action program (CAP) as condition report (CR)-JAF-2012-00584 and CR-JAF-2012-02288.

The finding was more than minor because it is associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the unsatisfactory low voltage response of the 66UC-22H fan motor contactor, along with the unavailability of another

east CAVC unit cooler due to maintenance, could have degraded the capability of ECCS systems in the east crescent area during an accident concurrent with degraded voltage conditions. In light of FitzPatrick staff's determination that there was reasonable assurance that the remaining three operable unit coolers would have been capable of removing required post-accident heat loads, the inspectors determined that the finding was of very low safety significance (Green) in accordance with Inspection Manual Chapter (IMC) 0609.04, "Initial Characterization of Findings," and Exhibit 2 of IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," because the performance deficiency was not a design or qualification deficiency, did not involve an actual loss of safety function, did not represent the actual loss of a safety function of a single train for greater than its TS AOT, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Resolution, because FitzPatrick staff did not take effective corrective actions to address the low voltage pickup issue in a timely manner commensurate with its safety significance (P.3).

Inspection Report# : 2014005 *(pdf)*

Barrier Integrity

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)*

Significance: G Mar 31, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
Incomplete Fuel Support Piece Seating Not Identified During Post-Refueling Core Verification
A self-revealing, Green NCV of Title 10 of the Code of Federal Regulations (10 CFR) 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified because the existence of a partially seated fuel support piece at reactor cell location 38-39 was not identified when FitzPatrick staff performed the procedure for reactor core verification at the conclusion of refueling operations during the 2014 refueling outage (RO21). Specifically, the fact that the four fuel assemblies associated with cell 38-39 were elevated by an estimated 1.5 inches above the top of the rest of the fuel assemblies in the reactor core was not identified during visual verification of fuel assembly seating performed after the conclusion of core alterations in accordance with procedure EN-RE-210, "BWR [boiling water reactor] Reactor Core and MPC [multi-purpose canister] Cask Fuel Verification." As immediate corrective action, FitzPatrick staff engaged the fuel vendor, who provided an interim thermal limit penalty to be applied to the four affected fuel assemblies pending completion of a formal analysis. The issue was entered into FitzPatrick's corrective action program (CAP) as condition report (CR)-JAF-2015-00789.

The finding was more than minor because it was associated with the Configuration Control attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the fuel support piece not being completely fitted into the top of the control rod guide tube resulted in increased bypass flow around the cell 38-39 fuel assemblies, which reduced the margin to thermal limits for these assemblies during normal, transient, and accident conditions. Since the performance deficiency associated with the finding occurred during shutdown operations and also had potential safety significance during normal at-power operations, the inspectors screened the finding for significance using both Inspection Manual Chapter (IMC) 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," and IMC 0609 Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process." The inspectors determined that the finding was of very low safety significance (Green) because the displaced fuel bundles did not have any negative impact on safety during shutdown conditions, and through application of a thermal limit penalty, did not negatively impact the safe operation of the reactor at power. This finding had a cross-cutting aspect in the area of Human Performance, Procedure Adherence, because FitzPatrick staff did not follow the procedure requirement for reactor core verification to verify that the tops of the fuel channels and bail handles were all at approximately the same height (H.8).
Inspection Report# : 2015001 *(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 : December 15, 2015