Dresden 3
2Q/2015 Plant Inspection Findings

Initiating Events

Mitigating Systems

Significance: G May 29, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
Procedure Revisions Resulted in Isolation Condenser Unable to Meet Design Basis
The inspectors identified a finding of very-low safety significance, and an associated Non-Cited Violation (NCV) of Title 10, Code of Federal Regulations (CFR), Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to ensure that applicable regulatory requirements and the isolation condenser's (IC's) design bases were correctly translated into procedures. Specifically, the licensee added steps to the IC control procedures which directed operators to secure the IC in order to prevent the water level in the shell from going below 3.5 feet. The added steps would result in the IC being shutdown when required to operate per the IC's design bases. The licensee entered the issue into their Corrective Action Program (CAP) as Action Request 02506445, "NRC MOD/5059 Inspection: ISCO [Isolation Condenser] Operating Procedures," dated May 28, 2015.
The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of Procedure Quality, and affected the cornerstone's objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the inadequate procedures would drive the operators to stop the IC during a design bases event and prevent the IC from performing its design function of removing decay heat from the reactor. The finding has a cross-cutting aspect in the area of Human Performance; Teamwork, because the licensee did not communicate and coordinate activities within and across organizational boundaries to ensure nuclear safety is maintained. Specifically, the Operations Department failed to communicate and coordinate with the Engineering Department when developing the procedural changes. [H.4]

Inspection Report# : 2015007 *(pdf)*

Significance: G May 29, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
EDG Usable Fuel Calculations Failed to Consider Appropriate EDG Frequency Variations
The inspectors identified a finding of very-low safety significance, and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to account for increased fuel oil consumption during the development of the Emergency Diesel Generator (EDG) Calculation 10553 CALC 07, "Dresden Station Emergency Diesel Generators Endurance Calculations," Revision 2, which resulted in non-conservative Technical Specifications (TS). Specifically, the licensee failed to account for the increased fuel oil consumption at an EDG frequency of 61.2 Hertz (Hz), and ensure that the minimum fuel oil level in the EDG day tanks, as required per TS 3.8.1.4, was adequate to support the EDGs' mission time at 110 percent for one hour. The licensee entered the issue into their CAP as Action Request 02506869, "NRC MOD/5059 Inspection: Emergency Diesel Generator Fuel Consumption," dated

May 28, 2015.
The performance deficiency was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone's objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the licensee failed to account for the increased fuel oil consumption resulting from operation at a higher EDG frequency. Therefore, the licensee did not ensure that the minimum fuel oil level in the day tanks, as required per TS 3.8.1.4, was adequate to support the EDGs' mission time at 110 percent for one hour. This finding has a cross cutting aspect in the area of Problem Identification and Resolution; Identification, because the licensee did not did not thoroughly evaluate the EDG fuel oil consumption when considering EDG frequency variation. Specifically, the licensee failed to translate applicable design bases into specifications which resulted in non-conservative TS. [P.1]

Inspection Report# : 2015007 *(pdf)*

Significance: G Dec 31, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Unit 3A Low Pressure Coolant Injection (LPCI) Heat Exchanger Supports Returned to Service with Unacceptable Indications
A finding of very low safety significance and associated non-cited violation (NCV) of 10 CFR 50.55a(g)(4) was identified by the inspectors for the licensee's failure to maintain American Society of Mechanical Engineers (ASME) Code Class 2 components in accordance with ASME Code Section XI requirements. Specifically, the licensee failed to repair or replace the Unit 3A Low Pressure Coolant Injection (LPCI) heat exchanger support welds identified to have unacceptable linear flaws prior to return to service.
The inspectors determined that the licensee's acceptance of linear flaws in the Unit 3A LPCI heat exchanger supports that are determined to be unacceptable for continued service IAW with the ASME Code Section XI, Article IWC–3000 requirements was a performance deficiency (PD). The inspectors determined that the PD was more-than-minor, and a finding, because if the PD remained uncorrected it could lead to a more significant safety concern. Absent NRC identification, the LPCI support welds with unacceptable linear flaws would have remained in service without repair or replacement. This condition could potentially lead to the failure of the Unit 3A LPCI heat exchanger supports, which in turn, could lead to a potential failure of the Unit 3A LPCI heat exchanger. The inspectors reviewed the finding using Attachment 0609.04, "Initial Characterization of Findings," Table 3–Significance Determination Process (SDP) Appendix Router. The inspectors answered 'No' to the question in Section A of Table 3; and therefore, evaluated the finding using the SDP in accordance with IMC 0609, "The Significance Determination Process for At-Power Operations," Appendix A, Exhibit 2, "Mitigating Systems Screening Questions." The inspectors answered "No" to the questions in Exhibit 2 and determined that this finding did not result in a deficiency affecting the structures, systems, and components (LPCI heat exchanger) to maintain its operability or functionality. Therefore, the finding was determined to have very low safety significance. The inspectors determined that this finding has a cross-cutting aspect in the area of Human Performance, training, for the licensee's failure to provide training and ensure knowledge transfer to maintain a knowledgeable, technically competent workforce and instill nuclear safety values. Specifically, the licensee staff dispositioned unacceptable flaws in the LPCI heat exchanger supports for continued service using an engineering evaluation because the licensee staff lacked the specific ASME Code knowledge concerning disposition of the unacceptable indications. Therefore, the licensee failed to return the LPCI heat exchanger supports to within ASME Code acceptable flaw limits via repair or replacement prior to return to service. [H.9]
Inspection Report# : 2014005 *(pdf)*

Significance: W Dec 31, 2014
Identified By: NRC
Item Type: VIO Violation

Failure to Ensure Continued Operability of Unit 3 Electromatic Relief Valve 3-0203-3E Following Implementation of Extended Power Uprate Plant Conditions
An apparent violation (AV) of 10 CFR Part 50, Appendix B, Criterion III, Design Control, having a preliminary low to moderate safety significance, was self-revealed on November 6, 2014, following the discovery that one of the Unit 3 electromatic relief valves (ERVs) would not have performed its intended safety function. Increased vibrations experienced while operating at extended power uprate (EPU) power levels resulted in the degradation of multiple ERV actuator components which rendered the valve inoperable. The inspectors determined that the licensee fully implemented the Unit 3 EPU following a main generator rewind in November 2010,
but failed to verify that the ERV actuator design was suitable for operation at the continuously increased vibration levels experienced at EPU power levels. This finding does not represent an immediate safety concern in that the licensee has replaced all four Unit 3 ERV actuators with a hardened design successfully utilized at the Quad Cities Generating Station, which also experienced significant steam line vibrations post EPU.
The inspectors determined that the licensee's failure to ensure the continued operability of the Unit 3 ERVs following the establishment of EPU plant operating conditions was a performance deficiency warranting a significance evaluation. The inspectors determined that the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because it was associated with the Mitigating Systems Cornerstone attributes of design control and equipment performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. A Significance and Enforcement Review Panel (SERP), using IMC 0609, Appendix A, "Significance Determination Process For Findings At-Power," dated June 19, 2012, preliminarily determined the finding to be of low to moderate safety significance (White). The inspectors determined that this finding has a cross-cutting aspect of operating experience in the area of Problem Identification and Resolution, since it involves the failure to implement relevant internal and external operating experience in a timely manner. [P.5]
Inspection Report# : 2014005 *(pdf)*
Inspection Report# : 2015008 *(pdf)*

Barrier Integrity

Significance: G Dec 31, 2014
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
Failure to Maintain Configuration Control in the Unit 3 Containment Pressure Suppression System
A finding of very low safety significance and associated non-cited violation of Technical Specification (TS) 5.4.1, "Procedures," was self-revealed on November 19, 2014, for the licensee's failure to maintain configuration control in the Unit 3 containment pressure suppression system. Specifically, the licensee failed to maintain the instrument air stop valve to the actuator for Unit 3 torus vent 3–1601–60 open with the reactor in the Start-up and Run Mode following refueling outage D3R23.
The inspectors determined that the licensee's failure to maintain configuration control of the Unit 3 containment pressure suppression system was contrary to procedures for the emergency depressurization of containment as well maintaining TS required atmospheric conditions inside the primary containment with the reactor in Mode 1 and was a performance deficiency. The inspectors determined that the finding was more than minor because it was associated with the barrier integrity cornerstone attribute of configuration control in how containment design parameters are maintained while affecting the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The inspectors determined that the finding was of very low safety significance based on answering "No" to all of the Barrier Integrity screening questions in IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," Exhibit 3. The finding has a cross-cutting aspect of conservative bias in the area of Human Performance because the licensee did not implement

appropriate robust barriers to prevent bumping of the 3–1601–60SV in response to corrective action 511878–02. Specifically, the licensee previously evaluated 3–1601–60SV and non-conservatively determined that this particular valve did not require a seal to prevent inadvertent operation. [H.14]
Inspection Report# : 2014005 *(pdf)*

Emergency Preparedness

Significance: G Sep 30, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
Inadequate Evacuation Time Estimate Submittals
The NRC identified a NCV of 10 CFR 50.54(q)(2) associated with 10 CFR 50.47(b)(10) and 10 CFR Part 50, Appendix E, Section IV.4, for failing to maintain the effectiveness of the Dresden Nuclear Power Station Emergency Plan as a result of failing to provide the station evacuation time estimate (ETE) to the responsible offsite response organizations (OROs) by the required date.
Exelon submitted the Dresden Nuclear Power Station ETE to the NRC on December 12, 2012, prior to the required due date of December 22, 2012. The NRC completeness review found the ETEs to be incomplete due to Exelon fleet common and site-specific deficiencies, thereby preventing Exelon from providing the ETEs to responsible OROs and from updating site-specific protective action strategies as necessary. The NRC discussed its concerns regarding the completeness of the ETE, in a teleconference with Exelon on June 10, 2013, and on September 5, 2013, Exelon resubmitted the ETEs for its sites. The NRC again found the ETEs to be incomplete. The issue is a performance deficiency because it involves a failure to comply with a regulation that was under Exelon's control to identify and prevent. The finding is more than minor because it is associated with the emergency preparedness cornerstone attribute of procedure quality and because it adversely affected the cornerstone objective of ensuring that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The finding is of very low safety significance because it was a failure to comply with a non-risk significant portion of 10 CFR 50.47(b)(10). The licensee had entered this issue into their corrective action program (CAP) and re-submitted a new revision of the Dresden Nuclear Power Station ETE to the NRC on May 2, 2014, which was found to be complete by the NRC. The cause of the finding is related to the cross-cutting element of Human Performance, Documentation. [IMC 0310, H.7]
Inspection Report# : 2014004 *(pdf)*

Occupational Radiation Safety

Significance: G Mar 31, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
10 CFR 20.1701; Failure to Implement Effective Radiological Engineering Controls
A finding of very-low safety significance, and an associated NCV of 10 CFR 20.1701 was self-revealed during work activities associated with the failure to effectively implement planned radiological engineering controls during reactor head reassembly that resulted in personal contaminations and unintended radiological intakes to workers. On November 14, 2014, during the cleaning of the reactor head studs, several workers on the refuel floor were contaminated, and received unplanned and unintended intakes of radioactive material. Corrective actions included revising applicable procedures to improve the engineering and contamination controls during reactor head reassembly.

The inspectors determined that that the finding was more than minor in accordance with IMC 0612, in that the finding impacted the program and process attribute of the Occupational Radiation Safety Cornerstone, and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation. Specifically, the failure to implement effective radiological engineering and contamination controls during the cleaning of the contaminated reactor head studs resulted in personal contaminations and intakes to several workers. The inspectors concluded that the radiological hazards had the potential to result in higher exposures to the individuals had the circumstances been slightly altered. The finding was determined to be of very-low safety significance in accordance with IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," because it was not an as low as reasonably-achievable planning issue, there was neither overexposure nor substantial potential for an overexposure, and the licensee's ability to assess dose was not compromised. The inspectors concluded that the cause of the issue involved a cross-cutting component in the human performance in that the licensee's management did not ensures that effective radiological engineering controls was either managed or coordinated commensurate to the work activities. [H.5]
Inspection Report# : 2015001 *(pdf)*

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

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