La Salle 1
4Q/2014 Plant Inspection Findings

Initiating Events

Significance: G Dec 31, 2014
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
Inappropriate Instructions Led to Failure of MSIV
A finding of very low safety significance and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self revealed for the licensee’s failure to establish instructions for an activity affecting quality that were appropriate to the circumstances. Specifically, when the Unit 2 ‘C’ inboard main steam isolation valve (MSIV) failed shut due to a stem to disc separation on August 5, 2014, inspectors reviewed the circumstances leading to the failure and determined that engineering change (EC) 340595 was deficient. This EC was created in response to 2003 industry operating experience (OE) for the same failure mechanism (loss of pretension on the shaft-to-pilot-disc) at another facility, with the purpose of establishing inspection acceptance criteria to determine if the OE applied to LaSalle. The inspectors concluded that the acceptance criteria were inappropriate to the circumstances because they contained no guidance for identifying or dispositioning the actual failure mechanism reported in the OE. Even though two of the five MSIVs inspected at the time by the licensee displayed evidence of the OE reported failure mechanism (loss of pretension), the acceptance criteria as written were satisfied, so the MSIVs passed their inspections and future rebuild activities were deferred based primarily on these false negative inspection results. It was due to these deferrals that the August 5th failure occurred. All MSIV internals have since been rebuilt with a more robust design that is not susceptible to a loss of pretension failure, and a root cause evaluation was performed.

The performance deficiency was determined to be more than minor because it was associated with the Initiating Events Cornerstone attribute of procedure quality and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Since the valve failure caused a reactor scram and loss of condenser as the normal heat sink due to the Group I MSIV isolation, a detailed risk evaluation was required. The RIII Senior Reactor Analysts (SRAs) performed a detailed risk evaluation using the NRC’s Standardized Plant Analysis Risk model for LaSalle, version 8.24, and calculated a conditional core damage probability estimate of 8.4E-7, which represents a finding of very low safety significance, or Green. Because this performance deficiency occurred in 2003, no cross cutting aspect was assigned because it was not considered current performance. (Section 4OA3)

Inspection Report# : 2014005 *(pdf)*

Mitigating Systems

Significance: G Oct 03, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
Unit 1 Reactor Protection System Limit Switch Testing Failure

A self revealed non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was identified for the licensee’s failure to provide instructions appropriate to the circumstances for an activity affecting quality. Specifically, the installation instructions for the reactor protection system limit switch 1C71 N006B did not contain sufficient guidance to allow the component to be adjusted so that adequate clearance would exist during normal operation to ensure operability, and the component failed its first in service surveillance test. The licensee performed an apparent cause evaluation and planned to evaluate maintenance training needs, potential procedure enhancements, and potential enhancements to model work orders.

The finding was determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of procedure quality and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, reactor protection system limit switch 1C71 N006B is a safety related component that, in conjunction with other inputs, can initiate a reactor scram. The finding was determined to be of very low safety significance (Green) in accordance with the Significance Determination Process because the inspectors answered “No” to each of the screening questions. This finding has a cross-cutting aspect in the area of Human Performance, Training, because the licensee did not provide sufficient training and ensure knowledge transfer to maintain a knowledgeable, technically competent workforce and instill nuclear safety values. Specifically, in this instance, the level of individual training was at such a level that the procedure needed to be of greater detail to be appropriate to the circumstances (H.9).
Inspection Report# : 2014004 *(pdf)*

Significance: G Jun 30, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Adhere to Postings Led to Prohibited Items Being Left in ECCS Corner Rooms
Inspectors identified a finding of very low safety significance (Green) and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to follow written instructions, prominently displayed on signs and placards near the entrances of the emergency core cooling systems corner rooms, which prohibit the storage of items that can potentially clog the floor drains and adversely affect the systems’ ability to maintain a water level below the maximum safe operating level during a flooding event, as specified in LGA-002, “Secondary Containment Control.” Specifically, upon numerous occasions, inspectors identified materials that were placed either on the floor or on a surface that was below the maximum safe operating water level for the room, such that the materials would have posed a potential clogging hazard for the floor drains during a flooding event. Upon notification by the inspectors of the presence of prohibited materials, the licensee promptly removed the prohibited items from the areas. The most recent occurrence was entered into the licensee’s corrective action program as Action Request 01661788, and a number of interim compensatory measures, such as shiftly walkdowns by operations and radiation protection, were implemented to ensure that the areas remain clear of prohibited items until more permanent corrective actions are developed and put in place. At the time of this report, an apparent cause evaluation was in progress to evaluate the underlying cause of the performance deficiency, and to develop appropriate corrective actions.

The finding was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone’s objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to adhere to the written instructions of the postings in the emergency core cooling systems corner rooms led to the storage of prohibited items within those areas, which could have potentially challenged equipment availability during a flooding event. The finding was determined to be of very low safety significance (Green) in accordance with the Significance Determination Process because the performance deficiency did not result in the inoperability of any structures, systems, or components. This finding had a cross cutting aspect in the area of Human Performance, Training, because the organization did not ensure that the appropriate knowledge was

transferred to the staff. Specifically, the staff was not effectively trained on the features of the emergency core cooling systems corner rooms, such that the importance of keeping prohibited items out of the area for flood mitigation purposes was not sufficiently understood.
Inspection Report# : 2014003 *(pdf)*

Significance: G Mar 31, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow the Drywell Closeout Procedure when Declaring Primary Containment Ready for Power Operations
The inspectors identified a finding of very low safety significance (Green) and associated non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," for the licensee's failure to conduct primary containment (drywell) close-out activities in accordance with site procedures. Specifically, during the NRC's drywell closeout inspection, the inspectors identified that the licensee had not secured the reactor shield steel doors around three of the six feedwater system penetrations into the reactor pressure vessel (RPV). As a result, the steel doors design function to resist transient pressure loadings within the shield annulus would have been impacted. Licensee corrective actions included securing the reactor shield doors prior to power operation and placing the issue into the corrective action program (CAP).

The finding was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone, and affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using NRC IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," dated February 28, 2005, the finding was determined to be of very low safety significance (Green) because the inspectors qualitatively determined that the finding involved adequate mitigation capability and was not an event that could be characterized as a loss of control. This finding had a cross-cutting aspect in the area of Human Performance, Work Management, because the licensee did not implement a process for controlling and executing work activities such that nuclear safety is the overriding priority. Specifically, the licensee's process did not ensure that the reactor shield was intact prior to the completion of the drywell closeout procedure. [H.5]
Inspection Report# : 2014002 *(pdf)*

Barrier Integrity

Significance: G Mar 31, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
Untimely Test, Evaluation, and Report on Reactor Vessel Surveillance Material
A finding of very low safety significance (Green) and an associated Severity Level IV non-cited violation (NCV) of 10 CFR 50.60 "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation" was identified by the inspectors for the licensee's failure to conduct a timely test, evaluation and report on the material contained in the 120° reactor vessel (RV) surveillance capsule within one year of capsule withdrawal to validate the RV pressure-temperature (P/T) limits. Specifically, in February 2010, the licensee withdrew the RV capsule at the 120° azimuth and did not report test results until November of 2011, and did not report the impact of these results on P/T limits until January of 2013. The licensee entered this issue into their Corrective Action Program as AR 01598777 and submitted a request for a Technical Specification amendment with revised P/T curves to the NRC for approval that reflected the test results from the 120° azimuth RV surveillance

capsule.

This issue was more than minor in accordance with IMC 0612, Appendix B "Issue Screening," dated September 7, 2012, because it adversely affected the Barrier Integrity Cornerstone attribute of Design Control. The finding was determined to be more-than-minor because If left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, the failure to conduct a timely test, evaluation and report on the material in the 120° RV surveillance capsule could have resulted in plant operation in an unacceptable region that would increase the possibility of vessel failure by brittle fracture (similar to pressurized thermal shock event for a pressurized water reactor). The inspectors performed a Phase I SDP screening using IMC 0609, Attachment 0609 Appendix A, Exhibit 3-Barrier Integrity Screening Questions, dated June 19, 2012, and selected the box under the Reactor Coolant System Boundary (e.g., pressurized thermal shock issues) which required a detailed risk-evaluation. A Region III Senior Reactor Analyst performed a detailed risk-evaluation of this finding. A potential increase in the probability for RV failure would exist if the plant was inadvertently operated in the unacceptable P/T region. Because the plant was not operated outside analyzed P/T limits for the RV, the driving force for crack propagation (e.g., K1) remained unchanged. However, to bound the delta risk evaluation, it was assumed that the initiating event frequency for a RV failure increased by 10 percent. From the LaSalle Standardized Plant Analysis Risk Model Version 8.21, the initiating event frequency for reactor vessel failure from any cause is 1E-7/yr. Core damage is expected to occur if reactor vessel failure occurs. The exposure time for the finding was the maximum of one year. Thus, a bounding risk assessment yields a delta risk of 1E-8/yr. Therefore, based on the detailed risk-evaluation, this finding is of very low safety significance (Green). This violation was similar to an example of a severity level (SL) IV violation identified in Section 6.9.d.9 of the NRC Enforcement Policy, which identifies an example related to failure to make a required report under 10 CFR 50.72 or 10 CFR 50.73. Because the report timeliness requirements were not met for reporting the 120° RV surveillance capsule results, the NRC did not have the opportunity to review and approve the revised P/T curves prior to the plant exceeding the 21 Effective Full Power Year limit for application of the existing NRC approved P/T curves. Therefore, the failure to provide a timely report on this surveillance capsule had the potential to have impeded the regulatory process. Because of the very low risk significance, this issue was considered similar to an example of a SL IV violation. This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, because the licensee did not take appropriate corrective actions to address safety issues in a timely manner, commensurate with their safety significance. Specifically, the licensee failed to develop a procedure or process for monitoring the timeliness of surveillance capsule testing, analysis and reporting. [P.3]
Inspection Report# : 2014002 *(pdf)*

Significance: G Mar 31, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Ensure that Activities Affecting Quality were Performed in Accordance with Current Procedure Revisions
A finding of very low safety significance and associated non-cited violation (NCV) of Technical Specification 5.4.1.a, "Procedures," was identified by inspectors for the licensee's failure to ensure that activities affecting quality were conducted in accordance with current, approved, revisions of procedures as required by licensee procedure HU-AA-104-101, "Procedure Use And Adherence," Revision 4. Specifically, on three separate occasions, inspectors identified that work groups were using superseded procedure revisions in the field and that no supervisory review had been performed to allow the use of those superseded procedures. The licensee entered this issue into its corrective action program (CAP) as action reports (ARs) 01623438 and 01625505, and performed an apparent cause evaluation. Corrective actions to prevent future occurrences of this performance deficiency have been developed and the activities in question were reviewed to ensure that the use of the incorrect procedures had no detrimental effect on the effected systems or components.

The finding was determined to be more than minor because, if left uncorrected, it would become a more significant safety concern. Specifically, failing to ensure that the most up to date procedures are used for a given activity affecting

quality, or failing to approve a superseded procedure for execution, could lead to a degraded or non-conforming condition if a crucial procedure step had been significantly revised. Using NRC IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process,” dated February 28, 2005, the finding was determined to be of very low safety significance (Green) because the inspectors qualitatively determined that the finding involved adequate mitigation capability and was not an event that could be characterized as a loss of control. This finding had a cross-cutting aspect in the area of Human Performance, Resources, because the licensee supervisors involved did not ensure that the appropriate procedures were available to the workers and adequate to support nuclear safety. Specifically, the cognizant supervisors did not obtain copies of the controlling documents from a controlled document set immediately prior to the performance of the tasks. [H.1]
Inspection Report# : 2014002 *(pdf)*

Emergency Preparedness

Significance: G Oct 03, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
Inadequate Evacuation Time Estimate Submittals
The NRC identified an non-citied violation 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 LaSalle County 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 LaSalle County
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 involved 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 (Green) 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 its corrective action program (CAP) and re submitted a new revision of the LaSalle County Station ETE to the NRC on April 30, 2014, which was found to be complete by the NRC. The cause of the finding is related to cross cutting element of Human Performance, Documentation (H.7).

Inspection Report# : 2014004 *(pdf)*

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

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