UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE RD. SUITE 210
LISLE, ILLINOIS 60532-4352

February 14, 2019

Mr. Bryan C. Hanson
Senior VP, Exelon Generation Company, LLC
President and CNO, Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2—NRC INTEGRATED INSPECTION REPORT 05000373/2018004 AND 05000374/2018004

Dear Mr. Hanson:

On December 31, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your LaSalle County Station, Units 1 and 2. On January 9, 2019, the NRC inspectors discussed the results of this inspection with Mr. W. Trafton, and other members of your staff. The results of this inspection are documented in the enclosed report.

Based on the results of this inspection, the NRC has identified two issues that were evaluated under the risk significance determination process as having very-low safety significance (Green). The NRC has also determined that two violations are associated with these issues. Because the licensee initiated condition reports to address these issues, these violations are are being treated as Non-Cited Violations (NCVs), consistent with Section 2.3.2 of the Enforcement Policy. These NCVs are described in the subject inspection report. Further, inspectors documented a licensee-identified violation which was determined to be of very low safety significance. The NRC is treating this violation as a non-cited violation consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement; and the NRC Resident Inspector at the LaSalle County Station.

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001; with copies to the Regional Administrator, Region III; and the NRC resident inspector at the LaSalle County Station.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Kenneth Riemer, Chief
Branch 1
Division of Reactor Projects

Docket Nos. 50–373; 50–374; 72–070
License Nos. NPF–11; NPF–18

Enclosure:
IR 05000373/2018004; 05000374/2018004

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Letter to Bryan Hanson from Kenneth Riemer dated February 14, 2019

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2—NRC INTEGRATED INSPECTION REPORT 05000373/2018004 AND 05000374/2018004

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Numbers:	50–373; 50–374; 72–070
License Numbers:	NPF–11; NPF–18
Report Numbers:	05000373/2018004; 05000374/2018004
Enterprise Identifier:	I–2018–004–029
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	Marseilles, IL
Dates:	October 1 through December 31, 2018
Inspectors:	W. Schuap, Senior Resident Inspector, LaSalle J. Havertape, Resident Inspector, LaSalle C. Phillips, RIII Project Engineer J. Cassidy, Sr. Health Physicist D. Sargis, Resident Inspector, Clinton M. Holmberg, Sr. Reactor Inspector V. Meghani, Reactor Inspector G. Hansen, Sr. Emergency Preparedness Inspector C. Zoia, Sr. Operations Engineer R. Zuffa, Illinois Emergency Management Agency
Approved by:	K. Riemer, Chief Branch 1 Division of Reactor Projects

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring licensee's performance by conducting an integrated quarterly inspection at LaSalle County Station, Units 1 and 2, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information. Findings and violations being considered in the NRC's assessment are summarized in the table below. A Licensee identified Non-Cited violation is documented in report section: 71153

List of Findings and Violations

Failure to Implement Scaffolding Program			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000373/2018004–01 Opened/Closed	[H.13] — Consistent Process	71111.05
The inspectors identified a finding of very low safety significance (Green) and an associated non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (CFR) Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to follow station procedure MA–AA–716–025, "Scaffold Installation, Modification, and Removal Request Process," Revision 15. Specifically, the licensee erected two scaffolds that were in close proximity to safety-related equipment without engineering approval for less than minimum clearances. Additionally, these scaffolds were not being tracked administratively, and as a result were installed in the plant for greater than 90 days with neither a 10 CFR 50.59 review nor engineering approval to make the installation a permanent scaffold.			

Technical Specification Surveillance Procedure Aceptance Criteria Did Not Consider Instrument Uncertainty			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000373/2018004–02 Opened/Closed	None	71152
The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to include an appropriate acceptance criterion for drywell oxygen concentration in surveillance procedures. Specifically, the acceptance criterion did not account for instrument uncertainty.			

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
LER	05000373/2018–003–00 and 05000373/2018–003–01	Two Main Steam Safety Relief Valves Failed Inservice Lift Inspection Pressure Test	71153	Closed

TABLE OF CONTENTS

PLANT STATUS

Unit 1 began the inspection period at rated thermal power. On October 17, 2018, the unit was down powered to approximately 92 percent power due to a feedwater heater (15A) emergency drain valve failure and power excursion. The affected drain valve was repaired and the unit was returned to full power on October 18, 2018. On December 14, 2018, the unit was down powered to approximately 77 percent for a rod pattern adjustment and to complete turbine valve testing. The unit was ruturn to full power the following day. The unit remained at or near rated thermal power for the remainder of the inspection period.

Unit 2 began the inspection period at rated thermal power. On November 2, 2018, the unit was down powered to approximately 80 percent power to perform a rod sequence exchange and control rod testing. The unit was returned to full power the following day. On November 20, 2018, the unit was down powered to approximately 85 percent power to perform a rod sequence exchange, control rod testing and turbine valve testing. On December 20, 2018, the unit was down powered to approximately 85 percent power to perform a rod sequence exchange, control rod testing and turbine valve testing. The unit was returned to full power the same day. The unit remained at or near rated thermal power for the remainder of the inspection period.

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors performed plant status activities described in IMC 2515 Appendix D, "Plant Status" and conducted routine reviews using IP 71152, "Problem Identification and Resolution." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

REACTOR SAFETY

71111.01—Adverse Weather Protection

Seasonal Extreme Weather (1 Sample)

The inspectors evaluated readiness for seasonal extreme weather conditions prior to the onset of seasonal cold temperatures on November 14, 2018.

External Flooding (1 Sample)

The inspectors evaluated readiness to cope with external flooding on October 23, 2018.

71111.04—Equipment Alignment

Partial Walkdown (2 Samples)

The inspectors evaluated system configurations during partial walkdowns of the following systems/trains:

(1) Unit 2 standby gas treatment system during maintenance on the Unit 1 standby gas treatment system on October 1, 2018; and
(2) Unit 2 'B' low pressure coolant injection mode of residual heat removal (RHR) on October 18, 2018.

Complete Walkdown (1 Sample)

The inspectors evaluated system configurations during a complete walkdown of the Unit 1, Division II electrical distribution system during Division I battery replacement on November 14, 2018.

71111.05AQ—Fire Protection Annual/Quarterly

Quarterly Inspection (4 Samples)

The inspectors evaluated fire protection program implementation in the following selected areas:

(1) Fire zone 3H3, Unit 2 B RHR, 694' elevation;
(2) Fire zone 3I3, Unit 2 B RHR pump, 673' elevation;
(3) Fire zone 3I4, Unit 2 low pressure core spray system, 673'4" elevation; and
(4) Fire zone 3H4, Reactor Core Isolation Cooling (RCIC) pump cubicle, 694' elevation.

Annual Inspection (1 Sample)

The inspectors evaluated fire brigade performance on Unit 1, condensate pump aisle, on November 1, 2018.

71111.06—Flood Protection Measures

Internal Flooding (2 Samples)

The inspectors evaluated internal flooding mitigation protections in the Unit 2, Division II corner room and Unit 2 raceway, both at 673" elevation, on October 23, 2018.

71111.07—Heat Sink Performance

Heat Sink (1 Sample)

The inspectors evaluated the Unit 1, Division III emergency diesel generator cooling performance on November 20, 2018.

71111.11—Licensed Operator Requalification Program and Licensed Operator Performance

Operator Requalification (1 Sample)

The inspectors observed and evaluated the out-of-the-box (OBE) ESG–80 on October 10, 2018.

Operator Performance (1 Sample)

The inspectors observed and evaluated operators in the control room during the Unit 2 down power to support control valve testing, scram time testing and post maintenance testing on motor-driven reactor feed pump on December 8, 2018.

Operator Exams (1 Sample)

The inspectors reviewed and evaluated requalification examination results on December 18, 2018.

71111.12—Maintenance Effectiveness

Routine Maintenance Effectiveness (3 Samples)

The inspectors evaluated the effectiveness of routine maintenance activities associated with the following equipment and/or safety significant functions:

(1) Reactor building flood seals on October 23, 2018;
(2) Licensee a(3) evaluation; and
(3) Reactor building ventilation check dampers.

71111.13—Maintenance Risk Assessments and Emergent Work Control (3 Samples)

The inspectors evaluated the risk assessments for the following planned and emergent work activities:

(1) Mobile crane exterior to Unit 1 and Unit 2 reactor buildings;
(2) Unit 1 and Unit 2 online risk yellow due to 0 emergency diesel generator maintenance;
(3) Unit 1, Division I and II protected equipment during yellow online risk for Division III maintenance; and
(4) Unit 1 and Unit 2 yellow online risk during blizzard.

71111.15—Operability Determinations and Functionality Assessments (5 Samples)

The inspectors evaluated the following operability determinations and functionality assessments:

(1) Unit 1 ‘A’ standby liquid control pump gearbox vibration in Alert range;
(2) Unit 1 drywell high temperature annunciator during ‘B’ containment chiller maintenance window;
(3) Unit 1, Division I 125 volts direct current (VDC) battery online replacement;

(4) Secondary containment operability during forced helium dehydration modification installation; and
(5) Unit 2 RCIC turbine inboard bearing oil level low.

71111.18—Plant Modifications (1 Sample)

The inspectors evaluated the following temporary or permanent modification:

(1) Category II over I qualification for abandoned high pressure core spray suction line after a temporary modification on November 20, 2018.

71111.19—Post Maintenance Testing (4 Samples)

The inspectors evaluated the following post maintenance tests:

(1) Unit 2 diesel generator on November 15, 2018;
(2) Unit 1 standby gas treatment system on October 18, 2018;
(3) Unit 2 B RHR heat exchanger outlet valve breaker replacement on October 16, 2018; and
(4) Unit 1, A control room heating, ventilation and cooling, D radiation monitor post-maintenance testing on October 29, 2018.

71111.22—Surveillance Testing

The inspectors evaluated the following surveillance tests:

Routine (1 Sample)

(1) Diesel fire pump 0A (0FP01KA) operational check on December 18, 2018.

In-Service (1 Sample)

(1) Unit 2 B RHR system operability and inservice test on December 19, 2018.

71114.04—Emergency Action Level and Emergency Plan Changes (1 Sample)

The inspector completed the evaluation of submitted Emergency Action Level and Emergency Plan changes on November 30, 2018. This evaluation does not constitute NRC approval.

RADIATION SAFETY

71124.02—Occupational As Low As Reasonably Achievable Planning and Controls

Radiological Work Planning (1 Sample)

The inspectors evaluated the licensee’s radiological work planning by reviewing the following activities:

(1) LA–02–17–00502; L2R16 drywell RP Department Activities;
(2) LA–02–17–00506; L2R16 drywell Scaffold;

(3) LA–02–17–00513; L2R16 drywell Control Rod Drive (CRD) Exchange;
(4) LA–02–17–00547; L2R16 drywell RR Motor Replacement; and
(5) LA–01–18–00510; L1R17 drywell Steam Safety Relief Valve Activities.

Verification of Dose Estimates and Exposure Tracking Systems (1 Sample)

The inspectors evaluated dose estimates and exposure tracking.

Implementation of As Low As Reasonably Achievable and Radiological Work Controls (Partial Sample)

The inspectors reviewed as low as reasonable achievable practices and radiological work controls by reviewing the following activities:

(1) LA–02–17–00502; L2R16 drywell RP Department Activities;
(2) LA–02–17–00506; L2R16 drywell Scaffold;
(3) LA–02–17–00513; L2R16 drywell Control Rod Drive (CRD) Exchange;
(4) LA–02–17–00547; L2R16 drywell RR Motor Replacement; and
(5) LA–01–18–00510; L1R17 drywell Steam Safety Relief Valve Activities.

OTHER ACTIVITIES – BASELINE

71151—Performance Indicator Verification (7 Samples)

The inspectors verified licensee performance indicators submittals listed below:

(1) MS05: Safety System Functional Failures—2 Samples; October 1, 2017 – September 31, 2018;
(2) MS08: Heat Removal Systems—2 Samples; October 1, 2017 – September 31, 2018;
(3) MS10: Cooling Water Support Systems—2 Samples; October 1, 2017 – September 31, 2018; and
(4) OR01: Occupational Exposure Control Effectiveness—1 Sample; October 2017 – September 2018.

71152—Problem Identification and Resolution

Semiannual Trend Review (1 Sample)

The inspectors reviewed the licensee's corrective action program for trends that might be indicative of a more significant safety issue.

Annual Follow-Up of Selected Issues (1 Samples)

The inspectors reviewed the licensee's implementation of its corrective action program related to the following issues:

(1) Drywell continuous oxygen monitor.

71153—Follow-Up of Events and Notices of Enforcement Discretion

Licensee Event Reports (1 Sample)

The inspectors evaluated the following licensee event reports which can be accessed at https://lersearch.inl.gov/LERSearchCriteria.aspx:

(1) Licensee Event Reports 05000373/2018–003–00 and 05000373/2018–003–01, Two Main Safety Relief Valves Failed Inservice Lift Inspection Pressure Test, on July 25, 2018.

INSPECTION RESULTS

71111.05AQ—Fire Protection Annual/Quarterly

<table>
<tr><td colspan="4">Failure to Adhere to Scaffolding Program</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-Cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Mitigating Systems</td><td>Green
NCV 05000373/2018004–01
Open/Closed</td><td>[H.13] –
Consistent Process</td><td>71111.05</td></tr>
<tr><td colspan="4">Introduction:

The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure follow station procedure MA–AA–716–025, “Scaffold Installation, Modification, and Removal Request Process,” Revision 15. Specifically, the licensee erected two scaffolds that were in close proximity to safety-related equipment without engineering approval for less than minimum clearances. Additionally, these scaffolds were not being tracked administratively, and as a result were installed in the plant for greater than 90 days with neither a 10 CFR 50.59 review nor engineering approval to make the installation a permanent scaffold.</td></tr>
<tr><td colspan="4">Description:

On October 17, 2018, during a fire protection walk down, NRC inspectors identified a scaffold in contact with safety-related piping on the 687 feet elevation in the Unit 2, Division II core standby cooling system (CSCS) pump room. The inspectors noted that there was no engineering approval of the scaffold installation as required by Step 4.1.5 of MA–AA–716–025 for scaffolds in close proximity of safety-related equipment. The inspectors discussed the issue with the licensee, as well as questioned the acceptability of a scaffold contacting safety-related equipment. The inspectors concerns were documented in the corrective action program by the licensee as AR 04185035.

On October 18, 2018, NRC inspectors identified a scaffold in contact with safety-related piping on the 694 feet elevation in the Unit 2, Division II RHR corner room. Again, the inspectors noted there was no engineering approval of the scaffold installation as required by Step 4.1.5 of MA–AA–716–025. Additionally, the inspectors reviewed the “Non-Permanent Scaffold Request Form” posted on the scaffold and noted that Section C, “Pre-Erection Operations Review,” of the form documented that Operations did their jobsite review via teleconference.</td></tr>
</table>

Procedure MA–AA–716–025, Steps 4.1.4.2 and 4.1.2.3, required operations staff to identify bump sensitive equipment in the area of the scaffold and to determine if an operations inspection of the installation was required (i.e. whether the scaffold was installed near safety-related equipment). The inspectors questioned the ability of operations department personnel to make that type of determination without physically walking down the area of the scaffold installation. The inspectors discussed their concerns with the licensee, who documented them in the corrective action program as AR 04185256.

In response to the inspectors' questions, the licensee evaluated the scaffolds for the potential impact on the equipment during a postulated seismic event. In each case it was determined that there would be no adverse impact. To determine whether engineering approval was required by step 4.1.5 of MA–AA–716–025, the licensee referred to Table 2 of engineering standard NES–MS–04.1, "Seismic Prequalified Scaffolds," Revision 7. Note 3 of this standard states in part, that scaffolds to be installed within 4 inches of safety-related equipment in the Auxiliary Building below 815 feet elevation and within 3.5 inches of safety-related equipment in the Reactor Building below 843 feet elevation be approved by engineering. Therefore, engineering approval was required for the scaffolds installed in the Division II CSCS pump room and RHR corner room.

The inspectors also discussed the administrative tracking requirements in Step 3.6 of MA–AA–716–025 with the licensee. Step 3.6 of MA–AA–716–025 required the following:

> "Scaffold Coordinator/Designee– Is responsible for the coordination of erection and removal of all scaffolds on site. Maintaining a log or electronic equivalent of the status of all scaffolds, and reviewing the log to ensure that any scaffolds approaching their 90 day limit are removed or converted to a permanent scaffold or requesting that an individual 10 CFR 50.59 review be performed for the individual scaffold required to be left in place beyond 90 days."

The inspectors noted that since the previously discussed scaffolds were installed in March and April of 2017, greater than 90 days, that Step 3.6 would require that the licensee either perform a 10 CFR 50.59 review of the temporary installed scaffolding or get engineering approval to convert the temporary installed scaffolding to permanent scaffolding. The licensee acknowledged that they did not perform either of these actions, contrary to Step 3.6 of MA–AA–716–025. Further, the licensee determined that the previously discussed scaffolds were not being tracked in the scaffolding log, contrary to Step 3.6 of MA–AA–716–025.

Corrective Actions: The licensee removed the scaffolds in the Division II CSCS pump room and RHR corner room to comply with MA–AA–716–025. Additionally, the licensee completed a Corrective Action Program Evaluation. During this review the licensee discovered two additional scaffolds in close proximity to safety-related equipment without the required engineering approval. The licensee documented the issues in the corrective action program as AR 04186864 and 04186868. The licensee evaluated the scaffolds for the potential impact on the equipment during a postulated seismic event and determined there would be no impact on equipment operability.

Corrective Action References: ARs 04185035, 04185256, 04186864, and 04186868

Performance Assessment:

Performance Deficiency: The inspectors identified that multiple examples of the failure to follow procedure, MA–AA–716–025, as related to control of temporary scaffolding was contrary to 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," and was a performance deficiency. Specifically, the licensee built scaffolds in close proximity to safety-related equipment without engineering approval; and did not follow administrative requirements to ensure temporary scaffolds installed in the plant greater than 90 days were either reviewed under 10 CFR 50.59 or made a permanent scaffold.

Screening: The inspectors determined the performance deficiency was more than minor in accordance with IMC 0612, Appendix E, "Examples of Minor Issues." Specifically, the inspectors concluded that this issue was similar to the more than minor criteria established in Example 4.a, "Insignificant Procedural Issues," since the licensee routinely failed to perform the required engineering evaluations on seismically qualified scaffolds. Therefore, this performance deficiency also impacted the Mitigating Systems Cornerstone objective of protection against external events (i.e. seismic events).

Significance: The inspectors assessed the significance of the finding using SDP Appendix A, "The Significance Determination Process for Findings At-Power." The finding screened as very low safety significance because it did not result in the loss of operability or functionality of a Mitigating System.

Cross-Cutting Aspect: The inspectors determined this finding affected the cross-cutting area of human performance in the aspect of consistent process, where Individuals use a consistent, systematic approach to make decisions. Risk insights are incorporated as appropriate. The inspectors found several scaffolds in the plant that incorporated all of the necessary elements of MA–AA–716–025 related to control of temporary scaffolding. However the scaffolds installed in the Division II CSCS pump room and RHR corner room did not. [H.13]

Enforcement:

Violation: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed by instructions, procedures, or drawings, of a type appropriate to the circumstance and shall be accomplished in accordance with these instructions, procedures, or drawings.

Step 3.6 of procedure MA–AA–716–025, "Scaffold Installation, Modification, and Removal Request Process," Revision 15, required:

> "Scaffold Coordinator/Designee– Is responsible for the coordination of erection and removal of all scaffolds on site. Maintaining a log or electronic equivalent of the status of all scaffolds, and reviewing the log to ensure that any scaffolds approaching their 90 day limit are removed or converted to a permanent scaffold or requesting that an individual 10 CFR 50.59 review be performed for the individual scaffold required to be left in place beyond 90 days."

Step 4.1.5 of procedure MA–AA–716–025, "Scaffold Installation, Modification, and Removal Request Process," Revision 15, required:

“Engineering **APPROVE** post erection inspections required by Engineering.”

Step 4.1.5 of procedure MA–AA–716–025 is implemented by Table 2, Note 3, of engineering standard NES-MS-04.1, “Seismic Prequalified Scaffolds,” Revision 7. NES-MS-04.01, Table 2, Note 3, required:

> “Movement of in-place systems/components are not included in the above clearances and should be increased accordingly. For Byron/Braidwood/Clinton, a 3” clearance shall be provided to account for movement of in-place systems/components, unless otherwise approved by engineering. This clearance is measured from anywhere on the outside surface of the in-place item to the closest point of the scaffolding. For the other stations, Engineering shall be contacted to provide clearance requirements, as required.”

Contrary to the above:

From April 18, 2017, until October 25, 2018, scaffold 1867810 was installed in close proximity to safety-related piping in on the 687 feet elevation, and from March 21, 2017, until October 25, 2018, scaffold 1929990 was installed in close proximity to safety-related piping in on the 694 feet elevation in the Unit 2, Division II core standby cooling system (CSCS) pump room, for a period in excess of 90 days, without engineering approval, was not recorded in the scaffolding log, and was not removed or converted to a permanent scaffold or requested that an individual 10 CFR 50.59 review be performed.

From March 21, 2017, until October 25, 2018, scaffold 1929990 was installed in close proximity to safety-related piping in on the 694 feet elevation in the Unit 2, Division II RHR corner room, for a period in excess of 90 days, without engineering approval, was not recorded in the scaffolding log, and was not removed or converted to a permanent scaffold or requested that an individual 10 CFR 50.59 review be performed.

Disposition: This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy.

71152—Problem Identification and Resolution

Acceptance Criteria for Technical Specification Surveillance Requirement Did Not Consider Instrument Uncertainty			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000373/2018004–02 Open/Closed	None	71152

Introduction:

The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to include an appropriate acceptance criterion for drywell oxygen concentration in surveillance procedures. Specifically, the acceptance criterion did not account for instrument uncertainty.

Description:

Technical Specification (TS) 3.6.3.2, "Primary Containment Oxygen Concentration," required, in part, that primary containment oxygen concentration remain less than 4 percent oxygen by volume during operation in Mode 1. In order to ensure that an event that produces any amount of hydrogen (e.g. a loss of coolant accident) does not result in a combustible mixture inside primary containment, which affects the safety-related function of primary containment, TS surveillance requirement (SR) 3.6.3.2.1 required the licensee to verify primary containment hydrogen remain within the specified limits on a weekly basis. The licensee implemented this surveillance requirement with procedure LOS–AA–W1, "Technical Specifications Weekly Surveillances," Revision 83. This procedure contained instructions to re-inert the drywell at 3.5 percent oxygen to prevent exceeding the 4 percent oxygen limit of TS 3.6.3.2.

On 29 May, 2018, the licensee documented in AR 04141949 that the Unit 1 drywell continuous oxygen monitor (1PL78J) had failed. This was determined while performing a monthly channel check of the post-LOCA containment monitoring system, during which the 1PL78J indicated 1 percent oxygen and the post-LOCA oxygen detection system indicated 4.2 percent oxygen in the drywell. In response to the high oxygen condition in the drywell, the licensee entered TS 3.6.3.2 and re-inerted the drywell to less than the TS oxygen limit. The licensee determined that the issue was not reportable under 10 CFR 50.73(a)(2)(i)(B) for a condition prohibited by TS.

Since 1PL78J is an instrument permitted by procedure LOS–AA–W1 to accomplish SR 3.6.3.2.1, the inspectors reviewed the calibration data for 1PL78J under work order 4718759 and 4789722. During their review, the inspectors noted that instrument calibration procedure, LIP–CM–510, "Unit 1 Primary Containment Continuous Oxygen Monitor Sensor Maintenance and Standardization," Revision 7, Step E.2.10.5, required that oxygen concentration stabilize at approximately 20 to 22 percent and greater than 19.5 percent oxygen as read in air. The inspectors determined that Step E.2.10.5 of LIP–CM–510 contained a leave alone tolerance of approximately 1.4 percent oxygen since air normally contains 20.9 percent oxygen.

The inspectors concluded that since LOS–AA–W1 contained approximately 0.5 percent of margin, and LIP–CM–510 allowed for as much as 1.4 percent deviation in the instrument response of 1PL78J, that the acceptance criteria of LOS–AA–W1 had the potential to allow a condition prohibited by TSs. Further, the inspectors found several instances of instrument drift in the continuous oxygen monitor. This was revealed by abnormally low oxygen readings upon de-inerting the drywell for refueling outages documented in AR 1170271, 1325742, 1689590, 2626000, and 3970533. In some instances, drywell oxygen read as low as 14 percent when normal atmospheric conditions, approximately 20.9 percent oxygen, were expected in the drywell. The inspectors discussed their concerns with the licensee who documented them in the corrective action program as AR 4209125.

Corrective Actions: The licensee intends on performing a loop uncertainty determination for the continuous oxygen monitor and to re-evaluate the acceptance criteria of LIP–CM–510 and LOS–AA–W1 based on the results of the uncertainty determination.

Corrective Action Reference: AR 4209125

Performance Assessment: Performance Deficiency: The inspectors determined that the failure to include appropriate acceptance criteria for drywell oxygen concentration in TS surveillance procedures, an activity affecting quality, was contrary to 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," and was a performance deficiency. Screening: The inspectors determined the performance deficiency was more than minor because it adversely affected the procedure quality attribute of the barrier integrity 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 failure to include instrument uncertainty as part of the acceptance criteria that performs the weekly TS SR had the potential to allow a condition prohibited by TS. Significance: The inspectors assessed the significance of the finding using SDP Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors determined that the finding was of very low safety significance (Green) it did not represent an actual open pathway in the physical integrity of the reactor containment, containment isolation system, or heat removal components. Cross-cutting Aspect: The inspectors determined that there was no cross-cutting issue because the issue was not indicative of current plant performance.
Enforcement: Violation: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures, or drawings include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, from January 16, 1999, to December 31, 2018, the licensee failed to include appropriate acceptance criteria in procedures that implement TS surveillance requirements when the continuous oxygen monitor was installed under engineering change 55203. Specifically, the drywell oxygen acceptance criteria in procedure LOS–AA–W1 did not account for instrument uncertainties to ensure compliance with TS limits. Disposition: This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy.

71153—Follow-Up of Events and Notices of Enforcement Discretion

Licensee Identified Non-Cited Violation	71153—Follow-Up of Events and Notices of Enforcement Discretion
This violation of very low safety significant was identified by the licensee and has been entered into the licensee corrective action program and is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy.	

Violation: Technical Specification 3.4.4 limited condition for operation (LCO) is (applicable for Modes 1, 2 and 3) states: The safety function of 12 safety relief valves (S/RVs) shall be OPERABLE; and action statement A states “One or more required S/RVs inoperable - A.1 be in mode 3 in 12 hours and A.2 be in Mode 4 in 36 hours.” In addition, TS SR 3.4.4.1 stated “Verify the safety function lift setpoints of the required S/RVs are as follows:“

Number of S/RVs	Setpoint (psig)
2	1205 ± 36.1
3	1195 ± 35.8
2	1185 ± 35.5
4	1175 ± 35.2
2	1150 ± 34.5

Contrary to the above, during a portion of the previous two Unit 1 operating cycles from February 7, 2014 through February of 2018, two main steam S/RVs did not meet these lift pressure setpoint requirements. Specifically S/RV 1B21 – F013R lifted at 1167 psig instead of 1168.9 – 1241.1 psig and S/RV 1B21 – F013U lifted at 1109 psig instead of 1115.5 – 1184.5 psig (reference; Licensee Event Report 05000373/2018–003–01, Two Main Safety Relief Valves Failed Inservice Lift Inspection Pressure Test). The licensee replaced the two affected valves and submitted a license amendment request on February 27, 2018, to revise TS 3.4.4.1 and lower the setpoint tolerance (minus five percent) to account for S/RV minor setpoint drift in the conservative direction.

Significance/Severity: This licensee identified finding affected the Initiating Events Cornerstone and was screened in accordance with IMC 0609.04, “Initial Characterization of Findings,” and Exhibit 1 of IMC 0609, Appendix A, “The Significance Determination Process for Findings At Power.” The two affected SRVs lifted low outside of their setpoint band, which was conservative with respect to maintaining the reactor coolant system overpressure protection safety function of these valves. Therefore, the inspectors determined that this finding is of very low safety significance (Green) because after a reasonable assessment of degradation, the finding would not have resulted in exceeding the reactor coolant system leak rate for a small LOCA and did not affect other systems used to mitigate a loss-of-coolant accident.

Corrective Action References: AR 04110929 and AR 04110933

EXIT MEETINGS AND DEBRIEFS

The inspectors confirmed that proprietary information was controlled to protect from public disclosure. No proprietary information was documented in this report.

- On November 9, 2018, the inspector presented the radiation protection program inspection results to Mr. J. Washko, Plant Manager, and other members of the licensee staff.
- On December 7, 2018, the inspector presented the emergency preparedness program inspection results to Mr. M. Hayworth, Site Emergency Preparedness Manager.
- On December 19, 2018, the inspector discussed the completed 2018 Licensed Operator Requalification Program annual operating test inspection results with Ms. D. Fuson, Operations Training Manager.
- On January 9, 2019, the inspector presented the quarterly integrated inspection results to Mr. W. Trafton, and other members of the licensee staff.

DOCUMENTS REVIEWED

71111.01—Adverse Weather Protection

- AR 4172878; EST, TCCP, and Config Control Review for LOS–ZZ–A2
- AR 4181689; EED Switchyard Winterization Inspection
- LOS–ZZ–A2; Preparation for Winter/Summer Operation; Revision 56
- WC–AA–107; Seasonal Readiness; Revision 20
- WO 4717497–01; LOS–ZZ–A2 Winterize Station; 12/6/2018

71111.04—Equipment Alignment

- 4185035; NRC Identified Potential Scaffold Issue in U2 CSCS Pump Room
- MA–AA–716–025; Scaffold Installation, Modification, and Removal Request Process; Revision 15
- LAP–100–65; Equipment/Parts Storage in Plant Areas Containing Safety-Related Equipment; Revision 9
- MA–AA–796–024; Scaffold Installation, Inspection, and Removal; Revision 11
- AR 4179046; SBGT 1FS–VG009 Indication Out of Calibration
- LaSalle NPS 90 Day Scaffold Report; 10/24/2018
- NES–MS–04.1; Seismic Prequalified Scaffolds; Revision 7

71111.05AQ—Fire Protection Annual/Quarterly

- Pre-Fire Plan FZ 3H4; RX Bldg. 694'–6" Elevation U2 RCIC/LPCS Cubicle
- Pre-Fire Plan FZ 3I4; RX Bldg. 673'–4" Elevation U2 LPCS/RCIC Pump Cubicle

71111.06—Flood Protection Measures

- AR 1413252; FUK: Fukushima Flooding Elevation Surveys —Doors 20 and 164
- C467110014-9939; ISOLRB3, Terminate Flood Before HPCS, RHR A & LPCS/RCIC Rooms Reach Critical Height; 2013
- EC 388864; Evaluate Leakage by MS Tunnel Dampers During Flood; Revision 000
- EC 399280; Beyond Design Basis Flooding Analysis for NRC Fukushima NTTF Recommendation 2.1—Plant LIP Ingress; Revision 004
- LOR–1PM13J–A304; RB NE/NW Equip DRN Sump Trouble; Revision 2
- LSCS-UFSAR; 3.4; Water Level (Flood) Design; Revision 20
- PMRQ 64856–01; Inspection of Magenetrol for the U–1 RCIC Pump Room; 10/25/2018
- WO 1112731–01; Perform Inspection of Magentrol; 1/19/2010

71111.07—Heat Sink Performance

- EC 347674; Loop Accuracy for the RHR A & B Heat Exchanger Service Water Inlet Flow; 3/2/2004
- EC 626435; Evaluation of Unit 2B RHR Heat Exchanger Thermal Performance Data Using Alternate (EPRI) Methodology; Revision 000
- WO 1816960–01; 2E12–B001 BRHR HX Heat Xfer Test per LTS–200–17; 12/7/2018

71111.11—Licensed Operator Requalification Program and Licensed Operator Performance

- L2C17–15; L2C17 December 2018 Rod Pattern Adjustment and Quarterly Surveillances; 12/1/2018

71111.12—Maintenance Effectiveness

- NRC Question on Unused Penetrations
- 1E–0–3073; Electrical Installation Fire-Stop & Fire-Barrier Details; Revision H
- LS–PSA–012; LaSalle PRA Internal Flood Analysis; Revision 1
- LTS–1000–29; Water Tight Penetration Inspection; Revision 15
- AR 4193160; NRC Id'd—Inadequate Detail in VR MR Evaluation
- AR 3972328; Left Blade on Damper 2VR08Y is Sticking
- WO 1814934-01; Inspect Steam Tunnel, Check Dampers 2VR08Y thru 2VR14Y; 2/9/2017
- WO 821498–01; Inspect Steam Tunnel Check Dampers 2VR08Y thru 2VR14Y; 2/18/2011
- WO 1909964–01; IVR90Y Press Relief Damper Open Torque Verification; 2/24/2018
- WO 1911148–01; Inspect Steam Tunnel Check Dampers 1VR08Y thru 1VR14Y; Recor *(sic)*; 2/24/2018
- ER–LA–450–1006; LaSalle Structures Monitoring Instructions; Revision 3
- ER–LA–450; LaSalle Structures Monitoring Program; Revision 002
- ER–AA–450; Sturctures Monitoring; Revision 7
- ER–AA–310–1003; Maintenance Rule—Performance Criteria Selection; Revision 5
- 5423 (Drawing); Steam Tunnel Check Dampers; Revision 2
- PMRQ 76697–01; Inspect Steam Tunnel Check Dampers 1VR08Y thru 1VR14Y; 10/10/2018
- PMRQ 75772–01; Inspect Steam Tunnel Check Dampers 1VR08Y thru 1VR14Y Record In; 10/10/2018
- AR 912656; Apply Lubricant to Chain Hold Down Bolt on VR Check Damper
- AR 891020; NOS ID: OPS CPA PRA Key Operator Actions
- M–1438; High Pressure Core Spray Switchgear Room and High Radiation Sampling Ventilation System Elevation 687'–0"; Revision J
- M–3460; HVAC/C&I Diagram Turbine Building Ventilation System; Revision C
- DWG 5559; General Arrangement Steam Tunnel Check Damper; Revision 2
- DWG 5561; Electrical and Control Schematic Steam Tunnel Check Dampers Unit 2 Turbine Building Air Return Risers; Revision 1
- M–1460, Sheets 1 & 2; P&ID Turbine Building Ventilation System; Revision J
- AR 4186903; NRC MR Questions 10/23/2018
- Current Installed Protected Pathway List;12/11/2018
- IR 4302231, "NRC Question On Div 3 Protected Paths; 12/11/18
- S–237; Reactor Building Framing Secion 8–8 Lower Area; Revision X
- WO 1916774–01; Watertight Penetration Inspection, Unit 1; 2/24/2018
- WO 1807623–01; Watertight Penetration Inspection, Unit 2; 4/27/2017
- IE–2–4085AM; Schematic Diagram Turbine Building Ventilation Syst. VT Pt. 12; Revision E

71111.13—Maintenance Risk Assessments and Emergent Work Control

- ECR 436522; Request Approval of Crane and Pad to Support LaSalle FHD Installation, Reactor Building; 7/24/2018
- ECR 436569; Request Approval of Scaffold Plans in Support of LaSalle FHD Installation; 7/30/2018
- L–004116; HCVS Steel Tower Load Drop Analysis, EC 392353–02; Revision 000
- RP–01; Plan on Crane Setup for Scaffold and Pipe Installation; Revision 4

- RP–02; Elevation on Crane Setup for Scaffold and Pipe Installation; Revision 2
- RP–03; Crane Pad Layout Plans, Sections & Details; Revision 3
- WO 4776605–09; Install/Remove Crane Pad for FHD Cooling Project, Reference EC 622967; 7/11/2018
- WO 4776605–11; Erect Outside Tower Scaffold for FHD Cooling Modification
- WO 4776605–27; Install Supports per EC 622967 AWA #3

71111.15—Operability Determinations and Functionality Assessments

- 1E–1–4000CU; Key Diagram 480V MCC 135X–3 (1AP73E); Revision P
- 1E–1–4000CU; Key Diagram 480V MCC 135X–3 (1AP73E); Revision P
- 1E–1–4000FB; Key Diagram 125V DC Distribution Essential Div. 1; Revision T
- 67062E; Turbine File 36687 Drawing; 12/31/1969
- A–261; Reactor Building Wall Sections; Revision D
- AR 4195336; NRC ID'd—RB Penetration Tracking
- AR 4195744; 4.0 Crew Critique and LL of 1DC07E Battery Replacement; 11/16/2018
- CC–AA–204; Control of Vendor Equipment Manuals; Revision 12
- EC 625223; Technical Evaluation of Procedure LEP–DC–116; Revision 0
- EPRI; Technical Report, Nuclear Maintenance Applications Center, RCIC; 2017
- HLA Brief for 1DC07E Div 1 125V Battery Replacement Unit 1 710' Aux Building; Undated
- LEP–DC–116; Division 1 and 2 Switchgear Room 125 Volt Battery Cell Replacement for Units 1 and 2; Revision 1
- LMP–RI–01; Replacement of Outboard Mechanical Seal Assembly; Revision 7
- LOS–DC–Q2; Battery Readings for Safety-Related 250 VDC and Div 1,2,3 125 VDC Batteries; Revision 36
- LOS–RI–Q4; RCIC; Revision 22
- M–1776, Sheets 2 & 4; ASME Weld Map; 11/13/2018
- Money/Johnson Risk Management Communication; Paragon and PRA Model Update; 5/22/2018
- NRC ID: Low Oil Level in U2 RCIC Turbine Sight Glass
- RCIC GS–1, GS–2; Lubrication System; Undated
- WO 4758919–01; Install New Battery Cells for 1DC07E; Undated
- WO 4776605–59; Install Pipe Penetration "A" Through the Reactor Building Metal Siding Wall on 843'; 11/13/2018

71111.18—Plant Modifications

- 4196929; NRC Identified—Questions Regarding CSCS Project Work
- CC–AA–402; Maintenance Specification: Installation of Temporary Rigging; Revision 5
- EC 622658; HP Pipe Loading Impact; Revision 000
- M–74; P&ID Cycled Condensate Storage; Revision AD
- M–766; Outdoor Piping; Revision AD
- M–938; High Pressure Core Spray Piping; Revision F
- NES–MS–04.2; BWR Stations Temporary Rigging Load Criteria; Revision 2
- SK–M–622658–1/EC 622658; Temporary Support for 10: Hot Tap/Plagging Machine; 11/9/2018
- WO 4777021–10; CM Remove Section of 2HP01C–24" / EC 622658; 11/19/2018
- WO 4777021–10; Remove A Section of Line 2HP01C-24" To Provide Access for the Line Stop Machine as Per EC 622658

71111.19—Post Maintenance Testing

- WO 1564237–03; Replace Hydramotor for 2VD19Y; 9/25/2018
- WO 1665024–02; Replace Hydramotor for 2VD03YA/YB; 9/26/2018
- WO 1853112–02; EM Cubicle Insp. LES–GM–108 @ 243–1 7B–2VD07C (DG Inop); 9/25/2018
- WO 1853113–02; Perform LES–GM–108 for Dist Trans @ MCC 243–1 CUB 6D (2AP79E); 9/25/2018
- WO 1853115–02; EM Cubicle Insp. LES–GM–108 @ 243–1 6A–2VD05C (DG Inop); 9/25/2018
- WO 1853409–02; EM Cubicle Insp. LES–GM–108 @ 243–1 4E–2VD01C (DG Inop); 9/26/2018
- WO 1853412–02; EM Cubicle Insp. LES–GM–108 @ 243–1 5B–2D002P (DG Inop); 9/25/2018
- WO 1882382–05; 2E12–F003B Klockner Moeller MCC 2AP82E–A6 Cubicle Replacement; 10/15/2018
- WO 1882382–08; 2E12–F003B Klockner Moeller MCC 2AP82E–A6 Cubicle Replacement; 10/15/2018
- WO 1923264–02; Perform LES–DG–202 Att A, B & C if Applicable on the U2 B D; 9/26/2018
- WO 1937824–03; MM Replace 2E22–F316 With SS Valve per IT–7000–M–PP–16; 9/25/2018
- WO 1964571–02; Inspect 2B Diesel Generator Start Air Moisture Separate; 9/26/2018
- WO 4791067–02; Replace Jacket Water Outlet Gasket on 2B DG; 9/26/2018
- WO 4801283–01; LRA LOS–DG–M3 2B DG Fast Start ATT 2B-Fast; 9/26/2018
- WO 484154–01; IM–EWP–1D–K751D 1D VD Rad Monitor Downscale; 10/23/2018

71111.22—Surveillance Testing

- LOS–FP–M6; Diesel Fire Pump 0A (0FP01KA) Operational Check; 12/19/2018
- LOS–RH–Q1; Unit 2 B RHR System Operability and Inservice Test; 12/18/2018

71114.04—Emergency Action Level and Emergency Plan Changes

- 10 CFR 50.54(q) Evaluator Qualification Spreadsheet; Dated May 30, 2018
- 50.54(q) Evaluation No. 17-105; EP-AA-1005, Addendum 3, Emergency Action Levels for LaSalle Station (Revision 3) Evaluation and Effectiveness Review; Dated October 13, 2017
- 50.54(q) Evaluation No. 17-72; EP-AA-1005, Exelon Nuclear Radiological Emergency Plan Annex for LaSalle Station (Revision 40) Evaluation and Effectiveness Review; Dated August 28, 2017
- 50.54(q) Evaluation No. 17-89; EP-AA-1005, Addendum 3, Emergency Action Levels for LaSalle Station (Revision 3) Evaluation and Effectiveness Review; Dated July 24, 2017
- 50.54(q) Evaluation No. 18-13; EP-AA-1005, Addendum 3, Emergency Action Levels for LaSalle Station (Revision 4) Evaluation and Effectiveness Review; Dated February 21, 2018
- AR 04096437; Typographical Error Identified in EAL Matrix
- AR 04109555; Enhancement Opportunity to Improve Knowledge for the C6 EAL
- EP-AA-1000; Exelon Nuclear Standardized Radiological Emergency Plan: Revision 29
- EP-AA-1005 Addendum 1; LaSalle Station On-Shift Staffing Technical Basis; Revision 1
- EP-AA-1005, Addendum 3; Emergency Action Levels for LaSalle Station; Revisions 2, 3, and 4
- EP-AA-1005; Exelon Nuclear Radiological Emergency Plan Annex for LaSalle Station; Revisions 39 and 40
- EP-AA-120; Emergency Plan Administration; Revision 21

- EP-AA-120-1001; 10 CFR 50.54(q) Change Evaluation; Revision 9

71124.02—Occupational As Low As Reasonably Achievable Planning and Controls

- AR 03949782; NRC Rad Protection Baseline Inspection Self-Assessment; 12/11/2017
- AR 03977765; Increased CRD Dose Rates in L2R16; 02/23/2017
- AR 03983436; Radiation Protection Baseline (71124.02) Self-Assessment; 08/30/2018
- Issue Report 4005125; ALARA Post-Job Review - L2R16 RB/DW Chemical Decontamination Project; 04/17/2017
- LaSalle Station RP / ALARA Refuel Outage Report; L1R17; 2018
- LaSalle Station RP / ALARA Refuel Outage Report; L2R16; 2017
- Radiation Work Permit and Associated ALARA File; LA-01-18-00510; L1R17 DW Steam Safety Relief Valve Activities
- Radiation Work Permit and Associated ALARA File; LA-02-17-00502; L2R16 DW RP Department Activities
- Radiation Work Permit and Associated ALARA File; LA-02-17-00506; L2R16 DW Scaffold
- Radiation Work Permit and Associated ALARA File; LA-02-17-00513; L2R16 DW Control Rod Drive (CRD) Exchange
- Radiation Work Permit and Associated ALARA File; LA-02-17-00547; L2R16 DW RR Motor Replacement
- RP-AA-400; ALARA Program; Revision 15
- RP-AA-400-1001; Establishing Collective Radiation Exposure Annual Business Plan Goals; Revision 5
- RP-AA-400-1004; Emergent Dose Control and Authorization; Revision 9
- RP-AA-400-1006; Outage Exposure Estimating and Tracking; Revision 8
- RP-AA-400-1008; Exposure Goal Recovery Plans; Revision 3
- RP-AA-401; Operational ALARA Planning and Controls; Revision 24

71151—Performance Indicator Verification

- LS-AA-2140; Attachment 1; Monthly Data Elements for NRC Occupational Control Effectiveness; October 2017 through September 2018
- Periodic Assessment of Maintenance Rule Program; LaSalle Station, Units 1 & 2; July 2016 through June 2018
- Periodic Assessment of Maintenance Rule Program; LaSalle Station, Units 0, 1 & 2; July 2014 through June 2016

71152—Problem Identification and Resolution

- AR 00002269; Document Actions Being Taken to Correct Water That is Leaking; 03/29/1999
- AR 2420888; Unit 2 Reactor Cavity Skirt Plate to Drain Line Leakage; 12/04/2014
- AR 4193011; NRC ID: Potential NCV for LaSalle Unit 2 Cavity Leakage; 11/08/2018
- ATI 1470953-18-47; Actions Supporting Issues Identified by LS-AA-2001; 07/29/2014
- Drawing S-326; Sections and Details, Reactor Containment Liner Plate, Sheet 1; Revision AE
- PI-AA-125; Corrective Action Program (CAP) Procedure; Revision 6

71153—Follow-Up of Events and Notices of Enforcement Discretion

- AR 04110929; 1B21-F013R Fails Set-Pressure Test; 02/27/2018
- AR 04110933; 1B21-F013U Fails Set-Pressure Test; 02/27/2018

- Licensee Event Report, 05000373/2013-003-01, Two Main Safety Relief Valves Failed Inservice Lift Inspection Pressure Test, 07/25/2018
- NWS Technology, Letter; LaSalle SRV s/n N63790-05-0016- As found Failure; Revision 0
- NWS Technology, Letter; LaSalle SRV s/n N63790-05-0076- As found Failure; Revision