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

August 12, 2015

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

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2 NRC INTEGRATED INSPECTION REPORT 05000373/2015002; 05000374/2015002

Dear Mr. Hanson:

On June 30, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your LaSalle County Station, Units 1 and 2. On July 8, 2015, the NRC inspectors discussed the results of this inspection with Plant Manager, Mr. H. Vineyard, and other members of your staff. The inspectors documented the results of this inspection in the enclosed inspection report.

One self-revealed finding and two NRC-identified findings of very low safety significance (Green) were identified during this inspection. These findings involved violations of NRC requirements. Two of these violations were determined to be Severity Level IV under the traditional enforcement process. The NRC is treating these violations as non-cited violations (NCVs) 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, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; and the NRC Resident Inspector at the LaSalle County Station.

If you disagree with the cross-cutting aspect assignment 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 Regional Administrator, Region III, and the NRC Resident Inspector at the LaSalle County Station.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, "Public Inspections, Exemptions, Requests for Withholding," of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records System (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA John Jandovitz Acting for/

Michael Kunowski, Chief
Branch 5
Division of Reactor Projects

Docket Nos. 50–373 and 50–374
License Nos. NPF–11 and NPF–18

Enclosure:
Inspection Report 05000373/2015002;
05000374/2015002 w/Attachment:
Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000373; 05000374
License Nos:	NPF–11; NPF–18
Report No:	05000373/2015002; 05000374/2015002
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	Marseilles, IL
Dates:	May 1, 2015 – June 30, 2015
Inspectors:	R. Ruiz, Senior Resident Inspector J. Robbins, Resident Inspector D. Krause, Acting Resident Inspector A. Dunlop, RIII Senior Reactor Engineer D. Chyu, RIII Reactor Engineer M. Holmberg, RIII Senior Reactor Engineer C. Hunt, RIII Reactor Engineer *(Observer)* R. Zuffa, (Illinois Emergency Management Agency), Resident Inspector
Approved by:	M. Kunowski, Chief Branch 5 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

Inspection Report 05000373/2015002; 05000374/2015002; 04/01/2015 – 06/30/2015; LaSalle County Station, Units 1 & 2; Maintenance Risk Assessments and Emergent Work Control; and Plant Modifications

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. One Green finding was self-revealed and two Green findings with associated Severity Level IV non-cited violations (NCVs) of NRC regulations were identified by the inspectors. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," effective date December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated February 4, 2015. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process," Revision 5.

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance (Green) and associated NCV of Technical Specification (TS) 5.4.1, "Procedures," was self-revealed when the licensee failed to properly preplan and perform maintenance in accordance with written procedures and instructions appropriate to the circumstances. Specifically, on May 14, 2015, the Work Order (WO 1643222) for testing of the motor for the Unit 2 reactor core isolation cooling (RCIC) water leg pump and involving operation of the motor's breaker did not include precautions or restrictions to prevent the inadvertent operation, by bumping, of the adjacent breaker for the safety-related Unit 2 "A" residual heat removal (RHR) suppression chamber spray isolation valve. Workers inadvertently bumped and opened the breaker for the RHR valve and rendered the system inoperable.

  The finding was determined to be more than minor because it was associated with 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 licensee failed to provide a work order appropriate to the circumstances of the juxtaposed breakers. The subsequent, inadvertent opening of the 2A RHR suppression chamber spray isolation valve breaker, unexpectedly rendered the valve inoperable. This negatively impacted the RHR suppression chamber spray system's ability to reduce suppression chamber pressure by removing one of the required two spray paths. The inspectors determined the finding to have very low safety significance (Green). This finding has a cross-cutting aspect in the area of Human Performance, Avoid Complacency, because configuration control and error prevention techniques (robust barriers) in an existing licensee procedure were not appropriately implemented due to the failure of individuals to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes (H.12). Specifically, licensee staff failed to implement the guidance found in procedure HU-AA-101, "Human Performance Tools and Verification Practices." (Section 1R13)

Cornerstone: Barrier Integrity

- Severity Level IV. The inspectors identified a Severity Level IV NCV of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, “Changes, Tests, and Experiments,” having very low safety significance (Green), for the licensee’s failure to provide a written safety evaluation supporting the determination that a license amendment was not required for operation with jet pump seal plugs (lost in the reactor vessel in February 2015 during a refueling outage) that could negatively impact fuel bundle cooling during an anticipated operational occurrence (misplaced fuel bundle). The licensee entered this issue into the corrective action program (CAP) as action report (AR) 02486215 and considered the core operable because additional testing demonstrated that with sufficient time (approximately 11 days) at operating temperature, the rubber plugs would degrade and pass through the affected flow orifices.

  The finding was determined to be more than minor because the inspectors could not reasonably determine that the activity, to operate with jet pump plugs blocking peripheral fuel bundle flow, would not have required prior NRC approval. Specifically, if the licensee operated a peripheral blocked fuel bundle coincident with a misplaced fuel bundle, the minimum critical power ratio limits/margins may not have been assured. Additionally, this finding was more than minor because the underlying technical issue adversely affected the Barrier Integrity Cornerstone objective of design control and cladding performance. The finding involves the potential for a misplaced fuel bundle concurrent with complete flow blockage to a fuel assembly. Given standard refueling practices, an error that results in plant operation with a misplaced fuel bundle is very unlikely due to strict procedural controls and multiple verifications of fuel assembly placement. In addition, the misplaced fuel assembly would have to be located at a peripheral core location to be susceptible to a jet pump plug that could possibly block bundle cooling and this was very unlikely. Further, the inspectors considered the relatively short duration of time where the plug material parameters were sufficient to cause plugging of an orifice coincident with plant power levels that could challenge the fuel integrity limits. Given these factors, the inspectors determined that the likelihood of a misplaced fuel assembly combined with a blocked orifice that could result in fuel clad damage was very low. Given the very low likelihood of the event scenario to occur and the low consequences if it were to occur, the inspectors concluded that the finding was of very low safety significance (Green). The inspectors identified a cross-cutting aspect associated with this finding in the area of Human Performance, Conservative Bias, because the licensee staff did not use a decision-making practice that emphasized prudent choices over those that are simply allowable (H.14). (Section 1R18)

- Severity Level IV. The inspectors identified a Severity Level IV NCV of 10 CFR 50.36, “Technical Specifications,” having very low safety significance (Green), for the licensee’s failure to ensure that limiting conditions for operation (LCOs) were contained in the station’s Technical Specifications (TSs). Specifically, as of March 15, 2015, through the Unit 2 Core Operating Limits Report (COLR), Cycle 16, Revisions 1 and 2, the licensee introduced new “Operating Limits for Lost Jet Pump Plug Seals Mitigation Strategy,” that created new LCOs as defined by §50.36(c)(2) but did not incorporate these LCOs into the TSs. The licensee incorrectly believed that because the COLR was revised via the 50.59 process and the special content that accounted for the existence of the plugs was developed using NRC-approved methodologies, the change was acceptable and no change to the TSs was obtained from the NRC.

This finding was considered more than minor because it was associated with the design control attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective of providing 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 Unit 2 COLR was revised in a manner that created new LCOs, and further, could have resulted in the operation of Unit 2 outside of its approved TSs and license. Operating the unit in accordance with its NRC-approved TSs could have resulted in the plant operating in an unanalyzed condition that could have resulted in fuel failure.

The finding involves the potential for a failed safety/relief valve (SRV) or turbine bypass valve concurrent with complete flow blockage to a peripheral fuel assembly, with a simultaneous breakdown of control room operator knowledge of the special steps required by the COLR revision. Given standard operating practices and the significant amount of extra attention and sensitivity placed on the jet pump plugs and their potential effect, an error that results in licensed operators failing to comply with the restrictive limits of the COLR would be very unlikely. Additionally, a read-and-sign was required of all Unit 2 control room operators and supervisors delineating the special compensatory measures to be taken in the event that a COLR base case component, such as an SRV, were to fail. Further, the inspectors considered the relatively short duration of time (March 15–23, 2015) where the plug material parameters were sufficient to cause plugging of an orifice coincident with plant power levels that could challenge the fuel integrity limits. Given these factors, the inspectors determined that the likelihood of a failed COLR base case component, combined with the operation of the unit in an unanalyzed condition in accordance with the NRC-approved TSs, combined with a blocked orifice that could result in fuel clad damage was very low. Given the very low likelihood of the event scenario to occur and the low consequences if it were to occur, the inspectors concluded that the finding was of very low safety significance (Green). The inspectors determined that this finding had a cross-cutting aspect in the area of Human Performance, Change Management, because the licensee leaders did not ensure the use of a systematic process for evaluating and implementing change so that nuclear safety remained the overriding priority (H.3). (Section 1R18)

REPORT DETAILS

Summary of Plant Status

Unit 1

The unit began the inspection period operating at full power. On May 30, 2015, power was reduced to approximately 65 percent to perform control rod sequence exchange and scram time testing. The unit was restored to full power the next day.

Unit 2

The unit began the inspection period operating at full power. On May 20, 2015, the licensee identified potential fuel degradation based on chemistry samples. On May 23, power was reduced to approximately 55 percent for power suppression testing in an attempt to determine the location of the fuel leak. The testing results confirmed the presence of a leak but the results were inconclusive with respect to its exact location in the core. Power was restored on May 28. On May 30, power was reduced to approximately 65 percent to perform control rod sequence exchange and scram time testing. The unit was restored to full power the next day. On June 23, power was reduced to approximately 85 percent due to a vibration in the Unit 2 Train B (2B) turbine-driven feed water pump. After repairs were completed, the unit was in the process of being restored to full power on June 30, and had reached 90 percent on the last day of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness of Offsite and Alternate AC Power Systems

a. Inspection Scope

The inspectors verified that plant features and procedures for operation and continued availability of offsite and alternate alternating current (AC) power systems during adverse weather were appropriate. The inspectors reviewed the licensee’s procedures affecting these areas and the communications protocols between the transmission system operator (TSO) and the plant to verify that the appropriate information was being exchanged when issues arose that could impact the offsite power system. Examples of aspects considered in the inspectors’ review included:

- coordination between the TSO and the plant during off-normal or emergency events;
- explanations for the events;
- estimates of when the offsite power system would be returned to a normal state; and
- notifications from the TSO to the plant when the offsite power system was returned to normal.

The inspectors also verified that plant procedures addressed measures to monitor and maintain availability and reliability of both the offsite AC power system and the onsite alternate AC power system prior to or during adverse weather conditions. Specifically, the inspectors verified that the procedures addressed the following:

- actions to be taken when notified by the TSO that the post-trip voltage of the offsite power system at the plant would not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply;
- compensatory actions identified to be performed if it would not be possible to predict the post-trip voltage at the plant for the current grid conditions;
- re-assessment of plant risk based on maintenance activities which could affect grid reliability, or the ability of the transmission system to provide offsite power; and
- communications between the plant and the TSO when changes at the plant could impact the transmission system, or when the capability of the transmission system to provide adequate offsite power was challenged.

Documents reviewed are listed in the Attachment to this report. The inspectors also reviewed corrective action plan (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into the CAP in accordance with station corrective action procedures.

This inspection constituted one readiness of offsite and alternate AC power systems sample as defined in Inspection Procedure (IP) 71111.01–05.

b. Findings

No findings were identified.

.2 Readiness For Impending Adverse Weather Condition—Heavy Rainfall/External Flooding Conditions

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the expected flooding conditions based on predicted rainfall and rises in local river and lake levels. The evaluation included a review to check for deviations from the descriptions provided in the updated final system analysis report (UFSAR) for features intended to mitigate the potential for flooding. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during the predicted flood conditions or allow water ingress past a barrier. The inspectors also walked down underground bunkers/manholes subject to flooding that contained multiple train or multiple function risk-significant cables. The inspectors also reviewed the abnormal operating procedure and compensatory measures for mitigating the expected flooding conditions to ensure they could be implemented as written. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition sample as defined in IP 71111.01–05

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Unit 1 Train A standby liquid control (SBLC);
- Unit 2 Train A diesel generator (DG) air start system; and
- Unit 2 Train B service water residual heat removal (RHR) system.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, UFSAR, technical specification (TS) requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted three partial system walkdown samples as defined in IP 71111.04–05.

b. Findings

No findings were identified.

.2 Semiannual Complete System Walkdown

a. Inspection Scope

On April 6, 2015, the inspectors performed a complete system alignment inspection of the Unit 2 Class 1E A-C Power System to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; component labeling; component lubrication; component and equipment cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

These activities constituted one complete system walkdown sample as defined in IP 71111.04–05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- fire zone 4E4, Unit 2 Division 2 essential switchgear room at elevation 731´;
- fire zone 4D1, Unit 1 cable spreading room at elevation 749´;
- fire zone 8C1, Unit 2 Division 3 diesel fuel tank room at elevation 674´;
- fire zone 7B6, Unit 1 Division 1 diesel day tank room at elevation 710´; and
- fire zone 4F1, Unit 1 Division 1 essential switchgear room at elevation 710´.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and implemented adequate compensatory measures for out-of-service (OOS), degraded, or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment to this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted five quarterly fire protection inspection samples as defined in IP 71111.05–05.

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the UFSAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. The documents reviewed are listed in the Attachment to this report. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's CAP documents with respect to past flood-related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant area(s) to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- Unit 2 area 3I4, low pressure core spray (LPCS)/ reactor core isolation cooling (RCIC) cubical.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted one internal flooding sample as defined in IP 71111.06–05.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07T)

.1 Triennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed operability determinations, completed surveillances, vendor manual information, associated calculations, performance test results, and cooler inspection results associated with the Unit 1 high pressure core spray (HPCS) DG (1E22–S001) heat exchanger. The heat exchanger was chosen based on its risk-significance in the licensee's probabilistic safety analysis, its important safety-related mitigating system support functions, its operating history and its apparent degrading margin.

For the HPCS DG heat exchanger, the inspectors verified that testing, inspection, maintenance, and monitoring of biotic fouling and macrofouling programs were adequate to ensure proper heat transfer. This was accomplished by verifying: (1) the test method used was consistent with accepted industry practices, or equivalent; (2) the test conditions were consistent with the selected methodology; (3) the test acceptance criteria were consistent with the design basis values; and (4) results of heat exchanger performance testing. The inspectors also verified that the test results appropriately considered differences between testing conditions and design conditions, the frequency of testing based on trending of test results was sufficient to detect degradation prior to loss of heat removal capabilities below design basis values, and test results considered test instrument inaccuracies and differences.

In addition, the inspectors verified the condition and operation of the HPCS DG heat exchanger was consistent with design assumptions in heat transfer calculations and as described in the UFSAR. This included verification that the number of plugged tubes was within pre-established limits based on capacity and heat transfer assumptions. The inspectors verified the licensee evaluated the potential for water hammer and established adequate controls, and operational limits to prevent heat exchanger degradation due to excessive flow-induced vibration during operation. In addition, eddy current test reports and visual inspection records were reviewed to determine the structural integrity of the heat exchanger.

The inspectors verified the performance of ultimate heat sinks (UHS) and safety-related service water systems, and their subcomponents, such as piping, pumps, and valves, by tests or other equivalent methods to ensure availability and accessibility to the in-plant cooling water systems.

The inspectors reviewed the results of the licensee's inspection of the UHS excavations. The inspectors verified that identified settlement or movement indicating loss of structural integrity and/or capacity was appropriately evaluated and dispositioned by the licensee. In addition, the inspectors verified the licensee ensured sufficient reservoir capacity by trending and removing debris or sediment buildup in the UHS as necessary.

The inspectors reviewed the licensee's performance testing of safety-related service water systems and the UHS. This included the review of the licensee's performance test results for key Unit 1 components, including pumps, a motor-operated valve, a check valve, and a relief valve; and the safety-related service water systems flow balance test results. In addition, the inspectors compared the flow balance results to system configuration and flow assumptions during design basis accident conditions.

In addition, the inspectors reviewed condition reports related to the heat exchanger and heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues and to evaluate the effectiveness of the corrective actions. The documents reviewed are listed in the Attachment to this report.

These inspection activities constituted two heat sink inspection samples as defined in IP 71111.07–05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review of Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On April 14, 2015, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program simulator sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation During Periods of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On May 26, 2015, the inspectors observed the Unit 2 reactor power increase following power suppression testing. This was an activity that required heightened awareness or was related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The performance in these areas was compared to pre-established operator action expectations, procedural compliance, and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- Unit 1 RCIC;
- Unit 0 charcoal filter replacement 0VE01FA;
- Unit 1 Train A DG ‘B’ air start motors and oiler replacement; and
- Unit 0 Train A ammonia detectors.

The inspectors reviewed events, such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems, and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for systems, structures, and components (SSCs)/functions classified as (a)(2), or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted four quarterly maintenance effectiveness samples as defined in IP 71111.12–05.

b. Findings

(Opened) Unresolved Item 05000374/2015002-01: Both Ammonia Detectors Non-Functional on Unit 0 Train A of Control Room Ventilation

Introduction: The inspectors identified an unresolved item (URI) concerning the failure of both Unit 0 Train A (0A) control room ventilation ammonia detectors. These detectors were previously removed from the TSs by amendment.

Description: On June 26, licensee staff identified that both ammonia detectors in the 0A Train of control room ventilation were non-functional. The inspectors identified that Amendments 61 and 42 contained language referencing these detectors. The amendments indicated that a commitment was made with respect to maintenance of these detectors. The licensee maintains a listing of current commitments, but this item was not on that list. The actual commitment predated the implementation of the licensee's formal commitment tracking program. The particulars of the commitment were contained in referenced correspondence. A review of the correspondence will allow the inspectors to determine if additional regulatory action is warranted. This issue has been entered into the CAP under AR 2520223, "NRC Question on Ammonia Detector Commitment."

This issue is a URI pending NRC evaluation of the details found in the correspondence referenced above **(URI 05000373/2015002-01, Both Ammonia Detectors Non-Functional on Unit 0 Train A of Control Room Ventilation).**

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- yellow risk for planned Unit 1 Train A DG work window;
- unanticipated operation of Unit 2 Train A RHR suppression chamber/pool and spray isolation valve;
- Unit 2 Train B turbine-driven reactor feed pump with high vibrations;
- technical support center (TSC) heating, ventilation and air conditioning (HVAC) failure (TSC Inoperable);
- Unit 1 average power range monitor Channel B Rod Block Failure; and
- site response to tornado warning for east-central LaSalle County on June 22.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

Documents reviewed are listed in the Attachment to this report.

These maintenance risk assessments and emergent work control activities constituted six samples as defined in IP 71111.13–05.

b. Findings

Inadvertent Operation of Breaker for Unit 2 Train A Residual Heat Removal Suppression Chamber Spray Isolation Valve

Introduction: A finding of very low safety significance (Green) and associated non-cited violation (NCV) of TS 5.4.1, "Procedures," was self-revealed when the licensee failed to properly preplan and perform maintenance in accordance with written procedures appropriate to the circumstances. Specifically, plant personnel inadvertently bumped and actuated the 2A RHR suppression chamber spray isolation valve breaker while attempting to operate the adjacent breaker for the U2 RCIC water leg pump motor as part of work activities under WO 1643222.

Description: On May 14, 2015, while attempting to operate the breaker of the U2 RCIC water leg pump motor for testing, workers inadvertently bumped and opened the adjacent breaker for the 2A RHR suppression chamber spray isolation valve (235Y-2C3). Opening this breaker caused the inoperability of the 2A suppression chamber spray isolation valve (2E12-F027A) and, therefore, the 2A suppression chamber spray system, as well as position indication for the 2A suppression chamber spray isolation valve post-accident monitoring. This rendered the 2A suppression chamber spray system inoperable and caused the entry into TS 3.6.2.4 Required Action A.1, "Restore the 2A Suppression Chamber Spray system within 7 days," and TS 3.3.3.1 Required Action A.1, "Restore the required channel to operable within 30 days." The licensee restored power approximately 30 minutes after the inadvertent opening and initiated an apparent cause investigation of the event.

The licensee's apparent cause investigation indicated that staff involved in the opening of the incorrect breaker were aware of station procedure OP-AA-108-101, "Control of Equipment and System Status," step 3.4 of which stated that all personnel were responsible to ensure changes in plant configuration were performed in a carefully planned and controlled manner and that status of equipment was required to ensure the safety of personnel and equipment, ensure that systems would perform as required, and ensure that configuration management was maintained. The licensee had developed in support of this stated objective procedure HU-AA-101, "Human Performance Tools and Verification Practices," which included tools and practices to prevent the type of inadvertent breaker opening that occurred on May 14, 2015.

For example, section 4.6 of HU-AA-101, "Flagging/Robust Operational Barriers," directs workers to use robust operational barriers during work near "trip-sensitive" or otherwise risk important equipment (to prevent inadvertent manipulation of the equipment / component). Step 4.6.6(2) states: "Apply robust operational barriers to designated components that are not to be manipulated (i.e., rope, paper, or other physical barrier). Electrical safety and effect on sensitive equipment should be considered." In this instance, the handle for the breaker that was inadvertently moved was manufactured with a location for a locking pin to be inserted. If the workers had used a locking pin (robust barrier) on the suppression chamber valve breaker, they could have prevented the handle of the breaker from being repositioned even in the case of inadvertent contact.

Also, section 4.8 of the procedure, "2 Foot Zone Rule," states in a note: "the purpose of the 2 Foot Zone rule is to heighten awareness of plant personnel to the risk of inadvertent bumping and mispositioning of plant components. The rule applies to all site personnel and contractors. […]" Step 4.8.3 states, "evaluate installing robust operational barriers to positionable components within 2 ft. of the work area."

Notwithstanding the plant procedure for preventing inadvertent operation of equipment, the inspectors noted that WO 1643222 covering the inspection of the RCIC water leg pump motor did not contain the necessary controls to ensure that safety-related equipment was not adversely affected during maintenance.

<u>Analysis</u>: The inspectors determined that the inadvertent opening of the RHR system breaker was a performance deficiency warranting further evaluation. Using the guidance in Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, the inspectors determined the performance deficiency was more than minor, and therefore a finding, because it was associated with 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 licensee failed to provide written procedures (a work order) appropriate to the circumstances of juxtaposed breakers. In addition, workers did not follow procedural guidance regarding robust barriers during planned work on the U2 RCIC Water Leg Pump circuit breaker (EQ Motor Inspection). The subsequent, inadvertent opening of the 2A RHR Suppression Chamber Spray Isolation Valve circuit breaker (235Y-2 C3), unexpectedly rendered the 2A RHR Suppression Chamber Spray Isolation Valve inoperable. This negatively impacted the RHR Suppression Chamber Spray system's ability to reduce suppression chamber pressure by removing one of the required two spray paths.

Using Exhibit 2 of IMC 0609, Appendix A, "The SDP for Findings At-Power," dated June 19, 2012; the finding was determined to have very low safety significance (Green) because all the screening questions were answered "No."

This finding has a cross-cutting aspect in the area of Human Performance, Avoid Complacency, because configuration control and error prevention techniques (robust barriers) were not appropriately implemented due to the failure of individuals to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes (H.12). Specifically, licensee staff failed to implement the guidance found in HU-AA-101, "Human Performance Tools and Verification Practices," due to overconfidence and the perceived lack of intrusiveness of the work.

Enforcement: Technical Specification Section 5.4.1 states, in part, that "written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Section 9 of the Regulatory Guide states, in part, that maintenance that can affect safety-related equipment should be properly preplanned and performed in accordance with written procedures and instructions appropriate to the circumstances.

Contrary to the above, on May 14, 2015, maintenance on the U2 RCIC water leg pump motor conducted under WO 1643222 was not properly planned and performed in accordance with written procedures and instructions appropriate to the circumstances in that the WO required operation of the breaker for the RCIC water leg pump motor but did not include precautions or restrictions to prevent the inadvertent operation, by bumping, of the adjacent breaker for the safety-related Unit 2 RHR suppression chamber spray isolation valve.

The licensee planned to implement the following corrective actions to address this issue: 1) supervisors would verify the use of robust barriers for 2 months and would evaluate the need to extend this date based on performance over that period, and 2) work planning would add explicit guidance relating to the use of robust barriers in future work packages associated with removal or replacement of motor control center breakers.

Since this issue was entered into the licensee's CAP (as ARs 02500451 and 02501055), this violation of TS 5.4.1.a, is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy **(NCV 05000373/15002-02, Inadvertent Operation of Circuit Breaker Affecting Unit 2 Train A Residual Heat Removal Suppression Chamber Spray Isolation Valve)**.

1R15 Operability Determinations and Functionality Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Unit 1 Train A DG, air-start motor oiler problem;
- Unit 1 control rod drive pump room water-tight door, damaged seal;
- Unit 2 Train A DG, air system check valve failure;
- Unit 0 DG, loose mounting bolts; and
- reactor building overhead crane, analysis software question.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to the licensee's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sampling of CAP documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted five samples as defined in IP 71111.15–05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following Unit 2 permanent modifications:

- engineering change (EC) 400019 lost parts evaluation for LaSalle Unit 2 Cycle 15 refueling outage (L2R15) activities;
- EC 400989; lost parts evaluation—miscellaneous parts from L2R15; and
- EC 401294; supplemental evaluation to Unit 2, Cycle 16, lost parts evaluation.

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation and safety evaluation screening against the design basis, the UFSAR, and the TSs, as applicable, to verify that the modification did not affect the operability or availability of the systems. The inspectors, as applicable, observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed are listed in the Attachment to this report.

This inspection constituted three permanent plant modification samples as defined in IP 71111.18 05.

b. Findings

Inadequate 10 CFR 50.59 Evaluation for Jet Pump Plugs Affecting Fuel Bundle Cooling

Introduction: The inspectors identified a Severity Level IV NCV of 10 CFR 50.59 "Changes, Tests, and Experiments," having very low safety significance (Green) for the licensee's failure to provide a written safety evaluation for the determination that operation with unretrieved jet pump seal plugs that were postulated to block cooling flow to a peripheral fuel bundle during an abnormal operational occurrence (AOO) event (misplaced fuel bundle) did not require a license amendment. Specifically, the licensee failed to provide a basis as to why power operation with jet pump plugs that were postulated to block cooling flow to a peripheral fuel bundle did not create a more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety (fuel bundle or fuel cladding) or result in a design basis limit for a fission product barrier being exceeded, and as such required NRC prior approval.

Description: The inspectors identified that the licensee did not provide a basis as to why power operation with jet pump plugs that were postulated to block cooling flow to a peripheral fuel bundle did not create a more minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety (fuel bundle or fuel cladding) or result in a design basis limit for a fission product barrier being exceeded. Specifically, the licensee failed to evaluate the impact of blocked fuel bundle cooling flow on a misplaced fuel assembly, which was an AOO event as described in UFSAR Section 15.4.7 “Misplaced Fuel Assembly and Subsequent Operation.”

In February 2015, during the removal of the jet pump plugs after a maintenance evolution in the Unit 2 refueling outage, three rubber seals were missing from the plug assemblies and were presumed to be entrained at the orifice entrance to the fuel bundles, reducing flow through the affected bundles. The three missing jet pump plug seals were 1.75 inches in diameter by 0.375 inches thick with a 0.18-inch diameter center bore. They were fabricated from an Ethylene Propylene Diene Monomer peroxide cured rubber with a vendor product name of Nordel 1700, produced over 20 years ago, which does not readily decompose under normal reactor operating conditions. This material included polymers, oils, and carbon black, was not subject to thermal breakdown at temperatures below 662 °F, nor subject to rapid decomposition in radiation fields below 1 billion units of radiation adsorbed dose. The licensee initially completed an evaluation to return to power operation with these plugs unaccounted for, and presumed the plugs would pass through the orifice openings to a debris filter located on the bottom of the fuel bundles and result in minimal flow blockage. However, after additional testing of the Nordel 1700 material, the licensee concluded that it was possible that substantial flow blockage could occur to the peripheral fuel bundles (98.5 – 100 percent). The licensee determined that the TS 3.2.2 minimum critical power ratio (MCPR) would be the most limiting fuel integrity limit challenged by bundle flow blockage and initially limited power to less than 25 percent to ensure compliance with this TS (This TS is not applicable when reactor power is below 25 percent power). Subsequently, the licensee completed a number of analyses and a 10 CFR 50.59 evaluation to support increasing power above 25 percent.

As discussed in the Basis for TS 3.2.2, the MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit is set such that 99.9 percent of the fuel rods are expected to avoid boiling transition if the limit is not violated. The operating limit MCPR is established to ensure that no fuel damage results during AOOs. Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition, the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion. Additionally, the TS 3.2.2 basis states “To ensure that the MCPR Safety Limit is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR. When the largest change in CPR is added to the MCPR Safety Limit, the required operating limit MCPR is obtained.”

In UFSAR Section 4.2.3.1, “Fuel Damage Analysis,” fuel damage is defined as a perforation of the fuel rod cladding which would permit the release of fission products to the reactor coolant. The mechanisms which could cause fuel damage in reactor operational transients are: (1) severe overheating of the fuel rod cladding caused by inadequate cooling, and (2) rupture of the fuel rod cladding due to strain caused by relative expansion of the uranium-oxide (UO_2) pellet. The UFSAR Section 4.2.1.1, “Safety Design Bases,” states “The fuel assembly is designed to ensure, in conjunction with the core nuclear characteristics (Section 4.3), the core thermal and hydraulic characteristics (Section 4.4), the plant equipment characteristics and the instrumentation and protection system, that fuel damage does not result in the release of radioactive materials in excess of the guideline values of 10 CFR 20, 50, and 100.” Also, UFSAR Section 4.2.1.2, “Power Generation Design Basis,” states “The fuel assembly is designed to ensure, in conjunction with the core nuclear characteristics, the core thermal and hydraulic characteristics, the plant equipment characteristics and the instrumentation and protection system, that fuel damage limits will not be exceeded during either planned operation or abnormal operational transients caused by any single equipment malfunction or single operator error.” The analysis supporting the conclusions in the preceding UFSAR sections did not include an evaluation for the effect of blocked cooling flow through the fuel bundles. The licensee had included assessments of the effect of loose parts on fuel rod behavior in UFSAR Section 4.2.3.13, “Fuel Rod Behavior Effects from Coolant Flow Blockage,” and UFSAR Section 15.C.6, “Lost Part Analyses.” These evaluations did not assume an AOO event or accident occurred coincident with bundle flow blockage caused by loose parts and the supporting analysis discussed in these Sections have not been reviewed or approved by the NRC.

The inspectors reviewed Revision 2 of 10 CFR 50.59 Evaluation L15–47, “Evaluation of Fuel Licensing Impact of Lost Jet Pump Seals in Primary System.” In this document, the licensee stated that “The lost jet pump plug seals have the potential to block the fuel support piece orifice flow to fuel assemblies with peripheral fuel support pieces. This potential flow reduction could lower the margin to a peripheral assembly MCPR limit (Technical Specification 2.1 and 3.2.2), leading in turn to transition boiling, dryout, and fuel damage.” In L15–47, the licensee had included the results and conclusions from a vendor analysis that assessed thermal-hydraulic stability of the fuel bundle with flow blockage for each of the applicable AOOs identified in Chapter 15 of the UFSAR except for one. Specifically, the licensee had reviewed and accepted a supporting vendor document—GE Hitachi Nuclear Energy Document 002N5595 R0, “LaSalle County Nuclear Power Station, Unit 2 Cycle 16 Jet Pump Seal Flow Blockage AOOs Evaluation for Power Ascension,” which evaluated the AOOs coincident with flow blockage, except for the effect of a misplaced fuel bundle coincident with flow blockage. The vendor had not completed an analysis for this AOO, because the vendor judged this scenario to be highly unlikely. Without this information, the licensee had not provided an adequate basis for the “No” response to the following questions and support the determination that a license amendment was not required.

- Does the proposed activity result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR?
- Does the proposed activity result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered?

Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” endorses Nuclear Energy Institute (NEI) 96–07, “Guidelines for 10 CFR 50.59 Implementation,” as providing acceptable guidance for performing 50.59 evaluations. In addressing the questions above, the licensee had not followed the NEI 96–07 guidelines. Specifically:

- NEI 96–07, Section 4.3.2, “Does the Activity Result in More Than a Minimal Increase in the Likelihood of Occurrence of a Malfunction of an SSC Important to Safety?”, identifies that departures from the design as outlined in the General Design Criteria (reference 10 CFR 50, Appendix A) are not compatible with the “no more than minimal increase standard.” In this case, Criterion 10, “Reactor Design,” of the General Design Criteria identified that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. Therefore, accepting fuel bundle flow blockage, without an evaluation of the misplaced fuel bundle AOO event to demonstrate that fuel design limits were not exceeded, potentially created a more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety.

- NEI 96–07 Section 4.3.7, “Does the activity result in a design basis limit for a fission product barrier being exceeded or altered?”, identifies that for the fission product barriers (e.g., fuel cladding) that the design basis limit pertains to the controlling numerical values in the UFSAR which includes the MCPR and Linear Heat Generation Rate limits. In this case, the licensee had accepted a blocked fuel orifice affecting fuel bundle cooling without demonstrating that the fuel cladding design basis limits would not be exceeded. Specifically, UFSAR Section15.4.7 described that the single loop MCPR and the Linear Heat Generation Rate limits would ensure the fuel cladding is protected during power operation with a misplaced fuel assembly. Therefore, accepting fuel bundle blockage to a misplaced fuel assembly without evaluation of the impact to the MCPR or Linear Heat Generation Rate limits could have resulted in the design basis limits for a fission product barrier (e.g., fuel cladding) being exceeded.

In L15–47, the licensee had not included or addressed the 10 CFR 50.59(c)(2)(vi) criterion - Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR. For this criteria, NEI 96–07, Section 4.3.6, “Does the Activity Create a Possibility for a Malfunction of an SSC Important to Safety with a Different Result?”, describes a malfunction that involves an initiator or failure whose effects are not bounded by those explicitly described in the UFSAR as a malfunction with a different result. In this case, the licensee had accepted a malfunction (blocked fuel orifice affecting fuel bundle cooling) without demonstrating that the effects were bounded by those explicitly described in UFSAR Section 15.4.7. Specifically, Section 15.4.7 described that the single loop MCPR and the Linear Heat Generation Rate limits would ensure the fuel cladding was protected during power operation with a misplaced fuel assembly. However, if cooling flow to a misplaced fuel assembly was blocked by a jet pump plug, the design basis limits for the fuel clad fission product barrier could potentially be exceeded and result in a breach of the fuel cladding (e.g., a different result).

The licensee entered this issue into the CAP (AR 02486215) and considered the core operable because additional testing demonstrated that with sufficient time (approximately 11 days) at operating temperature, the rubber plugs would degrade and pass through the affected flow orifices.

Analysis: The inspectors determined that failure to perform a written safety evaluation for the effect of jet pump seal plugs on fuel bundle cooling during an AOO event (misplaced fuel assembly) was contrary to 10 CFR 50.59(d)(1) and a performance deficiency.

The finding was determined to be more than minor because the inspectors could not reasonably determine that the activity to operate with jet pump plugs blocking peripheral fuel bundle flow would not have required prior NRC approval. Specifically, if the licensee operated a peripheral blocked fuel bundle coincident with a misplaced fuel bundle, the MCPR limits/margins may not have been assured. Additionally, this finding was more than minor because, the underlying technical issue adversely affected the Barrier Integrity Cornerstone objective of design control and cladding performance.

Because violations of 10 CFR 50.59 potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process. In accordance with Section 6.1.d.2 of the NRC Enforcement Policy, this violation was categorized as Severity Level IV because the licensee's failure to perform a review for the effect of jet pump seal plugs on fuel bundle cooling during an AOO event was of very low safety significance. This violation was also associated with a finding that has been evaluated by the significance determination process (SDP) and communicated with a SDP color reflective of the safety impact of the deficient licensee performance. The SDP, however, does not specifically consider regulatory process impact. Thus, although related to a common regulatory concern, it is necessary to address the violation and finding using different processes to correctly reflect both the regulatory importance of the violation and the safety significance of the associated finding.

Based upon a review of IMC 0609, Attachment 4, "Initial Characterization of Findings," and IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors and the RIII Senior Reactor Analyst determined that the existing SDP was not capable of evaluating the finding and the associated degraded condition. Specifically, a bounding quantitative risk evaluation could not be performed for this degraded condition because neither licensee probability risk analysis models nor the existing standardized plant analysis risk models address loss of cooling to a single fuel bundle and the subsequent localized fuel damage that may occur. Current SDP metrics are change in core damage frequency and change in large early release frequency. Localized fuel damage events that cannot result in widespread core damage with the potential for a significant release are not addressed by existing SDP tools and methods. Therefore, the inspectors performed a qualitative evaluation of the safety significance in accordance with guidance provided in IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Attributes."

The finding involves the potential for a misplaced fuel bundle concurrent with complete flow blockage to a fuel assembly. Given standard refueling practices, a misplaced fuel bundle is very unlikely due to strict procedural controls and multiple verifications of fuel assembly placement. In addition, the misplaced fuel assembly would have to be located at a peripheral core location to be susceptible to a jet pump plug that could possibly block bundle cooling and this was also very unlikely. Further, the inspectors considered the relatively short duration of time (March 15-23) where the plug material parameters were sufficient to cause plugging of an orifice coincident with plant power levels that could challenge the fuel integrity limits. Given these factors, the inspectors determined that the likelihood of a misplaced fuel assembly combined with a blocked orifice that could result in fuel clad damage was very low. Additionally, a vendor topical report, GEH NEDO–10174, "Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor," had been submitted to the NRC and in this document the fuel vendor concluded that a fuel bundle flow blockage incident would not result in local high pressure production, propagation to adjacent assemblies, or offsite doses in excess of small fractions of 10 CFR 100 guidelines. Given the very low likelihood of the event scenario to occur and the low consequences if it were to occur, the inspectors concluded that the finding was of very low safety significance (Green).

The inspectors identified a cross-cutting aspect associated with this finding in the area of Human Performance, Conservative Bias, because the licensee staff did not use a decision making-practice that emphasized prudent choices over those that are simply allowable (H.14). Specifically, the failure to perform a complete safety evaluation for the effect of jet pump plugs on fuel bundle cooling during an AOO event (misplaced fuel assembly) was based upon the licensee's belief that it was permissible to initiate at-power operations rather than the correct and prudent choice to complete this analysis prior to at-power operations.

<u>Enforcement</u>: Title 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments as described in the UFSAR. These records must include a written evaluation which provides a basis for the determination that the change, test, or experiment does not require a license amendment.

Contrary to the above, on March 14, 2015, the licensee made changes pursuant to 10 CFR 50.59(c) to the plant as described in the UFSAR and had not performed a written evaluation which provided the bases for determining that these changes did not require a license amendment. Specifically, the licensee did not provide a written evaluation for the determination of the effect of jet pump seal plugs on fuel bundle cooling during an AOO event (misplaced fuel assembly) did not require a license amendment. In accordance with the Enforcement Policy, Section 6.1.d.2, the violation was classified as a Severity Level IV violation. Because this violation was of very low safety significance, was not repetitive or willful, and was entered into the licensee's CAP (AR 02486215), this violation is being treated as a Severity Level IV NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy **(NCV 05000374/2015002–03; Inadequate 10 CFR 50.59 Evaluation for Jet Pump Plugs Affecting Fuel Bundle Cooling)**.

<u>(Closed) Unresolved Item–COLR Revision Potentially Created Non-Conservative Technical Specifications</u>

<u>Failure to Include Limiting Conditions for Operation in the Technical Specifications</u>

<u>Introduction</u>: The inspectors identified a Severity Level IV NCV of 10 CFR 50.36, "Technical Specifications," having very low safety significance (Green), for the licensee's failure to ensure that limiting conditions for operations (LCOs) were contained in the

station's TSs. Specifically, through the Unit 2 Core Operating Limits Report (COLR), Cycle 16, Revisions 1 and 2, the licensee introduced new "Operating Limits for Lost Jet Pump Plug Seals Mitigation Strategy," which created new LCOs as defined by §50.36(c)(2), but had not included them in the TSs.

Description: As part of the overall followup review of the Unit 2 Jet Pump Plug issue and the associated URI, the inspectors reviewed the changes made to the Unit 2 COLR, Cycle 16, Revisions 1 and 2. The inspectors assessed the changes with respect to their impact on the current licensing basis, e.g., TSs and regulations, such as 10 CFR 50.36.

In Revision 1 of LaSalle Unit 2's Cycle 16 COLR, the licensee introduced a new section in the form of an Appendix, entitled "Operating Limits for Lost Jet Pump Plug Seals Mitigation Strategy." This appendix stated, "The following limits apply while the jet pump plug peripheral bundle blocked orifice condition exists." Specifically, item 4, entitled "Other Requirements," stated in part that "All equipment must be in-service. This included the EOOS [equipment out-of-service] assumed in the Base Case mentioned in Footnote 1 of COLR Section 10 EXCEPT LPRMs (local power range monitors) and TIPOOS (traversing in-core probe out-of-service) … In the event of an EOOS, take action in accordance with TS 3.2.2 ACTION statements." Those TS actions were to "Reduce THERMAL POWER to < 25% RTP (rated thermal power)" within 4 hours.

The equipment referenced in the COLR Section 10 Base Case that had associated TS LCOs were SRVs (LCOs 3.4.4 and 3.5.1) and turbine bypass valves (LCO 3.7.7).

- LCO 3.4.4 stated "The safety function of 12 SRVs shall be OPERABLE." Unit 2 had a total of 13 safety/relief valves (SRVs), so this LCO essentially allowed one SRV to be OOS indefinitely with no further action required; however, since the COLR created a new operational restriction to prohibit any SRVs from being OOS in order to maintain the unit in an analyzed condition, the inspectors noted the non-conservatism that the COLR created for LCO 3.4.4. Specifically, under an identical condition of 1 SRV OOS, the COLR would have required the unit to downpower to less than 25 percent power, while the TSs would have allowed continuous operation at full power.
- LCO 3.5.1 stated "[…] the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE." Unit 2 had a total of seven ADS SRVs, so this LCO essentially would have allowed one ADS SRV to be OOS indefinitely with no further action required; however, since the COLR created a new operational restriction to prohibit any SRVs from being OOS in order to maintain the unit in an analyzed condition, the inspectors noted the non-conservatism that the COLR created for LCO 3.5.1. Specifically, under an identical condition of 1 ADS SRV OOS, the COLR would have required the unit to downpower to less than 25 percent power, while the TSs would have allowed continuous operation at full power.
- LCO 3.7.7 states "The Main Turbine Bypass System shall be OPERABLE. OR LCO 3.2.2, 'MINIMUM CRITICAL POWER RATIO (MCPR),' limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable." In this example, because the existing TS LCO had a standing reference to the COLR as containing the controlling limits when applicable, this LCO was not considered to be rendered non-conservative as a result of the COLR revision.

This condition only applied to Unit 2 from March 15–23, 2015, when rated thermal power was greater than or equal to 25 percent, which made the TS 3.2.2. MCPR limits in the COLR applicable; and while the plugs were still believed to pose a risk of blocking peripheral fuel orifices, i.e., prior to point that testing results showed sufficient material degradation from the harsh reactor environment to push the plugs through the orifice. Even though the Unit 2 COLR still contained the jet pump plug-related restrictions, they were not in effect because the plugs were shown through analysis to no longer pose a risk to blocking fuel bundle flow.

Discussions with the NRC Office of Nuclear Reactor Regulation confirmed that the above examples of COLR restrictions qualified as LCOs, and as such, were required to be in the TS per 10 CFR 50.36. Changes to a station's TS LCOs require prior NRC review and approval via the license amendment process. This did not take place. Instead, the licensee incorrectly believed that because the COLR was revised via the 50.59 process and the special content that accounted for the existence of the plugs was developed using NRC-approved methodologies, that the change was acceptable.

Although the physical risk associated with the jet pump plugs no longer exists, the current TSs remain non-conservative as the logical pathways discussed above still exist. One potential path forward, should the licensee choose to retain the current COLR content, would involve the process outlined in NRC Administrative Letter 98-10. Per that process, the licensee may create administrative controls that ensure continued safe operation, while they process an amendment request to modify the current TSs. Other options exist and at this time, the licensee has not selected a path for final resolution of this issue.

Analysis: The inspectors determined that the licensee's failure to ensure that LCOs were contained within the TS was contrary to 10 CFR 50.36(c)(2) and a performance deficiency.

This finding was considered more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," issued September 07, 2012, because it was associated with the Design Control Attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective of providing 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 Unit 2 COLR was revised in a manner that created new LCOs, and further, could have resulted in the operation of Unit 2 in a manner outside of its approved TSs and license. Operating the unit in accordance with its NRC-approved TSs could have resulted in the plant operating in an unanalyzed condition that could have resulted in fuel failure.

Because violations of 10 CFR 50.36 potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process. In accordance with Section 6.1.d.2 of the NRC Enforcement Policy, this violation was categorized as Severity Level IV because the licensee's failure to include LCOs in the TSs was of very low safety significance. This violation was also associated with a finding that has been evaluated by the SDP and communicated with a SDP color reflective of the safety impact of the deficient licensee performance. The SDP, however, does not specifically consider regulatory process impact. Thus, it is necessary to address the violation and finding using different processes to correctly reflect both the regulatory importance of the violation and the safety significance of the associated finding.

Based upon a review of IMC 0609, Attachment 4, "Initial Characterization of Findings," and IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors and the RIII Senior Reactor Analyst determined that the existing SDP was not capable of evaluating the finding and the associated degraded condition. Specifically, a bounding quantitative risk evaluation could not be performed for this degraded condition because neither licensee probability risk analysis models nor the existing standardized plant analysis risk models address loss of cooling to a single fuel bundle and the subsequent localized fuel damage that may occur. Current SDP metrics consider only change in core damage frequency and change in large early release frequency. Localized fuel damage events that cannot result in widespread core damage with the potential for a significant release are not addressed by existing SDP tools and methods. Therefore, the inspectors performed a qualitative evaluation of the safety significance in accordance with guidance provided in IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Attributes."

The finding involves the potential for a failed SRV or turbine bypass valve concurrent with complete flow blockage to a peripheral fuel assembly, with a simultaneous breakdown of control room operator knowledge of the special steps required by the COLR revision. Given standard operating practices and the significant amount of extra attention and sensitivity placed on the jet pump plugs and their potential effect, an error that resulted in licensed operators failing to comply with the restrictive limits of the COLR would be very unlikely. Additionally, a read-and-sign was required of all Unit 2 control room operators and supervisors delineating the special compensatory measures to be taken in the event that a COLR base case component, such as an SRV, were to fail. Further, the inspectors considered the relatively short duration of time (March 15, 2015 through March 23, 2015) where the plug material parameters were sufficient to cause plugging of an orifice coincident with plant power levels that could challenge the fuel integrity limits. Given these factors, the inspectors determined that the likelihood of a failed COLR base case component, combined with the operation of the unit in an unanalyzed condition in accordance with the NRC-approved TS, combined with a blocked orifice that could result in fuel clad damage was very low. Additionally, a vendor topical report, GEH NEDO-10174, "Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor," had been submitted to the NRC and in this document, the fuel vendor concluded that a fuel bundle flow blockage incident will not result in local high pressure production, propagation to adjacent assemblies, or offsite doses in excess of small fractions of 10 CFR 100 guidelines. Given the very low likelihood of the event scenario to occur and the low consequences if it was to occur, the inspectors concluded that the finding was of very low safety significance (Green).

The inspectors determined that this finding had a cross-cutting aspect in the area of Human Performance, Change Management, because the licensee leaders did not ensure the use of a systematic process for evaluating and implementing change so that nuclear safety remained the overriding priority (H.3). Specifically, the licensee incorrectly assumed that because the COLR was revised via the 50.59 process and that the special content that accounted for the existence of the plugs was developed using NRC approved methodologies, that the change was acceptable; however, the licensee failed to comply with 10 CFR 50.36 by failing to ensure that LCOs were in the TSs.

Enforcement: Title 10 CFR 50.36(c) states, in part that "Technical Specifications will include items in the following categories: […] (2) Limiting conditions for operation," which defines LCOs as the "the lowest functional capability or performance levels of equipment required for safe operation of the facility."

Contrary to the above, on March 15, 2015, the licensee revised the Unit 2 Cycle 16 COLR and introduced a new section in the form of an Appendix entitled "Operating Limits for Lost Jet Pump Plug Seals Mitigation Strategy," which created new LCOs and were not included in the TSs.

In accordance with the Enforcement Policy, Section 6.1.d.2, the violation was classified as a Severity Level IV violation. Because this violation was of very low safety significance, was not repetitive or willful, and was entered into the licensee's CAP (as AR 2482812), it is being treated as a Severity Level IV NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy **(NCV 05000374/2015002-04; Failure to Include Limiting Conditions for Operation in the Technical Specifications)**.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance testing (PMT) activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Unit 1 RCIC;
- Unit 1 Train A DG air start replacement fast start PMT;
- Unit 1 Train A hydraulic control unit pump functional/leak check PMT;
- Unit 1 Division I reactor water cleanup (RWCU) timer remove and replace;
- Unit 1 Division II RWCU timer remove and replace;
- Unit 2 Train A DG check valve maintenance; and
- Unit 1 Train A SBLC pump replacement PMT.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TSs, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed CAP documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted seven post-maintenance testing samples as defined in IP 71111.19–05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- Unit 1 average power range monitor channel B rod block surveillance (routine);
- Unit 1 LPCS pump evaluation (routine);
- Unit 2 Train A DG idle start (routine);
- Unit 1 Train B DG idle start (routine);
- Unit 1 Train A RHR, service water biennial comprehensive in-service testing (IST) pump test (IST);
- Unit 2 Train B RHR service water biennial IST LOS–RH–Q1 (IST); and
- Unit 2 Train C RHR (IST).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- the effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the UFSAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers code, and reference values were consistent with the system design basis;

- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted three inservice testing samples and four routine samples as defined in IP 71111.22, Sections –02 and –05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Security

4OA1 Performance Indicator Verification (71151)

.1 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS leakage performance indicator (PI) for Units 1 and 2 for the second quarter 2014 through the first quarter 2015. To determine the accuracy of the PI data reported, PI definitions and guidance contained in NEI 99–02, "Regulatory Assessment Performance Indicator Guideline," dated August 31, 2013, were used. The inspectors reviewed the licensee's operator logs, RCS leakage tracking data, issue reports, event reports, and NRC Integrated Inspection Reports for April 2014 through March 2015 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two RCS leakage samples as defined in IP 71151–05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

To assist with the identification of repetitive equipment failures and specific human performance issues for followup, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6-month period of January 2015 through June 2015, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

This review constituted one semiannual trend inspection sample as defined in IP 71152–05.

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On July 8, 2015, the inspectors presented the inspection results to Mr. H. Vineyard, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

An interim exit was conducted for inspection results of the triennial review of heat sink performance with Mr. P. Karaba, Site Vice-President, on April 24, 2015. The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

P. Karaba, Site Vice-President
H. Vinyard, Plant Manager
J. Kowalski, Engineering Manager
K. Aleshire, Corporate Emergency Preparedness Manager
V. Cwietniewicz, Corporate Emergency Preparedness Manager
M. Jesse, Corporate Regulatory Assurance Manager
G. Ford, Regulatory Assurance Manager
J. Houston, Nuclear Oversight Manager
J. Moser, Radiation Protection Manager
M. Hayworth, Emergency Preparedness Manager
G. Brumbelow, Emergency Preparedness Coordinator
T. Dean, Operations Training Manager
D. Wright, NRC Examination Coordinator
L. Blunk, Regulatory Assurance
S. Shields, Regulatory Assurance
B. Hilton, Design Manager
A. Baker, Dosimetry Specialist
J. Bauer, Training Director
T. Dean, Operations Training Manager
J. Shields, Program Engineering Manager
D. Anthony, Non-Destructive Examination
B. Casey, Inservice Inspection

Nuclear Regulatory Commission

M. Kunowski, Chief, Reactor Projects Branch 5

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000373/2015002–01	URI	Both Ammonia Detectors Non-Functional on Unit 0 Train A of Control Room Ventilation (1R12)
05000374/2015002–02	NCV	Inadvertent Operation of Circuit Breaker Affecting Unit 2 Train A Residual Heat Removal Suppression Chamber Spray Isolation Valve (235Y-2 C3) (1R13)
05000374/2015002–03	SLIV, NCV	Inadequate 10 CFR 50.59 Evaluation for Jet Pump Plugs Affecting Fuel Bundle Cooling (1R18)
05000374/2015002–04	SLIV, NCV	Failure to Include Limiting Conditions for Operation in the Technical Specifications (1R18)

Closed

05000374/2015002–02	NCV	Inadvertent Operation of Circuit Breaker Affecting Unit 2 Train A Residual Heat Removal Suppression Chamber Spray Isolation Valve (235Y-2 C3) (1R13)
05000374/2015002–03	SLIV, NCV	Inadequate 10 CFR 50.59 Evaluation for Jet Pump Plugs Affecting Fuel Bundle Cooling (1R18)
05000374/2015002–04	SLIV, NCV	Failure to Include Limiting Conditions for Operation in the Technical Specifications (1R18)
05000374/2015001–03	URI	COLR Revision Potentially Created Non-Conservative Technical Specifications (1R18)

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R01 Adverse Weather Protection

PROCEDURES

Number	Description or Title	Revision
LOA–GRID–001	Low Grid Voltage	14
LS–AA–129	NERC Reliability Standard Compliance Program	5
OP–AA–102–101	Management of Nuclear Generation	11
OP–AA–108–107–1001	Station Response to Grid Capacity Conditions	6
OP–AA–108–107–1002	Interface Procedure Between ComEd/PECO and Exelon Generation (Nuclear/Power) for Transmission Operations	8
OP–AA–108–111–1001	Severe Weather and Natural Disaster Guidelines	12
OP–LA–101–111–1002	LaSalle Operations Philosophy Handbook	60
WC–AA–8000	Interface Procedure Between ComEd/PECO and Exelon Generation (Nuclear/Power) for Construction and Maintenance Activities	7
WC–AA–8003	Interface Procedure Between ComEd/PECO and Exelon Generation (Nuclear/Power) for Design Engineering and Transmission Planning Activities	5
WC–LA–8003–1004	LaSalle County Station Units 1 and 2 Nuclear Plant Interface Requirements (NPIRs)	0

MISCELLANEOUS

Number	Description or Title	Date
	Letter from P. Karaba to D. Enright; Certification of 2015 Summer Readiness	5/14/2015
	Letter from T. McGuire to D. Enright; ComEd Nuclear Station Switchyard Readiness Certification for Summer 2015	5/15/2015
	2015 ComEd Nuclear Station Switchyard Summer Readiness Review	3/10/2015
STA1	SY Work to Date Listing	6/15/2015
WC–AA–107	Summer Readiness Timelines and AT's	10/2014 – 5/2015

1R04 Equipment Alignment

FIGURES AND DRAWINGS

Number	Description or Title	Revision
M–83	P & ID Diesel Generator Auxiliary System	BB, AV, AF
M–87	P & ID Core Standby Cooling System Equipment Cooling Water System	BC

MISCELLANEOUS

Number	Description or Title	Date
	Equipment, Procedure, Location List	*undated*

1R05 Fire Protection

ACTION REQUESTS

Number	Description or Title
2494055	NRC Question on Fire Protection in 2B DGCW Pump Room

MISCELLANEOUS

Number	Description or Title	Revision
FZ 4F1	LaSalle Pre-Fire Plan; Unit 1 Elevation 710'0", Division 1 Essential Switchgear Room	Rev. 1
FZ 7B6	LaSalle Pre-Fire Plan; DG Bldg. 710'0" Elev. U1 Div 1 Diesel Day Tank Room	Rev. 1
FZ 8C1	LaSalle Pre-Fire Plan; DG Bldg. 674'0" Elev. U2 HPCS Diesel Fuel Tank Room	Rev. 0

1R06 Flood Protection Measures

MISCELLANEOUS

Number	Description or Title	Date
C467110014-9939	LaSalle Internal Flood Report	2/14/2013

1R11 Licensed Operator Requalification Program

MISCELLANEOUS

Number	Description or Title	Date
	L2C16 Ramp Up MCR Observe	

1R12 Maintenance Effectiveness

PROCEDURES

Number	Description or Title	Revision
LIP–VC–903	Control Room HVAC System Ammonia Detector Operability Test	8
LOR–2PM06J–B204	Control Room HVAC Ammonia Detector Alarm	6
LOS–ZZ–SR2	Charcoal Adsorber Laboratory Testing	4

ACTION REQUESTS

Number	Description or Title
081916	0XY–VC125a fault 30 found during LIP–VC903
585240	CDBI FASA: Concerns with Overly Conservative Calc LTS–900–14 RCIC Underground Piping Results
1176385	2E51–F084 Failed IST Closure Test
1257160	Perform an Examination of Valves 1(2)E51–F082/84
1457528	RCIC Condenser Cond Pump Run Caused 250 VDC UV Alarm
2495451	No Functioning Ammonia Detectors on a VC
2517220	A VC Ammonia Detector “Fault 30”
2520223	NRC Question on Ammonia Detector Commitment
2529301	NRC Unresolved Issue (URI) on Ammonia Commitments

FIGURES AND DRAWINGS

Number	Description or Title	Date or Revision
B453–003	General Arrangement Control Room HVAC Supply Air Filter	F
B453–9973	Standby Gas Treatment HECA Bed Assembly	C
B453–9974	Internal Fill Hopper	8/2/1977
M–1443	P & ID Auxiliary Electrical Equipment Rooms Airconditioning System	Q

WORKING DOCUMENTS

Number	Description or Title	Date
WO 1634377–01	Last Charcoal Sample Pulled from A VE Recirc Filter	5/12/2015
WO 1673449–01	Inline Oiler is Not Entraining Proper Amount of Oil	4/23/2015
WO 99144924–01	Repair Valve As Required in the Event of IST Failure	2/11/2000

MISCELLANEOUS

Number	Description or Title	Date
	System Health Report, Unit 1, RCIC	10/1/2014 – 12/31/2014
ML021120640	Letters from C. Allen Commonwealth Edison	1/6/1989
GL–95–10	Relocation of Selected Technical Specifications Requirements Related to Instrumentation (Generic Letter 95-10)	12/15/1995
89010609	Licensing Dept. ASMCF; LaSalle County Station Unit 2 Change Request to Allow Removal of the Ammonia Detectors from the Technical Specifications	1/6/1989
	Letter from E. Tidd, LaSalle County Emergency Services and Disaster Agency to G. Diederich, Exelon, Confirming Emergency Notification of Ammonia Incidents	3/27/1990

1R13 Maintenance Risk Assessments and Emergent Work Control

PROCEDURES

Number	Description or Title	Revision
LES–EQ–102	Testing of Environmentally Qualified Motors	10
LOA–TORN–001	High Winds / Tornado	16
MA–AA–716–100	Maintenance Alterations Process	12
OP–AA–101–111–1001	Operations Standards and Expectations	15
OP–AA–108–101	Control of Equipment and System Status	12
OP–AA–108–112	Plant Status and Configuration	8
OP–AA–108–117	Protected Equipment Program	4
WC–AA–104	Integrated Risk management	23

ACTION REQUESTS

Number	Description or Title
2501055	Operations Crew 5 Clock Reset
2511176	TSC Room Temperature Hi Alarm
2511928	2B TDRFP Hi Vibe Alarm
2512073	Vibration trending Points for FW Turbines Not on PPC
2512320	TSC HVAC MCR Unexpected Alarm 2PM06J–B208
2515603	Unexpected Alarm TSC Low Air Flow
2511928	2B TDRFP Hi Vibe Alarm
2513023	4.0 Critique of TDRFP Vibe Hi Alarm 2B TDRFP
2513285	Generate ODM Assignments for 2B TDRFP High Vibes
2518110	Security—Tornado Warning Issued for LaSalle Station
2519736	NRC Question on LOA–TORN–001

FIGURES AND DRAWINGS

Number	Description or Title	Revision
1E–2–4226AL	Schematic Diagram Reactor Core Isolation Cooling System RI (E51) Part 11	T
1E–2–4389AC	Internal/External Wiring Diagram Reactor Building 480V MCC 235Y–2 (2AP76E) Part 3	O
1E–2–4220BK	Schematic Diagram Residual Heat Removal System “RH” (E12) Part 34	M

WORKING DOCUMENTS

Number	Description or Title	Date or Revision
Op Log	Unofficial Copy of Operations Log	6/10/2015
WO 1643222–01	Take Current Readings and Monitor Noise	5/14/2015

MISCELLANEOUS

Number	Description or Title	Date or Revision
1E–2–4000CV	Key Diagram 480V MCC 235Y–2–2AP76E	Y
AR 2500451	Configuration Control Alert, Breaker Cycled to “Off” Position After Being Bumped	5/14/2015
AR 2500451	Human Performance Issue, Verbal Report, Breaker Cycled to “Off” Position After Being Bumped	5/14/2015
AR 2511176	Event Report – TSC Room Temperature Hi Alarm	6/2015
AR 2511928	Event Report – 2B TDRFP Hi Vibe Alarm	6/8/2015
Message 141385	EMnet Emergency Message – Tornado Warning	6/22/2015
WO 1643222–01	Job Hazard Analysis Referral for Work Order to “Take Current Readings and Monitor Noise”	

1R15 Operability Determinations and Functionality Assessments

PROCEDURES

Number	Description or Title	Revision
LMS–ZZ–04	Water Tight Door Inspection	6
LOP–DG–04	If B Air Start Compressor 1(2) DG08CB Needs to be Isolated	63
LOP–PF–01	Closure of Water Tight Doors	6
LOS–DG–Q2	2A DG B Air Compressor Discharge Check Valve Test	61
LOS–PF–M1	ECCS/CSCS Water Tight Door Surveillance	0

ACTION REQUESTS

Number	Description or Title
249834	Engine Mounting Bolts on SE Corner Found Loose
1518770	Door 17 — Water Tight Door Not Sealing Properly
1527139	IEMA Question on Water Tight Door Operability Guidance
2485587	NRC Id'd Observation on DG Air Start Oiler
2489882	2A DG B Air System Check Valve Failure
2504640	Impact of IR 2503008 on the Reactor Building Crane Qualification
2513665	Operability Evaluation Requested for Reactor Building and Reactor Building Overhead Crane 10 CFR Part 21
2514058	1" Tear in Water Tight Door Seal

FIGURES AND DRAWINGS

Number	Description or Title	Revision
M–83	P & ID Diesel Generator Auxiliary System	BB

WORKING DOCUMENTS

Number	Description or Title	Date
WO1792520–01	LOS–DG–M2 1A DG Fast Start ATT 1A–FAST	4/8/2015

MISCELLANEOUS

Number	Description or Title	Revision
OE 2015-002	Operability Evaluation; Reactor Building Overhead Crane and Reactor Building	000
B 3.5.1–1	Emergency Core Cooling System (ECCS)	1
3.8.3–1	Electrical Power Systems, Starting Air	Amd. 191/178

1R18 Plant Modifications

ACTION REQUESTS

Number	Description or Title
02449326	FME Jet Pump Plug Seal Missing
02450946	Potential Safety Concern—Fuel Bundle Flow Blockage
02486215	Issues with Safety Evaluation L15–47 and Screening L15-044

WORKING DOCUMENTS

Number	Description or Title	Revision
EC 400019	Lost Parts Evaluation for L2R25	3
EC 400989	Lost Parts Evaluation—Miscellaneous Parts from L2R25	1

WORKING DOCUMENTS

Number	Description or Title	Revision
EC 401294	Supplemental Evaluation to LaSalle 2 Cycle 16 Lost Parts Evaluation	1

MISCELLANEOUS

Number	Description or Title	Date or Revision
50.59 No. L15–47	Evaluation of Fuel Licensing Impact of Lost Jet Pump Plug Seals in Primary System	2
50.59 Screening No. L15–044	Lost Parts Evaluation for L2R15	3
GE 002N5595R0	GE Hitachi Nuclear Energy; LaSalle County Nuclear Power Station, Unit 2 Cycle 16 Jet Pump Seal Flow Blockage Anticipated Operational Occurrences Evaluation for Power Ascension	3/2015

1R19 Post-Maintenance Testing

PROCEDURES

Number	Description or Title	Revision
LOP–RR–15	Reactor Recirculation Hydraulic Power Unit Pump Stroke Adjustment	8

ACTION REQUESTS

Number	Description or Title
1434660	Vacuum Pump has Shaft Packing Leak While Running and Stby
1621324	Component Classified as AQ With Q basis As #

WORKING DOCUMENTS

Number	Description or Title	Date
WO 1460591–02	U–1 RCIC Replace the EGM Module and RGSC	4/22/2015
WO 1673449–02	InLine Oiler is Not Entraining Proper Amount of Oil	5/09/2015
WO 1809603–03, 04	A RR Subloop 1 HPU Not Maintaining Pressure Between Starts	6/16/2015
WO 1469775–01	1E31–R621B RWCU Strtup ISOL Bypass Hi Flow Timer/Relay	6/16/2015
WO 1469776–01	1E31–R621B RWCU Strtup ISOL Bypass Hi Flow Timer/Relay	6/16/2015
WO 1822021-02	1A SBLC Did Not Meet Flow Criteria	6/18/2015
WO 1468644-02	Disassemble, Clean, Inspect Starting Air Check Valve	6/24/2015

WORKING DOCUMENTS

Number	Description or Title	Date
WO 1645902-02	Inspect 2A Diesel Generator Start Air Moisture Separator	6/24/2015

MISCELLANEOUS

Number	Description or Title	Date or Revision
LOS–DG–M2	LOS–DG–M2 1A DG Fast Start Voltage and Frequency Graph	5/15/2015

1R22 Surveillance Testing

PROCEDURES

Number	Description or Title	Revision
LIS–MR–303B	Unit 1 Average Power Range Monitor Channels B, D, and F Rod Block and SCRAM Functional Test	20
LOS–DG–M3	1B(2B) Diesel Generator Operability Test	92

ACTION REQUESTS

Number	Description or Title
1548147	Unit 1 LPCS Pump requires balance Adjustment
2490700	NRC ID: U2 SF Spool Piece has Possible Floatable Material
2515876	Vibration readings on U1 LPCS Pump Entered Alert Range

WORKING DOCUMENTS

Number	Description or Title	Date
WO 1666230–01	Unit 1 LPCS Pump Requires Balance Adjustment	9/18/2013
WO 1709206–01	Tech Spec Surveillance; Unit 2 RHR System Biennial Comprehensive IST Pump Test LOS–RH–Q1 AH 2C	
WO 1709214	Tech Spec Surveillance, Biennial Comprehensive IST Pump Test	4/16/2015
WO 1709217–01	Tech Spec Surveillance, IST Comprehensive Pump Test for 2E12-C300C & 2E12-C300D	6/24/2015
WO 1817736–01	Tech Spec Surveillance: LOS–LP–Q1 U1 LPCS Att 1A	6/18/2015
WO 1835210–01	Tech Spec Surveillance, LOS–DG–M2 2A Diesel Generator Att 2A-Idle	6/26/2015

4OA1 Performance Indicator Verification

WORKING DOCUMENTS

Number	Description or Title	Date or Revision
LOS–AA–S201	Unit 2 Shiftly Surveillance for Mode 1,2, or 3	11/09/2014

MISCELLANEOUS

Description or Title	Date or Revision
DWEDS, DWFDS, Days, Minutes, Counter Delta data for Units 1 and 2	2014 – 2015

4OA2 Identification and Resolution of Problems

PROCEDURES

Number	Description or Title	Revision
LOS-AA-S201	Unit 2 Shiftly Surveillance	92

ACTION REQUESTS GENERATED FROM NRC OR IEMA INSPECTION

Number	Description or Title
2481375	NRC Id's Breaker FME Covers with Slit Openings
2482812	NRC Id'd URI on COLR Change Impact on TS
2485000	Hand Monitor on Argos at 4 Line was Not Retracting Properly
2485587	NRC Id'd Observation on DG Air Start Oiler
2489835	NRC Identified 0FP078 Valve Degraded Performance
2489847	NRC Identified: Errors in IST Program Documentation
2489857	NRC Identified Procedure Compliance Issue
2490700	NRC ID: U2 SF Spool Piece has Possible Floatable Material
2494055	NRC Question on Fire Protection in 2B DGCW Pump Room
2502347	OE Security—NRC Issued Violation for Failure of 10CFR50.54(p)
2502444	NRC RP Baseline Inspection—Check-In
2502652	NRC Id'd Issue with MOC EOC Timeliness
2503136	NRC Identified Issues 5/19/15
2503551	2015 PI&R Id'd EFR for RCR 1627300 Not Measurable
2503558	NRC PI&R: Safety Culture Components Applicable RCR 2471718
2503805	PI&R Inspection Id'd Procedure Revision Prioritization
2504236	Untimely Actions to Address 2DG06A UT Exam Results
2505415	LaSalle License Renewal—NRC RAIS
2508347	NRC Green Finding and NCV on PT Procedure
2516970	Potential Failure to Notify NRC of Licensed Operator Retirement
2519736	NRC Question on LOA–TORN–001
2519808	NRC Identified, Water on Offgas Building Floor

ACTION REQUESTS GENERATED FROM NRC OR IEMA INSPECTION

Number	Description or Title
2519863	NRC Question Concerning Water Observed Inside Door 554
2520223	NRC Question on Ammonia Detector Commitment

ACTION REQUESTS

Number	Description or Title
1603566	MCR DWFDS Counts Indicated with No Pumpdown
1603821	MCR DWFDS Counts Indicated with No Pumpdown
1604383	NSO ID: DWFDS Totalizer Adding Counts Without Pumpdown
2404844	DWFDS Excessive Run Time Alarm
2418417	U2 DWFDS Excessive Runtime
2504222	Observed Increase in U2 Offgas Pre-Treat Activities
2505781	Unit 2 PST Results
2535598	Current Status of LaSalle Unit 2 Fuel Defect

WORKING DOCUMENTS

Number	Description or Title	Revision
WO 1831021–02	Revise Setting of Unit 1 DWFDS Pumps Excess Run Timer	0
WO 1831023–02	Revise Setting of Unit 2 DWFDS Pumps Excess Run Timer	0
EC 401851	Design Change Package for Revise Setting of U1/2 DWFDS Pumps Excess Run Timer to 15 Minutes	0

MISCELLANEOUS

Number	Description or Title	Date or Revision
	Plan of Day, Unit 2 Calculated DWFDS Fill Up Rate (gpm)	6/23/2015
B 3.4.5–1	Reactor Coolant System, RCS Operational Leakage	0
B 3.4.7–1	Reactor Coolant System, RCS Leakage Detection Instrumentation	53

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Documents Access Management System
AOO	Abnormal Operational Occurrence
AR	Action Report
CAP	Corrective Action Program
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
DC	Direct Current
DG	Diesel Generator
EC	Engineering Change
EOOS	Equipment Out-of-Service
EQ	Environmental Qualification
HPCS	High Pressure Core Spray
HVAC	Heating, Ventilation, and Air Conditioning
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
IST	In-service Testing
LCO	Limiting Condition for Operation
LPCS	Low Pressure Core Spray
L2R15	LaSalle Unit 2, Cycle 15, Refueling Outage
MCC	Motor Control Center
MCPR	Minimum Critical Power Ratio
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OOS	Out-of-Service
PARS	Publicly Available Records System
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PMT	Post-Maintenance Testing
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
SBLC	Standby Liquid Control
SDP	Significance Determination Process
SRV	Safety/Relief Valve
SSC	System, Structure, and Component
TS	Technical Specification
TSC	Technical Support Center
TSO	Transmission System Operator

UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
UO_2	Uranium-Oxide
URI	Unresolved Item
Vdc	Volts Direct Current
WO	Work Order

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Sincerely,

/RA John Jandovitz Acting for/

Michael Kunowski, Chief
Branch 5
Division of Reactor Projects

Docket Nos. 50–373 and 50–374
License Nos. NPF–11 and NPF–18

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